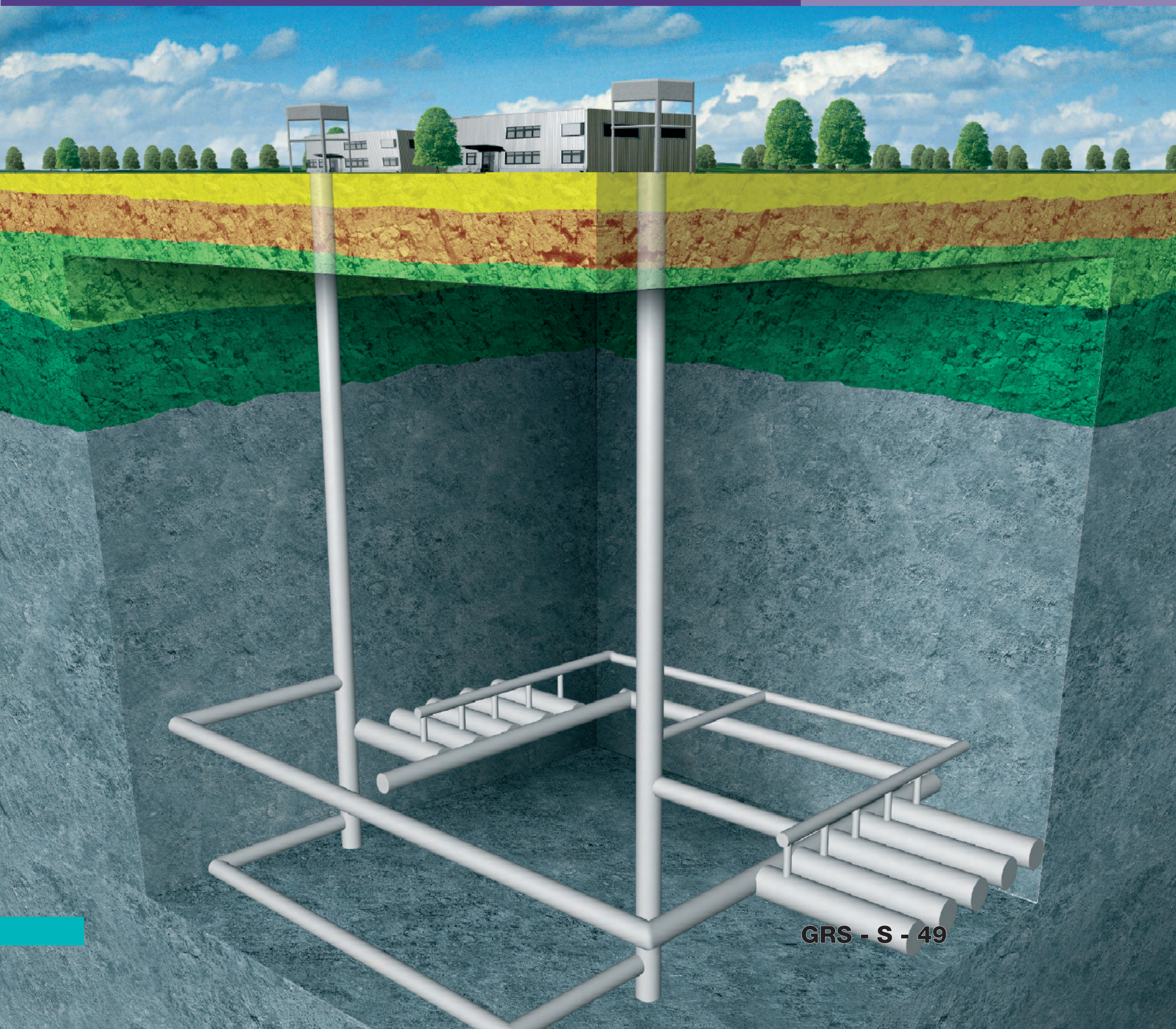
Bundesamt für Strahlenschutz

Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH

Radioactive Waste Disposal in Geological Formations

International Conference
Braunschweig ("City of Science 2007")
November 6 – 9, 2007

PROCEEDINGS



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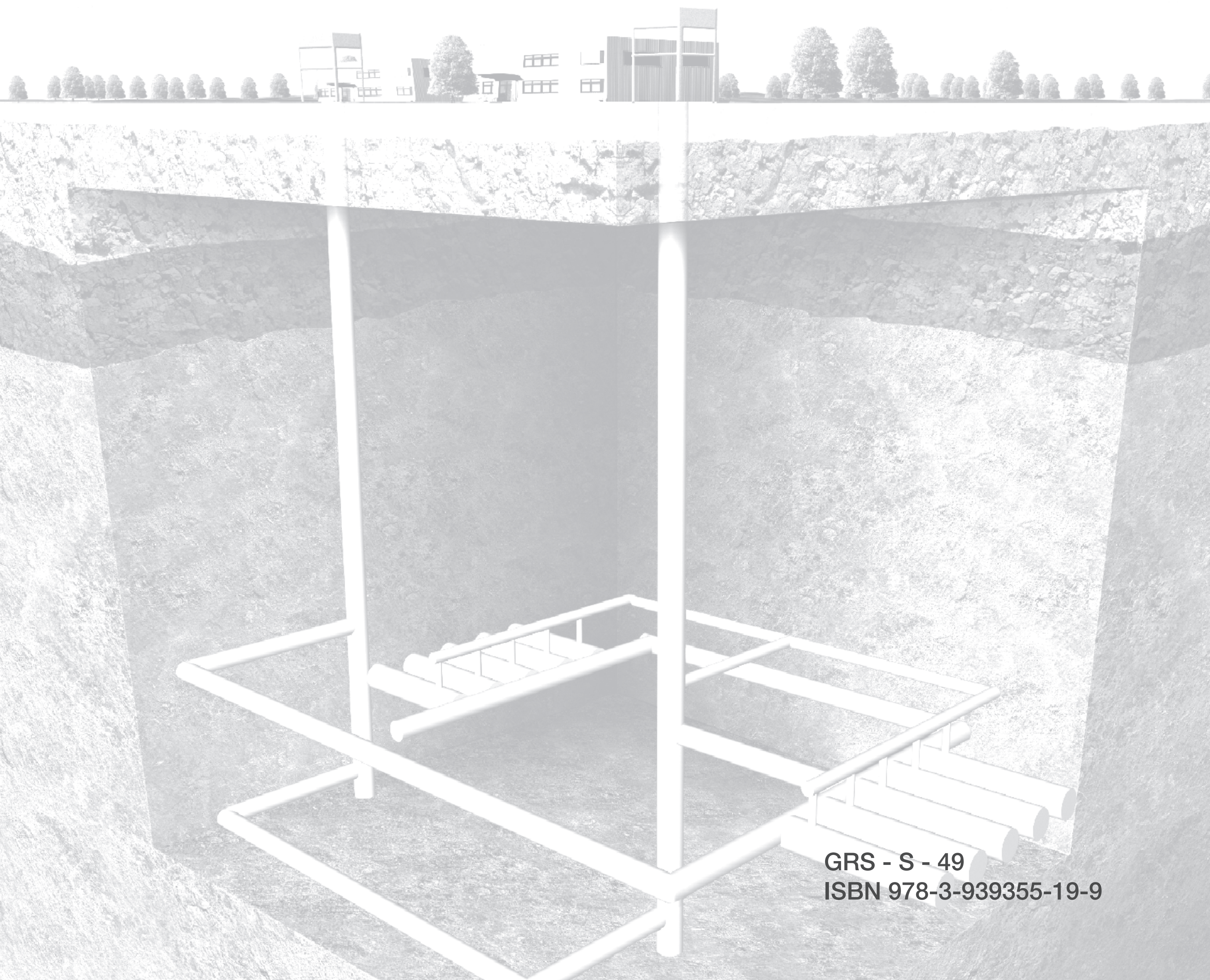

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Summary

Offene Diskussion zur Endlagerung hochradioaktiver Abfälle

Bei der nach wie vor ungelösten Entsorgung hochradioaktiver Abfälle setzen die auf diesem Gebiet am weitesten fortgeschrittenen Länder in der Regel auf vergleichende Auswahlverfahren mit umfassender Beteiligung der Bevölkerung. Dies ist ein zentrales Ergebnis der Fachtagung „RepoSafe“ in Braunschweig, auf der vom 6. bis 9. November international führende Experten auf Einladung des Bundesamtes für Strahlenschutz (BfS) und der Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) die neuesten Erkenntnisse bei der Entsorgung radioaktiver Abfälle präsentiert haben. Insgesamt nahmen rund 280 Teilnehmer aus 16 Ländern an der Tagung teil.

Vertreter der nationalen Entsorgungsinstitutionen der Schweiz, Schwedens und Finnlands betonten, wie wichtig die umfassende Einbindung der Öffentlichkeit für die derzeit laufenden Standort-Auswahlverfahren und deren Akzeptanz in der Bevölkerung sei. Dies sei die zentrale Lehre aus den bisherigen Erfahrungen. Eine Mehrheit der an der Konferenz teilnehmenden Wissenschaftler sprach sich zudem dafür aus, dass die Endlagerung radioaktiver Abfälle von heutigen Generationen und im eigenen Land gelöst werden muss. Dabei müssen sicherheitstechnische Fragestellungen Priorität genießen. Viele Experten betonten außerdem, dass in einzelnen Teilbereichen weiterer Forschungsbedarf bestehe, z.B. bei Fragen der langzeitlichen Wirkungsmechanismen für die Radionuklidrückhaltung, der Verfüllung und des Verschlusses von Endlagern und der Bewertung der Endlagersicherheit über lange Zeiträume.

Unbeschadet vieler Gemeinsamkeiten diskutierten die Teilnehmer auch Probleme und unterschiedliche Ansätze bei der Entsorgung radioaktiver Abfälle. So werden weltweit unterschiedliche Wirtsgesteine als Endlagermedium untersucht. Länder wie Schweden oder Finnland planen Endlager in Granitgestein, Frankreich oder die Schweiz erforschen dagegen Ton, während Deutschland und die USA bisher auch Salz untersucht haben. Während eine Mehrheit auf die Endlagerung in tiefen geologischen Schichten setzt, verfolgen einige Länder, beispielsweise die Niederlande, das Konzept der langfristigen Zwischenlagerung.

Die Veranstalter legten bei der von Bundesumweltminister Sigmar Gabriel und dem Oberbürgermeister der Stadt Braunschweig, Dr. Gert Hoffmann, am 6. November eröffneten Tagung Wert auf ein möglichst breites Meinungsspektrum und einen offenen Dialog. So wurden auch Vertreter verschiedener Bürgerinitiativen eingeladen, um ihre Sichtweise einzubringen und Erfahrungen zu sammeln. Am ersten Konferenztag gaben Vertreter verschiedener internationaler, mit der Endlagerung betrauter Organisationen Einblicke in die internationale Entwicklung und den Stand bei der Endlagerung in den jeweiligen Ländern. An den Folgetagen präsentierten Wissenschaftlerinnen und Wissenschaftler aus dem In- und Ausland aktuelle Ergebnisse auf dem Gebiet der Endlagerung. Im Fokus der Diskussionen standen u. a. neue Erkenntnisse für verschiedene Wirtsgesteine wie Ton oder Salz und Entwicklungen beim Langzeitverschluss von Endlagern für radioaktive Abfälle. Die deutsche Situation bezüglich der Endlagerung hochradioaktiver Abfälle wurde ergänzend zu den Ausführungen von Herrn Bundesminister Gabriel in Beiträgen des Bundesamtes für Strahlenschutz (BfS), des Bundesministeriums für Wirtschaft und Technologie und der Energieversorgungsunternehmen näher beleuchtet.

Im Anschluss an die Vorträge und die fachlichen Diskussionen besuchten etliche Teilnehmerinnen und Teilnehmer der Konferenz bei so genannten Field Trips die Endlagerprojekte Konrad, Morsleben, Asse und das Erkundungsbergwerk Gorleben. Hier konnten sich die Teilnehmerinnen und Teilnehmer der Konferenz ein Bild über die jeweilige Situation vor Ort machen. ■

Open Discussion on High-Level Radioactive Waste Disposal

To solve the still open question of high-level radioactive waste disposal, the countries having made the greatest progress in this field usually choose to carry out comparing selection procedures including broad involvement of the public. This is a central result of the “RepoSafe”symposium which took place from November 6 to 9, 2007, in Braunschweig. Within the scope of this symposium, internationally leading experts, invited by the Federal Office for Radiation Protection (BfS) and Gesellschaft für Anlagen- und Reaktorsicherheit (GRS), presented the most recent findings in the field of radioactive waste management. Altogether, about 280 participants from 16 countries participated in the symposium.

Representatives of the national waste management institutions of Switzerland, Sweden and Finland emphasised the importance of a broad involvement of the public to the currently running site selection procedures and their acceptance in the population. They stated that this was the central lesson learnt from previous experience. A majority of the scientists participating in the symposium furthermore argued that the issue of radioactive waste disposal must be solved by today´s generations and in one´s own country, putting priority to safety-related issues. Many experts additionally emphasised that in there was further need for research in individual subareas, such as questions relating to the long-term mechanisms of action for radionuclide retention, the backfilling and sealing of repositories and the evaluation of repository safety over long periods of time.

Irrespective of many mutualities, the participants also discussed problems and different approaches to radioactive waste management. For example, different host rocks are investigated world-wide for their capabilities to host a repository. Countries such as Sweden or Finland plan repositories in granite, France or Switzerland, on the other hand, explore clay, while Germany and the USA have also investigated salt so far. While a majority pursues disposal in deep geological formations, some countries, such as the Netherlands, pursue the concept of long-term interim storage.

In the symposium, which was opened on November 6, 2007 by the Minister for the Environment, Nature Conservation and Nuclear Safety, Sigmar Gabriel, and the Mayor of Braunschweig, Dr. Gert Hoffmann, the participants attached importance to a broad spectrum of opinions and an open dialogue. Thus, representatives of various citizens´ initiatives were also invited to convey theier views. On the first symposium day, representatives of various international and national organisations entrusted with disposal delivered insight into the international developments and the state of disposal in the respective countries. On the following days, national and international scientists presented current results in the field of disposal. Among others, new findings for different host rocks were in the focus of discussion, such as clay or salt, and developments in the long-term sealing of radioactive waste repositories. In addition to the remarks made by Federal Minister Gabriel, the German situation with respect to high-level radioactive waste disposal was looked at in more detail in contributions made by representatives of the Federal Office for Radiation Protection (BfS), the Federal Ministry of Economics and Technology and the utilities.

Subsequent to the lectures and the technical discussions, many participants of the symposium visited the repositories of Konrad, Morsleben and Asse as well as the Gorleben exploration mine. Thus, where they could get an idea of the respective situation on site. ■

Contents

Contents

Contents

Contents

Welcome and Opening Remarks

Sigmar Gabriel
Federal Minister for Environment, Nature
Conservation and Nuclear Safety

"Taking Responsibility – Achieving Consensus on a final Repository"

Dr Hoffmann, Ladies and Gentlemen,

as a member of the Bundestag for a constituency where one final repository is currently being decommissioned and another is being constructed, this issue affects me personally.

My home town of Goslar is very close to Braunschweig – about 50 km – and I have felt a close connection to the city and region of Braunschweig since my childhood. It therefore gives me particular pleasure to be here in Braunschweig today to open the International Conference on Radioactive Waste Disposal in Geological Formations. I would especially like to thank the organisers of this conference, the Federal Office for Radiation Protection (BfS) and GRS - Gesellschaft für Anlagen- und Reaktorsicherheit mbH - the expert and research organisation in the field of nuclear safety.

Taking account of the state of the art in science and technology is a key element both for the selection of a final repository site and for developing a final repository concept. In this regard, this conference on radioactive waste disposal is happening in precisely the right place and at the right time: As the "City of Science 2007" Braunschweig is presenting itself as a city which offers a fascinating variety of research disciplines. 27 scientific organisations and research institutions have their headquarters in Braunschweig. Scientists make up 4 percent of the overall workforce – a higher proportion than in any other European region. Many developments from the area spring from the close cooperation between the scientific community and industry. As City of Science 2007, Braunschweig also aims to arouse in its citizens an enthusiasm and curiosity for science. They should participate in the range of knowledge which the region has generated. I believe that participation is also a key element in the question of final disposal. Local people must be involved in the various processes, they must feel that their concerns and fears are being taken seriously. The only way to ensure that the measures taken are accepted is if we, the responsible actors from politics and society, can create this sense of participation. In this light, Braunschweig's approach of encouraging public participation should be an example to us all.

Germany is currently experiencing all phases of final disposal. The Morsleben repository for low- and intermediate-level radioactive waste in Saxony-Anhalt and the former research mine Asse in Lower Saxony, also used for low- and intermediate-level waste, must both be decommissioned. Decommissioning these two facilities will ultimately cost more than two billion euros.

These two projects are lessons in how not to go about searching for a final repository or deciding on a site. This misguided approach must not result in future generations having to put up with heavy financial burdens or living with doubts as to the safety of the repository. Final repository safety begins with the site selection. Radioactive wastes should not be stored unless the decommissioning concept has been proved fully feasible and the long-term safety of the final repository established beyond question. Unfortunately, neither of these aspects were taken into consideration with regard to Morsleben or Asse.

In May 2007, construction began on the Konrad repository which has been approved for 303,000 m^3 of radioactive wastes with negligible heat generation and which received planning approval confirmation from the Federal Administrative Court last spring. Under the current plans the repository will go into operation at the end of 2013.

However, the question of how to select the site for a final repository of high-level, heat generating waste, remains open in Germany. The Land of Lower Saxony decided on the Gorleben site in 1997 in the course of a technical and administrative site selection procedure. Exploration of the site has been suspended since October 2000 in order to clarify some safety and planning issues. The selection procedure did not address the socio-political dimension of the final repository issue or the relevant site decisions. The Gorleben site was decided on within one year and without any participation of the local public at all - although the massive protests triggered by the nomination of the potential sites Lutterloh, Lichtenhorst and Wahn ought to have served as a warning in the run-up to the nomination of Gorleben.

While high-level radioactive wastes occurring in Germany only make up 10 percent of the waste volumes predicted for the country up to 2040, they contain more than 99 percent of the radioactivity in existing and future waste. We anticipate that around 24,000 m^3 of high-level waste will have to be stored in the repository to be established for this purpose.

A working final repository for this high-level waste ought to be available by 2035 at the latest, since from that date licences for transport cask storage and licences for on-site interim storage facilities will gradually expire.

In its 2005 Coalition Agreement, the German Government set itself the goal of solving this problem. Past experience shows that in solving the problem of a final repository, the process was always impeded by the fact that there was no broad consensus on how to decide on a final repository site. The grand coalition of CDU, CSU and SPD now has the opportunity - and most especially the obligation - to solve this issue. In my view, the major parties will only be able to organise a social consensus which is broad enough to survive beyond the legislative period if they work together.

Therefore, the 1977 decision to earmark the Gorleben salt dome as a final repository for all types of radioactive waste should be reviewed in a selection procedure according to the current state of the art in science and technology. Such a site selection procedure is the best way to secure legal certainty and facilitate planning for the energy industry, to ensure transparency and comprehensibility for the public and to protect future generations from the dangers of radioactive waste.

In autumn last year, my ministry therefore developed a concept for implementing a site selection procedure which can form the basis for the continuing process. This concept "taking responsibility – achieving consensus on a final repository" is currently still being discussed within the Federal Government.

The concept is based on the following principles:

- Acknowledging national tasks: radioactive waste originating from the use of nuclear power in Germany must also be disposed of in Germany, not exported to other countries which may have lower safety standards.

- Taking responsibility: the generation which benefits from nuclear power must also dispose of the waste. For this reason, the question of a repository must be addressed now.

- In final disposal, safety must take priority over all other aspects. Therefore, the most suitable site must be decided on based on a comparison of several alternatives.

- The selection and designation of a final repository site calls for a comprehensible and transparent procedure.

I also feel it is important to stress here that this concept is based on requirements which correspond to the state of the art in science and technology, as described in international standards of the IAEA and published documents of the OEC/NEA. These requirements include:

- step-by-step procedure with clearly defined review steps;

- establishing selection criteria and safety requirements before the selection procedure begins;

- transparency of the procedure and creation of extensive participation options;

- definition of protection targets and safety requirements as assessment criteria;

- independent collection and evaluation of relevant site data.

Such a procedure is in keeping with international practice. In Switzerland e.g. there is to be a selection procedure for deep geological repositories based on geological, planning and socio-economic criteria, with a decision on a site by 2018. The Swiss approach is characterised by a high degree of participation. The neighbouring communities on the German side of the border have the support of the Federal Environment Ministry in representing their interests. What we ask of Switzerland should be a matter of course for us in Germany.

The concept which I presented in 2006 for discussion in the Federal Government is based on the proposals developed in 2002 by the Committee on a Selection Procedure for Repository Sites (AkEnd). In contrast to the selection procedure proposed by AkEnd, which starts off with a "blank map" of Germany and gives equal consideration to all sites, this procedure gives special consideration to Gorleben because of the extensive exploration already carried out there and the 1.4 billion euro invested in the project. The concept envisages a review aimed at ascertaining whether any alternative sites to Gorleben demonstrate any striking indication of a higher level of safety or give reason to expect such a higher level of safety.

According to this concept, another site should only be selected and explored if it promises substantial safety advantages over Gorleben. Safety-related advantages and disadvantages can only be communicated in a plausible way if selection criteria and safety requirements are first laid down with the participation of the public. An initial draft for such safety requirements based on the recommendations of the International Atomic Energy Agency (IAEA) and the proposals of the GRS mbH, is currently being elaborated in the BMU. Discussions have already begun among the experts in the field – with representatives from the Reactor Safety Commission and the Commission on Radiological Protection.

As I described earlier, Germany has had a wide range of experience regarding the issue of final disposal. This experience should be used and also incorporated into the discussions at this conference. The discussions should aim at determining how we can meet the public's justified claims to the highest possible level of safety in the final disposal of radioactive waste and to fairness in the site selection decision. After decades of work and billions of euros of investment, we cannot allow any serious doubts regarding the safety of a site or the objectivity of the selection procedure to remain. In this respect, we should learn from the debate in Germany about Gorleben, and from the growing debate surrounding the final repository site Yucca Mountain in the US.

Amidst all the discussions, it is also important to bear in mind that the future role of nuclear power in electricity production has a key influence on the issue of final disposal. A clear stance with regard to the time limit on nuclear power utilisation makes it easier for many people to accept projects on the final disposal of radioactive waste.

I would like to wish you all a successful conference with interesting and open discussions and hope that here in Germany we will soon also achieve a clear position on the further procedure regarding final disposal. In this way we will give life to the title of the concept developed by my ministry:

"Taking responsibility – achieving consensus on a final repository!"

Thank you very much. ■

Dr. Gert Hofmann
Mayor of the City of
Braunschweig

My dear Minister, Mr. President, Mr. Hahn, Ladies and Gentlemen,

I have great pleasure in welcoming all participants of this important international conference to Braunschweig, Germany`s City of the Sciences 2007. In particular I would like to thank the organisers for bringing this highly-esteemed conference to our city. I hope that you will retain good memories of the city itself, as well as this event.

This conference ist well suited to Braunschweig this year, not only because, as I mentioned, we have been designated Germany's City of the Sciences 2007 and therefore, of course, are hosting a significant number of renowned international congresses and events. These events bring many visitors from across the globe and your presence, ladies and gentlemen, also illustrates the important role that the sciences play in our city, and draws attention to the number of celebrated, well-known scientific establishments located in the city and region of Braunschweig. The German Federal Government Agency for Radiation Protection, for example, ist one such institution.

Braunschweig, therefore, is an equally suitable location for the conference on this same topic, because the issue of permanent disposal of nuclear waste is one that has involved both scientists and the general population here, both in the past and in the present.

Scientists from our city were important pioneers in research work on natural environmental radioactivity. And in the present, it is not only the Association for Plant and Reactor Safety that is based here in Braunschweig, but also Federal institutions an other important scientific establishments, such as the Federal Agency for Physics and Technology, which also has links to the topic of nuclear science.

You are probably all aware that our region ist currently involved in intensive discussion in political and scientific spheres, as well as among the general public, regarding the topic of permanent storage of nuclear waste, whether as final or interim storage.

The issue of peaceful use of nuclear energy is highly disputed, not only among us. My very personal opinion is that we should continue to use our own national, sound nuclear technology for the purposes of climate protection, at least on a transitional basis. Others, as you know, have completely different views. Either way, permanent storage must be controlled in a safe and stable manner. Unfortunately, this is not yet the case, which is a cause of concern to many. Personally, I find it particularly regrettable when people who point out this problem develop activities themselves that impede concrete plans for permanent storage.

It would be of great benefit to all of us if this conference, for instance, could make a substantial contribution to finding solutions and, if possible, also assist in establishing a general consensus.

This, too, would be a very valuable contribution to the City of the Sciences 2007.

Ladies and gentlemen, it would seem appropriate to include some information about our city and what it has to offer in this greeting. However, since I have the pleasure and the honour of welcoming you once again at a reception this evening, I would like to address such points then. Therefore, I will conclude for now with the hope that this congress will be a successful one.

Session 1

International Developments

1.01 Euratom Research: Making the Case for Geological Disposal

Simon Webster
*Head, Unit DG Research "Fission", European Commission, Brussels, Belgium**

Abstract
EU support for R&D in nuclear science and technology is channelled principally through multi-annual Euratom research Framework Programmes (FPs) covering all areas linked to the peaceful uses of nuclear energy. Radioactive waste management, including in particular geological disposal, has been a priority area of research for many years and remains so in the current programme FP7 (2007-2011). Over the years, FPs have covered the full range of scientific topics, with a progression from more fundamental research in the first programmes, to applied research, development and demonstration in the host rock environment of underground research laboratories in later programmes. In FP6, €45M have been committed to this research effort, funding a small number of large integrating projects grouping all research in key domains: near field and engineered barriers; far field; engineering systems and demonstration; performance assessment. Future support will seek to maximise effectiveness of the Euratom programme by further enhancing co-ordination between respective efforts in EU Member States. Though important research and investigation is still required on a number of key issues, these are increasingly related to local site conditions and linked to the licensing of a particular repository. Crucially, the Euratom research programme, in collaboration with national efforts, has consistently pointed to the technical feasibility of geological disposal.

1 Introduction – Euratom Research

As the executive institution of the EU, the European Commission (EC) is responsible for the planning and implementation of the EU research programme. For more than two decades, the principle method of providing this support has been the Framework Programme (FP). This is a shared-cost grant-based programme (projects are partly funded by the participating organisations), each FP having a duration of at least four years. These programmes are implemented via calls for proposals published at regular intervals in the EU's Official Journal. Evaluations of submitted proposals are usually carried out with the help of independent experts.

Ever since the start of European integration back in the 1950s there has been a separate Treaty covering nuclear issues. The Euratom Treaty [1], short for Treaty establishing the European Atomic Energy Community, was one of the original Treaties of Rome at the inaugural signing in 1957. The Euratom Treaty covers everything that was important in the nuclear field in the late 1950s – nuclear safeguards, radiation protection, supply of fissile material, and also research. The inclusion of research has meant that all EU support for R&D in applied nuclear science and technology (including radioactive waste, reactor systems – fusion as well as fission, etc.) comes under a separate legal basis from the rest of EU research, and is covered by its own legally distinct FP. These Euratom FPs have traditionally been run in parallel with the "non-nuclear" FPs and are implemented in a similar fashion using the same type of funding instruments, at least as far as research in nuclear fission and radiation protection is concerned.

Back in the early 80s, during the first FPs, the Euratom Programme accounted for something like 25% of the total EU research spending. Today, this percentage is much less – in FP7, launched at the beginning of 2007, the nuclear (Euratom) component is some €2.75 billion (over 5 years) compared with some €50.52 billion (over 7 year) for the non-nuclear FP covering the whole of non-Euratom research. The majority of the Euratom research budget is spent on the fusion programme – approx. €2 billion – the rest is split between the nuclear activities of the EC's own Joint Research Centre (€517 million) and the programme in fission and radiation protection (€287 million). One of the priority areas of the latter is, and has been ever since the first FP, management of radioactive waste. Though the scope of this area now includes partitioning and transmutation (P&T), for the last two decades the principal topic of research has been geological disposal of high-level / long-lived waste.

* The views expressed in this paper are those of the author and do not necessarily reflect those of the European Commission.

The level of research spending on fission reactor technology in general has fallen over this period (as would be expected with the coming to industrial maturity of the 2nd generation of power reactors), but the research effort on managing the back end has remained a priority, reflecting the fact that this is the only remaining issue to be resolved before a full industrial implementation of the whole nuclear cycle. In the past, the Euratom programme also funded research on low-level waste management and decommissioning activities, though these are now considered to have reached a high level of industrial maturity and further such support at the EU level is no longer necessary.

The research on P&T is examining the potential to reduce the amounts of some of the longest-lived radionuclides in the most radiotoxic wastes. This involves chemical separation of key radionuclides followed by nuclear transmutation, either in a sub-critical nuclear reactor coupled to a particle accelerator or in a critical power reactor. This is a very long-term research programme, which is increasingly being linked with research on advanced reactor systems and associated fuel cycles as part of the development of the next generation of more sustainable nuclear reactors. However, it is clear that no matter what the efficiency and effectiveness of these separation and transmutation processes, there will always be some ultimate waste that must be disposed of by geological disposal.

2 Supporting Research in Geological Disposal

There has of course been a marked shift in focus of the research in geological disposal over the period of the Euratom FPs. Initially work concentrated on more fundamental aspects of the physical, chemical and geological processes affecting deep disposal. Projects tended to be smaller and there was less emphasis on technology and engineering. With the construction of dedicated underground research laboratories (URLs) in the various host rock environments, research projects have become more focussed on the specific conditions prevailing underground, the engineered barriers, the required engineering systems and associated demonstration experiments and the overall performance assessment. Since there are only relatively few URLs in Europe, they have naturally become magnets for all EU research in host rock conditions, which in turn has resulted in enhanced co-operation between research teams and waste agencies in different EU Member States. The research often involves large and costly experiments, again encouraging interested research teams to combine efforts in order to reduce costs. The key role played by URLs means that they are also important focal points for Euratom FP funding.

2.1 FP4 and FP5

By the time of FP5 (1998 – 2002), all key areas of research in geological disposal were under investigation in the EU programme. Many of the projects followed on from research supported in FP4, with an important development towards large-scale 1:1

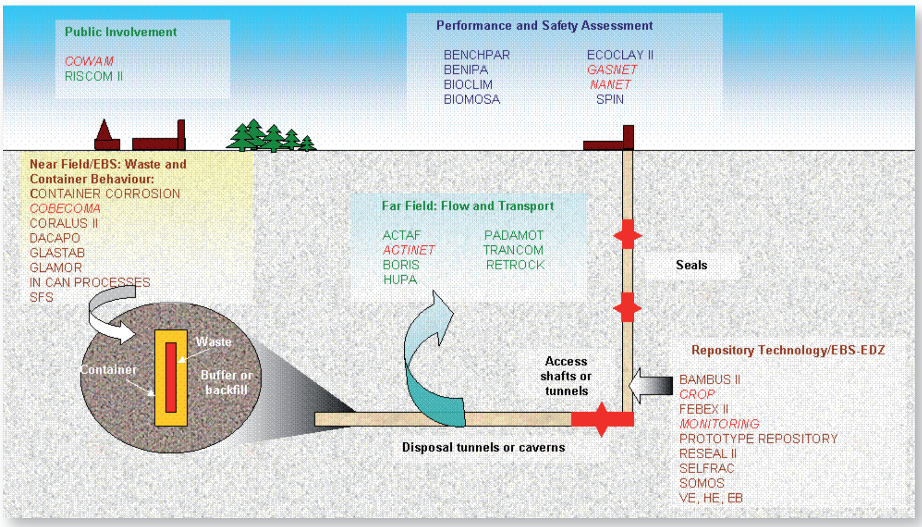

Fig. 1: FP5 coverage of geological disposal – names in red indicate projects aimed at coordination and networking (not all projects shown; see [2] and [3] for project details)

demonstration of processes such as heating and hydration in the actual rock environment. FP5 also saw the first projects studying societal and public involvement issues associated with waste management, in particular new ways of dealing with and communicating risk, and local democracy and governance issues associated principally with site selection. The inclusion of this topic (since retained in FP6 and also in the scope of FP7) is indicative of a recognised need to deal with waste management issues on a more holistic rather than purely technical level, and is a reaffirmation of the importance of political and strategic considerations in the planning of future research programmes. The extent of the FP5 research effort is depicted schematically in Figure 1.

The following headings summarise the principal achievements of FP4 and FP5 in the main thematic areas.

2.1.1 Development of Repository Technology

As a result of several large-scale tests (heating, hydration, etc.) carried out under real conditions in URLs, repository designers and engineers are now increasingly able to develop operational plans for repositories that can be refined over the years before construction work begins. Much has also been learned that will be useful when considering how to monitor real repositories through their operational lives and beyond. This includes both the technical and societal requirements for such activities – a truly cross-cutting issue where there are diverse views on what is needed and how and when monitoring data could be used. This is closely linked to the whole issue of "phasing" repository development so as to establish a broad consensus of confidence before moving from one step of the disposal process to the next. Many of these issues are philosophical in nature, but each has a practical implication. In many cases, there is no requirement to reach decisions today, but it is important to realise that choices will always exist and it is the responsibility of society to be aware of the issues and be prepared to take decisions when required. This is closely linked to the issue of retrievability, which is increasingly being incorporated as a requirement to be studied in national programmes.

2.1.2 Long-term Behaviour of Wastes and Containers

Spent fuel and HLW are extremely stable materials and will remain so in the deep, stable environment provided by geological disposal. As a result of many years of study, their properties and behaviour are well understood. There are a range of suitable metals in which to contain these wastes, enabling them to be safely deposited and isolated inside the buffer and the rock of a deep repository.

2.1.3 Groundwater and Radionuclide Movement around Repositories

Overall understanding of the most important aspects of radionuclide chemistry in natural waters continues to improve and important gaps in this knowledge, where there have been unanswered questions for many years, are being filled – one key example being colloid behaviour. In the future, new analytical techniques will provide ever-increasingly detailed information, thereby enabling refinement in the models and in the understanding of processes. However, in the meantime, the safety assessment modellers are generally satisfied with the adequacy with which they can represent chemical transport aspects.

2.1.4 Safety Assessment of Geological Disposal

Safety assessment is an established, everyday activity. Today, it takes a practical and conservative approach by overestimating impacts, but future developments will allow it to become more realistic. In particular, these developments will enable a better treatment of uncertainties and refinement of the coupling of the various processes involved in geological disposal of high-level waste.

2.1.5 Public Involvement in Repository Programmes

Controversial construction projects will always be affected by the NIMBY ("not in my backyard") syndrome – none more so than radioactive waste facilities. Experience has taught the implementing organisations and the proponents in general that gaining the trust of the local populations is essential, and this can only be achieved via constructive dialogue, transparency, and involvement and empowerment of local communities in the decision-making process. However, this remains a very complex socio-political issue, which continues to be linked, especially by opposition groups, to the continued use of nuclear power.

2.2 FP6 – The new Funding Instruments

In FP6 (2002 – 2006) the EC introduced new funding instruments to enable a more efficient and effective structuring of EU research, to reduce fragmentation and to promote European centres of excellence and mobility of researchers, all in line with the objectives of the European Research Area (ERA). To attract EU support, research groups were encouraged to join forces in collaborative partnerships called Networks of Excellence (NoE) and Integrated Projects (IP). An NoE is a means to promote sustainable integration of key research organisations in a given field. An IP on the other hand focuses more on the product rather than the process, bringing together key research players in an ambitious project to go beyond the current state of the art.

On a purely technical level, the aims of research in geological disposal, as stated in the Council Decision establishing FP6 Euratom, were the establishing of a sound scientific basis for demonstrating the safety and feasibility of geological disposal. However, the Euratom programme also actively encourages more cooperation between research bodies in Europe. Table 1 shows the trends and programme emphasis in recent FPs.

Table 1: The changing emphasis in geological disposal – comparison of the last four FPs

Framework Programme	Total Euratom contribution [€]	No. of projects	No. of projects aimed at coordination and networking	Programme emphasis[1]
FP4 (1994-1998)	33.5 M	42	2	RS
FP5 (1998-2002)	29.0 M	43	10	RT, RS
FP6 (2002-2006)	45.0 M	17	all major projects	I&N, RT, RS
FP7 (2007-2011)	-	-	all major projects	I&N, PA, LI

[1]RS = repository system behaviour (near-field / far-field basic phenomena); RT = repository technology / URLs; I&N = integration and networking; PA + LI = performance assessment & licensing issues

The introduction of the new funding instruments in FP6 was an opportunity to improve still further the degree of collaboration between research players. During FP6, five major research projects, totalling some €32M of EU funding, were launched in the field of geological disposal; four of these are large IPs and the fifth is an NoE (see Tables 2 and 3). The IPs cover the four principal areas in which achievements of previous FPs have been listed in 2.1.1 – 2.1.4: engineering and repository design; near-field behaviour; far-field studies; performance and safety assessment.

Though there was initial reticence on the part of the research community to go along this route of large multi-partner projects, and indeed the added administrative burden has occasionally been considerable, there is nonetheless consensus regarding the overall benefits, especially from the point of view of increased networking, integration of practices and results and the development of an harmonised EU vision on the key issues in geological disposal. The first of the major projects will come to an end in the next few months, and the others will follow over the next year or so. These projects are not only pushing back the frontiers of knowledge in Europe, therefore providing a firm scientific basis on which to proceed towards implementation of actual disposal systems over the coming decade, but are also greatly enhancing the effectiveness and the efficiency of the overall research effort in this field. Both these aspects must be capitalised upon during FP7.

Table 2: On-going FP6 Integrated Projects and Networks of Excellence in the field of geological disposal

Project title & description[1]	Instrument[2]	Coordinating organisation	Number of consortium partners[3]	EU contribution / total cost [€]	Start date & duration
NF-PRO – Understanding and physical and numerical modelling of the key processes in the near-field and their coupling for different host rocks and repository strategies www.nf-pro.org	IP	SCK.CEN (B)	40 (10)	8M / 16.8M	1/1/04 4 years
ESDRED – Engineering Studies and Demonstrations of Repository Designs www.esdred.info	IP	ANDRA (FR)	13 (9)	7.32M / 18.1M	1/2/04 5 years
ACTINET-6 – Network for Actinide Sciences www.actinet-network.org	NoE	CEA (FR)	27 (13)	6.35M / 10.5M	1/3/04 4 years
FUNMIG – Fundamental processes of radionuclide migration www.funmig.com	IP	FZK-INE (DE)	51 (15)	8M / 15M	1/1/05 4 years
PAMINA – Performance Assessment Methodologies in Application to Guide the Development of the Safety Case www.ip-pamina.eu	IP	GSF (DE)	25 (10)	4M / 7.62M	1/10/06 38 months

[1]Refer to http://cordis.europa.eu/fp6-euratom/projects.htm (click on "management of radioactive waste") for more complete information
[2]NoE = Network of Excellence; IP = Integrated Project
[3]The figures in parentheses indicate number of different European countries represented.

Table 3: Brief descriptions of the major on-going FP6 projects

NF-PRO has been investigating dominant processes and their couplings affecting the isolation of nuclear waste within the near-field. It has been applying and developing conceptual and mathematical models for predicting the source-term release of radionuclides from the near-field to the far-field. Results and conclusions of experimental and modelling work are being integrated in performance assessments. To understand the performance of the overall near-field system, an adequate insight in both the performance of the individual near-field sub-systems and their interactions is essential and this constitutes the core of the integration component of the project. The consortium of 40 partners represents 7 European waste management agencies, 25 research institutions and 8 Universities.

ESDRED has the overall objective of demonstrating the technical feasibility of deep disposal on an industrial scale, especially as regards the activities required during construction, operation and closure of a deep geological repository. The project will also show how these activities comply with requirements regarding long-term safety, operational safety, safeguards and monitoring, and is a joint research effort by the major European radioactive waste management agencies (or their subsidiaries).

ACTINET-6 is encouraging sustainable integration of European research on the physics and chemistry of actinides. The goals are, more specifically, to co-ordinate the use of the major actinide research facilities within the European scientific community, improve human mobility between member institutions (in particular between academic institutions and national laboratories) and to promote excellence through a process of selecting R&D projects and support for training activities. ACTINET-6 has a broad participation of research organisations and academic institutions with expertise in actinides science, as well as effective links with the user community.

FUNMIG is a complement to NF-PRO, the main objectives being the fundamental understanding of radionuclide migration processes in the geosphere, their application to performance assessment and the communication of the results. An understanding of processes involved in the transport of key radionuclides and their retardation at the molecular level is fundamental, but this must be scaled up to the dimension of host rock strata being considered in Europe (clay, granite, salt). The migration processes can then be studied at scales of interest in performance assessment (PA), and this integration and abstraction to PA are key issues. The knowledge acquired during the project will be disseminated to the wider scientific community and other stakeholders by active training and other dedicated knowledge management activities. A large consortium of research organisations, waste management agencies and universities across Europe are implementing the project.

PAMINA is the most recent of the large projects to be launched that integrates key activities undertaken in previous FPs. It has the objective of improving and harmonising integrated performance assessment (PA) methodologies and tools for various disposal concepts of long-lived radioactive waste and spent nuclear fuel in different deep geological environments. The IP PAMINA aims at providing a sound methodological and scientific basis for demonstrating the safety of deep geological disposal of such wastes, that will be of value to all national radioactive waste management programmes, regardless of waste type, repository design, and stage that has been reached in PA and safety case development. The project is organised into four components oriented towards research and technological development and one component oriented towards training and transfer of knowledge. There are important synergies with the output from NF-PRO and FUNMIG.

2.3 FP7 and the Future

The geological disposal of high-level and long-lived radioactive waste remains a priority research area in the latest Euratom programme – FP7 (2007-2011, [4]). However, unlike previous Euratom FPs, there are no ring-fenced budgets for the various priority areas. This will maximise flexibility in programme implementation and ensure that limited funds are allocated as effectively as possible. In particular, since there are large projects still on-going in all programme areas, with two or three years left to run, it was important not to earmark FP7 funding without having a clear idea of the results from these projects and where there may be a need for continued support at EU level. This includes in the area of geological disposal.

The Euratom FP7 Specific Programme for nuclear research and training activities, approved unanimously by the EU Member States towards the end of 2006, has this to say on the scope of the research to be supported in the field of geological disposal:

"Research and technological development in the field of geological disposal of high-level and/or long-lived radioactive waste involving engineering studies and demonstration of repository designs, in-situ characterisation of repository host rocks (in both generic and site-specific underground research laboratories), understanding of the repository environment, studies on relevant processes in the near field (waste form and engineered barriers) and far-field (bedrock and pathways to the biosphere), development of robust methodologies for performance and safety assessment and investigation of governance and societal issues related to public acceptance."

This, therefore, represents a progression from the objectives of previous programmes, with the emphasis now much more on "implementation oriented" research. In the light of the progress being made during FP6, and also in the socio-political arena in a number of countries, these clearly defined objectives in FP7 represent the next logical phase of support to the development of geological disposal in Europe.

The EC's technical services responsible for the Euratom FP are also considering how future EU funding in the general field of nuclear science and technology can be spent most effectively. In the area of geological disposal, the most important stakeholders are the national radioactive waste management agencies, responsible for the management of national waste arisings and the implementing of national programmes. As such, they are the "drivers" behind the lion's share of the research effort in the EU and are involved in the majority of the projects at EU level, certainly those involving research in URLs.

However, it should be appreciated that all agencies are under constraints imposed on them by their national circumstances. Important differences exist between countries as a result of the varying degrees of progress in the development of disposal systems, the choice of host rock and the differing regulatory requirements imposed by governments and national safety authorities. The most marked differences are between countries with identified sites and those with no clear policy regarding long-term management. This highlights the need for decisions to be taken at the political and strategic level in several Member States as well as at the level of research priorities.

However, the trend is clear. Even in those countries lagging behind in the development of national repositories, but where more extensive stakeholder consultation has been undertaken, the end result has always been that geological disposal is the only feasible end point for much of today's nuclear waste. Of course, the process of getting from this statement in principle to actual implementation is long and fraught with difficulties. Importantly, even though the research conducted to date has demonstrated that technical issues are not the most serious difficulties to be overcome in this process, there is still progress to be made in a number of areas, and the Euratom programme will continue to provide important support where necessary.

2.3.1 Introducing Technology Platforms

A Technology Platform (TP) brings together key stakeholders – industry, academia, regulatory authorities, research community, national research coordinators – in a particular sector of research. The resulting "forum" is then responsible for the establishing and implementation of a strategic research agenda (SRA) in this sector. The TP members decide amongst themselves how best to conduct future research and must each contribute significant resources in the implementation of the SRA. Clearly the stakeholders need to have a shared vision regarding the direction in which the research should go and be willing to collaborate in order to further the platform's agenda.

A TP belongs to its stakeholders, not to the EC, though the EC can be instrumental in providing the initial impetus and high-level political support needed at start-up. Initially, the stakeholders usually come together over the endorsement of a "vision" document drafted by a high-level group at the request of the EC. Once the TP has been officially launched and its governance structure established, the important work of drafting the SRA can begin. Later, this can be used by the EC to orient the FP calls for proposals in this field. In this way, the EU can bring a significant degree of support to the platform's activities.

There are currently some 30 or so European TPs, covering a range of technical and scientific domains. Each is helping to coordinate and integrate EU research in its respective field, with industry or the end-users of the research being key driving forces. Recently, on 21 September 2007, the first TP in the nuclear field was formally launched in Brussels. The Sustainable Nuclear Energy TP (SNE-TP [5]) brings together all the principal nuclear research and industrial stakeholders in Europe in a broad-based TP covering the whole sector of nuclear systems and safety, including cross-cutting issues such as research infrastructures and education and training. Industry, including nuclear suppliers, utilities and large users of electricity are all on board, as are the major research organisations and institutes, academia and the Technical Safety Organisations (TSOs). The vision document [5], endorsed at a high level by the stakeholders, presents the prospects for developing nuclear technology in the coming years, with a special emphasis on increased sustainability through the use of fast breeder reactors, cogeneration of electricity and process heat using very-high temperature reactors, and the continued safe operation of current light-water reactors. The whole of the fuel cycle is included in the scope of SNE-TP, with the exception of geological disposal. This exclusion is quite deliberate and reflects the sensitive nature of the disposal issue and the need for the implementing organisations – the national radioactive waste management agencies – to keep their distance from any activities linked to promotion of nuclear technology in order to maximise credibility and trust in the eyes of the local population at potential host sites.

Nonetheless, in the field of geological disposal there are several aspects of the research effort in Europe that lend themselves to coordination via a „TP-like" structure. First and foremost, the main end-users, the waste management agencies, are well defined and there is broad agreement amongst them on the common objectives. This vision is also shared amongst the other research stakeholders, for example the principal research institutes, and is reflected in national programmes (even though geological disposal is not yet officially recognised as the favoured management option in one or two countries). There is also a good degree of co-operation in ongoing projects, and the relatively small number of URLs also promote a converging of national research programmes. A TP could also enable an exchange of experiences, sharing of technology and planning of research tasks of common interest, as well as identifying issues that are of purely national or bilateral interest.

The important work carried out in the FP5 project NET.EXCEL [6] has shown that a greater degree of integration is possible. A follow-up study – the CARD project [7] – was launched in November 2006 and will elaborate further the possibility of establishing a TP in geological disposal. It will report back at the end of 2007. Any structure to enhance coordination must be capable of handling the different requirements and speeds of the various national programmes. Another important aspect is the involvement of the major research institutes as well as the TSOs, who work closely with the regulatory authorities.

3 Conclusions

The earliest work on geological disposal was undertaken in the USA in the 1950s and 60s, when deep salt formations were first considered as host rocks for high-level radioactive waste repositories. In 1975, the Euratom programme identified potentially suitable rock formations in Europe, producing an atlas of hard igneous and metamorphic rocks (such as granite and gneiss), clay-rich rocks and salt formations that might be considered for disposal. All were selected on the basis of their stability, low permeability and good containment properties. Since then, work has been focused on these three geological environments and European countries have progressively developed their own active R&D programmes, supported by the Euratom programme. Basic R&D, both in the field and in the laboratory, has been built upon by practical tests and experiments in specially constructed URLs that have now been operating for more than 20 years.

The steps from concept to implementation will therefore take many decades and the further operational steps leading to final closure of geological repositories are expected to take at least as long. Compared with other industrial or environmental developments, geological disposal programmes evolve slowly and cautiously, reflecting not only the complexity and multidisciplinary nature of the science involved, but also the need for time-consuming demonstration experiments in the host rock environment. In addition, delays can also result from a “wait and see” attitude by those in authority as a result of the low volumes of waste involved and the relative ease with which these can be temporarily stored. However, delays are now largely a result of the complex socio-political issues at stake, in particular the lack of public and therefore political acceptance resulting from an often inexact and irrational public perception and the link with continued nuclear power, which is widely exploited by opponents of this technology.

These political and social problems are most crucial when it concerns the actual selecting of repository sites. This atmosphere was prevalent throughout the 1980s and 1990s. The first national repository programmes to have overcome these setbacks are in Finland and Sweden, and both countries should have operating disposal facilities by 2020. In Finland, mining operations at the actual repository site have been underway for more than 3 years. In addition, since 2006, there is a legal framework in place in France that should allow its national repository to be in operation by 2025. Other countries have engaged in consultation and review processes that have led to the selection of phased geological disposal as the reference solution for the management of their high-level long-lived radioactive waste.

The Euratom FP continues to provide an essential service to Member States in the field of geological disposal and is encouraging greater collaboration between waste agencies and other research organisations. The current Euratom FP projects are ambitious and far reaching undertakings that, through significant investments of public money, are contributing to major advances in the research effort as well as bringing about a restructuring of the research community in this field.

Working together within Euratom programmes has provided tremendous added value by bringing together numerous academic and professional scientists, waste disposal agencies and regulatory bodies, and the involvement of the EC has resulted in wide and efficient dissemination of results to end-users. Today, as the geological disposal concept moves towards maturity and implementation, the emphasis is very much on integration of efforts in order to rationalise and optimise solutions that can be achieved in Europe. The Euratom programme is at the heart of the these initiatives.

To achieve this enhanced degree of collaboration, a suitable mechanism must be found that caters for national programmes of different speeds, but also allows the slower Member States to benefit from advances in other countries. These general issues of knowledge management are crucial to this future research effort, and one possible mechanism could be a Technology Platform, which has already been widely implemented in other domains of EU research. In this regard, the support of the waste management agencies and other key European research players is essential. ■

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[4] FP7 Euratom and SP on Nuclear Research available from http://cordis.europa.eu/fp7/find-doc_en.html

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[7] Refer to http://cordis.europa.eu/fp6-euratom/projects.htm (click on „management of radioactive waste"; projects are listed alphabetically)

1.02 The Application of IAEA Safety Standards in the Management of High Level Waste

Phil Metcalf
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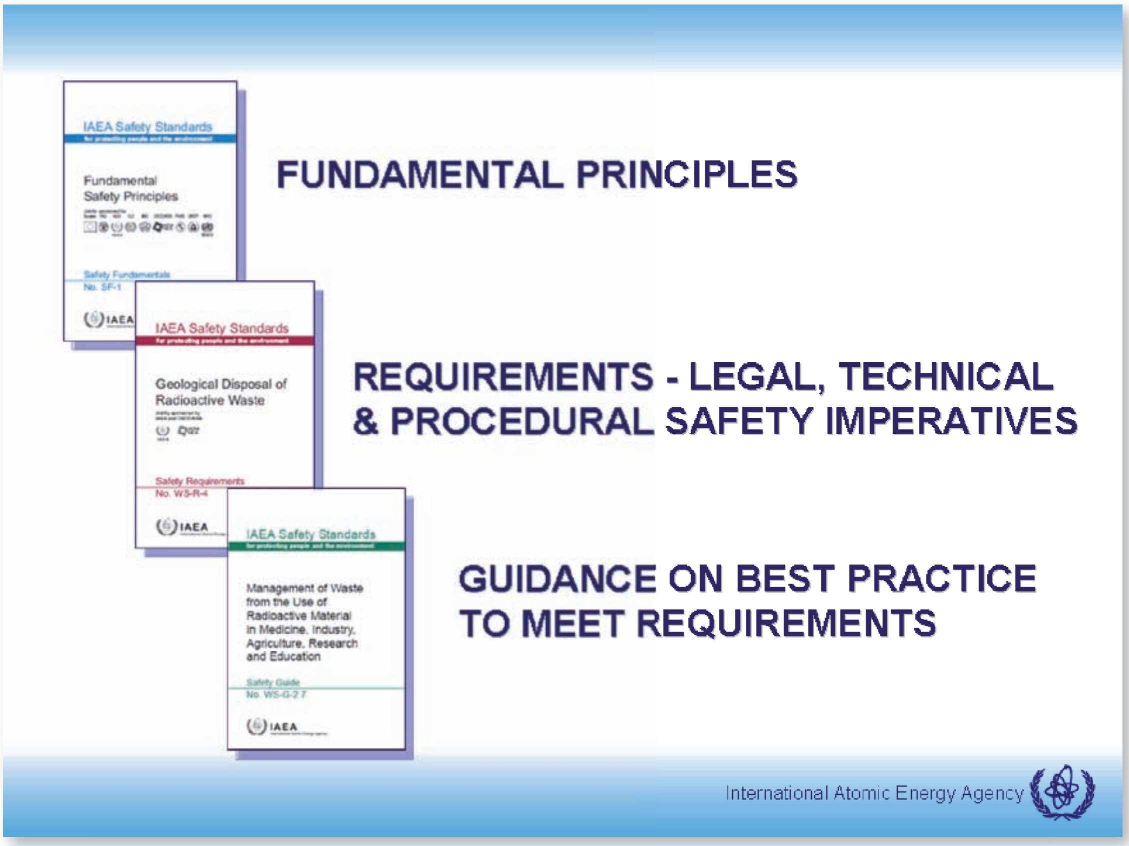
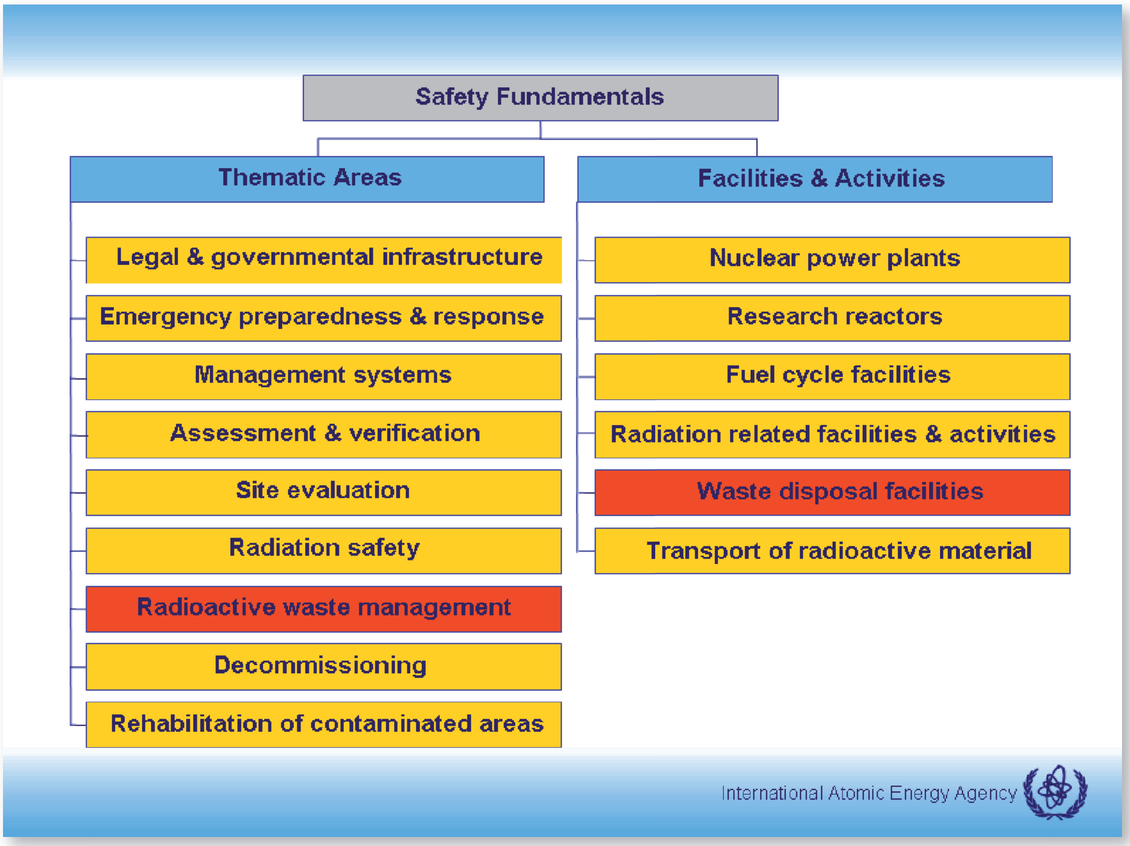
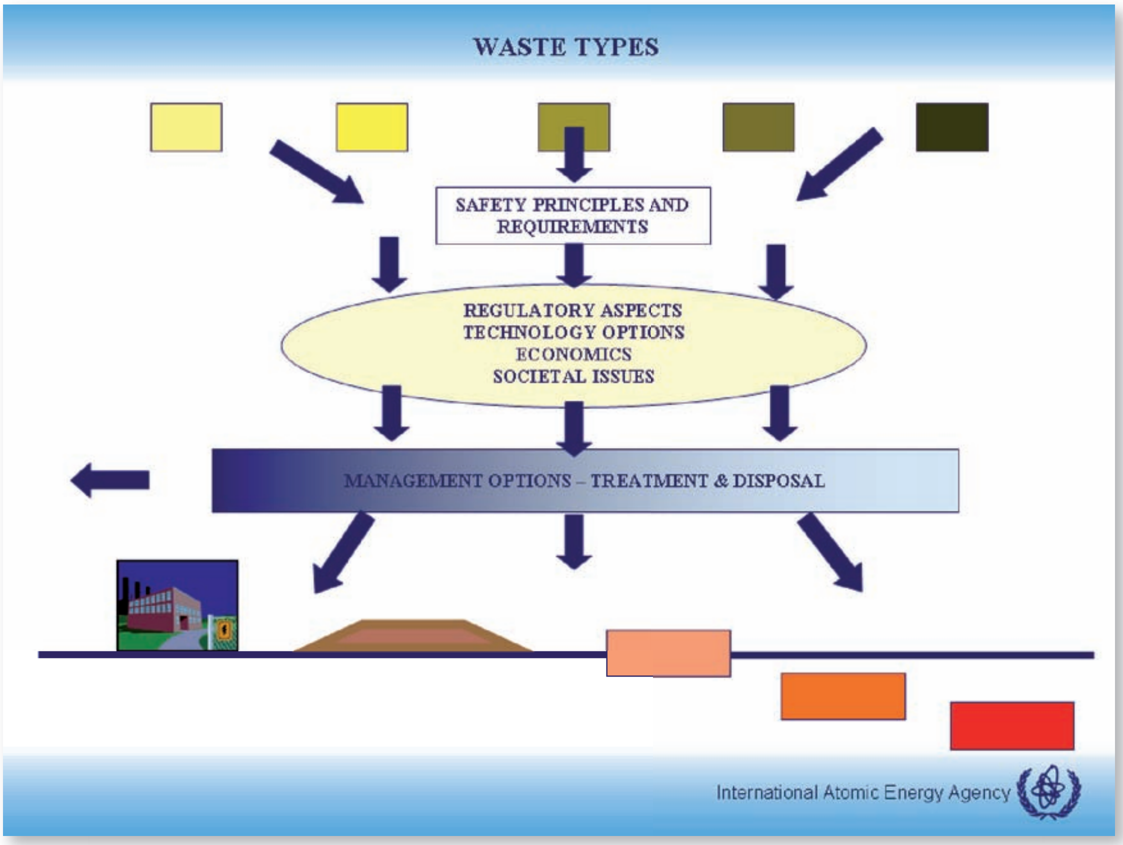
INTERNATIONAL SAFETY STANDARDS
International Atomic Energy Agency


IAEA Safety Standards
Fundamental Safety Principles
Safety Fundamentals
No. SF-1
IAEA Safety Standards
Geological Disposal of Radioactive Waste
Safety Requirements
No. WS-R-4
IAEA Safety Standards
Management of Waste from the Use of Radioactive Material in Medicine, Industry, Agriculture, Research and Education
Safety Guide
No. WS-G-2.7
FUNDAMENTAL PRINCIPLES
REQUIREMENTS - LEGAL, TECHNICAL & PROCEDURAL SAFETY IMPERATIVES
GUIDANCE ON BEST PRACTICE TO MEET REQUIREMENTS
International Atomic Energy Agency


Safety Fundamentals
Thematic Areas
Facilities & Activities
Legal & governmental infrastructure
Emergency preparedness & response
Management systems
Assessment & verification
Site evaluation
Radiation safety
Radioactive waste management
Decommissioning
Rehabilitation of contaminated areas
Nuclear power plants
Research reactors
Fuel cycle facilities
Radiation related facilities & activities
Waste disposal facilities
Transport of radioactive material
International Atomic Energy Agency

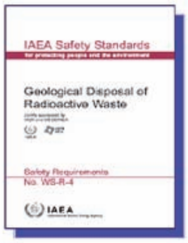

Safety Standards- Pre-disposal
Management
DS 353
WS-R-2
LILW
WS-G-2.5
HLW
WS-G-2.6
Small Amounts
WS-G-2.7
Storage
WS-G-6.1
Management NORM Residues
DS 352
Classification of Radioactive Waste
DS 390
Management Systems Pre-disposal
DS 337
Safety Assessment Pre-disposal
DS 284
International Atomic Energy Agency


WASTE TYPES
SAFETY PRINCIPLES AND REQUIREMENTS
REGULATORY ASPECTS
TECHNOLOGY OPTIONS
ECONOMICS
SOCIETAL ISSUES
MANAGEMENT OPTIONS – TREATMENT & DISPOSAL
International Atomic Energy Agency


Safety Standards- Disposal
Disposal
DS 354
Classification of Radioactive Waste
DS 390
Near Surface Disposal
DS 356
Borehole Disposal
DS 335
Geological Disposal
DS 334
Management of Norm Residues
DS 352
Safety Case & Assessment Disposal
DS 355
Monitoring and Surveillance
DS 357
Management Systems Disposal
DS 337
International Atomic Energy Agency

Safety Requirements – Geological Disposal

- Planning
  - Legal and organisational
  - Safety approach
  - Safety design principles
- Development, operation and closure
  - Framework for disposal facility development
  - Safety case and safety assessment
  - Steps in the development of a facility
- Assurance
  - Management system
  - Limits, conditions controls e.g. WAC
  - Regulatory control



International Atomic Energy Agency

Safety Guide – Safety Case and Safety Assessment for Disposal

- INTRODUCTION
- SAFETY FUNDAMENTALS AND REQUIREMENTS
- SAFETY CASE FOR DISPOSAL
- SAFETY ASSESSMENT
- SPECIFIC ISSUES
  - Graded approach
  - Defence in depth
  - Timeframes
  - Intrusion
  - Institutional control
  - Retrievability
- REGULATORY REVIEW PROCESS
- ANNEX I; CHARACTERISATION AND ANALYSIS OF UNCERTAINTIES
- ANNEX II; CONFIDENCE IN EACH STAGE OF THE SAFETY ASSESSMENT

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USE OF SAFETY STANDARDS

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- DEVELOPMENT AND MAINTENANCE OF NATIONAL REGULATIONS & REGULATORY GUIDANCE
- REGULATORY COMPLIANCE ASSURANCE ACTIVITIES
- JOINT CONVENTION
- INTERNATIONAL PEER REVIEW
- TECHNICAL ASSISTANCE
- COORDINATED RESEARCH PROJECTS
- INFORMATION EXCHANGE

International Atomic Energy Agency

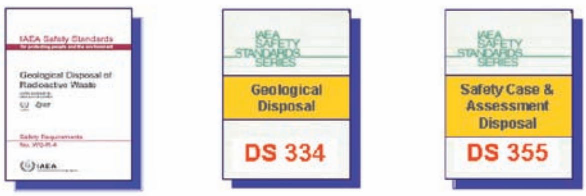
WENRA

- Safety reference levels for
  - Radioactive waste and spent fuel storage
  - Decommissioning
- Based on IAEA Safety Standards
- Liaison IAEA Secretariat and WENRA Working Group on Waste and Decommissioning (WGWD)

International Atomic Energy Agency

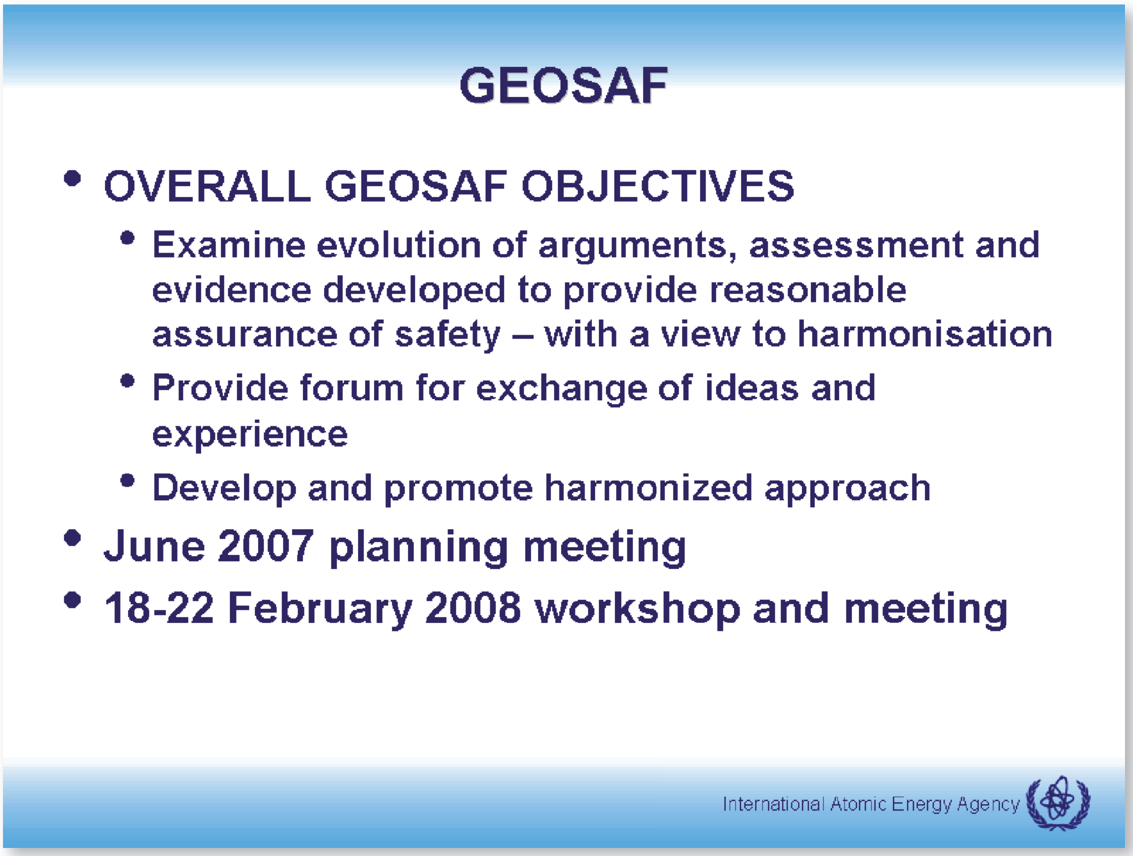
GEOSAF

- Increasing interest in safety demonstration for geological disposal



- European initiative on safety of geological disposal – harmonised approach
- Success of ISAM/ASAM, SADRWMS, DeSa, EMRAS

International Atomic Energy Agency


GEOSAF
• OVERALL GEOSAF OBJECTIVES
• Examine evolution of arguments, assessment and evidence developed to provide reasonable assurance of safety – with a view to harmonisation
• Provide forum for exchange of ideas and experience
• Develop and promote harmonized approach
• June 2007 planning meeting
• 18-22 February 2008 workshop and meeting
International Atomic Energy Agency

1.03 Demonstrating and Communicating Safety of Geological Disposal

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Abstract
It has been more than 15 years since NEA concluded that the methods were available to adequately evaluate safety and make decisions regarding geological radioactive waste disposal systems. In the intervening years, further international work by NEA has affirmed these conclusions and contributed significantly to defining fundamental concepts of safety, advancing the technical and scientific basis---as well as the methodologies---for evaluating safety, and exploring the societal factors that contribute to confidence in safety. All these aspects converge in safety cases, which have evolved into tools to both assess safety and aid in decision making. This presentation describes the modern concept of safety cases, including a review of recent advances in safety cases.

1 Introduction

Central to successfully implementing long-term deep disposal of radioactive waste is the ability to evaluate and illustrate the safety of a disposal system after closure and far into the future in a manner that is clear, scientifically sound, and persuasive to decision-makers and the public. The Radioactive Waste Management Committee (RWMC) of the OECD Nuclear Energy Agency (NEA), a unique world wide forum of senior regulators, implementers, policy makers and managers of R&D institutions, and its Integration Group for the Safety Case (IGSC) have been working for many years in this area and have been instrumental in broadening the scope of classical safety assessments towards a new broader concept of a safety case.

1.1 Endpoint for Long-term Radioactive Waste Management

The technical concepts for the management of high-level radioactive waste differ largely between the various NEA countries. However, they all share some common views regarding the ethical framework for long-term management of radioactive waste as laid down in the Rio Declaration, the Joint Convention on waste and spent fuel management, and the IAEA Safety Standards; and they share a commitment for sustainable development.

The concept of sustainability implies that a waste management strategy should not rely on speculations concerning future societal, scientific or technological developments. To be complete, the strategy must have a well-defined endpoint, and the path (or alternative paths) to reach that endpoint must be specified. There is wide spread agreement that Geological disposal is the only scientifically and technically credible long-term solution, without reliance on active management. Deep underground repositories ensure long-term safety through a system of natural and engineered safety barriers and most nations where long-lived radioactive waste is an issue ultimately aim for geologic disposal.

1.2 Stepwise Decision Making

Today, consideration is increasingly being given to concepts such as “stepwise decision making” and “adaptive staging”, in which the public, and especially the local public, are to be meaningfully involved in the review and planning of repository development. This should be accomplished at all major stages, e.g. repository planning, construction, operation, or even closure. A stepwise approach or adapted staging for repository development recognises the long duration of these kind of projects (> ~ 100 years) and acknowledges that not everything needs to be known from the start.

The work of the Forum on Stakeholder Confidence (FSC), a RWMC group that serves as a venue for sharing international experience in achieving effective dialogue with the public and embraces a wider representation of civil society through national workshops that include local stakeholders clearly showed that the most important issues for stakeholder are the design of the decision-making process, with clearly defined roles and responsibilities, and the assurance of safety.

Basically, stepwise decision making and confidence in decision building for repository development rests on three pillars:

- First, there should be a general agreement regarding the ethical, economic and political aspects of the appropriateness of the underground disposal option;
- Second, there is a need for confidence in the organisational structures, and the legal and regulatory framework for repository development; and
- Third, there should be confidence in the practicality and long-term safety of disposal.

It is this last area, where the Safety Case should provide a platform for a meaningful dialogue with the public, and especially the local public.

1.3 The Safety Case

Over the years, there has been now a notable convergence in documents published nationally and internationally in the development of long-term safety cases for geological disposal. They all show a significant assimilation of the principles and main elements, which allows having a definition of the safety case.

According to a recent OECD/NEA publication, a safety case is defined as “the synthesis of evidence, analyses and arguments that quantify and substantiate a claim that the repository will be safe after closure and beyond the time when active control of the facility can be relied on”. This definition has also been taken up in the IAEA/NEA Safety Requirements for Geological Disposal of Radioactive Waste.

1.4 Main Elements of the Safety Case

To evaluate safety, three main areas should be addressed by the Safety Case:

- The System Concept, that describes how safety should be achieved;
- the Safety Strategy, which shows how safety is proved, and
- the Assessment Basis, which assures the assessment is well founded.

The System Concept provides the high-level integrated approach adopted for achieving safe disposal. The goal should be to arrive at a robust systems, a system characterised by a lack of complex, poorly understood or difficult to characterise features and phenomena, and by an absence of detrimental phenomena arising either internally within the repository, or externally in the form of geological and climatic phenomena.

Two components of a System Concept can be differentiated:

- The overall management strategy of the various activities required for repository planning, implementation and closure, including siting and design, safety assessment, site and waste form characterisation and R&D; and
- the siting and design strategy to select a site and to develop practicable engineering solutions.

The Safety Strategy is the strategy to perform safety assessments and define the approach to evaluate evidence and analyse the evolution of the system. Arguments and analyses for the safety case should be supported, where possible, by multiple lines of evidence. An emphasis should be put on a limited number of processes or features that are well-understood and reliable, such as long-lived corrosion resistant canisters and stable properties of the host rock. The most appropriate ways of quantifying performance or safety may vary with time, as the repository and its environment evolve and different phenomena and uncertainties become relevant. Thus, it may be convenient to divide the post-closure period into a number of discrete "time frames", that are characterised by particular types of phenomena or uncertainties, and for which particular types of indicators or arguments are most suitable.

The Assessment Basis must also be clearly and logically presented. Comprehensive research and site investigation programmes should have been implemented; the assessment methods, models, computer codes and databases should be evaluated and arguments for their reliability should be given. The presentation of scientific data and understanding in a safety case should highlight evidence

- that the information base is consistent, well founded and adequate for the purposes of safety assessment;
- that the diverse sources of information (and methods of acquisition) that have been brought together form a consistent picture of the characteristics and history of a site, from which a reliable prognosis of future evolution can be made.

By addressing these three areas - System Concept, Safety Strategy and Assessment Basis - the Safety Case supports decision making by demonstrating that

- a transparent description of the system and its possible evolutions has been compiled giving adequate confidence to support the decision at hand;
- a strategy exists to deal at later stages with any remaining uncertainties that have the potential to compromise safety;
- confidence exists in the means applied, in the quality of the work presented, and in future steps where uncertainty will be further characterised and reduced.

1.5 Confidence vs. Uncertainty

Uncertainties will inevitably remain, but the safety case should indicate the reasons why these uncertainties do not undermine primary arguments for safety. The connection needs to be made between key uncertainties that have been identified and the specific measures or actions that will be taken to address them.

Many uncertainties can be bounded or even quantified and methods exist to take these uncertainties into account. For data or model uncertainties probabilistic techniques can be used to explore a wide range of scenarios. When it comes to long-term predictions a mixture of quantitative and qualitative analyses and arguments (still science-based) will have to be provided to engender confidence of both the provider and the reviewer.

The extent and depth of the knowledge base needed to support the Safety Case for a specific decision at hand cannot be predefined, but will be established based on a national consensus process involving regulators, implementers, decision makers, researchers and other relevant stakeholders. This process needs to be repeated at further stages of the development process, and at various decision making levels. The depth of understanding and technical information available to support decisions will increase from step to step.

2 Symposium

Over the last decade, the concept of the safety case has continued to evolve and important advances have been made in terms of a much expanded pool of scientific and experimental data; improved understanding of processes at various spatial and temporal scales; advancement of modelling techniques; and better appreciation of the importance of openness, communication, and stakeholder involvement in developing and presenting safety cases. A recent NEA symposium on “Safety Cases for the Deep Disposal of Radioactive Waste: Where do we stand?” reviewed progress and practical experiences at both the technical and managerial level in preparing for and developing safety cases. The symposium, organised in cooperation with the European Union and the International Atomic Energy Agency, provided the opportunity to take stock of recent progress and remaining challenges in evaluating and supporting the safety of long-term disposal of radioactive waste.

The symposium showed that safety cases have evolved into tools to both evaluate and illustrate safety, and to aid in decision-making. There is a good, shared understanding of what a safety case is and what comprise its main elements. Key aspects of this evolution in the past decade include:

- improved and structured documentation to favour clarity and traceability of the argumentation;
- evidence and arguments that showcase the knowledge basis (and scientific understanding) build up by the project;
- the development of more sophisticated analytical tools and databases;
- the introduction of new conceptual tools such as the concept of safety function;
- the utilisation of a breadth of performance and safety indicators besides the traditional dose- and risk-indicators;
- the open discussion—in the safety case itself—of extant issues of concern and the identification of a path forward to their resolution.

The symposium afforded additional verification that the current, shared understanding of the purpose and contents of a safety case allows for better discussions and exchange of experience.

3 Conclusions

A modern Safety Case is an important vehicle to demonstrate and communicate safety in Geologic Disposal; it can be an important tool that supports decision making at various levels of repository development, making confidence tangible.

The development of such a broad concept of a Safety Case for geologic disposal fits timely current needs. At the time being many national waste management programmes are entering key stages in the stepwise decision process towards disposal. Also significant technical progress is reported, e.g. in databases, in the integration of science in safety cases, in strategic areas such as those dealing with timescales in safety cases, and with the definition and judging of compliance with long-term safety criteria.

Given the foreseeable, undiminished importance of the disposal safety case for future decision-making in national programmes, the NEA and its Integration Group for the Safety Case are well positioned to continue to provide a key service to the international community of radioactive waste management for many years to come. NEA has a rich programme of work addressing technical, policy, regulatory, and societal aspects of confidence building. Integration of all the above aspects will be enhanced in line with current developments and trends in the NEA Member countries. ■

1.04 The Decision-making Process for Managing Radioactive Waste in France

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Abstract

Over the last 20 years or so, radioactive-waste management went through remarkable changes. Developments in scientific and technical knowledge helped to provide responses to issues relating to long-term safety. However, failing to be known to both the public at large and political circles, their implementation required changes in governance before any sound decision could be made involving the participation of the different levels of public representatives. After presenting a brief historical background of the subject, this paper addresses with the results generated by the last 15 years of French research, mostly from the standpoint of the support to the decision-making process. The adoption of the Planning Act of 28 June 2006 marks the conclusion of the first step and, obviously the beginning of a new one, which should lead to the application for a reversible deep geological repository for high-level and intermediate-level long-lived waste.

1 Historical Background

The history of radioactive-waste management in France starts in 1969 with the creation of the Centre de stockage de la Manche (CSM), the first disposal facility for low-level and intermediate-level short-lived waste (Figure 1).

◄ Fig. 1: Aerial view of the Centre de la Manche Disposal Facility, closed down in 1994 and currently in its post-closure monitoring phase.

The first waste packages were placed in open-ground trenches on the site. In 1974, the operations were rationalised with the installation of the first disposal structures designed to provide performance requirements that are consistent with the new 1973 nuclear regulations. The significance attributed to the disposal of radioactive-waste packages was confirmed in 1979 by the creation of the French National Radioactive Waste Management Agency (Agence pour la gestion des déchets radioactifs – Andra) within the French Atomic Energy Commission (Commissariat à l'énergie atomique – CEA), shortly after came into effect the London Convention prohibiting all radioactive waste from being dumped into the sea. That turning point led the facility to adopt an industrial scope. The search for a new site was launched, while anticipating the quantity of waste to be generated over the next decades. A site, located at Soulaines, in the Aube District, was selected for its well-suited geological and hydrogeological properties for safe disposal. In 1992, the Centre de l'Aube Disposal Facility (CSA) was built on the site. It is designed to accommodate 1 million cubic metres of waste, which is almost twice the capacity of the CSM, and should remain in operation for about 60 years.

Throughout the same period as the CSA was being designed at the end of the 80s, a question was raised concerning the disposal of waste resulting from the nuclear-power industry. Those residues are processed at the COGEMA Treatment Plant, located in La Hague. They consist mostly in glass matrices containing non-recoverable high-level long-lived materials. Due to safety reasons, their specific radioactive and thermal characteristics do not allow for their disposal of in surface facilities over the long term. Studies were necessary in order to locate suitable formations for deep geological disposal. With that objective in mind various survey campaigns started in 1989 in four different geological media: clay, salt, granite and schist, with the prospect to further investigations in underground laboratories. Activities were rapidly met with opposition, sometimes violent, and came to an abrupt end. The Prime Minister announced a moratorium that led to the adoption of the Law of 30 December 1991, which structured research activities regarding the management of high-level and intermediate-level long-lived waste. The Law also granted Andra the status of an independent establishment from radioactive-waste producers. Over the following 15 years, the Agency pursued not only its research programmes, but also its industrial mission relating to the design, construction and operation of radioactive-waste disposal facilities. During the summer of 2003, a disposal facility for very-low-level waste (CSTFA) was commissioned at Morvilliers, close to the CSA.

2 The Law of 1991: A Political Leap Forward

2.1 The Evolution of the Structure

The crisis that the first attempts to implement research sites for the disposal of high-level and intermediate-level long-lived waste generated in 1989 marked a major turning point in France. Until then, only technicians and scientists working in the nuclear sector had been dealing with radioactive-waste management, when it suddenly became a societal issue. With the mission entrusted upon the Parliamentary Office for Scientific and Technological Options (Office parlementaire pour l'évaluation des choix scientifiques et technologiques – OPECST), followed by the adoption of the Law of 30 December 1991, radioactive-waste management gained a national scope under the supervision of Parliament. A formal structure was given to research and investigations on radioactive waste with a strict framework and clear deadlines. Roles, missions and responsibilities of every stakeholder were clarified.

Within that context, Andra became a public establishment, as a structure allowing the State to manage radioactive waste, as well as to design and to carry out research programmes. Consequently, the Agency lost its reporting relationship with the CEA and was independent from waste producers. Through its creative financing, it constitutes an original structure, since it operates without any contribution from the State Budget, except for the mission dealing with the preparation of the National Radioactive Waste Inventory. Indeed, all industrial and research activities are financed through agreements with waste producers in accordance with the "polluter-pays" principle.

2.2 Governance and decision-making Process

Governance in matters relating to radioactive-waste management also evolved and new intervention modes were developed. First and foremost, by being removed from the jurisdiction of the CEA, the leadership for research regarding the disposal of radioactive waste promoted the mobilisation of the overall scientific community around the different issues raised by the management of such waste. Programmes and results are placed under the control of a national review board that reports directly to the government and Parliament. The new mechanism involves a high level of transparency as a driving force for the intensification of research programmes and especially for the quality of methodological developments and scientific knowledge. Throughout the events that followed, it was clear that mobilising various skills proved to be one of the richness and success factors of the research programme.

Secondly, the Law of 30 December 1991 was also innovative with respect to the decision-making process. It set up an open and progressive process with a 15-year research phase at the end of which a new deadline was prescribed to hold a national discussion with a view to determining future steps. In the light of the results achieved with the different alternative solutions under consideration (partitioning and transmutation, long-term storage), the decision was taken that the implementation of a repository for high-level and intermediate-level long-lived waste was feasible.

Andra's missions were clarified and focus on:

- industrial activities, including the design, construction and operation of radioactive-waste disposal facilities, followed by their closure and post-closure monitoring;
- information activities, including especially the preparation and publication of the National Radioactive Waste Inventory, listing all radioactive waste in France;
- research activities, mostly in order to implement a deep geological repository for high-level and intermediate-level long-lived waste.

3 15 Years of Research

3.1 The Search for Sites

Following the adoption of the Law of 30 December 1991 and the publication of the implementing Order, the government entrusted upon Mr Christian Bataille, Member of the French National Assembly, a consultation mission aimed at searching for suitable sites for the construction of underground laboratories where investigations could be carried out on the deep geological disposal of high-level and intermediate-level long-lied waste. In 1993, four candidate sites were selected among approximately 30 volunteer communities:

- one site located in a granite formation, under a sedimentary cover, in the Vienne District;
- one site in deep marls located in the Gard District, near the Rhône River;
- two sites located in Callovo-Oxfordian argillites, in the Meuse and Haute-Marne Districts respectively; due to their proximity, both sites were quickly combined to form the Meuse/Haute-Marne Site.

Detailed investigations were conducted from the surface. Borehole-drilling operations and geophysical-measurement campaigns were launched as early as 1994 and lasted for 2 years. The qualities of each site could be verified during this phase. A 400-m-thick formation in the subsoil of the Gard District, near the Rhône River, was also discovered during those investigations.

Local Information and Oversight Committees (commission locale d'information et de suivi – CLIS) together with incentive funds were implemented on every site during that period. Following the studies it conducted, Andra submitted three applications to authorise the implementation of an underground laboratory, in the Vienne, Gard and Meuse/Haute-Marne Districts. Those applications took stock of all existing information concerning the three sites, described the work programmes involved with the implementation of underground laboratories, as well as the research and experimentation programmes intended to complement the body of required data in order to meet the 2006 deadline, as prescribed by the Law of 1991.

In 1997, the applications were the subject of public inquiries. All candidate communities confirmed their willingness to host an underground laboratory and therefore agreed to its construction.

3.2 The Decision concerning the Underground Laboratory

Once the applications were reviewed by relevant services, the government decided to continue work activities on the Meuse/Haute-Marne Site by implementing an underground laboratory at Bure. On the other hand, both the Gard and Vienne Sites were abandoned. In the latter case, the National Review Board (Commission nationale d'évaluation – CNE) had reservations because the sedimentary series covering the granite block contains water resources that are used for agricultural purposes. As regards the clay formation, the Callovo-Oxfordian formation on the Meuse/Haute-Marne Site is better known and its geometry is more suitable than the Gard Site. In parallel, the government instituted a research mission to seek a new granite site. However, the mission did not meet local support and was confronted with a large number of demonstrations, so much so that it folded in 1999. In the meantime, preparatory activities were undertaken for the construction of the Meuse/Haute-Marne Underground Research Laboratory. The construction and the first experiments took place between 2001 and 2006; they are described in the Dossier 2005 Argile [1], which was submitted to the government in mid-2005 (Figure 2).

◄ Fig. 2: Experimental drift at a depth of 490 m in the Meuse/Haute-Marne Underground Laboratory.

During the same period, the Dossier 2005 Granite was also prepared on the basis of the overall information gathered on that type of geological formation by integrating the experience acquired through foreign underground laboratories and programmes in Canada, Switzerland, Sweden and Finland.

The Dossiers presented to the government contain a compilation of the overall information acquired on the geological formations, waste packages and potential means for the implementation of a deep geological repository.

Analyses demonstrate the feasibility of a repository within the Callovo-Oxfordian formation; its reversibility may be guaranteed beyond a century. Safety functions consisting in retarding radionuclide migration are satisfactory. Safety calculations show that the radioactivity level likely to be released into the human environment is several orders of magnitude lower than the regulatory limit, with a peak at a few hundreds of thousands of years.

4 Decision-making Steps

4.1 Assessments

During the preparation of the Dossier 2005, Andra called upon various personalities among the French scientific community and representatives of foreign counterpart agencies in order to carry out a critical review of the main documents. In 2005, the Dossier 2005 Argile was the subject of a triple assessment at the request of public authorities:

- a scientific and technical assessment conducted by the CNE, as prescribed by law;
- a safety assessment performed by the Nuclear Safety Authority (Autorité de sûreté nucléaire – ASN), as part of its prerogatives;

- a peer review carried out at the request of Andra's supervisory ministries by a group of international experts under the aegis of the OECD Nuclear Energy Agency (OECD/NEA) in order to verify the soundness of approaches and results in relationship to international standards.

The CNE continued to monitor the advance of the research programme on a continuous basis and submitted its final assessment report to the government on 30 January 2006. It addressed the three research areas prescribed by law. More specifically, it recommended that disposal be selected as the reference solution. It also considered that the work conducted on that theme met "the best international standards". The CNE also felt that investigations had shown that the Callovo-Oxfordian formation constitutes a "remarkable achievement, both in quality and quantity"; the work also demonstrated that, from such a standpoint, the rock of the Meuse/Haute-Marne Site was very homogeneous and free of water-conducting faults.

At the request of the ASN, the Institute for Radiation Protection and Nuclear Safety (Institut de radioprotection et de sûreté nucléaire – IRSN) also reviewed the Dossier 2005 Argile, and published an assessment report that was in turn submitted to the Standing Waste Group whose final opinion was transmitted to Andra as follows: *"The Standing Group emphasises that the Dossier 2005 Argile contains a thorough and high-quality presentation, constituting a significant advance. [It] issues a favourable opinion concerning Andra's assessment and considers that a radioactive-waste repository within a clay formation for which ongoing investigations are taking place in the Bure Underground Laboratory, is feasible. […] the Standing Group also considers that there is no safety-related obstacle to the search for a repository site within the perimeter of Andra's delineated transposition zone".*

In the opinion it presented to the government on 1 February 2006, the ASN mentioned that "a deep geological waste repository is a final management solution that appears to be unavoidable".

The peer review of the Dossier 2005 Argile carried out by the International Review Team (IRT) set in place by the OECD/NEA concluded especially that the programme met fully the best international practices and was leading in several fields. Andra's reversibility approach is considered as innovating without compromising the safety of the repository [2].

4.2 The Public Debate

At the government's request, a national debate was also held on the long-term management of radioactive waste under the auspices of the National Commission on Public Debate (Commission nationale du débat public) with six months of preparation and 13 meetings between September 2005 and January 2006. Scientific and technical issues, as well as management and governance strategies, were discussed at length. In its report, the National Commission underlined the general view that all radioactive waste be taken into account by the new act, the need to improve governance with regard to radioactive-waste management, the benefits of a stepwise decision-making process and the need for a concrete territorial project for the districts in which a potential waste repository may be implemented.

Lastly, the OPECST report, published in March 2005 by Messrs Birraux and Bataille, analyses investigation results from the standpoint of management strategies and confirms the complementarity of the three research areas examined pursuant to the Law of 30 December 1991: partitioning and transmutation, deep geological disposal and long-term storage.

5 A New Planning Act

The Planning Act of 28 June 2006 [3] extends the scope of the Law of 1991 by prescribing specific deadlines for the different management solutions to come into force. For partitioning and transmutation, for instance, the industrial prospects associated with investigations in the framework of the fourth generation of reactors must be assessed by 2012. With regard to the implementation of a reversible waste repository within a deep geological formation, the prescribed schedule requires all relevant elements for the review of the licence application to implement such facility to be ready by 2015 at the latest. The commissioning date of the repository is set in 2025, a date that is compatible with the production estimates for high-level and intermediate-level long-lived waste generated by the French nuclear power cycle.

The Act also provides two essential elements in fields that were not covered by the Law of 1991. First of all, in response to the wish expressed during the public debate, it proposes a concrete national management plan not only for radioactive waste, but also for radioactive materials, whether recoverable or not, by creating the National Radioactive Waste Management Plan. The Act prescribes strict deadlines for high-level and intermediate-level long-lived waste, as well as for the commissioning in 2013 of a disposal facility for graphite and radium-bearing waste, consisting of low-level short-lived waste. Hence, within a few years from now, all categories of radioactive waste will have found an adequate outlet.

Furthermore, the Act provides a legislative framework for the dismantling of nuclear facilities and particularly for an amount of 68 billion euros, currently considered appropriate, to be constituted by operators and made available. Parliament will participate in the control of those financial provisions and in their specific appropriation in the companies' accounts.

Lastly, the Act reinforces socio-economic incentives in the territories concerned by a potential waste repository. Hence, it strengthens the existing public interest groups promoting local development in the Meuse and Haute-Marne Districts. It also aims at encouraging nuclear industrialists to participate in local industrial projects and confirms the statute of the CLIS.

Beyond its industrial mission to manage radioactive waste, its research mission and its information mission to disseminate knowledge, the major evolutions in Andra's mandate concern:

- the leadership of investigations relating to storage;
- the assessment of accruing costs to the implementation of long-term management solutions for radioactive waste, which will serve as the basis for calculating the producers' provisions;
- the conditioning of waste for which Andra is authorised to provide an opinion;
- the take-over of radioactive waste and polluted orphan sites (public-service mission).

6 The New Challenges of Geological Disposal

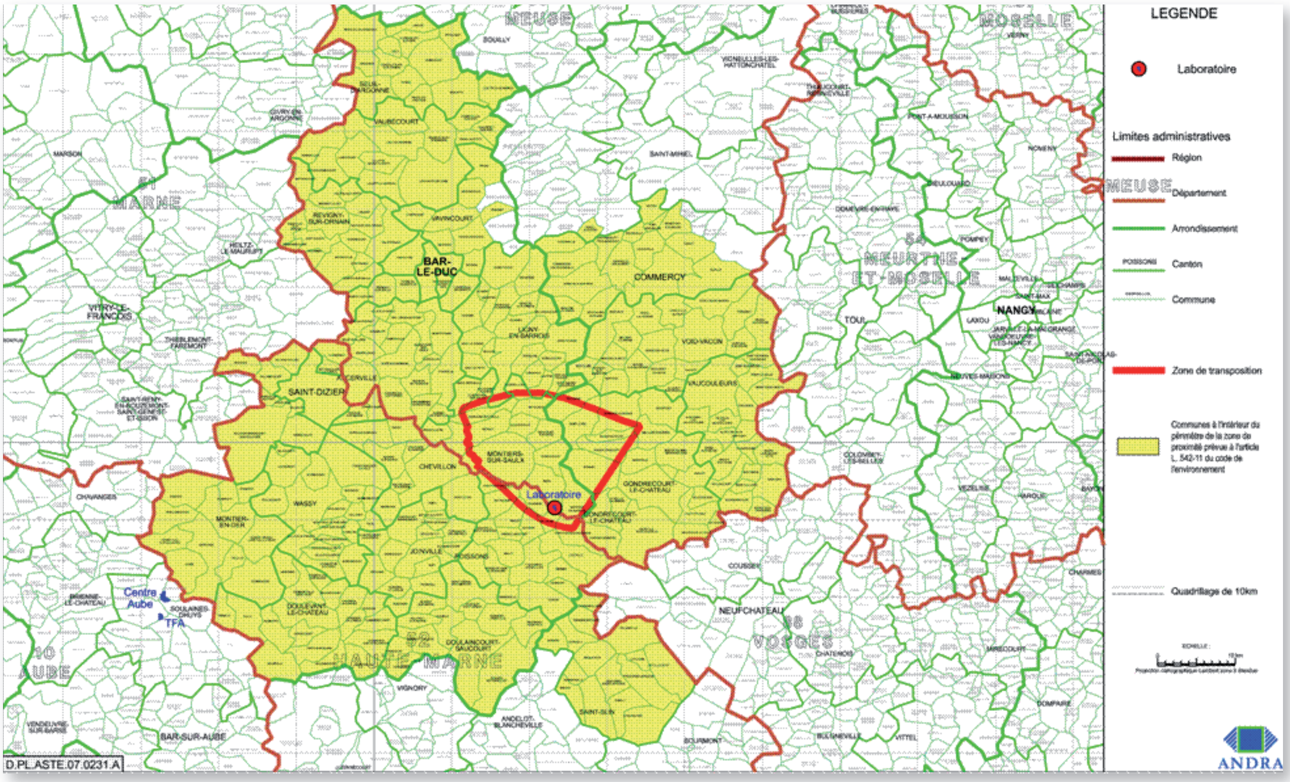
The Planning Act of 28 June 2006 details the framework and the objectives relating to radioactive-waste management for the years ahead. The first edition of the National Radioactive Waste Management Plan was published at the beginning of 2007 [4] and was subject to a first assessment. The major next step for high-level and intermediate-level long-lived waste will consist in preparing in time for 2015 the licence application for the implementation of a deep geological repository.

The licence application involves the identification of a specific site as well as the definition of a satisfactory repository concept and architecture with respect to safety and reversibility requirements. Over the next decade, efforts will be dedicated to gathering all relevant information in order to build a convincing case. Those efforts will require the mobilisation of all stakeholders in order to provide technical, social and political responses.

6.1 In Search of a Site

The identification of a suitable site requires not only that geological characteristics be consistent with the requirements imposed by the repository, but also that local populations accept its implementation. In the framework of the Dossier 2005 Argile was delineated a 250 km^2 transposition zone corresponding to the area within which geological properties are similar to those found on the Meuse/Haute-Marne Site, as confirmed mainly from the Bure Underground Laboratory. The ASN has recommended that a suitable area be drawn within that perimeter for the implementation of a waste repository. It will be Andra's task to continue finer and finer survey activities throughout the transposition zone, notably on the basis of 2-D seismic profiles and later at a more restricted scale on 3-D seismic measurements; new probe boreholes will also be used in order to enhance the geological data of the zone.

As regards the territorial policy, the prospect of implementing a waste repository is only possible within the framework of a joint project with community, government and industry representatives providing, among other measures, for the economic development of the site being considered. Incentives for the regional development are provided for in the Law of 28 June 2006. The eligible proximity zone for economic incentives was also determined by order and encompasses more than 300 villages and towns around the transposition zone (Figure 3).



▲ Fig. 3: Map of the proximity zone with the transposition zone (in red).

Year 2007 was marked by intense regional-exchange and project-development activities with a view not only to promoting local business activities, but also the social values that encourage local citizens to accept the implementation of a waste repository. The nuclear-power sector is mobilising itself in order to stimulate development in the areas located near the Meuse/Haute-Marne Site with various local job-creation prospects being announced. Local and regional installations and infrastructures also experience a new boost due to the impulse of the public-interest group created specifically for that purpose and funded through an additional tax paid by operators of basic nuclear facilities. In 2007, the budget amounts to approximately 20 million euros for each of the two districts concerned.

Among the social values that would be able to support the local implementation of the repository, the high scientific and technological level of a deep geological repository provides an excellent opportunity for developing an international showcase. A broad series of initiatives designed to promote the proximity of the underground laboratory and later of the waste repository may be contemplated by the various stakeholders: industrial tours, information centre, museums of science and technology, training pole or very large scientific equipment. They all have the potential to contribute to the social promotion of the populations by reinforcing the sense of pride in belonging to a recognised community for its state-of-the-art expertise. They may also improve the reputation of the region and the development of its economy.

All those arguments support community dialogue throughout the site-selection process. However, a certain number of guarantees must be provided to the populations in order to build confidence. The most important guarantee will be the safety of the repository and is addressed under various aspects that are taken into account in the repository concept and architecture themselves, and in its operating modalities.

6.2 The Repository Project

The architecture of the repository, which is still at the feasibility stage, is contemplated according to a horizontal development in the core of the Callovo-Oxfordian formation, at approximately 500 m in depth. It is designed to dispose of the various categories of high-level and intermediate-level long-lived waste within structures adapted especially to the thermal characteristics of each of those categories. The distribution of shafts and access drifts, of disposal cells or cavities, their size and the choice of building materials were guided by safety criteria with a view to retarding as long as possible any release of the radionuclides contained within the waste and to retaining them as effectively as possible within the disposal structures and the Callovo-Oxfordian formation.

Disposal structures were designed to ensure reversibility (i.e., allowing for waste removal) in accordance with the government's request in 1998. It was demonstrated that such possibility was feasible for more than 100 years without altering long-term safety conditions. A progressive approach for closing down the structures was developed with a concern to allow society to make decisions step by step and the operator to revert back to a previous state. Hence, in case of anomaly or significant gap in the behaviour of the repository compared to the expected evolution, it would be possible to remove waste packages or to adapt structures accordingly.

The finest knowledge possible of the different physico-chemical phenomena that may affect the evolution of the repository has been undergoing continuous improvement for many years in order to assess the safety at different timescales. That knowledge lies at the basis of the study on suitable means to observe and to monitor the repository; it constitutes the technical foundation for any decision to be made concerning the shutdown of the repository and reversibility.

The repository design and the information-acquisition process follow an iterative procedure and are consistent with the safety of the repository, the demonstration of which remains the essential element for building confidence. Complementary guarantees are provided to local populations and communities by monitoring the repository's evolution and reversibility. However, those technical guarantees are only meaningful if they are also supported by the national community, whose role will have proven determining in any decision made concerning the implementation of the repository. The process described by the Planning Act of 28 June 2006 provides for a political intervention to take place through a public debate and decisions relating to reversibility and the closure of the repository by Parliament.

Holding a public debate prior to the submission of the licence application to authorise the implementation of a deep geological repository provides an additional opportunity for all stakeholders to express their views in the light of the entire body of knowledge made available by Andra. The purpose is to ensure collectively that current and future generations, even over the very long term (hundreds of thousands of years), will benefit from suitable conditions in order to manage the repository both safely and efficiently. The current investigation programme includes a public debate to be held between the end of 2012 and the beginning of 2013.

By adopting a stepwise approach, Parliament wished to be consulted once again before the government makes any licensing decision. Two control points have been established. The first prescribes that the reversibility conditions of the repository be determined before any authorisation to implement a repository is granted. According to the schedule set by the Planning Act of 28 June 2006, a corresponding law should be adopted after the submission of the licence application, in other words, in 2016. Later, once operations will have come to an end after several decades, only a new law may formalise the final closure of the facility. By observing and monitoring the behaviour of the disposal structures and of waste packages throughout those decades will provide technical information on which will be based that decision.

7 Conclusions

Besides high-level and intermediate-level long-lived residues, which form Andra's most visible challenge with regard to radioactive-waste management, the Agency keeps receiving very-low-level waste at its Morvilliers facility as well as low-level and intermediate-level waste at the Centre de l'Aube Disposal Facility. Operating procedures are consistent with a continuous-improvement policy in accordance with Andra's quality processes. A large number of studies are carried out in parallel to the operation of the disposal facilities in order to assess acceptance conditions regarding non-standard waste packages that were not taken into account during the design stage. Over recent years, that was notably the case for reactor-vessel covers from EDF nuclear power plants. Safety models are also constantly updated on the basis of the collected data concerning not only incoming and disposed packages, but also the monitoring of facilities and of their environments.

The Centre de la Manche Disposal Facility, has been shut down for 10 years and is closely monitored by Andra. Monitoring activities do not only cover the environment, but also the covering structures whose purpose is to limit as much as possible any seepage of surface waters through the repository structure. Periodical re-assessments are also carried out on the site.

The next disposal facility to be commissioned will be dedicated to radium-bearing and graphite waste. The search for a suitable site, together with the design and construction of the facility, must be completed in time for the first waste packages to be processed in 2013.

The quality in the implementation and operation of surface structures represents an essential showcase for Andra in order to reinforce confidence in its more ambitious projects relating to deep geological disposal. Information and communication, being one of the Agency's explicit missions besides its industrial and research missions, has always been an integral part of all Andra projects. Their varying scientific, technical, social and political scopes require that they be introduced, explained, discussed both at the local scale around every site and at the national scale where decisions are taken concerning every citizen who consumes electricity or services involving radioactive substances. The efforts dedicated to communication, training and the dissemination of the scientific and technological culture pertaining to radioactive-waste management, and to the promotion of Andra's know-how abroad are also prescribed in the Planning Act of 28 June 2006. Those constitute essential conditions for the success of the new projects thanks to the sharing and enactment of Andra's approaches by all stakeholders who, at one point or another, are likely to express their views. ■

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1.05 Preparing for Licensing: Progress in Underground Rock Characterisation of Olkiluoto Bedrock in Finland

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Abstract
In May 2001 the Finnish Parliament approved of the Decision-in-Principle on disposal of spent fuel from innish nuclear power plants in the Olkiluoto bedrock near the Olkiluoto nuclear power plant. The site investigations are now focused at Olkiluoto and the development of the disposal concept is continued on the basis of the KBS-3 type concept. The Finnish safety regulations require that the suitability of the site be confirmed through underground investigations before the submission of the application for the construction license of the repository. For that purpose the construction of an underground investigations facility, ONKALO, was started in 2004. By the end of June 2007 the excavation of the access tunnel had reached the chainage of 2100 metres. Some 180 metres of the first shaft have also been excavated. The underground investigations are in full progress. Besides regular tunnel mapping and various monitoring activities, various hydrological and geochemical investigations have been started. Modelling predictions are made at regular intervals for the tunnel conditions ahead. In parallel with the ONKALO programme, work is underway on disposal technology and safety case of geologic disposal at Olkiluoto. All the activities aim at submission of the application for construction license in 2012.

1 Introduction

Preparations for nuclear waste management were started in Finland in the 1970's when the power plants were under construction. In 1983, the Government set a target schedule for spent fuel management, in which the construction of the final disposal facility was planned take place in the 2010s and the start of final disposal in 2020. With the Decision-in-Principle of 2001 the Finnish Parliament reconfirmed the time schedule, approved the siting of the repository in Eurajoki municipality, near the Olkiluoto nuclear power plants, and agreed that the disposal can be based on a KBS-3 type concept. According to the present target schedule, Posiva is planning to submit the application for the construction licence in 2012. At the moment this sets the framework for all investigations, research and technology development related to spent fuel disposal in Finland.

One of the prerequisites for the licence application is the confirmation of the site suitability through underground characterisation of the intended host rock of the repository. For this purpose an underground rock characterisation facility, ONKALO, is now under construction at Olkiluoto. More than two kilometres of access tunnel have been excavated by June 2007, and underground investigations have been started. The target depth of the main characterisation level is 420 metres, but excavations are planned to be continued down to a depth of 520 metres.

Since the plan is to use the ONKALO as an access way to the actual repository as well, the construction of the ONKALO follows the rules and requirements for nuclear facilities in Finland. Special emphasis is given to measures to limit the disturbance caused by the construction activities on the host rock to the minimum possible.

In parallel with the design and construction of ONKALO and the related investigations, progress is being made in the development of the technology and safety case for the license application. However, in this paper the focus will mainly be on the underground rock characterisation work.

2 Programme Framework

The repository siting programme defined in the early 1980's was based on stepwise proceeding from general surveys to detailed characterisation, first through surface investigations and then to final suitability check by underground investigations. Several intermediate check-points were defined before the actual licensing stage. The Decision-in-Principle (DiP) of 2001 to locate the repository at Olkiluoto was based on the results from ten years of surface investigations at several candidate sites, during which about ten deep boreholes were drilled at each of them and extensive modelling was carried out to develop the preliminary site descriptive models for them. These models then formed the basis of the safety assessment TILA-99 (Vieno & Nordman 1999) and the Environmental Impact Assessment report (Posiva 1999) that were used to support the application for the DiP . The site descriptive model for Olkiluoto was reported by Anttila et al. (1999) and the whole siting process before the DiP has been summarised in McEwen & Äikäs (2000).

On the basis of the site assessments made by Posiva all the candidate sites would be suitable for spent fuel disposal, and the Finnish Radiation and Nuclear Safety Authority, STUK, did not object that conclusion in the preliminary safety assessment they prepared for the Government (Ruokola 2000). However, according to the general safety requirements that had been ruled by the Government the site suitability should be finally confirmed through underground investigations at the actual site and in the intended host rock of the repository.

Accordingly, after DiP Posiva started preparing for the underground rock characterisation at Olkiluoto. In the long-term RTD programme of 2000 (Posiva 2000) Posiva stated the goals of the underground work as to

- verify the current conclusions on suitability of the Olkiluoto site
- define and identify suitable rock volumes for the repository
- characterise those rock volumes in detail for the design, safety assessment and construction of the repository.

Characteristics of importance included

- nature of fracturing
- structural features to be avoided
- magnitude and direction of the rock stress field
- strength of rock mass
- thermal properties of rock
- rock mechanical response to excavation
- detailed hydrogeological and hydrogeochemical properties.

To enable the underground characterisation a facility would be needed that

- would provide the access to the bedrock at depth of 400—500 metres below sea level
- would make it possible to apply various underground investigation techniques safely and reliably
- would not badly distort or deteriorate the site conditions of importance for the long-term safety of disposal.

An additional important design consideration was that the access ways to the underground investigations facility should later serve as access ways to the actual repository, i.e., the facility would become a part of the repository. Because of this, the facility should be built according to the norms applicable for nuclear facilities.

The facility was given the name “ONKALO”, which can be considered as a stylised acronym for the Finnish words meaning Olkiluoto Rock Investigations Facility for Final Disposal, but the word also means a cave in Finnish.

3 ONKALO Layout

After a systematic comparison of various conceptual alternatives, the decision was made in 2002 that the access to the repository depth would be provided both by an access tunnel and a vertical shaft. The main aspects in favour of the combined tunnel-shaft concept were the increased flexibility as regards the planned future use of the facility as a part of the repository, the logistics benefits as well as the greater opportunities for characterisation during construction.

The conceptual design of ONKALO at the main drawings stage is presented in Figure 1. The main characterisation level is at a depth of 420 metres below sea level; the lower characterisation level is 100 metres below the main level. The inclination of the tunnel is 1:10, which means that the length of the access tunnel will be approximately 5.5 km. A total of 330 000 m³ of rock will be excavated.

The location of the ONKALO entrance is in the central part of the Olkiluoto island, some two kilometres away from the Olkiluoto nuclear power plant, near the southern border of the existing site investigation area (Figure 2). The location was decided on the basis of a comparison between a number of alternatives. In this comparison one of the main criteria was the expected disturbance to the host rock of the repository; in particular, the inflow of groundwater to the tunnel was to be kept to the minimum.

The main characteristics of the original design are summarised in Posiva (2003a). The first design had the tunnel and only one shaft; later decisions have been made to build three shafts already during the ONKALO stage. These will ease the ventilation and logistics, and also provide economies in the excavation. All the three shafts will be fairly close to each other.

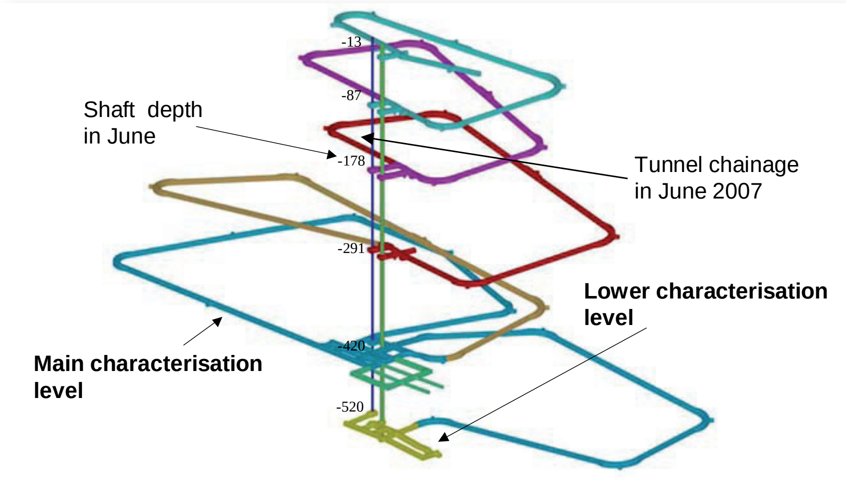

Fig. 1: ONKALO Underground Rock Characterisation Facility.

Site preparations for the facility were started in 2003 and the actual excavation work began in September 2004. The tunnelling work is carried out using drill & blast techniques. Special attention is paid to minimising of groundwater leakages into the tunnel by means of careful grouting. By midsummer 2007 the tunnel has passed the chainage of 2000 metres at the depth of about –190 metres. The first shaft has now been built down to the level of –180 metres.

4 Investigations

One of the aspects in favour of an access tunnel – in addition to shafts – was that the tunnel would make it possible to start the underground characterisation work earlier than in the shafts. Although the main interest was in the actual host rock at the depth of 400 – 500 metres, the upper parts of the tunnel could be used to develop the investigations techniques and train the cooperation between design, construction and investigations staff. After all, one central benefit from the ONKALO period was to be gained in learning. In practice, the underground investigations were started at the same time with the excavations.

A programme for underground characterisation and research (UCRP) to be carried out in ONKALO was set up in 2003 (Posiva 2003b). The investigations programme consists of mapping of the tunnel roofs and walls, drilling of characterisation holes with subsequent geologic, geophysical and geohydrological studies and tests, hydrogeochemical sampling and measurements, determination of fracture and flow data plus various rock-mechanical tests and measurements. The investigations programme was built in a way that would minimise delays in construction schedules but still would not sacrifice the possibility of obtaining any important information.

◄ Fig. 2: The location of the ONKALO at Olkiluoto: the mouth of the access tunnel is seen in the foreground.

In addition to the UCRP a separate monitoring programme was launched. Its main purpose is to follow the possible changes in surface and underground conditions due to excavations activities.

The mapping is done in two stages: first a quick round mapping soon after blasting; after the tunnel face has advanced to a safe distance a systematic geological mapping is carried out for the tunnel walls. The first mapping is mainly done for the purpose of construction planning and to determine the possible needs for temporary rock reinforcement. The main input for the actual site characterisation is gathered from the more detailed second mapping campaign.

Probe holes of 20—25 metres are primarily used for hydraulic measurements to determine the grouting needs. In addition, cored pilot holes of 100 to 200 metres are drilled for most of the tunnel length and serve for advance planning of the excavation work. Probe holes and pilot holes are kept inside the tunnel perimeter to avoid unnecessary openings in the tunnel vicinity.

Several ground water sampling stations are now in place. At this stage they mainly serve for the monitoring purposes, but later the ONKALO should enable obtaining representative groundwater samples also from the less conductive rock volumes.

The groundwater inflow to the tunnel is monitored at measurement weirs. The main leakage points are recorded separately.

The first investigations niche is now available for more time-consuming tests and experiments. Plans are made for studying both the groundwater flow conditions and the hydrogeochemistry in low transmissive fractures.

The ONKALO should also improve the current picture of the rock mechanical conditions of the Olkiluoto site. The stress measurements made earlier from surface drillholes have suffered from technical problems affecting the reliability of some of the measurements. The underground measurements from tunnel niches are expected to provide a more reliable data set for the understanding of the local stress conditions. One stress measurement campaign has so far been made, but it seems that technical improvements in measurement methods would still be highly desirable for obtaining reliable data. Inverse modelling of the rock response measurements may help getting additional information on stress conditions.

Besides the rock characterisation activities in the ONKALO, surface investigations are continued. By the end of summer 2007 there will be 48 deep core-drilled holes at Olkiluoto. Some of the holes drilled in recent years have served for the design of the ONKALO; at present the main purpose is to extend the area "covered" by the drillholes to the eastern parts of the Olkiluoto island. This would enhance flexibility in the layout design of the repository. In addition to drillholes, extensive use has been made of investigation trenches, in which the rock surface is uncovered from soil for direct visual mapping.

5 Modelling and Interpretation

The data from the surface and underground investigations are used for developing a consistent icture of the bedrock at Olkiluoto. A special group, the Olkiluoto Modelling Task Force, is set up to coordinate and integrate the work done by experts in various geoscientific disciplines for the development of the site descriptive model of Olkiluoto. The latest model version was published in the early 2007 (Andersson et al. 2007) and the next one is scheduled for the late 2008. The next version will largely form the basis of the layout design for the licensing application.

The description of the bedrock in the ONKALO area is updated at shorter intervals. The purpose is to keep the picture of the local geological structures and properties continuously up to date in a way that makes it possible to consider needs for revisions of the tunnel layout before it is too late from the design and construction point of view.

An important part of the investigation programme is, indeed, the prediction-outcome process in which models are used to predict rock conditions further along the tunnel line. These predictions assist in the further design of the facility and play an important part in the learning process built into the whole ONKALO programme.

Predictions are made on three levels: type A predictions are based on the latest accepted site model, type B predictions may use additional information that has become available before the model publication, e.g., the pilot hole data; type C "predictions" are made after the excavation and try to estimate whether the modelling method was in fact capable of producing the real outcome by adjustment of model parameters.

So far the rock quality and structure has turned out to be very much alike it was predicted and no significant surprises have been encountered. The site of the ONKALO was selected on the basis of expected good rock conditions, and so far the outcome from the excavated part corresponds to the expectations. Some modifications have been made in the structural bedrock model of the ONKALO site since the early design period, and they imply small layout modifications in the lower parts of the ONKALO, but in general the ONKALO outcome information seems to well conform to the picture predicted.

6 Special Issues and Challenges

Since the ONKALO is built at the repository site the construction and operation of it will have an effect on the conditions in the repository host rock. It will cause some rock mechanical disturbance in the rock, but the most important disturbance is likely to be caused by the inflow of groundwater into the tunnel and shafts. The leakage will directly affect the hydrogeological conditions, but it may also indirectly change the geochemical conditions of the repository environment. Another important disturbance may be caused by the introduction of various "foreign" materials, especially cement based grouting masses, into the underground rock volumes.

Temporary changes in the host rock conditions are inevitable. After closure of the repository the original conditions will soon resume and there will be no serious impact on repository safety. In some cases the recovery of the original conditions may take more time, and in some conceivable cases the impact may be non-trivial even if limited in time. Therefore, any changes need to be assessed in terms of their meaning for long-term safety and, inversely, the smaller the changes can be kept, the less there will be need for extra analysis.

On the basis of the assessment made by Vieno et al (2003) the main concerns about the disturbance caused are related to the performance of the buffer and backfill. Pumping effects of the leakage waters together with the lowering of the groundwater table in the ONKALO area may raise the salinity of the groundwater to levels that may be harmful for bentonite materials. Grouting can reduce the groundwater leakage, but the grouting materials may also be deleterious for the buffer performance.

To limit the potential negative disturbances special procedures and instructions have been defined for the construction and investigation activities that are deemed critical from the long-term safety point of view. The critical activities are concerned with

- management of groundwater inflow
- management of the EDZ
- use of foreign materials
- drillings in the ONKALO and above it.

For example, two warning levels are defined for the groundwater inflow: passing the first level requires that increased attention be given to minimisation of the inflow in the following construction periods; passing the second level requires that either the higher level are justified by new analyses or more efficient means to reduce the inflow are found.

Normal cement-based grouting materials have been used to keep the inflow below the warning levels. The present total inflow is about 20 litres per minute into the 2 km of excavated tunnel and shafts, which should not pose problems as regard the first warning level of 140 litres per minute for the whole facility, which is based on model simulations. However, the use of cement-based grouting materials is considered a risk in the actual host rock volumes because of the possibility of a high pH plume that might affect the behaviour of compacted bentonite components.

A special programme is now underway to develop a procedure for the inflow management that would efficiently limit the inflow when needed but would also be chemically acceptable. After the well conductive bedrock sections near the surface the main inflow sources to the ONKALO are likely to be the two subhorizontal fault zones “R19” and “R20” that cannot be avoided in the ONKALO layout design (Figure 3). The first of these, R19, has already been crossed without major problems. The goal is now to have the efficient and chemically acceptable grouting procedure in place when crossing the second zone, R20. According to the present construction schedules the level of R20 will be reached in mid-2008.

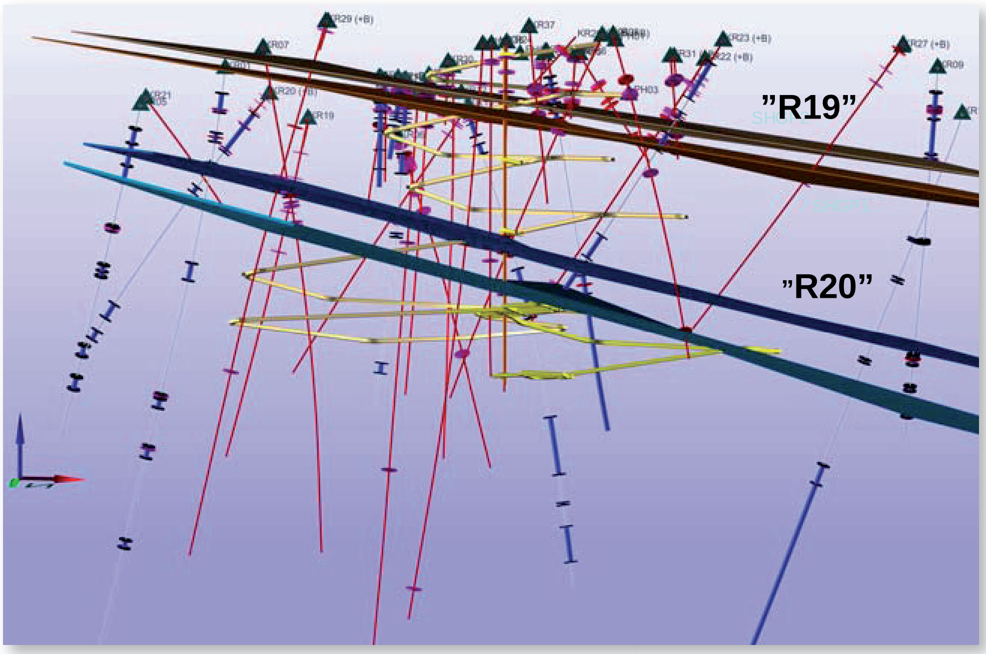

Fig. 3: Main hydrogeologically important fracture zones in the ONKALO area. The picture also shows the current deep drillholes crossing these structures.

The control of the EDZ is another issue that is subject to special studies in the ONKALO. Tentative limits have been defined for the acceptable depth of the excavation damage zone, but their relevance is subject to re-evaluation in light of the safety case considerations presented lately, for instance, in Sweden. Independent of the specific figures for acceptable EDZ, the methods for verification of compliance with these limits requires further work.

A new review of the expected disturbances caused by the ONKALO has been recently compiled by Alexander & Neall (2007). The main conclusions remain similar as in the earlier assessment by Vieno et al. (2003); a number of proposals for further research into the subject has been made.

As the ONKALO is built on the actual repository site and will access the actual host rock, it is natural to plan to use it also an access way to the repository. This means that the ONKALO must comply with the rules and regulations applicable for the repository. Therefore, although the ONKALO is not a nuclear facility, yet, it will be built according to the same rules and procedures as the repository would. The regulatory agency, STUK, has launched a special project to control and supervise the construction process. In this context STUK and Posiva have agreed on a number of procedures on inspections, documentation, quality management and mutual communication. As regards research and investigations, regular meetings are held and a joint issue list is maintained on matters calling for resolution.

7 Concluding Remarks

The construction of the underground rock characterisation facility is now well underway. After some difficulties in the early phase of the excavations, a steady progress is now being made and, barring surprises, the depth of the main characterisation level at –420 metres will be reached by the end of 2009. After that the access is open to the actual host rock. However, even before that, the rock volume intended for the first deposition tunnels will be explored using characterisation holes drilled from niches built in the upper ONKALO sections.

The main goal for the activities at the –420 level is to show that suitable rock volumes are available for repository tunnels and deposition holes. This requires that applicable criteria for suitability have been established by that time. The suitability judgements will be based on the host rock classification (HRC) scheme that is under development. A tentative classification system has already been proposed as a result of a project carried out in 2002—2005 (Hagros et al 2005), but the system still needs further development and testing in the Olkiluoto conditions. Once agreed, the applicability of the HRC system will be demonstrated in the ONKALO as a part of the licensing process.

It is foreseen that some site-specific technology demonstration projects will also be carried out in the ONKALO, but the need for such tests will be assessed on the basis of the results from the joint cooperative projects carried out in the Äspö Hard Rock Laboratory in Sweden. In many respects the experience from Äspö is applicable for Olkiluoto as well; however, some tests may be sensitive to the groundwater salinity level, which is generally much higher at Olkiluoto than at Äspö.

The success of the licensing application will depend on the strength of the safety case that will support the application. According to Posiva's safety concept, the long-term safety of disposal will primarily be based on long-term containment provided by the engineered barrier system and the spent fuel canister, in particular. In this respect, the Olkiluoto site investigations should give sufficient evidence that benign and sufficiently stable conditions prevail in the Olkiluoto bedrock to guarantee the longevity of the containment. The activities in the ONKALO are expected to provide important contributions to that evidence. Posiva's overall RTD programme in the ongoing threeyear period 2007—2009 was recently published in Posiva (2006).

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1.06 Progress in Implementation of a Deep Geological System for Spent Nuclear Fuel in Sweden

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In Sweden the nuclear power industry is responsible for the management and disposal of all radioactive waste from its plants. The owners of the nuclear power plants have therefore jointly formed SKB which has been given the task of organizing the waste management. Over the past three decades a system has been built up for disposing of different types of radioactive waste in a safe manner.

The facilities required for final disposal of the spent nuclear fuel have however not yet been built. Since the 1970‘s SKB has been working with the development of the Swedish method for disposing of spent nuclear fuel, called the KBS-3 method. It entails encapsulating the spent nuclear fuel in copper canisters, which are embedded in bentonite clay at a depth of about 500 metres in the Swedish crystalline bedrock. In addition to the final repository we also have to develop a canister factory and build an encapsulation plant for encapsulation of the spent fuel. SKB has built a number of laboratories to carry out research and development projects on a full scale and in a realistic setting.

The development of the KBS-3-method has been carried out in parallel with the work to find a suitable site for the final repository. Site investigations have now been going on for five years at two sites. An environmental impact assessment is being made by SKB. Stakeholder involvement by concerned municipalities and the interested public is an essential part of the licensing and implementation process.

We are now reaching the end of this stage. The licensing of the two new nuclear facilities is planned to take place during the period 2006-2010. The permit application for the final repository on one of the two investigated sites will be submitted within a few years. Meanwhile SKB already in November 2006 applied for a permit to build the encapsulation plant adjacent to the present-clay Clab interim storage facility in Oskarshamn. Both applications will be scrutinized by the regulatory authorities and reviewing bodies. Provided that permits then are issued the building of the facilities could start around 2012, and the first canister may be disposed in 2020. ■

1.07 Prospects of Research and Technical Development in the Swiss HLW Programme

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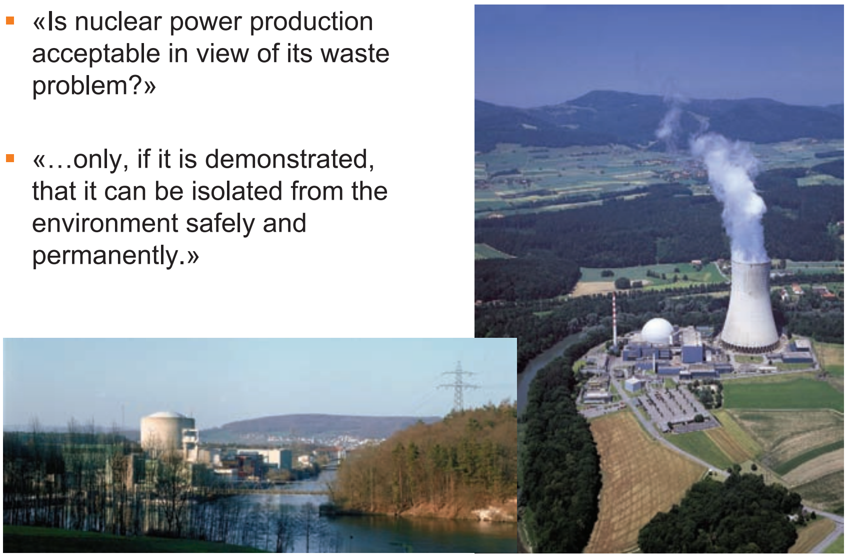
Wettingen, Switzerland

Political and legal requirements

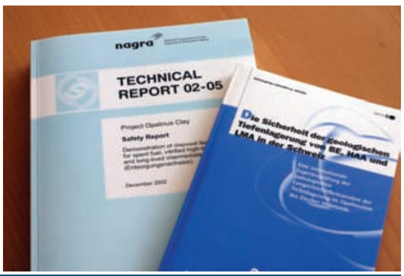
- «Is nuclear power production acceptable in view of its waste problem?»
- «…only, if it is demonstrated, that it can be isolated from the environment safely and permanently.»



2 Issler/05112007 Braunschweig Nov 2007 nagra

Demonstration of Feasibility and Safety of Disposal

- A **legal requirement** for licensing new nuclear power plants in Switzerland
- Such a feasibility **project** must include :
  - **the technical feasibility** of construction and operation of a repository
  - availability of adequate **host rocks and regions**
  - the demonstration of **long term safety**



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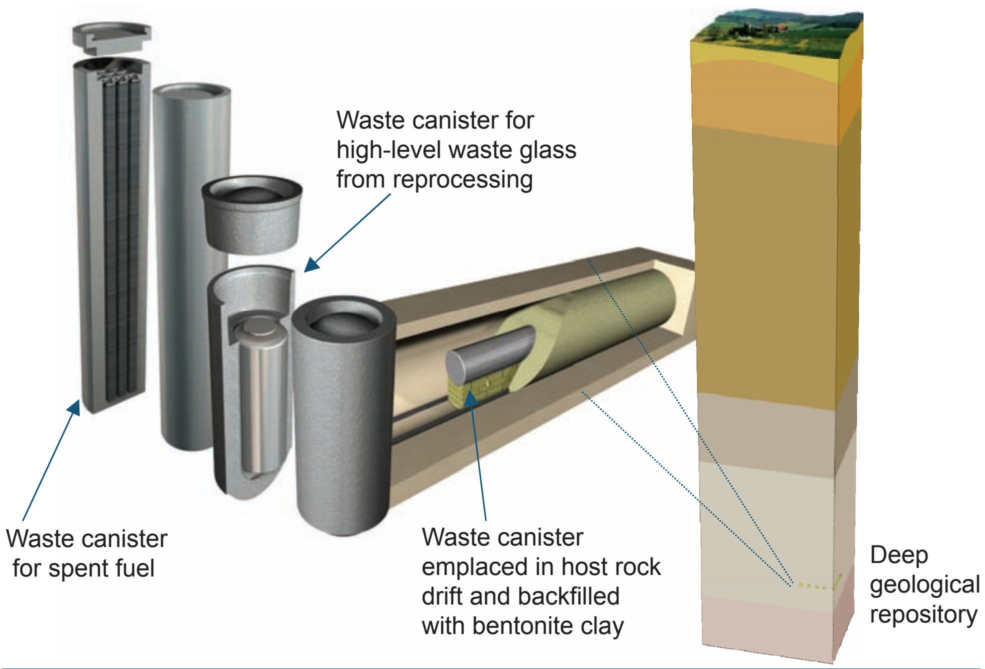
Building the scientific basis: a 30 year foundation

We have:

- **Developed geological disposal concepts** for combined disposal of Spent Fuel/HLW/L/ILW based on multi barrier systems
- **Built up comprehensive infrastructure** (Universities, 2 URLs), tools (methods, models, data) & experienced team
- **In-depth understanding of geological possibilities** in crystalline and sediments for implementation
- **Developed projects and safety cases,** each followed by regulatory review
  - Project Gewähr 1985
  - Project Wellenberg
  - Kristallin-I
  - Project Opalinus Clay 2002 («Entsorgungsnachweis»)

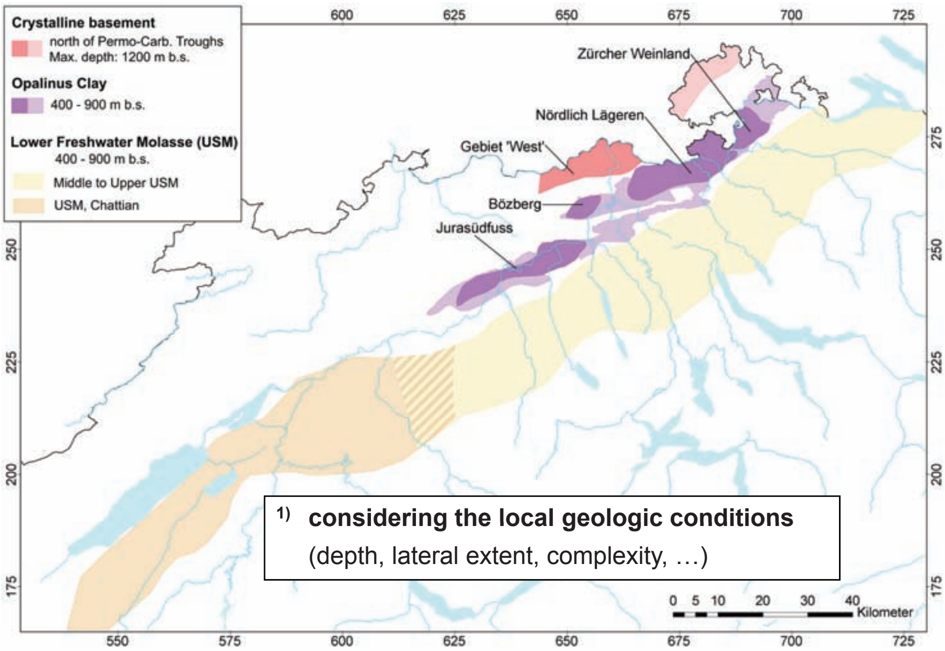
4 Issler/05112007 Braunschweig Nov 2007 nagra

Swiss Multi-barrier System for SF and HLW



5 Issler/05112007 Braunschweig Nov 2007 nagra

HLW repository – potential regions[1)]



1) **considering the local geologic conditions** (depth, lateral extent, complexity, …)

6 Issler/05112007 Braunschweig Nov 2007 nagra

Feasibility project: Project Opalinus Clay

- Nagra project sent to Government — December 2002
- International Expert Review by OECD/NEA — April 2004
- Review by Nuclear Safety Inspectorate and Federal Commission on Nuclear Safety — April 2005
- Nagra report on host rocks alternatives — September 2005

- Government report for public review/consultation — September to December 2005
- Generic socio-economic studies (Government/Region) — June 2006

- Government approval of project — June 28, 2006

7 Issler/05112007 Braunschweig Nov 2007 nagra

The next stage is...

Taking a decision to build repositories

- **Based on**
  - the possibilities that Switzerland offers (available understanding of geology)
  - the understanding available on repository safety
  - the needs of society (land use, environmental impact, socio-economic issues, …)

  …using a transparent societal process, deciding on where to build the repository
  (integration of information & decision-making process based on *Sectoral Plan 'Geological Disposal'*)
- **In parallel: maintain & improve scientific/technological basis**
  - broaden scientific support & enhance confidence
  - maintain high level of scientific competence

8 Issler/05112007 Braunschweig Nov 2007 nagra.

The development of repository projects

- Repository projects take a long time
  (> 100 years from start of project until closure of repository)
  → RD&D has to focus on the next step
  in a long, staged Implementation process
- Development of repositories requires …
  - **regional geological investigations:** seismic investigations, boreholes, … (→ input to site selection)
  - **research on host rock properties** (barrier properties, geotechnical aspects): boreholes, underground research laboratories, …
  - **design work** (layout of engineered barriers adjusted to waste properties and local geological situation)
  - **safety analyses** to evaluate safety and assess adequacy of proposed project (→ importance of feedback!)
- Underground research facilities play an important role in this process

9 Issler/05112007 Braunschweig Nov 2007 nagra.

The importance of international cooperation

- **Grimsel** and **Mont Terri** rock laboratories (granite; clay stone)
- URL studies in Switzerland – with important contributions from many partners
  - Development of methodologies
  - understanding of mechanisms and processes as well as their uncertainties for host rock and engineered barriers EBS
  - Concept testing and demonstration
  - Platform for interaction with the scientific and engineering community <u>and</u> the public

10 Issler/05112007 Braunschweig Nov 2007 nagra

URLs – an increasingly important role

- **Generic URLs** (1st generation URLs) – facility for RD+D at a site that will not be used for waste disposal
  - allows all types experiments (including destructive experiments)
  - provides flexible approach (modifications possible whenever desirable)
- **Site-specific URLs** (2nd generation URLs) – facility for specific investigations at a potential site & may become part of future repository
  - must not unduly affect potential future repository (e.g. negative impact on host rock performance)
  - is part of development of specific repository project (more formal requirements)
- **Performance confirmation facility**[1] («3rd generation URLs»): facility for experiments and monitoring to confirming key phenomena
  - is of importance during and after waste emplacement (→ input to decision-making for final closure of repository)

[1] «pilot facility» in the Swiss context

11 Issler/05112007 Braunschweig Nov 2007 nagra

Nagra - RD&D emphasis for the next stages

- **Site selection stages for SF/HLW and L/ILW repositories (to ~2014)**
  - Stage 1 – Selection of potential **siting regions**
  - Stage 2 – Selection of at least **two proposed sites**
  - Stage 3 – General licence application and confirmation in context of a **general licence** – **selection of a project for realisation**
- **RD&D implications of schedule and steps**
  - Stages 1 and 2
    - **Geological synthesis** needed for several sites that would fulfil safety and geological requirements
    - **Facilities design** (surface and underground) as required for support of dialogue on land planning aspects
    - **Multiple safety analyses studies** required, with updates of data and models in various areas
  - Stage 3 – **Major geosynthesis and safety assessment studies**, including improvements in models, databases and technology development areas
  - URL at site in ~2020-25
  - L/ILW repository in service ~2030
  - SF/HLW repository in service ~ 2040

12 Issler/05112007 Braunschweig Nov 2007 nagra

Key areas of work until general licence application (1)

- Geoscience
  - Host rock properties, state conditions, long-term evolution
- Safety and systems analysis
  - Safety concept and strategy, code development, synthesis of system understanding, operational safety, performance assessment, monitoring concept, requirements management
- Waste characterisation and acceptance
  - Inventory, logistics, acceptance criteria, advice
- Facilities design
  - Surface and underground facilities, tunneling, constructability
- Technology development
  - Canister development and corrosion studies
  - Sealing materials engineering development

13 Issler/05112007 Braunschweig Nov 2007 nagra

Key areas of work until general licence application (2)

- Geochemistry of radionuclide retention
  - bentonite, host rock and cementitious materials
- Waste-form processes
  - Spent fuel, HLW dissolution
  - C-14 behaviour (L/ILW)
- Near-field and host rock processes and evolution
  - EDZ formation and self-sealing
  - Bentonite thermal properties; thermal and chemical alteration
  - Thermo-hydro mechanical coupled processes – parameters, model development and application (including URL studies)
  - Gas transport – laboratory and URL studies, design aspects (also L/ILW)
  - Cement/rock interaction (also (L/ILW)

14 Issler/05112007 Braunschweig Nov 2007 nagra.

Summary & conclusions

- It has been demonstrated and approved that **geological disposal** in Switzerland is **feasible and safe**.
- The **future RD&D** will focus on the next steps of site evaluation and for a **general licence application** around 2014
- **International cooperation** has played an important role in contributing to existing know-how and the advanced stage of technical development
- **URL investigations** and **field studies** play a critical role in RD&D
- **Collaboration** will play an increasingly important role as various countries move into the repository implementation phase

15 Issler/05112007 Braunschweig Nov 2007 nagra.

1.08 Status of Radioactive Waste Disposal in Germany

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Abstract

In May 2007, the German Federal Office for Radiation Protection (BfS) was ordered to construct Konrad mine as a repository for radioactive waste with negligible heat generation. The Konrad repository was licensed in 2002, with a capacity of 303,000 cbm. A non-appealable plan-approval decision for Konrad repository was passed on April 3, 2007. The work has begun and is planned to be completed within a period of six years.

The abandoned salt mine at Morsleben was used respectively from 1971 to 1991 and from 1994 to 1998 as a repository. In total, 36,000 cbm of low- to intermediate level wastes were disposed of. For safety reasons, BfS abandoned further acceptance and disposal of radioactive waste in 2001. Backfilling of selected rooms of the mine started in October 2003, in order to maintain geo-mechanical stability and integrity. An appeal has been filed for the closure. The next step shall be an active public involvement by 2008.

Investigations into the conceptual and safety-related issues have accumulated a host of information on the state-of-the-art of science and technology, with respect to planning and erection of repositories for high-level waste. These investigations indicate that none of the host rocks provide, a priori, the highest level of repository safety. At present, the safety requirements for the disposal of high-level waste are being compiled. The implementation of a site selection procedure for a high-level waste repository is recommended.

1 Introduction

The objective of this paper is to give an introduction to the current status of radioactive waste disposal in Germany.

According to German Federal Atomic Act, the federal office for radiation protection (BfS) is the operator of repositories, under the supervision of the Federal Minister of Environment. It is the law, that repositories have to be licensed by the state ministry of the respective state, where the site is located. The plan approval procedure is defined by the Atomic Act, and includes public involvement.

The disposal of high level waste is mainly an activity problem, not a volume problem. More than 99.9 percent of the activity to be disposed of is concentrated in the high level waste, which has only a percentage of 10 % of the waste volume. On the other hand, 90 % of the volume comes from low to intermediate level wastes with lower than 0.1 % of the radioactive inventory (Figure 1).

Inventory of conditioned wastes
dated: 31.12.2005

low and intermediate level waste	high level waste
ca. 86,000 m^3	ca. 1,800 m^3

Projection for 2040

low and intermediate level waste	high level waste
ca. 200,000 – 280,000 m^3	ca. 24,000 m^3

< 0.1 % **percentage of activity** **> 99.9 %**

▲ Fig. 1: Inventory and projection of radioactive waste in Germany

2 KONRAD Repository

In May 2007, the German Federal Office for Radiation Protection (BfS) was ordered to construct Konrad mine as a repository for radioactive waste with negligible heat generation.

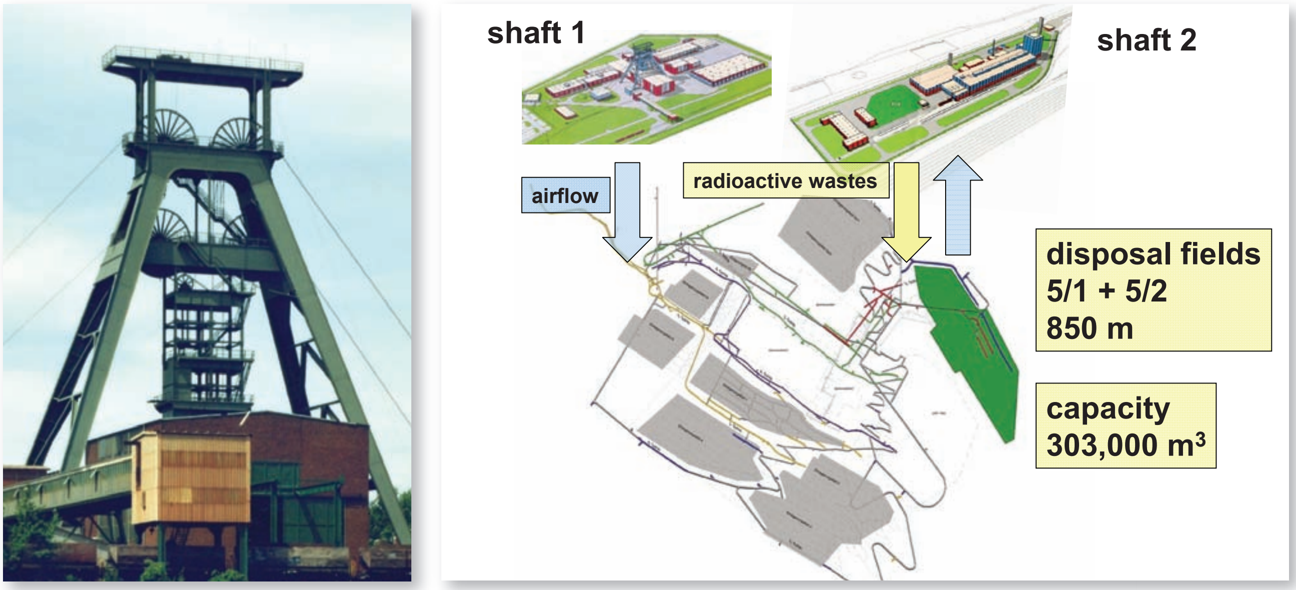
The abandoned ore mine produced about 7 million tons of iron ore from 1965 to 1976. After ore production became uneconomical, mine workers brought this mine into discussion for radioactive waste disposal, due to the extreme dry conditions.

The planning application was filed in 1982. Twenty years later, plan approval for the disposal of 303,000 cbm of radioactive waste with negligible heat production (or low and intermediate level waste) was granted by the state environmental minister of Lower Saxony. All suits and complaints against this decision were dismissed by Higher and Federal Administrative Courts.

The geological situation is unique (see also [1]). Host rock for the disposal of radioactive waste is a 15 to 18 m thick ore body of Upper Jurassic age. It guarantees high stability for the mine, which is an important difference to repositories in abandoned salt mines. The Geological barrier is formed by more than 300 m thick clay rocks of mainly Lower Cretaceous age. These rocks guarantee extreme dry conditions and the suitability for radioactive waste disposal.

The disposal fields 5/1 and 5/2 are at the 850 m level and are expected to contain the whole licensed capacity of 303,000 cbm.

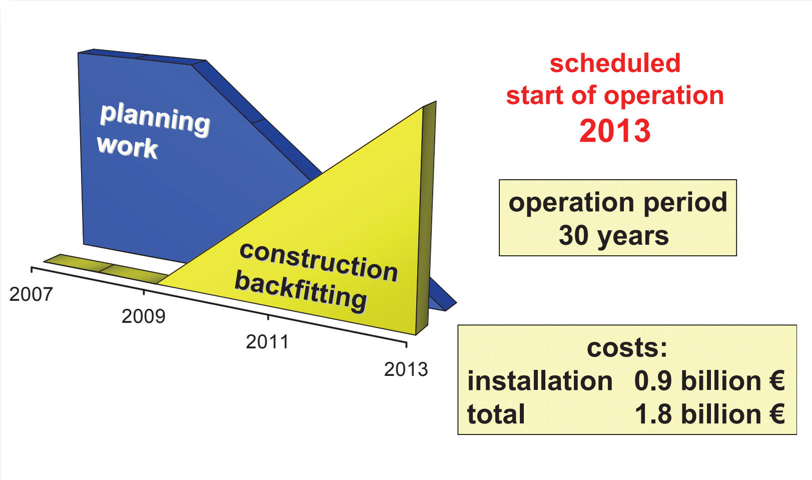
The repository has two shafts (Figure 2). Shaft 1 is for material and personal transports only, as well as downcast airflow. From shaft 2 emplacement of radioactive wastes to the disposal fields will take place. Therefore, the whole facilities for delivery, downloading, buffering and exhaust air cleaning will be located here (see also [1]).



▲ Fig. 2: KONRAD – disposal fields

All plans for the repository are from the beginning of "the 1980s" and, therefore, have to be reworked. For example, the electronic equipment is based on 386 IBM computers, which can be found in museums today. We expect a period of about two years for the planning work. Backfitting of the old mine and construction work will need another 4 years.

The opening of repository is scheduled for the year 2013.



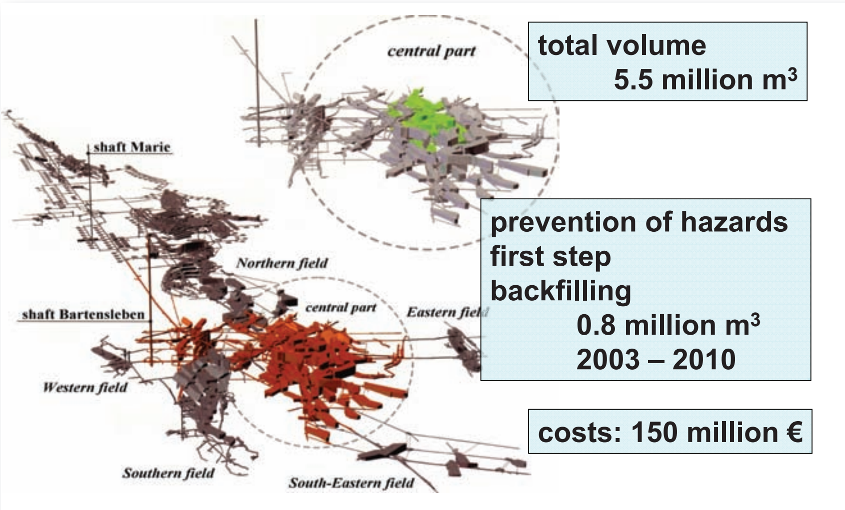
▲ Fig. 3: KONRAD – timescale

Installation and operation until opening will produce costs of about 900 million Euros. Together with the costs accumulated until today the total costs will be 1.8 billion Euros. An operation period of 30 years is said to be sufficient for the disposal of all radioactive waste with negligible heat production. While Konrad will be the new state-of-the-art repository there are two abandoned repositories for low and intermediate level wastes, that have to be closed.

3 MORSLEBEN Repository

The Morsleben repository will be the first one, worldwide, that has to be closed according to Atomic Act.

The abandoned salt mine has a history of more than hundred years. In 1971, the former German Democratic Republic (GDR) started disposal of about 14,000 cbm low and intermediate level waste. In contrast to Western Germanys Atomic Act the GDR-license for permanent operation does not include the closure of repository. After German Reunification the disposal of more than 22,000 cbm radioactive waste went on, until it was stopped by court in 1998.



▲ Fig. 4: MORSLEBEN – mine openings

The activity of the disposed waste lies in the range of 10^{14} Bq. After BfS renounced further disposal in 2001, the plan for closure was filed for the plan approval procedure four years later.

The entire cavities are shown in threedimensional model of in the Morsleben repository (Figure 4).

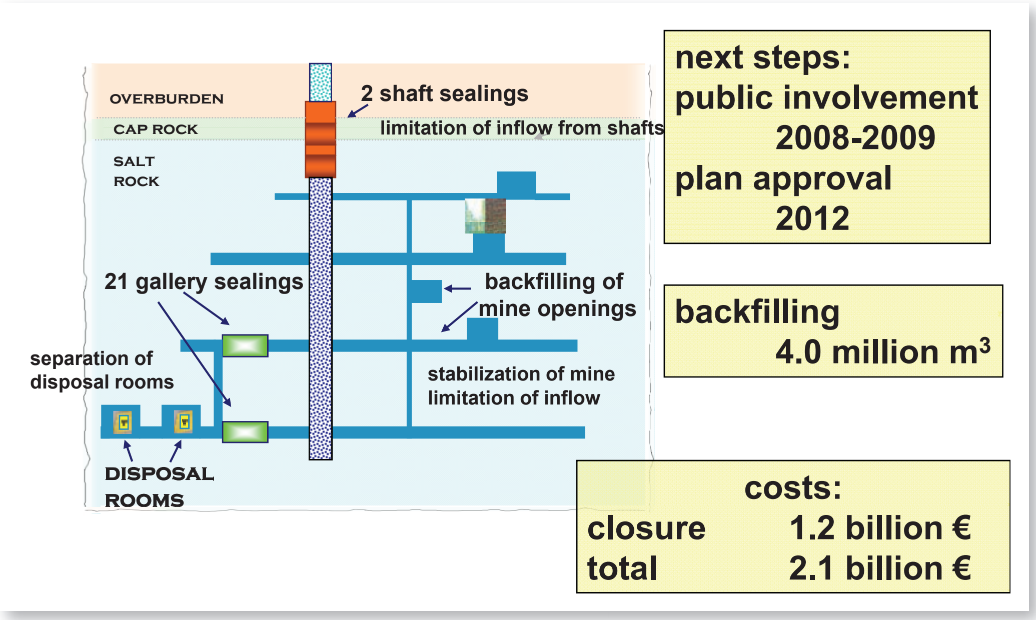
Due to extensive mining activity the total volume of mine openings is in the range of 5.5 mio. cbm. To prevent hazards from destabilized parts of the mine, and to make a controlled closure possible, backfilling of the Central Part with salt concrete started in 2003.

Until 2010, about 0.8 mio cbm will be backfilled, which will cost in total about 150 mio €.

The disposal of radioactive waste occurred on the 500 m level in different fields and manners. In the Western and Eastern Field the bins were stacked. Contrary to that, in the Southern field they were dumped in a haphazard manner, which makes retrievability nearly impossible.

The 3 main components of the closure concept are shown in Figure 5.

The backfilling of about 4 million cbm of mine openings with salt concrete shall stabilize the mine and limit the inflow of unsaturated brines. 21 gallery sealings will separate the disposal rooms from the rest of the mine. 2 shaft sealings shall prevent the inflow of groundwater from the overburden.



▲ Fig. 5: MORSLEBEN – closure concept

We believe, that long-term safety can be guaranteed by this closure concept. Public participation will begin next year and in the next 5 years we expect the plan approval.

The closure concept is estimated to cost about 1.2 billion €, in total more than 2 billion €, which will have to be paid by the German taxpayers in the end.

4 ASSE Research Mine

It was not only Eastern Germany, which had its example of wrong site selection, but West Germany too.

Until 1978, trial disposal of more than 47,000 cbm took place in the abandoned salt mine Asse.

There are several similarities between Morsleben and Asse, such as the age of mine (more than 100 years), the volume of mine openings and resulting safety problems, the kind of dumping, and the radioactive inventory (about 10^{15} Bq).

But, this mine was licensed as a research mine by mining act and is operated by the National Research Centre for Environment and Health (GSF), and not by the operator of repositories BfS. There is a paper by Günther Kappei from GSF [2], which is recommended for further reading.

At present there is a public discussion going on, how long stability of the mine can be assured and whether the closure should be licensed under atomic act.

4.1 Lessons learned from the “Oldies”

From these two examples several lessons can be learned:

- First, if we have an abandoned salt mine – we must not use it as a repository.
- Second, Asse and Morsleben show the importance of a good site selection. The costs for remedy and closure are much higher than the costs for site selection.
- And, sceptics are not always wrong; they should be involved in the site selection procedure.

If the site selection is important for low and intermediate level waste, this is even more true for high-level waste. This leads to the open solution for high-level waste (HLW) disposal, where we still have no political consensus in Germany.

4.2 Steps to a HLW repository

Gorleben (Figure 6) was selected in the 1970´s as possible site for nuclear disposal centre. It was mainly a political decision, and not the result of a transparent site selection procedure to find the best available repository. Surface exploration started in 1979 and was extended between 1996 and 1999. Underground exploration started 1983 and was extended 1996.

A lot of information was collected, but the suitability has still not been verified.

The verification of suitability will need performance assessments and more exploration.

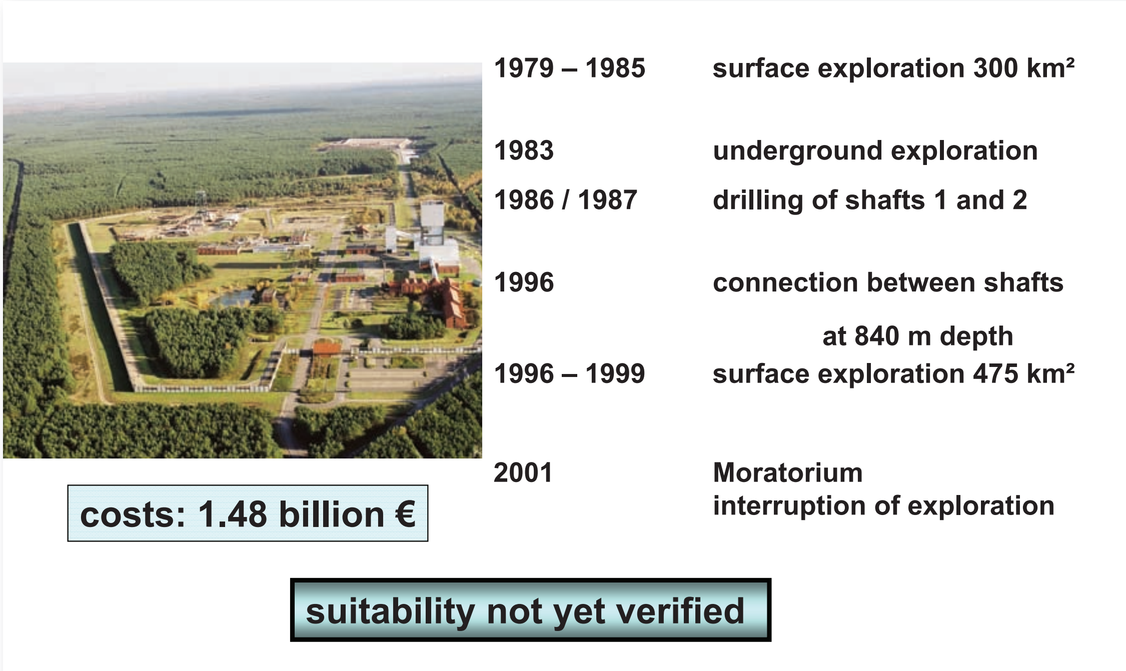
In 2001, exploration of the Gorleben salt dome was interrupted by a moratorium, based on the agreement between government and energy providers. The moratorium should be used for the clearing of fundamental questions.

Since 2001, several steps for site selection of a high-level waste repository were made, or are still under progress:

- **Committee on Repository Site Selection (AKEnd)**

    final report 2002

    basic conditions / selection criteria



▲ Fig. 6: Exploration mine GORLEBEN

- **conceptual and safety related issues**

  BfS report 2005

  no preference of one host rock in Germany

- **safety requirements**

  BfS workshop 2007

  expected by 2008

- **project "VerSi" - comparing performance assessments**

  first results expected by 2008

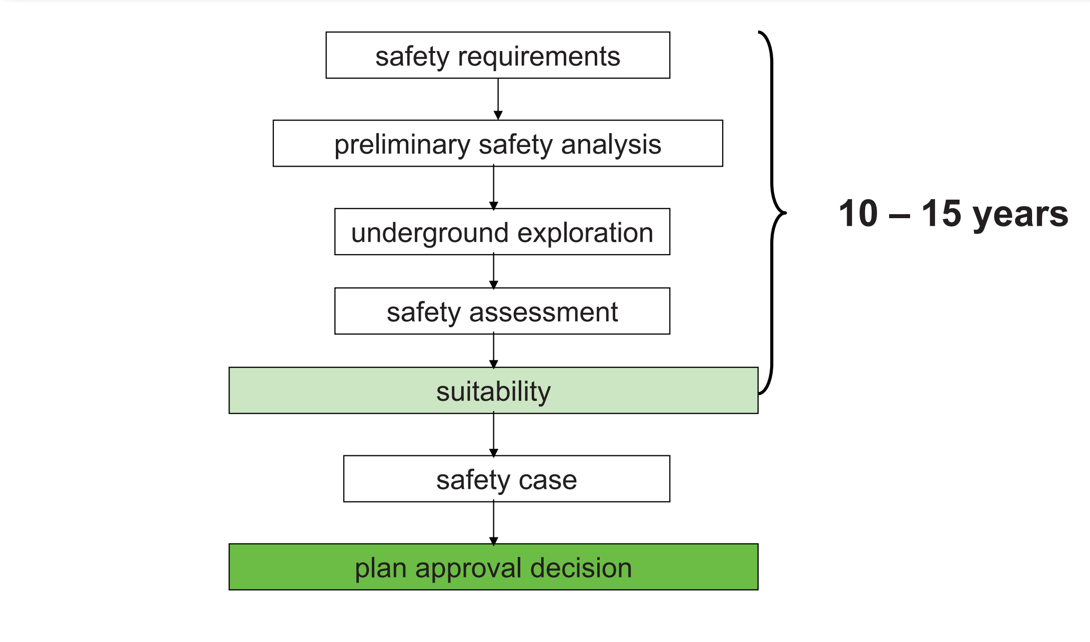
In 2002, the Committee on Repository Site Selection (called AKEnd) published its final report [3], in which basic conditions and site selection criteria were developed. In 2005, the results of 12 conceptual and safety related issues were published by BfS [4].

At present, the safety requirements for disposal of high-level wastes are under discussion [5], [6], [7], [8] and will be published as an ordinance soon. We will have two talks on this tomorrow. With our new project VerSi, the tools for comparing performance assessments will be developed.

Based on the agreement between the government and energy providers 12 conceptual and safety related issues were answered by BfS. The approach was to solve these issues on an objective scientific basis, which comprised commissioning of contractors, peer reviews by international experts, and discussion on a workshop [4].

The main results are:

- no host rock in Germany generally provides the highest level of safety.
- Moreover, repository concepts can be adopted to all possible host rock types.
- Different options for repositories can only be compared by site specific comparison.
- Therefore we conclude, that a site selection procedure is necessary to find the best option for a high-level waste disposal.



▲ Fig. 7: Required steps for suitability of HLW repository

A site selection procedure to find the best option for repositories is international state-of-the-art. But there are some open questions concerning the comparison of performance assessments in different host rocks. That is the objective, where our new scientific project, VerSi for comparing performance assessments comes in.

The works include definition of the extent of comparison, the identification of evaluation parameters and the development of an array of evaluation. After basic work, performance assessments for different host rocks and sites will be executed. First results will be expected by 2008.

Figure 7 shows, where we stand now, on the way to a suitability declaration for one or more sites. After definition of safety requirements, preliminary safety analysis shall deliver the targets for further surface and underground exploration. The results of exploration will be the basis for further safety assessments.

Even for the well explored site, Gorleben, we assume a timeframe of 10 to 15 years until its suitability can be verified. This is enough time for a site selection procedure to find the best available option for a high-level waste repository.

5 Conclusions

A solution was found for the disposal of low and intermediate level wastes in Germany with the Konrad repository, which will be opened by 2013.

The safe closure of the old repositories Asse and Morsleben, which seemed to be bargains in the seventies, are big challenges today, and have been producing high costs.

Based on that experience, we believe that an effort for site selection is a necessary investment for future safety and should be implemented for a high-level waste repository in Germany. ■

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1.09 Establishing a Safety and Feasibility Case for Geological Disposal of Belgian HLW-ILLLW into Boom Clay

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Abstract

This paper describes the sequence of decisions that will be taken in a reference scenario for stepwise development of a solution for the long-term management of high-level and intermediate-level, and/or long-lived wastes in Belgium, based on disposal in Boom Clay. It also explains the roles of the safety strategy and of the safety and feasibility case in this scenario. The process and current stage of application of the safety strategy is then described in more detail. The safety strategy relates to the stepwise and iterative development of a repository concept and design, and its assessment in terms of both safety and feasibility. According to the safety strategy, the assessment of a concept and design is a two-part process that involves firstly the development of a structured set of safety and feasibility statements, and secondly the evaluation of the levels of support for these statements. In order to guide the progressive refinement of a concept and design, safety and feasibility statements are evaluated according to:

- *How well supported a statement needs to be, or how critical the statement is, in the context of the current programme stage and of future stages?*
- *What level of support is available, or is likely to become available according to the current planning of the RD&D Programme?*

Safety and feasibility statements are generally expressed initially as hypotheses and are subsequently developed into increasingly well-substantiated claims as the design and implementation procedures are developed and optimised, and the evidence, arguments and analyses that support a statement are acquired or developed. A key objective of RD&D can be seen as developing the assessment basis needed to convert safety and feasibility statements from hypotheses to well-substantiated claims. The evaluation of the support available for safety and feasibility statements at a given programme stage can guide the prioritision of RD&D work at subsequent stages.

1 Introduction

Radioactive waste management in Belgium is the responsibility of the Belgian Agency for Radioactive Waste Management, ONDRAF/NIRAS. ONDRAF/NIRAS is, in particular, responsible for developing and implementing a solution for the long-term management of high-level and intermediate-level, and/or long-lived-wastes (HL-ILLLW), which are called B&C wastes in the Belgian Programme.

Studies in Belgium related to geological disposal of category B&C wastes were launched by the Belgian Nuclear Research Centre SCK·CEN, in the mid-seventies, and quickly focused on clays, and, more specifically, on Boom Clay. A key development in these studies was the construction of the underground research laboratory in the Boom Clay at Mol, in northeastern Belgium. These studies have been continued by ONDRAF/NIRAS, with continuing close collaboration with SCK·CEN. Promising results have led to geological disposal in Boom Clay becoming adopted as the ONDRAF/NIRAS reference solution for the long-term management of category B&C wastes, although without prejudging the choice of a disposal site, which is a decision for the responsible authorities to make in due course. According to the ONDRAF/NIRAS SAFIR [1] and SAFIR 2 reports [2], which assessed safety and feasibility of a disposal system based on this reference solution, and as confirmed by the international peer review of the SAFIR 2 report by the Nuclear Energy Agency of the Organisation for Economic Cooperation and Development [3], Boom Clay shows good promise as host formation for a disposal system for considered waste (mainly vitrified HLW and spent-fuels). This provides sufficient confidence for ONDRAF/NIRAS to move forward.

Despite international recognition of the quality of the scientific and technical work carried out by the B&C Programme, geological disposal in Boom Clay has never been discussed at the societal level in Belgium and has never been formally confirmed by the Federal Government as its policy for the long-term management of category B&C wastes. The ONDRAF/NIRAS work programme is thus still at the stage of "methodological" RD&D (research, development and demonstration).

Whilst preparing for the launch of a societal dialogue around its B&C Programme, ONDRAF/NIRAS is currently working towards preparing a first safety and feasibility case[1] (SFC 1), on the basis of which it intends to ask the Federal Government for authorization to proceed with the designation of a site for a disposal facility. Disposal in Boom Clay is taken as a working hypothesis for the development of SCF 1. Should this working hypothesis not be confirmed, the objective and content of SFC 1 will have to be adapted accordingly.

2 Stepwise Development of the Belgian B&C Programme

2.1 Caveats and Basis Hypotheses

The orientation of work being carried by the ONDRAF/NIRAS B&C Programme is currently based on a set of working hypotheses, the most important of which are as follows.

- The long-term management of category B&C wastes must take place on Belgian national territory.
- The reference solution for the long-term management of category B&C wastes is disposal in a deep geological formation (geological disposal).
- The potential host formations for a geological repository are limited to argillaceous formations.
- The argillaceous formations that are likely to host a repository are poorly indurated clays.
- The reference host formation for most RD&D activities will continue to be Boom Clay.
- The long-term management solution will be implemented as soon as possible, taking account of the relevant scientific, technical and societal factors and of the availability of the wastes.

2.2 Reference Repository Development Scenario

Planning within the B&C Programme envisages a sequence of key decisions related to repository development that the competent authorities will take. This sequence of decisions, assuming these to be positive, will allow development to proceed according to the reference repository development scenario. According to this scenario, ONDRAF/NIRAS will firstly,

- obtain confirmation of the main work hypotheses of the B&C Programme;

and secondly,

- move the B&C Programme in a stepwise manner towards the final choice of a disposal site and licence application.

[1]The term "safety case" is more widely used internationally (see, e.g. NEA 2004), although aspects of feasibility are generally included. By using the term "safety and feasibility case", ONDRAF/NIRAS emphasises its view that ensuring feasibility (including operational safety), is as important a consideration as post-closure safety in developing a geological repository.

The main decisions to be taken are as follows.

- **"Go for disposal in Boom Clay".** ONDRAF/NIRAS will seek a decision-in-principle confirming a preference for geological disposal in Boom Clay and requiring that the final disposal site should be located in a specific zone within the Boom Clay that is designated at the regional scale. A positive decision currently seems justifiable from a scientific and technical point of view, but also needs to be accepted from a societal point of view.

- **"Go for siting".** ONDRAF/NIRAS will seek authorization to launch the process that should lead to the final choice of a disposal site, where this process comprises, among other tasks, the identification of one or several potential disposal sites on the basis of expressions of interest by potentially interested municipalities and the confirmation of the scientific and technical suitability of this or these sites by RD&D.

- **"Go for licensing".** Having selected a disposal site, ONDRAF/NIRAS will seek authorization to launch the detailed studies that will be necessary to build the licence application files required for the next decision.

- **"Go for implementation".** ONDRAF/NIRAS will prepare for the application of the licences (nuclear and non-nuclear) required for implementation.

According to the reference repository development scenario, a positive decision to "go for disposal in Boom Clay" would be taken by the competent authorities on the basis of the ONDRAF/NIRAS Waste Plan, currently in preparation. The Waste Plan will be a global, strategic plan for the long-term management of radioactive wastes in Belgium, and will include a comparison of various options. It will play a key role in the societal dialogue that will be launched by ONDRAF/NIRAS.

Two successive safety and feasibility cases, SFC 1 and SFC 2, will serve as the scientific and technological basis on which the competent authorities will judge the achievements and working hypotheses of the B&C Programme and whether these are suitable to support positive decisions to "go for siting" (SFC 1) and "go for licensing" (SFC 2). The wider information base to support decision making will also include other considerations, such as the outcome of dialogue with stakeholders.

A safety and feasibility case is defined as follows (based on [4]): *« A safety and feasibility case is an integration of scientific and technological arguments and evidence that describe, substantiate and, if possible, quantify the safety and feasibility of, and the level of confidence in, the proposed long-term management solution for HLW/LILW-LL, i.e. geological disposal, at a given stage of development. It consists of a series of documents supporting the statements that:*

- *the proposed disposal system provides long-term safety if implemented according to design specifications;*

- *the proposed repository can be constructed, operated and closed according to these specifications (i.e. it is feasible).*

An SFC discusses the significance of any remaining uncertainty or open issue in the context of the. decision at hand in the process of repository development and provides guidance for work to resolve these issues in future development stages. »

SFC 1 is currently planned for 2013. To support the "go for siting" decision, it will aim to provide all necessary scientific and technical elements to make the case that, for a given zone in the Boom Clay, and for all the waste fluxes that can currently be predicted,

- the disposal system can ensure passive safety on the long term;

- the proposed design is feasible (where the notion of feasibility includes that of operational safety and of costs).

Should the decision be taken to proceed with siting, a SFC 2 will then be prepared, aimed at supporting the competent authorities in making the "go for licensing" decision, which will involve choosing (or confirming) the final disposal site and deciding on whether to give their go-ahead for the detailed engineering studies that will then be needed to prepare the license application files.

	Exploratory studies phase	Pre-project phase		Project phase
Target decision	Go for disposal in Boom Clay	Go for siting	Go for licensing	Go for implementation
Supporting scientific and technological case	Waste Plan	SFC 1	SFC 2	Licence application files

This reference repository development scenario provides an essential tool for ONDRAF/NIRAS planning. Responsibility for decision making, however, lies with the competent authorities, and B&C Programme planning may have to be adapted according to the actual decisions taken.

3 Overall Safety Strategy Approach

3.1 Process defined by the Safety Strategy

The long-term safety of any proposed solution will be a key consideration in the decision making process described in the previous section. A safety strategy has been developed by ONDRAF/NIRAS to define the process for deriving:

- a concept and design for the disposal of wastes in a suitable repository and procedures for repository implementation (system development); and
- the evidence, arguments and analyses to show that such disposal is both feasible and safe (system assessment and the development of a safety and feasibility case).

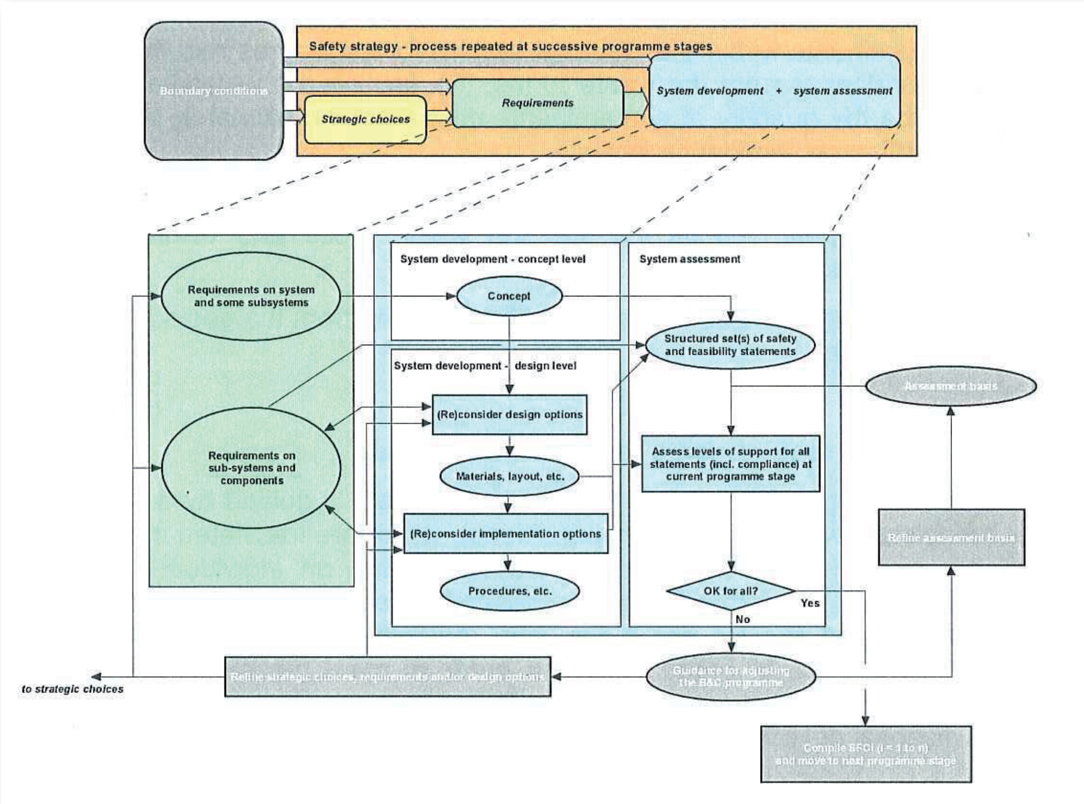
The process is outlined in Figure 1, and described in more detail in [5] and [6].

The term “concept” refers to a broad-brush description of the facility and the functions that the different components are intended to perform in order to protect the workforce during construction, operation, closure and institutional control of the facility (operational safety) and to protect the public and the environment in the longer-term (long-term safety). The term “design” refers to more detailed specifications of the surface and underground facilities.

In the safety strategy, the development of the concept and design is constrained by boundary conditions, and guided by certain strategic choices and requirements (that are themselves constrained by boundary conditions) and by the findings of safety and feasibility assessments. Development proceeds iteratively and concurrently with the development of the assessment basis - the scientific basis of these assessments - which comprises the empirical and theoretical scientific knowledge and understanding that are relevant to the disposal system under consideration, as well as various models, computer codes and data (also termed the “toolbox”) that can be applied in the assessments.

Refinements to the detailed design - and if necessary the underlying concept - may be made as the programme progresses through successive stages, in order (i), to better adapt the repository to relevant boundary conditions (ii), to take advantage of advances in science and engineering and (iii), to enable a better-substantiated safety and feasibility case to be made.

A particular role of the safety and feasibility assessments in this context is to identify any weaknesses in the concept and design and in the assessment basis, and to help define and prioritise that RD&D needed to resolve these weaknesses.



▲ Fig. 1: Overview of the safety strategy

3.2 Boundary Conditions

In implementing the safety strategy, various boundary conditions must be taken into account that are either outside the direct control of ONDRAF/NIRAS, or were set by ONDRAF/NIRAS early in the programme. Boundary conditions include, for example, the principles, standards, directives, recommendations and elements of good practice that are internationally recommended. They also include the long-term safety functions that a disposal system should fulfil, namely:

- Function of isolation (I function): the function that consists of isolating the wastes durably from man and the environment, by (1), preventing direct access to the wastes and (2), protecting the disposal system from potentially detrimental processes occurring in the environment of the disposal system.

- Function of engineered containment (C function): the function that consists of preventing for as long as required the dispersion of contaminants from the waste forms and the escape of gaseous substances, by using one or several appropriate barriers.

- Function of delay and attenuation of the releases (R function): the function that consists of retaining the contaminants within the disposal system for as long as required, by (1), limiting contaminant releases from the waste forms, (2), limiting the water flow through the system and hence the quantity of contaminants migrating and ultimately leaving the system and (3), retarding contaminant migration.

These safety functions apply to any geological repository, irrespective of programme-specific considerations, such as national regulations and the waste types and volumes to be disposed of, although terminology may differ.

Further boundary conditions should also include conditions arising from the societal dialogue and from the local stakeholders.

3.3 Strategic Choices and Requirements for HLW and ILW-LL

According to the safety strategy, concept and design development is guided by a number of strategic choices (consistent with the boundary conditions), which are translated into a set of requirements on the system as a whole, on sub-systems and on individual repository components. Consideration is given to what are the essential elements of the repository and its environment, and strategic choices are made, and requirements set, concerning what these elements will need to do, in broad terms, in order to meet the most fundamental objective of providing long-term passive safety. The following strategic choices have been made regarding the disposal of HLW and IL-LLW, constrained by current boundary conditions:

- Given the working hypothesis that the Boom Clay will provide the host rock for a repository for these wastes, the repository shall be constructed at depth within this formation, with the overlying sedimentary formations providing the geological coverage.
- The materials and implementation procedures shall not unduly perturb the safety functions of the Boom Clay, or any other component on which safety depends.
- In the case of heat-generating wastes, the engineered barriers shall be designed to provide complete containment of the wastes and associated contaminants at least through the period when the heat output from the wastes is high (no similar requirement for a containment is set for other wastes) to avoid the necessity to model contaminant transport during the thermal phase.
- Wastes shall be divided into groups to be emplaced in separate sections of the repository.
- Repository construction and operation shall proceed as soon as possible, but taking due account of scientific, technological, societal and economic considerations.
- The different repository sections and the repository as a whole shall be closed (access routes backfilled and sealed) as soon as practically possible following emplacement of the wastes.
- There are preferences for permanent shielding, and for minimisation of operations in the underground.
- There are preferences for materials and implementation procedures for which broad experience and knowledge already exists.
- Repository planning shall assume that post-closure surveillance and control will continue for as long as reasonably possible, in order to reduce the likelihood of deliberate (unauthorised) or inadvertent human intrusion, but taking into account the resources that this will require.

Some of the boundary conditions and strategic choices described above translate directly into requirements on the system, subsystems and components. An example of a subsystem requirement translated directly from a strategic choice is the requirement that the engineered barriers must provide complete containment of heat-generating wastes and associated contaminants at least through the period when the heat output from the wastes is high.

Further high-level requirements derive from the need to meet the twin objectives of safety and feasibility. For example, for feasibility, there is a requirement that the costs for the construction, operation and closure of the repository, for the dismantling of the surface facilities and for the post-closure surveillance and control are covered by the current funding mechanism. High-level safety and feasibility requirements give rise to more specific requirements on subsystems and components (including implementation procedures), which are derived concurrently with the concept and design in an iterative process, as the subsystems and components are defined.

4 Current Disposal Concept

Based on current strategic choices, a reference concept and design for a repository for B&C wastes has been developed.

In the reference concept, the following broad elements contribute to long-term safety:

- the host rock and geological coverage;
- the access route backfill and seals;
- the engineered containment barriers; and
- the waste matrices and primary containers.

These elements are common to all concepts considered internationally, although their realisation in design (materials, dimensions, etc.) varies depending on boundary conditions such as the specific wastes to be disposed of and the geological settings under consideration.

In the current ONDRAF/NIRAS concept [7], consistent with the strategic choices, the repository is constructed at depth in an approximately 100 m thick Boom Clay layer, with the overlying sedimentary formations providing the geological coverage. The concept for surface and underground facilities is illustrated in Figure 2, which also shows the emplacement of HLW and ILW-LL in approximately horizontal disposal galleries in spatially separated sections of the repository. Access to the disposal galleries is through a series of shafts and an access gallery. The underground facility includes engineered containment barriers that provide complete containment of heat-generating wastes and associated contaminants at least through the period when the heat output from the wastes is high.

The engineered barrier design for vitrified high-level waste and spent fuel has been extensively reviewed and modified since the SAFIR 2 report (Figure 3 and [6]). In the case of vitrified high-level waste and spent fuel, wastes will be emplaced in primary containers that will be placed in the galleries in metallic overpacks, surrounded by a buffer contained in an envelope. The galleries themselves will be located centrally within the Boom Clay layer. The primary containers, overpacks and buffer together comprise the engineered containment barriers (see also [8]).

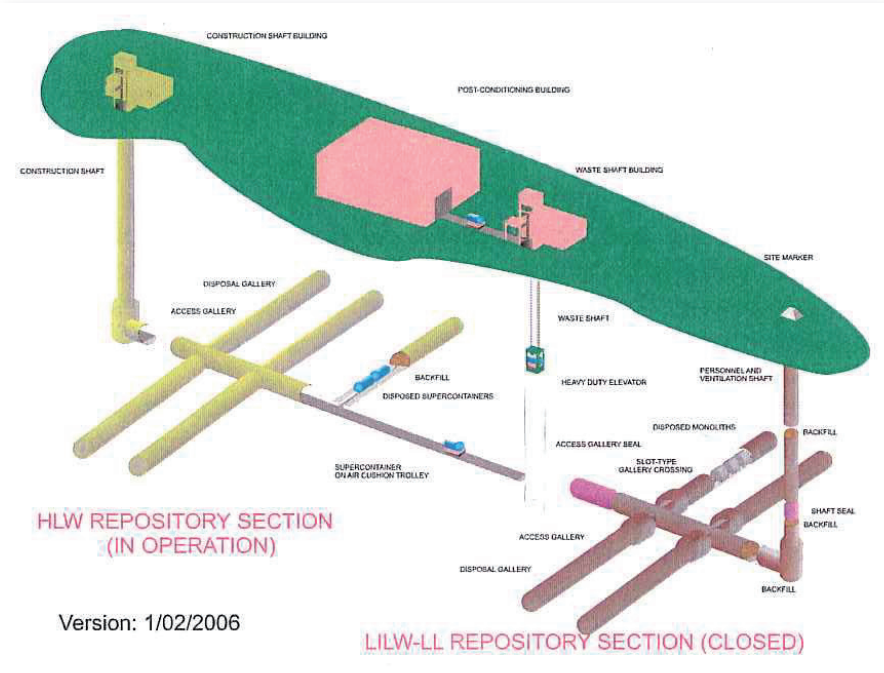


Fig. 2: Schematic lay-out of the underground repository facilities and of the related surface facilities

The galleries for high-level waste and spent fuel have typical internal diameters of 3 m, and are lined with concrete wedge blocks to stabilise the excavated galleries against clay convergence (Figure 3). The reference materials for the overpack, the buffer and the envelope are carbon steel, concrete and stainless steel, respectively. Disposal galleries are backfilled with a cement-based material.

The chain of activities needed to implement the current reference concept and design has not so far been planned in detail. Some tentative decisions regarding implementation have, however, been made, consistent with the strategic choices outlined above. Examples are as follows.

- The decision to opt for phased implementation, whereby different parts of the repository will be constructed, operated and closed separately and according to different time schedules.
- The decision to assemble key engineered components of the repository at the surface (for instance, in the case of vitrified high-level waste and spent fuel, the overpack containing the wastes, the buffer material and a surrounding envelope).
- The decision to maintain some form of post-closure surveillance and control for as long as reasonably possible.

In future programme stages, the disposal system components related to the feasibility of implementation, as well as the implementation procedures themselves, will be specified in detail.

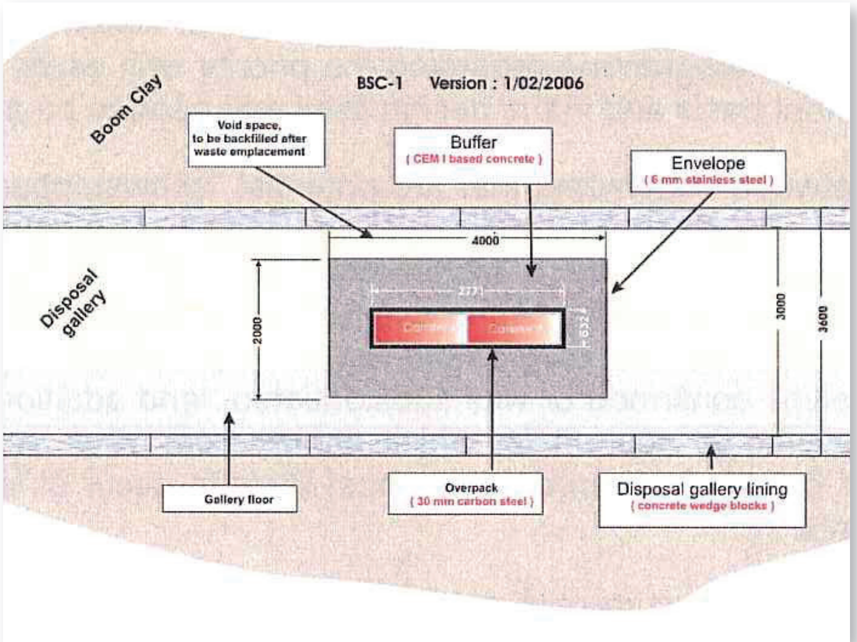


Fig. 3: The disposal gallery design for vitrified high-level waste

5 From Hypotheses to Claims: An Approach to Prioritise RD&D

5.1 Approach

Given the range, complexity and long duration of the RD&D undertaken in the course of repository planning, it is important that the safety strategy ensures RD&D activities remain focussed on safety and feasibility objectives, and concentrate on any major uncertainties or open questions that could call into question either the safety or feasibility of a given concept and design.

According to the safety strategy process illustrated in Figure 1, the assessment of a concept and design in terms of its safety and feasibility is a two-part process that involves:

- development of a structured set of safety and feasibility statements; and
- evaluation of the levels of support for these statements, in which the assessment basis and safety and feasibility assessments playa key role.

Safety and feasibility statements generally begin as hypotheses (i.e. statements of the type "the repository and/or its components should ... "), which may initially be tentative. These are developed into increasingly well-substantiated claims (statements of the type "the repository and/or its components are expected to ...") as the design and implementation procedures are developed and optimised, and the evidence, arguments and analyses that support a statement are acquired or developed. Thus, a key objective of RD&D can be seen as developing the assessment basis needed to convert safety and feasibility statements from hypotheses to well-substantiated claims.

In order to guide the progressive refinement of a concept and design, safety and feasibility statements are evaluated according to:

- How well supported a statement needs to be, or how critical the statement is, in the context of the current programme stage and of future stages?
- What level of support is available, or is likely to become available according to the current planning of the RD&D Programme?

A classification scheme is proposed in Table 1 (although it has yet to be fully implemented and may be modified in the light of experience). The classification of statements is intended to provide guidance to the RD&D programme regarding the priority with which uncertainties or deficiencies in the assessment basis and in the design itself that need to be addressed.

The scheme differentiates between statements that are potential "showstoppers" and those that are simply "nice-to-have". Potential "showstoppers" are those statements that, if not confirmed or well substantiated, undermine a higher-level potential "showstopper", and, ultimately, the long-term safety and feasibility of the overall concept and design and hence the possibility of moving forward to the next programme stage. "Nice-to-have" statements are those that, if they are themselves confirmed or well substantiated, lend additional support to a higher-level statement, the primary support for which comes from other sources. These "nice-to-have" or "confidence building" statements are most likely to occur at lower levels in the structured set of statements.

Potential "showstoppers" for which the current level of support falls into categories R or Y (i.e. SR and SY statements, in the scheme shown in Table 1) are clearly the highest priorities for further RD&D work (SY statements are judged to be adequately covered by the existing RD&D Programme, whereas SR statements are not). "Nice-to-have" statements for which the current level of support falls into these categories are potential lower-priority targets for the RD&D Programme.

It should be noted that there are other considerations in planning a RD&D Programme in addition to the classification assigned to safety and feasibility statements. There are, for example, some fundamental areas where the maintenance of expertise within the programme is considered important, even if the level of understanding already available is considered adequate for future safety and feasibility cases. Furthermore, even if the level of support for a statement is judged already to be adequate, general scientific development must be monitored in order to have confidence that this judgement continues to hold in the future.

Table 1: Proposed classification scheme tor safety and feasibility statements

		Type of statement	
		Potential "Showstopper" (S)	"Nice-to-have" (N)
Adequacy of the level of support judged to be available (or potentially available) with respect to the programme milestone at hand	Adequate support by the next programme milestone is likely to require changes in the planned RD&D Programme, or in the proposed design (R - red)	SR	NR
	Adequate support is likely to be available by the next programme milestone, based on current work planned in the RD&D Programme (Y - yellow)	SY	NY
	Adequate support for the statement is judged to be already available (G - green)	SG	NG

The meaning of "adequate support", as used in Table 1, will change as the programme proceeds. At early programme stages, scoping calculations alone provide adequate support for at least some safety and feasibility statements. At later stages, and especially before starting construction and operations, all potential "showstoppers" will need to be well substantiated, ideally backed up by multiple lines of argument, including, for example, fullscale demonstrations.

5.2 The Structuring of Safety and Feasibility Statements

Safety and feasibility statements are developed and structured in a top-down manner, starting with the most general (highest-level) statements and progressing to increasingly specific (lower-level) statements. Lower-level statements are generally statements about what the system is designed or intended to do, or properties that it should have, in order to satisfy higher-level statements.

In terms of long-term safety, the objective of concept and design development is to arrive at a disposal system which, *if implemented according to design specifications, together with its environment, will conform to relevant regulatory requirements and general guidance concerning long-term safety*. The italicised text is regarded as the most general, highest-level safety statement (claim) that requires substantiation or underpinning by lower-level safety statements.

Supporting the highest-level safety statement are firstly (assuming these to be available) the positive findings of a formal safety assessment, which includes the results of quantitative analyses of dose or risk for a wide range of evolution scenarios. Secondly, but at the same level in the hierarchy, are statements that are underpinned, though generally in a more qualitative sense, directly by the assessment basis. These relate to evidence concerning the individual long-term safety functions that the disposal system is expected to provide and concerning the dilution and dispersion of any contaminant releases from the repository in the biosphere and aquifers. They include, for example, the statement that the disposal system will delay and attenuate releases to the environment. This will be supported, at the next level down, by the statement (among others) that the Boom Clay will delay and attenuate releases over a prolonged period. This will, in turn, be supported by statements about the low permeability, fine homogeneous pore structure and favourable geochemical conditions in the Boom Clay, evidence for which forms part of the assessment basis.

Although formal safety (and feasibility) assessments are carried out only every few years, support from statements underpinned directly by the assessment basis is informally assessed continuously in the intervening periods in order to judge whether or not the design and assessment basis are sufficiently well developed in terms of long-term safety to justify a positive decision to proceed with a formal assessment. If the judgement is negative, then the design and assessment basis must be enhanced before proceeding with the assessment.

5.3 Some Examples of Assessment and Substantiation

There is already substantial evidence underpinning some lower-level safety and feasibility statements for the current reference concept and design, developed, for example, in the context of SAFIR 2 [2], as well as relevant studies carried out internationally. A first assessment of the available support for the safety and feasibility statements has been carried out recently, to test the proposed methodology.

Below are presented the high-level statements for heat-emitting waste that are underpinned directly by the assessment basis. These statements essentially represent the long-term safety functions that the disposal system is expected to provide, as well as the dilution and dispersion role of the aquifer.

- **SS.1:** The disposal system and its environment isolate the waste to minimize the probability and consequences of human intrusion, and to protect the repository against surface and subsurface events and processes during the "stable geological environment phase".
- **SS.2:** The engineered barrier system of vitrified HLW and spent fuel (is expected to) provide complete containment of radionuclides during the thermal phase, ensuring zero release.
- **SS.3:** The disposal system (is expected to) delay(s) and attenuate(s) releases to the environment during the "system containment phase", ensuring that releases remain below regulatory targets/standards and general guidance.
- **SS.4:** The actual dilution and dispersion by the environment (biosphere and aquifers) of the disposal system and transfers of radionuclides in the biosphere to man can be sufficiently quantified and their (this) safety roles can be conceptualised for long-term safety assessment calculations.

The next textbox (last page of the paper), illustrates how one of these high-level statements is supported by a set of lower-level statements. The development of a safety and feasibility statement hierarchy is, however, work in progress and many of the statements are still under discussion. The aim is that relevant information, including uncertainties and open questions, are incorporated in the statements and in their assessment according to the scheme shown in Table 1. Lowest-level of the statements make reference to reports of past research, or to research plans if they concern work in progress. If a statement is judged not to be adequately well supported for the purposes of the next programme milestone (currently SFC 1), the kind of activity that will be needed to acquire the necessary support is indicated. In the future, a completeness check of the statements against a disposal system specific FEP list is foreseen. This completeness checking will be undertaken to ensure, as far as possible, that the scientific basis for the safety assessment considers the full range of relevant FEPs.

5.4 Some RD&D Priorities for the Belgian Programme up to SFC 1

As explained in section 5.2, the safety strategy provides a methodology for reviewing RD&D activities and establishing priorities for future work.

ONDRAF/NIRAS and its scientific partners have recently finalised a first application of the methodology. All relevant information that is currently available, including uncertainties and open questions, has been considered in formulating and assessing the various statements (see the example given in the box at the end of the paper). Some priority issues for RD&D have already been identified that will need to be addressed to support future discussions on whether or not the statements are sufficiently well substantiated for the purposes of SFC 1.These include the need to:

- define the requirements related to the supercontainer and monolith design,

- define allowable perturbations of the Boom Clay properties,
- define requirements on the gallery and shaft seals (the large-scale Praclay Heater Test will be important in this respect, as it will address the THM(C) behaviour of the Boom Clay),
- provide a full overview of the expected waste fluxes (or groups of fluxes) and characteristics and check that these characteristics do not represent a thread vis-à-vis the current concept and design,
- estimate the gas generation and further transport for category B waste, considering the repository lifetime aspects (i.e. timing of gas generation, accumulation and migration vs. timing of radionuclide migration), and
- develop a lay-out for the global repository and surface installations.

The consolidation of the available knowledge, including RD&D financed through ONDRAF/NIRAS and results available in the open literature, into assessment basis reports is also considered to be a major challenge, as is explaining the strategic basis of the current RD&D programme. ■

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SS.3: The disposal system (is expected to) delay(s) and attenuate(s) releases to the environment during the "system containment phase", ensuring that releases remain below regulatory targets/standards and general guidance.

SS.3.1: In the predicted geochemical/thermal conditions in the buffer bulk, the vitrified HLW and SF waste forms spread the release of radiotoxic and chemotoxic waste components in the environment over x years (x to be specified).

SS.3.2: The transport in the host rock is diffusion dominated and this characteristic is assured for a long-term.

SS.3.3: The host rock and to a lesser extent the EBS have favorable characteristics to retard the radiotoxic and chemotoxic components and the host rock assures these characteristics for a long-term.

SS.3.3.1: The host rock has a favorable geochemical environment provoking a low solubility for most of the radionuclides (/contaminants).

SS.3.3.2: The host rock exhibits considerable sorption and ion-exchange capacity.

SS.3.3.3: The host rock has such a pore structure, that results in an effective filtering of colloids.

SS.3.3.4: The host rock possesses a sufficient chemical buffering capacity against changes stemming from the waste emplacement (so that SS.3.3.1 to 3.3.3 are not significantly altered or to a limited spatial extent).

SS.3.3.4.1: The chemical disturbances stemming from the alkaline perturbation from the EBS are known, have a limited extent and do not jeopardise the necessary thickness of host rock.

SS.3.3.4.2: The chemical disturbances of oxidation resulting from excavation and ventilation are known and the extent of the disturbance do not jeopardise the necessary thickness of the host rock.

SS.3.3.4.3: The chemical disturbances related to the temperature increase from the waste are known and will not jeopardise the favourable characteristics of the host rock.

SS.3.3.4.4: The chemical disturbances related to products released from the waste are known and do not jeopardise the favourable characteristics of the host rock ($NaN0_3$, (in-)organic ligands, ...).

SS.3.3.5: The favorable geochemical properties of the host rock are not jeopardised by external geological events and processes.

SS.3.3.6: Sorption characteristics of the EBS will make up a supplementary safety function and can be qualitatively assessed.

SS.3.3.7: Solubility characteristics of the EBS will make up a supplementary safety function.

1.10 Eight Years WIPP Progress

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Abstract

Contact-handled (CH) transuranic (TRU) defense waste has been trucked to the Waste Isolation Pilot Plant (WIPP) and disposed of 655 m deep in massive Permian rock salt for 8 1/2 years without releasing radionuclides to the environment or causing harm to employees or the public. Shipments of remote-handled (RH) waste commenced January 23, 2007. Virtually flawless performance helps maintain local and regional support. As of July 29, 2007, the repository accommodated 49 966 m³ CH-TRU waste and 15 m³ RH-TRU waste. Waste emplacement continues in Panel 4, while Panel 5 is being prepared. Disposal of up to ~175 000 m³ of waste at WIPP is projected to be complete about 2033, followed by five years of dismantling, decommissioning, and constructing permanent markers, with active institutional controls continuing 100 years beyond. Annual repository operating costs are in the $ 80-90 million range. The life-cycle cost (1977 through 2070) of national TRU waste management and disposal at WIPP (in constant 1998-dollars) has been estimated at > $ 10 billion. Lessons learned at WIPP present opportunities for other repositories to match or surpass its safety record while improving efficiency.

1 Basics

The Waste Isolation Pilot Plant (WIPP) east of Carlsbad, New Mexico, is the world‘s first deep geologic repository specifically designed and built to permanently isolate intermediate-level, long-lived, and negligible heat generating radioactive waste in

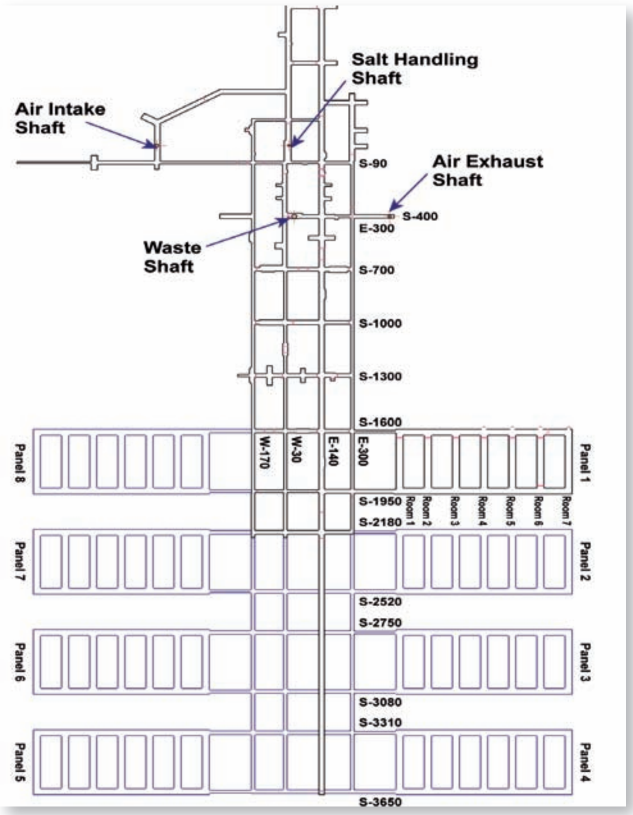


Fig. 1: Repository plan view (numbers indicate distances in feet from the Salt Handling Shaft)

Permian salt. The WIPP disposal area consists of excavations 655m below the surface that are not part of an active or former mine. The WIPP's principal isolation feature is the inherent nature of its host rock: impermeable and geologically ancient salt. The disposal horizon has not been affected by groundwater in 250 million years. More than 300 m of rock salt and an additional 300 m of non-salt rocks separate the repository from the surface. Under the weight of the combined overburden, the salt surrounding the excavations gradually creeps (at a vertical closure rate of up to 9cm per year) into the openings and encapsulates the waste. Four shafts form the only (temporary) connections between the surface and the repository level. All will be plugged with state-of-the-art multi-component seal systems (engineered barriers) when the repository is full. In addition, panel closure systems divide the disposal area into separate cells.

The U.S. Department of Energy (DOE) owns the surface and essentially all mineral rights in the so-called WIPP Land Withdrawal Act Area (LWAA), a square tract of land encompassing 42 km^2. No drilling or mining is permitted there in perpetuity. In about the center of the LWAA lies the disposal area proper (Figure 1), measuring less than 0.5 km^2. The disposal area footprint consists of rooms about 90m long and 10m wide, pillars between rooms 30m wide, and pillars between panels (each made up of seven rooms and six pillars in parallel) 60m wide. These dimensions result in an aggregate extraction ratio of less than 25 percent.

The official WIPP web site "www.wipp.energy.gov" offers a convenient portal to additional technical, scientific, and regulatory details. German language WIPP information is available at "www.endlagerung.de" under "Internationale Projekte". An interim status report in English was presented at the 2004 DisTec conference [1]. A German language status report on WIPP was given last fall at a TÜV Nord seminar in Hannover [2].

2 Safety

Inadvertent disturbance by drilling is the only barely credible scenario that could potentially cause the release from the WIPP of waste constituents in concentrations of regulatory concern. The repository is situated in an area rich in a variety of currently identified mineral resources. These range from near-surface caliche, through potash seams 420 to 540 m deep, to hydrocarbon reservoirs 1 400 m to more than 7 000 m deep. To mitigate concerns expressed by regulators and oversight groups, the DOE has prohibited any mining and drilling within the WIPP LWAA. The likelihood of inadvertent intrusion becoming reality is further reduced to an almost negligible chance by a variety of preventive administrative, institutional, and engineering measures. Consequences of any inadvertent intrusion scenario are actually fairly innocuous [3].

Underground stability has been a recurring theme during the life of the WIPP excavations (since 1982). Project delays (more than 10 years) caused some excavations to remain open far longer than originally intended. Mined openings close at vertical rates of up to 9 cm per year, and ground support requires continuous effort. But these conditions are similar to those in any salt or potash mine and are likewise managed routinely and safely. Beyond standard mining practice, however, the WIPP has installed more than 1 000 instruments to monitor rock behavior. Comprehensive geotechnical surveillance identifies deteriorating conditions long before they become hazards and leaves plenty of time for implementing remedial measures. Here, as in all other aspects of the WIPP, safety is always the first priority and is never compromised.

3 Changes

3.1 Accelerated Disposal

After about two years of operating the repository, the DOE decided to accelerate waste disposal beyond the relatively slow and steady throughput rate assumed during design and construction of the facility. This change required faster than anticipated mining. Excavation rates could be increased fairly simply by hiring more employees and running additional shifts. But the salt handling shaft hoist (Figure 1) could not accommodate the mined salt in concert with the accelerated pace of mining. The problem was solved by stockpiling the excess salt underground during the early (mining) shift, and hoisting it to the surface during the late (no mining) shift. This process involves double handling of a portion of the mined salt. The additional cost was justified by faster removal of dangerous waste from generator sites. The main lesson to be drawn from this experience is to design a repository to be very robust and capable of being adapted to changes in priorities.

3.2 Elevation of Disposal Horizon

The immediate roof above the original repository horizon consists of bedded rock salt, as does the entire host formation. Thin anhydrite and polyhalite beds underlain by very thin layers of clay are commonly interbedded with the salt (Figure 2). One thin anhydrite-clay bed combination (Anhydrite B and Clay G) lies about 2m above the roof in filled Panels 1 and 2 and future Panels 7 and 8. Creep deformation results eventually in bed separations along the anhydrite-clay interface, accompanied by downward bending and fracturing. Pattern rock bolting and routine maintenance are needed to maintain safe working conditions beyond a few years after initial excavation.

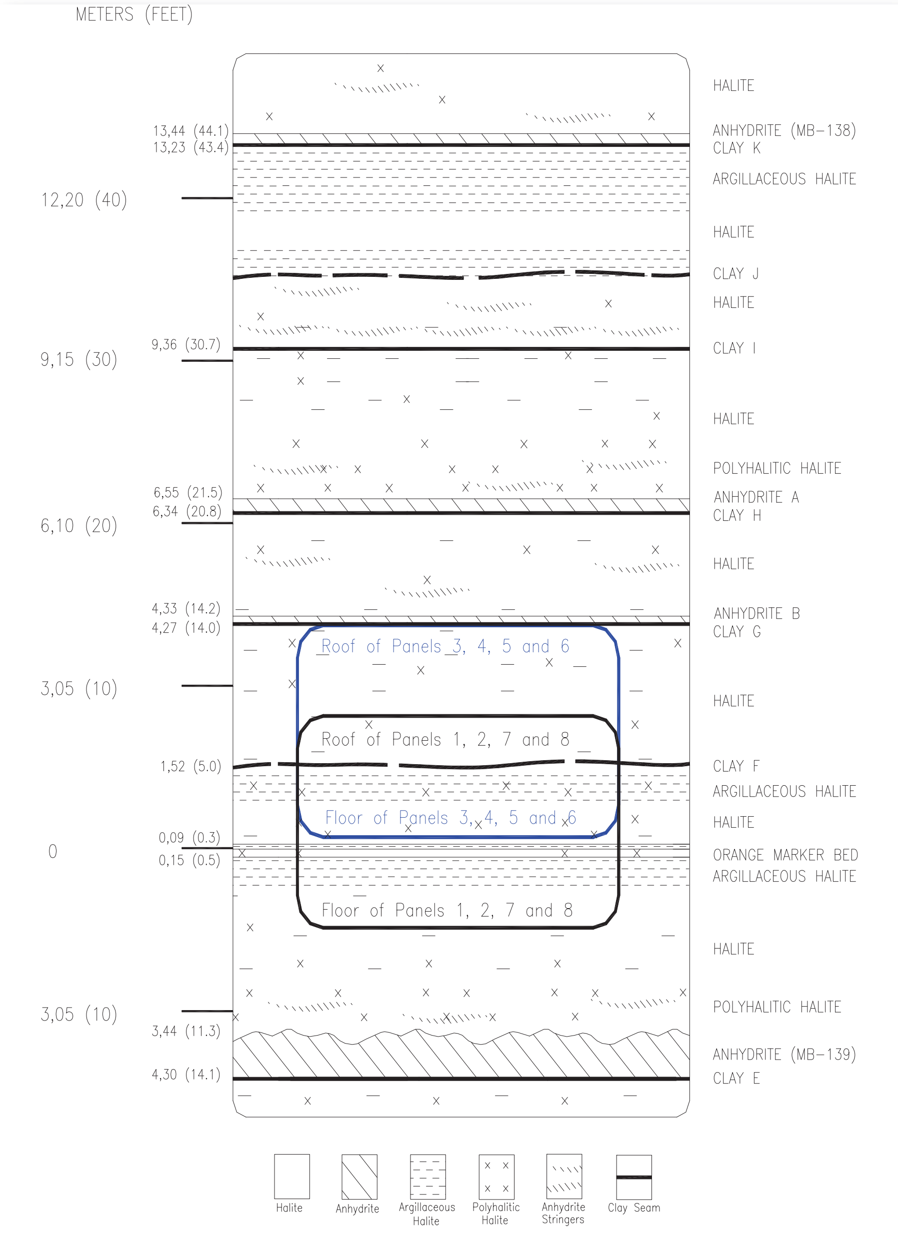


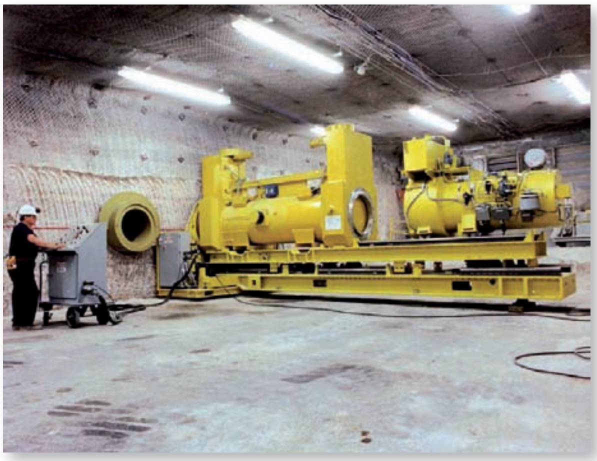
Fig. 2: Disposal Horizons

Practical experience and rock mechanical modeling indicated that removing the 2 m thick roof beam might increase the stability of excavation without significantly affecting long-term repository performance. This change, including raising the floor by a commensurate distance, was implemented beginning with the access drifts toward Panels 3-6 (Figure 1). But the hoped-for advantages of raising the repository horizon did not quite materialize. The 20-45 cm interval immediately above the new roof includes discontinuous anhydrite stringers, along which the roof beam began to separate soon after mining. Immediate bolting with epoxy anchors solved the problem, but also increased the cost of roof support and maintenance. The thin anhydrite-clay bed combination (Anhydrite A and Clay H) about 2 m above the roof in filled Panel 3, current Panels 4 and 5, and future Panel 6 behaves in a fashion similar to the Anhydrite B and Clay G combination.

3.3 Remote-Handled Waste

TRU waste is divided into two subcategories: contact-handled (CH) and remote-handled (RH). The maximum allowable container surface dose rate of CH waste is 0.002 Sv/h, and of RH waste 10 Sv/h, but the maximum allowable rate at the surface of the shielded RH casks is the same (0.002 Sv/h) as the one at the surface of the un-shielded CH containers. Ninety-six percent of the total permitted volume of WIPP will be filled with CH waste. RH waste canisters are inserted into horizontal holes drilled into the ribs of disposal rooms, while CH containers are stacked three layers high inside the rooms (Figures. 4 and 5). WIPP received only CH waste for almost eight years. Receipt of RH waste had originally been anticipated two to three years after initial receipt of CH waste. Instead, the canisters finally started arriving January 23, 2007. Panels 1-3 therefore contain only CH waste. Perhaps the most important lesson of the drawn-out RH permit and approval process is that dogged persistence succeeds eventually.

▲ Fig. 3: CH Waste Emplacement



▲ Fig. 4: RH Waste Emplacement

3.4 Panel Closures

Both access drifts are closed after each panel (Figure 1) is filled with waste. Panels 1 and 2 were closed with 3.7m-thick walls consisting of solid concrete blocks laid in a three-dimensionally interlocked pattern. Panel 3 was filled earlier this year and was closed with a wire mesh and brattice cloth combination whose lower 1/3 to 1/2 is embedded in a pile of run-of-mine salt. This arrangement allows continued monitoring of conditions inside the panel, while the Panel 1 and 2 closure systems are not amenable to penetration for monitoring.

4 Challenges

4.1 Backfill

Current US regulations require engineered barriers in addition to WIPP's natural rock salt barrier. WIPP emplaces bags of magnesium oxide on top of each three-high stack of CH waste containers as chemical backfill intended to keep the pH neutral to alkaline under theoretical humid or wet repository conditions. These conditions are highly unlikely, and their consequences to the environment and human health and safety would be negligible. Discussions continue whether to aim for elimination of this superfluous requirement in a future re-certification. The Environmental Protection Agency (EPA) must re-certify WIPP every five years so it can continue operating. WIPP must submit its next re-certification application by 2009.

4.2 Exhaust Shaft Water

The 5m-diameter exhaust shaft is lined with concrete through the overburden from the surface to the shaft key in the top of the salt 257m below the surface. The salt section below the key is lined with wire mesh anchored by rock bolts.

Three seepage zones within the interval from 16-28 m below the collar leak water into the shaft from a perched aquifer that is recharged as the result of surface modifications at the site. The combined flow rate from these zones is about 4ℓ per minute. Condensation and precipitation of warm, moisture-laden ascending air constitutes a much larger source of shaft water. Water from both sources can flow or rain down the shaft, can be absorbed by the shaft wall, or can be blown out of the shaft by the ventilation system.

Since fluid was first detected in 1995, catch basins at the base of the exhaust shaft collected accumulating fluid for disposal. An interception well system replaced the catch basins in early 2006. This system consists of four 10m-deep holes in the mine floor down-dip from the exhaust shaft bottom. Fluid is pumped from each borehole to a series of storage tanks. Volumes of fluid removed from the collection points fluctuate as a function of several variables, among them ventilation rate, humidity, and temperature. The ventilation rate is the primary factor: low ventilation allows fluid to accumulate, whereas high ventilation removes most of the fluid from the shaft.

How fluid collected from the exhaust shaft is managed depends on its chemical fingerprint. Occasional batches contain concentrations of lead that exceed drinking water standards. Under U.S. regulations, such fluid is classified as hazardous waste and must be treated or disposed of in specifically authorized facilities at great expense. The likely source of the lead is the galvanized wire mesh covering the lower shaft wall. Fortunately, most of the collected fluid can be disposed of in evaporation ponds on the surface.

4.3 Waste Retrieval

On July 18, 2007, WIPP officials were notified that one of four CH drums over-packed inside a standard waste box and shipped from Idaho had not been certified in accordance with WIPP waste acceptance criteria or cleared for shipment. Drums that do not meet Department of Transportation requirements are over-packed into containers that do.

A worker at Idaho National Laboratory misread a six-digit drum identification number. As a result, the wrong drum was sent. An x-ray showed that it contained liquid, which is prohibited in WIPP waste except for trace amounts (no more than 1% residual). The shipment left Idaho on June 23 and arrived at WIPP two days later, where it was subsequently placed underground. The mistake was not noticed until an inventory in Idaho on July 16. In the three weeks before the error was discovered, WIPP stacked 36 rows of waste containers in front of the problem drum, making retrieval quite complicated. All shipments to WIPP from Idaho were suspended pending corrective actions.

In a July 25 letter to the New Mexico Environment Department (NMED), WIPP officials expressed sufficient certainty about the drum to assess its potential impact and concluded that it posed no risk to human health and the environment. The only prohibited item in the drum was three-fourths of a cup of liquid that is not ignitable, reactive, or corrosive (most probably water-based solutions). To determine the danger that might be posed by the liquid, WIPP assumed a worst-case scenario - that instead of being water-based, the liquid was acid. Even under that scenario, the liquid would still not have a chance of eating through the drum and leaking out. Leaving the drum would be protective of human health and the environment. The letter specifically stated that more safety risks would be associated with retrieving the problem container than with leaving it in place.

Because free liquids are prohibited in WIPP waste, the state of New Mexico has the legal authority to force DOE to remove the drum from WIPP. Despite the evidence offered that leaving the container in place would be the safest course of action, and without offering a persuasive safety rationale in support of his decision, NMED Secretary Ron Curry on Aug. 3 ordered the drum removed.

Workers spent several days practicing on drums filled with sand before tackling the real thing. Actual retrieval began August 10. Waste handlers slowly and deliberately removed waste containers and transferred them to two adjacent waste disposal rooms. Upon reaching the problem container, it was brought to the surface and shipped back to Idaho. Since the NMED order and throughout the retrieval operations all waste shipments to WIPP were suspended.

The lesson from this recent experience is the potential for regulatory compliance to trump risk minimization. That assessment in no way detracts from the superb job done by WIPP employees in executing this difficult assignment.

5 Recent Achievements

Recent WIPP performance highlights include:

- January 23, 2007 - WIPP receives the first RH waste shipment.
- December 3-9, 2006 - WIPP receives 36 CH shipments in one week.
- November 15, 2006 - WIPP employees complete more than 4 million hours without missing a single day of work due to on-the-job injury or illness (2- and 3-million-hour milestones were reached several times before).
- October 16, 2006 - New Mexico Environment Department issues permit for disposal of RH waste.
- March 29, 2006 - Environmental Protection Agency (EPA) issues first re-certification (required every five years).
- September 10, 2006 - WIPP receives 5 000th CH waste shipment.
- February 5-11, 2006 - WIPP receives 33 CH shipments in one week. Employees reduced processing time per shipment from 8½ hours in 1999 to 2½ hours in 2006.
- April 20, 2005 - Final (2 045th) CH waste shipment from Rocky Flats arrives at WIPP, completing the disposal of 15 000 m^3 from that site since 1999.

6 Food for Thought

6.1 Virgin Ground or Mature Mine

The search for a deep geologic repository site began in many countries with the question: “Which rocks or formations are suitable, and where are they located?” A better question may be: “Which mines (active or inactive) show some promise?”

Surprises such as WIPP experienced after raising its repository horizon are less likely in a mature mine than in a new excavation. Konrad appears to be somewhat less susceptible to surprises than WIPP or Gorleben. The widely held but hardly examined assumption that new excavations are better than old ones deserves much closer scrutiny. After all, active and former German repositories use pre-existing mines.

6.2 Conflict of Interest?

In the U.S. the same federal department (DOE) manages both waste generator sites and repositories. It pays the private WIPP contractor for, among other criteria, the quantity of waste it places into WIPP. Contractor personnel operate not only the repository and transportation system but also assist substantially with waste characterization and certification at the generator sites. Conflicts of interest appear possible, although oversight and quality assurance by independent agencies have so far ensured safety and success.

6.3 Contract Terms

The standard contract term for management and operating contracts within the DOE complex is five years. It appears questionable whether semi-decadal changes in organization and personnel favor the long-term safety of a geologic repository. WIPP has had the good fortune of being managed by the same company –although under changing names and ownership- since 1985. The continuation of this trend remains uncertain.

6.4 Cooperation across Borders

Current national boundaries are rather meaningless for projects that must function properly for 10 000 years or many multiples thereof. International information exchange and continuous cooperation are in the interest of all humanity, including our descendants. Permanent isolation must be seen and evaluated less under the perspective of constantly changing ad-hoc priorities and budgets and more "sub specie aeternitatis".

7 Conclusions

The first deep geologic repository in salt for intermediate-level, long-lived radioactive waste has proven to regulators, oversight groups, critics, and neighbors that it protects people and the environment from some deleterious byproducts of nuclear activities. In more than eight years, WIPP operations did not encounter any fatal surprises. Characterization, transportation, and disposal did not contaminate or hurt anybody. Functioning in fully operational mode, the WIPP offers a model for future efforts to safely isolate other dangerous wastes deep in the earth's crust. Lessons learned include experiences to emulate as well as some to avoid. Information exchanges with other operating repositories offer opportunities for continuous improvement. ■

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Disclaimer

This contribution was not submitted for review or approved by employees or contractors of the U.S. Government or WIPP. The author is solely responsible for its content.

1.11 Key Issues in the Dutch R&D Programme

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Abstract
A stepwise approach in finding waste management options that are feasible, suitable and technological and societal acceptable is a central policy consideration of the Dutch Government. Based on three policy documents, published respectively in 1984, 1993 and 2002, the current strategy in the Netherlands can be summarised as follows:

- *long-term interim storage in a purpose-built facility at COVRA for at least 100 years,*
- *ongoing research, preferably in international collaborative programmes,*
- *eventually retrievable deep geological disposal.*

In the Netherlands, potential host formations are salt domes and plastic Boom clay. As result of the 1993 retrievability requirement, existing generic designs were modified to allow retrieval of the waste packages after emplacement and sealing of the disposal cells. The safety assessment shows that these designs meet the relevant criteria for operational and long-term safety.

The Commission CORA (Commission on radioactive waste disposal), appointed by the Dutch government to co-ordinate the R&D programme, identified the following technical key-issues for the ongoing and future research with respect to geological disposal:

- *analysis of all potential hazard scenarios for the underground options, taking into account the probability of their occurrence;*
- *in-situ experiments in underground laboratories to study the response of salt and clay to the combined effects of pressure, temperature and radiation under in-situ conditions;*
- *development, construction and testing of the monitoring systems for the period of retrievability, for all options;*

The present paper explains why CORA felt that these issues are important. Further, an overview of the Dutch contributions to various EC research projects (such as BAMBUS, ESDRED, NF-Pro, SAPIERR, and upcoming projects) will be given in order to show the underlying structure of these individual project specific contributions in regard of the identified key issues.

1 Introduction

In the Netherlands, research on radioactive waste disposal started in the early seventies. It was recognised that rock salt formations in the Netherlands could serve as host rock for a disposal facility. The objectives of the research were laid down in a policy paper on radioactive waste that was discussed and accepted by Parliament in 1984: *"Therefore a site must be found in the Netherlands where storage of all categories of radioactive waste can take place. During the storage period further considerations can be given to final disposal, international developments can be followed and even an international facility could be used".*

In the eighties an extensive research programme was started with the goal to build an underground test facility in the Netherlands. Feasibility studies and safety studies concluded that building a disposal facility is technically feasible and that underground disposal of the waste is safe. The safety studies involved desktop studies as well as experimental studies in the Asse research facility. The safety studies involved two stages: the first study (VEOS) showed on a deterministic basis that the dose rates in the biosphere would always remain very low for various scenarios. The second study (PROSA) followed and expanded on international practice in the area of probabilistic safety analysis as applied to geological disposal. In addition, it was attempted to use all international efforts regarding the definition of so-called FEPs (Features, Events and Processes) and to develop the scenarios to be addressed probabilistically on a systematic basis.

In 1993 the Dutch Government issued a policy directive, which states that underground disposal of highly toxic waste (including high level radioactive waste) is permitted provided it is retrievable for a long period of time. Therefore, the R&D programme was adjusted to focus on three options for retrievable storage or disposal: long-term aboveground or underground in either rock-salt formations or deep clay deposits. The Commission CORA (Commission on radioactive waste disposal) was established to coordinate the R&D programme. In this programme, for each of three waste management options under consideration, the retrievability and safety aspects have been evaluated. For underground disposal, the studies resulted in the expectation that implementation of the condition for retrievability of the waste is feasible for both host rocks without compromising safety.

The CORA research programme comprised 21 projects, with contributions from 20 research institutes both in the Netherlands and abroad. The programme was carried out during the period from 1996 to 2000 at a cost of about € 3.5 million. The CORA report is available in Dutch, with an executive summary in English.

The present R&D policy intention (published in 2002) is mainly cooperation with other countries within the European Framework Programmes. The policy's long-term objective is to realise a regional facility for retrievable disposal for radioactive waste jointly operated with other countries. A regional facility not only pools technical and financial resources, but also provides a wider choice of suitable geological formations and guarantees international supervision.

2 Radioactive Waste Policy

The radioactive waste policy in the Netherlands is that all kinds and categories of radioactive waste are managed by one central organisation (COVRA[1]) and stored for at least 100 years at one site, above ground in engineered structures. This allows retrieval at all times. This step is to be followed by deep disposal for all categories (low, intermediate and high level waste) in one single repository. The long-term storage is in full operation now and necessary provisions for the next step have been taken as well. The capital growth fund to finance the final disposal exists, waste generators pay for this and there is a clear choice for the ownership of the waste: all liabilities are transferred to the waste management organisation COVRA.

The economy-of-scale of the nuclear power programs limits the options for the geological disposal of radioactive waste in the Netherlands. For countries with small nuclear power programs, such as the Netherlands, economies-of-scale will force them either to implement long-term storage and wait for decades, and/or to share a repository with others.

2.1 Interim, Long-term Storage

Implementing a small repository and operating it for very long times is very costly. Moreover, the small volume of waste can easily be kept under control in above ground structures. This "interim" storage provides time to let the volume of waste accumulate and to let the amount of money, needed for disposal, grow in a capital growth fund.

With only 450 MWe installed nuclear capacity in the Netherlands, there is no need for disposal in the short term. The volume of all categories of radioactive waste generated over 60 years is only a few thousand m^3. Such small volume will make the disposal costs per m^3 very high. For waste generators other then the nuclear power plants these costs per m^3 are too high.

[1]COVRA (Centrale Organisatie Voor Radioactief Afval) is the central organisation in the Netherlands for collecting, processing and storing the radioactive wastes.

Because of these reasons, building a repository is very unattractive as long as the accumulated amount of waste is small. Beyond that, the infrastructure for interim storage is needed anyway. The additional costs of prolonged interim storage are relatively small compared to the economical ineffectiveness of building a repository for small amounts of waste.

Although the additional costs of prolonged interim storage are relatively small, the challenge is to engineer buildings that last for a hundred years or more, comply with present and (as far as possible) future environmental legislation, and are acceptable for the public during this long time. Other uncertainties are caused by policy chances every few years, even lack of a clear policy, or lack of political decisions. This easily leads to additional costs, e.g. the construction of a facility meant for short term waste storage that later on has to be refurbished several times as the interim storage period appears longer as expected.

Apart from the fact that building a repository for is only economically attractive for a sufficient large amount of waste, delaying the disposal is also expected to result in capital growth of the funds available for disposal. In 100 years time, growth by one order of magnitude can be obtained with a real interest rate of 2.5%. On the other hand, there is also a possibility of severe losses.

2.2 Shared Regional Repositories

Although there are sound environmental, safety and security reasons for disposing the radioactive waste in national geological repositories as soon as possible, there are also good reasons why this will not take place in all countries. Over the past years it has increasingly been acknowledged that regional or multinational repositories can potentially enhance safety, security and the economics of radioactive waste disposal. The need for supranational surveillance also points to shared solutions and there are past and current examples of countries being prepared to accept radioactive waste from others if a better environmental solution is thus achieved.

In the EU, all 27 member states generate radioactive waste. Of course there are large differences in type and quantities between the member states. But even a country with only waste from radium containing lightning rods needs a long-term solution. The 1600 year half-life of radium does not fit in a solution with a time span of a few hundred years. When the 27 European countries would each create a repository adequate to their needs, this will be a tremendous waste of resources. Regarding the total volume of radioactive waste in the European Union, one repository would be sufficient. Because of the present public attitude as well as legislative obstacles, the solution of a single EUrepository will be very difficult to obtain. A more realistic goal is the creation of a few regional repositories, parallel to national programmes.

Although both the European Parliament and the Commission support the concept of shared regional repositories in Europe, (national) political and societal constraints have hampered the realisation of such facilities up to now. Development of a shared repository implies that sharing does not exclude being the host. All possibilities should be kept open and the pros and cons of a shared facility will have to be studied first. The Netherlands have been involved in a first study of this kind: the SAPIERR[2] project, which can be seen as an important first step towards an outline of a European facility. Another important initiative is the creation of ARIUS[3], the Association for Regional and International Underground Storage. The Netherlands is one of the ARIUS partners.

3 Research Policy

Long-term waste management strategy in the Netherlands is based on the main components of the Dutch policy regarding radioactive waste management. This policy, recorded in Governmental documents in respectively 1984 [2], 1993 [3] and in 2002 [4], will be briefly outlined below in order to present a total overview [1].

[2] V. Štefula. 2006. SAPIERR Support Action: Pilot Initiative for European Regional Repositories FINAL REPORT, Deliverable D-7.

[3] www.arius-world.org

Beside the issues of minimisation and reduction of (the amount of) radioactive wastes the policy in the Netherlands has two main components:

- Interim storage of all kinds of radioactive wastes at a centralised site of COVRA for at least 100 years;
- Ongoing research on retrievable[4] disposal in deep geological formations.

Retrievability, the second main component in radioactive waste management, is an important prerequisite, fully in line with the (fourth) National Environmental Policy Plan (NMP) [5] in which the Government established a strategy for sustainable development. This policy directive, which stated that eventually underground disposal is permissible if the waste remains retrievable over the long-term, is a consequence of a NMP initiative, implemented in 1993 [3].

Long-term retrievable disposal of radioactive waste is considered to be feasible in both available host environments in the Netherlands: salt formations and clay layers [6]. The Commission CORA[5], that was appointed by the Dutch government to co-ordinate the R&D programme, recommended further investigations on societal and ethical issues regarding waste management, in addition to the ongoing technological research. The responsible ministry in the Netherlands (VROM[6]) reflected to these recommendations with a policy document that describes the governments perspective on radioactive waste management [4]. This policy envisages the expansion of knowledge in an international framework on all aspects of (deep geological) disposal, including technological, societal and ethical issues, during the interim storage phase.

Keeping all options open is also part of this strategy, which avoid any exclusion of potential waste management opportunities as result of (irreversible) decisions taken now.

4 Key Issues

The Commission CORA identified the following technical key-issues for the ongoing and future research with respect to geological disposal:

- analysis of all potential hazard scenarios for the underground options, taking into account the probability of their occurrence;
- in-situ experiments in underground laboratories to study the response of salt and clay to the combined effects of pressure, temperature and radiation under in-situ conditions;
- development, construction and testing of the monitoring systems for the period of retrievability, for all options;

To be able to point out the key issues, first the 'retrievability issue' is explained in Section 4.1 and the Dutch disposal concept is introduced in Section 4.2. In Section 4.3 'Safety and design' it is explained how the 'in situ experiments in URL' are addressed, Section 4.4 'Performance Assessment' deals with the 'analysis of potential hazard scenarios', and Section 4.5 'Monitoring' deals with the development of monitoring systems.

The R&D policy's long-term objective is to realise a regional facility for retrievable disposal for radioactive waste jointly operated with other countries. Section 4.6 deals with Multilateral approaches'. CORA also recommended further investigations on societal and ethical issues regarding waste management: Section 4.7 deals with Societal aspects.

[4] Retrievability means the deposition of radioactive waste in a way that it is reversible forthe long-term by proven technology without re-mining.

[5] Commission on Radioactive Waste Disposal; in Dutch: Commissie Opberging Radioactief Afval.

[6] Volkshuisvesting, Ruimtelijke Ordening en Milieubeheer (Housing, Spatial Planning and the Environment)

4.1 Retrievability

Retrievability of the waste allows future generations to make their own choices, but implies that the facility is being kept accessible for a long time for inspection and monitoring. In the Netherlands a research programme on retrievable disposal of radioactive waste was initiated in line with the 1993 policy directive of the Dutch Government. This directive states that underground disposal of highly toxic waste (including radioactive waste) is permitted provided it is retrievable for a long period of time.

The issue of retrievability is being considered important in an increasing number of countries. It seems that a staged decision-making process in which decisions are not irrevocable, would reassure several stakeholder groups. Staged decision-making requires control and surveillance of the disposal process and the option to be able to revert to previous stages in the disposal process.

With respect to the safety provided by the facility in relation to retrievability two different issues should be dealt with:

- the design changes (compared to a design that offers no options to retrieve the waste) that are needed to allow retrieval of the waste package should not affect the isolation performance of the facility

- the option of retrieval of the waste may result in extended operation times, i.e. closure of the facility may be postponed even when all waste has been emplaced.

There is a consensus opinion that the first issue can be dealt with by proper engineering and designing [8]. The second issue prompted the Dutch scientific advisory committee CORA to the conclusion that "a disposal facility should be fail-safe, in case of the possible loss of human control over the facility."

In general 'safety' addresses (apart from the risks that have to be very small) also the level of oversight over the system (e.g. monitoring to assure that the system is behaving as expected) and the level of control over the system (e.g. possibilities to intervene in the disposal process if the system shows undesired behaviour). Introducing retrievability has a significant positive effect on the level of control over the disposal system, if the facility can be designed to be fail-safe.

4.2 Dutch Reference Design for a Geological Disposal Facility

The Dutch reference disposal concept contains short horizontal disposal cells drilled into the side walls of a gallery, each accommodating one HLW container. The annular space around the container is filled with suitable backfill material and each cell is sealed off seperately. Retrieval requires a special machine to remove the seal and backfill and to withdraw the waste container from the disposal cell. Figure 1 shows an artists impression of the reference design.

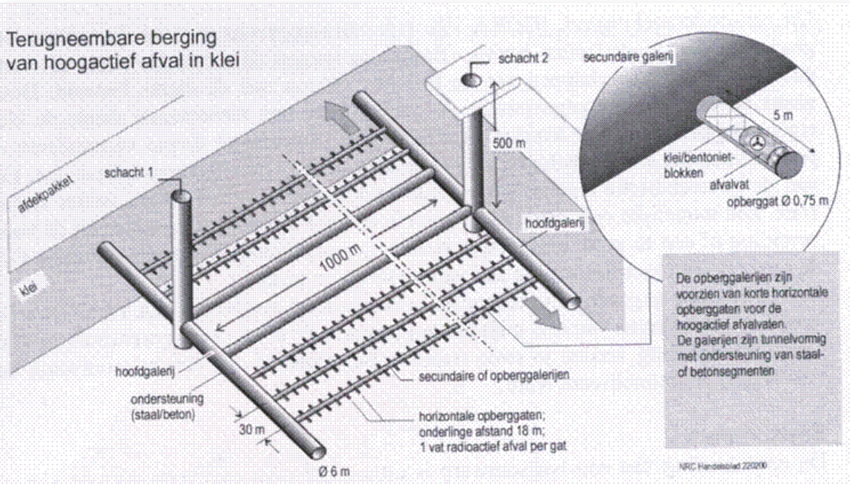
Two practicable host formations for deep geological disposal of radioactive waste exist in the Netherlands, namely rock salt formations mainly located in the north-east part of the country and (Boom) clay layers of which the minimum required depth can be found in the south near the Belgium border. Engineering studies [6] showed that that:

- retrievable disposal of HLW in a rock-salt formation is technically quite feasible.

- retrievable disposal of HLW in a clay deposit seems feasible.

For the Dutch reference design, the construction, operation and closure of a retrievable nderground disposal facility in rock salt will cost about 250 M€, and in clay deposits about 600 M€. Keeping an underground disposal facility open to allow retrieval of the waste would entail an annual expenditure of around 1.8 M€ for maintenance, management and inspection [6].

It should be noted that the present Dutch reference designs do not deal with long-lived low level waste and intermediate level waste. Attention for this waste is needed, especially with respect to practical issues such as gas-development, retrievability of large volumes of waste, and long-term chemical stabilisation of the waste. There are indications that the (far future) radiological impact from a repository containing HLW, ILW and LLW is dominated by the LLW and ILW, because of the superior conditioning of the HLW.



◄ Fig. 1: Generic design for a geological disposal facility (Source: NRC Handelsblad)

4.3 Safety and Design

From a safety perspective it is required that the repository is fail-safe during all steps of the waste disposal operation. I.e. even in case of abandonment of the repository before proper closure, the waste must not become a threat to our environment. Given that a repository will be in operation for about 50 to 100 years, events of concern are:

- Economic distortion
- War, national disaster
- Mining disaster

These events could lead to abandonment of the repository without proper closure. This could lead to the following chain of events:

- Flooding of unsealed galleries
- Water is gradually pressed through the seals of the disposal cells
- Soluble parts of the waste dissolve in the water
- Advective flow and diffusion through the remains of the underground infrastructure
- Radioactive material reaches aquifer and biosphere
- Exposure of humans to radioactive material

This chain of events describes the “abandonment scenario”.

In this scenario the only barriers that are in place are the waste matrix, the container, the backfill and the seals of the disposal cells. For the Dutch reference disposal concepts, the long-term behaviour of the seals of the disposal cells dominates this scenario.

Therefore, a key issue is the long-term performance of the seals of the disposal cells in the abandonment scenario. This resulted in various contributions to EC projects, international cooperation and some national studies, see Table 1.

4.4 Performance Assessment

Table 1: Dutch contribution to projects in the scope of "safety and design" and focussing on seal behaviour

Project - rock salt	Description of the Dutch contribution
BAMBUS I BAMBUS II	Behaviour of crushed salt - benchmark of FE-elements "material codes"
NF Pro	Process level understanding of low porosity (< 5%) crushed salt behaviour
Theresa	Benchmark of FE "material codes" EDZ behaviour
Dutch R&D programme	Radiation damage of halite, alternative sealing materials, radon tracing
ESDRED	Review of the reference design (DBEtec- Germany) with respect to retrievability
Project - clay	**Description of the Dutch contribution**
TIMODAZ	Constitutive modelling of the clay rock, in a continuum framework to incorporate thermal effects, long-term creep and the development, evolution, and eventual sealing of micro cracks.
PRACLAY	Bilateral cooperation between NRG and SCK.CEN in the framework of EURIDICE and the PRACLAY project
ESDRED	Review of the reference design (ANDRA - France) with respect to retrievability
Dutch R&D	Retrievable disposal in Boom clay at 500 m depth

Long-term safety studies give estimates of potential dose rates and the associated risks as a consequence of radioactive material that might escape from the repository. For most repository designs there is a non-negligible probability that a small amount of radioactive material will be able to escape from the waste container, move through the remains of the underground facility, and enter a deep aquifer. Eventually some of this material reaches the surface groundwater and can enter the food chain. If the radioactive material is ingested by humans, the radiation from the radioactive material will cause potential health damage. This is quantified using the effective dose rate. For the Dutch disposal concepts, the long-term safety studies show that this effective dose rate is orders of magnitude smaller than the normal background dose rate resulting from natural background radiation. The dose rates and risks are within legal limits and are considered to be safe.

The recent Dutch studies focus on "abandonment scenarios". Although "retrievability" was the initial reason for introducing these scenarios, this type of scenarios is equally relevant to all facilities that will be operational for several decades. The treatment of this type of scenarios required extensions to existing PA-tools/techniques and the systematic FEP treatment procedure. An overview of PA projects is given in Table 2.

4.5 Monitoring

Surveillance (and thus monitoring) is one of the three main components of Government's radioactive waste policy in the Netherlands. For the steering commission on radioactive waste disposal CORA surveillance and supervision is a "societal" rather than strictly "technical" requirement in gaining public acceptance for waste disposal. Hence, monitoring will form a key element

Table 2: PA projects with a focus on abandonment scenarios

Project	Description of the Dutch contribution
PAMINA	Credibility of the sequence of events in the abandonment scenario; Sensitivity and uncertainties resulting from plug behaviour
SPA	Scoping analyses of the abandonment scenario for direct disposal of spent fuel
Dutch R&D	Direct disposal of spent fuel from Dutch research reactors, scoping PA for the Dutch reference designs

in the waste management strategy. In the research programme monitoring issues must play a very important role, because no (final) strategy has been developed yet. Furthermore, it is essential to enhance the durability and reliability of the instruments according to specific disposal situations and monitor time frames. It is expected that a long-term in-situ monitoring approach will be chosen when taking into account the Dutch disposal requirements, especially the need to ensure the possibility of waste retrieval over a long time period [1].

In [7] the following is stated:

“Monitoring needs to contribute to the decision-making process by providing:

- site-specific information on the evolution of the surface and underground environment during all phases of repository implementation, as well as on the short- and long-term evolution of the engineered barrier system (EBS);
- overall confirmation of a regular and safe operation of the disposal facility (including the requirements of nuclear material safeguards);”

The Dutch scientific advisory committee CORA concluded:

- that allowing surveillance and supervision of retrievable disposal may decrease he societal resistance against disposal.
- that future technical research should focus amongst others on the development, construction and testing of monitoring-systems to be used during the retrievabilityperiod, for surface and for underground facilities.

On the other hand it is sometimes argued that the operation of underground laboratories, demonstrations and prototypes give sufficient evidence that the repository will behave as expected. Monitoring would then only be required for operational safety (of the workers) and fulfilling legal obligations.

Table 3: Contribution to Monitoring projects

Project	Description of the Dutch contribution
TN Monitoring	Role of monitoring in a phased approach to geological disposal – Dutch position
Modern (FP7 proposal)	Wireless transmittance of signals from a geological repository to the surface
Dutch R&D	Monitoring and safety aspects of retrievability

Table 3 shows that so far only minor attention has been given to monitoring. However, we expect that monitoring will become of increasing importance as several waste management programmes are moving from concept development and research towards more detailed site investigation and implementation stages (during which monitoring programmes must be defined).

4.6 Multilateral Approaches

Two Specific Support Actions: SAPIERR-1 (Support Action: Pilot Acton for European Regional Repositories) and CATT (Co-operation And Technology Transfer on long-term radioactive waste management for Member States with small nuclear programmes) and one Co-ordination Action SAPIERR-2 (Strategic Action Plan for Implementation of European Regional Repositories) have been undertaken within the 6th EC Framework Programme to explore how member states with relatively small amounts of nuclear waste can implement long-term solutions through collaboration.

In the period 2003 to 2005, the Netherlands participated in the EC funded SAPIERR I, a project devoted to pilot studies on the feasibility of shared regional storage facilities and geological repositories, for use by European countries. The studies showed that shared regional repositories are feasible. The first step would be to establish a structured framework for the work on regional repositories. Moreover, if shared regional repositories are to be implemented, even some decades ahead, efforts must already be increased now.

This is the goal of SAPIERR II (2006-2008): to develop possible practical implementation strategies and organisational structures. These will enable a formalised, structured European Development Organisation (EDO) to be established in 2008 or afterwards for working on shared EU radioactive waste storage and disposal activities. The EDO can work in parallel with national waste programmes. Participating EU Member States will be able to use the structures developed as, when and if needed for their individual national policies and programmes.

The Dutch WMO COVRA will co-ordinate the follow-up project SAPIERR II (Strategic Action Plan for Implementation of European Regional Repositories), which builds on the pilot studies of SAPIERR I to develop options for organisational frameworks and project plans that could lead to the establishment of an European Development Organisation (EDO) for European regional repositories. To clarify issues related to the structure and future programme of the potential EDO, presently a series of specific studies are being carried out on organisational structures, legal liabilities, economics, safety and security as well as public and political acceptability. The options distilled from these studies will be presented and discussed at a workshop for interested countries and organisations, in order to identify potential end-users and to achieve consensus on a preferred way forward: the first steps of implementation or a further programme of preparatory work.

The Netherlands also participated in the EC CATT workshops. The overall objective of CATT is to investigate the feasibility of Member States with small nuclear programmes implementing long-term radioactive waste management solutions within their national borders, through collaboration for technology transfer with those Member States that have advanced disposal concepts.

Although CATT assumes that each nuclear member state has its own repository, whereas SAPIERR assumes that there will be both, national and also shared repositories in Europe, the two Actions are complementary. Together with the IE, DG-JRC, DPRAO, NDA, ARIUS, IAEA, COVRA is organizing a combined CATT-SAPIERR workshop to further explore these multilateral approaches, in particular future collaborations models to support Member States with small amounts of radioactive waste. Irrespective of whether the repositories are shared or national, it is a very likely that the implementation will take advantage of the concepts that are already being developed for different geological formations.

4.7 Societal Aspects

CORA recommended further investigations on societal and ethical issues regarding waste management, in addition to the ongoing technological research. In order to develop a politically and publicly acceptable solution for radioactive waste a research programme (1) on public attitudes to geological disposal and to national/shared repositories, (2) on public participation processes, and (3) on the (societal) costs and benefits of geological disposal for host communities, and (4) on political governance has to be developed. Although societal aspect can be decisive for an eventual implementation of a national or an international repository, until now limited work was carried out on the subject of societal acceptability.

At present the Netherlands participate in two EC projects that investigate societal aspects of geological disposal: governance and public attitudes.

- The OBRA workshops on governance processes: As the development of a suitable and acceptable repository is a (inter)national challenge with a strong local dimension, this has to be reflected in an appropriate cross-linking of the national and the regional governance processes. The EC OBRA project investigates the feasibility of creating an "Observatory" in order to address the concerns and information needs of different stakeholders on radioactive waste management. As such, the OBRA project structures the governance processes.

- Through SAPIERR II, the Netherlands will participate in a review of public and political attitudes in Europe towards the concept of shared regional repositories. This review will be based on input from literature studies by representatives of organisations participating in SAPIERR II, complemented by a review by project team members of the situation in other European countries. An overview will also be prepared on how national programmes now and in the past have tried (or not) to encourage public participation in nuclear decisions.

5 Conclusions

- Countries with a small nuclear program, such as the Netherlands, should not necessarily develop a national geological disposal facility. If shared regional repositories are to be implemented, even some decades ahead, efforts must already be increased now. Before increasing efforts on regional repositories, a structured framework should be established.

- In the Netherlands, waste must be disposed in a way that it is possible to retrieve the waste. Presently, most countries require something similar.

- Retrievability can be implemented without major changes to existing designs; the additional costs are minor.

- Given the long lasting operational period of a geological disposal facility, special attention is required for the sealing of disposal cells and disposal galleries.

- Safety studies should not only address the long-term safety consequences of post-closure events, but also the long-term safety consequences of potential preclosure events, such as abandonment.

- Remote (wireless) monitoring techniques (pre- en post-closure) are necessary to retain trust in the long-term functioning of the facility.

- Attention for long-lived low level waste and intermediate level waste is needed, especially with respect to practical issues such as gas-development, retrievability of large volumes of waste, and long-term chemical stabilisation of the waste.

- Finally, in order to developed a politically and publicly acceptable solution for radioactive waste also a research programme on the societal aspect of geological disposal has to be developed, preferably in international collaborative programmes.

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1.12 Searching for a suitable Site in Switzerland – The Deep Geological Repository Sectoral Plan

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Abstract
In Switzerland, the procedure for selecting potential sites for deep geological repositories is defined in the Deep Geological Repository Sectoral Plan. This sectoral plan consists of a general concept and an implementation part. The general concept defines the federal government's objectives and the rules for the selection procedure. In addition to safety and security criteria, it deals with other requirements relating to the selection of suitable regions or sites, in particular spatial planning and socio-economic aspects, and organises the co-operation with the regions concerned.

The implementation of the actual selection procedure starts once the Federal Council has approved the general concept. This implementation procedure takes place over 7 to 10 years. It comprises three stages and shall result in the identification of sites for deep geological repositories for low and intermediate level as well as high level radioactive waste.

With public acceptance being crucial, the Deep Geological Repository Sectoral Plan sets forward a significant involvement of the cantons and regions concerned, as well as of the neighbouring countries. This includes the creation of regional participation bodies in which the municipalities concerned, as well as their inhabitants, can participate. National borders shall not play a role in the determination of involvement. After the completion of each stage, formal consultations take place. Once the three step procedure has been completed, the option of a national referendum will remain available, which means that the Swiss electorate has the final say.

Switzerland's first nuclear power plant, Beznau I, was put into operation in 1969. The National Co-operative for the Storage of Radioactive Waste (Nagra) was established three years later, in 1972. It was entrusted with the mandate of finding a safe long-term solution for the disposal of all radioactive waste produced in Switzerland – a mandate that still has to be fulfilled. At present, radioactive waste is being stored safely in special containers in well-secured halls at an interim storage site in Würenlingen, and at the nuclear power plants themselves. However, this radioactive material needs to be safely stored for up to a million years, and for this purpose, storage above ground is not a suitable solution.

Today, fully developed concepts exist for the permanent storage of radioactive waste in geological formations, but so far it has not been possible to find a site for a deep geological repository in Switzerland.

With the new legislation that entered into effect in February 2005 (Nuclear Energy Act and Nuclear Energy Ordinance), the federal government adopted a new approach: the search for a suitable site is now to be carried out within the scope of a sectoral planning procedure. Sectoral planning is a tried and tested instrument that the Swiss federal government uses for planning and co-ordinating major national infrastructure projects. The objective of the Deep Geological Repository Sectoral Plan is to ensure that, as major projects of national importance, deep geological repositories can be decided upon and constructed on the basis of an independent, transparent and fair procedure. This process is managed and co-ordinated by the Swiss Federal Office of Energy (SFOE).

1 Legal Provisions governing the Management of Radioactive Waste

In addition to specifying the implementation of the Deep Geological Repository Sectoral Plan, the Swiss Federal Nuclear Energy Act and its accompanying Ordinance stipulate a variety of other fundamental provisions governing the management of radioactive waste:

- All radioactive waste produced in Switzerland must be disposed of within Swiss sovereign territory.

- Producers of radioactive waste are responsible for ensuring its safe long-term disposal ("polluter pays" principle).

- Radioactive waste must be disposed of in such a manner as to ensure the permanent protection of human beings and the environment. Experts agree that this can be accomplished by storing waste material in stable geological formations. The Nuclear Energy Act therefore stipulates that deep geological repositories are to be used for the disposal of all radioactive waste. Once a deep geological repository has been put into operation, it must be monitored for several decades, and the waste material must remain recoverable.

The new legislation no longer recognises a veto right on the part of cantonal authorities. However, it provides for a national (optional) referendum which may be initiated against the award of a general licence for a deep geological repository.

2 Transparent and fair Selection Procedure

The Swiss government has developed sectoral plans for numerous major infrastructure projects of national importance, for example in the areas of civil aviation, transport, defence and high voltage transmission lines. Sectoral plans are developed on the basis of close co-operation between the federal government, cantonal governments, competent authorities, other organisations and the authorities of Switzerland's neighbouring countries. Furthermore, the involvement of the general population in the sectoral planning process is also called for in the Swiss Federal Spatial Planning Act.

The Deep Geological Repository Sectoral Plan represents a new concept in a variety of ways: for example, never before has a sectoral plan been conceived for infrastructure of such a lasting nature, and it is probably safe to say that never before have such widely diverging political values clashed with respect to a sectoral plan. For many, the disposal of radioactive waste is closely intertwined with the issue of a continued utilisation of nuclear energy. The demands placed on the selection procedure are thus extremely high, and it is essential to ensure that clear guiding principles are defined for the Deep Geological Repository Sectoral Plan:

- The permanent protection of human beings and the environment is of the highest priority; it takes precedence over area planning, economic and social aspects.

- The search for a suitable site is based on the quantities of waste that are produced from the operation of the existing nuclear power plants in Switzerland. After these facilities have been decommissioned and dismantled, there will be a total volume of around 110,000 cubic metres of radioactive waste (including containers) to be disposed of. The definitive quantity is to be specified when an application for a general licence for a specific site is submitted. In this regard, the Deep Geological Repository Sectoral Plan does not establish a prejudice for or against future nuclear power plants.

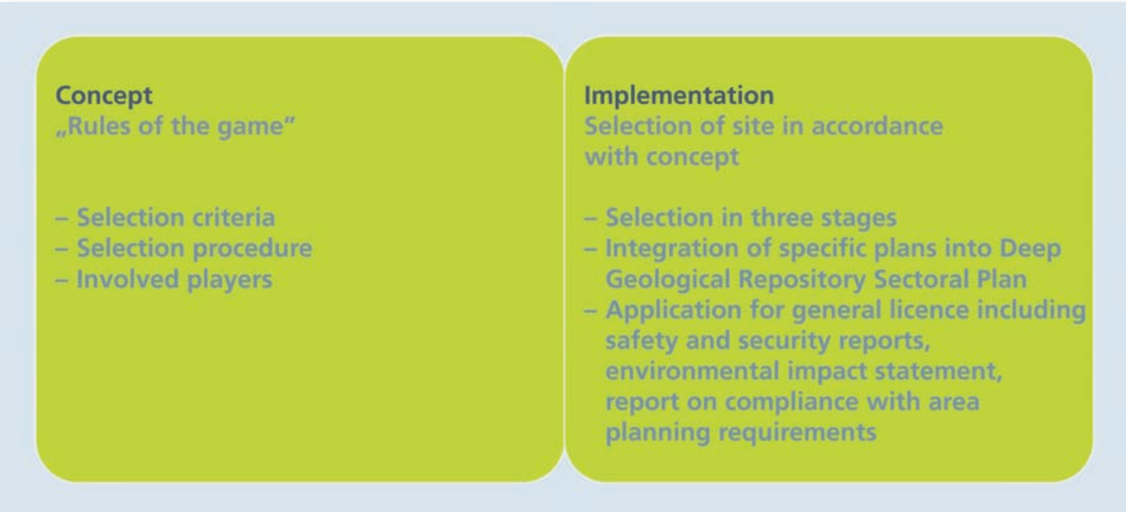
- A transparent and fair procedure is an essential prerequisite for achieving the principal objective of the plan, i.e. finding acceptable sites for deep geological repositories. This will only be possible if both the procedure and the decisions that are subsequently taken meet with the necessary degree of acceptance.

3 Involvement of all Interest Groups as a Crucial Factor

Sites for facilities like deep geological repositories, which are of major importance and of a lasting nature, cannot simply be decreed "from above". Rather, they have to be selected on the basis of a public debate. In Switzerland, citizens can voice their opinions either through instruments of direct democracy (votes, referendums, etc.) or via procedural processes (hearings, objections, etc.). But these instruments are only effective at the end of a given procedure. Findings obtained over the past 30 years at the international level have shown that a site selection procedure can only be successful if it is perceived by all parties and interest groups as a fair and transparent process in which everyone concerned can play a part. It is therefore essential to ensure that the applicable "rules of the game" governing the selection of suitable sites are defined in a clear and transparent manner. Social concerns have to be taken seriously, and it is important to secure an ongoing and open dialogue. Otherwise it will not be possible to gain the necessary degree of trust in the actors involved and confidence in the various processes. The aspect of consultation is therefore a highly important component of the Deep Geological Repository Sectoral Plan, and the possibilities for involvement extend beyond the legally prescribed minimum. While this does not guarantee success, it nonetheless creates a sound basis for increasing the chances of acceptance of proposed sites for deep geological repositories.

4 Concept and Implementation

The Deep Geological Repository Sectoral Plan is divided into two sections: a conceptual component, in which the rules governing the selection procedure are defined, and an implementation component, in which the individual steps of the selection process lead to detailed plans and specific locations are named as suitable sites for deep geological repositories in Switzerland.



▲ Fig. 1: The two components of the Deep Geological Repository Sectoral Plan: concept and implementation

5 Two Repositories

Switzerland's geology has been the focus of intensive research for more than 200 years. The findings have been documented in numerous geological maps and scientific reports, and the resulting know-how has also been intensified through geotechnical and seismic studies conducted for the purpose of exploring for raw materials. During the past twenty-five years, Nagra has been contributing towards a better understanding of Switzerland's geology by carrying out drillings and other activities in two Swiss field laboratories. The Deep Geological Repository Sectoral Plan documents the studies carried out to date as well as the geological findings. They form a broad basis for the search for suitable sites.

Switzerland's Nuclear Waste Management Concept calls for two repositories: one for low and intermediate level waste and another for high level waste. Alpha toxic waste may be stored in either of the sites. In the event that one particular site should be identified as suitable for all three waste categories, the selection procedure can result in the proposal for a single site for the repositories.

6 Selection Criteria: Safety is the highest Priority

The concept of the Deep Geological Repository Sectoral Plan specifies the criteria that potential sites have to meet. Here, criteria relating to safety head the list. A proposed geological formation must guarantee the safe storage of radioactive matter over the required extended time frame.

The safety criteria are initially defined in qualitative terms. This includes aspects such as the extent and depth of the rock formation, its degree of long-term stability and its hydraulic barrier effect. The safety of a given site is based on the interaction between various natural local factors, but it also depends on the nature of the waste matter to be stored and on other technical aspects (e.g. storage containers). As the procedure progresses, the requirements relating to safety become more specific in accordance with the relevant guidelines of the Swiss Federal Nuclear Safety Inspectorate ("Protection Objectives for the Disposal of Radioactive Waste" – HSK-R-21).

Only those sites that meet the specified safety requirements can be potential candidate sites. In addition, area planning and socio-economic aspects of these potential candidate sites are evaluated and the corresponding impacts of a deep geological repository on the region concerned are closely examined.

7 Site Selection Procedure: Three Stages

The site selection procedure is carried out in three stages:

Stage 1: Selection of geologically suitable Regions for Low-/Intermediate-Level Waste and High-Level Waste

Nagra, representing the producers of radioactive waste and responsible for waste disposal, proposes geologically suitable regions on the basis of the relevant safety criteria, and substantiates its choice in a report for the attention of the Swiss Federal Office of Energy (SFOE).

The federal government then notifies the authorities of the cantons and municipalities concerned. A cantonal committee is then formed, in which the involved cantons and neighbouring cantons are represented. Neighbouring countries that are affected are also entitled to sit on this committee. Together with the committee, the SFOE then specifies the suitable regions and initiates regional consulting bodies in these regions.

The next step, which is initiated by the relevant federal authorities in co-operation with the cantons concerned, is to take stock of the current situation and defining the methodology for determining the most important area planning aspects and their subsequent assessment in stage 2.

Once this criterion is met and the safety assessment completed, the geologically suitable regions are adopted. This is done in the form of preliminary proposals – following a three-month consultation period (in accordance with the Swiss Federal Spatial Planning Act) and the subsequent approval by the Federal Council. The suitable regions identified are then adopted into the Deep Geological Repository Sectoral Plan.

Stage 2: Selection of at Least Two Sites each for Low-/Intermediate-Level Waste and High-Level Waste

In stage 2, at least two sites each for low / intermediate level waste and high level waste have to be selected. The safety assessment continues to be of the highest priority. Nagra will carry out provisional safety analyses of the geologically suitable regions named in the Plan.

An evaluation of land use and socio-economic impacts is carried out in collaboration with the cantons concerned and regional consultation bodies. Here, a land use register is prepared that includes existing and future land use, and socio-economic background studies are carried out.

At this stage, the regional consulting bodies are given the opportunity to formulate sustainable regional development strategies, or to further develop existing ones.

Nagra designs the layout of the underground sections of the repository and, together with the regional consulting bodies, designates potential sites within the planning perimeter of geologically suitable regions.

Based on the evaluation of these potential sites, Nagra proposes at least two potential sites each for low / intermediate level waste and high level waste. Some of these sites might be regarded as suitable for all three waste categories.

The sites are then examined by the federal authorities and, if approved, are adopted in the form of intermediate proposals following a three-month consultation period (in accordance with the Swiss Federal Spatial Planning Act) and the subsequent approval by the Federal Council.

Stage 3: Selection of Sites and Licensing Procedures for Low-/Intermediate-Level Waste and High-Level Waste

In the final stage, the remaining sites are subject to a more detailed examination and, where necessary, the geological data is updated by carrying out seismic tests and drillings. This allows the carrying out of in-depth safety assessments on a comparative

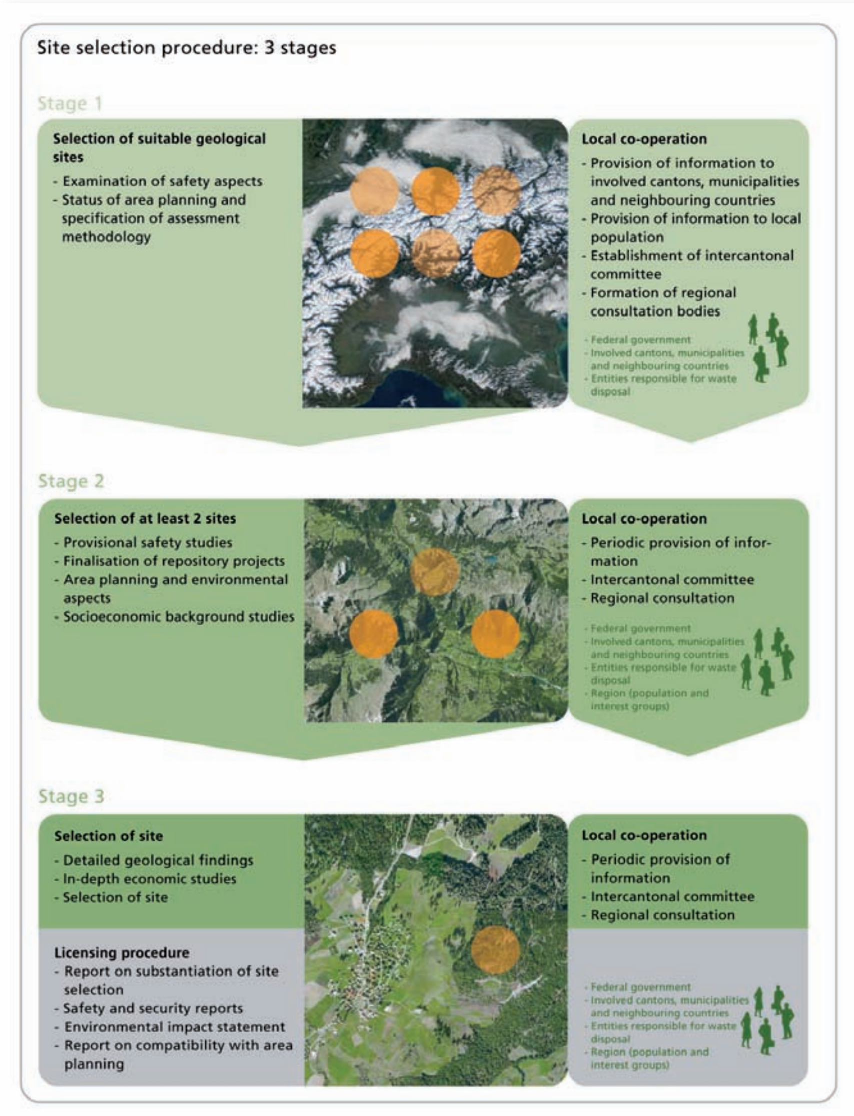


Fig. 2: Planning procedure in three stages. Local co-operation plays a major role.

basis in preparation for the licensing procedure. The socio-economic aspects are subjected to closer scrutiny and the regional consultation bodies propose regional development projects. The socio-economic studies also form the bases for the monitoring of economic and ecological impacts of the geological depository, as well as for any compensation measures. If remuneration is foreseen, it is negotiated – and made known – in stage 3. Nagra then proceeds to propose the location at which the deep geological repository is to be constructed (one site each for low / intermediate level waste and high level waste, or one site for all waste categories).

The work and the activities in stage 3 also pave the way for the licensing procedure and the first stage of the environmental impact assessment. Stage 3 is completed after the site has been specified in the Deep Geological Repository Sectoral Plan and the Federal Council has granted a general licence. The next steps concern the approval of the general licence by Parliament and the organisation of a referendum, the latter if an optional referendum should be initiated against the award of the licence.

8 Timetable and Outlook

The goal is to put deep geological repositories into operation within 25 to 35 years. High level radioactive waste takes around 40 years to cool down to a temperature low enough to allow for permanent storage. A large proportion of low and intermediate level waste will only arise after the existing nuclear power plants have been decommissioned and dismantled.

The Federal Council will be taking its decision on the concept of the Deep Geological Repository Sectoral Plan at the beginning of 2008. This move will create the prerequisites for a broad-based, goal-oriented selection procedure. Nagra will most likely submit its proposals for geologically suitable regions in the course of 2008. The population in the regions concerned will thus for the first time be faced with the possibility of a nearby deep geological repository. The submission of the proposals will then lead over to the decisive phase of co-operation and participation described above. ■

1.13 Geologic Disposal of High-Level Waste in the United Kingdom

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The Nuclear Decommissioning Authority (NDA) is a non-departmental public body, which began operation in April 2005 with a remit to secure the decommissioning and clean-up of the UK's civil public sector nuclear sites. This remit was widened when the Government announced on 25 October 2006 that, following recommendations from the Committee on Radioactive Waste Management (CoRWM), higher activity wastes will be managed in the long-term through geological disposal. Government also announced the NDA was to be given the responsibility for planning and implementing geological disposal. A new directorate within the NDA was created, the Radioactive Waste Management Directorate (RWMD), to address this additional remit. The UK's former waste management organisation Nirex was legally wound up and its staff integrated into RWMD.

The NDA's mission is *"to deliver safe, sustainable and publicly acceptable solutions to the challenge of nuclear clean-up and waste management. This means never compromising on safety or security, taking full account of our social and environmental responsibilities, always seeking value for money for the tax payer and actively engaging with stakeholders"*.

The NDA has selected a provisional repository concept for the UK's HLW and SF that satisfies a number of broad criteria. The concept will become more fully defined as it is progressively tested and refined, and as it is adapted to a specific site, once one is selected. In order to begin this process, an outline design for the concept has been developed in sufficient detail to enable an initial assessment to be made of factors such as long-term safety and cost.

The concept selected by the NDA for further investigation and development is based on the KBS-3V concept developed by the Swedish Nuclear Fuel and Waste Management Company (SKB) for the deep geological disposal of spent fuel in Sweden. The dimensions of the disposal canisters and deposition holes have been modified in length and diameter to accommodate UK HLW and SF.

Currently, about 4.500 stainless steel containers containing approximately 675 m³ vitrified High-Level Waste (vHLW) are stored at the BNFL Vitrified Products Store (VPS) at Sellafield. The majority of this material has its origin from reprocessing of UK nuclear fuel (a part of the vHLW stored at VPS originates from reprocessing contracts and belongs to overseas customers). There are some radioactive materials that are not currently classified as waste but that may, if decided so, need to be managed through geological disposal. About 1,200 tons of PWR and 7,000 tons of AGR spent fuel (SF) exist in the UK. These fuels are currently not destined for reprocessing. They could be instead packaged and disposed of in a geological disposal facility, as is planned in several other countries.

The UK government and the devolved administrations are exploring how an approach based on voluntarism could be made to work in the UK context, and how a staged partnership approach to the development of a geological disposal facility could be applied. In the absence of a specific site, NDA-RWMD undertake "generic" research into the mechanisms and phenomena which ultimately will control the safety and associated cost margins of a disposal concept for UK HLW and SF. In order to approach this goal, both a strategy of exploring various options, as well as close collaboration with other national geological disposal programmes are employed. ■

Reference

[1] Managing Radioactive Waste Safely – A framework for implementing geological disposal, Department for Environment, Food, and Rural Affairs (DEFRA), London/United Kingdom, 25 June 2007.

1.14 Multinational Cooperation in Geological Disposal: Sharing of Concepts, Results, Research Projects and Facilities

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Abstract

Since the earliest days of research into geological disposal of radioactive wastes, multinational cooperation has played a prominent role. The early concepts developed by some nations have been adopted and/or adapted by many programmes that started later. The repository design work and the analyses of the safety of these facilities depend on a vast range of data that was being produced in parallel in laboratories across the world. In practice, the most intensive collaboration between nations in the area of disposal related R&D may be the work carried out in underground laboratories (URLs). In addition to the specific technical tasks, there are key strategic challenges in geological disposal where multinational collaboration has also played a major role. These include the development of siting strategies, approaches to communicating with the public, and in preparing and presenting convincing arguments on the safety of repositories. Multinational cooperation in repository construction and operation will also occur by sharing technology and expertise. Furthermore, although, in many national repositories, cooperation involving the acceptance of foreign wastes is not foreseen, there is increased interest in expanding the possibilities for operation of multinational repositories, either as a service or as a regional partnership between willing countries.

1 Introduction

Since the earliest days of research into geological disposal of radioactive wastes, multinational cooperation has played a prominent role. One reason for this was that commercial restrictions were very much less than in other areas of nuclear technology. Developments of nuclear power plants and of fuel cycle facilities for the front-end and for reprocessing were competitive commercial fields. Ensuring that wastes could be disposed of safety and economically was increasingly recognised as a common challenge that, although not a major contributor to its costs, is a pre-requisite to obtaining the necessary public acceptance of nuclear power. Solving this universal problem was seen to be more important than trying to exploit commercially the advances being made. In addition, in many countries, including the USA, UK, France, Japan and Germany, development work has been led by government organisations with little interest in commercially exploiting the work.

The specific aspects of geological disposal programmes that have been, and are being, strongly influenced by multinational cooperation are many and varied. Since intensive cooperation in waste disposal has been taking place for decades, it is impracticable to document here the numerous specific projects that have been carried out. Instead, this overview identifies the principal types of collaboration that take place between national programmes and illustrates these with examples. The following activities that can benefit from cooperation are addressed:

- Developing strategies and concepts for geological disposal
- Knowledge exchange and transfer
- Jointly developing methodologies and producing experimental results
- Organising large scale joint research projects
- Communication activities
- Implementing waste management facilities

The very close relationships between national programmes can also result in potential disadvantages that must be guarded against. These risks are also discussed.

2 Developing Strategies and Concepts

The very principle of geological disposal has become almost universally accepted largely due to cooperation between nations. The original idea can be traced back to the US National Academy in 1957, but in the 50 intervening years it has been consolidated and refined in innumerable multinational efforts, including those organised by the IAEA and the NEA. Today, it is almost universally accepted that geological disposal is the only available approach to ensuring very long term safety and security for certain types of radioactive wastes. Of the major nuclear nations, the UK and Canada were the most recent ones to reaffirm this formal conclusion after extensive public consultation processes.

At the more specific level of repository design concepts there have been strong influences of international cooperation. The early concepts developed by some nations have been adopted and/or adapted by many programmes that started later. The pioneering work was sometimes a reaction to public or political pressures tied to continued use of nuclear power. In such countries, reference disposal concepts were established that have been taken up widely, e.g. the KBS3 design developed in Sweden for hard rock or the horizontal emplacement in small diameter tunnels proposed by Switzerland for hard rock or clays. Other examples are German concepts for repositories in salt domes or the US work on bedded salt which led to the implementation of the first custom-built deep geological repository at WIPP. All these basic designs were published openly. The choices of potential host rocks for a deep repository have also been affected by international trends – although local geology obviously also determines preferences strongly. Early work in granite in Scandinavia and salt in the USA led many countries to look first at these options. Later, increased interest in clays could be observed as various countries moved in this direction.

These technical aspects are less affected by societal considerations than are the challenges of choosing a repository development strategy. Despite the frequent references to “cultural differences” between nations that affect societal issues, however, there has still been a convergence of approaches. There are key strategic and policy challenges in geological disposal where multinational collaboration has clearly played a major role. These include the development of siting strategies (where a staged approach has become common), approaches to communicating with the public (where dialogue rather than one-way information flow is now preferred) and in preparing and presenting convincing arguments on the safety of repositories (where the term “Safety Case” is now common). Many of these developments have been the result of direct cooperation between nation programmes in the framework of joint projects or of bilateral or multinational working groups.

Specifically, siting programmes in many countries have progressed from a “decide, announce, defend (DAD)” strategy to an approach requiring local acceptance or even community volunteering. Repository implementation plans have evolved from relatively short, one-off construction projects run by experts to extended staged programmes requiring continuous interactions with the public. The growing consensus on these strategic issues results, in large measure, from the exchange of positive (and negative) experiences between national disposal programmes.

A final, very topical, strategic issue that is increasingly affecting multinational cooperation on waste disposal strategies relates to the growing global concerns about nuclear security. There is wide realisation that national and global security depends on maintaining tight control over nuclear materials, including spent fuel and HLW. This encourages multinational cooperation efforts aimed at emplacing such sensitive materials underground in secure repositories in either a national or a multinational context.

3 Knowledge Exchange and Transfer

As pointed out in the introduction, there has always been a very free exchange of knowledge in the waste management field. A key mechanism for this is open publication of results, including comprehensive report series published by implementers, detailed reviews by regulators and independent studies by research labs. In addition there are numerous workshops and conferences (which some believe to be almost too frequent throughout the year), and exchanges in the scope of bilateral agreements (of which a typical implementing body may have 5-10) and also through international organisations (most prominently the IAEA and the NEA).

Exchanging polished final results in journal articles and conference proceedings does not, however, suffice to transmit the in-depth knowledge that is gained more when projects fail than when they are successful. Accordingly, various national programmes have established international advisory groups which include chosen experts with detailed knowledge of foreign programme developments. The International Technical Advisory Group (ITAC) of the Japanese implementing organisations, NUMO, is a good example of this with its members having been heavily involved in 8 different national programmes. In practice virtually all major national waste disposal programmes have engaged experts from other countries as sub-contractors, advisors or reviewers. Another mechanism for transferring in-depth knowledge is by exchange of personnel between programmes. This has, however, been used to a relatively limited extent.

Finally, commercial approaches are also employed to transfer knowledge, in particular to young disposal programmes. Some of the larger national implementing bodies have a fully commercial consulting wing of their own (e.g. SKB, Nagra, Posiva, Nirex) or else combine their resources to provide consulting services (e.g. the CASSIOPEE group of waste management organisations). At a more generic level there are also various initiatives today aimed at transferring knowledge by means of formal tuition in courses. These are sometimes provided by fully commercial organisations, sometimes by not-for-profit organisations (e.g. ITC in Switzerland or the WNU in the UK) and occasionally funded or organised by the IAEA.

4 Developing Methodologies and Producing Experimental Results

The repository design work and the analyses of the safety of these facilities depend on a vast range of data that has been produced in laboratories across the world – and again, virtually all of these data were freely available. In fact, since a large subset of the data is generic, it was soon realised that cooperative efforts between national could generate such data more efficiently. Joint efforts on defining data collection needs and on sharing the work have been encouraged by international organisation like the EC, the IAEA and the NEA.

Many of the experimental or theoretical methodologies used in geological disposal programmes have been developed in one country and then shared with others. Laboratory techniques for measuring leach rates, sorption values, diffusion rates, etc. have been refined by continuous exchange of experience between organisations in different countries. Key in-situ field measurements e.g. in seismics, hydrology, rock mechanics and core-mapping have been similarly shared, As a result, a rather universal state-of-the-art has been established in such areas.

One of the most successful vehicles for encouraging multinational cooperation in the area of in-situ measurement has been the underground research laboratory (URL). This type of facility was developed at an early date in several countries and in various host rocks. Sweden hosted multinational research projects in granite at the STRIPA mine and now does so at Aspo; Germany ran Asse as an experimental facility in salt; Switzerland operates the Grimsel laboratory in granite and also the Mont Terri facility in clay; Belgium inaugurated the underground facility in the Boom clay at Mol. These early facilities led to other underground laboratories in Canada, Japan, France, etc. and in most cases the experimental programmes were extremely international. The early URLs were generic in nature, aimed at investigating the basic properties of different potential host rocks and at developing investigation tools that could be used at later site-specific URLs. Therefore, the results could be widely applied, which made this a fertile area for multinational projects. Many of these were (or are being still) carried out in facilities in relatively few countries to which teams of researchers travel from partner nations. For some nations, an added incentive for seeking access to a URL in a foreign country has been that these facilities are expensive and that it has not always been straightforward to achieve that local acceptance from a host community. This latter aspect led to the abandonment of URL plans in a number of countries, including Spain, France, the UK and the Czech Republic.

On the theoretical side, the most intensive interactions have occurred during development and testing of the suite of calculational models needed to analyse the safety of a repository. Included are specific process models for hydrogeology, geochemistry, nuclide transport, and biosphere transport. Equally important has been the joint development of the fundamental framework of safety analysis, involving scenario, consequence and risk analyses. In practice, the still wider framework for judging and communicating repository safety – now commonly known as safety case development - is today a widely accepted approach that has been developed in a real multinational effort.

5 Organising Joint Research Projects

In addition to the numerous multinational projects centred on underground research laboratories, there have been many other joint projects with participation of organisations from different countries. The driver for such collaboration has often been the wish to share the high cost of projects. One early example was the joint Japanese-Swedish-Swiss (JSS) Project on the leaching of HLW glasses, involving expensive experimental work in active laboratories. In a similar fashion, the complexities of working with real HLW materials led to a proposed ASSE project involving 30 full-sizes HLW glass cylinders produced in the USA. Unfortunately, this particular experiment was never completed because German priorities switched to disposal of un-reprocessed spent fuel, but the handling and emplacement systems were constructed and tested underground. Further examples of experimental projects that are so large that they virtually necessitate multinational participation are full scale underground tests such as the FEBEX experiment at Grimsel, the heater and buffer mass tests at Stripa and the prototype repository projects at Aspo.

In other cases, wide international participation in specific studies has emerged because of the uniqueness of the actual study objects. The most obvious examples here are the natural analogue studies on geological anomalies around the world. Investigations of this type have taken place at, for example, Poços de Caldas in Brazil, Oklo in Gabon, Oman, Jordan and Alligator River in Australia. These studies were all organised, funded and performed by ad-hoc multinational groupings involving scientists from many nations.

Today major multinational research efforts are organised most often through the international bodies concerned with waste management. Foremost, because of its substantial budget, is the European Commission. A long list of collaborative projects has been run under its Framework programmes. The types of collaboration supported by the EC have evolved over the years and in FP6 emphasis was placed on new structures, such as integrated Projects (IP) and Networks of Excellence, which are designed to encourage very wide international involvement. In practice these have certainly resulted in large participation; for example the FUNMIG IP has 51 participating organisations and 28 Associated Groups. The large administrative overheads in organising such collaborations are, however, a drawback and the difficulty in maximising coverage of the topics studied while minimising overlap are considerable.

6 Communication Activities

It is obvious that approaches for communicating with national and local publics and politicians will be dependent on national cultures in ways that make developing common methods less straightforward than in the scientific and technical arenas. Nevertheless, there are many initiatives making use of multinational experience to aid communication in national programmes. Standard methods have been to share documentation produced for communication purposes, to inform in-depth on experience, and to provide foreign speakers for national public events. On rare occasions, a joint product has been developed for use in communication activities in a range of countries. A good example is the film of natural and archaeological analogues produced in a joint project involving organisations from 8 countries and dubbed into different languages.

One extremely successful example of collaboration on communication is the study visits of foreign groups that many advanced disposal programmes have hosted over the years. Representatives of many nations have toured the Swedish SFR repository, the Grimsel, Mol and Asse underground laboratories, the repository excavations at Yucca Mountain, Gorleben and Onkalo, etc. A direct viewing of the benign conditions in an actual deep underground facility can be more informative for the public and politicians than thick documents on technical issues.

In recent years, there has been a move to developing more structured multinational projects aimed at exploring approaches to involving the public in waste disposal issues. Work of this kind has been encouraged by the EC which is supporting a number of studies on the governance of waste management programmes. In addition, the EC publishes periodically on its Eurobarometer a survey of public attitudes towards a variety of waste management issues.

7 Implementing Geological Disposal Facilities

Today, repository programmes in some countries are finally moving towards the implementation phase. The prime candidates for a first geological repository for HLW or spent nuclear fuel are Finland, Sweden, France, and the USA. Will multinational cooperation continue into the era of repository construction and operation? Certainly, in the first three of the countries mentioned, cooperation involving the acceptance of foreign wastes is not foreseen. The USA does accept spent fuel from research reactors and has proposed in its recent GNEP initiative extending this take-back service to commercial fuel. With the growth in interest in nuclear power, there is increased interest in expanding the possibilities for operation of multinational repositories. A prime driver is the security concerns of the major countries if spent nuclear fuel and HLW is to be held at numerous places around the world, including in countries starting new nuclear programmes. From the point of view of small or new nuclear nations, the main attraction is spreading the high costs of geological disposal. Multinational repositories may eventually be implemented, either as a service (as in proposals made in Russia and the USA) or as a regional partnership between willing countries (as proposed in the EC supported SAPIERR studies).

The issue of multinational disposal remains controversial, however, and there are also other less far-reaching approaches that may offer help, in to particular less advanced nuclear countries that choose to implement national repositories. These include sharing technology, and expertise and perhaps even access to expensive facilities other than repositories (e.g. to waste encapsulation plants). These types of approaches are the subject of the CATT project currently being carried out in the EC FP6 framework.

8 Potential Down-sides to Multinational Exchanges

The long list of the different aspects of multinational cooperation discussed above make it obvious that all national waste management programmes – especially in new or small nuclear nations – stand to gain considerably from open exchanges. Are there any possible drawbacks resulting from the close interactions between countries?

Some negative aspects can certainly be perceived by individuals directly engaged in national programmes – in particular by scientists involved in the development work. If knowledge or developments can be taken over from foreign countries, this may result in less work for nationals. This should not be a real problem for any country that will develop a national repository since the generic aspects that can be taken over are counterbalanced by the extensive in-situ work on siting that must be done locally. However, for the ultimate form of cooperation on geological disposal, multinational repositories, there would certainly be a much reduced need for national capacity – and this realisation has led to some scientists opposing multinational facilities, even though their country, as a whole, would benefit economically.

But there are also potential drawbacks from the point of view of the repository implementer, be it the government or a private entity. One real danger is that the free flow of all information can lead to a "one size fits all" mentality. For example, if site characterisation costs in the USA are known to be several billion dollars, how can smaller programmes justify programmes costing a tenth of this? If the much admired Swedish and Finnish repository concepts use copper containers, how can one justify steel overpacks, or even no overpack? Of course, the opponents of geological disposal are practised at making such comparisons and at using these to attack their national programmes. It is not only in small countries that this occurs. For example, the State of Nevada tried to have the Yucca Mountain safety assessments disqualified because USDOE did not use the exact terminology "safety case" that has been introduced in many national programmes.

However, perhaps the most serious potential danger in intense international cooperation is that free thinking will be limited and flexibility reduced. Even today there is a noticeable tendency for waste disposal concepts to be chosen to a too great extent on the basis of their current "popularity". If the KBS-3 disposal layout is being adopted in so many programmes, why look at fresh ideas? If the currently favoured host rocks are clays and granites, why should we consider salt? The freedom of thought can potentially be endangered also in another way. Communication and exchanges of views are so common, that really independent review of scientific work can be lacking, unless special measures are taken. For example the well-known national disposal programme reviews organised by the NEA and the IAEA are often performed by committees staffed with key participants of other national programmes with which the subject country has a long history of collaboration.

9 Conclusions

The development of geological disposal strategies, concepts and technologies across the world has unquestionably been helped by the extremely open collaboration between national programmes. The non-commercial approaches adopted widely at the outset by waste management organisations that were primarily interested in ensuring the availability of safe solutions to a common problem encouraged the spread of knowledge and the free transfer of data. Nevertheless, there has also been considerable duplication of effort on research issues that are of a generic nature and need not be reproduced at multiple locations. Small waste management programmes would be better advised to concentrate their studies on the site-specific aspects that can not be taken over from other countries or else tackled in a multinational framework.

Very successful multinational projects have been organised on an ad-hoc basis by groups of countries with common interests. International organisations have also strongly encouraged cooperation between their members. This has been very positive when information and experience are exchanged in working groups, committees or conferences. Some large experimental projects organised by international bodies such as the EC have also been successful, but the explosive growth in participation in some recent joint projects presents great challenges for efficient and cost-effective organisation.

The need for collaboration in waste management between countries will grow as the revival of interest in nuclear power leads to wider demands for safe and affordable disposal facilities. The more advanced nations are continuing to collaborate with each other and with newer waste programmes. However, commercial goals are becoming somewhat more apparent as countries that have invested heavily in developing mature disposal concepts begin to consider whether some of the costs can be recouped. Discussions on intellectual property rights are a relatively recent feature of interactions between programmes. How will these issues develop in the future?

The positive news is that, still today, the major nuclear nations realise that it is in the interest of all countries to ensure that the technology is recognised as safe, secure and environmentally acceptable. Waste disposal continues to be recognised as a key element in this respect. Failures resulting in conventional or radiation harmful effects to persons in any region of the world will impact on all other countries. Accordingly, collaboration will still be a key component and will continue to be relatively open. Advanced waste disposal programmes will exchange information and will transfer expertise to newer programmes without excessive commercial constraints. International organisations are intensifying their efforts to support collaboration.

However, the big step – the actual construction and operation of geological disposal facilities – has scarcely begun. Disposal concepts, repository designs, experimental data and project plans are not enough. Eventually, all countries that produce long-lived radioactive wastes will need to have access to state-of-the art deep disposal facilities. This will be possible only if the long history of multinational collaboration in this area leads on to further intensive transfer of expertise and technology that enables every nuclear nation to emplace its wastes in a national or multinational disposal facility. ■

Session 2A

Material Parameters and Experiments

2A.01 The Mont Terri Rock Laboratory: Research in an Argillaceous Formation for Geological Disposal

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Repositories for radioactive waste have to provide long-term safety and security for radioactive materials. Generic and site specific rock laboratories play an important role in the characterisation of such repositories. The experimental results gained in these underground facilities are used, together with information from natural analogues, deep drilling programmes and modelling, to assess the evolution and performance of a geological disposal-repository.

Over the past ten years, the 12 Mont Terri Partner organisations ANDRA, BGR, CRIEPI, ENRESA, GRS, HSK, IRSN, JAEA, NAGRA, OBAYASHI, SCK-CEN and SWISSTOPO have jointly carried out and financed a research programme in the generic Mont Terri Rock Laboratory, an underground research facility adjacent to the security gallery of the Mont Terri motorway tunnel, in the vicinity of St-Ursanne, Canton Jura, Switzerland (Figure 1). The aim of the project is the geological, hydrogeological, geochemical and geotechnical characterisation of a clay formation, specifically of the Opalinus Clay. The experiments can be assigned to the following three categories:

- process and mechanism understanding in undisturbed Opalinus Clay,
- experiments related to repository-induced perturbations and
- experiments related to repository performance during the operational and post-closure phases. The experimental results provide input tor assessing different phases of repository evolution and the performance assessment.

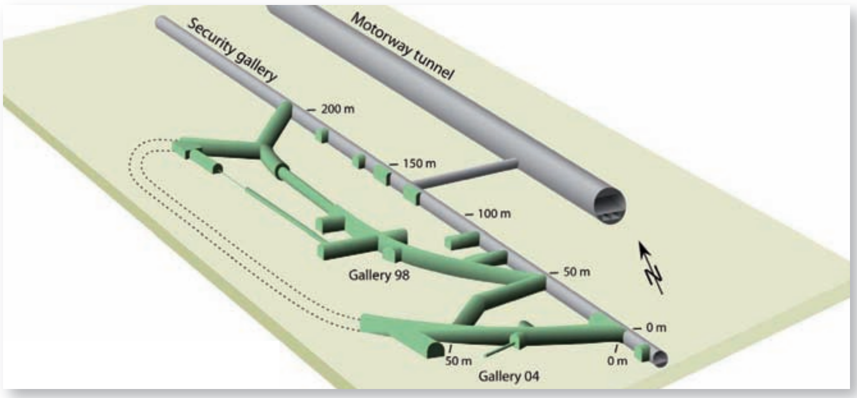

Fig. 1: Layout of the Mont Terri Rock Laboratory

Key experiments tor geological disposal, based on a questionnaire completed by the Delegates of the Mont Terri partners, are: self-sealing of open discontinuities in the host rock and in the excavation damaged zone (EDZ), characterisation of rock-pore water interactions (undisturbed and disturbed argillaceous formations), optimisation of sealing materials for different geological disposal concepts, understanding of the THM processes in the near-field (bentonite backfill and host rock), understanding of gas flow processes and identification of corresponding flow paths and, finally, diffusion and retention processes of various radionuclides in the bentonite and argillaceous host rock. The performance and results of some of these key experiments will be presented in more detail.

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2A.02 Investigation of the Hydaulic-mechanical Behaviour and the Gas Migration Issue in the Opalinus Clay at the Mont Terri URL

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Abstract

In order to obtain the knowledge necessary for the safe disposal of radioactive waste in a repository in argillaceous rock, various investigation programmes are performed in the Mont Terri Underground Laboratory. A central issue which GRS has investigated in the frame of various projects is the question of gas migration. Waste containers and the metallic components will corrode resulting in the generation of hydrogen, with the potential of building up gas pressure. Starting with a summary of gas-related results obtained during the HE-D heater test, this paper presents current work and plans for future investigations, with the emphasis on coupled hydraulic-mechanical behaviour and gas migration. The focus of investigation is on the clarification of relevant gas migration mechanisms in the undisturbed clay rock (project HG-C/BET) and the investigation of clay-sand mixtures as backfill material which combines low gas entry pressures and gas permeabilities with reasonable sealing capability versus water migration (project SB).

1 Introduction

Due to their low permeability, clay formations have a high potential to act as natural barriers and are therefore under worldwide consideration as a host rock for radioactive waste disposal. Construction, operation, and sealing of a repository require high standards of planning, implementation, and safety assessment. For proving the integrity of the geological and geotechnical barriers a detailed knowledge of the physico-chemical conditions and processes in the repository near-field is essential. Therefore, underground research laboratories (URLs) have been constructed in various countries, e.g., Belgium (Mol), Switzerland (Mont Terri), and France (Bure).

In a high-level waste repository the waste forms will be emplaced in boreholes or galleries and the remaining void will be backfilled. After healing and resaturation of the excavation disturbed zone (EDZ) and saturation of the backfill, the waste containers and the metallic components will corrode resulting in the generation of hydrogen. Rübel et al. [1] show that the amount of water available will be sufficient to corrode all iron that is present. Additionally, carbon dioxide will be released as a result of oxidation and thermal decomposition of the organic components in the clay. If the disposal boreholes and galleries are sealed gas-tight, high gas pressure may be produced leading to the potential generation of fractures in the host rock which could influence the integrity of the repository. There is, however, the possibility that, at the low gas production rates that can be expected, the elevated gas pressure will result in micro-fracturing allowing for gas migration (dilatancy-controlled gas flow). After pressure reduction the clay rock has the potential of self-sealing due to resaturation and swelling. Even if macro-fractures occur, self-sealing of the clay may neutralize this issue. For assuring repository safety it is essential that the mechanisms of gas migration through the technical barriers (backfill) or into the surrounding host rock are well understood.

Consequently, the gas issue has been one of the central questions in GRS' investigation programme during the recent years. Measurements of gas production and migration in clay backfill and host rock were performed in various projects, such as FEBEX [2], CORALUS [3], or HE-B [4]. In this paper, the results of the recently finished heater experiment HE-D and current programmes related to the gas issue are presented.

2 Results Obtained in the Frame of the HE-D Heater Experiment

From October 2003 to December 2005, the HE-D heating experiment was performed at the Mont Terri Rock Laboratory by the French Agence Nationale pour la gestion de Déchets Radioactifs (ANDRA) and GRS in order to test the capability of a prototype

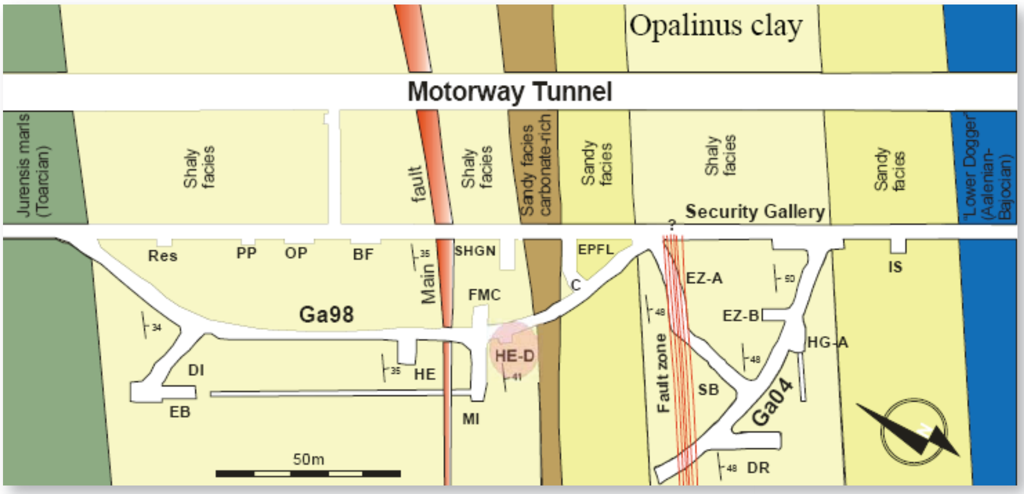


Fig. 1: Horizontal cross section of the Mont Terri Rock Laboratory showing the location of the HE-D test field

heater equipment, develop and test measurement techniques for monitoring of the relevant petrophysical parameters, observe the response of the Opalinus clay to heating and cooling, and to characterize and model the thermo-hydraulic-mechanical (THM) behaviour of the clay. GRS' main tasks in the project were the hydraulic in-situ measurements, the determination of material parameters in the laboratory, and the THM modelling [5].

The location of the HE-D test field in the Mont Terri URL is shown in Figure 1. Figure 2 shows the test layout with the horizontal heater and the measurement boreholes drilled parallel and perpendicular to the heater borehole.

The horizontal heater borehole was drilled in March 2004. Heating started on April 6, 2004 with a total power of 650 W. On July 6, 2004, heater power was increased to 1950 W. A maximum temperature of 100 °C was reached at the heater/rock interface. On March 13, 2005 the heaters were shut down and the cool-down phase began.

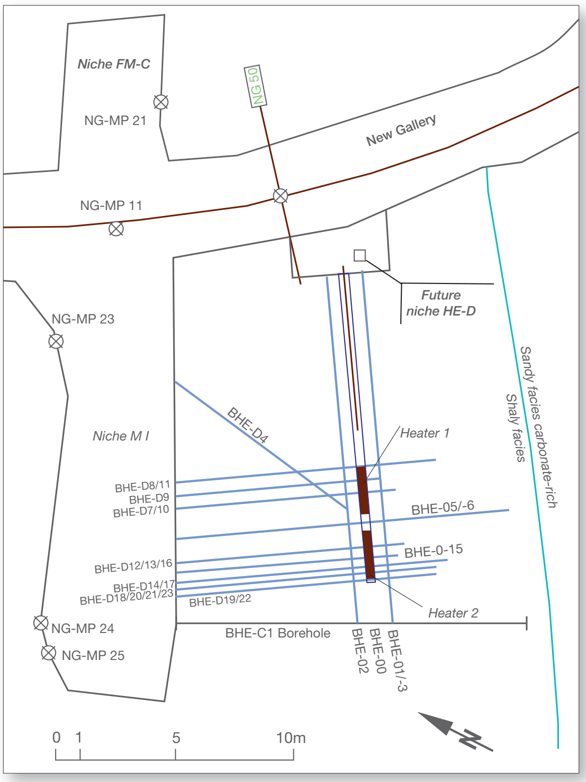


Fig. 2: Layout of the HE-D experiment with heater and measuring boreholes

2.1 Temperature and Pore Pressure Evolution

Exemplary results on temperature and pore water pressure evolution are shown in the Figures 3 to 5, together with finite element modelling results. For modelling the computer programme CODE_BRIGHT [6] was used. It solves the balance equations for internal energy, solid mass, liquid mass, air mass, and stress equilibrium. The following assumptions were made:

- Heat transport includes conduction (Fourier's law) through the porous medium, advection of liquid water and vapour flow;
- Water transport is controlled by liquid water advection (Darcy's law), vapour diffusion in air (Fick's law), and the liquid/gas phase changes (psychrometric law);
- Flow of dry air due to air pressure gradients (Darcy's law) and air dissolved in the liquid phase (Henry's law);
- Description of the elasto-plastic behaviour of the clay rock by the Barcelona Basic Model (BBM) [7, 8] with the main features of thermal expansion/contraction, swelling/shrinking;
- The clay rock is assumed isotropic and homogeneous.

The parameters associated with the constitutive models were first estimated on basis of laboratory tests and data from literature and then calibrated by back-calculations of lab mock-up heating tests.

Figure 3 and 4 show the measured and calculated temperature evolution at the interface between heater 2 and the rock and at different locations inside the rock mass, respectively. A maximum temperature of 100 °C of the rock was reached at the end of the second heating phase. There is a good agreement between the measured and calculated temperature curves, although the marked anisotropy in thermal conductivity of the Opalinus clay had to be neglected and an average value was used. Even a power interruption during the second heating phase is well reproduced.

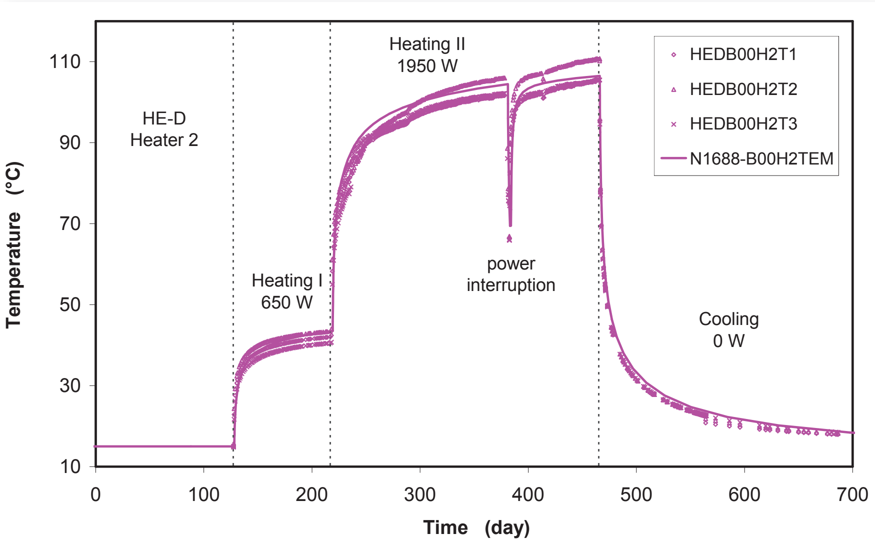


Fig. 3: Measured and calculated temperature evolution at the heater/ rock interface

Hydraulic measurements throughout the HE-D showed the expected increase and decrease of pore pressure due to heating and cooling of the rock, respectively (see Figure 5). The initial pore water pressure between 0.2 and 1.2 MPa dropped slightly after drilling of the heater borehole. The first heating phase caused a rapid increase in pore pressure, which then reached a plateau. After increase of the heater power in the second phase, pressure again rose rapidly and then, after reaching a peak value between 2.6 and 4 MPa, steadily reduced, with a transient drop during a power failure. At the end of the second heating phase, the pressure decayed smoothly to zero. The smooth decay of pore pressure during cooling at all measuring points indicates that no cool-down fracturing occurred. Figure 5 also shows that the pore pressure changes in the vicinity of the heater could again be well reproduced by modelling. Farther away in the rock mass, the agreement is less good due to a complicated initial pore pressure distribution and the simplifications of the model.

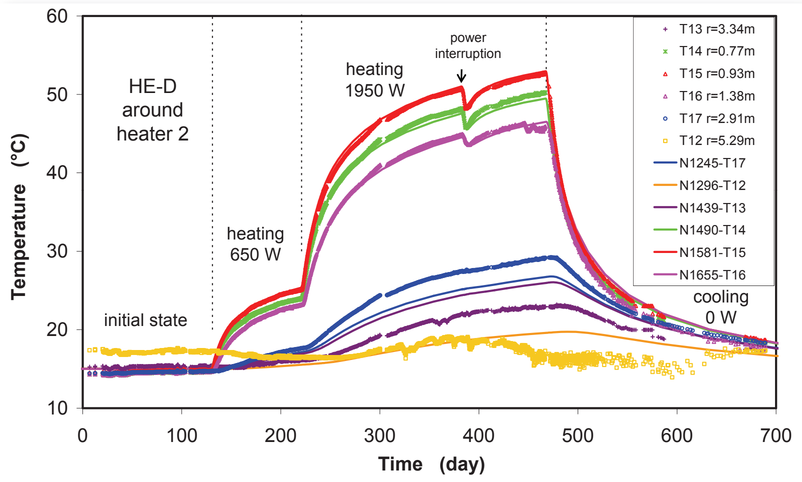


Fig. 4: Measured and calculated temperature evolution in the rock mass

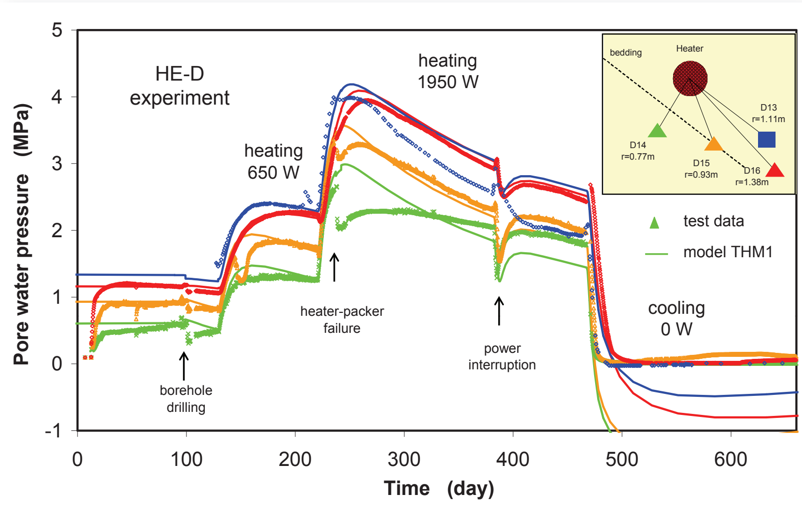


Fig. 5: Measured and calculated pore water pressure evolution in the rock mass. Calculation with a relative humidity of 85 % on the niche wall, drained conditions at the heater-rock interface, and a permeability of the rock to water of $2 \cdot 10^{-20}$ m²

2.2 Gas Entry Pressure and Gas Fracturing

After cool-down, the minipacker probes for measurement of pore water pressure were used for measuring rock permeability to water and gas entry pressure. Water injection tests with artificial pore water (Pearson water) yielded rock permeability values between 10^{-20} m² and 10^{-19} m². Afterwards, the water in the probes was displaced with gas, and gas injection tests were performed at different pressure steps. The respective pressure curves are shown in Figure 6. For the cases where constant pressure is reached during the shut-in phase (D11, D13, D16), this value can be interpreted as the gas entry pressure. In the other cases, the gas entry pressure can be estimated to range between the end prressure and the next-lower pressure step where no gas flow was detected. Most gas entry pressures range between 0.8 and 1.8 MPa, except for D13 (close to the MI niche) which is somewhat lower and D17 (above 2.8 MPa) beyond the heater.

Gas injection testing in 2 intervals of a multi-point packer array was performed during different stages of the experiment. During heating the rock remained gas-tight, from which can be derived that no significant desaturation due to heating occurred. After heating, effective gas permeabilities below 10^{-21} m² were estimated when the gas entry pressure of about 1.5 MPa was exceeded.

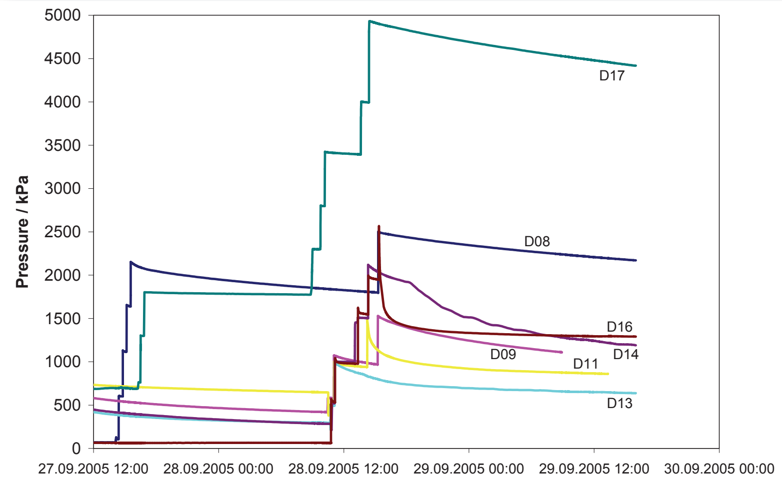


Fig. 6: Pressure evolution in the minipacker probes during gas entry pressure testing

When the gas injection pressure was increased above 3 MPa, gas fracturing occurred (see Figure 7), and the shut-in pressure dropped. Increasing the gas pressure again resulted in a frac extension with pressure responses in adjacent intervals. Form the pressure responses an orientation of the fracture parallel to the bedding plane could be deduced (Figure 8).

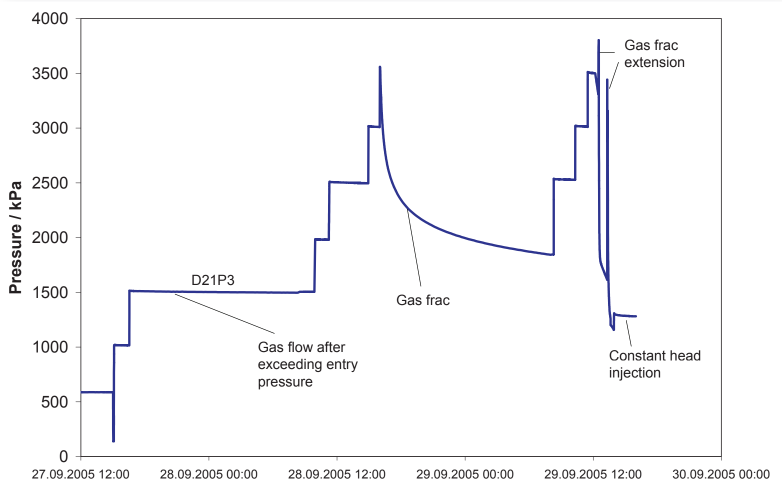


Fig. 7: Gas injection tests in interval 3 of borehole BHE-D21

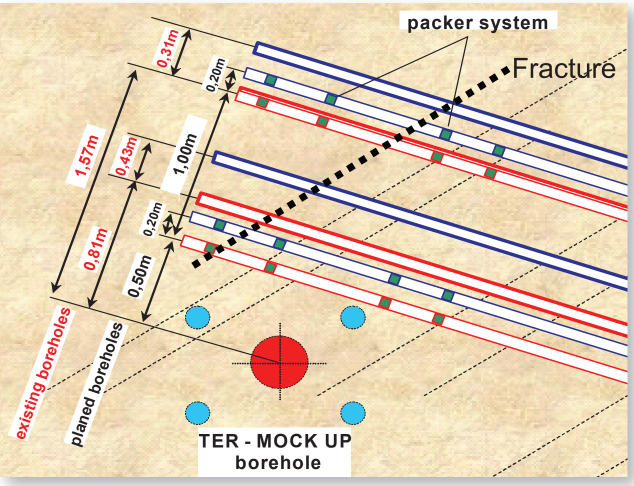


Fig. 8: Boreholes BHE-D18 to BHE-D23 with the assumed fracture after gas injection into interval 3 of borehole BHE-D21

3 Current Investigation Programmes

3.1 Hydraulic-Mechanical Behaviour of Argillaceous Rock

The results of the HE-D show that in the vicinity of the heater borehole gas migration below the frac pressure as well as gas fracturing were achieved under suitable conditions. In 2006, a new project (HG-C/BET) was started in cooperation with NAGRA at the Mont Terri URL in order to further clarify the relevant mechanisms of gas migration in the undisturbed clay formation. While NAGRA contributes to the project by investigation of hydro-mechanical coupling regarding water flow (pressure-dependent water permeability/transmissivity), GRS‘ investigation programme comprises:

- Investigations on gas migration at low pressure (dissolution and diffusion in the liquid phase or advective flow?)
- Investigation of high-pressure gas transport (dilatancy-controlled gas flow or gas fracturing?) and self-sealing capacity
- Laboratory investigations of gas-induced damage and self-sealing

The in-situ experiments are performed in boreholes oriented perpendicular and parallel to the bedding. Two sets of arrangements are used: Multi-point packer probes similar to those employed in the HE-D, and small 20-mm diameter sinter metal piezometer probes which have the advantage of minimizing the disturbance introduced by drilling. The testing programme includes the determination of the initial conditions (pore water pressure and permeability to water) and measurement of pressure-dependent gas flow into the rock by stepwise increase of gas injection pressure until a significant gas flow (below and/or above minimum stress) is detected; the effective permeability at each step is determined. The measurements will be repeated after 6 – 12 months for investigation of self-sealing.

The additional laboratory testing programme comprises

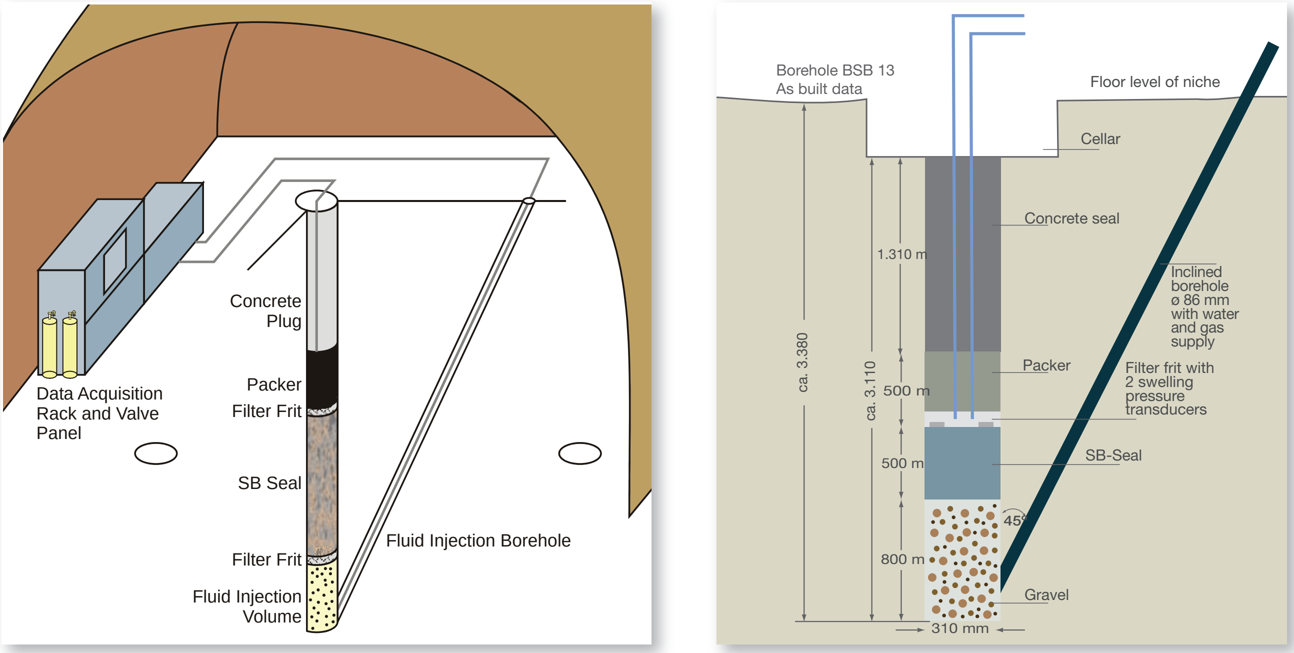
- Investigation of gas-pressure-induced damage by gas injection into the central borehole of a core sample in an autoclave until damage is reached, subsequent measurement of gas permeability, and injection of a coloured tracer liquid to determine the nature of damage (discrete fracture / micro-fracture network)
- Investigation of self-sealing by reconsolidation of a damaged sample under increasing hydrostatic stress and monitoring of gas permeability decrease, resaturation of the reconsolidated sample under hydrostatic load, measurement of permeability to water and subsequent gas injection for measurement of gas permeability
- Supplementary tests on scale-effect using large samples (280 mm diameter)

The experiments will give information about the nature of gas pressure induced damage, the degree of damage (in terms of permeability), and the self-sealing capacity of the Opalinus clay. The results will provide a basis for developing a conceptual model for the coupling between mechanics and gas transfer.

3.2 Clay-Sand Mixtures as Buffer Materials

A different approach to the issue of avoiding a gas pressure build-up in disposal boreholes or galleries is to engineer the backfill in a way that allows gas flow. Currently, various clay/sand mixtures are investigated in situ in terms of their ability of sealing against liquids when saturated while keeping a low gas entry pressure. This work is also performed at the Mont Terri URL in the frame of the SB project which has been running since 2004. The objective of the project is to test and demonstrate the sealing properties of clay/sand mixtures determined preliminarily in the GRS laboratory and to show they can technically be realized and maintained under repository relevant in-situ conditions.

Four experiments in vertical boreholes of 0.31 m diameter drilled into the floor of a test niche are performed at the Mont Terri URL (see Figure 9). The lower part of the boreholes, the injection volume, is filled with gravel as porous medium. At the



▲ Fig. 9: Principle design of one of the SB boreholes experiment at the Mont Terri URL

top of the porous medium a filter frit is placed for ensuring a homogeneous distribution of the injected water over the entire borehole cross section. Above the filter frit, the seal is installed in several layers. Above the seal another filter frit is installed for water and gas collection. The whole borehole is sealed against the ambient atmosphere by a gas-tight packer. At the bottom of the packer two swelling pressure sensors are installed. The uppermost part of the test borehole is grouted for keeping the packer in place at higher swelling pressure of the SB seal.

Various seal materials are investigated in the different boreholes: Clay/sand mixtures with ratios of 35/65 and 50/50 and crushed clay pellets (NAGRA's reference material). The seals are currently resaturated by water injection. After finishing resaturation, the permeability to water, the gas entry pressure, and effective gas permeabilities will be determined.

4 Conclusions

The issue of gas migration in an argillaceous host rock for the disposal of high-level waste is being investigated at the Mont Terri URL. The results of the HE-D heater test showed that, while cool-down fracturing does not seem to be relevant, gas pressure induced fracturing may occur if the required boundary conditions are met. A gas production that can potentially create high gas pressures is, as a consequence of corrosion, very likely. On the other hand, the results show that gas migration already sets in at pressures well below frac pressure. All these results, however, were obtained in a thermally disturbed zone. Current investigations concentrate on the gas migration mechanisms in the undisturbed rock. Moreover, the self-sealing potential of the rock after a gas-frac is investigated.

While the HE-D could be reasonably well modelled by finite element calculations, features like anisotropy could not yet be included in the GRS' modelling calculations. The hydro-mechanical coupling that is needed for the modelling of dilatancy-controlled gas flow or fracturing and self-sealing would be an even more advanced feature. The investigations performed in the frame of HG-C/BET are intended to yield the information necessary for developing the conceptual model.

Parallel to the investigation of the gas migration issue in the host rock, GRS works on the development of seal materials that maintain a low gas entry pressure. If this work proves successful, the question of high gas pressures may no longer occur. ■

Acknowledgements

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2A.03 A Comparative Evaluation of the Modelling Efforts of Gas Transport in Clay Formations as Repository System

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Abstract

Modelling of transport phenomena in clay formations is more complex than that of the other host rocks. The obvious insufficiency of Darcy's approach, the local and global heterogeneity, lack of experimental data on EDZ's, the interactions of present fluids with clay minerals are some of the features creating the complexity. Empirical correlations are used to describe the gas breakthrough pressure and gas flow rate as function of gas saturation, vertical stress and temperature. The attempts of defining gas entry pressure in terms of permeability can also be cited among the empirical, semi-empirical formulations. Simplified conceptual models by using capillaries (capillary bundles) and fractures as pathway with stress dependency geometry are affirmed to be an effective model applicable for gas, two-phase flow in plastic as well as indurated clays with and without TC coupling. In the model called "Multiple Front Propagation" fluids (wetting and non-wetting) are assumed to advance through the clay of the tip of the propagating gas flow channel. The gas-filled pathways are assumed to dilate elastically for small displacements. The number of paths is set by an input channel density. Some other approaches conceptualize also the clay formations as a crack-fracture network where the opening of the cracks is governed by deformation. An executive compilation of the modelling efforts of gas transport in clay formations is presented. Modelling concepts are evaluated and compared. Their implementations and validations are summarized. The application potential of these models for safety analysis in repository concepts is discussed. The models to describe the flow and the effect of the internal and external factors in bentonite buffers are also briefly examined.

1 Introduction

The sealing properties of especially deep clay rich formations seem promising for the isolation prerequisites of underground storage and therefore a great number of studies have been conducted and are conducted on the related subjects such as HADES, MONT TERRI, CORALUS, SELFRAC, FEBEX. One of the critical issues of these projects is the determination of the hydraulic properties and the sensitivity of these properties to thermal, mechanical and chemical perturbations in and around the storage locations. An enormous volume of literature from various fields such as hydrogeology, petroleum geology, geochemistry, thermodynamics, soil and rock mechanics, engineering geology accompanies these research activities.

Fluid movement in an underground repository system is examined generally as transport through geological and engineered barriers. The possible pathways of the transport of the fluids in a repository can be schematized as given in Figure 1. The fluids may move through engineered barriers, backfills in the repository chambers, galleries and drifts (No 1 and 2). The excavation damaged and disturbed zone (No 2) is commented generally as a hydraulic transition zone between the repository and host rock offering a more favourable medium for the transport of the fluids. The transport in host and overlying rocks and formations (No 3 and 5) might be disturbed by the structural discontinuities like faults. The thermal, chemical and mechanical disturbances due to the presence of the wastes and geologically and externally created forces act on these flow pathways separately and/or interactively. This general scenario is also valid for the repository systems in clay formations. However, the difficulties on describing the behaviour of clays pose additional problems on the description of the transport in clay formations.

The physical and numerical modelling of flow of any fluid in clay formations and backfill necessitates first of all the knowledge of the classical terminology of fluid flow in porous media as well as in fractures. The thermo and flow dynamics are two important disciplines dominating the modelling of the gas transport in clay formations. The petrophysical and mechanical properties of the host rock and backfill material are the main components. These properties differ greatly based on the location and time. The typical example of these variations occurs especially in EDZ where the permeability and porosity are expected to change dramatically shortly after the excavation and at long term.

In this paper, an executive compilation of the modelling efforts of gas transport in clay formations is presented. Modelling concepts are evaluated and compared. Their implementations and validations are summarized. The application potential of

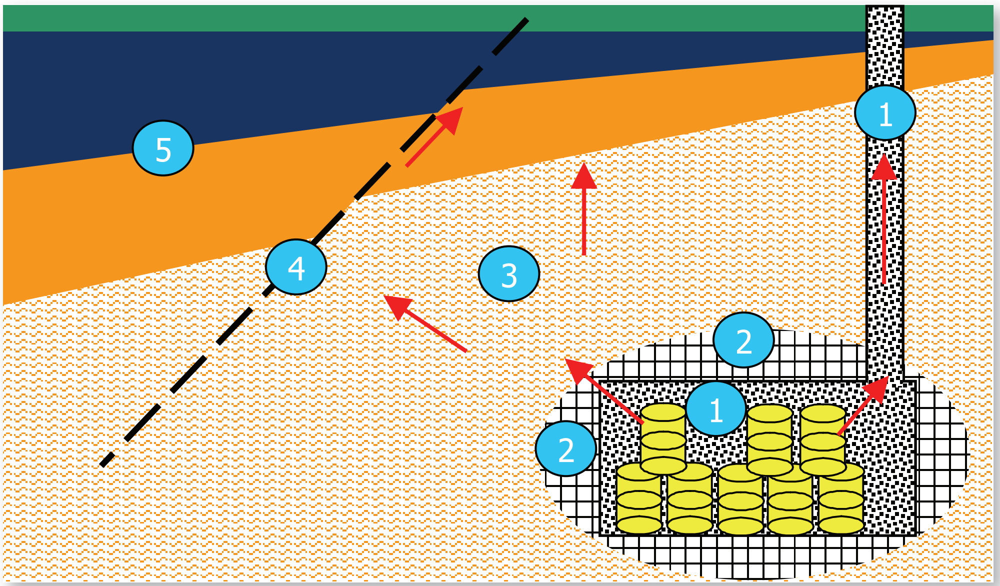


Fig. 1: The pathways of gas transport in underground repository systems

these models for safety analysis in repository concepts is discussed. The models to describe the flow and the effect of the internal and external factors in bentonite buffers are also briefly examined.

2 Gas Generation and Thermodynamics

A number of mechanisms that can contribute to the gas generation are identified in the literature. Their importance is relative to the physico-chemical, thermodynamical, geological milieu and waste characteristics. The main mechanisms of gas generation in an underground waste repository are corrosion of metals in the waste, radiolysis of water and organic materials and microbial degradation. Studies can be found in the literature on the mechanisms and rates of the gas generation in the underground waste repositories [1-4]. The modelling efforts for microbial degradation need more validation and qualification studies.

The thermodynamical system in a repository is generally assumed to consist from air-steam and generated gases as gas phase, saline water and rock. The physical properties of air or steam or a mixture of both is generally well known and can be described with simple half-empirical and empirical correlations. If other gases than air and/or steam are present in the system the effect of these gases on the thermodynamic properties should be taken into consideration if their fraction in the total composition is relevant. The calculation of the thermodynamic properties of individual gases and mixture of gases is relatively simple and accurate compared with the calculations for the liquid and gas mixtures. The thermodynamics of saline water in equilibrium with non-condensable gases like CO_2 exhibit complexities compared to air-steam-water systems. The use of Henry's law for modelling the solubility behaviour of the gas is generally applied and for the equilibrium of the gases with liquid fugacity concept is used. Especially in the last years many studies on the phase behaviour of water-saline water-CO_2 and/or CH_4 are published based on increasing importance of geological CO_2 sequestration [5]. The transport of free and/or dissolved gas or the retardation/retention as liquid and/or solid phase through a clay repository system should take the thermodynamic into account.

3 Gas Transport in Clay Formations; Concepts

In basic, the transport of gases is modelled by diffusion and dispersion supported advection. However, this approach is limited for clay formations because following of reasons. First, capillary and physico-chemical forces are higher in clay formations and play a major role in controlling the flow conditions. Second, the poro-elastic crack dilation due to effective stress concept causes intermittent, episodic flow processes in clay formations. With a physical model it is suggested that the pressure in the source reservoir rises until the sealing criterion is no longer met and the incident flow pathways dilates with an increase of the fracture aperture (Figure 2) [6]. The flow rate along this pathway then exhibit a highly nonlinear dependency on effective stress, with the flow rate increasing dramatically as the source pressure continues to rise. The flow to the sink will then drop off dramatically and it is possible that the source reservoir may become totally depressurized.

The phenomenological description of the transport mechanisms for clay formations are made by NAGRA on the basis of Opalinus Clay with geomechanical and transport considerations [7]. These include firstly the advection and diffusion of dissolved gas governed by Darcy and Fick's law successively in the case where the generated gas dissolves directly in the fluid phase because of the low gas production rates. If free gas is formed as a second phase due to the saturation of the liquid phase and/or higher gas generation rates, two phase flow begins. The controlling factor for the two-phase flow is the gas entry pressure. Once the gas entry pressure has been exceeded, the gas mobility is controlled mostly by the intrinsic permeability k of the formation, the permeability-saturation relationship (relative permeability), and the relationship between the capillary pressure and the water saturation (suction or water retention curve). If gas pressure increases further, because two-phase flow cannot transport gas quickly enough, microscopic gas pathways may form as the result of the poroelastic variations. Tensile gas fracs is reported to be expected when the pressure build-up in the rock is rapid. The macroscopic fracture is initiated quasi-instantaneously and propagates at about the velocity of a shear wave. Gas flow in such a macroscopic tensile fracture can be seen as a single-phase flow process. The propagation stops when the gas pressure in the fracture becomes less than the value of the minimum principal stress (shut-in pressure). Macroscopic gas fracs require gas pressures in the order of the minimum tensile strength. According to this plausible concept, the hydraulics in clay formations is controlled mainly by the multiphase flow in porous media based on the semi-empirical generalisation of Darcy's Law with the application of relative permeability. The works on capillarity and gas flow in low permeability rocks are helpful reporting on the observations on departures from Darcy's law at high flow-rates due to inertia and turbulence. The Fick's Law for the diffusion of the generated gases into the liquid phase and the Henry's Law for evaluating the thermodynamic of the two-phase conditions are therefore the other main mathematical expressions of hydraulics in clay formations. The hydraulic modelling should be coupled with the expressions for poroelactic and if exits with irreversible deformations.

The complex interaction between the mineral phases, water and solutes is the other complex features of the hydraulics in clay formations. This characteristic together with the sub-microscopic dimensions of the pore channels, the deformable matrix and the generally low tensile strength are of key importance to the problem of gas transport in compact clays. Although the transport of gas in solution is a very slow process in unfractured clay formations, its general importance in geological repository systems is obvious.

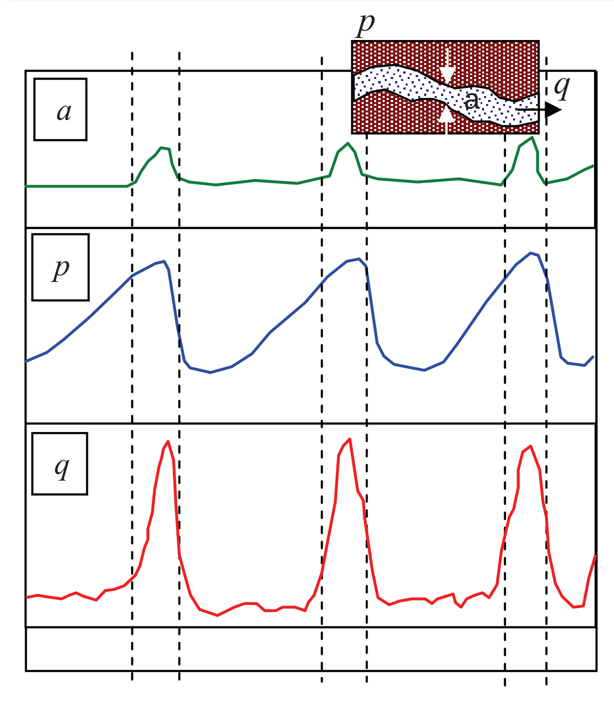


Fig. 2: The changes of pressure (p), flow rate (q) and fracture aperture (a) during the intermittent flow processes.

Three possible basic mechanisms have been proposed for gas movement in water saturated compacted bentonite as a conventional porous medium; gas flow governed by conventional concepts of capillary pressure and relative permeability, microfissuring of the clay, in which small fissures are created or opened by the invading gas to provide the pathways for the gas to enter the clay and macroscopic fracturing of the clay to provide fracture pathways for gas flow [8].

4 Modelling Approaches for Gas Transport in Clay Formations

4.1 Darcy Law Related Approaches

From a philosophical point of view, Darcy law is a turning point of hydrogeological studies. It comprises the know-how and experience of previous studies whereas it offered a large research area on the subject. For that reason all attempts to model flow in clay formations based on similar assumptions as Darcy law have been cited here under the same title. Darcy law describes the laminar flow of an incompressible fluid which saturates an isotropic porous media. The well-known empirical equation is modified successively for the flow of the gases taking the compressibility and slip of the gas molecules on the walls of the pore as well as the multi-phase flow into account.

4.1.1 Capillary Bundle Model [9]

In capillary bundle model, porous media in clay formations is assumed to compose from a series of non intersecting capillaries (Figure 3). The capillaries have a uniform radius of r, and the number of capillaries per unit cross sectional area of core, with radius in the range [r, $r+dr$] is $N(r)dr$. The length of each capillary is τL where τ is known as the tortuosity factor. If it assumed that the portion of the capillaries filled with water is $R(r,t)$, with the assumption of the slow flow of water in the capillaries and introducing the capillary pressure concept, the position of the water $R(r,t)$ is calculated based on Navier-Stokes equation as follows:

$$R^2(r,t) = \tau^2 L^2 - \frac{t}{4\mu_w}(r^2(P_g - P_w) - r\sqrt{2}\sigma) \qquad (1)$$

This equation is valid in the region $0 \leq R(r,t) \leq \tau L$.

The gas pressure must exceed the water and capillary pressure combined in order for a capillary to start desaturating. The smallest capillary radius, r_s, which starts desaturating at given gas and water pressure, is given by:

$$r_s = \frac{\sigma\sqrt{2}}{(P_g - P_w)} \qquad (2)$$

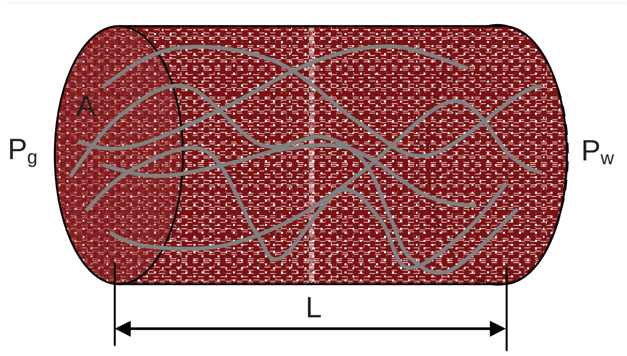


Fig. 3: Capillary bundle model [9]

4.1.2 Multiple Interacting Media

Considering the clay formations as microfissure systems, attempts were made to model transport in clay formations as a problem of fractured media. The studies of Barrenblatt and Warren and Root on formulating the dual porosity (DP) or multiple interacting media were developed by Gerke and van Genuchten to model the fluid movement in microfractured clay systems [10,11]. The model is combined with two systems: a macropore or fracture pore system at the macroscopic level by fracture continuum (FC) and a less permeable matrix pore system by matrix continuum (MC). In DP model, the transfer value between these two continua plays an important role to model accuracy [12].

Figure 4 shows a finite element composed by a rock matrix and a series of *n* fractures designed to model the flow in clay formations. The number of fractures in an element depends on the width associated with each fracture, *a*, which is a characteristic size of the material, and the element size *s*. The flow phenomenon through a single fracture is considered first. Liquid and gas flow is calculated using Darcy's law. The intrinsic permeability can be calculated, assuming laminar flow with cubic law for discrete fractures. When a set of *n* fractures is included in a finite element the equivalent intrinsic permeability of the element can be calculated as:

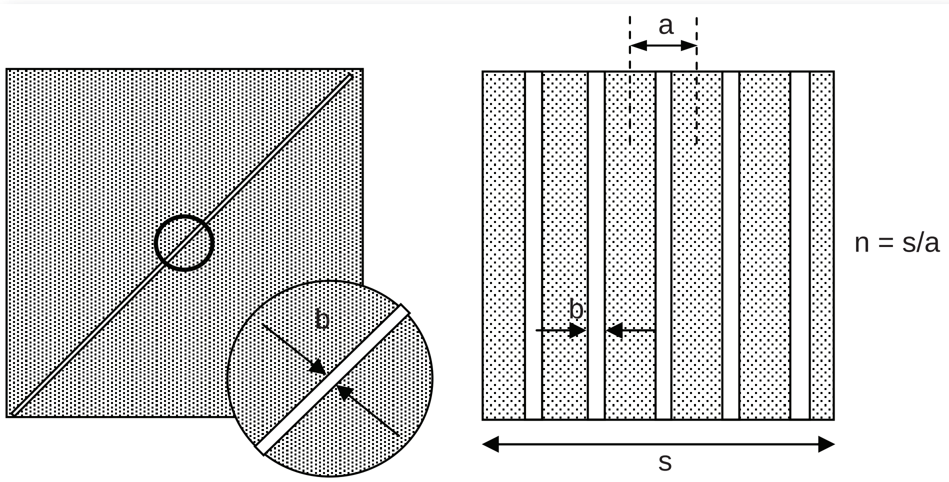
$$k = k_{matrix}\left[\frac{s-nb}{s}\right] + \sum_{i=1}^{n}\left[k_{fracture}\frac{b}{a}\frac{1}{n}\right] \cong k_{matrix} + \frac{b^3}{12a} \tag{3}$$

where k_{matrix} is the reference intrinsic permeability of the matrix. *s* is the element size (width normal to flow direction), *a* is the width associated with each fracture, and $n=s/a$ is the number of fractures in the element. The basic capillary pressure definition in a fracture of width b can be used directly to calculate the air entry value of the element by combining the definition of capillary pressure to start desaturation is obtained as:

$$P = P_o \frac{\sqrt[3]{k_{fracture\_o}}}{\sqrt[3]{k_{fracture}}} \tag{4}$$

where subscript *o* refers to a reference (initial) aperture.

It is reported that the model calculations is in good agreement with the experiments using Boom Clay samples conducted in the framework of MEGAS Project [12].



▲ Fig. 4: Rock with a single idealized fracture and a series of parallel fractures uniformly separated [12]

4.1.3 Matrix Diffusion in Fractured Clay Rock, Brush Model [13]

To better consider the matrix diffusion in clay formations a model called brush model is proposed. Gravitational effects are neglected and thus the schematization of nature yields a system of subhorizontal fractures with an attached rock matrix. If it assumed that the dissolved substances are transported with the same velocity over the entire rock column, a model of one horizontal fracture is representative for the entire system. Due to the symmetry considerations, only one half of the system is modelled. The fracture plane is discretized by 2-D quadrilateral finite elements in the x, y plane with Cartesian coordinates. Advective-dispersive transport without retardation is assumed. The transport of a substance that is not sorbed through a fracture is modelled using the following 2-D advection-dispertion equation:

$$\frac{\partial c}{\partial t} = D_L \frac{\partial^2 c}{\partial x^2} + D_T \frac{\partial^2 c}{\partial y^2} - \frac{v_D}{\phi_{eF}} \frac{\partial c}{\partial x}, \tag{5}$$

where *C* is solute concentration [mg/l]. It is indicated that the tracer velocity, υ_D/ϕ_e, is sufficiently high that the term D_m does not contribute significantly to the total dispersion coefficient D_L and D_T. Darcy law is applied for the flow velocity in the fracture. Concentration in the rock matrix satisfies the 1-D diffusion equation:

$$\frac{\partial c}{\partial t} = D_e \frac{\partial^2 c}{\partial z^2}, \tag{6}$$

where D_e is the effective diffusion coefficient.

4.2 Non-Darcy Appproches; Mechanical Coupling

According to Darcy law the water flux in a porous medium is linearly proportional to the hydraulic potential gradient. Except geomechanical effects, clay minerals exhibit exceedingly complex interactions with both water and solutes creating anomalies in flow behaviour as aqueous solutions moves through the sub-microscopic flow channels of clay formations as well as compacted bentonit. Direct laboratory evidence for non-Darcy flow through clays has been widely reported, but is still fairly inconclusive. In the analysis of experimental data three possible types of deviation from Darcy‘s law can be defined:

- a nonlinearity in the relationship between flux and hydraulic gradient,
- a threshold hydraulic gradient, below which the flux is zero,
- the occurrence of components of flux which cannot be attributed to the hydraulic gradient.

Many attempts for describing the transport in clay formations are proposed. They are primarily focused on the elimination of the effects of the internal and external forces on the transport thought clay media.

4.2.1 Empirical Correlations

Due to the experiments performed with Boom and Pontida clays with a uniaxial compaction up to a predetermined vertical (axial) stress (0,4 to 2,5 MPa), the gas specific discharge was shown to be dependent on the inlet pressure, on the saturation and on the vertical stress [14]. The following relationship was found for Pontida Clay :

$$q = (3{,}984 + 9{,}01x10^{-4}\sigma_v)\, p_g^{(1{,}2348-4{,}29x10^{-5}\sigma_v)}\, 17{,}663\left[\frac{(S_t - S)}{S}\right] \tag{7}$$

where q is the gas discharge, μl /s^{-1}, p_g the excess gas pressure, σ_v the vertical stress, Pa, S the saturation in the sample and S_t is the saturation limit below which the threshold pressure needed for the gas to penetrate the medium is equal to zero. Attempts were also made to correlate the gas entry pressure to permeability and bentonite swelling pressure for compacted clays [15]. The empirical correlations are specific to the cases studied rather than universally accepted.

4.2.2 Planar Pathway Model

A simple modelling approach to pathway flow is given using the cubic law for determining the flow rate through a planar fracture [6]. The schematic of the considered model is given in Figure 5 in which the source and sink terms are determined with pressures p_1 and p_2. It is assumed that the aperture of the pathway (fracture) is a function of the applied overburden (confining) stress (σ) and pore pressure (p_g):

$$b = f(\sigma - p_g) \tag{8}$$

Following this concept a critical value of (σ-p_g) is supposed which leads to total closure of the flow pathway with b=0. This is called as sealing criterion and is written as:

$$q = 0 \qquad for \qquad (\sigma - p_g) > (\sigma - p_g)_{crit} \tag{9}$$

As fluid pressure gradually build up in the source reservoir, a point will be met at which the sealing criterion is no longer valid and fluid starts to flow towards the sink. The introduction of the pathway aperture relationship into the cubic law makes the effective permeability of the pathway a highly nonlinear function of the local fluid pressure and stress distribution. The resulting flow law therefore exhibits both a threshold and nonlinearity. An increase in fluid (gas) pressure above the critical pressure leads to dilation of the flow pathway and a substantial increase ineffective permeability. This critical pressure is physically comparable with the threshold or entry pressure. The numerical application of the stress dependent pathway is easy if the flow through fracture is modelled by using a model depending on the geometry of the fractures. However the modelling of the transport phenomena in clay formations only with formulation of flow in an idealized fracture is an obvious oversimplification. Some potential deficits of this assumption are plastic yielding of stress-supporting asperities, compaction of a clay-rich fault aperture under high effective stress, swelling of the rock, precipitation of minerals from in-situ or flowing liquid.

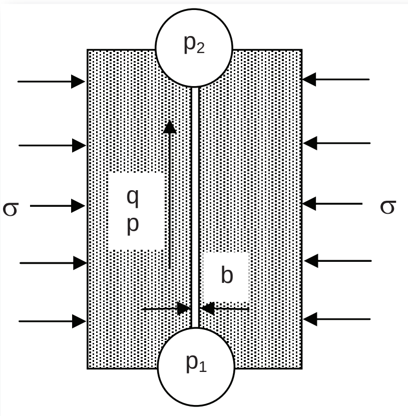


Fig. 5: Simple conceptual model of pathway flow in clay formation [6]

4.2.3 The Extended DP Model

The fracture model presented in section (4.1.2) is extended to include the deformation, ε by estimating the fracture aperture in the following way:

$$b = b_o + \Delta b$$
$$\Delta b = a\Delta\varepsilon = a(\varepsilon - \varepsilon_o) = (s/n)(\varepsilon - \varepsilon_o) \qquad (10)$$

Here, it has been assumed that deformation is localized and results in changes in aperture. A threshold value (ε_o) is considered. Therefore the changes in aperture start when deformation reaches this value. Deformation perpendicular to the fracture plane is used when aperture changes have to be obtained. The threshold value (ε_o) is associated with fracture initiation. This parameter will be set to zero if the fractures already exist and have an initial aperture b_o. In practice, the initial aperture can be essentially zero when the fractures exist but are closed. Introducing the expressions of fracture aperture to the Eq. (3) the element permeability of the system shown in Figure 4 is obtained as follows:

$$k = k_{matrix} + \frac{b^3}{12a} = k_{matrix} + \frac{(b_o + a(\varepsilon - \varepsilon_o))^3}{12a} \qquad (11)$$

The so-formulated permeability is used as the Darcy permeability in flow equations as function of elastoplastic deformation in clay formations. Some examples to show the capabilities of the formulation are presented in various studies [12,16,17]. Modelling attempts of the tests conducted to study gas flow through compacted clays and rocks were presented [12]. The analyses showed the sensitivity of results to a number of factors controlling the tests usually performed.

4.2.4 Transient Flow in a Deforming Poroelastic Medium

Terzaghi's theory is commonly used for the description of consolidation and compaction processes in saturated low-permeability soils (e.g. clay) due to extended load acting on such a medium. The deformation of a finite volume of porous medium involves the compressible behaviour of both the solid matrix and the pores. Changes in the volume can be induced by changes in the internal fluid pressure *p* or an externally imposed stress σ. The mathematical framework for the analysis of transient flow in a saturated, linear elastic, porous medium is provided by Biot's theory of three-dimensional consolidation [18]. A formulation of the theory for clay formations is made in [6]. Assuming that a porous medium is mechanically isotropic, relatively incompressible, and its volumetric strain ε_V, can be described by the linear thermoelastic constitutive equation which can be in turn introduced in Darcy Equation.

$$\varepsilon_V = -\frac{\Delta V}{V_0} = \frac{1}{K}(\sigma_m - \frac{K}{H}p_w) - \alpha_b(T - T_0) \qquad (12)$$

This modelling attempts, although addresses to one-phase liquid flow, gives a theoretical overview on the modelling of transient and steady-state advection in poroelastic medium as the case of clays can be applied for 2 phase (liquid-gas) flow too.

4.2.5 Capillary Bundle Model with Geomechanical Effects [19]

The model is the extention of the capillary bundle model to incorporate generic gas migration behaviour, in which pathways (bundles) are dynamically opened as the gas pressure exceeds a local threshold value, propagate thought the material and close if the gas pressure drops is based on generic formulation of gas pathway generation (GPG).

It is assumed that the gas moves in the positive direction and that the gas pathway is radially symmetric. The GPG model is formulated by considering the forces acting at the interface between the pathway (assumed to be filled with gas) and the transport medium, which includes granular material (for example clay particles) and free or bonded pore water. The position of the gas-medium interface is given by y=s(x,t). For this model the balance of the forces that need to be overcome if the gas-medium interface is to move, whilst the drag term controls the rate at which the gas-medium interface propagation occurs. In this balance the forces acting normal to the gas-medium interface that need to be overcome if the gas-medium interface is lumped as σ_t, threshold tensor.

These are for example capillary pressure and effective stress. The magnitude of the surface tension acting tangentially to the surface is also taken into account as force per unit length of interface. Departing from this basic momentum conservation approach, with the assumption of constant surface tension and simple diagonal forms for the drag and generalized threshold tensor, the equation of motion for the infinitesimal section at position r can be written as function of time. In a further step the equation of motion of gas flow in the pathway itself which are derived from the mass conversation, gas flow and the gas equation of state. Two specific cases namely capillary pressure dominated pathways and effective stress dominated pathways are studied using this model.

4.2.6 Crack Dilation Model [20]

A modelling approach was proposed to primary gas-oil migration in shales which can also be used especially the EDZ's in clay formations. The model takes the linear elastic fracture mechanics as the theoretical basis. The cracks are represented by ellipsoidal cavities with dimensions defined by semi-axes *a,b* and *c*. These parameters are regarded as suction-dependent variables. The cracks are arranged in a regular pattern. The scale is defined by a fixed characteristic distance, *L*. As a uniaxial case, an increase in suction can be represented by an equivalent radial tensile stress, σ, applied to the boundaries of the plug. If the distance between water molecules is expressed as δ_w, the crack propagation criterion can be relied to the gas entry phenomenon with the following expression:

$$\sigma = \frac{2\gamma_w}{\delta_w}\left(\frac{a}{c}\right) \tag{13}$$

affirming that gas can enter to the rock if the above given stress is reached. Note that in this equation γ_w can be taken as the interfacial tension of water-gas binary system. Analysis of the three-dimensional problem is quite complex and demands that the strain energy and the surface energy terms be expressed as functions of the semi-axes of the cracks. A possible relation of the effective gas permeability to the spatial density and dimensions of the cracks in the plane normal to gas flow is of primary importance.

4.2.7 Multiple Front Propagation Model [21]

As in the case of the implementation of the capillary bundle model described in the previous section, the "multiple front propagation model" has been designed to simulate laboratory gas migration experiments on cylindrical clay cores. This model is again based on the solution of mass conservation and Poisseuille flow equations for both fluids (wetting and non-wetting) and on the following equation for the advance through the clay of the tip of the propagating gas flow channel. The gas-filled pathways are assumed to dilate elastically for small displacements.

Each path *i* is constructed stochastically from a series of channel segments *j*. The number of paths is set by an input channel density $\psi_{channel}$ and the area *A* of the core end face. Propagation of gas fronts along channels is allowed when the inlet gas pressure p_i exceeds the input threshold p_{start} and the front velocity *v* is positive. The front velocity along path *i* and channel *j* is modelled as:

$$v_{i,j}(t_n) = \left(\frac{1}{\beta}\right)\left[p_i(t_n) - p_{out} - p_{tension}\frac{r_{0,\max}}{r_{0,i,j}}\right] \tag{14}$$

where β is the drag coefficient, $p_{tension}$ is a measure of the gas-gel surface tension. At any time t_n, a path with $v_{i,j}(t_n)<0$ is assumed to have stopped. Fronts are propagated during each time step. During propagation, they may emerge at the outlet core face (breakthrough) or stop because they encounter a channel that is too narrow for propagation at the current inlet and outlet pressures. As the pressure change, stopped fronts may resume propagation. The radius of gas-filled channels behind it is all assumed to be at pressure p_i. However after breakthrough, the pressure throughout the open path is implicitly a function of the position along the clay core.

4.2.8 Damage Induced Gas Transport

The implication of EDZ's on the transport processes is the dilatancy which is controlled by the mechanical disturbance and damage. The permeability induced by the dilatancy is the principal parameter representing the EDZ's in transport equations. The coupling of the permeability to the modelled dilatancy is investigated for various host rock types intensively. For crystalline and salt rocks as repository host rock, the dilatancy induced permeability, the geometry and the extent of the EDZ's are well studied [22]. In clay formations, the EDZ is also commented as a dilatancy related fracture opening and development phenomena. Principally, the concepts originated from crystalline host rocks can be applied to plastic and indurated clays too. However the EDZ studies for clay formations are not advanced enough to set a universal solution of coupling the changes on stress equilibrium on to the hydraulic properties. Some attempts were made to couple the permeability to the dilatancy empirically and relates this to the transient hydraulic diffusion such as:

$$t_h = \frac{r_i^2}{k_h}, \tag{15}$$

where t_h is the characteristic time for hydraulic diffusion, r_i is the excavation radius and k_h the hydraulic diffusivity defined by

$$k_h = \frac{k}{\rho_w g} M \frac{3K^{'} + 4G}{3K + 4G} \tag{16}$$

Using the typical mechanical parameters of Boom Clay, the value k_h = 1,21·10^{-7} m^2/s is obtained. Then, considering r_i = 4 m (excavated radius), the characteristic time for hydraulic diffusion can be found as t_h = 4,2 year, which is much larger than the times observed in-situ measurements [23].

Another model provides a typical example of accounting the notions of crack damage and double porosity [24]. The formulation is first obtained by simulation of a circular crack distribution system inside a REV (Representative Elementary Volume) and homogenisation at the macroscopic scale. The hydraulic behaviour of the rock is drawn from the classical Darcy law for the matrix permeability and Poiseuille law for the cracks. Percolation threshold phenomenon is observed by increasing the crack density. The anisotropic permeability tensor before and after the percolation threshold appears to fit to a non-linear law with the crack damage tensor. Extensive measurements on clay samples have allowed to valid the following power law between the trace of the permeability tensor and the trace of the crack damage tensor.

$$T(k) = \alpha\, T(D_c)^b \quad \text{with } \alpha \approx 10^{-14} \text{ and } \beta \approx 3$$

The above given crack damage law is derived from the mechanical damage law by projection onto the deviatoric plane. The model has been successfully implemented in Code_ASTER and Cesar for numerical modelling of the storage concepts.

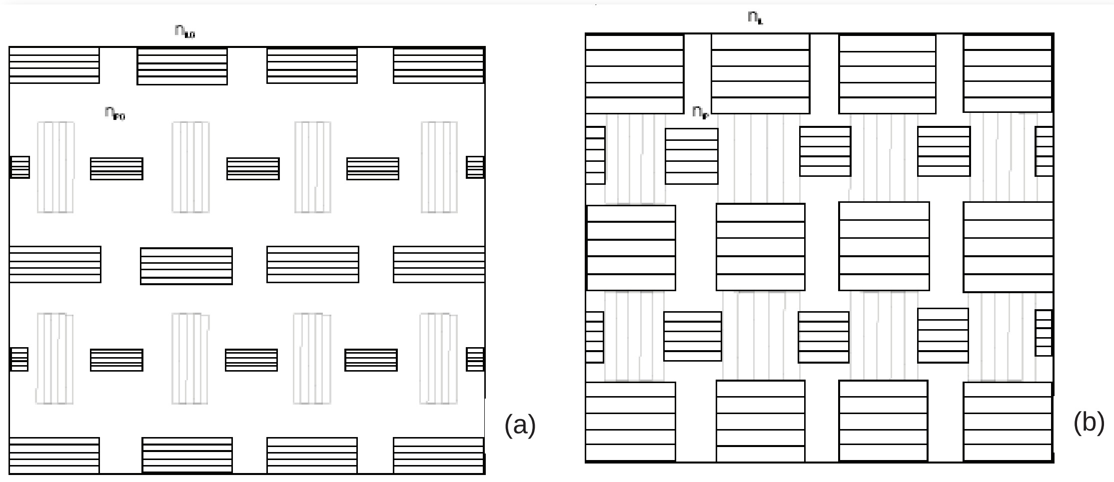
4.2.9 Two-phase Flow through a Deformable Porous Medium

In the framework of GAMBIT-GW project, a model was developed by considering the coupled mechanical deformation of and flow through a porous medium for the case in which only a single fluid phase is present, in practice water. The theory was originally developed by Biot [18]. The necessary extensions for Biot theory to be applicable to gas invasion of water-saturated bentonite are presented in [8].

For this formulation, the solid grains are assumed incompressible. The model consists of a force balance equations, which determine the equilibrium state of a small element of the porous medium and of the macroscopic stress-strain relations that are the generalisation of Equation Hooke Law for a porous medium with both a liquid phase and a gas phase. The change in water saturation is expressed as function of elastic deformation as well as with respect to the change in capillary pressure. The continuity equations for the pore fluids are given based on the Darcy velocities. Continuity equation for gas phase is generalised as combined equation for dissolved and free gas flow. The elastic moduli; poisson ratio and elastic modulus are taken constant from measured values. The functional form of other elastic parameters and permeability tensor are given for bentonite.

4.2.10 Coupling with Swelling [25]

An upscaling concept for swelling/shrinking phenomena in expansive porous media, which are triggered by the mineral composition of the material, is presented using a simplified structural model of bentonite [25]. The microscopic approach is based on the mineral structure and the diffusive double layer (DDL) theory. A constitutive equation to transform microscopic volume into macroscopic porosity changes is derived, which is the link between the two scales. Main processes during swelling/shrinking are hydraulic (water transport, re-saturation, de-saturation), chemical (particle-water-cation interaction) and mechanical ones (volume and porosity changes, swelling pressure), which are highly coupled. These processes are described at the macroscopic level. For modelling the swelling of bentonite which is in which montmorillonite is the main mineral for swelling, a simplified microstructural model of bentonite is created as shown in Figure With fluid intrusion a film of water/solution builds up around the surface of particles owing to the special structure of the net negative charge on the surface of the expansible minerals in bentonite (e.g. montmorillonite). In the meanwhile fluid can also be absorbed into the interlayer gaps of bentonite particles and results in expansion of the particles. Consequently the bentonite will swell macroscopically. If the volumetric expansion is confined, a swelling pressure appears. This mechanism is schematically illustrated in Figure 6.



▲ Fig 6: Simplified microstructural model of bentonite before and after saturation [25]

The number of layers *m* within one particle of expansive minerals in bentonite is a mineral structural parameter. Effective layer number is defined here as those layer numbers that contribute to the swelling effect. As the montmorillonite particles are very fine in size, the average value is used to calculate the interlayer porosity. So that it can be determined from the specific surface experiments as the ratio of maximal specific surface area S_{max} (including the surface between layers, which can separate by swelling) and external specific surface area $S_{external}$ (only the surface of particles). Thus the maximal interlayer porosity for one particle with m effective layers can be calculated:

$$\phi_{IL} = \frac{2m.A.\delta}{V_0}, \qquad (17)$$

where a simplification for the surface area *A* of the particle can be made assuming that the particles are circles with diameter δ.

Microscopic volume changes estimated based on above given model is translated into macroscopic changes of porosity. As a consequence, hydraulic properties of the porous media, such as capillarity and permeability, have to be affected as well. The concept of the swelling model for the first step of coupling is based on the microstructural models in Figure 6 and the Gouy-Chapman theory. For the Chemo-Hydraulic-Mechanic (CHM) model it is assumed that the total porosity ϕ can be divided into porosity between particles ϕ_{IP} and that within the particles or interlayer porosity ϕ_{IL}. It is assumed that for bentonite or bentonite/sand mixture ϕ_{IL} is a function of water saturation S_l and volume fraction of expansive minerals β:

$$\phi_{IL} = f(S_l).\beta.\phi_{IL\max} \tag{18}$$

For completing the model, a two-phase flow system is considered; the mass balance and fluid momentum balance equations are developed in a conventional way. For calculating the capillary pressure and relative permeability curves van Genuchten Model is proposed. After the discretization of the formula with finite differences approach and implementation in the software ROCKFLOW, the obtained model was validated using two experimental drying/wetting cases [25].

5 Conclusion

A summary of the studied conceptual and numerical models for gas transport in clay formations has been given. In basic, the transport of gases is modelled by diffusion and dispersion supported advection. Gas moves either dissolved in liquid phase or together with liquid phase. In natural clay systems the flow of one-phase gas is not expected. There are few open questions regarding on the phase behaviour of gas-liquid phase properties excepting support the equation of states with empirical parameters such as solubility of gases in water with various ionic activities. The main problem arises from two sides. The first one is the phsyico-chemical interactions in gas-water-clay interfaces. The second one is the geometrical coupling of the flow pathways coupled with mechanics. Both are key features for the formulation of the gas retention in transport models. More modelling efforts have been made for plastic clays than for indurated clays. Capillary bundle model or similar models coupled with geomechanics intent to conceptualize the preferential pathways introducing the capillary effects i.e. threshold pressure in two-phase flow. The validations of these models are reported to give sufficiently good results in matching the experimental data. The modelling of gas transport in indurated clays necessitates a better understanding of the fracture physics and its application in conventional and probably non-conventional flow formulations. If indurated clay contains already open fractures, then the problematic reduces to the flow in fractured geological media in which the petroleum industry offers a very big experience with network or continuum models with various concepts for taken the configuration of the fractures into considerations. Modelling of the gas transport, as two-phase or with diffusion supported one-phase flow in tight indurated clays necessitates the strong coupling with fracture mechanics to model induced fracturing.

As the case in other host rocks, the EDZ's should be taken in modelling efforts into consideration. The most important feature is the coupling of the hydraulic properties with mechanical changes. In clay formations the chemical coupling in modelling EDZ's is also important. The fracture mechanics considered in modelling the gas transport in near and far field of the indurated clay formations can also be used for EDZ calculations. However a validated hydraulic-mechanical-chemical coupling does not exist for clay formations. ■

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2A.04 Results on Pu Diffusion Experiments in the Opalinus Clay

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The Opalinus Clay (OPA) is a potential host rock for a repository for spent fuel, vitrified high-level waste and long-lived intermediate-level waste in Switzerland. Owing to its small hydraulic conductivity (10^{-14}-10^{-13} ms^{-1}), it is expected that transport of solutes will be dominated by diffusion. The diffusion is a very sensitive parameter in performance assessment (PA). The process is well understood for non-retarded solutes with simple chemistry, but little is known for retarded solutes. Therefore, the objective of this work is to understand the Pu-238 diffusion in clay mineral-rich geological formations in order to provide support for improved representation of these processes in performance assessment and to enhance safety case credibility.

A sample cell - autoclave system (SCAS) was developed for carrying out actinide diffusion experiments in clay stones under their natural, confining pressure. We started our investigation with sorption experiments of Pu-238 on the Opalinus Clay (OPA) in the OPA porewater. According to the results of the batch sorption data, a strong Pu-238 sorption under the experimental conditions of the diffusion experiments is expected. In our presentation we will compare the obtained diffusion data of Pu-238 with published data for non retarted solutes (e. g. HTO). The oxidation state of Pu-238 on the solids and in solution will be described. With different techniques (e. g. autoradiography, abrasive techniques, laser ablation) we will characterize the diffusion profiles within the OPA.

These ongoing investigations will provide the necessary basis for credible description of sorbing radionuclide mobility in clay for the nuclear waste disposal safety case. ■

2A.05 Efficiency of Clay Barriers

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Abstract

Clay is investigated in many countries as a potential host rock for radioactive waste repositories. The key feature of clay is its efficiency as a barrier to the flow of water and the migration of radionuclides. The barrier efficiency of a clay formation is governed by the barrier thickness, the permeability, the prevailing hydraulic gradient, the diffusion constant, the transport relevant porosity, and the radionuclide-specific sorption constants. Depending on the specific conditions, radionuclide migration in clay may be dominated either by diffusion or by advection. Therefore, the question arises, which combinations of the factors stated above constitute a "sufficiently efficient barrier". This question is addressed here by evaluating formerly completed safety analyses with a set of performance and safety indicators. The main conclusions are that (i) sorbing radionuclides are very efficiently confined by a wide range of barrier properties, (ii) non-sorbing long-lived radionuclides are only marginally confined by any realistic clay barrier, (iii) the relevant barrier effect for non-sorbing long-lived radionuclides is a delayed radionuclide release through the barrier at low rates, and (iv) a clay barrier can tentatively be assessed as "sufficiently efficient" if it provides a characteristic time of diffusive transport of about 1 mio. years or more and a characteristic time of advective transport of more than about 100'000 years.

1 Introduction

Clay is investigated in many countries as a potential host rock for radioactive waste repositories, among others in Belgium, France and Switzerland. In Germany, Lower Cretaceous clay forms the main geological barrier at the Konrad site, and clay formations are considered as potential alternative host rocks for the future radioactive waste repository.

The key feature of clay is its efficiency as a barrier to the flow of water and the migration of radionuclides. When starting to search for a new site on a "white map", this leads to two questions: (i) What are the "minimum" quantitative requirements with respect to the properties of a clay barrier to render a site suitable? (ii) How can two sites be compared without carrying out full-scale safety analyses?

Relevant properties of a clay barrier are the low permeability, the prevailing hydraulic gradient, the sorption capacity for many radionuclides, the small diffusion constant for anions, the thickness of the barrier, and the possibility to extrapolate data within the barrier over rather long lateral distances. The permeability and the diffusion constant generally show a substantial anisotropy.

The performance of specific sites with consolidated clay as host rock has been extensively analysed in [1] for the Opalinus clay at the Benken site in Switzerland and in [2] for a site with Callovo-Oxfordian argillites in France. The properties of the clay barriers investigated in these projects cover with the respective reference cases, the alternative cases and the "what-if"-cases broad ranges of conditions. The correlation between the properties of the clay barrier and the site performance (calculated doses) in [1] and [2] has been analysed in a research project [3] using different performance and safety indicators. The derived indicators allow to assess, in a structured manner, the performance of a clay barrier with deviating properties.

In this paper some of these indicators are described, and their values and limitations illustrated through the application to conditions with known system performance (Benken site of Nagra, Opalinus clay [1]). Finally, conclusions are drawn for a clay barrier with different properties.

2 Barrier Properties of Clay Barriers

Table 1 lists the properties of the clay barriers from the safety analyses in [1] and [2]. For comparison, the table also gives the properties of the clay barrier at the Konrad site as they were used in the safety analysis for the Konrad repository [4].

Table 1: Properties of clay barriers (reference values and investigated parameter ranges, incl. "what-if"-cases)

	Benken [1]	Bure [2]	Konrad [4]
Relevant transport distance [m]	40 (30 – …)	65 (… – 75)	100 (5)
Vertical groundwater flow (1) (Darcy velocity) [m/s]	2E-14 (… – 2E-12)	1E-14 (… – 4E-13)	5E-13 (… – 3E-12)
Hydraulic transmissivity (2) of (hypothetical) fracture zone [m²/s]	– (… – 1E-9)	–	n.a.
$D_e^{=}$ / $D_e^{\perp}$ (anions) (3) [m²/s]	5E-12 / 1E-12 (… – …/ 3E-12)	5E-12 (… – 1E-11)	1E-11
$D_e^{=}$ / $D_e^{\perp}$ (non-anions) (3) [m²/s]	5E-11 / 1E-11 (… – …/ 1E-10)	2.5E-10	1E-11
Porosity for anions [–]	0.06	0.05 (0.03 – 0.08)	0.15
Porosity for non-anions [–]	0.12	0.18	0.15
K_d (selenium) (4) [m³/kg]	0	0	5E-4
K_d (chlorine) (4) [m³/kg]	0	0	0
K_d (iodine) (4) [m³/kg]	3E-5 (3E-6 – 4E-4)	0 (… – 1E-3)	0
K_d (uranium) (4) [m³/kg]	20 (0.5 – 200)	8	0.02

(1) product of vertical hydraulic conductivity and vertical hydraulic gradient; (2) product of hydraulic conductivity and thickness; (3) D_e = effective diffusivity = product of pore diffusion constant and porosity with the indices "=" and "⊥" indicating the diffusivity parallel and perpendicular to the layering of the clay respectively; (4) element-specific sorption constant according to the K_d concept; (5) assumed here for the comparison in section 3.3

The parameter values in Table 1 disclose the difficulty encountered when attempting to compare – as an example – the barrier efficiency at the Benken site with that at the Konrad site.

3 Performance and Safety Indicators

3.1 Indroductory Definitions

The advective/dispersive and the diffusive transport of radionuclides through the clay barrier depend on the following physical parameters:

H transport distance [m]

v_D groundwater flow density (= Darcy velocity) [m/s]

n porosity [–]

D_e effective diffusivity [m²/s] parallel to layering ($D_e^{=}$) and perpendicular to layering ($D_e^{\perp}$)

a_L longitudinal dispersion length [m]

R element-specific retardation factor [–]

with

$$R = 1 + \frac{\rho_B \cdot K_d}{n} \tag{1}$$

K_d element-specific sorption constant [m³/kg]

ρ_B bulk rock density [kg/m³]

The well-known transport equation for decaying species reads for vertical transport through the barrier, i.e., perpendicular to the layering of the clay (v_D = vertical Darcy velocity):

$$n \cdot R \cdot \frac{\partial C}{\partial t} = D' \cdot \frac{\partial^2 C}{\partial x^2} - v_D \cdot \frac{\partial C}{\partial x} - \lambda \cdot n \cdot R \cdot C \tag{2}$$

$$D' = D_e^{\perp} + a_L \cdot v_D \tag{3}$$

C radionuclide concentration in the pore water [Bq/m³]

λ decay constant [s⁻¹]

The transport process is characterised by the following derived parameters:

$$t_d = \frac{H^2 \cdot n \cdot R}{D_e^{\perp}}$$ characteristic time of diffusive transport (4)

$$t_a = \frac{H \cdot n \cdot R}{v_D} \qquad \text{characteristic time of advective transport} \qquad (5)$$

$$\alpha_d = \frac{D_e^{\perp}}{H \cdot v_D} = \frac{t_a}{t_d} \qquad \text{inverse diffusive Peclet number} \qquad (6)$$

$$\alpha_L = \frac{a_L}{H} \qquad \text{inverse dispersive Peclet number, here taken to be} = 0.1 \quad (7)$$

These introductory definitions of physical parameters and derived characteristic parameters are the basis for the definition and evaluation of a set of performance and safety indicators for the transport efficiency of a clay barrier. This is done in the following section.

3.2 Definition of Indicators and their Application

Performance indicator F1: Radionuclide retention by the geological barrier under steady state conditions
The performance indicator F1 is a measure of the decay of a radionuclide species during its transport through the barrier under steady state conditions. Given a constant radionuclide concentration C_0 at the interface between the engineered barrier and the host rock, the steady state radionuclide concentration at distance H, C(H), can be calculated using equation (2). The indicator F1 is defined as the ratio $C(H)/C_0$ and it is fulfilled, if this ratio is smaller than a given criterion ε. With the downstream boundary condition being equivalent to an infinitely extended host rock, the result is

$$\frac{C(H)}{C_0} \leq \varepsilon \quad \Leftrightarrow \quad \lambda \cdot t_d = \frac{t_d \cdot \ln(2)}{T_{1/2}} \geq -\frac{1}{\alpha_d} \cdot \ln\varepsilon + \left(1 + \frac{\alpha_L}{\alpha_d}\right) \cdot (\ln\varepsilon)^2 \qquad (8a)$$

$$\Leftrightarrow \quad \lambda \cdot t_a = \frac{t_a \cdot \ln(2)}{T_{1/2}} \geq - \cdot \ln\varepsilon + (\alpha_d + \alpha_L) \cdot (\ln\varepsilon)^2 \qquad (8b)$$

Figure 1 shows the performance indicator F1 for different values of ε. For $\alpha_d > \sim 1$ the almost vertical shape of the curves illustrates that the transport is dominated by diffusion, that the barrier efficiency depends mostly on the ratio $t_d/T_{1/2}$ and that the Darcy velocity has little impact. For such conditions the characteristic time of diffusive transport t_d must exceed the half life time $T_{1/2}$ of the radionuclide by more than a factor 30 for the barrier to retain 99% of the radionuclides. For advection dominated conditions ($\alpha_d = 0.1 = \alpha_L$, say) the characteristic time of advective transport t_a must exceed the half life time of the radionuclide by a factor 14 in order for the barrier to retain 99% of the radionuclides according to equation (8b).

The red crosses in Figure 1 correspond to the characteristic values of the clay barrier at the Benken site for reference conditions according to [1]. The transport is clearly diffusion dominated for non-anions ($\alpha_d \approx 10$) and – due to the smaller diffusion constant – weakly diffusion dominated ($\alpha_d \approx 1$) for anions. The non- or weakly sorbing long-lived anions (^{129}I, ^{79}Se and ^{36}Cl, the latter not shown in Figure 1) do not fulfil the performance indicator F1, i.e. they are not substantially confined by the geological barrier under steady state conditions. The uranium isotopes, on the other hand, do meet the indicator (Figure 1 shows the uranium isotope with the smallest ratio $t_d/T_{1/2}$). However, it must be noted that steady state conditions are attained for strongly sorbing radionuclides only after extremely long times which are beyond any prognostic range. The performance indicator F1 therefore is a valuable indicator for weakly sorbing or medium long-lived radionuclides. Of course, it cannot comprehensively assess the suitability of a geological barrier.

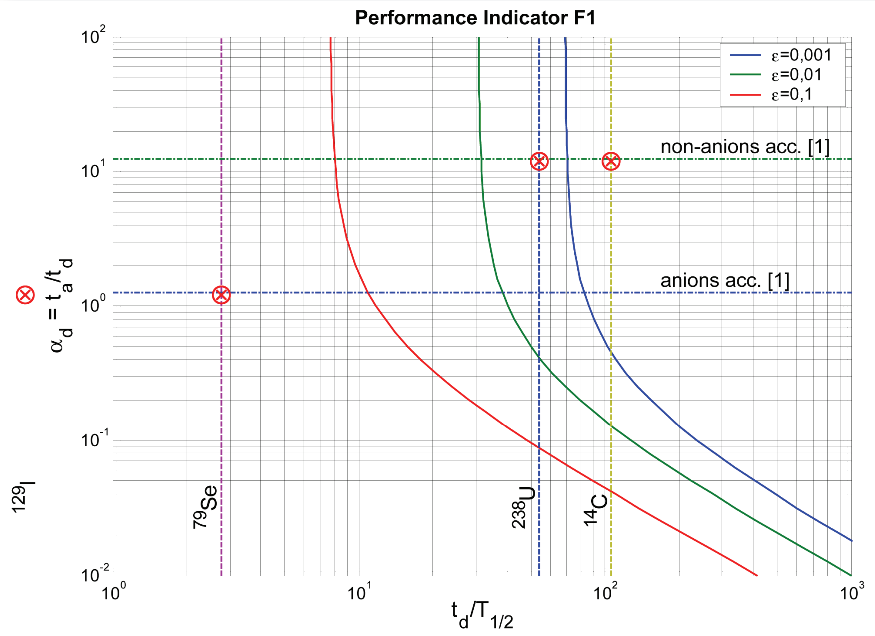

Fig 1: Performance indicator F1 for different values of ε. The indicator is fulfilled for points on the right hand side of the corresponding curve. The red crosses show the performance of the clay barrier at the Benken site for reference values.

Performance indicator F2: Radionuclide retention for continuous release from the repository under transient conditions

The performance indicator F2 is a measure of the efficiency of the geological barrier to retain radionuclides for a predefined time period, the assessment time period (0, t_{BZ}). It is defined as the maximum of the ratio of the radionuclide concentration at distance H from the repository and at time t (0<t<t_{BZ}), C(H,t), and the initial radionuclide concentration at the interface engineered barrier / host rock, C_0, with the following boundary conditions: C(x=0,t) = $C_0 \cdot \exp(-\lambda t)$ and an infinitely extended host rock. The indicator F2 is fulfilled, if the maximum of this ratio is smaller than a given criterion ε.

Solving equation (2) with these boundary conditions yields for the performance indicator F2:

$$\max_{0<t<t_{BZ}}\left(\frac{C(H,t)}{C_0}\right) = \max_{0<t<t_{BZ}}\left\{e^{-\lambda t}\cdot\frac{1}{2}\cdot\left[\operatorname{erfc}(\beta_-(t)) + \exp\left(\frac{1}{\alpha_d+\alpha_L}\right)\cdot\operatorname{erfc}(\beta_+(t))\right]\right\} \leq \varepsilon \tag{9a}$$

$$\beta_\pm(t) = \frac{1}{2}\cdot\left(1+\frac{\alpha_L}{\alpha_d}\right)^{-\frac{1}{2}}\cdot\sqrt{\frac{t_d}{t}}\cdot\left(1\pm\frac{t}{\alpha_d\cdot t_d}\right) \tag{9b}$$

For long-lived radionuclides the concentration ratio will attain its maximum value at the end of the assessment time period and the performance indicator F2 corresponds to the ratio C(H,t_{BZ})/C_0. Figure 2 shows the performance indicator F2 for extremely long-lived radionuclides.

Figure 2 shows that for diffusion dominated transport the characteristic time of diffusive transport must exceed the assessment time t_{BZ} by a factor of about 15 for the geological barrier to retain 99% of the corresponding radionuclide for this time period.

The strongly sorbing elements uranium and neptunium have characteristic times of diffusive transport of more than 100 mio. years so that their isotopes clearly fulfil the performance indicator F2. Again the non- or weakly sorbing long-lived anions fail. Figure 2 shows that the indicator F2 rises to about 0.2 for ^{79}Se (including the effect of radioactive decay) and to about 0.1 for ^{129}I after 1 mio. years at the Benken site for reference parameters (α_d = 1.25).

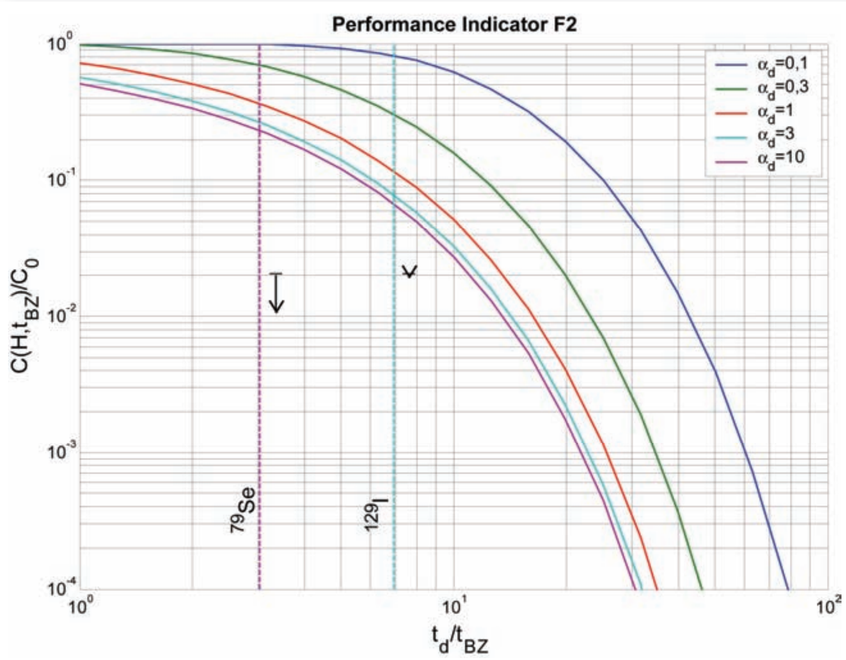

Fig 2: Performance indicator F2 for different values of α_d and for extremely long-lived radionuclides ($T_{1/2}$ >> t_{BZ}). The arrows show the effect of radioactive decay for ^{79}Se and ^{129}I for t_{BZ} = 1 mio. years.

Performance indicator F3: Radionuclide retention for delta-peak release from the repository

The performance indicator F3 is a measure of the cumulative radionuclide release from the geological barrier during the assessment time period for a delta-peak release from the repository at the interface engineered barrier / geological barrier. The performance indicator F3 is the ratio of the time integrated advective / dispersive / diffusive radionuclide flux j at distance H from the repository and the total radionuclide inventory released from the repository I_0. The indicator is fulfilled if this ratio is smaller than a given criterion ε.

Solving equation (2) for a delta-peak release at x=0 and an infinitely extended host rock yields

$$\frac{1}{I_0}\cdot\int_0^{t_{BZ}} j(H,t)\,dt = \frac{1}{\sqrt{4\pi}}\cdot\int_0^{t_{BZ}}\frac{1}{t}\cdot e^{-\lambda\cdot t}\cdot\beta_+(t)\cdot\exp\left(-\beta_-^2(t)\right)dt \le \varepsilon \qquad (10)$$

Figure 3 shows the performance indicator F3 for long-lived radionuclides ($\lambda\cdot t_{BZ}$ << 1) for different values of α_d. The performance of the geological barrier as indicated by the performance indicator F3 is very similar to those of indicator F2 (Figure 2): A retention efficiency of 99% requires for diffusion dominated transport a characteristic time of diffusive transport t_d of more than about $15\cdot t_{BZ}$. Like for the performance indicator F2 this is fulfilled by the strongly sorbing radionuclides and not fulfilled by the non- or weakly sorbing anions ^{79}Se and ^{129}I.

Spent fuel releases a certain percentage (between 4% and 15%) of activation and fission products like ^{79}Se and ^{129}I within a short period of time after canister breaching (instant-release-fraction). Figure 3 shows that about 20% of ^{79}Se (neglecting the effect of radioactive decay) and about 6% of ^{129}I of the instant-release-fraction will be released from the geological barrier within the assessment time period at the Benken site for reference conditions (α_d = 1.25).

In order to assess the potential consequences of such releases into the biosphere the safety indicator S1 is defined below.

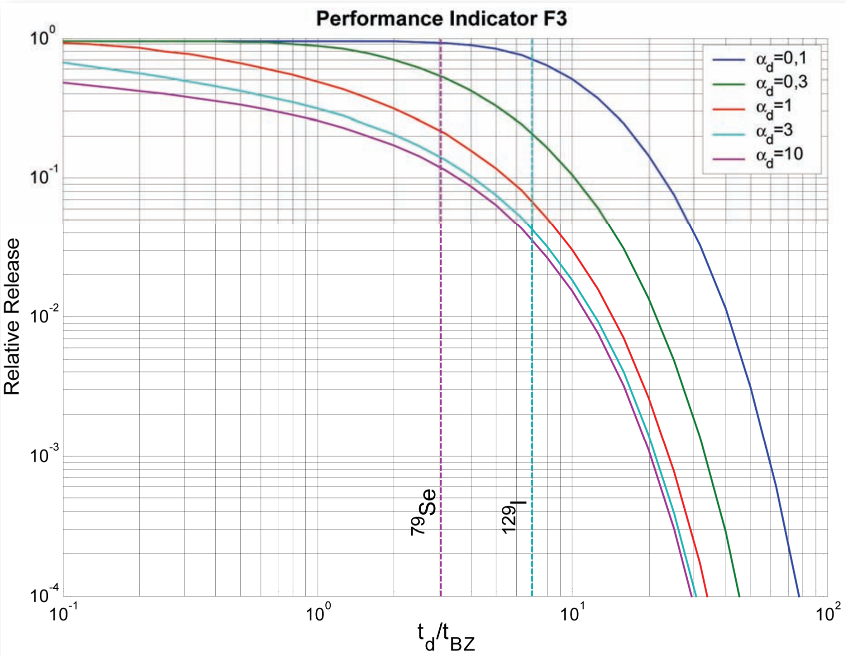

Fig 3: Performance indicator F3 for different values of α_d and for very long-lived radionuclides ($T_{1/2} >> t_{BZ}$).

Safety indicator S1: Approximate dose from maximum annual radionuclide release from the geological barrier as a consequence of a delta-peak release from the repository

Given the total radionuclide inventory I_0 released within a short period of time from the engineered barriers of the repository, the near-surface groundwater flow Q_{Bio} which determines the dilution of the radionuclides released from the geological barrier, and a representative dose-conversion factor DKF which quantifies the annual dose rate as a function of the radionuclide concentration in the near-surface groundwater, the contribution of the maximum annual release of the radionuclide to the annual dose D within the assessment time period is

$$D = \frac{I_0 \cdot DKF}{Q_{Bio}} \cdot f_{r,max} \qquad (11a)$$

$$f_{r,max} = \max_{0<t<t_{BZ}}\left(\frac{j(H,t)}{I_0}\right) = \frac{1}{\sqrt{4\pi}} \cdot \max_{0<t<t_{BZ}}\left(\frac{1}{t} \cdot e^{-\lambda \cdot t} \cdot \beta_+(t) \cdot \exp\left(-\beta_-^2(t)\right)\right) \qquad (11b)$$

Figure 4 provides the maximum relative release rate $f_{r,max}$ from the geological barrier for different values of α_d, for the assessment time period t_{BZ} of 1 mio. years and for very long-lived radionuclides. For diffusion dominated transport the maximum relative release rate $f_{r,max}$ decreases inversely proportional with the characteristic time of diffusive transport t_d up to about $t_d \approx$ 10 mio. years. Under these conditions, the radionuclide concentration peak traverses the geological barrier within the assessment time period and the decrease of the maximum relative release rate is a consequence exclusively of the diffusive (and dispersive) broadening of the radionuclide concentration peak. Only if the characteristic time of diffusive transport exceeds about 10 mio. years, the radionuclide concentration peak is substantially confined within the geological barrier for the assessment time period of 1 mio. years and the maximum relative release rate drops off steeply with increasing t_d/t_{BZ}.

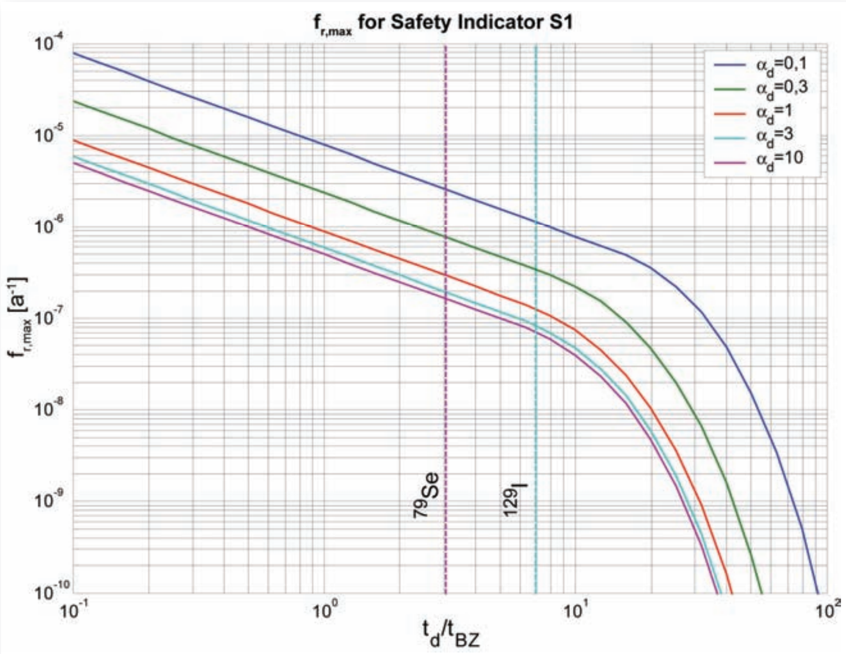

Fig 4: Maximum relative release rate $f_{r,max}$ [a^{-1}] for different values of α_d, for the assessment time period t_{BZ} = 1 mio. years and for very long-lived radionuclides ($T_{1/2} >> t_{BZ}$).

If the safety indicator S1 is applied to the conditions at the Benken site, the estimated dose according to equation (11a) is about a factor of 3 smaller than the result of the model calculations in [1]. However, these model calculations include the contributions of ^{79}Se and ^{129}I release from the spent fuel which is not part of the instant-release-fraction and which is, therefore, not included in the safety indicator S1. The fairly good correspondence between the safety indicator S1 and the model results in [1] illustrates that the safety indicator S1 is reasonable to grossly assess the effect of the release of non- or weakly sorbing radionuclides from the geological barrier.

The performance indicators F1, F2 and F3 and the safety indicator S1 constitute a set of indicators which allows to assess the efficiency of an intact clay barrier, i.e. of a barrier with no steeply inclined fracture zone or other feature with an enhanced permeability. In order to estimate the potential effect of a vertical fracture zone with transmissivity T_D crossing the host rock from the aquifer below to the aquifer above the repository, two alternative performance indicators F4 and F4" are defined. A fracture zone is of minor relevance for the transport of a radionuclide species if one of the two indicators is fulfilled.

Performance indicator F4: Radionuclide retention in a fracture zone for continuous release from the repository under transient conditions (relevant for sorbing radionuclides)

The performance indicator F4 is a measure of the retention of a radionuclide species during its transport along a fracture zone, including matrix diffusion from the fracture zone into the adjacent rock matrix. It is defined as the maximum of the ratio of the radionuclide concentration in the fracture zone at distance H from the repository, $C_D(H,t)$, within the assessment time period and the initial radionuclide concentration at the interface engineered barrier / fracture zone, C_0, with the same boundary conditions as for indicator F2. The indicator F4 is fulfilled if the maximum of this ratio is smaller than a given criterion ε. Obviously, this indicator is only applicable to those radionuclides that fulfil the performance indicator F2 for an undisturbed geological barrier, i.e. for sorbing radionuclides.

Using an approximate closed solution for the effect of matrix diffusion [3], the performance indicator F4 can be written as

$$\max_{0<t<t_{BZ}}\left(\frac{C_D(H,t)}{C_0}\right) = \max_{0<t<t_{BZ}}\left\{e^{-\lambda\cdot t}\cdot \mathrm{erfc}\left(\frac{H\cdot k_{WG}}{T_D\cdot v_D}\cdot\sqrt{\frac{D_e^{=}\cdot R\cdot n}{t}}\right)\right\} \leq \varepsilon \qquad (12a)$$

k_{WG} vertical hydraulic conductivity of the clay barrier [m/s]

T_D hydraulic transmissivity of the fracture zone [m²/s]

and with Ω as a dimensionless quantity for the transmissivity of the fracture zone

$$\Omega = \frac{T_D}{H \cdot k_{WG}} \cdot \sqrt{\frac{D_e^{\perp}}{D_e^{=}}} \tag{12b}$$

$$\max_{0<t<t_{BZ}} \left(\frac{C_D(H,t)}{C_0} \right) = \max_{0<t<t_{BZ}} \left\{ e^{-\lambda \cdot t} \cdot \operatorname{erfc}\left(\frac{\alpha_d}{\Omega} \cdot \sqrt{\frac{t_d}{t}} \right) \right\} \leq \varepsilon \tag{12c}$$

For very long-lived radionuclides the factor for the radioactive decay can be neglected and the maximum value of the concentration ratio occurs at the end of the assessment time period. Using $\varepsilon = 0.01$ as a criterion, the argument of the complementary error function must then exceed 1.82, i.e.

$$\Omega \leq \frac{\alpha_d}{1.82} \cdot \sqrt{\frac{t_d}{t_{BZ}}} \qquad \text{for } T_{1/2} >> t_{BZ} \text{ and } \varepsilon = 0.01 \tag{12d}$$

Uranium at the Benken site has a t_d/t_{BZ} ratio of roughly 240'000 when using reference values. Inserting this into equation (12d) gives (not rounded) $\Omega \leq$ 3'365 and, with equation (12b) and $k_{WG} = 2 \cdot 10^{-14}$ m/s, $T_D \leq 6 \cdot 10^{-9}$ m²/s as uranium-specific criterion for the fracture zone. Non-anions with medium strong sorption (R = 1000) are characterized by $t_d/t_{BZ} = 600$. For a fracture zone to be of negligible importance they require $\Omega \leq$ 168 (not rounded) which corresponds to $T_D \leq 3 \cdot 10^{-10}$ m²/s.

Performance indicator F4": Radionuclide release rate along the fracture zone is smaller than the release rate through a given area of the intact clay barrier (relevant for non- or weakly sorbing radionuclides)

The performance indicator F4" requires that the radionuclide release rate along a fracture zone of horizontal length L should not exceed the radionuclide release rate through an area of size L·X of the intact clay barrier, X being a given criterion. For ease of notation the performance indicator F4" is applied to very long-lived radionuclides and evaluated at the end of the assessment time period only. Considering the typical area of a nuclear waste repository, X is selected to be 1000 m.

Using the dimensionless quantity Ω for the transmissivity of the fracture zone defined by equation (12b), the performance indicator F4" becomes [3]

$$\Omega \leq \Omega_G \tag{13a}$$

with Ω_G being the solution of

$$\frac{H}{X} \cdot \sqrt{\frac{D_e^{=}}{D_e^{\perp}}} \cdot \Omega_G \cdot \operatorname{erfc}\left(\frac{\alpha_d}{\Omega_G} \cdot \sqrt{\frac{t_d}{t_{BZ}}} \right) = \frac{1}{2} \cdot \operatorname{erfc}(\beta_-(t_{BZ})) + \sqrt{\frac{\alpha_d \cdot (\alpha_L + \alpha_d)}{\pi}} \cdot \sqrt{\frac{t_d}{t_{BZ}}} \cdot \exp\left(-\beta_-^2(t_{BZ})\right) \tag{13b}$$

The solution of equation (13b), Ω_G, is the maximum value of Ω for which the fracture zone fulfils the performance indicator F4". Figure 5 shows the solution Ω_G for different values of α_d. The factor on the l.h.s of equation (13b) is chosen to be which corresponds to X = 1000 m at the Benken site.

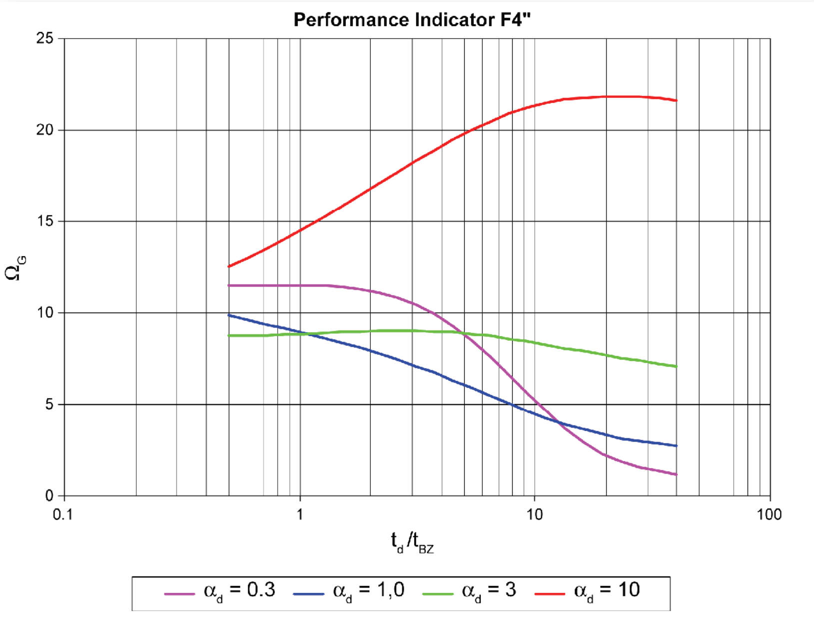

Fig 5: Upper limit of the dimensionless quantity for the fracture transmissivity Ω according to the performance indicator F4" for the criterion $\sqrt{D_e^+/D_e^-}\cdot X/H = 11.2$ corresponding to X = 1000 m at the Benken site.

The anions ^{79}Se and ^{129}I are characterized by α_d = 1.25 and t_d/t_{BZ} = 3 and 7, respectively, when using the reference values of the Benken site. According to Figure 5, a fracture zone which fulfils the performance indicator F4" must therefore obey $\Omega \leq \Omega_G \approx 7$ which corresponds to $T_D \leq 1.3\cdot10^{-11}$ m²/s. This compares fairly well with the results in [1] which calculated enhanced dose rates compared to the reference case for "what-if" cases with a hypothetical water-conducting discontinuity with TD = 10^{-10} m²/s or more.

3.3 Assessment of the Barrier Efficiency of 100 m Clay with the Properties Assumed in the Safety Report of the Konrad Site

The properties of the clay barrier which were assumed in the safety report of the Konrad site [4] are listed in Table 1. The corresponding characteristic values for the application of the performance and safety indicators are given in Table 2.

Table 2: Characteristic values of a 100 m clay barrier with the properties assumed in the safety report of the Konrad site [4]

	^{238}U	^{79}Se	^{129}I	^{36}Cl
Half life time $T_{1/2}$ [a]	4.47E9	1.10E6	1.57E7	3.00E5
Retention factor R [–]	300	8.4	1	1
Characteristic time of diffusive transport t_d [a]	1.4E9	4.0E7	4.8E6	4.8E6
$t_d/T_{1/2}$ [–]	0.3	36	0.3	16
t_d/t_{BZ} (1) [–]	1'400	40	4.8	4.8
Characteristic time of advective transport t_a [a]	2.8E8	8.0E6	9.5E5	9.5E5
Inverse diffusive Peclet number α_d [–]	0.2			

(1) with the assessment time period (0, $t_{BZ} = 10^6$ a)

Transport through the clay barrier is weakly advection dominated ($\alpha_d = 0.2$) with diffusion dominating over dispersion if, again, it is assumed that $\alpha_L = 0.1$.

According to the performance indicator F1 none of the listed radionuclides is confined to nearly 100% by the barrier under steady state conditions (Figure 1). ^{79}Se has the best score with a retention efficiency of somewhat more than 90%. But, as was mentioned earlier, the failure of uranium to satisfy the indicator F1 is meaningless because of the extremely long equilibration time for this element.

According to the performance indicator F2 uranium is, however, fully retained by the barrier for the assessment time period of 1 mio. years (Figure 2). Also, ^{79}Se is held back by almost 99.9% due to its (weak) sorption in the barrier. This result does not depend on the release mode from the engineered barriers since it follows from the performance indicator F2 (for constant release) and also from the performance indicator F3 (for delta-peak release, Figure 3). The non-sorbing radionuclides ^{129}I and ^{36}Cl are merely marginally confined by the barrier.

According to Figure 4, the maximum relative release rate from the clay barrier for non-sorbing radionuclides which were released from the engineered barrier within a short period of time (instant-release-fraction) is somewhat less than $f_{r,max} = 10^{-6}$ a^{-1}. The ^{129}I inventory in the instant-release-fraction in a future German repository for heat generating waste is estimated to be about $1.5 \cdot 10^{12}$ Bq. With a diluting near-surface groundwater flow of Q_{Bio} = 100’000 m^3/a and a dose conversion factor of DKF = $5.6 \cdot 10^{-7}$ (Sv/a)/(Bq/m^3) the safety indicator S1 gives an ^{129}I contribution to the dose of 0.008 mSv/a (equation (11a)). The result leads to the conclusion that a 100 m clay barrier with the properties assumed in the safety report of the Konrad site is also sufficient for non-sorbing anionic fission products.

The potential influence of a vertical fracture zone through the clay barrier is assessed with the performance indicators F4 and F4”. For a significant retention ($\varepsilon = 0.01$) of sorbing uranium on its transport along the fracture zone, the performance indicator F4 (equation (12c)) requires the dimensionless quantity for the transmissivity of the fracture zone to be $\Omega < 4$. The performance indicator F4” for non-sorbing radionuclides postulates $\Omega < 8$ (Figure 5, curve $\alpha_d = 0.3$) and is, therefore, less demanding. With $k_{WG} = 10^{-11}$ m/s and an isotropic effective diffusivity the criterion $\Omega < 4$ corresponds to a fracture transmissivity of $T_D < 4 \cdot 10^{-9}$ m^2/s according to equation (12b).

4 Conclusions

Consolidated clay is a very effective barrier for most of the safety relevant radionuclides. Its performance depends on the barrier thickness, the water flow rate through the barrier, the diffusion constant of contaminants, the transport relevant porosity, and the radionuclide specific sorption constants. Diffusion constant and transport relevant porosity may differ for anions and non-anions. Depending on the specific conditions, the transport of contaminants through the barrier may be dominated either by diffusion or by advection.

The performance of consolidated clay barriers has been demonstrated in several safety analyses for different sites. Instructive examples are those conducted by Nagra for a site with Opalinus clay at Benken in Switzerland [1], those done by Andra for a site with Callovo-Oxfordian argillites [2] and those for the Konrad site with Lower Cretaceous clay constituting the geological barrier [4]. These safety analyses have been evaluated using a set of performance and safety indicators with the objective to extract “necessary and sufficient” barrier properties [3]. A part of the evaluation has been presented in this paper. It is clear that rigorous conclusions can only be drawn from detailed model calculations and for a specific waste inventory. Yet, the application of a reasonable set of quantitative indicators has been shown to allow the derivation of some indicative concluding statements.

Important conclusions from the evaluation of formerly completed safety analyses and from the application of a set of performance and safety indicators are:

- The efficiencies of the clay barriers at the sites listed above are well above “critical limits”, i.e. all barriers bear safety reserves.

- Due to their strong sorption capacity, clay barriers confine sorbing radionuclides very effectively.
- However, long-lived, non- or weakly sorbing radionuclides like ^{129}I, ^{36}Cl and, possibly, ^{79}Se are not significantly confined by a clay barrier for the assessment time period of a safety analysis, which is typically about 1 mio. years. For these radionuclides the barrier effect is reflected in a delayed release from the barrier at low rates.
- Clay barriers are considered to be sufficiently effective if they provide a characteristic time of diffusive transport for non-sorbing radionuclides of about 1 mio. years or more and a characteristic time of advective transport of more than about 100’000 years.
- Vertical, hydraulically active discontinuities through the clay barrier (e.g. fracture zones) are of minor importance if their hydraulic transmissivity is less than about 10^{-10} m^2/s at sites with a relatively high vertical hydraulic gradient (about 1 m/m, situation at the Benken site) and less than about 10^{-8} m^2/s at sites with a low vertical hydraulic gradient of a few cm/m (situation at the Konrad site). ■

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2A.06 Self Closure Mechanisms for Underground Waste Repositories

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Abstract

The concept of phased geological repositories for radioactive waste, as under study in variants in many countries, implies that the repository is to some degree accessible for longer times, possibly up to some decades or even centuries. This increases the vulnerability to external events, as societal collapse due to natural or man-made catastrophes, leading possibly to a temporary or permanent loss of control over the repository. Depending on the phase of operation the repository is in at the time of the disruptive event, waste may be more or less accessible and the lost control over the repository can or can not be regained. As the nature of the loss of control can not be anticipated, one has to plan for all possible scenarios, which hints strongly at passive mechanisms as appropriate design and operation procedures to ensure a minimal vulnerability at any time to both short term interruptions and long term changes in the general environment. Self-closure has consequently to be understood, in our view, not only as the possibility to close rapidly some part of the access tunnels, but is a concept to be integrated from the beginning in the repository design and extends to the long-term management of the repository site.

1 Introduction

1.1 Definition

The idea of self-closure of repositories for radioactive waste is an often cited but rather elusive concept (e.g. EKRA, see [7]) meaning to ensure that in case of sudden inability of the society to further operate the repository the facility tends itself, with minimal or even without any further external intervention, towards a state that can be considered as safe or at least as significantly more so than an open, abandoned repository.

1.2 Scope

This paper tries to develop, from a viewpoint of safety, the main reasons in favour of including self-closure mechanism in the repository design and the main requirements that have to be met by those mechanisms and to give directions of further work needed.

1.3 Generic Repository Concept

We concentrate on the swiss situation concerning basic repository concept and boundary conditions, i.e. a repository for SF / HLW in a low permeability clay formation roughly 600m below the surface. It will be assumed that a phased repository concept will be applied, i.e. a operational phase of some decades during which the waste is emplaced will be followed by a period for monitoring and retrivability implying still administrative action to ensure safety, in particular site control, maintenance of equipment and active repository management such as preventing groundwater intrusion from higher geological layers or verification of structural stability (see [7]). This period is initially planned but further generations have the option of changing the schedule and reporting any final closure activity until they judge the time appropriate to undertake such action. This means essentially that this period may last at least several decades or even 100 years but may have, from a today's point of view, virtually any duration. This period of accessibility is likely to increase the vulnerability of the repository to external events. Once the decision for closure of the repository is taken, the necessary actions will come under way and the repository will be finally sealed within more or less a decade. After closure, monitoring of the repository may continue, but no further action is necessary to ensure safety in compliance with the safety case.

2 Societal Breakdown

To motivate the need for such a thing as a self-closure mechanism, a small excursion in social science and archaeology seems necessary. It is well known that in the past relatively advanced societies have completely disappeared, leaving behind a new societal structure significantly less complex, less rich and less skilled than their predecessors. The Roman Empire or Maya culture are only the best known examples. In fact, collapse of societies seems to be rather frequent, to the point that Tainter [1], who lists about 20 examples, states "…Collapse is a recurrent feature of human societies…".

Without entering the discussion why such collapse occurs, in the context of the present discussion the timescale of the process and the consequences and scale of societal disruption are here of interest. In fact, to present a threat for repository safety, the disruption in society must occur in a way that generates:

- Loss of knowledge on repository maintenance and closure
- Loss of interest in maintenance and closure
- Loss of necessary resources to protect the repository during the monitoring period
- Loss of necessary resources for appropriate closure

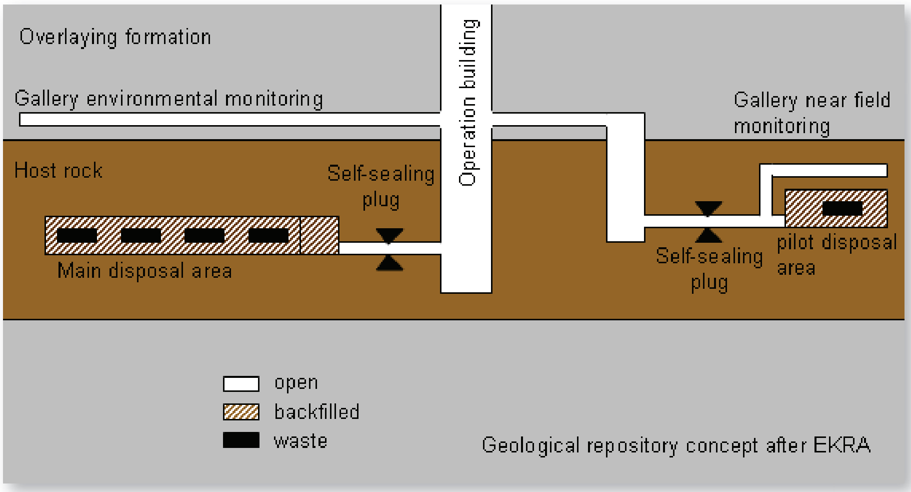

Fig 1: EKRA repository concept ([7])

Analysis of numerous societal breakdowns in past shows that in most cases several of the above conditions would have been met, and thus that it is at least possible that control of the repository will be lost. Admittedly the cases studied in [1] and [2] cover necessarily societies in the sometimes distant past and consequently at a very different level than todays. Nevertheless it is to be expected that the general features still apply. It seems recurrent that collapsing societies are subject to stress on resources and unable to maintain the level of public activity they where used to. This leads to cuts in activities that are regarded as not directly vital, and repository closure could very fall into this category. The timescale of decline is however generally decades to centuries, so plenty of time is available for corrective actions.

Examples of breakdowns occurring in shorter times are not mentioned in the above cited literature, however the fall of the former soviet union shows that a very rapid decomposition of governmental structures is possible and the challenges in modern society as widespread (nuclear) war, drastic climatic change or large scale economic crises, that would affect the planet as a whole are problems ancient societies where not faced with, at least not on the level possible today. In fact, modern interconnected societies are likely to be more vulnerable to rapid spread of crises and diseases (e.g. due to air travel) than this used to be the case in the past. A recent report by the US department of defence on effects of climatic change [5] establishes scenarios implying major societal disruption in the timescale of years.

Even a superficial analysis of existing literature shows that societal breakdown has occurred and will occur in the future, that developed societies are in no way immune to it and that it would, once occurring, lead very likely to a loss of control, during a specific time or permanently, of the repository. So at least the possibility of such an event has to addressed and if possible measures to minimize its impact on the repository will have to be taken.

3 Disruption Scenarios

Several configurations of disruptions, leading to a short or long term loss of the ability to maintain the site are possible (even though unlikely).

3.1 Possible Characteristics

Existing literature on societal breakdown cites a multitude of possible mechanism and forwards numerous hypotheses on the causes and underlying processes. However, there seems to be no consensus on the causes for even such a well studied case as the Roman Empire, far less an agreed on basic process that would lead to the decline of a given society. As mentioned above, this is here not really of relevance, it is sufficient to accept the fact that breakdown occurs. However, it would be helpful to have an idea on the processes going on in order to be able to design countermeasures efficiently.

Here the observation is helpful that social collapse is often associated with some recurrent effects on the system. Authors disagree whether these effects are the cause, triggers or even the consequence of the disintegration, but they are probably anyway the relevant characteristics. These effects include

- Resource depletion
- Catastrophes
- Insufficient response to circumstances
- Social disfunction
- Economic factors

As a recurrent feature during breakdown of a society it is advanced that the societies capability to support it's expenses is severely crippled and that the decision makers or the society as a whole are no longer able or willing to support ancient structures and therefore decision making processes start to fail [1] This implies that two major pillars of the phased repository concept for the repository closure are seriously endangered, namely the capability of a society to take a responsible and well-founded decision on the appropriate time for closure and the economic capability to mobilize the resources necessary to do so.

3.2 Time Scale of Events

Catastrophic events occurring during the period of operation or during the monitoring phase while access to the tunnel system or parts of it is still possible have the potential to interrupt normal procedures and safety assurance process both in the short and long term. Such events may be a major natural catastrophe (e.g. earthquake affecting large areas), a widespread disease disabling large parts of the population (e.g a bird flue pandemic) or man-made disturbances in the society (e.g. war). Such events have the potential to cause a rather sudden loss of physical control of the repository, in case the personal in charge of actual supervision and also of its administration are unable or unwilling to continue their task. In the worst case this could imply that considerable parts of the repository containing large amounts of waste are without proper access control and without maintenance of the technical installation necessary to keep the installations operational.

A slightly different situation arises in the case of comparatively slow changes in society occurring over years or decades, e.g. changes in economic structures leading to an unbearable weight of the cost of maintenance of the site (a major economic crises like the 1928 worldwide crises) or large migration movements leading to a shift of focus of the society (as has happened in Europe several times in the last 2 milliena). Recently, at least two incidents with radioactive material of the former soviet union caused casualties due to loss of control following change in society (the Tammiku and Lilo accidents [3], [4]).

		Loss of control	
		Short term	**Long term**
Repository phase	**Active phase (deposition)**	**main hazard:** readily accessible phase **potential:** very high **barrier status:** no barrier in place	**main hazard:** open deposition tunnels/shafts **potential:** very high **barrier status:** EBS not functional, no backfill in galleries
	Passive phase (monitoring)	**main hazard:** open galleries/ easy access **potential:** high **barrier status:** EBS in place, no backfill in galleries/no seals	**main hazard:** no site surveillance/ loss of knowledge **potential:** moderate **barrier status:** poor backfill in galleries/ poor seals

It is likely that such events arise in combination (e.g. economic crises triggered by a major natural catastrophe) and that the strain on public resources will come from different directions, leaving little or no margin to sustain public expenses not considered as absolutely necessary.

So basically one has to consider 4 different scenarios of loss of control of the repository under different conditions:

- Rapid (short term) loss of control during the deposition phase: this is probably the most drastic case, as a loss of control within days, lasting for some months or years (e.g. in case of war or widespread disease) may leave hazardous waste in virtually any state, including free in the reception facility or transfer
- Rapid (short term) loss of control during the monitoring phase can be considered somewhat less critical, as the waste would be safely stored and the repository tunnels backfilled. However, the access shafts and galleries would still be open, allowing human intervention and creating rapid pathways for transport.

Depending on the circumstances and events that let to the loss of control, it can be expected that control is regained by the society (in it's old or a new form) as soon as the situation begins to stabilize or, in the case of a major disruption, that the loss of control is permanent (in case the new societal structures in place are not interested in or capable of the necessary actions). In the latter case, the situation would shift into a long term loss of control.

- Gradual (long term) loss of control during the deposition phase would imply that the originally planned mode of exploitation is slowly altered, probably shifting to a less intense use with lower quality and safety levels, coming to a halt eventually. It is likely that at least the deposition tunnels can be backfilled and the EBS put in place there, although it is to be expected that the galleries and shaft would remain open. As the time scale is in decades or more, natural degradation processes and equipment failure start to get relevant, leading to structural damage, flooding and /or unplanned alteration of the barriers in place, reducing thereby the retention capability of the repository system.

- Gradual (long term) loss of control during the monitoring phase affects a repository in a stable, reasonably secure but not yet optimum state. As before its is likely that during the decline some action can be undertaken, even if it is not as efficient and technically sound as could be hoped for. This would lead to a sealed and backfilled repository with somewhat reduced performance of the backfill and consequently possible pathways and higher as assumed transport rates. Unauthorized access at this stage is rather unlikely, as there is no easy way to enter the widely backfilled tunnel system, even if active site control has ceased.

3.3 Consequences on the Repository

Two main classes of threats to the repository integrity have been identified above: human intervention and natural processes. Human intervention would consist in unauthorized access to the repository thereby creating damage to structures or barriers in place and, in extreme cases, leading to the removal of hazardous material. The latter case is the probably most dangerous in the short term, as radioactivity may be released directly to the environment. The two first cases are by themselves not critical, but may become relevant in combination with other processes.

In fact most of the transport processes addressed in the safety assessment will still be in place, probably enhanced by unforeseen conditions in the repository (as e.g. relevant quantities of free water due to flooding). Therefore, most transport mechanisms would apply, and the overall barrier conditions must be considered substantially less efficient than assumed in the safety case. In case of loss of control during the deposition period, the efficiency of the EBS must be considered to be seriously inhibited and preferential pathway along galleries and shafts must be expected be present. The latter holds also for events during the monitoring phase, although there the EBS in the disposal tunnels can be expected to function as planned.

Disregarding the question of proliferation, the main risk implied is the structural collapse of parts of the repository and the insufficient condition of the engineered barrier system for at least some parts of the waste (open disposal tunnels, waste in transfer). The failure of necessary equipment to keep the repository in a stable and safe state is also a possible consequence of a short or long term breakdown in maintenance. This would lead ultimately to a higher than predicted and earlier release of radioactivity to the geosphere.

4 Self Closure

Accounting for the above developed scenarios, we can now define the necessary functionality requirements for a self-closure mechanism.

4.1 Function of Closure Mechanism

The two main threats to repository performance due to loss of control and it’s consequences where identifies as unauthorized human intervention and with respect to the safety case increased transport capability through the barrier system. Both mechanisms are relevant in the case of short and long term loss of control and must be countered by the closure mechanism. To do so, the following functionality should be associated to self-closure:

unauthorized human intervention:

- Prevent access
- Protect deposition zone
- Make waste retrieval difficult

increased transport capability of the barrier:

- Prevent negative impact on barrier integrity
- Prevent any unplanned physical events (flooding)
- Reduce open pathways

This functionality can be achieved by numerous means; however important restrictions apply due to the special situation in which the self-closure has to take place. As the term self-closure implies, any active mechanism involved has to be triggered automatically or with minimal external intervention. The functionality should be met without unduly compromise any later improvement on repository safety, in case control can be regained. And finally, the self-closure should not lead to an easy option for later permanent repository closure.

4.2 Active Closure Mechanisms

The most obvious mechanism to achieve the above mentioned goals of self-closure is some means to block access tunnels and shafts, e.g using explosive charges to close some parts of the tunnel system. This is however a rather crude means and not without side effects on repository safety. The way of blocking the tunnels, the emplacement of these seals and the trigger have to be studied carefully, along with possible repercussions on the repository. Several places are possible to block access and construct some makeshift seals: the access tunnels and shaft, the galleries and the access leading to the deposition tunnels. In all cases, one would have to provide the appropriate material in form that it could move in place by itself or with minimal intervention, without damaging the engineered and natural barriers. Without detailed technical planning, it is hard to see how this could be accomplished in a way that is without drawbacks for the repository safety.

4.3 Passive Mechanisms

Until now, we assumed implicitly, as does the term self-closure, that some active mechanism is to seal up some part of the repository and thereby do more or less the work planned during the ordinary closure phase. To achieve the goals of self-closure mentioned above, this may be by no means necessary, and sometimes not even the most efficient method. A series of precautions during repository design and operation could help to achieve the goals.

Clearly during operation it would be helpful to keep the quantity of waste not yet placed in sealed disposal tunnels to a minimum. This implies limited stock of waste in the surface facilities, minimized transfer times and rapid backfilling and sealing of disposal tunnels after waste emplacement. This would significantly reduce the risk of removal of waste and at the same time ensure an important part of the EBS is functional.

It is likely, that repository safety during operation depends to some degree on technical equipment and maintenance. In order to ensure repository performance after loss of control, it would be helpful to have wherever possible simple, autonomous equipment without need of maintenance (e.g. wind powered pumps to prevent ground water intrusion).

Finally repository design has a key role in assuring passive safety: the layout of disposal tunnels has to allow for the sealing of individual tunnels and wherever possible small groups of tunnels in order to optimize the EBS functionality during operation. Any threat on repository safety should be countered by constructive rather than technical means (e.g. prefer a sealing and drainage system against groundwater inflow over the above mentioned pumps).

But not only the design of the actual repository is important, but also the organisation and fitting of surface facilities. If one assumes that society regains control of a temporarily abandoned repository, the society in question is likely to have no spare resources to perform a proper closure, let alone continue operation. Therefore, it seems useful to provide from the beginning the material and equipment necessary for a planned, even though simplified closure. Material and equipment should be stable, easy to use and most notably of no commercial interest outside the repository, to prevent them from being used elsewhere. As the option of an after-crisis closure implies that some sort of information on the repository state would be needed, an auxiliary monitoring system, separated both in space and organisation structure, from the repository itself could help to achieve efficient closure. This is particularly true if such closure scenarios have been addressed and properly planned for at the design phase, and consistent backup strategies have been defined and prepared for.

Furthermore, the repository itself has to be prepared for temporary shutdown: a possible way to prevent access is to disable systematically and thoroughly any transport equipment available, as lifts, vehicles etc. and to remove all items that could be used to damage the barrier and to handle waste containers.

As such actions are likely to be undertaken under pressure, with lacking time and resources, they have to be planned for, the required dedicated storage space for the equipment provided and the disabling procedures defined.

Not in passing that in the case of the discussed short term loss of control, the repository has to stay operational at least for the time necessary to finalize the emergency procedures, which necessarily include bringing all the waste container in the compound in a safe state, either deposing them as planned or storing them in a safe way. This would certainly need some days or weeks, during which, in extreme cases, the repository has to be able to function without any external support or supply.

5 Conclusions

As can be seen from the discussion above, the possibilities, both technical and from the viewpoint of repository safety, for true self-closure mechanisms are rather limited. In fact, it seems unlikely that a repository can be brought to an acceptable safe state by automatic closure alone, without proper preparative action.

As soon as one admits to extend the self-closure concept to additional mechanisms like mode of operation, preparation for rapid closure and appropriate equipment, the perspective of loss of control, both on the short and long term, with or without regaining control, becomes manageable. This property would be more appropriately referred to as “passive safety concept” rather than “self-closure”. It implies more than constructive measures to facilitate emergency sealing of parts or the total of the repository, which would be insufficient. Starting right from the design of the repository, passing management and operational procedures and including aspects or reversibility and retrieveability, such a concept has to be carefully planned for. ■

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2A.07 Rock-mechanical Back-analysis of Long-term Deformation Behaviour of Drifts in Claystone Formations using Advanced Constitutive Models

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1 Introduction

Internationally indurated clays will be host rocks as well as barrier rock formations in case of radioactive waste disposal in deep geological formations. Therefore, laboratory tests on indurated clays together with physical modelling as well as numerical calculations simulating the EDZ development and the long-term behaviour of rock mass surrounding drifts and disposal chambers are important topics of todays international research work. There are many results of lab tests with indurated clays available in the literature now, e.g. investigations with indurated clays from the well-known sites of Mont Terri, Bure or Tournemire, *Blümling, et al. (2005), Dossier (2005) Argile, Popp & Salzer (2007), Rejeb et al. (2006), Schulze & Hunsche (2005), Tsang et al. (2005), Zhang et al. (2007)*.

In Germany the knowledge with respect to salt formations seems to be advanced but regarding argillaceous rock mass formations the German experienced knowledge is more or less limited to some geological and geomechanical experiences and observations. These have been made in context with tunnelling on one hand and with mining as well as specialised research work many years ago at the site of an abandoned iron-ore mine on the other hand. In recent times there is a good cooperation with neighboured European countries, especially France and Switzerland. For further information on current investigations in Germany see *BGR (2007)* and *DBEtec (2007)*.

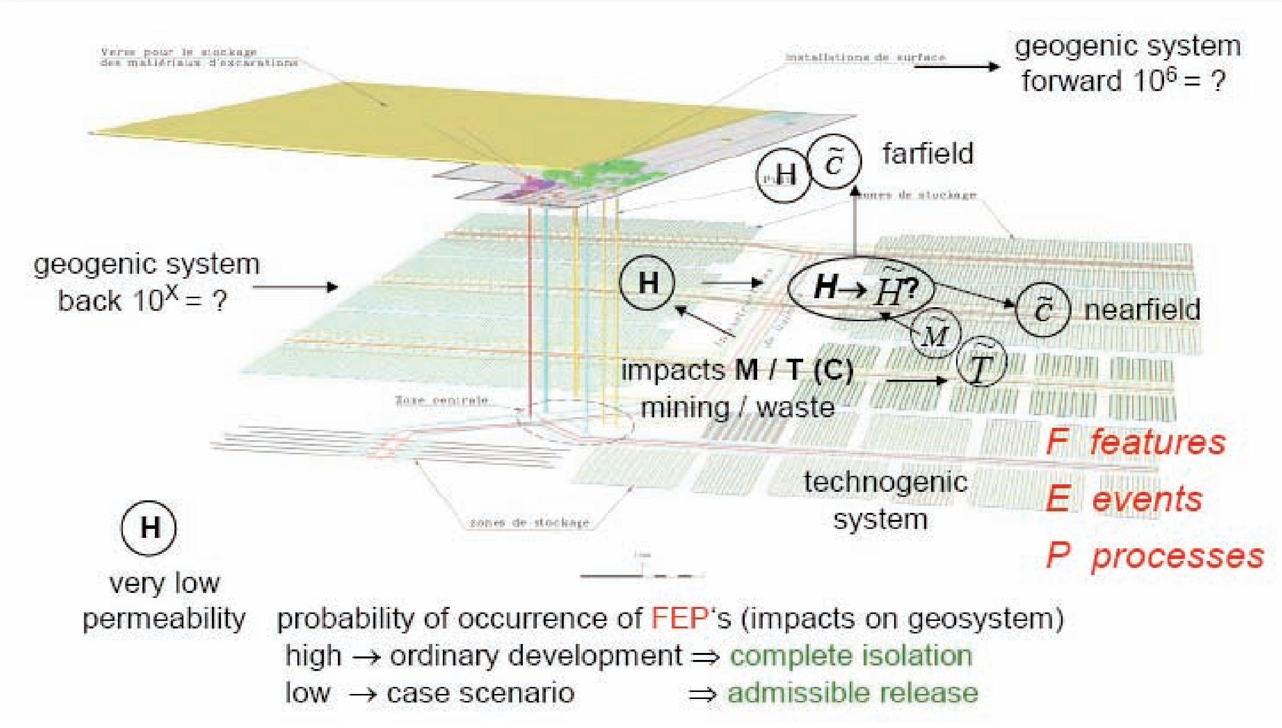
Looking for alternative geologic systems – in this case argillaceous rock formations - basic and site-specific knowledge and experiences are necessary for characterisation of their mechanical and hydraulic properties and their hydro-mechanical behaviour. Therefore, the development of advanced computer codes for numerical simulation of hydro-mechanical coupled processes during operation and in the long-term is of fundamental importance and demands improved and specialized constitutive laws on the one hand as well as in situ validation of these laws on the other hand.

2 Review the Reliability of Developed Simulation Tools to predict the Load Bearing Behaviour of Drift Openings in Indurated Clay

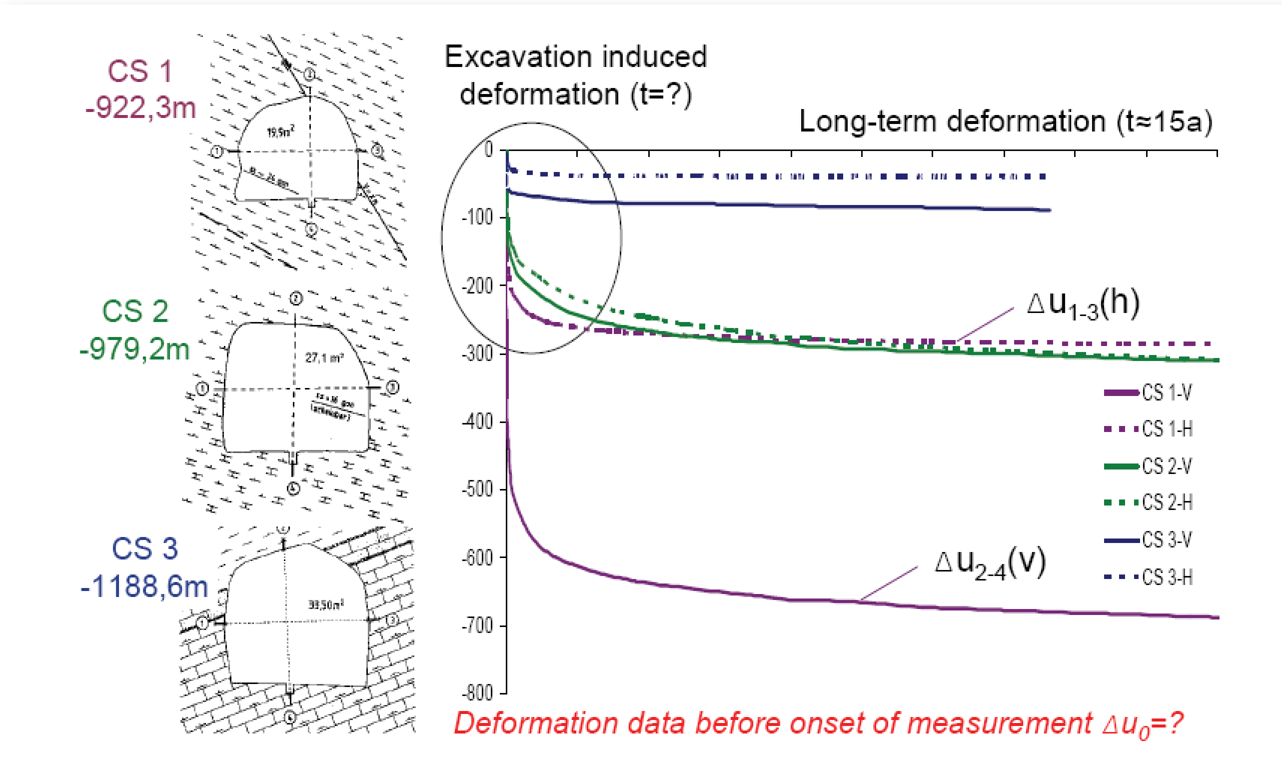
The evolution of a repository has to be described in time and space with a sufficient degree of confidence in the short-term as well as in the long-term (construction phase, operation phase, abandonment phase, long-term behaviour). The performance assessment analysis of a selected site for radioactive waste disposal will be mainly based on numerical simulations. These calculations should give a reliable prognosis of the future development of the site – including waste impacts (e.g. heat emission and gas generation) as well as possible changes of rock mass properties provided with load bearing or barrier functions within the geological system in the long-term. Figure 1 shows a schematic sketch of a repository system and the coupled processes that have to be mentioned in physico-chemical modelling.

The general objective therefore must be to show that based on site characterisation, lab test investigations as well as their transformation into physical models, in the framework of back-analysis (case history analysis) the results of in situ measurements and observations related to selected underground excavations for example can be simulated with the help of numerical models.

The *Hou/Lux-T* constitutive model has been developed for physical modelling of indurated clay formations. To demonstrate its quality of prediction capacity the results of numerical simulations have been exemplarily applicated to in situ measurements and observations that have been made during the excavation and operation phase of drifts in argillaceous rock mass at the site of an abandoned iron ore mine over a period of more or less than 15 years now. Figure 2 shows some typical cross-sections and the bandwidth of related deformation data, *Lux et al. (2005c)*.



▲ Fig.1: Schematic sketch of a repository system and coupled processes for physico-chemical modelling.

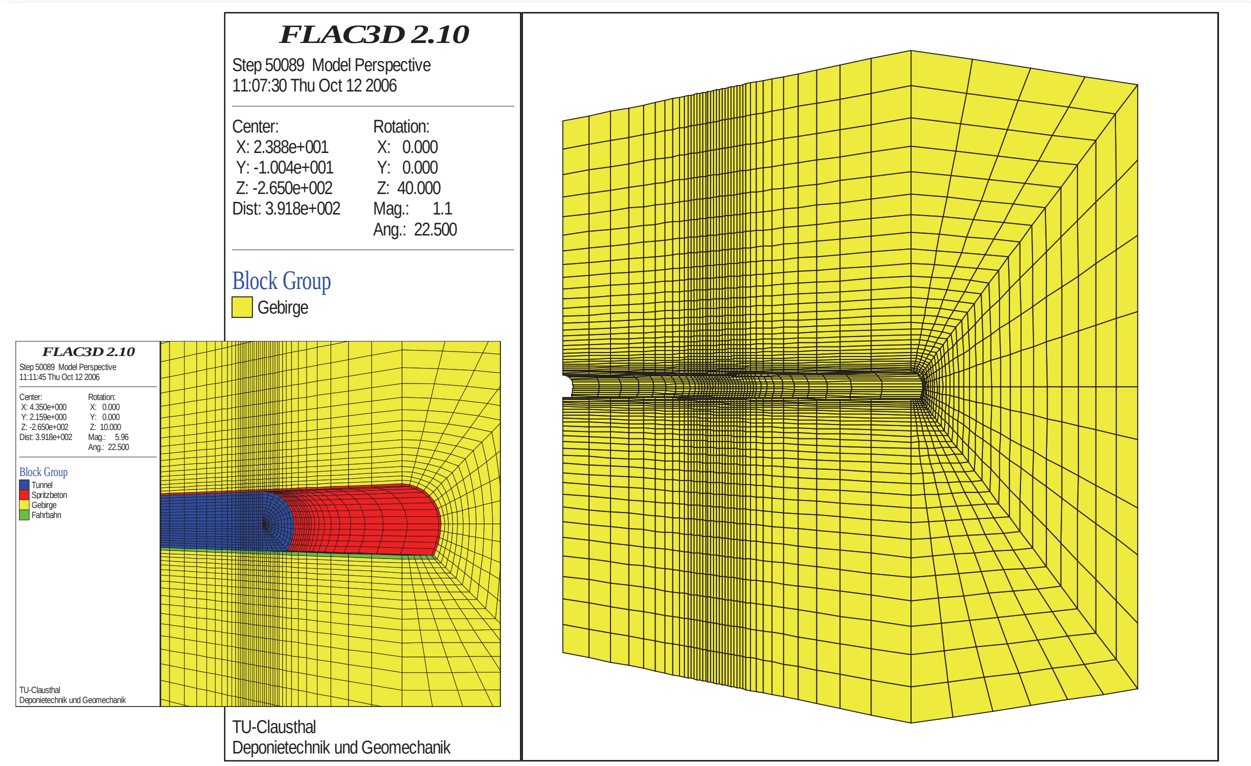


▲ Fig.2: Typical cross-sections and bandwidth of related deformation data

3 Comparison of Calculation Results and Measurement Data

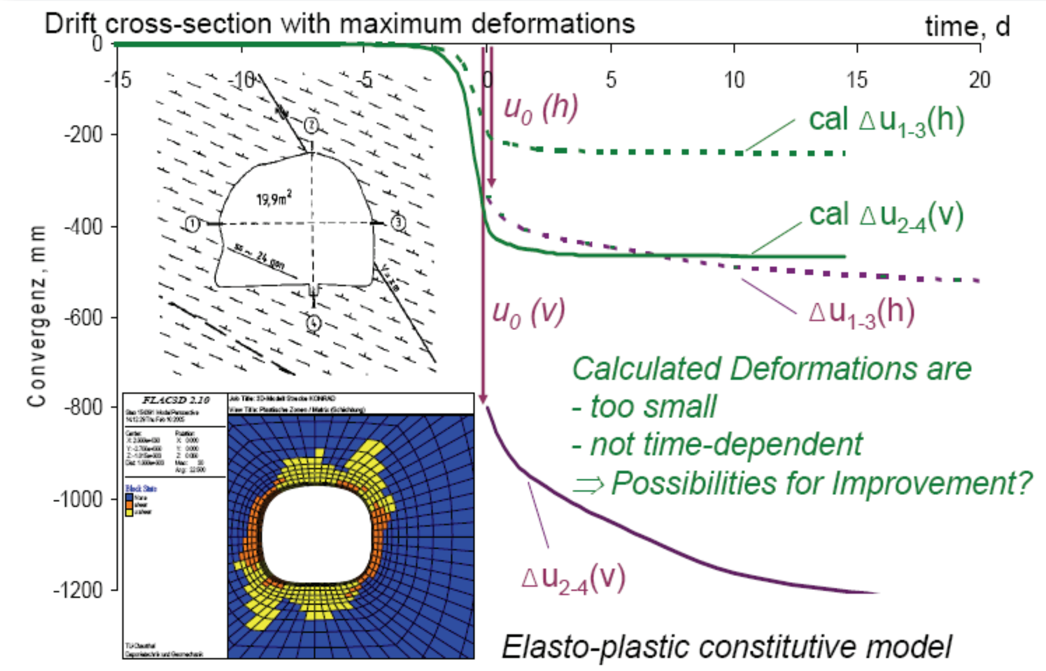
For comparison of calculation results and measurement data the constitutive model *Hou/Lux-T* will be used to model the rock mass behaviour, especially the weakening process of the rock mass in the so-called excavation damaged zone (EDZ) as well as the deformation processes of the rock mass in the long-term. This constitutive model has been originally developed some years ago for salt rock mass and is now being modified to describe the mechanical behaviour of indurated clays, *Hou (2002), Lux et al. (2005a,b,c), Czaikowski & Lux (2006), Lux et al. (2006a,b)*. In the meantime it has been implemented into the *FLAC3D*-code, *Itasca (2005)*. Due to the phenomenological-macroscopic orientation the model at this time takes into consideration the effects of different deformation mechanisms in a more phenomenological manner, based on the constitutive model *Lubby2* as well as on basic elements of Continuum-Damage-Mechanics (CDM). Argillaceous rock mass formations show a geotectonic induced laminated microstructure with partially heavy reduced strength on bedding planes. A constitutive model for describing this texture anisotropy and the belonging effects on the defomational behaviour with sufficient accuracy is naturally forced to capture these mechanisms. Therefore the texture anisotropy, which is not a dominant effect in rock salt mechanics, has been imperative for the enlargement of the *Hou/Lux-T* constitutive model. For further information see *Czaikowski & Lux (2006)*.

Figure 3 shows the 3-dimensional calculation model used in this back-analysis with special consideration to numerically simulate the excavation process of the drift in time and space in the short-term as well as in the long-term.



▲ Fig 3: 3-dimensional model with special consideration to numerically simulate the mechanical behaviour of the rock mass during the excavation process and in the long-term

Figure 4 shows a first comparison of the calculated excavation induced deformations for the selected cross-section with maximum deformations using the 3D-rock mass model and an elasto-plastic constitutive model with reduced strength on bedding planes as well as taking into account the non-measurable rock mass deformation u_0 before onset of measurement. These calculations are based on material parameters which have been estimated for the relevant rock mass area at the selected drift cross-section in the past and handed over to the investigator. These parameters have been used to start with numerical simulation.



▲ Fig 4: Comparison between measurement data and simulation data

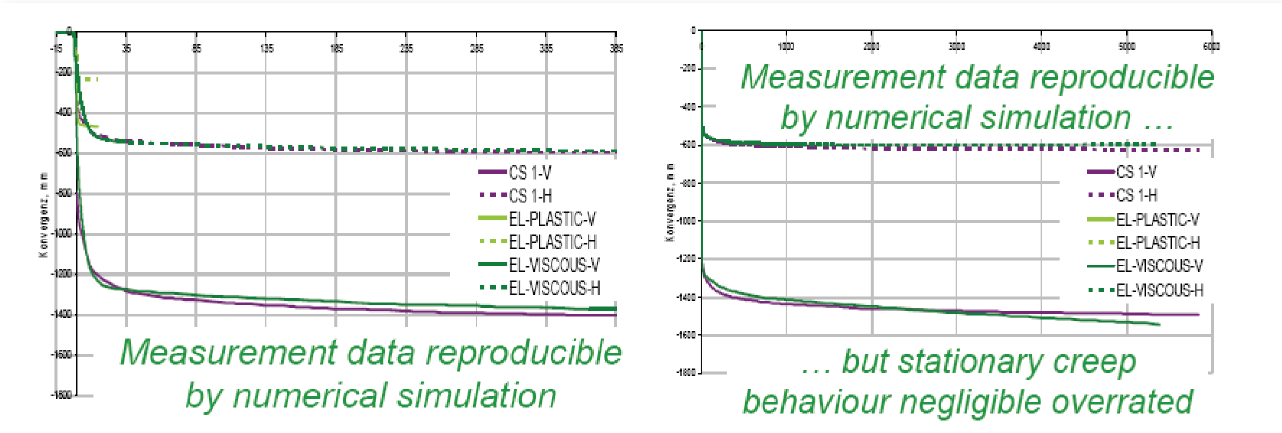
According to Figure 4 the calculated deformations are much too small and therefore the basic assumptions for the calculations need to be evaluated by additional in situ and laborative investigations. A more detailed investigation of the rock mass structure shows the existence of a regular system of closed fracture planes located at cross-section 1, Figure 5.

▲ Fig. 5: Additional detailed in situ investigations of rock mass structure and results

As a consequence these existing discontinuities in rock mass will reduce strength and enlarge deformations properties. Their mechanical effect has been taken into account in the simulation model by reducing the deformation modulus and reducing the matrix strength whilst retaining strength anisotropy.

In addition developed laboratory long-term compression tests show some creep deformation even for old claystone core material, which might be perhaps desaturated. This phenomenon could cause a reduction of creep properties compared to the original rock material.

The transfer of new results to numerical simulation using the *Hou/Lux-T* constitutive model shows an underestimation of transient creep potential as well as an overestimation of the stationary creep rate in the long-term. As consequence next steps for advanced simulations are the discontinuity-structure related increase of transient creep properties (see also discontinuity-structure related increase in elastic deformations), the in situ desaturation related reduction in long-term stationary creep rate and also possible additional elasto-plastic deformations (non-linear, short-term, time-independent during excavation). Figure 6a) shows a comparison of the time-depending deformations within the first year after excavation for the selected cross-section 1 with maximum deformations using the enlarged *Hou/Lux-T* constitutive model taking into account the above described texture anisotropy of argillaceous formations and the belonging effects on the deformational behaviour.



▲ Fig. 6: Comparison between measurement and long-term simulation data for a plotted period of a) one year and b) 15 years

In addition Figure 6b) shows a well defined agreement in case of consideration of a discontinuity structure related increase of transient creep properties and consideration of a desaturation- related reduction in long-term stationary creep rate even for a plotted period of 15 years.

4 Conclusions and Extended Outlook

The experience gained from the re-analysis of the reviewed measurement cross-sections with respect to reliability of prognosis could be summarized as follows (not really new, but worth to repeat):

- Blind prediction based on insufficient data will fail,
- Physical modelling without taking into account all relevant mechanisms just using simple constitutive models will fail,
- Reliable prognosis - even for mechanical processes only - demands a lot of experiences and fundamental as well as site-specific knowledge.

Fig. 7: Photographic view of the storage container

The experience show once more that only extensive back-analysis is able to give a better understanding of rock mass load bearing behaviour at a selected site. Essential precondition for reliable prognosis are therefore

- intensive characterisation of rock mass structure including especially discontinuities,
- primary state of stresses and pore pressures in the rock mass,
- intensive and careful laborative investigations,
- field measurements and field observations

and last but not least

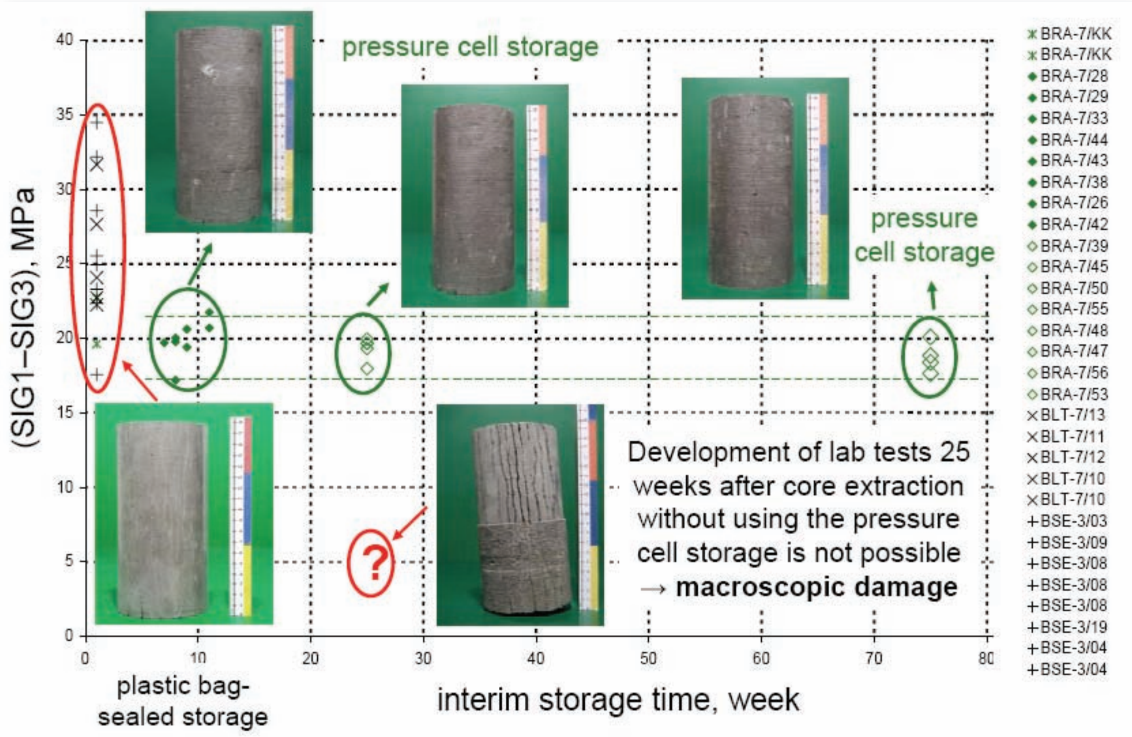
- advanced numerical simulation tools.

Finally some comments on careful laborative investigations with special respect to indurated clays and their mechanical properties together with latest experiences should be added:

The samples required for laboratory investigations are extracted from the rock mass by means of core drilling. In order to ensure that the material behaviour determined in the laboratory experiments for the different properties will sufficiently represent that of the on-site rock, it is necessary to appropriately protect the drill cores against ageing effects subsequent to their extraction and up to carrying out the rock mechanical experiments in the laboratory.

According to the rock structure, and with increasing storage time, desaturation, geochemical reactions with air as well as pore water pressures will lead to more or less strongly pronounced changes in the mechanical and hydraulic properties of the rock matrix (e.g. strengthening due to capillary forces and weakening due to microcracking).

A demonstrable successful method to avoid or minimize the mechanically, hydraulically as well as geochemically based ageing effects outlined above is the interim storage of the drill cores in special sample storage containers (pressure cells) directly after their extraction, Figure 7 *(patent specification 10 2005 053 360.4-2)*.



▲ Fig. 8: Current results of triaxial compression tests P-samples from Mont Terri (same drilling location) in dependence on interim storage time and type of storage (red – plastig bag sealed interim storage / green – interim container storage)

An important feature of this developed sample storage is that, because of the constructive design of the pressure plates and top and bottom cover plates, the hydraulic pressure is universally distributed, i.e. acts both radially and axially on the sample. The membrane reservoir and pressure relief valve in the hydraulic circuit ensure that the pre-determined pressure is held constant over the storage time, largely independent of temperature fluctuations.

Repetitive experiments on differently stored samples (plastic-bag sealed, pressure cell) coming from the same sample location have been carried out at intervals from 25 – 50 weeks over a total time span of significantly more than one year now. First of all the results show the possibility to achieve experimental results even after this relative long intermediate storage times without any visible macroscopic damaging of the drilled samples (Figure 8).

Furthermore the results show the possibility to achieve reproducible experimental results even after long intermediate storage while using the developed method for ageing resistant storage of argillaceous rock samples taking into account anisotropic deformational behaviour, typical for core samples from Mont Terri site. For further information see *Czaikowski & Lux (2007)*. ■

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2A.08 Numerical Investigation of the Long-Term Evolution of the Excavation Disturbed Zone

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Abstract

For the long-term performance of a repository in rock salt, the evolution of the "Excavation Disturbed Zone" (EDZ) and the hydro-mechanical behaviour of this zone represent important issues with respect to the integrity of the geological and technical barriers. In order to investigate these processes, numerical models with adequate constitutive laws are under development. A new constitutive model that can describe the volumetric strain and the damage of the rock salt has been implemented in the available finite-element codes. Additionally a preliminary relationship between permeability and the volumetric strain is used to calculate the hydraulic properties of the EDZ. The model parameters have been evaluated on various transient creep tests. The influence of several material parameters on the numerical results has also been studied. Furthermore, the creation and the long-term evolution of the damage zone around a gallery in a salt mine at about 700m below the surface has been analyzed and the numerical results have been compared with in situ measurements.

1 Introduction

The construction of deep geological repositories leads to the perturbation of the initial lithostatic stress state in the rock mass around the openings, creating micro cracks and degrading the hydro-mechanical properties. For the long-term performance of a repository in rock salt, the evolution of the "Excavation Disturbed Zone" (EDZ) and the hydro-mechanical behaviour of this zone represent important issues with respect to the integrity of the geological and technical barriers.

The scope of the numerical modelling is to predict the development and evolution of the EDZ under different conditions. To investigate the hydro-mechanical processes, numerical models with adequate constitutive laws are under development. A new material model for rock salt that can describe the damage of the rock has been implemented in the finite-element codes ADINA and MAUS [1], [2]. According to this model, the total strain rate is given as the sum of elastic and viscoplastic strain rates. The viscoplastic strain rate is decomposed into a part without volume changes of the material and a second one taking into account the volume changes due to the damage. In this case, the viscoplastic flow function depends on mean stress, deviatoric stress, and volumetric strain.

In the present contribution, the capability of the constitutive model to predict the volumetric deformation (dilatancy) of the rock salt has been tested for some existing transient creep experiments. The influence of several material parameters on the numerical results has also been studied. A preliminary relationship between permeability and the volumetric strain has been used to calculate the permeability of the test samples. Besides the parameters calibration, a first validation of the proposed model by comparison of the numerical results with experimental data is presented.

The last part of this paper is devoted to the simulation of the long-term evolution of the EDZ around a large gallery in the Sondershausen salt mine at about 700m below the surface [3]. The numerical results were compared with in situ measured closure rates, stresses, and rock salt permeability [4].

2 Material Model for Rock Salt

The constitutive model proposed is based on the assumption of small strains, where the total strain rate, $\dot{\varepsilon}_{tot}$ is split into elastic and viscoplatic parts as follows:

$$\dot{\varepsilon}_{tot} = \dot{\varepsilon}_{el} + \dot{\varepsilon}_{vp} \tag{1}$$

$\dot{\varepsilon}_{el}$: elastic strain rate tensor

$\dot{\varepsilon}_{vp}$: viscoplastic strain rate tensor

The elastic behaviour is assumed to be time-independent. Furthermore, the viscoplastic strain rate tensor is decomposed into a viscoplastic strain rate tensor by constant volume and a viscoplastic strain rate tensor due to damage that considers the volume change, such as dilatancy or compaction of the material:

$$\dot{\varepsilon}_{vp} = \dot{\varepsilon}_{vp}^{c} + \dot{\varepsilon}_{vp}^{d} \tag{2}$$

$\dot{\varepsilon}_{vp}^{c}$: viscoplastic strain rate without volume change

$\dot{\varepsilon}_{vp}^{d}$: viscoplastic strain rate due to damage which describes a volumetric strain

For each viscoplastic strain rate, an associated flow rule is used (i.e. the viscoplastic potential function is the same as the yield function and the viscoplastic strain increment vector will be associated with the yield surface [5])

$$\dot{\varepsilon}_{vp} = \gamma < \Phi\ (F(\sigma)) > \partial F / \partial \sigma \tag{3}$$

where $\gamma = a_1 \exp(-a_2 / T)$ is fluidity parameter, a_1 and a_2 are material constants and T is the temperature. The term Φ (F) denotes a monotonic function of the yield function (F). The meaning of the brackets $< >$ is as follows:

$$\begin{aligned} < \Phi\ (F) > &= 0 && \text{if } F \leq 0 \\ < \Phi\ (F) > &= \Phi\ (F) && \text{if } F > 0 \end{aligned} \tag{4}$$

The function Φ (F) is expressed by power law as follows:

$$\Phi\ (F) = (F - Fo)^{m} \tag{5}$$

in which m is an arbitrary constant and Fo is the uniaxial yield stress and set to zero for instance. For our viscoplastic model, the functions F^c and F^d are as follows:

$$F^c = q^2 \text{ (without volume change)} \tag{6}$$

$$F^d = n_1 p^2 + n_2 q^2 \tag{7}$$

where p is the mean stress and q is the standard stress deviator;

n_1, n_2 are material functions of the volumetric strain, ε_{vol}, and expressed as:

$$n_1 = c_1 (q^2/p^2 - c_2 (\eta_0 + \varepsilon_{vol})/(1 + \varepsilon_{vol})) \tag{8}$$

$$n_2 = 1 - c_3 . n_1 \, p^2/q^2 \qquad (9)$$

with c_1, c_2, and c_3 being material constants to be evaluated by laboratory tests. In the present approach η_0 is the initial porosity of the undisturbed rock salt.

This viscoplastic material model for damage is based on the mathematical formulation proposed by Hein [5] for granular materials, such as crushed salt, and was implemented in the available finite-element codes.

Furthermore, separate criteria are available for shear and tensile fracture [8] and a compression-dilation criterion [9] to judge the damage of rock salt (i.e. microcracks or fractures):

Shear stress criterion for compression

$$\tau_f \geq b \, |\sigma_m|^p \qquad (10)$$

where

τ_f : predicted shear stress at failure

σ_m : mean stress

b and p are fitting parameters.

A safety factor is defined as:

$$FS = \tau_{oct} / \tau_f \qquad (11)$$

with τ_{oct} being the actual octahedral shear stress. Under compression loads, failure will occur if $FS > 1$.

Tension-induced failure is assumed if the maximum principal stress exceeds a tension limit of 1 MPa.

Compression-dilation boundary:

$$\tau_{oct} \geq f_1 \sigma - f_2 \sigma_m^2 \qquad (12)$$

where f_1 and f_2 are fitting parameters.

A preliminary relation between permeability and the volumetric strain given in references [8] and [9] are used to calculate the permeability, *k* of the rock salt:

$$k = A \cdot \varepsilon_{vol}^{B} \qquad (13)$$

A and B are material parameters.

3 Validation of the Proposed Damage Model

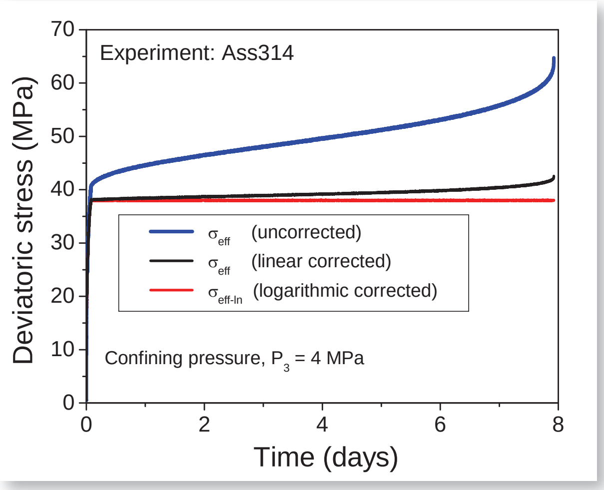
In order to verify the implemented material model and to demonstrate the applicability of the model to describe the dilatant volumetric strain of rock salt, a number of different triaxial laboratory tests were investigated numerically. The calculated strain rates were compared to experimental data. The influence of different material parameters on the numerical results was studied as well.

3.1 Numerical Simulation of Transient Creep Tests

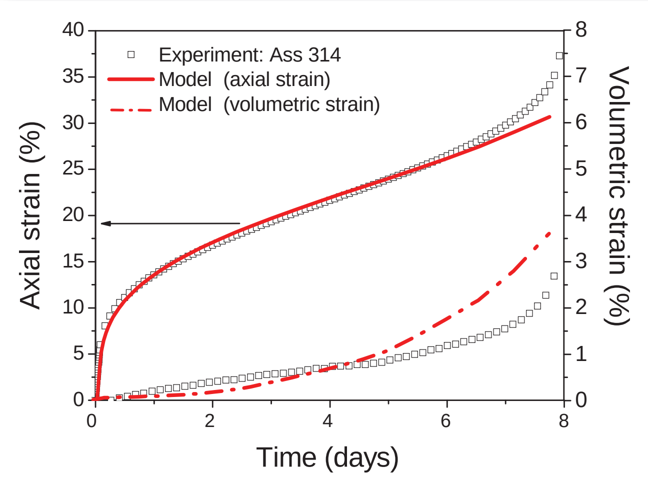
A transient creep test conducted on Asse rock salt sample [11] for which the volumetric creep strain rates are measured was selected for the numerical analysis. The cylindrical sample was subjected to an axial compression of 38 MPa and a lateral confining pressure of 4 MPa. The history of the experimental differential stress for different stress definition is shown in Figure 1. The tertiary creep test lasted about 8 days. For the calculation, a simple one-axial symmetric element was considered. The results are interpreted using the infinitesimal strain and Cauchy stress related to the current sample shape. The material parameters used for simulation are summarized in Table 1. In the calculation was assumed that the dilation of the samples start immediately after loading (i. e. the short time compaction of the sample was not simulated).

Table 1: Material constants used in the numerical analysis

Thermoelastic properties	E = 36 GPa; ν = 0.27; α = 4.2E-05 1/K
Viscoplasticity, equations (3), (5)	a_1 = 2.08E-06 1/s; a_2 = 6520; m = 2.25; T = 293 K
Dilatancy, equations (8), (9)	c_1 = 0.7; c_2 = 400; c_3 =1
Permeability, equation (13)	A = 3.2 E-11; B = 3.5; ref. [9]



▲ Fig. 1: Effect of the stress definitions on the experimental results



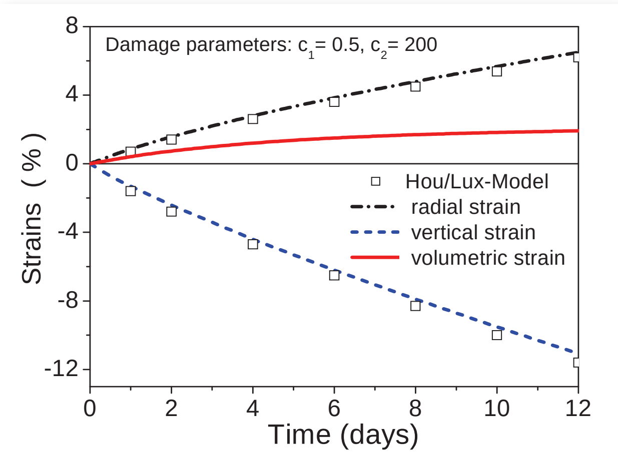
▲ Fig. 2: Comparison between measured creep strains and calculation results

Figure 2 shows the calculated development of axial and volumetric strains compared with the experimental data. Unfortunately, the measured radial strain is not available, but it can be calculated easily from axial and volumetric strains. As evident from this figure, the calculation predicts the dilatant volumetric creep strain of the sample quite well.

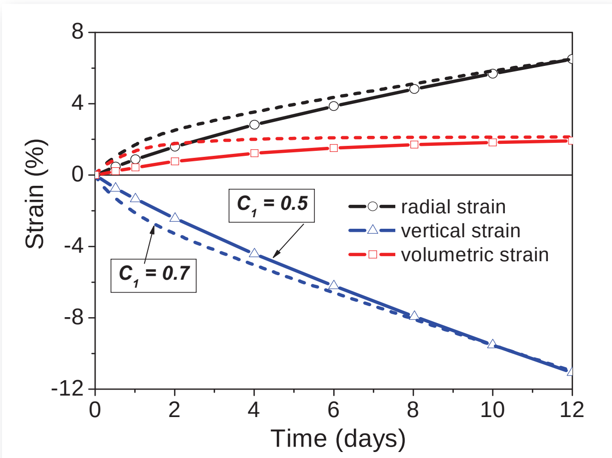
3.2 Influence of Damage Parameters on Numerical Results

The second problem considered was that of a similar triaxial creep test. For this test, the numerical results obtained using the Hou/Lux damage model [12] over a period of 12 days are given in [13]. The sample was loaded axial with 25.5 MPa and a confining pressure of 1.5 MPa. Figure 3 shows the numerical results obtained with the proposed constitutive model in comparison with the results from the simulation with the Hou/Lux model. The damage material constants c_1 and c_2 were adjusted to achieve a good overall correlation between both calculations. In order to study the influence of both damage parameters c_1 and c_2 on the calculated strains, a sensitivity analysis was performed. This study indicates that the parameter c_1 influences the shape of the strain curves, but not the final values of the strains if the assumed boundary conditions and the creep parameters of the rock salt remain the same (Figure 4). The parameter c_2 directly limits the maximal amount of the calculated dilatancy. As can be seen from Figure 5, the increase of c_2 from a value of 200 to 300 induces a relevant reduction of the volumetric strain maximum.

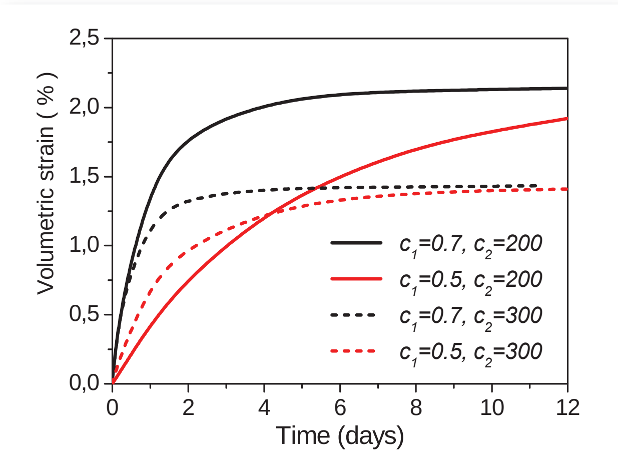
Figure 6 shows the resulting permeability of the sample as a function of calculated volumetric strain for two sets of damage parameters. As expected, both permeability curves at the end of the numerical experiment reach the same values. The calculated values are in the ranges of in-situ permeability of rock salt measured in the near field of a large excavation in the Asse mine [9].



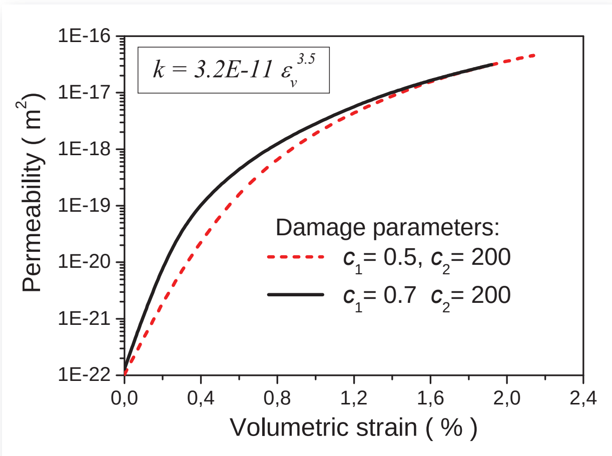
▲ Fig. 3: Comparison between results calculated with the proposed damage model and the results from Reference [13]



▲ Fig. 4: Development of strains for two different values of the damage parameter c_1



▲ Fig. 5: Development of volumetric strain for different values of damage parameters

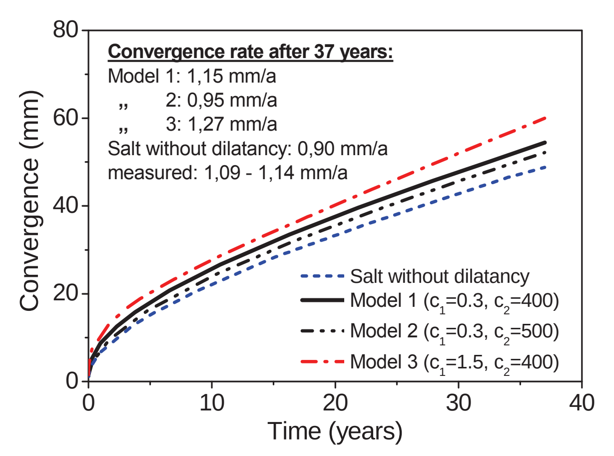


▲ Fig. 6: Calculated permeability of the sample during a transient creep test using two sets of damage parameters

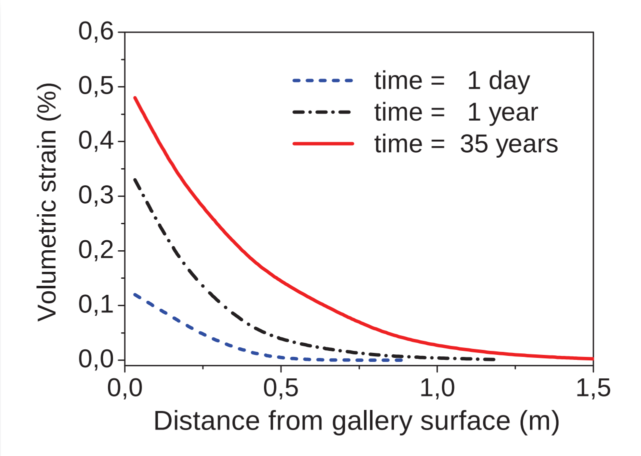
4 Modelling of the EDZ around a Gallery in Rock Salt

To illustrate the capability of the material model and the codes used, the long-term evolution of the EDZ around a 37 years old gallery in a rock salt formation was analyzed. The circular gallery with a radius of 1.5 m is located at about 700m depth from the surface in the Sondershausen salt mine. The room closure rates, the radial stress, and permeability distribution after 37 years were measured in situ and are given in references [3], [4].

For the numerical analyses a two-dimensional (2D) finite element model was used, assuming plane strain conditions. A detailed description of the geometrical model and the boundary conditions is given in [14]. The radial convergences of the gallery calculated for three sets of model parameters are presented in Figure 7. The measured and calculated closure rates after 37 years are also given in this Figure.



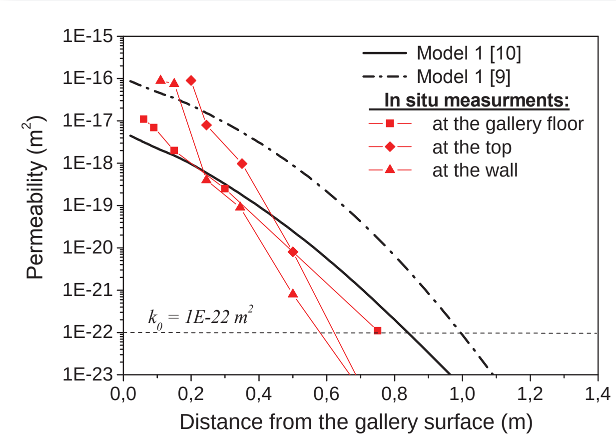
▲ Fig. 7: Gallery convergence calculated for different model parameters and the closure rates 37 years after excavation



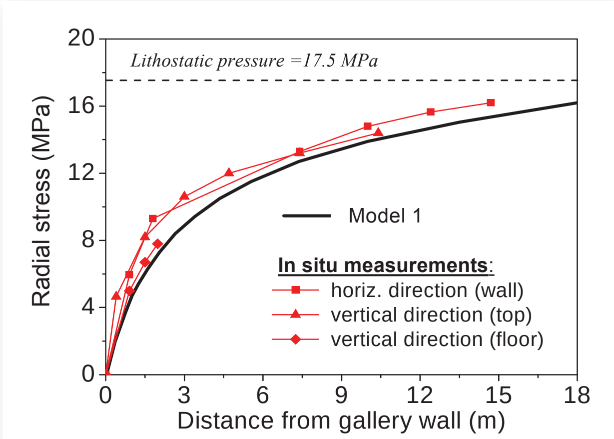
▲ Fig. 8: Development of volumetric strain in rock salt around the gallery at different time points after excavation

Figure 8 shows the evolution of the predicted volumetric strains around the gallery for different times. The volumetric strains increase with time within the rock (i.e. the EDZ expands). A comparison of the calculated with equation (10) and the material parameters given in references [9] and [10] and in situ permeability distributions measured by [3] are presented in Figure 9. The numerical prediction of the “Model 1” for both parameters sets agrees quite well with the measurement near the gallery surface, but it seems to overestimate the permeability at a distance larger than 0.5 m.

The measured minimal stresses around the gallery and calculated radial stresses are shown in Figure 10. The good agreement between the model calculation and the measured data confirms that the current model can predict stress distribution around the gallery with an acceptable accuracy.



▲ Fig. 9: Comparison of calculated and measured distribution of permeability 37 years after gallery excavation



▲ Fig. 10: Comparison of the measured and calculated minimal stress, 37 years after gallery excavation

5 Conclusions

A constitutive model for rock salt that permits the description of the development and evolution of the EDZ around the excavations has been implemented in the available numerical codes. Several laboratory experiments have been analyzed numerically to gain confidence in the proposed model. At the moment, the calibration of the model parameters is limited by the few reliable triaxial transient creep tests. Additional laboratory tests will help to improve the model performances.

The comparison of numerical results and in-situ measurements indicates that the implemented constitutive model with material parameters fitted to laboratory tests works quite accurately. The validation of the presented model on more complex loading conditions is under progress. For all that, the codes and material models used appear to provide good predictive data to assist future calculations for the hydro-mechanical behaviour of the EDZ around the openings in rock salt.

Acknowledgements

This work has been funded partly by the Commission of the European Communities in the framework of the NF-PRO project.

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2A.09 Integrity of a Salt Barrier during Gas Pressure Build-up in a Radioactive Waste Repository – Implications from Laboratory Investigations and Field Studies –

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Abstract

Rock salt formations are attributed to be impermeable for gases and fluids ensuring safe isolation of the disposed waste. Otherwise, significant quantities of gas are generated in the long-term. If the gas pressure would exceed the minimal principal stress fracturing processes may take place.

For an assessment of the provable impact of increasing gas pressures on the integrity of rock salt new results from long-term gas injection tests in gas-tight sealed boreholes are presented. Slightly above the primary stress state the gas-breakthrough (i.e. permeability increase of 3 - 4 orders) was observed, without pressure induced micro-acoustic activity. This contradicts the feared single gas-frac scenario but demonstrates enhanced gas permeation into the surrounding salt due to secondary permeability. Fortunately, self-sealing was confirmed by re-establishing, at least partial, gas-tightness after receiving a lower pressure level. The additional healing capability of salt is proven by investigating the post-frac situation of a natural gas-frac analogue that occurred at the salt mine Merkers in 1989 after a rock burst, i.e. by direct probing of the former gas-frac zone via a 250 m long core-drilling and hydrofrac measurements.

Based on the results final conclusions will be drawn about the likely consequences of a gas pressure build-up in a salt repository.

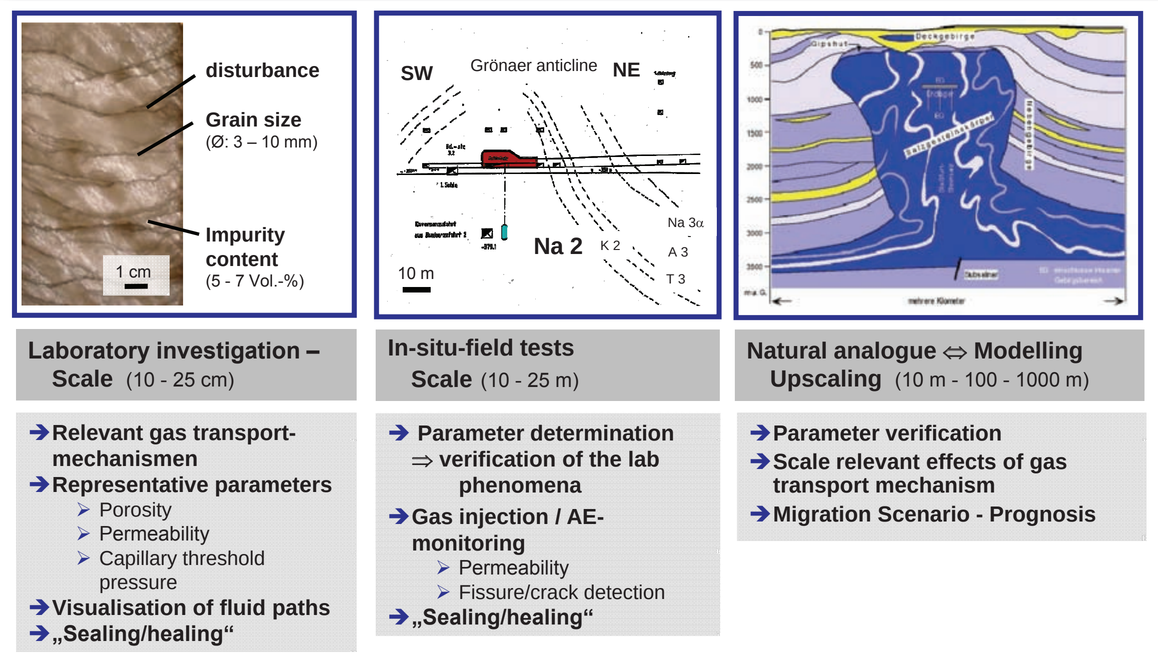
1 Introduction

Tightness is a fundamental prerequisite for guarantying the long-term safety of a radioactive waste repository in a salt formation, i.e. rock salt is attributed to be impermeable for gases and fluids as well documented by comprehensive investigations carried out in laboratory experiments and in-situ for the past 30 years, e.g. [1], [2]. However, from the perspective of gas storage, it has to be mentioned that impermeability means that the rock mass permeability is in the order of 10^{-20} - 10^{-22} m^2 or lower but not zero (e.g. [3]).

As a consequence of tightness in performance assessments (PA) of the long-term behaviour of radioactive waste repository in a salt formation it is usually assumed that a time dependent pressure build up will occur due to the following reasons:

- significant quantities of gas are expected to be produced inside the repository by various processes (e.g. corrosion and microbial degradation, and in addition, in the case of radioactive waste radiolysis.
- owing to the creep properties of salt, the underground opening will converge, thus reducing its volume, until the pressure inside the cavity equalizes the external stress in the salt bed.

If the gas pressure exceeds the primary formation stress localized dilatant shear formation and fracturing may take place in the overlying salt barrier, i.e. described as the so-called gas-frac scenario. The conceptual, theoretical and experimental framework for fluid-pressure driven fracture propagation is well documented in standard hydrocarbon exploration literature (e.g. [4]). The process of pneumatic fracturing is principally identical to common hydraulic fracturing except for the compressibility of gas. Exceeding the minor principal stress (plus the tensile strength of the medium as generally discussed in the literature) a macroscopic frac typically develops which is initiated quasi-instantaneously and propagates on a plane which is oriented normally to the direction of σ_3 until the gas pressure in the macroscopic fracture becomes less than the value of the minimum principal stress (shut-in pressure).



▲ Fig 1: Conceptual approach for investigation of gas pressure dependent rock salt integrity.

Due to the risk that pneumatic fractures might become preferential pathways for the repository fluids and could short-cut the barrier function of the host rock and engineered barrier system (EBS), the extent to which pneumatic fractures may propagate into the host rock is important for PA (e.g. [5]):

- Overpressures could expel contaminated porewater
- Gas flow could carry volatile radionuclides.

Thus knowledge of pressure dependent gas transport properties of rock salt is a key issue in the long-term assessment of storage of high level radioactive or toxic waste in salt formations. Despite the extensive geo-mechanical research done in the last decades the remaining fundamental questions referring to a pressure build up in underground openings in salt are:

- How the host rock salt reacts on the gas pressure build-up, i.e. what is the influence of the pressure build-up rate and the minor principal stress?
- If a gas-frac becomes likely is there a potential self healing capacity, i.e. how efficient are such processes?

Because pressure development and associated scenarios may strongly depend on scale factors, i.e. size of the underground openings and lithological effects, a comprehensive approach is applied for answering these questions consisting of different scales of investigation as schematically depicted in Figure 1.

Firstly, results from lab and field investigations are presented focusing on the barrier integrity of a salt repository during a hypothetical gas pressure build-up in the underground openings. Gas injection tests were performed under well controlled laboratory conditions on core samples and in two gas-tight sealed boreholes as long term field tests (duration up to three years), which are situated in the Bernburg salt mine in the Staßfurt (Na2) and the Leine salt (Na3) formation, respectively. In addition to observations of gas transport features, both test sites are equipped with very sensitive micro-seismic monitoring systems (operated by GMuG) to detect gas pressure induced rock disturbances or, at least, a potential gas-frac.

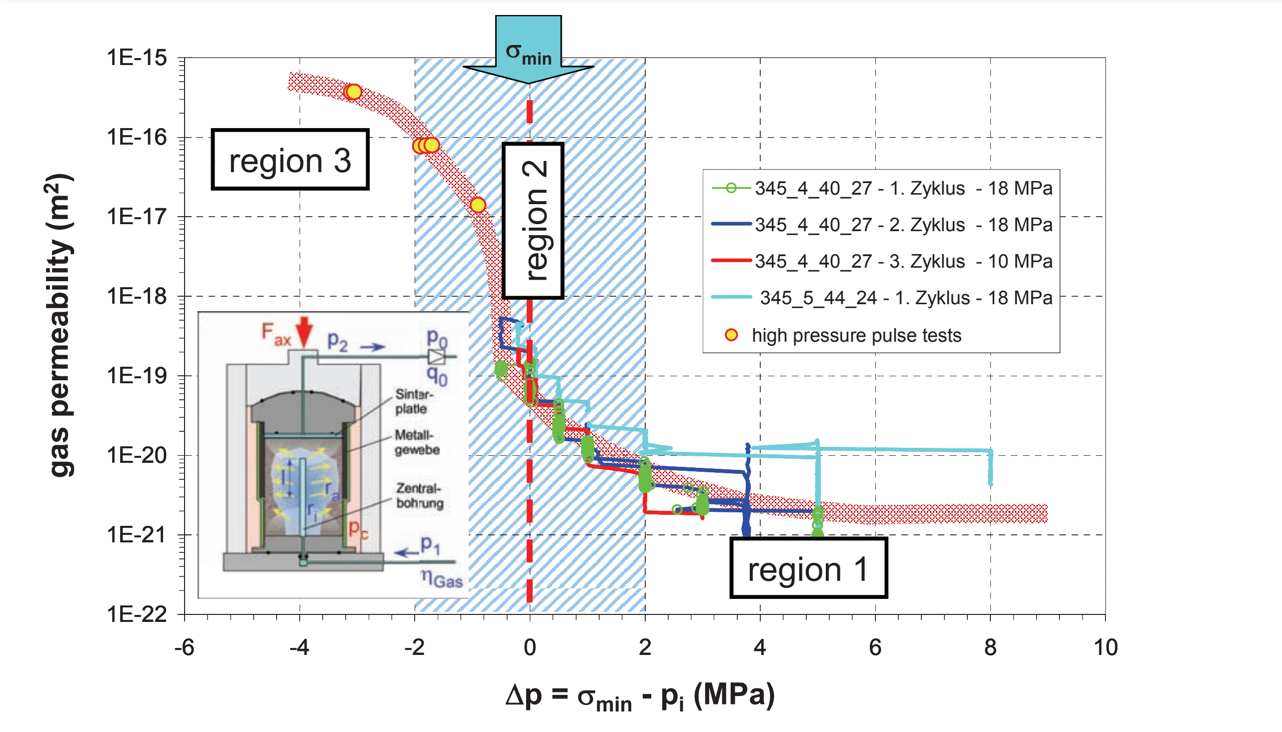
To extent the frame of survey to a more realistic scale, we also investigated the gas-frac situation after the rock burst that occurred in the Merkers potash mine in March 13th in 1989. This event is treated as unique example of a natural gas-frac analogue in a salt barrier. Remarkably, the frac-related gas release decreased between 1989 and 2000 to nearly zero (overall gas outflow 46 Mio. m³ CO_2) which has been attributed to the almost partial recovery of integrity in the Lower Werra salt beds. This assumption has been proven by direct probing the former gas-frac zone via a 250 m long core-drilling and hydrofrac measurements, and by numerical rock-mechanical back analysis of this rock burst.

Based on the well documented results of the field tests and the temporary recovery of barrier integrity at the Merkers site final conclusions will be drawn about the likely consequences of a gas pressure build-up in salt.

2 Laboratory Investigations

In the course of the project an extensive laboratory program on gas transport in rock salt (taken from various locations) has been performed under realization of a wide spectrum of experimental conditions (e.g. dilated and pre-compacted core samples). Some of the results are already published in [2].

For determining gas permeability under well defined loading conditions and at high gas pressures a conventional triaxial cell is used with a special sample arrangement, as schematically depicted in Figure 2. Corresponding to the field borehole geometry a cylindrical sample is equipped with a central sack hole. Measuring the radial gas outflow around the central injection borehole in the sample for a given pressure (i.e. pulse tests) respectively measuring the equilibrium pressure for constant injection rates facilitates the calculation of the gas permeability, using the well known Darcy law for gases. Alternatively, the gas pressure can be applied in a conventional way through plates on the sample end faces initiating axial gas flow.



▲ Fig. 2: Permeability vs. differential pressure. Summary of gas-injection tests performed at 10 MPa respectively 18 MPa quasi-hydrostatic pressures – stationary flow tests with constant injection pressure steps. In addition, results of high pressure pulse are indicated. The inset shows the scheme of the cell with radial gas flow geometry.

Focusing on the results of gas-breakthrough experiments at high gas pressures the observed permeability evolution of various injection tests performed at different minimal stresses is summarized in Figure 2 as a function of differential gas pressure which is simply the difference between the confining ($p_c = \sigma_{min}$) and the gas injection pressure (p_i). Generally, three pressure regions have to be discriminated:

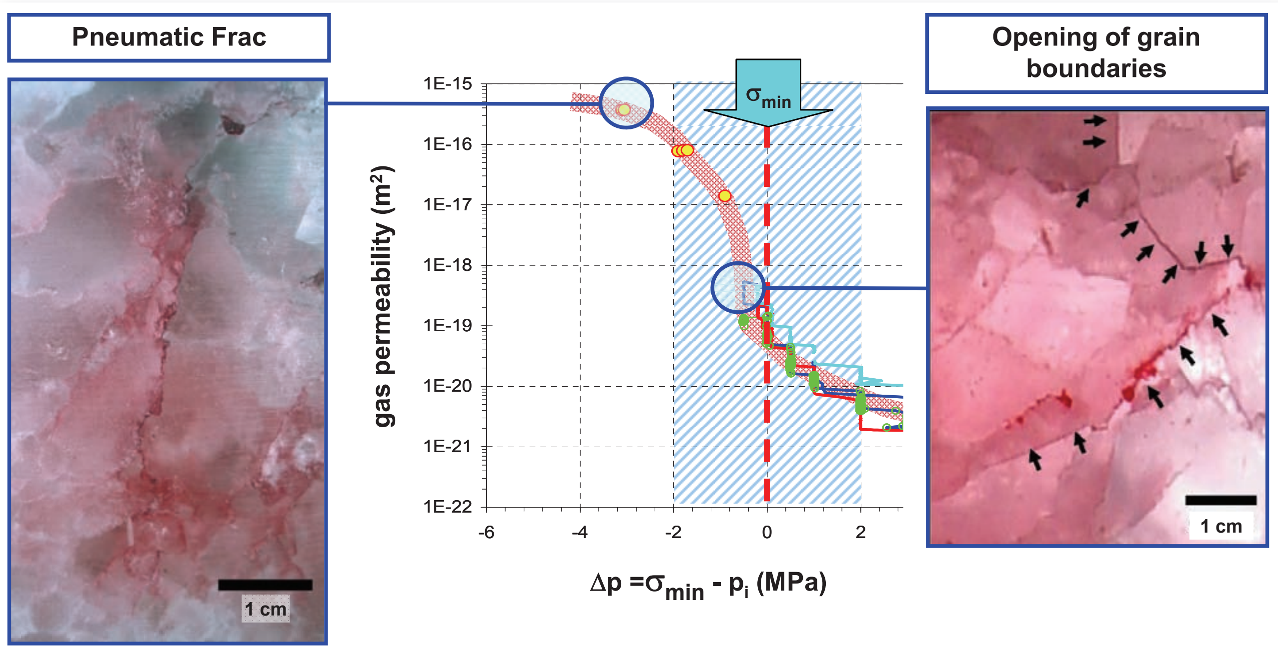
- Region 1 - at low gas pressures ($p_i \ll \sigma_{min}$) the initially measured permeability of rock salt is extremely low, i.e. $k < 10^{-20}$ m², but increases slightly during stepwise pressurization independently from the initial permeability state. At injection pressures some MPa below σ_{min} a significant increase of around one order occurs.

- Region 2 - when p_i approaches σ_{min} the gas breakthrough occurred resulting in a steep increase of permeability (up to 5 orders, whereby the lower the initial permeability the higher the rise).

- Region 3 - at pressures $p_i > \sigma_{min}$ a plateau of permeability at $k > 10^{-16}$ m² is reached. But it has to be mentioned that in this region the database is only weak. Due to the drastically increased permeability, the gas flow in region 3 became so high that the used flowmeters (up to 1000 ml/min) were not able to cover the appearing flow rates. In this stage pre-pressurized gas volumes were used to initiate high injection rates resulting in maximal gas presssure 30 bars > than the minimal stress, i.e. high pressure shut-in tests.

Importantly, if overpressurisation, i.e. $|\Delta p|$, is limited cyclic increasing and lowering of the injection pressure results only in a weak hysteresis curve indicating that after a decrease of the injection pressure the primary permeability state is nearly recovered. In addition, the pressure induced permeability increase depends only on the relative deviation of the gas pressure from the minimal stress and not on the absolute order of minimal stress.

Visual inspection of the moderate gas pressurized samples ($|\Delta p| \approx 0 – 1$ MPa) clearly reveals dilated grain boundaries acting as flow paths (see Figure 3). Importantly, only if the gas injection pressures are significantly higher than σ_{min}, i.e. $|\Delta p| \gg 2$ MPa, also



▲ Fig. 3: Microstructural observations on samples subjected to various injection pressures as indicated in the k vs. Δp-diagram. For better visualization the sample has been flooded with a red colored solution impregnating opened pore space. (right) Dilated grain boundaries (indicated by arrows) in the pressure region of the gas breakthrough (p_i is in the order of σ_{min}). (left) Singular fracture after gas injection at pressures with $\Delta p < -2$.

transgranular gas-fractures were observed. This differential stress value corresponds to the lower limit of experimentally measured tensional strength of intact rock salt, which would indicate a transition from pressure driven gas permeation to real pneumatic fracturing due to overpressurisation of some MPa. However, this gas pressure level is only reached at very high injection rates.

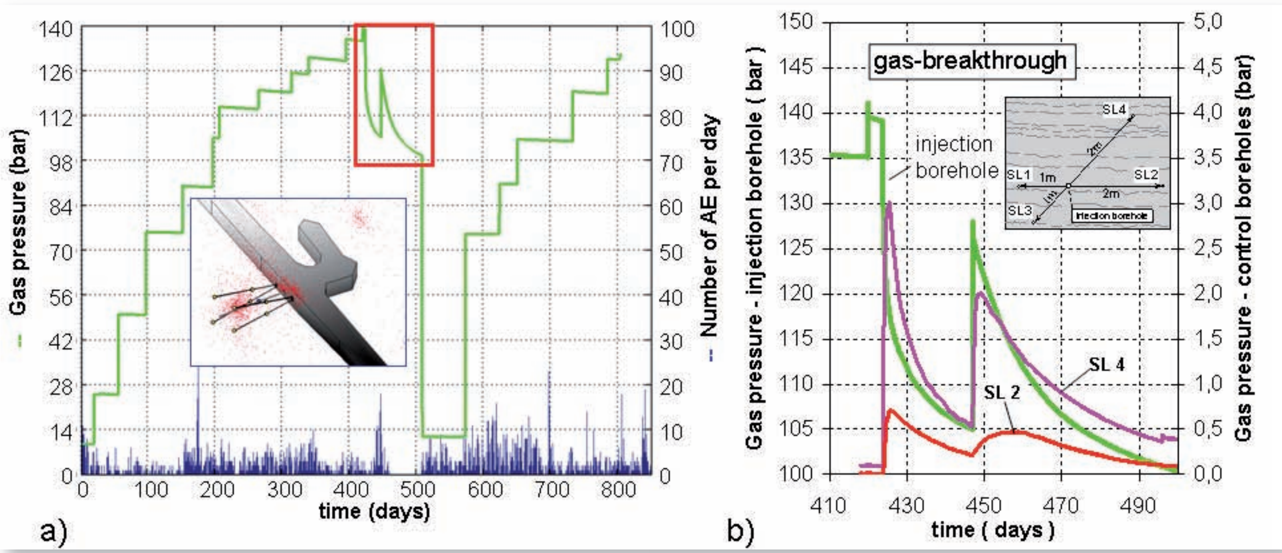
3 The Gas Threshold Field Tests at the Bernburg Site

Because in lab tests the pressurized gas volume is limited to some few tens of cubic centimetres additionally two field tests were performed in the Bernburg salt mine increasing the scale of investigation by a factor of 1000 to some tens of cubic decimetres. The test sites are situated in two different lithological units: (site 1 - "gas permeation") the older rock salt (Na2, Staßfurt rocksalt) where the geological situation corresponds to domal conditions and (site 2 – "BfS") the younger rock salt (Na3, Leine salt) where a bedded salt formation exists. Preliminary test results are already reported by [2]).

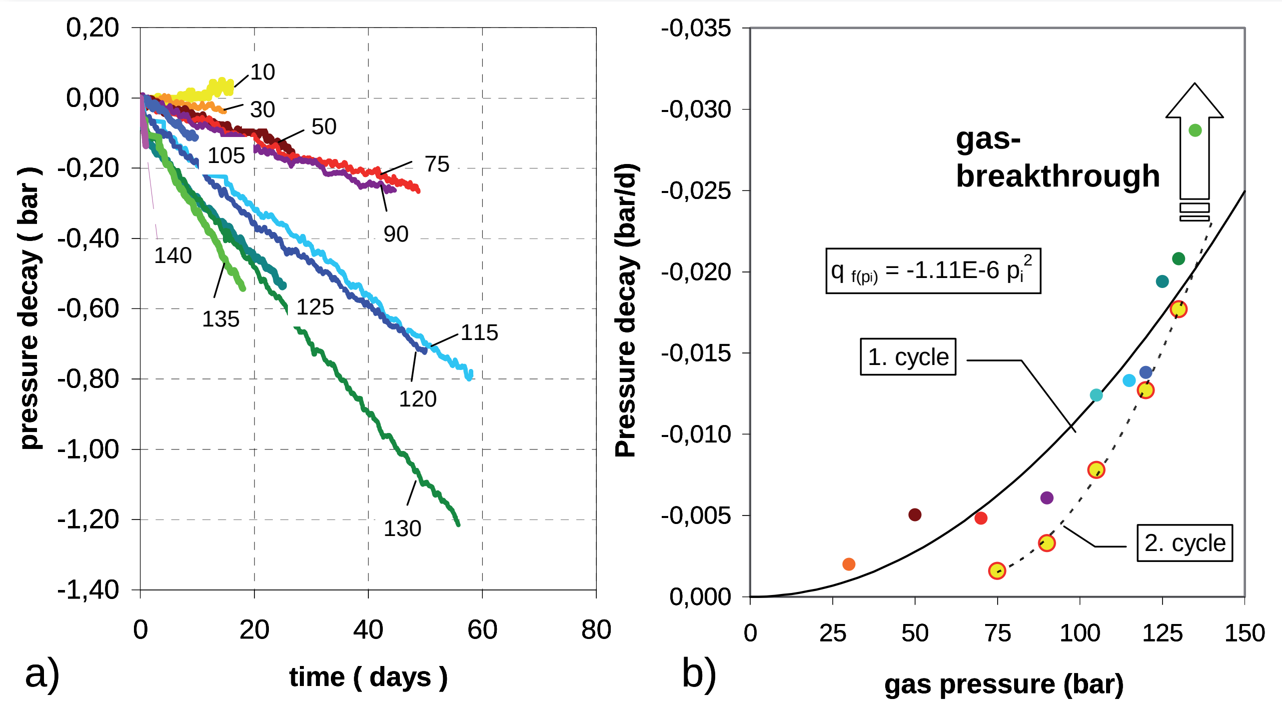
Both test areas are placed in a level of about 520 m depth. Fortunatly, at the test sites only minor mining activities were performed which warranted undisturbed conditions in deeper wall portions as a prerequisite for the reliability of the injection tests, i.e. by hydro-frac measurements. With an average pressure gradient of 0.024 MPa/m which is typical for the Bernburg location, a mean lithostatic stress level of around 13.0 MPa is estimated.

At both locations an array of 9 boreholes, each 25 m long, was installed consisting of the nearly horizontal central injection borehole (∅ = 60 mm) and four surrounding control boreholes (∅ = 42 mm), parallel drilled in distances of 1 respectively 2 m to detect gasbreakthrough. In addition, a micro-seismic monitoring array was installed in four funnel-shape oriented boreholes (∅ = 101 mm), each equipped with two seismic sensors, in the near field of the test (approx. 2 to 4 m distance). The measuring holes were sealed using a hydromechanical packer system positioned in around 10 m depth, i.e. behind the dilated contour.

Due to the hypothesized low rock permeability both field tests started with stepwise pressurization using dry N_2 for observation of the pressure decay in each step. The actual state of the complete course of the still running injection test at site 1 is illustrated in Figure 4 and the evaluation of the observed pressure decay of various injection steps is depicted in Figure 5.



▲ Fig. 4: Multi-stage gas pressure injection test: (site 1) the older rock salt (Na2). a) complete test duration with injection pressure (line) and measured micro-seismic activity (AE; scale bars): 1st and 2nd pressurisation cycle. The inset shows as a 3D-sketch the borehole array with the AE-network. b) Gas-breakthrough in the pulse test at nominal 140 bars. Note the coeval pressure decay in the injection borehole and pressure increase in the control boreholes SL 2 and SL 4. The inset shows the arrangement of the injection borehole (in the center) and the four surrounding control boreholes. The visible bedding consisting of non connecting anhydrite-portions is nearly horizontal.



▲ Fig. 5: Evaluation of the pressure decay. a) normalized pressure decay (referred to the initial pressure: $\Delta p = p_{f(t)} - p_{initial}$) curves for the various steps. At each curve the initial pressure level is indicated ($p_{initial}$ (bar)). b) Gas pressure decay rates from pulse tests vs. mean gas pressure – two pressurisation cycles.

The so performed pulse tests show very limited pressure decay, between nearly zero and 1 bar / 50 days, which required test durations for each step of between 20 and 50 days which indicated a very gas tight host rock. Because the shape of pressure decay corresponds roughly to straight lines, nearly-stationary gas flow is expected in the injection tests until 135 bar. The evaluation of the pressure decay rates as a function of pressure shows a progressive increase which can sufficiently approximated by a quadratic relationship as included in Figure 5b. Thus, the observed pressure dependence on pressure decay rates corresponds to Darcy-flow of compressible media and, in addition, no hints exist for capillary effects due to brine accumulations in the flow pore spaces, which would result in threshold pressures.

As can be seen from Figure 5 the pressure discharge accelerates when the injection pressure is increased to 135 bar which is slightly above the estimated primary stress state of 13 MPa. Further pressure increase up to 140 bar results in a more pronounced pressure decay (in the order of -0.15 bar/d) which was nearly 5 times higher than before. In addition, after 4 days in the transient phase of the pulse test a dramatic gas pressure drop occurred accompanied by the gas-breakthrough into two of the four control boreholes (compare detail section in Figure 4b). Amazingly, the pressure build-up occurred in the two more distant boreholes (d = 2 m), arranged diagonal above (SL4: $\uparrow p_p$ = 3.2 bar) respectively parallel (SL2: $\uparrow p_p$ = 0.7 bar) to the central injection borehole.

Remarkably, the rapid pressure decay during the break-through is characterized by transitional behaviour aspiring an extrapolated equilibrium state at around 100 bar, which would reached after approx. 50 d. Restoring the injection pressure to around 128 bar replicates nearly the same pressure decay. With respect to the potential gas-frac scenario, it is important to note that the highly sensitive micro-seismic monitoring gives no hints for a pressure induced change in the micro-seismic activity during the whole pressurization cycle, in particular also not during the gas-breakthrough phase (Figure 4a). However, during the test period discontinuous appearance of seismic events were observed but they are mostly related to EDZ phenomena in the drift contour.

After test duration of around 500 days the pressure was lowered to 10 bar to repeat the gas injection tests cycle (Figure 4a). Surprisingly no outflow of gas from the pressurized rock contour was observed during a period of around 60 days. Repetition of various pressure steps results in pressure decay rates which are slightly lower than in the 1st cycle. This is attributed to an existing gas pressure loading in the borehole contour, i.e. impregnation with gas but became nearly equal at pressures above 100 bar (Figure 5b). However, the reversibility of the gas transport behaviour clearly indicates closure of temporary opened pathways which were created in the former two gas breakthrough cycles as indicated by a pressure increase in the two metre distant control boreholes. Obviously the initial gas tightness around the injection borehole is restored due to the associated pressure drop.

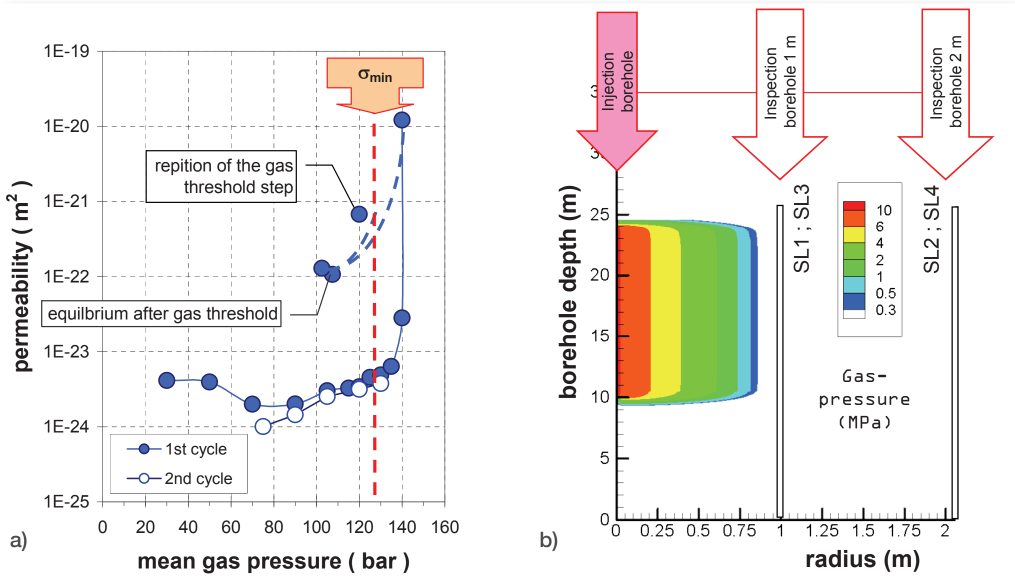
4 Evaluation of the Gas Breakthrough observed in Lab and Field Scales

Since the gas transport properties of rock salt are responsible for the required hydraulical integrity, knowledge about the relationship between the developments of stress respectively gas pressure induced damage and permeability is of utmost importance during an assumed pressure build up. In addition, the influence of gas pressure build-up rates and scaling factors need to be understood for an assessment of the risk of a potential “gas-frac scenario”.

4.1 Gas Pressure induced Permeability Changes of Rock Salt in Field Tests

Due the complex flow geometry and random conditions evaluation of field gas injection tests is not a simple task, in contrast to permeability determinations in laboratory tests (e.g. Figure 2). As prerequisite, the main process governing the flow properties of rock salt must be known (e.g. [6]). Here, as inferred from the observed pressure dependence on gas outflow rates (compare Figure 4b), the hydraulic system in rock salt is described on the base of Darcy-flow, as simplification, assuming pure gas phase flow. Two approaches (an analytical solution according to a simple radial gas flow model and the so called r-z- model which comprises a numerical simulator for spatial flow around a borehole, e.g. [7]) were used for estimating permeability.

Remarkably, both approaches were found to be sufficiently in agreement confirming a very low permeability of $<10^{-23}$ m^2 for both test sites at injection pressures below the undisturbed stress state of around 13.0 MPa (for details see [2]). Referring only to the data of approach (1) the permeability results obtained for the various injection steps performed at site 1 are summarized in Figure 6a, which shows the permeability evolution as a function of nominal gas pressure in the injection borehole.



▲ Fig 6: Evaluation of gas-injection tests (test site 1). (a) Gas-permeability evolution during stepwise gas-injection. Note, that at the pressure step of 140 bar two permeability values were estimated, before and during gas-breakthrough. Additionally, the gas breakthrough was repeated (b). Extent of the pressurized zone around the injection bore during gas-breakthrough.

At pressures ≥120 bar a slight increase of permeability is observed until the gasbreakthrough occurs at 140 bar resulting in a rise of three orders of magnitude to approx. 10^{-20} m². With the coeval transient pressure decay the permeability decreases until preserving stable flow conditions at around 100 bar, respectively at 10^{-22} m². The permeability drop is replicated in the so called re-frac cycle when the injection pressure is increased to around 128 bar (Figure 4b).

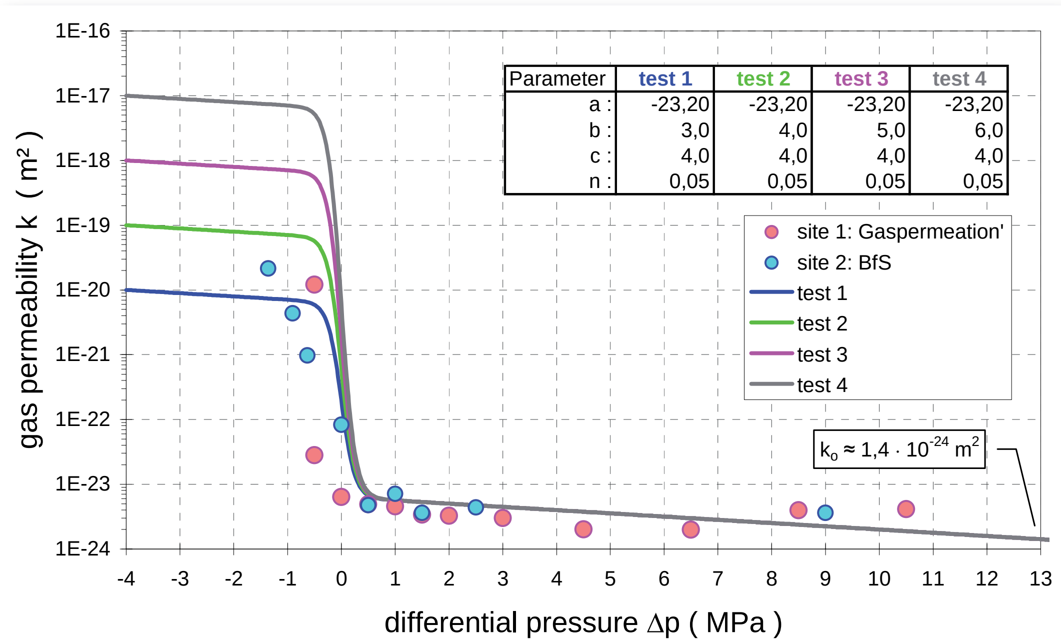
Qualitatively, the observed gas pressure permeability relationship corresponds fairly well to the results of the laboratory tests as described in chapter 2. Accordingly, the two regions 1 ($p_i \ll \sigma_{min}$) and 2 (p_i approaches σ_{min}) can be identified during the course of the field test. But region 3 ($p_i \gg \sigma_{min}$), where pneumatic fracturing occurs, was not reached in the field tests. Nevertheless, in both scales a drastic rise of permeability (up to 3 - 4 orders of magnitude in the field but probably higher in the lab tests, i.e 5 orders) is documented, which is obviously only a function of the difference (Δp) between gas pressure (p_i) and minimal stress (σ_{min}) independently from the order of σ_{min} itself.

For the description of the relationship k vs. differential stress $\Delta p = \sigma_{min} - p_i$ a preliminary empirical function is developed (based on the hyperbolic tangent), which is graphically depicted in Figure 8 (parameters are included in the table of the figure):

$$k = 10^{a + 0{,}5 \cdot b \cdot (1 + \tanh(-c \cdot \Delta p)) - n \cdot \Delta p} \qquad (4\text{-}1)$$

The parameter a refers to the estimated initial permeability value (k_o) at zero gas pressure, in this case $1.4 \cdot 10^{-24}$ m². The pressure induced only weak permeability increase permeability, as observed in the regimes 1 and 3, is described by the parameter n, whereas the shape of permeability rise approaching the threshold stage is approximated by the curvature parameter c. The factor b corresponds to the order of magnitude of permeability increase, but because the regime 3 is not documented by the field test results, this parameter is varied between 3 and 6. After parameter approximation the modelled permeability evolution fits were well to the experimentally measured values in the field tests.

Depending on the gas pressure induced permeability of the surrounding salt a pressurized zone will evolve around the injection borehole with time. The application of the r-z-model offers an estimate of gas pressure distribution during the gas-breakthrough corresponding to a permeability increase up to 10^{-20} m² in the pressure step of ~140 bar. As depicted in Figure 6b) the extent of the pressurized zone at the gas breakthrough is in the order of around 1 m, as can be seen from the



Parameter	test 1	test 2	test 3	test 4
a :	-23,20	-23,20	-23,20	-23,20
b :	3,0	4,0	5,0	6,0
c :	4,0	4,0	4,0	4,0
n :	0,05	0,05	0,05	0,05

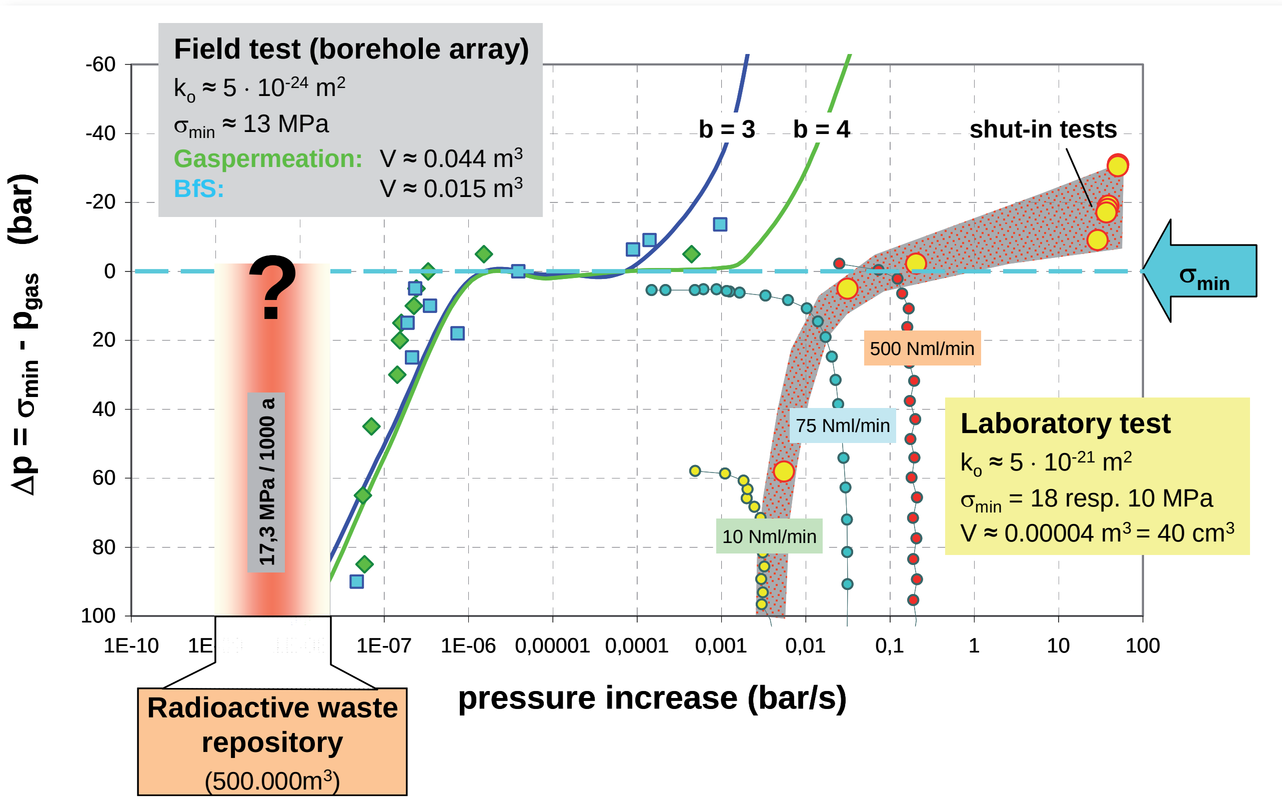
▲ Fig. 7: Permeability k vs. differential pressure Δp for the two test sites fitted by various parameter sets (σ_{min} ≈ 13 MPa). The inset shows the used parameter sets of formula (4-1).

pressure isoline of 3 bar. However, during the gas breakthrough an increase of pressure up to 3 bar was observed only in the more distant control boreholes which implies that the pressure distribution in the test site is not homogenous.

This infers that the reliability of the performed modelling is restricted to isotropic salt conditions. In addition it has to be mentioned, that the data evaluation does not consider the presaturation with gas obtained in the foregoing steps. In consequence, the here presented permeability results are in a preliminary state, because the lithological random conditions are up to now not well understood, and need to be supplemented by a more adequate modelling. Nevertheless, the new developed permeability / differential pressure relationship offers a reliable description of the gas pressure induced change in transport properties of rock salt.

4.2 Pressure Build-up Rates in Relation to Scale Effects

As already mentioned in chapter 1 the process of pneumatic fracturing depends mainly on the pressure build-up rate, i.e. when the gas production rate is no longer matched by the gas transport capacity of the rock mass. However, because the size of the pressurized volume strongly influences the outflow of gas, i.e. the larger the volume the tighter the system, the scaling factor needs to be addressed. For illustration of this aspect the order of the maximal reachable pressure state depending on the pressure build up rates at lab and field scales are depicted in Figure 8.



▲ Fig. 8: Synthesis of gas-pressurisation of rock salt in lab and field tests. For comparison the equilibrium states (p_{max}) referring to the initial pressure slopes are indicated by various symbols. (1) In the lab tests two experimental regimes are realized: constant gas-flow injection and shut-in tests with a high pressurized gas reservoir. (2) In the field tests the observed experimental data are additionally fitted by numerical modelling based on equation (4-1). For comparison a range of hypothetical pressure build-up rates for a radioactive waste repository in a salt formation with an average value of 17.3 MPa / 1000 a is included referring to a volume in the order of 500.000 m³.

Referred to laboratory conditions a hypothetical gas pressure build-up can easy simulated with constant injection rates (q_{in}) into a well defined volume drilled an initial low-permeable salt sample (compare Figure 2). As can be seen from Figure 8, the pressure will increase but with declining rate which is simply due to the pressure dependent rock permeability increase (k = f(p)), as described by formula (4-1), enhancing the fluid outflow from the pressurized reservoir, i.e. described by q_{out}. As a consequence, the pressure increase is limited to a final value, when q_{in} and q_{out} are in balance. For the later discussion only the final equilibrium pressure level referred to the initial pressure build-up rate are used for former discussion which are represented in Figure 8 by isolated symbols.

Focusing on the maximal reachable pressure it can be clearly seen, that if $p_i < \sigma_{min}$ the pressure build up is rapid corresponding to a small span of pressure build-up rates, but if the minimal stress is approached ($p_i \approx \sigma_{min}$) further pressure increase is suppressed, i.e. the curves is shifted in the order of σ_{min}. Due to fact of enhanced rock permeability of several orders extensive higher injection rates are required to reach pressures significantly above the minimal stress. In the lab such conditions could be realized only by shut-in tests.

In the field, due to the low rock permeability mostly pulse tests were performed but for the further data interpretation it is assumed that the observed pressure decay rates at each pressure step correspond to the necessary injection rates to establish the pressure equilibrium, i.e. $\dot{q}_{in} = \dot{q}_{out}$. Comparison with the few existing data from constant rate injection tests confirms the validity of this assumption.

Plotting the so obtained equilibrium pressures in Figure 8 it is apparent, that the necessary pressure build-up rates to reach the minimal stress are significantly lower than in the lab tests. This is obviously due to the extremely low permeability of the rock and also due to the increased reservoir volume. However, the general characteristics of pronounced gas permeation is also proven, when the gas pressure approaches the minimal stress.

5 The Natural Gas-frac Analogue Merkers

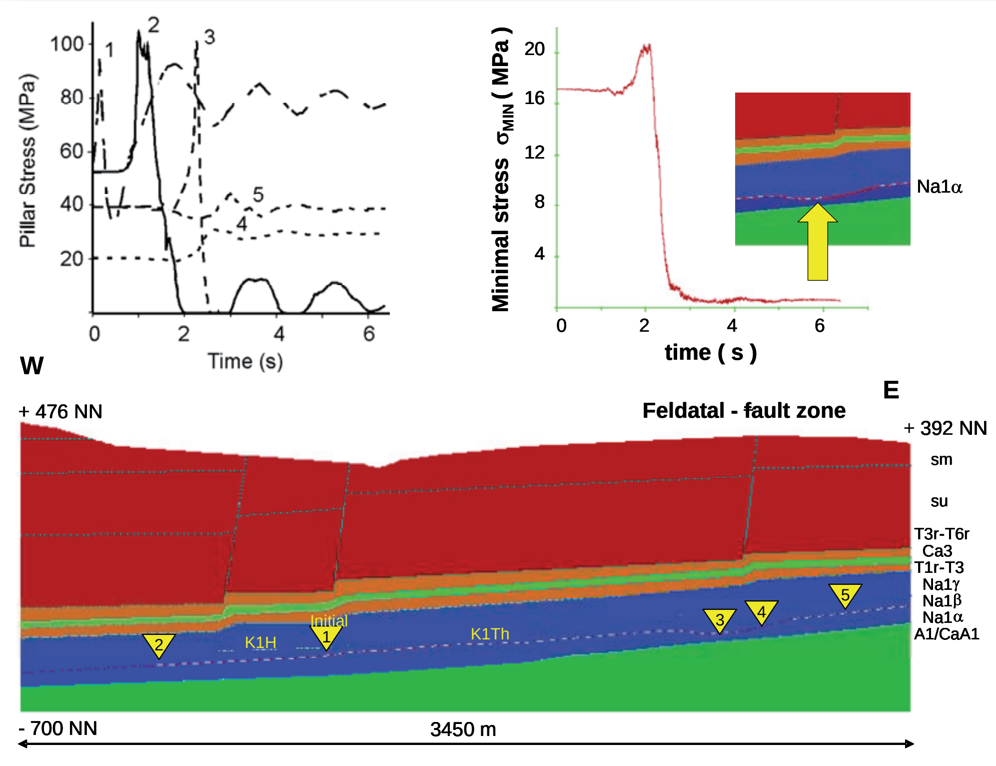
Although a potential gas-frac risk seems to become unlikely due to the observed permeation phenomena ones needs to know what happens if such a disastrous event has occurred. The natural gas-frac analogue Merkers offers a unique opportunity to study underlying conditions of barrier integrity loss due to a real gas-frac and the subsequent healing behaviour.

During the basaltic volcanism in the Tertiary formation 15 to 20 millions of years ago, in the Vorderrhön region CO_2 is accumulated below the Werra salt beds (Na 1) in the Rotliegendes and Lower Werra cycle, in liquid or supercritical phase with $p_{gas} \approx 7.5$ MPa. The overlying Werra Rock Salt and the intercalated potash seams “Thüringen” (K1Th) and “Hessen” (K1H) form the natural barrier to the CO_2 deposits, i.e. the seal.

The potash salt has been extensively mined in the Werra district, recently by room-and-pillar mining. Due to time-dependent destrengthening process in the load bearing pillar system, mainly composed of carnallitic rocks, a rock burst occurred in March 13th in 1989, in the region Völkershausen. It was one of the most devastating mining induced seismic events worldwide with a local magnitude M_L of 5.6. Within a few seconds 3200 pillars in a minefield with an area of 6.5 km² in 750 m to 900 m depth in the carnallitic seam Thüringen were destroyed. Immediately after the rock burst CO_2 was detected escaping from clefts and gaps in the gallery level at the south-eastern rib of the generated fracture zone.

To investigate the mechanism of this rock burst and the associated gas-frac numerical back analysis was performed with UDEC [8]. Modelling of the brittle failure of the salt rock carnallitite was accomplished with a specially developed visco-elasto-plastic constitutive law, which includes hardening/softening and dilation, and is available for use as a DLL-application [9]. Static and dynamic calculations were performed to investigate the processes that lead to the collapse of the ten-year-old mining field. The results are summarized in Figure 9.

The initialization of the rock burst was a blasting at a carnallitic pillar of the 2nd seam which initiated failure of carnallitic pillars first in the vicinity below the 1st seam before the fraction process continues towards the East (Figure 9 - lower part). The dynamics of the fracture process of sturdy carnallitic pillars is characterized by the transfer of the high pressure zones, which are close behind the deconsolidated contour areas, into the pillar center due to quick progress in contour deconsolidation. This leads



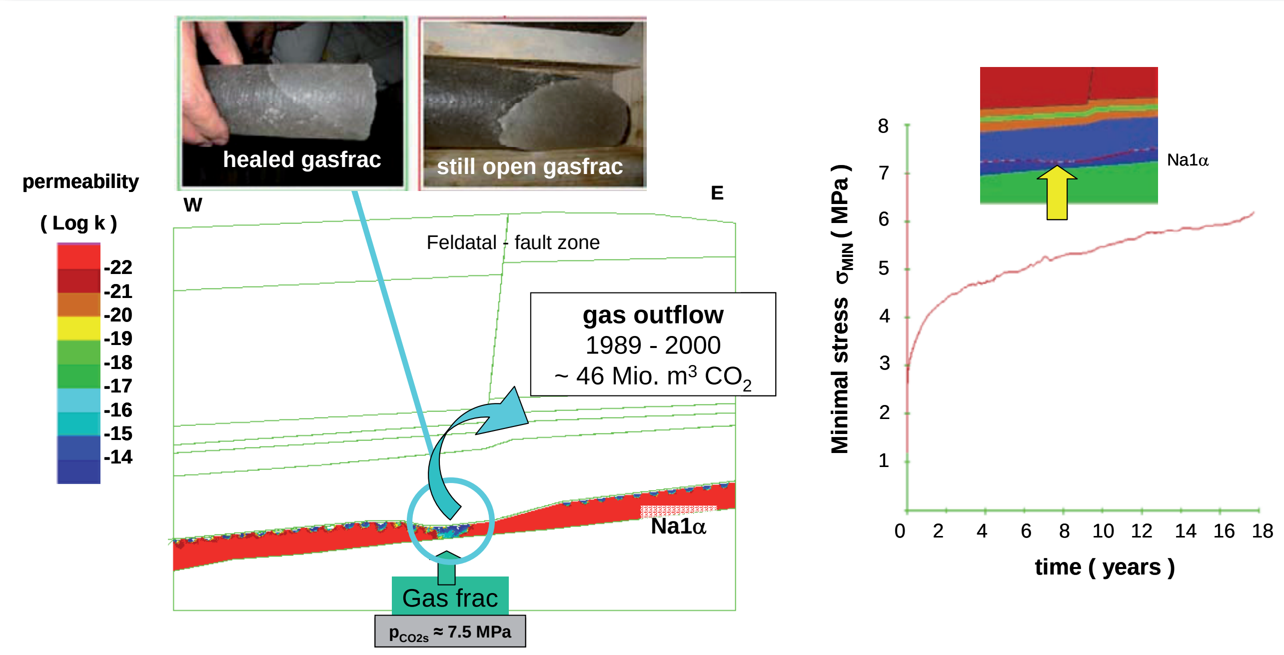
▲ Fig. 9: Synthesis of the rock burst simulation in model. (lower part) rock mechanical model and development of pillar failure in carnallitite at 2nd seam (K1H = seam ´Hessen`, hard salt; K1Th = seam ´Thueringen`, carnallitite). (top left) Pillar stress vs. time (location see lower part). (top right) Decrease of minimum principal stress σ_3 at base of Lower Werra rock salt Na1α.

to high local pillar stresses which are not sustained by the pillar cores who fail (Figure 9 - top left). According to the calculated chain reaction the fracture process for the 1600 m from the dynamically stimulated pillar to the eastern rib lasted ~2 seconds (Figure 9 –top left, curve 3). The coeval subsidence of the surface, which occurred dynamically, reaches about 1 m in the subsidence center.

Referring to the gas-frac scenario it is important to note that coevally with the convergence jump in the mining horizon the minimum principal stress (σ_{min}) is lowered suddenly from 20 MPa to some few Megapascal at the gas pressure bearing base of the Lower Werra-rock salt Na1α in the area of the thinned zone below the Feldatal-fault zone (Figure 9 - top right). According to the resulting pressurisation rate (dp/dt ≈ several MPa / s) and the arising amplitude of overpressurisation (|Δp| » several MPa) a real gas-frac occurred inducing shear dilatancy. This event nicely corresponds to the experimental observations (see chapter 4).

Using the permeability / differential stress relationship ($\Delta p = \sigma_{min} - p_i$) from equation 4-1 a localized highly permeable zone with $k \approx 10^{-14}$ m^2 could be documented by the back analysis formed exclusively in the bulk area of the rock salt rising up to the upper edge of the Lower Werra-anhydrite, which ideally corresponds with the observed gas outflow (Figure 10 – left part).

However, between 1989 and 2000 a time dependent decrease of gas release was observed (overall gas outflow 46 Mio. m^3 CO_2). This evolution has been attributed to the almost partial recovery of integrity in the Lower Werra salt beds as may inferred from numerical rockmechanical back analysis of this rock burst (Figure 10 – right part). For verification of this assumption the former gas-frac zone has been probed via a 250 m long horizontal borehole (Ø = 85 mm) with core-drilling and hydrofrac measurements.



▲ Fig. 10: Barrier integrity after the rock burst. In addition, typical frac patterns are shown, recovered by a 250 m long bore into the former gas-frac zone. (left side) Permeability in Lower Werra rock salt Na1α some few seconds after rock burst. (right side) Time dependent recovery of confinement, as characterized by the minimal stress σ_{min}.

Despite technical difficulties the potential fracture zone was reached in June 2005 where suddenly a CO_2-out blow occurs. After sealing the borehole several pressure build-up tests were performed, which indicate a pressure build-up in the order of 2 MPa, but with declining rate and weakly decreasing final equilibrium pressures during the test repetitions.

On the recovered core samples two types of fractures are distinguished. Coring-induced fractures are intersecting the cores more or less perpendicular to the coring axis, showing sometimes concave fracture planes. Only in the last 5 metres pressure induced gas-fractures were observed, which are, most important, both open **and** healed (see Figure 10 - upper part). Their orientation is accordance to the calculated stress trajectories inclined to the borehole axis (~45°) and are interpreted as tension fractures or joints, which generate perpendicular to σ_3 (smallest principal stress) and are opened in this direction. The observed healed fractures are attributed to the effectiveness of fluid assisted dynamical recrystallisation, which have restored almost partly the hydraulical integrity in the fracture zone.

In addition to gas injection tests, a complete profile of the minimal stress distribution along the 250 m long borehole has been measured with a special developed hydrofrac tool. The profile clearly indicates the transition from the non-affected salt region ($\sigma_{min} \approx 25$ MPa) to the rock burst zone, where σ_{min} actually varies between 10 and 18 MPa depending on the overlaying room-and-pillar system. Despite this, the proof of confinement in the originally gasfracturized zone, where σ_{min} was lowered to some few MPa, definitely confirms the recovery of mechanical integrity over a time period of only 18 years which is also documented by the mechanical back-analysis of the rock burst (Figure 10 – right side).

6 Concluding Summary and Recommendations

Summarizing the experimental test results the general characteristics of pressure driven gas transport in salt rocks has been convincingly enlightened, whereby in lab and field tests qualitatively the same phenomena are observed but scale effect are obvious. The outstanding observation is that if the gas pressure approaches σ_{min} the further pressure increase is diminished for a wide range of gas injection rates, due to the coeval permeability increase at the gas threshold. This permeation process is independently from the amplitude of σ_{min}. The relevant incident is the order of permeability increase during the gas threshold, as described by the parameter b of the new developed relationship (compare Figure 7). Only if the enhanced gas transport capacity of the surrounding salt is exceeded a further increase of pressure becomes likely (as confirmed by numerical simulations of the field tests) which could result in pneumatic fracturing if overpressures in the order of several MPa are reached. However, in the field tests no pressure induced micro-seismic activity was observed during the gas-breakthrough which clearly contradicts the pneumatic gas-frac scenario at realistic pressurisation rates. We believe that only local widening of bottle-necks or linking-up of preexisting pathways such as grain boundaries, causes the observed increase of permeability during the pressure threshold as may inferred from microstructural observations. This mechanism is generally described as "secondary permeability" which is not accompanied with a measurable increase in porosity.

Coevally with the permeability a pressure decay will occur in the reservoir resulting in a quasi-elastic closure of the prior opened path ways and thus to a recovery of hydraulic integrity. Because the observed permeability reversibility can be understood as "self sealing" this process may act as a "safety valve" if a gas pressure increase in salt occurs.

Beyond this, the self-healing capacity of rock salt is also convincingly demonstrated in the macroscopical scale of a natural analogue by the analyses of the post-gas-frac situation at the Merkers site. After the disastrous event of barrier fracturisation due to CO_2-expulsion a time dependent recovery of the mechanical **and** hydraulical barrier integrity is demonstrated, at least partly during a short period of only 18 years. In addition, it has to be mentioned that as well the rock burst itself and the associated event of a gas-frac, and the subsequent healing could be simulated by the performed rock mechanical back-analysis of this scenario.

In consequence, the risk of expulsion of radioactive gases or fluids from the repository through pneumatic fracturing is assessed to be low as long as the pressure build up-rates are compensated by permeation into the salt barrier. Preliminary modelling calculations (not shown here) document that for conditions specific to a German radioactive waste repository an increase of gas pressure up to the minimal stress is not excluded but even in this case no critical overpressurisation will occur. The results of lab and field tests show that, if the pressure build-up equalizes the critical stress state in the salt a rapid permeation process will take place ($\Rightarrow$ secondary permeability). Due to the proven integral permeability increase the range of tolerable pressure build up rates is significantly extended.

Evaluating our findings we can conclude that the gas-frac scenario resulting from an absolute tightness of rock salt becomes unlikely, which requires a new handling of gas pressure buildup in long-term assessments of a radioactive waste repository in salt formations. Because pressurized fluids can permeate into the salt, for future proof of safe enclosure the distance and extent of the permeation zone must be proven which depends on the gas storage capacity of the salt, i.e. porosity and mean pore pressure. For adequate modelling of radionuclide release it must be kept in mind that local circumstances often play a major role:

- There is both experimental and geological evidence that lithological intercalations inside the salt, e.g. anhydrite bearing layers or bedding planes can act as preferential path ways which could localize gas flow, e. g. [2].
- The in situ stress is, in addition to the design of the repository, the most significant factor determining the critical stress state of pressure controlled permeation. In many cases, i.e. bedded salt, however, it is adequate to assume that its magnitude is determined by the overburden and Poisson's ratio [10]. In contrast, under domal salt conditions the in situ stress in the salt can be different, i.e. the three principal stresses are not equal due to the fact that salt driven by its lower density flowed upward toform diapirs and pierced overlying units.

Finally, it is worth to note that hydrogen amounts and build-up rates strongly depend on the available amount of water, which is in dry rock salt rather limited contradicting generation of large amounts of gas for initiating critical pressures. As argued by [11] only if the geomechanical barrier-integrity is violated, i.e. proven by the minimal stress and the dilatancy criterion, a necessary inflow of water can occur. ■

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2A.10 Long-term Geochemical Evolution of HAW Glass under Repository Conditions: Achievements and Perspectives

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During the last 10-15 years it could be demonstrated that virtified high-level liquid radioactive waste is a suitable waste matrix for geological disposal. Without contact to groundwater HLW borosilicate glass provides a stable matrix over extended periods of time. However, upon contact with groundwater complex corrosion/dissolution/reprecipitation processes occur. The geochemical conditions of the groundwater has a siginifcant impact on the corrosion behaviour. Furthermore, due to the release of glass components and the formation of secondary phases, the geochemical conditions may change. Since the near-field environment of a waste repository system may involve several barriers within a multi-barrier system, the corrosion behaviour of the other barrier components (e.g. a steel container, backfill material, etc.) interact with the corrosion behaviour of the HLW glass. As a consequence, a highly coupled system with numerous simultaneous chemical reactions evolves.

The long-term corrosion behaviour of simulated and high-active glass R7T7 and of the simulated HLW glass GP WAK1 has been investigated in several EU projects by numerous corrosion tests in brines. More recently the focus has been on corrosion tests in synthetic clay pore solution. A mechanistic understanding has been derived for some aspects related to the interaction of HLW glass and ground water, including initial ion exchange reactions, element release from the glass matrix including empirical rate laws, the formation of a Si rich gel layer, Si saturation effects on glass corrosion and to some extend the formation of secondary phases. Over geolgical time scales a waste repository system is a dynamic system from a geochemistry perspective. In summary one could state that the key corrosion processes of HLW glass with resepct to deep geological disposal have been identified and to a large extend described empirically. Extrapolating these observations over geological time scales is possible with certain limitation for some processes. The reliablity of such extrapolations can be improved by understanding the involved reactions on a molecular level.

The formation of crystalline and amorphous secondary phases is one key process which controlls the mobility / the release of radionuclides and has been studied very recently in more detail. Some of these newly formed phases are mobile colloids whereas others are stationary (immobile) linked to the corroded waste matrix. Many radionuclides have a high affinity to these newly formed phases which are typically described as sorption phenomena and quantified by a KD value. However, on a molecular level many distinct sorption reactions occur. Most of them have not been identified umambigously. Also, the secondary phases which have been observed in long-term corrosion experiments do not neccesarily represent thermodynmically stable phases, but rather represent kintically controlled metastable precursor phases.

Here, the structural incorporation of trivalent actinides into the key secondary phases powellite ($CaMoO_4$) and hectorite (a smectite like clay mineral) will be discussed in more detail. The formation and stability of the $Ca_2(MoO_4)_2$ $NaAn(III)MoO_4$ has been studied using powder x-ray diffraction (XRD), extended x-ray absorption finestructure spectroscopy (EXFAS) and time resolved laser fluorescence spectroscopy (TRLFS) measurements in order to characterize long-range as well as short range ordering phenomena. Further solid solution – aqueous solution equlibria have been quantified on the basis of mixed-flow-reactor experiments. Similarily, the clay mineral hectorite was synthezised in the presence of trivalent actinides. For the first time, it could be shown that trivalent actinides can be incorporated into the octahedral layer of the clay host structure on the basis of EXFAS and in particular TRLFS measurements.

The combination of state of the art synthesis procedures under well defined boundary conditions and advanced spectroscopic techniques provides robust baseline information for deriving thermodynamic data on solid solution – aqueous solution equilibria. Such thermodynamic data are a prerequisite for including the formation of solid solution in geochmical modell calculation. With the availablity of thermodynamic data of the studied system (e.g. the multibarrier system), predictive capabilities with respect to modelling alteration process will be significantly improved. As a consequence uncertainties for the safety case will be reduced. ■

2A.11 CORALUS: An Integrated In Situ Corrosion Test on α-Active HLW Glass

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Abstract

Integrated in situ corrosion tests with α-doped SON68 (R7T7) glass samples have been performed in the HADES underground research facility in Mol, Belgium. The results on glass corrosion (effect of temperature, effect of addition of powdered glass frit, glass alteration mechanism) relate very well to the expectations. Taking into account the loss of part of the alteration layer of the in situ corroded samples and the fact that the hypotheses in the model are not always fully valid, the results agree also well with results from surface laboratory tests and modelling predictions with LIXIVER 2. The α-doped glass samples showed slightly thicker alteration layers, and the thickness seemed to increase with increasing $\alpha\beta\gamma$-activity of the sample. In combination with the decreased glass alteration, the addition of powdered glass frit to a Ca-bentonite decreased also the leaching of Np, Pu, and Am. In many cases, the migration profiles of these radionuclides showed a contribution of colloidal transport. Under the thermal gradient in the test tubes, the reaction of Ca-bentonite with the powdered glass frit resulted in the neo-formation of non-swelling 7Å minerals. The relevance of the results in view of the performance of a final repository for HLW glass is discussed.

1 Introduction

As part of the studies on the long-term performance of the French R7T7 HLW glass in a deep underground repository, we have designed and operated during the last 10 years the CORALUS tests (CORrosion of alpha-Active gLass in Underground Storage conditions). These are integrated in situ tests on the alteration of the SON68 reference glass* in conditions that are representative for those expected to prevail in a disposal system with a clay-based backfill [1-4]. In particular, the tests include(d) highly active alpha-doped glass samples, backfill materials exerting a high swelling pressure, contact with the clay host formation, controlled temperature, and the presence of γ radiation. The results of these tests were to be compared with results and insights from surface laboratory experiments and with results of modelling predictions. In 2004 we have dismantled two test tubes. In this paper, we report the most important results on the operation of the test tubes, on glass dissolution and clay alteration, and on leaching and migration of Np, Pu, Am, Th, Zr, Pd, U, and Cs.

2 The Coralus Project

In this part the CORALUS project is briefly summarised. A more detailed description can be found in [1-4]. The CORALUS in situ tests comprise four modular test tubes, placed in the Boom Clay through the underground research facility HADES (Mol, Belgium), for durations up to 10 years (Table 1).

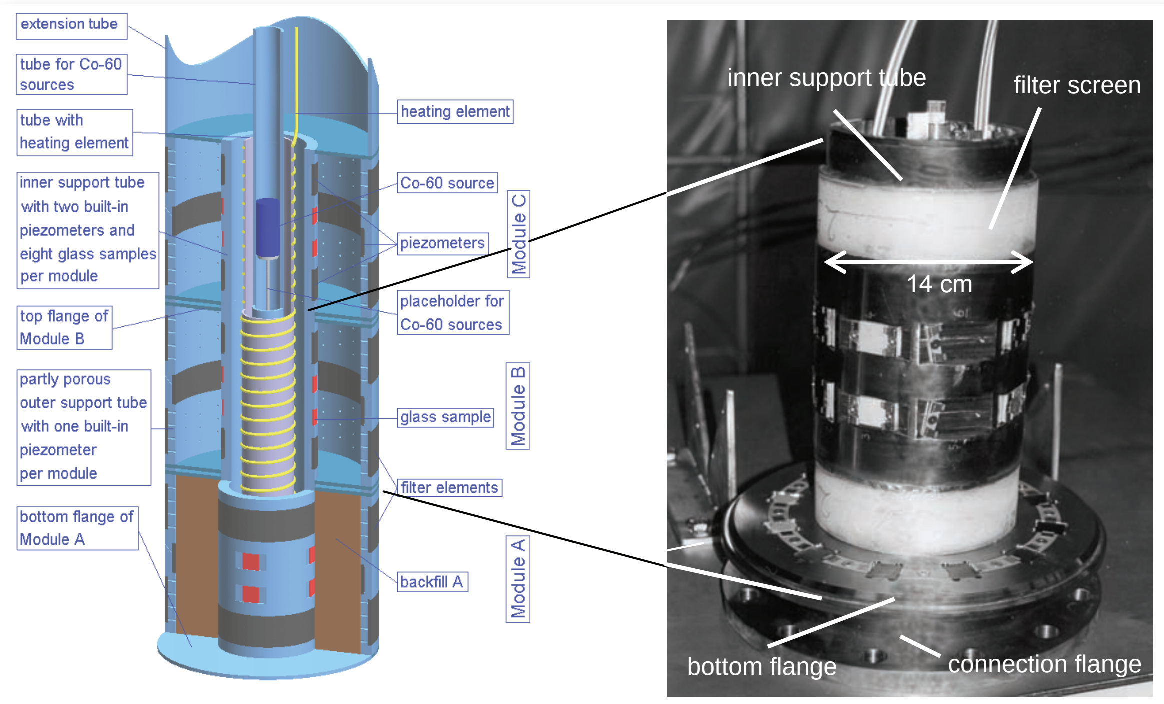
*SON 68 is the abbreviation of SON 68 18 17 L1C2A2Z1, a reference glass with a composition simulating the nominal composition of the AREVA NC R7T7 glass that is used for the vitrification of high-level and long-lived radioactive waste. Its main constituents are (wt%) 45.5 SiO_2, 4.9 Al_2O_3, 14.0 B_2O_3, 11.8 $Na_2O + Li_2O$, 4.0 CaO, 2.9 Fe_2O_3, 2.65 ZrO_2, 2.5 ZnO, 9.65 fission product oxides, 0.85 actinide oxides, 2.25 other elements.

Table 1: Experimental matrix and time-schedule for the CORALUS in situ tests

	Installation*	Operation	Duration (year)
Tube 2 (30 °C, no ^{60}Co)	2000.04.10	2001.02.21 - 2004.06.01	3.3
Tube 3 (90 °C, with ^{60}Co)	2001.04.25	2002.12.05 - 2004.02.20	1.3
Tube 4 (90 °C, with ^{60}Co)	2001.10.25	2003.04.15 - 2009.04.15	6
Tube 5 (30 °C, no ^{60}Co)	2003.11.06	2004.09.01 - 2014.08.31	10

*The long time between installation and operation is due to the slow convergence of the Boom Clay around the test tube and the slow saturation rate of the dry pre-compacted backfill materials.

Each test tube consists of three modules. In each module, inactive and alpha-doped SON 68 glass samples, placed on an inner support tube, are in contact with a backfill exerting a design swelling pressure of 2 MPa. Built-in piezometers allow to sample interstitial solution. Within the inner support tube a heating element and – for two of the four test tubes – ^{60}Co sources can be placed. A three-dimensional cut-away view of a test tube with ^{60}Co sources is given in Figure 1, together with a picture of part of a module during its assembly. Backfill materials were emplaced during the assembly of the test tube as dry pre-compacted half-cylindrical blocks. Hydration of the blocks was achieved with Boom Clay pore water infiltrating via the partly porous outer support tube.



▲ Fig. 1: (left) Three-dimensional cut-away view of a test tube with ^{60}Co sources. (right) Inner support tube, bottom flange, and connection flange of one module. The photo shows the position of the samples (SON68 and other glass types, and container materials) on the inner support tube and on the flange, and the filter screens and connecting tubes of the inner piezometers (in grey).

Radioactive glass samples contain ~0.85 wt% of $^{237}NpO_2$ (0.22 MBq/g ^{237}Np), $^{238\text{-}242}PuO_2$, (27 MBq/g $^{239/240}Pu$), or $^{241}Am_2O_3$ (1 GBq/g ^{241}Am). Inactive glass samples contain ~0.37 weight% of $(U,Th)O_2$, and inactive isotopes, or chemically similar elements, of other critical long-lived nuclides such as ^{135}Cs, ^{79}Se, ^{93}Zr, ^{107}Pd. The three backfill materials are (module A) **d**ried **B**oom **C**lay ("**DBC**"), (module B) Ca-**b**entonite mixed with **s**and and **g**raphite (60/35/5 wt%; "**BSG**"), and (module C) Ca-**b**entonite mixed with powdered SON68 **g**lass **f**rit (95/5 wt%; "**BGF**"). The addition of the glass frit aims at imposing high concentrations of the main glass constituents (Si, Al, B, Na, Ca, Zr, Li, and Zn), to reduce the chemical potential difference of these elements – especially Si – between glass and backfill, thus decreasing the glass dissolution rate [5].

Two test tubes ('tubes 2 and 5') are operated at 30°C, in the absence of γ radiation. They simulate the normal evolution scenario conditions beyond the thermal period. Test tubes 3 and 4 contain ^{60}Co sources (~100-130 Gy/h) and are operated at 90 °C. These conditions represent an early failure scenario, occurring ~50 years after disposal of the waste.

Complementary surface laboratory experiments were performed to be able to better interpret the results of the in situ tests [1,2]. Integrated glass corrosion tests with inactive samples have been performed under the conditions of the in situ tests, except for the presence of γ radiation, to provide reference data for comparison. In another test, the thermal gas generation in the three water saturated backfill materials at 90 °C was measured as a function of time, up to 1000 days. Additionally, modelling predictions were performed with the CEA LIXIVER 2 code [2,6], to provide estimates for the specific conditions of the in situ tests.

3 Results and Discussion

3.1 Operation and Dismantling of the Test Tubes

Saturation of the 35 mm thick backfill in the test tubes without ^{60}Co sources was achieved after ~6 months. The tubes were then heated to 30 °C. For the saturated "dried Boom Clay" backfill in tube 2, the effective permeability to water was ~ 5×10^{-19} m², which agrees with the value for slightly disturbed Boom Clay [7]. Values for the bentonite-based materials were of the same order of magnitude. Hydration of the 1.5 times thicker backfill in the test tubes with ^{60}Co sources proceeded very slowly, and even 1.5 year after installation full saturation was not yet achieved. Hence, after installation of the ^{60}Co sources, heating to 90 °C was started. We assumed that, in analogy with the results of other tests on backfill material behaviour [8], about 95 % of the saturation was already achieved. Yet, for module B (BSG) of test tube 3, it still took another 8 months to achieve full saturation. There is some evidence that for this module, the glass was not all the time in (intense) contact with the clay, locally resulting in smaller glass dissolution and radionuclide leaching and migration.

α spectrometry revealed the presence of up to 0.26 Bq/l of ^{241}Am and up to 0.01 Bq/l $^{238\text{-}240}Pu$ in the piezometer solutions sampled before heating, i.e. when the clay had not yet sufficiently swelled to fill the initial small gaps. Later, the activity of ^{241}Am and $^{238\text{-}240}Pu$ was below or around the detection limit of 0.01 – 0.05 Bq/l, demonstrating that after saturation the swelled backfill materials indeed constitute an effective seal against radionuclide migration.

The composition of the interstitial solutions of all three backfill materials is fingerprinted by the high Na_2SO_4 concentration in the partially oxidised Boom Clay surrounding the test tube (Table 2). The solutions of module A (DBC) contain also high Ca^{2+} concentrations due to pyrite oxidation and subsequent $CaCO_3$ dissolution [9]. The high Ca^{2+} concentrations in modules B (BSG) and C (BGF) are due to Na^+-Ca^{2+} ion exchange in the Ca-bentonite of these modules. Further differences were related to the type of backfill material (e.g. high B and Si concentration in module C). CO_2 was the major dissolved gas in all solutions (see below) [2,3]. CH_4 was found in small concentrations in a few solutions of tube 2, pointing to the presence of methanogenic bacteria in the piezometer [10].

By connecting flow-through cells with pressure-resistant electrodes to the piezometers, pH and E_h of the piezometer solutions could be measured on-line. The pH in the solutions of tube 2 (30 °C) was neutral to slightly alkaline (6.8 to 7.9), and the redox conditions were reducing (-170 to -350 mV SHE) [3]. pH values for tube 3 (90 °C) were considerably lower – for modules A and B almost 2 pH units – than for tube 2 (30 °C). This is very probably due to the 3 to 5 times higher concentration of dissolved CO_2 in tube 3. These higher CO_2 contents stem from the greater decomposition of organic matter (kerogen fraction in module A, DBC, [11]) and graphite (module B, BSG) at high temperature (this was confirmed by the surface laboratory experiment on thermal gas generation [2]) and under γ radiation [12]. Redox potentials were less reducing (0 to –80 mV SHE), which is in line with the lower pH.

Table 2: Concentrations (mg/l) of major cations and anions and total organic carbon (TOC) in the inner piezometer solutions that were sampled shortly after start of heating

	Tube 2 (30 °C, no ^{60}Co sources)			Tube 3/Tube4 (90 °C, ^{60}Co sources)		
Module→	A	B	C	A	B	C
Na^+	2700	350	560	1660/1930	410/283	680/550
Ca^{2+}	475	500	600	270/360	100/470	310/470
B	13	<2	530	38/42	9/<2	2280/1430
Si	31	<13	15	8/15	9/6	61/26
SO_4^{2-}	7000	1750	2260	4340/4930	546/1410	1530/2070
TOC	660	70	220	n.m./660	n.m./129	n.m./160

n.m. = not measured. Concentrations were determined after filtration over a pre-washed 0.45 µm membrane filter. The concentrations in the solutions that were sampled later were mostly slightly lower.

Test tubes were retrieved by over-coring. More details are given in [1-4]. Because of the high stresses (temperature, swelling pressure), about 65 % (tube 2) and 90 % (tube 3) of the glass samples were broken. The clay on the glass samples was removed with a small brush, after immersing the sample in demineralised water. It cannot be excluded that (part of) the alteration layer was damaged or even removed due to this treatment.

3.2 Glass Alteration

The analyses on the corroded glass samples (mass loss, SEM and EDS of surface and cross section, SIMS) allowed to conclude that both the inactive and the radioactive glass samples generally behaved as could be expected on the basis of the existing insights on the alteration of SON68 HLW glass in contact with clay.

3.2.1 Glass Dissolution: Mass Loss and Thickness of the Alteration Layer

The glass alteration at 90 °C is considerably higher than at 30 °C. We measured differences in specific daily mass loss of a factor 20 (Table 3). The alteration layers were up to 20 times thicker for the alteration during 1.3 years at 90 °C compared to the alteration during 3.3 years at 30 °C (Table 4). Avoiding contact between the HLW glass and water during the thermal phase could therefore avoid the “fast” dissolution (i.e. within the first 300 years) of up to ~15 weight% (~60 kg) of HLW glass (assuming a constant glass dissolution rate of 0.3 g/m²/day during the first 300 years, and neglecting the presence of fractures in the HLW glass monolith). At long-term glass dissolution rates of respectively 0.01 and 0.001 g/m²/day, it takes about 10,000 years and 100,000 years, respectively, to dissolve 60 kg of HLW glass (again neglecting the presence of fractures in the HLW glass monolith).

The results in this table should be interpreted with caution, because (i) on all glass samples very small quantities of clay (as a thin “film” layer) were still present (the results for the glass samples in module C (BGF) of test tube 2 may give an indication of the weight of this clay film), and (ii) we cannot exclude that very small glass fragments split off during the test or manipulations. However, in case the two duplicate samples were not broken (n=2), the difference between the two values was smaller than 15 %. This may indicate that the results are possibly still fairly reliable. The value for the Np-doped glass in DBC of test tube 2 is not realistic.

The lower and upper values give the range of thicknesses observed per sample. The central value gives the estimated average thickness, with an estimate of the uncertainty on this average. Especially for the radioactive samples and for the samples altered

Table 3: (Average) specific daily mass loss (g/m²/day) of the SON68 glass samples

		Tube 2 (30°C)	Tube 3 (90°C)			Tube 2 (30°C)	Tube 3 (90°C)
DBC (A)	U/Th	0.0093 (n = 2)	no data	BSG (B)	U/Th	0.0075 (n = 2)	No data
DBC (A)	Np	0.0008 (n = 1)	no data	BGF (C)	U/Th	-0.0019 (n = 2)	No data
DBC (A)	Am	0.0143 (n = 1)	0.301 (n = 2)	BGF (C)	Np	0.0013 (n=1)	No data

at 90°C, part of the alteration layer was lost, either because of the strong interaction with the (removed) clay or by the cleaning treatment; see below. This hampered the interpretation of these results.

For DBC (module A), the results on specific daily mass loss – except for Np (outlier) – correspond very well with the values obtained from integrated surface laboratory tests for the same conditions in undisturbed Boom Clay (i.e., 0.01 g/m²/day at 30 °C, and 0.3 – 0.5 g/m²/day at 90 °C but without gamma irradiation; test durations up to 2.5 years) [13-15]. We therefore conclude that if other processes such as the thermal decomposition of the natural organic matter in the backfill material result in an important decrease of the pH (see ‚operation' section), the contribution of gamma radiation to a lower pH and concomitant decreased initial glass alteration rate [9,16,17] may become negligible.

Table 4: Gel thicknesses (µm) measured on the glass samples of the in situ tests

	Tube 2 (30 °C, no ^{60}Co, 3.3 years)			Tube 3 (90 °C, ^{60}Co, 1.3 years)		
Mod	A (DBC)	B (BSG)	C (BGF)	A (DBC)	B (BSG)	C (BGF)
U/Th	0.5<1.5±0.5<5	0.5<2.3±0.3<5	0.2±0.1	10<25±3<55	15<20±5<50	1<2.5±0.5<5
Np	3<3.7±0.3<6	2<3.5±0.5<10	<0.5	30<35±3<50	55<80±10<130*	1<2.6±0.5<5
Pu	3<3.5±0.5<12	2<3.0±1<12	<0.5	30<40±10<50	45<50±5<55	2.3<2.8±0.5<3.3
Am	5<6.0±0.5<12	3<4.0±1<15	<0.5	40<45±3<55	45<50±10<90	gel layer lost

* In contrast to the other samples, the alteration layer of the Np doped sample in Module B (BSG) was intact, probably because of the weak contact between the glass and the backfill material (see text).

3.2.2 Effect of Addition of Powdered Glass Frit

The addition of powdered glass frit to the backfill material diminishes the glass dissolution with one order of magnitude or more (Tables 3 and 4). The effect of powdered glass frit occurs most probably through the rapid establishment of high solution concentrations of the main glass constituents – especially Si – , and by concomitant rapid saturation of sorption sites on the backfill material close to the surface of the radioactive waste glass [5].

3.2.3 Glass Alteration Mechanism

SIMS profiles – normalised to Zr, which proved to be rather stable in both pristine and altered glass – showed the leaching of Li^+, B, Na^+, Cs^+, and – but less pronounced – Sr^{2+}, Ba^{2+}, and REE; the influx of H^+, K^+, and Mg^{2+} from the clay (pore water), and the stability of Si, Al, and Fe. Ca^{2+} proved to be rather stable, probably because of the high Ca^{2+} pore water concentration. The profiles of Th, Np, Pu, and Am showed that these elements were stable in (that part of) the alteration layer (that was not lost). U, in contrast, seemed to be more easily leached. The loss of part of the alteration layer does not allow to present an overall picture of the distribution of these radionuclides in the alteration layer. Yet, the results of the radionuclide migration enable to calculate radionuclide retention factors – defined as the ratio of the mass loss and the normalised radionuclide release – which vary between 1 and 3. These values are similar to values reported by Lemmens et al. [18] for glass corrosion tests in contact with dense clay. They are much smaller than the values normally reported for glass alteration in water in the absence of solids (5 – 10 for Np, 50 – 100 for Pu, and 500 – 1000 for Am) [19]. This is due to the high affinity of the clay-based backfill material for the radionuclides.

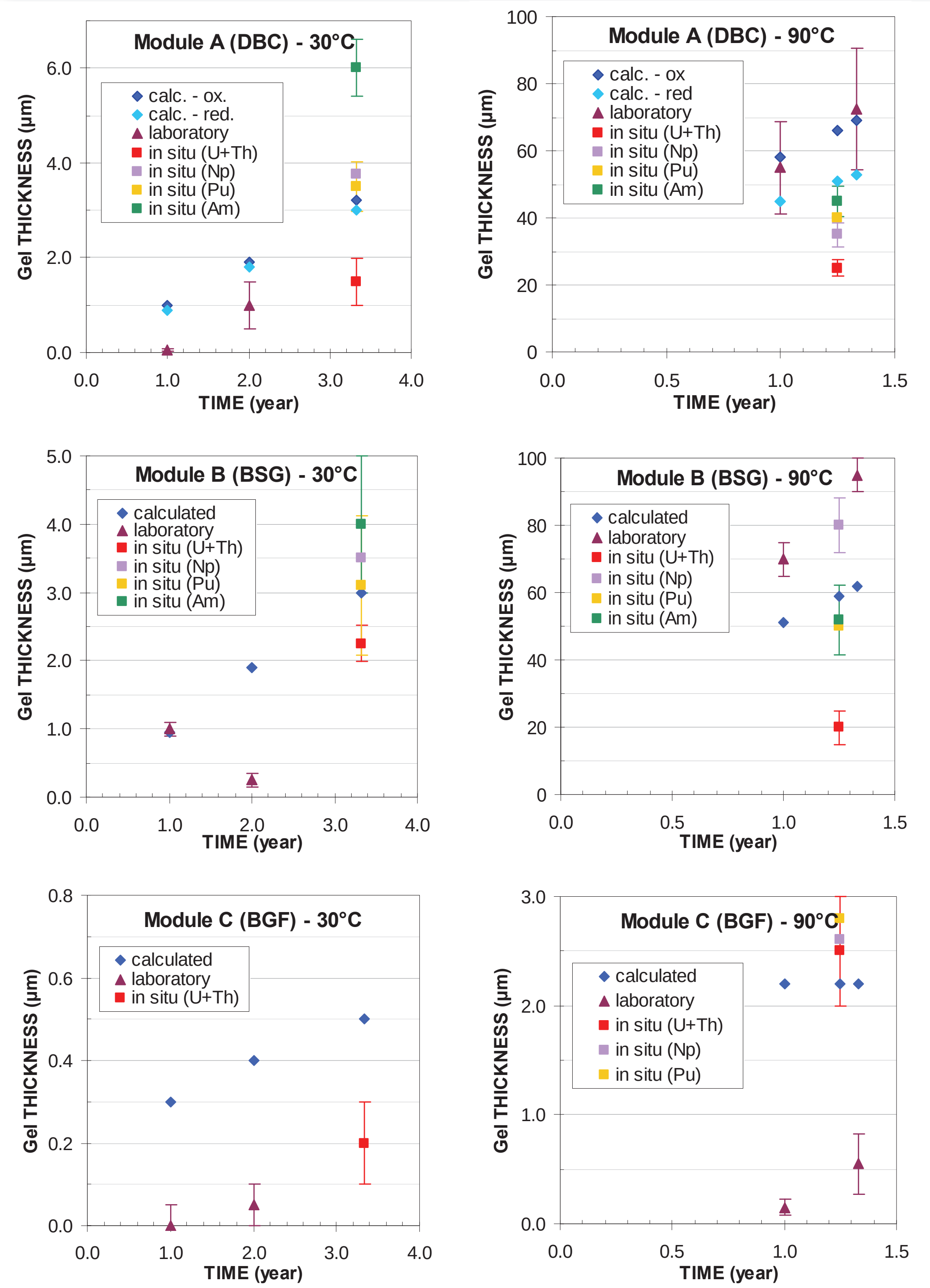
Taking into account the nearly complete elimination of B, Li^+, Na^+, and Cs^+ and the influx of H^+, K^+, and Mg^{2+} in the alteration layer of the SIMS profiles, we conclude that the dissolution of the SON 68 samples is predominantly selective-substitutional leaching, as observed for other SON 68 glass alteration tests [17,20]. Yet, the loss of a distinct part ("layer") of the alteration layer, either because of the sorption of the glass constituents onto the clay and/or secondary phase formation, or because of the treatment of the glass samples, can be interpreted as matrix dissolution. We hypothesise that this loss is due to the extensive matrix dissolution of (the most leached part of) the alteration layer in combination with the very high swelling pressure (1.5 – 3.5 MPa) exerted by the backfill material on the alteration layer. Given the low density and the high water content of the most leached part of the alteration (or: gel) layer, it is not inconceivable that the swelling clay intruded this gel to some depth. It is striking that this loss of part of the alteration layer was observed for the in situ tests but not for the laboratory tests with the same (target) density – and hence swelling pressure – of the backfill.

Considering the loss of (part of) the alteration layer of the samples of the in situ tests, and the fact that for the surface laboratory tests the alteration layer thickness was in some cases unreasonably low [2], the thicknesses of the alteration layers of the non-radioactive samples generally compared well with those of the surface laboratory tests (Figure 2). The results of the in situ tests and surface laboratory tests agreed also fairly well with the predictions of the LIXIVER 2 model (Figure 2). The LIXIVER 2 kinetic model couples the first-order rate law in which dissolved silica plays a major role, with silica diffusion through the alteration layer, and assumes that all altered glass remains in the gel layer (i.e. iso-volumetric alteration, with no reaction with the clay) [6]. For systems where part of the alteration layer has reacted with the clay, which is the case for the modules A (DBC) and B (BSG) in our tests, this assumption is not valid. Therefore, the calculations for these cases have to be considered as indicative.

3.2.4 Effect of Alpha-Beta-Gamma Activity

Despite the good agreement with expectations and results of laboratory tests and modelling predictions, there seem to be small but distinct differences between non-radioactive and radioactive glass samples. SEM analysis of surface and polished cross section revealed that the radioactive samples were more severely attacked. SEM images of the altered surface showed the inactive samples of modules A (DBC) and B (BSG) at 30 °C to be only slightly altered, with the surface being homogeneously covered with isolated spots of limited alteration, and with polishing lines still visible (Figure 3). In contrast, radioactive samples appeared to be more severely altered, and traces of polishing lines were no longer visible (Figure 3). Because of the lack of sufficient data (Table 3), we cannot conclude that is related to an increased mass loss. As expected, the alteration of the samples in contact with the BGF (module C) at 30 °C was very limited. At 90 °C, the glass samples from modules A (DBC) and B (BSG) were much more altered, and no difference could be seen between non-radioactive and radioactive samples (Figure 3). For the samples of module C (BGF), the surface alteration was again very limited. SEM analysis of the polished cross section showed that for modules A (DBC) and B (BSG) the alteration layers of the radioactive samples were thicker than for the inactive samples (Figure 3), and the thickness seems to increase with increasing $\alpha\beta\gamma$ activity of the samples (Np < Pu < Am; Table 4). The difference is not so large: a factor 4 at most. Also for module C (BGF) at 90 °C there is a tendency to thicker alteration layers for the active samples, but it is statistically not significant.

The observation of thicker alteration layers for the radioactive samples was thus also made for the samples from test tube 3 (90 °C, with ^{60}Co sources), for which the pH was already considerably lower, and for which the γ radiation dose rate was already very high (~130 Gy/h) and similar for both inactive and radioactive samples. pH effects and/or effects by $\beta\gamma$ radiation (affecting the interconnectivity in the alteration layer) were invoked by Advocat et al. to explain the slower decrease of the initial glass alteration rate observed for leach tests with highly $\alpha\beta\gamma$ active glass samples in contact with initially pure water [21]. Therefore, we conclude that this effect is (at least partly) due to the alpha activity of the samples. This conclusion differs from the conclusions from glass alteration tests in water or suspensions with samples doped exclusively with alpha emitters [22]. The reason for our observations is probably the high water and/or swelling pressures prevailing during the glass alteration, in combination with the increasing α and $\beta\gamma$ activity of the samples.



▲ Fig. 2: Comparison of calculated and measured (surface laboratory and in situ experiments) gel thicknesses for module A (DBC; top), module B (BSG; centre), and module C (BGF; bottom) at 30 °C (left) and 90 °C (right). For DBC (top), separate calculations were made for reducing and oxidising conditions (referred to as calc. red. and calc. ox., respectively). Values use in the model are (for the forward dissolution rate, g/m²/day) 1.0 (90 °C) and 7.5×10^{-3} (30 °C), (for the Si saturation concentration, M) $10^{-2.7}$ (90 °C) and $10^{-3.01}$ (30 °C), and (for the apparent Si diffusion coefficient through the alteration layer, m²/s) 5×10^{-13} (DBC, 30 and 90 °C), 5×10^{-13} (BSG, 30 and 90 °C), and 1×10^{-15} (BGF, 30 and 90 °C).

3.3 Radionuclide Migration

The migration profiles of Np, Pu, and Am at 30 and 90 °C are shown in Figure 4. These results warrant the following comments.

- Highly compacted clay-based backfill materials with good radionuclide sorption properties are efficient in retarding radionuclide migration. For all test combinations, the specific activity of Np, Pu, and Am decreased by several orders of magnitude over a distance of ~5 mm.

- A contribution of colloidal transport of Am and Pu can be observed in modules A (DBC) and B (BSG) (for a pure diffusion-dominated transport, we expect (one side of) a Gausian curve). These are probably radionuclide–organic matter pseudo colloids, formed by adsorption of radionuclides onto existing colloids in the backfill pore water or onto freshly formed glass alteration products [23]. They are expected to become unstable over larger distances [24]. This destabilisation theory is supported by the negligibly low concentrations of these elements in the piezometer solutions (see “operation” section).

- The addition of powdered glass frit to the backfill material reduces considerably the glass alteration and, as a consequence, the radionuclide leaching. In five of the six cases, this resulted also in a smaller radionuclide dispersion (migration) in the surrounding backfill.

The non-radioactive glass samples contained a.o. U (presumably as U(VI)), Th(IV), Zr (Zr^{4+}), Pd (Pd^{2+}), and Cs (Cs^{+}). Analysis of the sliced clay cores (extraction followed by ICP-MS) showed that Zr^{4+}, Th^{4+}, and Pd^{2+} , which easily precipitate or sorb onto the clay, did not migrate far (< 5 mm). In contrast, U(VI/IV) – which easily forms negatively charged carbonate complexes such as $UO_2(CO_3)_2^{2-}$ (lower pH) or $UO_2(CO_3)_3^{4-}$ (higher pH) – and Cs^{+} – which has to compete with the high Na^{+} concentration for the Cs^{+} specific sorption sites on clay minerals – migrated over larger distances (up to 10 mm for U, up to 20 mm for Cs).

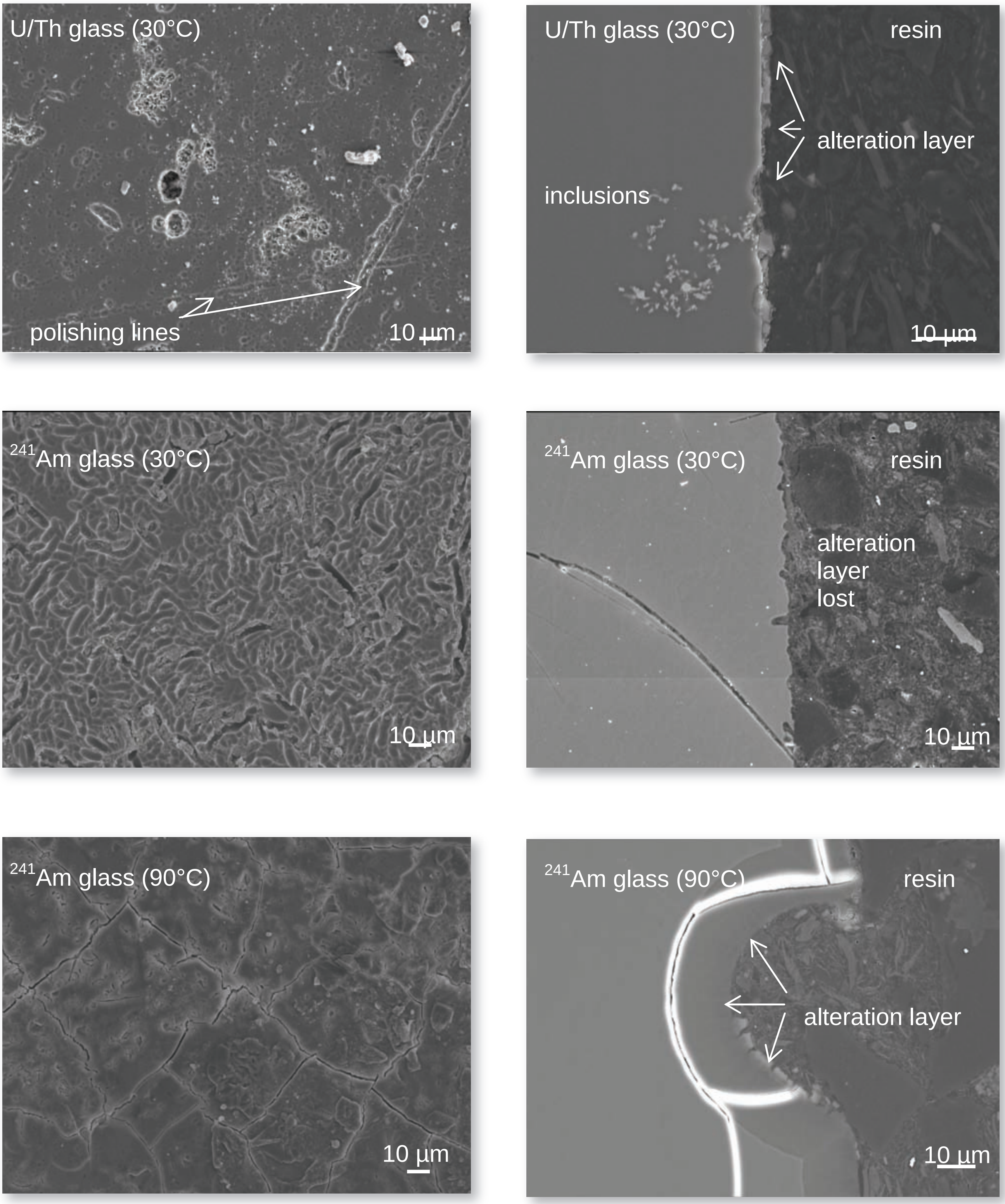
3.4 Clay Alteration

The characterisation of the Ca-bentonite with powdered glass frit (ESEM, SEM-EDS, TEM, XRD, and FTIR) showed that after ~1.3 years of heating to 90 °C and γ irradiation, the interface between clay and glass (both the glass samples and the glass powder in the backfill) was fully reactive. The most important process that occurred during the operational period is silica precipitation at the glass/clay interface [25]. Kaolinite and interstratified kaolinite/smectite of the Ca-bentonite had started to become unstable due to the high temperature and/or pH variation due to glass dissolution, and tended to form non-swelling 7Å minerals. This was also described for the same material under thermal gradient by Latrille et al. [26]. In our test, these minerals tended to be transformed at the hottest point to non-swelling 7Å minerals – type serpentine ($Mg_3Si_2O_5(OH)_4$), with Mg^{2+} coming from (degrading) interstratified minerals and from the (Boom Clay) pore water. This class of non-swelling 7Å minerals has also been observed in iron/clay interaction tests [27].

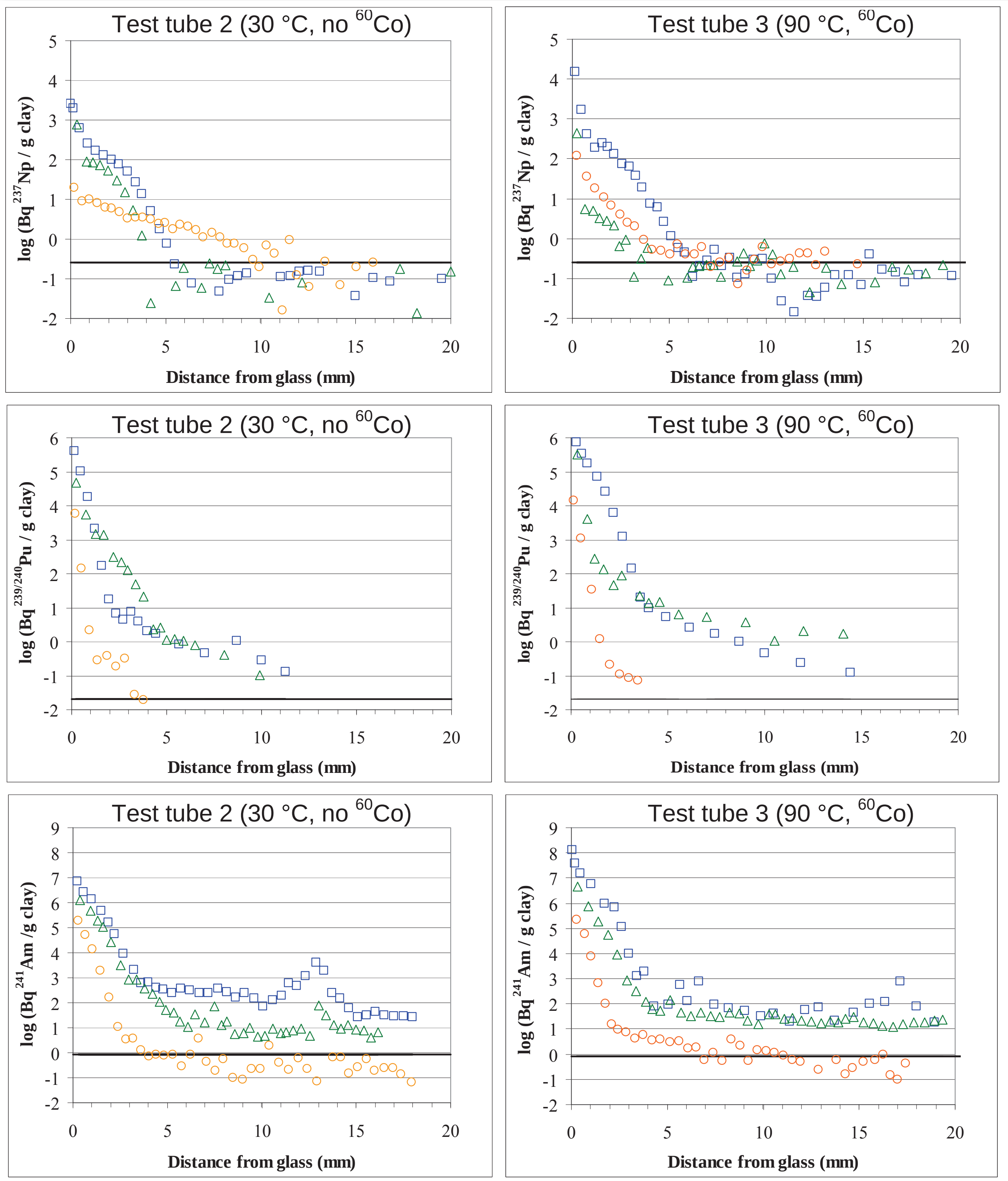
4 Discussion and Conclusions

The interpretation of all data of these integrated tests allows to conclude that the alteration behaviour of glass in realistic in situ conditions is very similar to its behaviour in surface laboratory tests (effect of temperature, effect of type of backfill material, alteration mechanism). The results compare also fairly well with the modelling predictions, even if the modelling hypotheses are not always fully valid.

Yet, the analyses of the glass samples revealed also some unexpected results. Although the α-doped glasses behave rather similarly as the inactive ones, we observed slightly thicker alteration layers for the active samples, with the thickness increasing with increasing $\alpha\beta\gamma$-activity of the samples. As explained, it seems that this effect is due to the effect of the increasing specific alpha activity, in combination with the very high swelling pressure of the backfill. This high pressure might result in a mechanical degradation of the less dense (i.e. the most leached part of the) alteration layer, explaining why for the in situ corroded samples part of the alteration layer was missing. It is striking that this loss of part of the alteration layer was only observed in the in situ



▲ Fig. 3: SEM observation of the surface (left) and the polished cross section (right) of in situ corroded glass samples of module A (DBC) at 30 °C (top + middle) and 90 °C (bottom). Top: U Th glass (30 °C); middle: ^{241}Am glass (30 °C); bottom: ^{241}Am glass (90 °C). The white line on the right figure in the lower row is an analytical artefact due to the fissuring of the gel. The alteration layer of the ^{241}Am doped glass of tube 2 is completely lost (no other layer between glass and resin).



▲ Fig. 4: Radionuclide migration profiles for Np (top), Pu (centre), and Am (bottom). Left: test tube 2 (3.3 years 30 °C, no ^{60}Co sources); Right: test tube 3 (13 years 90 °C, with ^{60}Co sources); (□) DBC, (△) BSG, (O) BGF. The horizontal black line represents the detection limit (0.25 Bq/g for ^{237}Np, 0.02 Bq/g for Pu, and 0.75 Bq/g for ^{241}Am).

tests, and not in the laboratory tests with the same (target) density – and hence swelling pressure – of the backfill materials. These somewhat "unexpected" results are not believed to have an important influence on the long-term HLW glass behaviour. Yet, they are of interest in that they provide information of the behaviour of HLW glass in realistic disposal conditions.

The results clearly show the high interest of avoiding contact of the HLW glass and formation water during the thermal phase, and the interest of adding powdered glass frit to keep the glass dissolution rate almost from the first contact with water to its low long-term residual dissolution rate. In view of the high reactivity of Si with a bentonite-based backfill material, resulting in the generation of non-swelling 7Å clay materials, it seems advisable to place the powdered glass frit within the water-tight "overpack" container that is designed to withstand contact with formation water during the thermal period. Such practice will avoid that Si reacts during several hundredths of years with the backfill material – possibly destroying the swelling capacity of this material – and it will assure that Si is available in the immediate neighbourhood of the HLW glass when it starts dissolving.

The fact that no important contamination was found in the piezometer solutions around the glass samples [3,4] demonstrates that clay-based backfill materials with good radionuclide adsorption properties, when saturated, constitute an effective seal against radionuclide migration. The effect of colloidal transport is possibly limited in space due to the destabilisation (dissociation) of the (pseudo)colloids.

On the basis of all results and observations, we conclude that the CORALUS integrated in situ tests largely contribute to the validation of the laboratory tests and modelling predictions. ■

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2A.12 Corrosion Behavior of Spent Fuel in Non-saline Solutions Experimental Studies at FZK/INE

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With respect to the assessment of the long-term behavior of the waste form spent fuel it is of high importance to understand in particular the UO_2 matrix dissolution behavior and the associated release or retention of mobilized radionuclides during an assumed contact with groundwater. Due to the radionuclide inventory of the fuel radiolysis of the groundwater will start, followed by oxidation and subsequent dissolution of the fuel. Additionally, the overall corrosion behavior of the fuel is governed by the geochemical constraints in the near field. Related to this objective we studied experimentally the impact of geochemical conditions possibly encountered in close vicinity to the spent fuel on fuel matrix dissolution under anoxic conditions. Special attention was focused on (1) the nature of solution contacting the matrix (5 M NaCl brine, granite water [low ionic strength, SiO_2-rich], granite bentonite water [high ionic strength, low in SiO_2]), the (2) presence/absence of CO_2, and (3) the presence or absence of corroding container material (Fe and Fe corrosion products). Pre washed powdered fuel samples (grain size > 3 µm) were used for these studies to achieve a faster reaction progress and to mask contributions from the grain boundary inventories. The fuel matrix dissolution rates were found to be within the same range in 5 M NaCl-brine, granite and granite bentonite water (about 10^{-4}/d). In the presence of iron and its corrosion products considerable amounts of H_2 were generated due to the corrosion of the added iron. In these cases, the matrix dissolution rates have been slowed down by a factor of 10 - 20, associated with strong retention effects for various radioelements. Studies upon spent fuel corrosion behavior in a clay environment are presently in the planning stage.

For performance assessment it is necessary to analyze, which fraction of mobilized safety relevant radionuclides is present in ionic, and which in colloidal form (increase of solution concentration beyond the thermodynamic solubility limit and ability to migration). In low ionic strength granite water the formation of possibly mobile colloidal fractions of actinides was found to a large extent. Colloid generation was reduced in granite-bentonite water due to higher ionic strength, as well as in the presence of iron powder. Almost no formation of colloids was observed in 5 M NaCl brine.

The independence of the dissolution rate of the UO_2 matrix upon the nature of solution (salt brine or granite water), pCO_2, can probably be explained by the fact that the controlling processes are governed mainly by effects of radiolysis, i.e. production of oxidative radiolysis products and their consumption for fuel oxidation and subsequent dissolution.

Moreover, it must be considered that the dissolution rates measured in the frame of this study can not be used directly for spent fuel performance predictions under repository conditions. Significantly lower dissolution rates will be resulting, because no pre oxidation of the fuel matrix is expected, radiation fields and the fuel surface area will be considerably lower than in these experiments.

2A.13 Interactions of Actinides with Hydroxyaluminosilicate Colloids in "Statu Nascendi"

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The formation of hydroxyaluminosilicate (HAS) colloids under near natural aquifer conditions and the incorporation of trivalent actinides in their oxo-bridged structure are studied. Colloid-borne actinides undergo migration in natural aquifer systems with little geochemical hindrance increasing the mobility of actinides. Hydroxyaluminosilicate (HAS) colloids are generated by heterogeneous nucleation in presence of trace amounts of actinides. The colloids are characterized by laser-induced breakdown detection (LIBD), AFM, EDX, and XPS. The particle size of colloids is in the range of 10 nm – 50 nm with a mass concentration of 10 – 50 ppb and a Si/Al ratio of 0.7-1.0. For speciation of the colloid-borne Cm(III), the nucleation process is investigated by time-resolved laser fluorescence spectroscopy (TRLFS) with the assistance of radiometry. Spectroscopic speciation shows the formation of two colloid-borne (surface-bound) Cm(III) species, Cm-HAS(I) and Cm-HAS(II) with peak maxima at 598.5 and 601.8 nm, respectively, and a third Cm(III) species with a peak maximum of 606.8 nm (Cm-HAS(III)) for pH > 6.3. The long fluorescence lifetime (τ = 518.5 µs) of the latter species proves that the Cm(III) has lost its primary hydration sphere and is imbedded into the molecular structure of the aluminosilicate colloids. The interaction of An(III), An(IV), An(V), and An(VI) are studied by X-ray absorption spectroscopy (EXAFS), using Eu(III), Th(IV), Np(V), and U(VI). Whereas An(V) does not form colloidal species, An(III), An(IV) and An(VI) are readily incorporated into HAS colloids. The stability of the colloids was studied by 2-D LIBD and TRLFS. Long-term measurements confirm the stability of the actinide containing HAS colloids under natural aquifer conditions (pH > 6). The present work gives new insights into the generation of colloid-borne actinide species in the near and/or far field of a given nuclear waste repository, which thus facilitates actinide migration. ■

2A.14 Self Sealing Backfill (SVV) — New Experimental Results

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A self sealing salt backfill (SVV) based on anhydrous magnesium sulphate has been developed and was patented by GRS. The results of a series of experiments on different scales with different typical boundary conditions for repositories in rock salt and potash salt formations demonstrate that SVV is well suited for the construction of seals in salt and potash mines. The backfill consists mainly of fine grained anhydrous $MgSO_4$ with a grain density of 2.63 g/cm^3. Other salt minerals such as rock salt and sylvite may be added in small amounts and varying proportions according to the desired final hydraulical and geochemical parameters of the SVV seal. The dry material can be emplaced pneumatically and reaches a bulk density between 1.35 and 1.40 g/cm^3. The initial pore volume of about 50% can be filled with brine. By artificial flooding of the pore space a reaction sets in that leads to the formation of a brine tight seal. After brine inflow a fast exothermic reaction sets in. As a result the brine is consumed completely and an impermeable new material with a new mineralogy and higher volume is formed.

The initial high porosity is reduced to 2-5 volume-% of isolated pores. At the beginning of the reaction temperatures rise up to 85 °C. These high temperatures however occur only for a couple of hours after the first contact of brine with the initial SVV material. Sharp temperature peaks start to build up immediately after brine inflow. The temperatures decrease to ambient values almost as fast as they have risen. The reaction SVV/brine leads to the formation of new hydrated salt minerals. Water from the intruding brine is consumed. The brine becomes oversaturated and precipitates further new minerals. These combined effects lead in the short run to a metastable mineral assemblage of the seal with a higher volume than the combined volume of the initial material and brine. Mineralogical assemblages obtained in the experiments and measured by X-ray diffraction agree well with results of geochemical modelling with EQ3/6. The geochemical modelling allows the quantification of the short and long term volume changes in the system and demonstrates that in the long run stable mineral paragenesis will be obtained. The volume increase due to the reaction SVV/brine leads to a considerable crystallisation pressure.

The actual values of the resulting pressures depend on the mineralogical composition in the seal after completion of the reaction. The mineralogy of the seal is dependant on the mineralogical composition of the initial SVV material, the composition of the initial solution, the achieved solid-solution ratio and the homogeneity of the fluid distribution after flooding. In laboratory experiments with different SVV materials and on different scales crystallisation pressures between 0.2 - 10 MPa have been obtained. Crystallisation pressures can vary in rather large margins within one sample indicating non homogeneous conditions. However in all cases the obtained crystallisation pressures were high enough in order to guarantee not only low mean permeabilities but also a tight contact with the host rock and a continuous reduction of the permeability of the excavation disturbed zone. The permeabilities obtained in laboratory experiments of different scales are low. They range between 10^{-18} and 10^{-21} m^2. Due to the crystallisation pressure the permeability tends to decrease in time. In large scale in-situ experiments in rock salt initial SVV permeabilities of 10^{-18} were measured. The porosity/permeability relationship is the same one that has been determined for highly compacted rock salt backfill and undisturbed rock salt. Not only the hydraulic properties but also the mechanical properties of the SVV after reaction with brine are comparable with undisturbed rock salt.

Rock mechanical parameters like Poisson ration, Young's modulus, fracture strength as well as strain and failure were determined and compared with rock salt data. Octahedral shear strength for instance lie at the octahedral shear strength curve of rock salt indicating excellent rock mechanical properties of the new seal material. We conclude that SVV is a material with similar properties as rock salt. SVV can be used for high quality seals in all kinds of salt formations including the extremely soluble carnallitite and tachydrite bearing potash rock formations. SVV is long term stable and fully predictable. ■

2A.15 Colloid/Radionuclide Migration Studies in Fractured Rocks; Lessons learnt from the Grimsel and Äspö System

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Investigations within the CRR (Colloid Radionuclide Retention) project in the migration shear zone (MI) of the Grimsel Test Site (GTS, Switzerland) and within a series of actinide migration experiments at the Äspö Hard Rock Laboratory (HRL, Sweden) are summarized.

The CRR experiments have shown that smectite colloids once eroded from bentonite can exhibit a considerable mobility in a granitic shear zone at high groundwater velocities. Under the given geochemical conditions, they act as carriers notably for the tri- and tetravalent actinide migration. Np(V) and U(VI) are not reduced and the interaction with colloids is negligible. Spectroscopic studies indicate actinide binding to aquatic colloids via inner-sphere surface complexation. Dissociation of such complexes may be considered slowly reversible and reversibility is indeed observed in batch sorption studies.

Furthermore, a series of experiments investigated the migration of Am, Np, Pu, U and Tc in a single granite fracture in the CHEMLAB 2 probe under in situ conditions of the drill hole KJ0044F01 at Äspö HRL. Np retention on Fe(II) containing minerals of the granite and altered material by reduction to Np(IV) could be observed by XPS. Pu sorption took place on a multitude of minerals and specific sorption processes could not be identified up to now. Both, batch-type and migration studies reveal significantly higher Pu sorption coefficients compared to Np and U. U migration behavior and the mobilization of natural uranium is interpreted as the influence of oxygen due to the artificial oxidation of the core fracture.

Combining the outcome of laboratory investigations with those of the field study reveals the necessity of considering kinetics of the radionuclide interactions at the mineral-groundwater and the colloid-groundwater interface. The quantitative assessment of radionuclide migration requires the deeper understanding of radionuclide reaction kinetics and the underlying mechanisms of colloid generation and mobility under conditions comparable to the situation in a future nuclear waste repository. This is the focus of the currently running projects CFM (Colloid Formation and Migration) at the GTS and the Colloid project at the Äspö HRL, whose latest results will also be presented. ■

2A.16 Ruprechtov Site (CZ): Geological Evolution, Uranium Forms, Role of Organic Matter and Suitability as a Natural Analogue for RN Transport and Retention in Lignitic Clay

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Abstract
Ruprechtov natural analogue site, notably for its unique association of U-rich and organic layers within clay sediments, being underlain by granite, is studied in order to understand the processes of U-enrichment as a natural analogue for radionuclide transport and retention in lignitic clay formations, which often represent constituent parts of the overburden of potential host rocks for radioactive waste disposal. The site investigations focus on the detailed characterization of immobile uranium phases methods. Based on all available results a model for the geological evolution of the site including the different processes of uranium enrichment is developed. Special focus is laid on investigation of organic rich layers (lignite) within clayey sediments with carbon content up to 40 %. The results show that organic matter plays a more important role in establishing a reducing environment than as U sorbent.

1 Introduction

To assess the safety of a repository for radioactive waste, one needs to look at the important aspect of the barrier function of surrounding geological environment, i.e. the host rock and overlying and the overburden. Possible future developments are simulated with the help of long-term safety analyses, which in turn are based on assumptions that have to be verified. So-called “Natural Analogues” are a particularly suitable instrument for such verification purposes as they represent naturally occurring systems in which physical and/or chemical processes that can be expected in repository systems (or parts thereof) are developing (or have developed) over geological time periods. In particular, the study of geochemical long-term processes in rock types with increased natural radionuclide content represents an important supplement to laboratory experiments, which for reasons of setup and realisable time periods are subject to many limitations. Natural analogues can be also a very useful tool for methodology development, scientific team building and communication with the public.

Starting from the general geological situation in the environment of possible repository sites in northern Germany, with their characteristic sequence of clayey-silty and sandy sediments as well as lignite sands, approx. 40 uranium ore deposits in Germany were studied initially by way of an evaluation of the relevant literature. Due to their high potential suitability (genesis, lithology, age), two localities were finally chosen for a more detailed investigation programme: a) Heselbach near Schwandorf/ Upper Palatinate (north-eastern Bavaria) and b) Ruprechtov in the north-east of the Czech Republic (between Jáchymov and Karlovy Vary). The latter is subject of the present paper. The investigations at both localities have been aimed to broaden the understanding of complex geochemical behaviour of radionuclides during transport and sorption in argillaceous sandy sediment under the influence of lignite horizons, affecting the geochemical milieu.

2 Geological Evolution and Uranium Accumulation

The overall geological situation of the Ruprechtov NA site is characterised by its position within the Sokolov basin, which forms part of the Ohre rift, a rift structure running in SW-NE direction and parallel to the Erzgebirge / Krušné hory mountains. This rift structure is filled with tertiary sediments (in the region of Ruprechtov mainly of volcanic origin) (Figure 1).

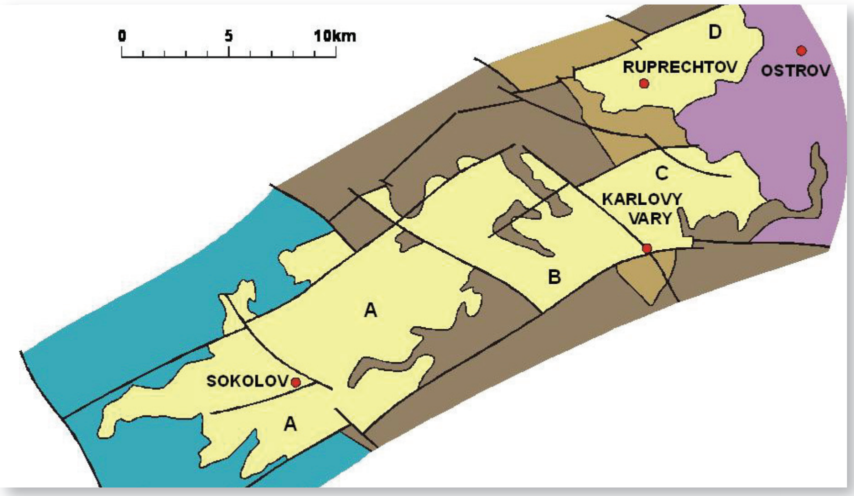


Fig. 1: Geological sketch map of Sokolov basin (part of the Ohre rift). Tertiary sediments are plotted in yellow, granitic rocks in brown, metamorphic rocks in cyan and the tertiary volcanic rocks of the Doupovské hory stratovolcano in purple colour. Characters A-D mark several sub-basins

In the investigation area the following stratification is represented (from top to bottom):

- Quaternary cover (soil, loam – up to 2 m deep)
- Pyroclastic sediments (tuff, mostly argillised into bentonitic clay - maximum thickness up to 50 m)
- Clay/Lignite-sand horizon (with the most important uranium accumulation – thickness of stratum approx. 1-2 m)
- Kaolin (weathering product of the underlying granite – thickness up to approx. 40 m)
- Granite (partially also outcropping).

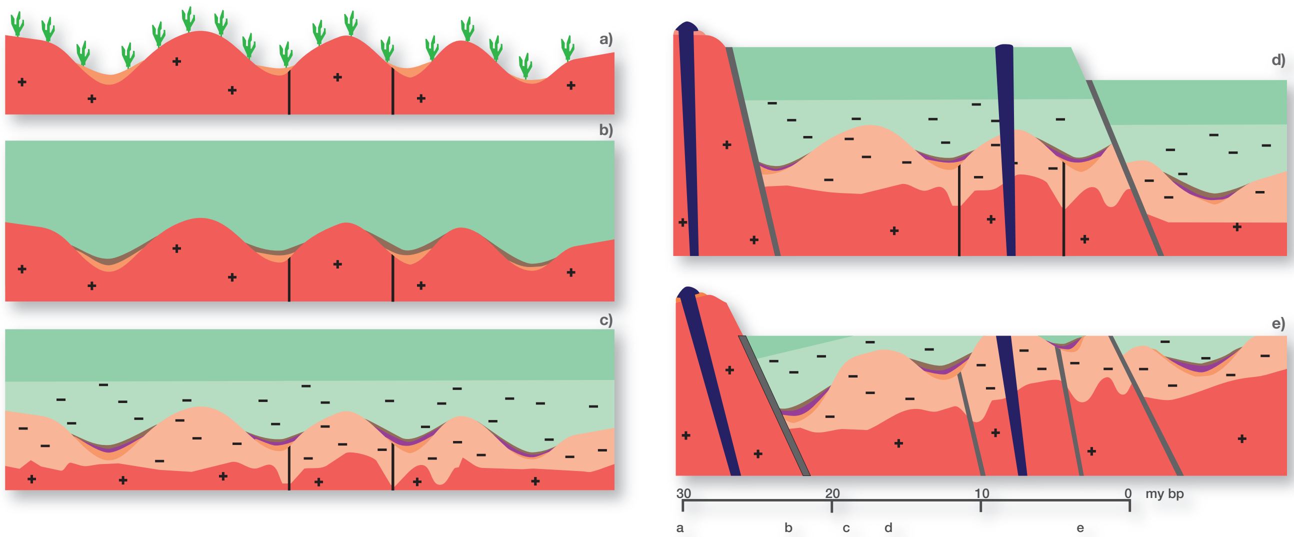
A first, in an early stage developed, relatively simple model of uranium mobilisation [1] was based on information found in the literature and on the results of a few drillings that had been restricted to the area of the uranium accumulation known at this time. This model involved the dissolution of uranium under oxidising conditions from the surrounding granite, its transport via groundwater in sediments with increased hydraulic conductivity, and its precipitation/sorption on lignite intercalations under reducing conditions in the form of a roll-front deposit.

More recent studies supported by drilling programmes spread over wider areas and involved more detailed laboratory tests have now shown that the bedding conditions are governed by strong morphology at the Tertiary base and that the uranium accumulations found in the clay/lignite-sand horizon have not been formed by one single process but by several different causes:

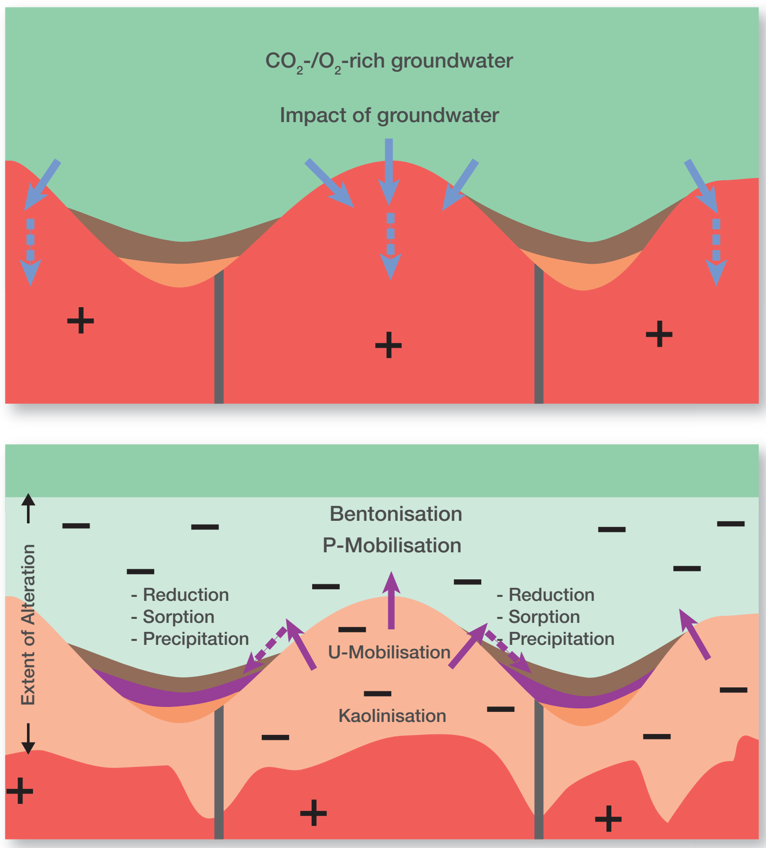
- Synsedimentary influx of detritic uranium minerals,
- Spatially limited mobilisation of uranium from the surrounding granite during the kaolinisation of the latter, diffusion and retardation (reduction, sorption, precipitation) within the clay/lignite-sand horizon,
- Influx from the underlying granite in zones of low kaolin thickness and/or via fault zones and retardation (reduction, sorption, precipitation) in the zone of the clay/lignite-sand horizon.

The assumptions based on current knowledge regarding the geological evolution of the Ruprechtov locality (as far as an U-accumulation is concerned) now comprise five main phases a – e (Figure 2).

Here, it is above all phase “c” (Figure 3) that makes a major contribution to uranium accumulation: Following the sedimentation of the pyroclastics, weathering of the underlying granite mainly took place as a result of the reaction of feldspars with CO_2-rich



▲ Fig. 2: Schematical representation of relevant phases of the geological evolution of the Sokolov basin: a) Pre-Oligocene (> 30 My): influx of detritic material (orange) through physical weathering of the granite present at the surface (red); b) Lower Oligocene - Miocene (30-16 My): deposition of organic material (brown) in trough areas; after that, main phase of volcanic activity with wide-area sedimentation of pyroclastics / tuff (green); c) Lower Oligocene - Miocene (30-16 My): alteration of granite (kaolinisation - light red, dashed) and tuff (bentonisation - light green, dashed); d) Miocene (16-15 My): rift formation, combined with fault zones (grey) and basalt intrusions (dark blue) e); Pliocene - Quaternary (< 5 My): further evolution of the Ohre rift and partial erosion of the pyroclastic sediments



◄ Fig. 3: Schematical representation of the alteration processes in granite (kaolinisation) and tuff (bentonisation) through the impact of CO_2- and O_2-rich groundwaters in phase “c” of the geological evolution. The previous wide area sedimentation of pyroclastic sediments has led to the condition that the thickness of kaolin is greatest where the granite previously showed morphological elevations. Uranium (magenta) is accumulated in those former valleys in which lignite or lignitic clay also have accumulated groundwaters and the formation of kaolin. CO_2-rich groundwaters furthermore initiated the mobilisation of the uranium from accessory minerals by formation of soluble UO_2-carbonate complexes. Within the kaolin, uranium was mainly transported through diffusion. At the boundary layer between the kaolin and the overlying pyroclastic sediments, advective transport was also possible over short distances in a local horizon with increased hydraulic conductivity. The main immobilisation processes that could be considered were uranium reduction in/close to lignite-rich sediments, combined with formation of secondary uranium(IV) minerals such as uraninite and ningyoite.

The strongly reducing conditions in lignite-rich horizons could be verified by means of pH-Eh measurements in the corresponding groundwaters. The Eh-values of the groundwaters in clay sediments are clearly below those of granitic waters. According to that uranium can be supposed to exist in immobile U(IV) form that could be also declared by low U groundwater concentration (see Figure 4).

The individual retention processes are currently quantified within the framework of FUNMIG EC Project (see http://www.funmig.com/). Here, the focus was paid on the in-depth study of the immobile uranium phases by means of different surface-spectroscopic methods, adsorption and desorption experiments, as well as geochemical studies.

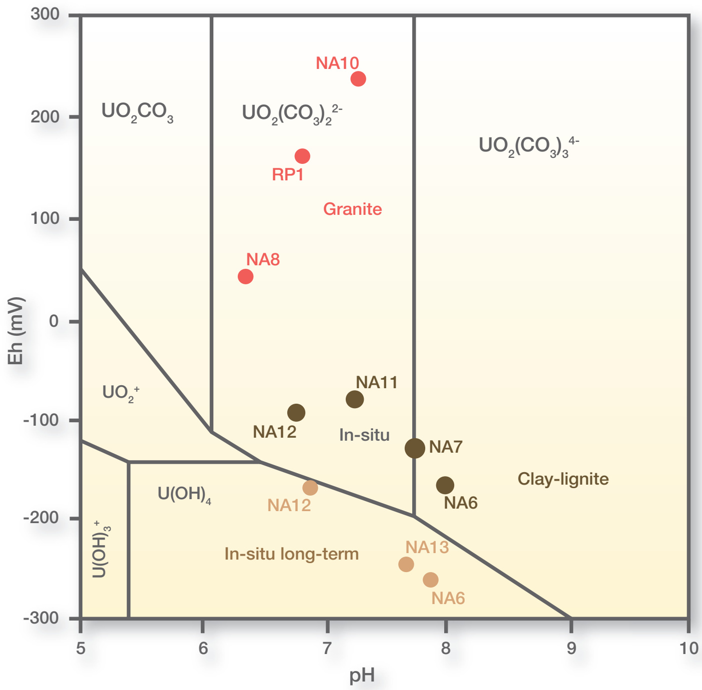


Fig. 4: pH-Eh-phase diagram of aqueous uranium species. The coloured dots represent groundwater measurements and illustrate the differences in the geochemical environment between granite (red) and the lignite-clay/sand horizon (dark brown) as well as problems associated with qualified pH-Eh-measurements: regarding the drillings NA6, NA12 and NA13, only in-situ long-term measurements (light brown) at a depth of approx. 40 m have shown that uranium(IV) can be expected to be the dominant species in the aqueous phase

The very thin clay/lignite-sand horizon therefore can act as a highly effective transport barrier for uranium. There are no indications whatsoever to a renewed release of uranium from the accumulation horizons. Details concerning the studies and the results are summarised in the recently published final report “Radionuclide Transport and Retention in Natural Rock Formations - Ruprechtov Site” [2].

3 Uranium Forms

Organic rich layers (up to 40 wt% of TOC) were localised within the sedimentary system, being heterogeneously distributed. Uranium enrichment found showed also heterogeneous distribution, however it was usually not directly incorporated within lignite-like matter, but in vicinity.

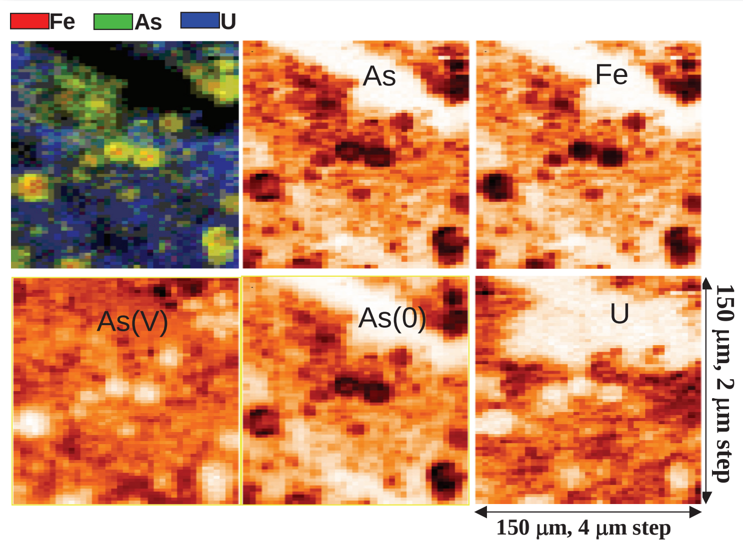


Fig. 5: µ-XRF distribution maps for a 150*150 µm² area of a thin section of a sample from NA5. The distribution of the total As, Fe and U measured with Eexcite = 18 keV and its corresponding red-blue-green (Fe, As, and U, respectively) image, as well as the arsenic chemical state distributions for As(V) and As(0) are shown [3]

Therefore, uranium forms in the sediment system and its relation to organic matter composition and degradation were extensively studied. In a first step, modern material surface analyses methods, i.e. micro X-ray fluorescence (µ-XRF) and micro x-ray absorption fine structure (µ-XAFS) on several core sections were employed [3]. Those were compared with the classical sequential extraction method, used up for trace element fractionation and form determination. The results showed uranium to be present as a U(IV) phosphate in association with As (see Figure 5 and Figure 6).

Simultaneously the sequential extraction (SE) scheme was applied to Ruprechtov samples of different origin (six boreholes, different depths, different rock types - kaolinised granite or tertiary clay). Samples were leached using different extractants in order to quantify various forms of U present. The SE scheme consists of alltogether 5 successive steps [4].

In tertiary clay samples (all samples except NA10 and NA15B) U in reduced form/bound onto organic matter was proportional to the total U content (U_{tot}) and on total organic carbon content (TOC). Those lead to conclusion U in reduced form was dependent on organic matter presence. Moreover, increased U total content only increased with more reduced U(IV) present. In granitic samples (NA10, NA15B) U was significantly bound either into residual phase or bound onto carbonate (Figure 7).

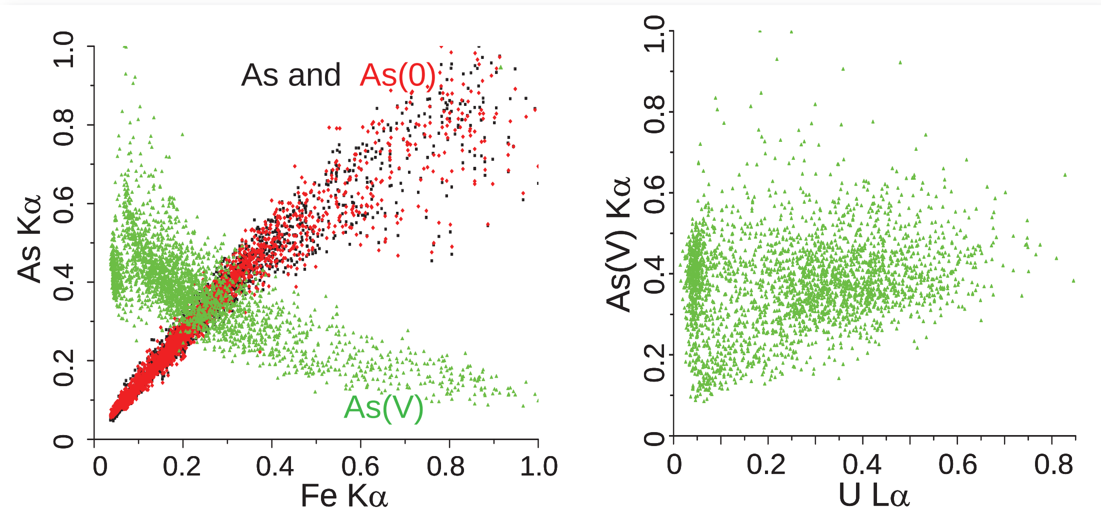
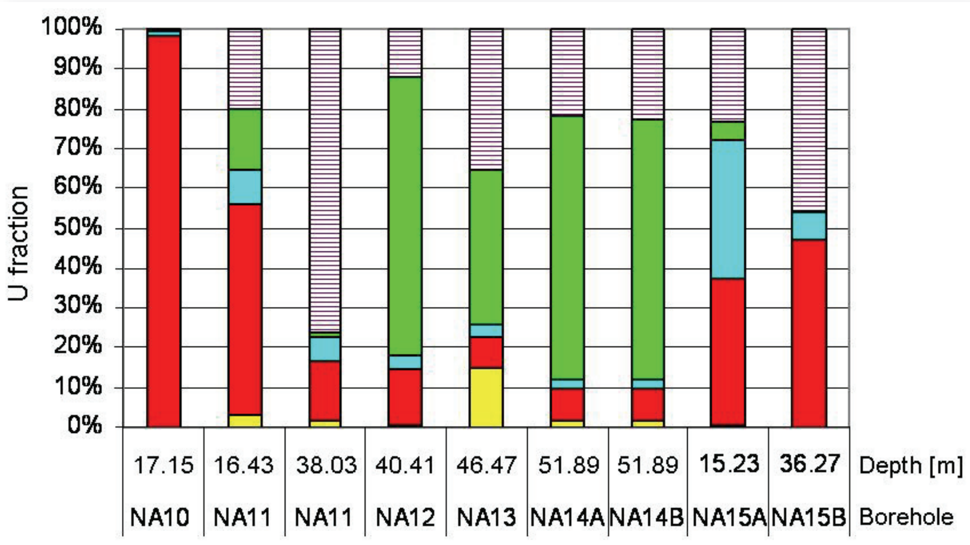


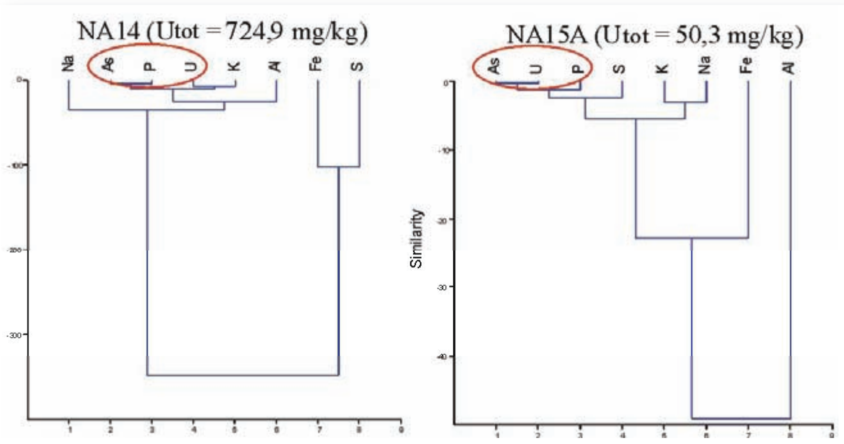
Fig. 6: Correlation plots between Fe and As(0) [◆], As(V) [▲], and total As [■] (left) and between U and As(V) (right) [3]



▲ Fig. 7: Uranium form fraction representation (in %) within Ruprechtov sediment samples as result of SE-experiments (step 1 [yellow] = exchangeable; step 2 [red] = carbonates; step 3 [cyan] = Fe/Mn oxides; step 4 [green] = U(IV) / reduced form; step 5 [dotted] = residuum)

Following these observations extended element analyses (Na, K, S, Fe, As, P) of SE-leachates were performed for two samples (NA14 and NA15A) in order to define phases dissolving in each extraction step. Cluster analyses could then be used to process the data in order to identify possible interrelations between elements and define phase dissolution/leaching within the sediment. [5] was used for the statistical procedure of U, P, As, Na, K, S, Fe in each SE-leachate sample (see Figure 8).

The results revealed that P, As and U were assigned into one group. This was in direct agreement with the µ-XRF and µ-XAFS measurement results. Fe and S were not interrelated with U, i.e. after all, there was no direct evidence of U dependence on Fe and S, although high content of those two elements were identified within the sediment. The major elements K and Na were mainly fixed in residual mineral phase (SE step 5, silicate origin).

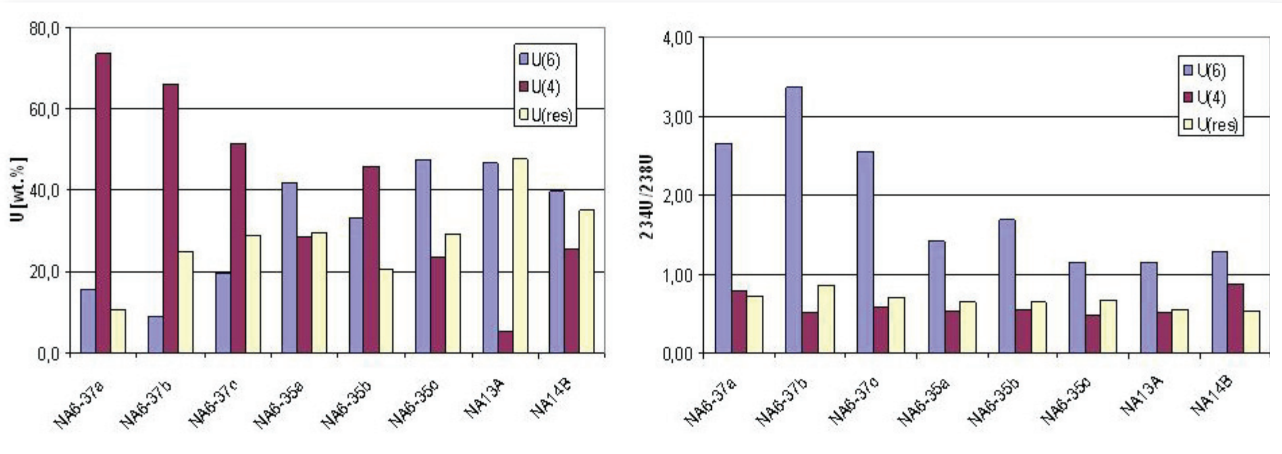


◄ Fig. 8: Cluster analyses (PAST) for trace element content in each sequential extraction leachate for two tertiary clay samples from Ruprechtov site. NA14 - tertiary clay, rich in organic matter (left), high U content, NA15A - tertiary clay, low organic matter and U content (right)

As K was identified to be leached in SE step 2, originally signed as carbonate bound form, although there is limited occurrence of carbonate minerals, we can assume that alumosilicates and/or carbonate complexes are more responsible for U distribution than carbonate minerals themselves.

In the same time, results of both microstructural analyses and sequential extraction were compared with mineralogical and microscopic study [6]. Here, U was identified in the forms of either primary U-bearing detritic minerals (xenotime, zircon, monazite) or secondary minerals (ningyoite, uraninite). In addition, very rare grains of non-identified uranium phases were observed. Ningyoite was also detected by micro structural analyses as small crystals, with dimensions around 5 µm, most commonly in association with iron sulphides (µ-XRF, µ-XANES, µ-EXAFS measurement [3]).

To identify uranium oxidation state fraction detailed complementary method for analysing uranium (IV) and uranium (VI) was applied for samples with different uranium content from different boreholes [7]. Both forms U(IV) and U(VI) existed in the tertiary clay horizon. The major part of uranium was U(IV) and its fraction varied from 55 to 90 % in the investigated samples. The generally low $^{234}U/^{238}U$-activity ratios (< 1) in the uranium(IV) fraction gave the evidence that this phase was rather stable and immobile over geological time scales (Figure 9). This evidence was confirmed by content of uranium in groundwater: even though U content in sediments was relatively high (up to 600 ppm), U concentration measured in groundwater was very low (app. 10 µg/l in granitic water, 0.5 µg/l in waters from Tertiary).



▲ Fig. 9: Fraction of uranium in U(IV), U(VI) and U(res) phase (left) and corresponding $^{234}U/^{238}U$ isotope ratios (right). Insoluble U(res) is assumed to be U(IV)

4 Sedimentary Organic Carbon Cycle

Natural isotopic data were used for identification of the hydrogeological flow regime and the processes on the site, including carbon and sulphur cycle within the Ruprechtov system (e.g. [2]). Moreover, sedimentary organic matter composition and humic substances content were studied.

Despite the high sedimentary organic carbon (SOC) content in the clay/lignite-sand horizon (up to 40 wt%), the concentration of dissolved organic carbon (DOC) in Ruprechtov groundwater was generally low (app. 4 mg/l). Comparing the system with conditions at Gorleben, striking difference was evident: The highest DOC concentrations at Gorleben reached up to 200 ppm [8]. If humic substances were present, those were identified as fulvic acids. Studying composition of SOC, humic acids formed only 0.15 wt% of SOC [9].

Preliminary results indicated that microbial SOC degradation probably takes place at Ruprechtov site [2]. The conceptual model of sedimentary organic matter (SOC) decomposition was described elsewhere. Microbial processes lead to oxidation of SOC and reduction of oxidising agents, like SO_4^{2-} or NO_3^-. SOC is then partly oxidised to dissolved inorganic carbon (DIC) but part is also released as DOC, i.e. humic, fulvic and hydrophilic acids. However, those degradation products could then be sorbed on clayish sedimentary material and got fixed as reported in [10]. Such effects could result in a DOC content decrease in Ruprechtov groundwaters and the lack of humic substances [8].

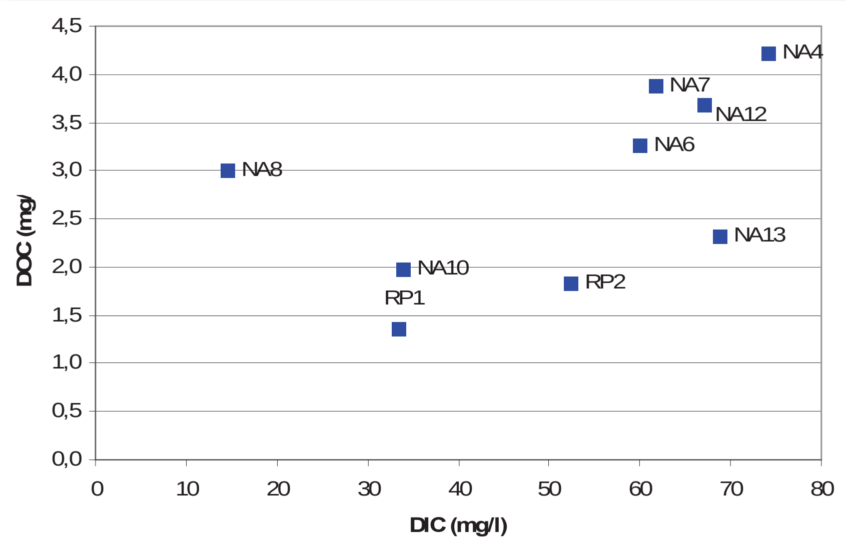


Fig. 10: DIC concentration vers. DOC concentration in Ruprechtov groundwater wells

In the next stage of degradation process more DIC is released followed by an increasing DOC oxidation. The formation of carbonic acid releases one proton and dissolves as HCO_3. In contact with carbonate containing sediments (sedimentary inorganic carbon - SIC) this process results in its dissolution and additional release of DIC of sedimentary origin.

The first indication of the microbially mediated process at Ruprechtov site is presented in Figure 10. Dissolved inorganic carbon concentration increase, particularly in clay/lignite horizon, is generally followed by increased DOC release (extraordinary NA8 borehole is the one with outcropping granite).

Furthermore, mineralisation (oxidation) of SOC can be accompanied by reduction of oxidising agents (SO_4^{2-} and NO_3^-), i.e. decrease of the species concentration with increasing DOC. The general trend is traceable within species concentration analyses in the Ruprechtov groundwater (see Figure 11). On the other hand, general reverse increasing trend can be spotted for phosphates (PO_4^{3-}). Those are produced by microbial mediated SOC metabolisation, causing reduction of sulphates and nitrates and release of originally SOC-bound phosphates into the solution.

Moreover, considering sulphur isotopic signatures, a complete oxidation of organic matter and the sulphate reduction can be written as $SO_4^{2-} + 2\,CH_2O = H_2S + HCO_3^-$. In presence of iron the reaction product will be fixed as mono-sulphide and afterwards

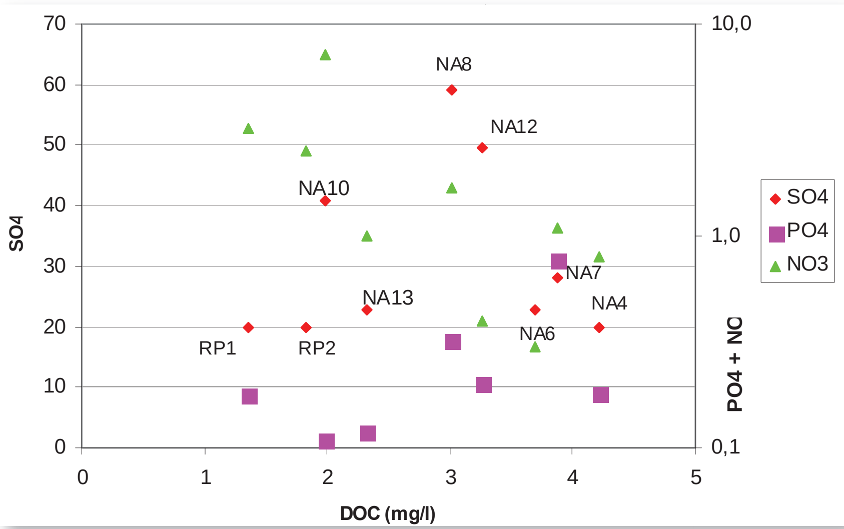


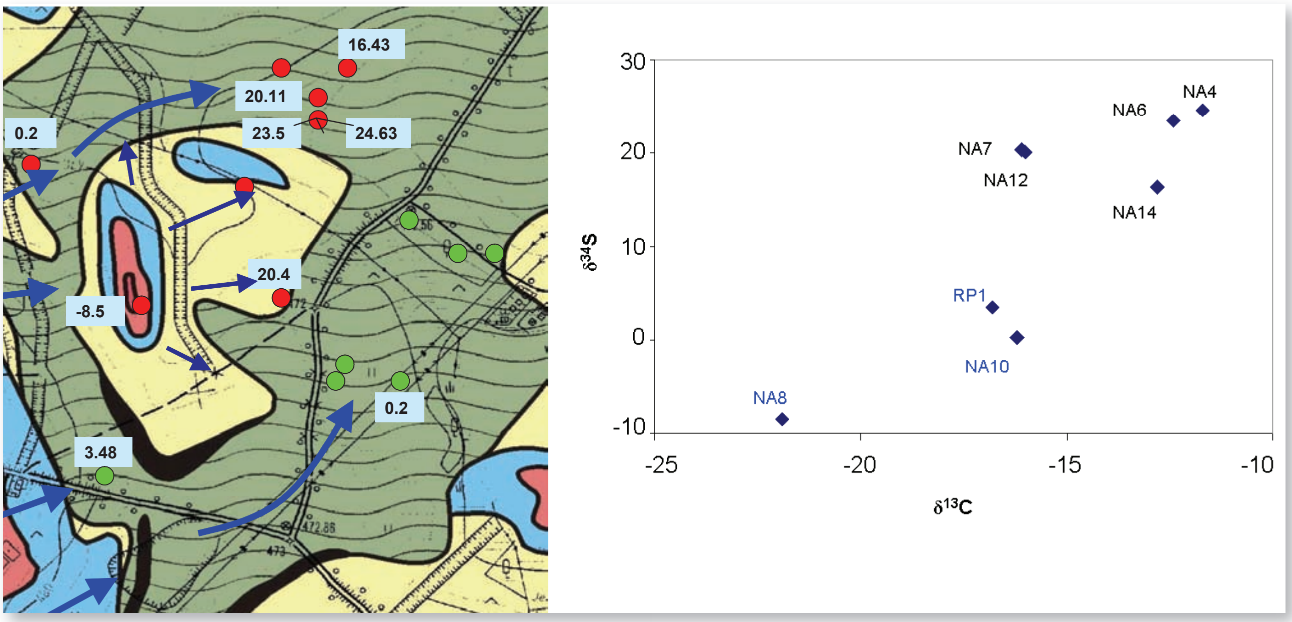
Fig. 11: SO_4^{2-}, NO_3^- and PO_4^{3-} concentration vers. DOC groundwater (RP1 and RP2 boreholes were not drilled as part of the research project and are not filtered in a distinct horizon as the other ones)

be transformed into pyrite. The microbial sulphate reduction is accompanied by isotope fractionation. The lighter isotope ^{32}S is preferentially metabolised by microbes and therefore residual sulphate molecules in groundwater become enriched in the isotope ^{34}S, whereas $\delta^{34}S$ values in sulphides decrease. The groundwaters from the clay/lignite-sand layer show increased $\delta^{34}S$-values in the range of 16.43 to 24.63 ‰ that is clear indication of microbial sulphate reduction. The results of $\delta^{34}S$ analyses in water samples are shown in Figure 12.

Considering the results of uranium form identification and organic carbon behaviour on the site, there are indications for proposed uranium immobilisation at Ruprechtov site. Organic matter degradation was a microbial mediated process contributing to U(VI) reduction and U(IV) immobilisation into secondary phases.

SOC within the sedimentary layers was microbially oxidised, releasing DOC and providing protons to dissolve SIC. Oxidation agents (SO_4^{2-} and NO_3^-) were reduced and reduction of sulphate caused FeS_2 formation. Dissolved As got sorbed onto FeS_2, forming FeAsS precipitate. Uranium, having source in the underlying granite, was reduced on FeAsS surface to U(IV) and reacted with phosphates PO_4^{3-}, produced by microbial SOC oxidation. Uranium phosphates (ningyoite) were thus formed. Uranium is predominantly in U(IV), stable and immobile.

Low content of DOC in the groundwater identified could be caused either by high sorptive affinity of clay for organic degradation products or low SOC availability for degradation. Organic degradation products, considered in PA as potentially strong complexing agent, causing colloidal facilitated transport of radionuclides, then could be effectively fixed in the sedimentary system and reduce probability of radionuclide release into the environment. The research on sorptive properties of Ruprechtov tertiary clay sediments for natural humic acids and their complexation are in progress within the frame of FUNMIG project.



▲ Fig. 12: Distribution of $\delta^{34}S$ in different boreholes at Ruprechtov site (left) and $\delta^{34}S$ vs $\delta^{13}C$ (right)

5 Conclusions and Outlook

Ruprechtov site is a good example to demonstrate that tertiary argillaceous sediments can exert a strong barrier function for uranium migration, when specific prerequisites are fulfilled. Major uranium transport occurred over distances of about tens to max. some 100 m only during Tertiary. Uranium was transported as U(VI) and was reduced in a lignite rich clay horizon with occurrence of pyrite and arsenopyrite minerals. It was immobilised by forming uraninite, and phosphate bearing minerals like ningyoite. There is no evidence of uranium mobilisation during the last million years. But there is indication that still during the last several 100'000 years further uranium enrichment occurred in this lignite rich clay horizon. This is probably due to transport from underlying granite through zones of low kaolin thickness and/or fault zones. The uranium concentrations in groundwater of the clay/lignite-sand horizon are low in the range of 0.5 µg/l although this horizon is only 25 to 65 m below the surface.

There are still several open questions to be answered at Ruprechtov site. In the frame of the integrated project FUNMIG, which is carried out from 2005 until 2008, some of these questions are addressed (e.g. role of organic matter, immobile uranium phases). Further interest is also given to an easier accessible uranium phase, which makes up about 10-20 % of uranium in the enriched horizon and probably represents a uranium(VI) phase, which is sorbed to organic material. This will be investigated by optimization of the U(IV)/U(VI)-separation method and application to further samples. All analyses of immobile uranium phases will be accompanied by further specific sequential extraction methods and specific uranium desorption (exchange) measurements with the non-naturally occurring ^{236}U tracer. Application of surface complexation models are planned to quantify the sorption process.

It is still not proved that the current model assumptions are complete and describe all relevant processes. The recently started kaolin open cast mining now offers the unique chance, to review the state of knowledge concerning genesis of uranium accumulation and model assumptions by a large-scale and 3-dimensional exposure of geological conditions of the site and therewith allows the proof of assumptions and amendment of model assumptions (if necessary). Furthermore, the scientific attendance of kaolin mining with its alterations of hydrogeological and hydrogeochemical conditions enables the investigation of the impact of such sudden changes (especially redox potential and pH-value) on the mobility of uranium and – in addition – may give hints on the demand of investigations needed to characterise a potential site for geological disposal of radioactive waste. Thus, it could help to improve the public acceptance of long-term safety assessments also. ■

Acknowledgements

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Session 2B

Performance Assessment and Modelling

2B.01 The Safe Disposal of Radioactive Waste in Rock Salt

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Abstract

In Germany rock salt is the preferred host rock for the disposal of hazardous industrial waste and heat generating radioactive waste. This preference is based upon the excellent understanding of the very positive features of rock salt, the century-old mining experience, the geologic situation and, above all, the scientific consensus confirmed by the scientific results of nearly 40 years of R&D.

In the 1960s the responsible authorities in Germany decided that high level vitrified waste and spent fuel should be disposed of in rock salt. In the site selection procedure the Gorleben salt dome was chosen and a reference disposal concept was developed. The exploration at this site started in 1979.

Underground disposal of chemotoxic waste started in 1972 as an initiative of the chemical industry in the facility "Untertagedeponie Herfa-Neurode". Meanwhile several millions of tons of chemotoxic waste have been disposed of at the operating facilities Herfa-Neurode, Heilbronn, Sondershausen, and Zielitz.

To date R&D has accompanied these activities. R&D on safe disposal of radioactive waste has been performed for about 40 years. In the last 20 years the R&D projects focused on issues like repository technology, safety and development of methodologies and instruments for performance assessment. The state of knowledge is advanced, the tool-box is well equipped and the remaining activities, which still have to be done, can be identified.

In principle argillaceous media have the potential to host a repository in Germany. However, a decision concerning this type of host rock and potential sites is pending. Despite these activities, there is still consensus in the scientific community that rock salt is an excellent host rock for the disposal of hazardous waste, especially heat producing radioactive waste.

In Germany crystalline rock ranks lowest as host rock mainly because of geological reasons. Therefore, R&D activities with regard to disposal in crystalline media are restricted to investigations of the engineered barrier system and on methods or tools which can be applied in rock salt and argillaceous formations as well.

This paper is focusing on R&D for the disposal of hazardous waste in rock salt, a disposal option which is practicable, sound and safe - especially with respect to long-term safety, a feature indispensable for radioactive waste disposal. Selected R&D achievements are addressed, showing the state of knowledge in underground waste disposal in salt.

1 Introduction

"Some of the most important scientific-technical and ecological problems in the present time and in the future, which could only be managed by interdisciplinary activities, are the production and the spreading of anthropogenic wastes on the Earth's surface and their long-term safe disposal" [1]. The authors Herrmann and Röthemeyer tried to draw attention to the relevance of the problems of hazardous waste and they claimed the utmost urgency to solve these problems by interdisciplinary research and development and a consequent deployment of R&D-results into disposal projects.

Safe disposal of hazardous wastes is attainable and is acceptable on the basis of a sound scientific knowledge about the disposal systems and their long-term behaviour. In Germany deep geological disposal of all types of radioactive waste and of selected types of hazardous non-radioactive waste is obligatory. This method of waste disposal is acknowledged as being the best suited disposal option which guarantees safety for future generations. Waste which cannot be reused or should not be disposed of at or near the Earth's surface is encapsulated into a system of natural and technical barriers in deep geologic formations. So a real and long-term stable sink for inventory-compounds is established which prevents radionuclides, heavy metals, salts and toxic organic compounds from spreading in the human environment, leading to adverse health- and environmental effects. The multiple barrier system of the disposal facility prevents the release of hazardous inventory with a high probability. Furthermore, the transport processes in the geosphere are very slow, compared with processes at the Earth's surface. Mobilization und transport of important inventory components are delayed. Moreover, the predicted consequences in case of a release of hazardous waste from a deep geological repository are presumably much lower than the consequences expected in the case of surface disposal.

Based upon the German geologic situation, the experience from mining activities in salt deposits and supported by scientific consensus, rock salt became the preferred host rock to dispose of hazardous waste.

In principle the underground disposal of both radioactive waste and chemotoxic waste is similar. For radioactive and for chemical waste as well, practically no alternative waste management options exist from an ecological or economical point of view. In both cases underground facilities act as a safe long-term sink for hazardous waste, preventing the waste inventory to endanger the biosphere and mankind over very long time periods.

In the early 1970s German industry took the lead by getting the first underground disposal facility for hazardous waste operational. The Herfa-Neurode-facility is still operated by K+S, a former daughter of BASF, but meanwhile one of the most important salt and potash producers world-wide. It can only be hoped that Germany becomes as successful in the disposal of radioactive waste.

Some experts and institutions see the need to answer several important questions in the field of radioactive waste disposal requiring efforts in scientific work and in disposal projects. But these open questions do not jeopardize the feasibility of underground disposal.

This paper deals with some selected R&D-aspects which are vital for the feasibility of deep underground disposal.

2 Current Status

2.1 Waste Disposal

The current status of waste disposal in Germany is determined by current disposal activities of hazardous waste, and a pending decision for further steps for radioactive waste disposal. In Germany chemotoxic waste and low and intermediate radioactive waste was disposed of in mined underground facilities. Presently, in German underground disposal facilities for hazardous waste (Herfa-Neurode, Zielitz, Heilbronn, Sondershausen) about 250.000 tons per year are disposed of.

Underground disposal of chemotoxic waste is managed by the respective waste producers in industry, mainly the chemical industry, and few specialized disposal facilities, experienced in mining and waste management issues. In 1972 the first German underground disposal facility at Herfa-Neurode became operational. Meanwhile underground long-term disposal in mined openings is practised in three additional facilities. In the 1990s a quite similar option to dispose of low contaminated solid waste was established by using mineral waste as a filling material in the mining industry. Up to 30 mines in different host rocks – most often salt – used waste classified as hazardous by legislation for their closure and backfilling measures.

2.2 R&D

The activities to dispose of chemotoxic waste have not always been supported by R&D. It was not before the late 1980s that the German Federal Government decided to start funding of scientific research for the underground disposal of waste. These research activities were to help improve the safety assessment of disposal facilities, set off by an advanced legal framework

(TA Abfall) developed at that time. One of the first project-funded activities was the investigation of the behaviour of solution-mined waste caverns, an option believed to have some advantages with respect to safety and maintenance compared with mines. Although the results were quite positive, the consequences of German Reunification for the mining industry prevented this option to be implemented. From the 1990s until now, it has become evident that a predicted “Müllnotstand” will not occur, due to the consequences of waste avoidance and recycling.

In 1991 the Federal Ministry of Science and Technology (BMFT) reorganized the scientific and technical fields of both disposal of radioactive and hazardous waste in one integrated research (funding) concept. Corresponding research projects where financed through typical means of project funding. An expert group was established to provide guidance and consultation for this integrated research concept, which was published in 1998 [2].

The importance of R&D in these fields and the economic and environmental relevance should be illustrated with a few data. In underground facilities in Germany more than 2 million tons of hazardous waste are disposed off. Every year about one to three hundred thousands tons are added.

In mines with backfilling operations a continuously increasing volume of solid waste is used as filling and stabilizing material. Waste volume increased from about 350.000 tons in 1994 to about 1.200.000 tons in the year 2001 as reported by the Federal Ministry of Environmental Protection (BMU) [3].

Radioactive waste was disposed of in former German salt mines beginning in the 1970s, too. In the ERA Morsleben, this is situated in the Federal State of Saxony-Anhalt, LLW/ILW disposal started in 1976. It took place under the then legislation of the German Democratic Republic, and was continued after the German unification until 1998. Since then, the Plan Approval Procedure for closure and decommissioning of the mine has been in progress. The Asse salt mine is situated in the Federal State of Lower Saxony. Starting in 1967, it was used as a research facility. From 1972 to 1978 research into disposal technologies was performed and emplacement of low and intermediate level waste from different sources and with different techniques occurred. After the stop of waste emplacement the Asse served as an underground laboratory. A lot of experiments with regard to HLW disposal were performed; the last one was finished in 2003. Presently, the Asse mine is in the closure phase. Since the Asse mine is legally not considered as a repository, the licensing procedure for closure and decommissioning is governed by Mining Law [4].

The KONRAD iron ore mine, situated in the Federal State of Lower Saxony, too, was foreseen to dispose of low and intermediate level waste (waste with negligible heat generation) and in 1982 a Plan Approval Procedure was started.

Since 2007 the KONRAD mine is licensed. The stage is set for a repository for low-level radioactive waste to be operational in 2013 [5].

The amount of radioactive waste disposed of to date or to be disposed of in the future [6] is by far outnumbered by the amount of hazardous waste managed in the field of chemotoxical waste.

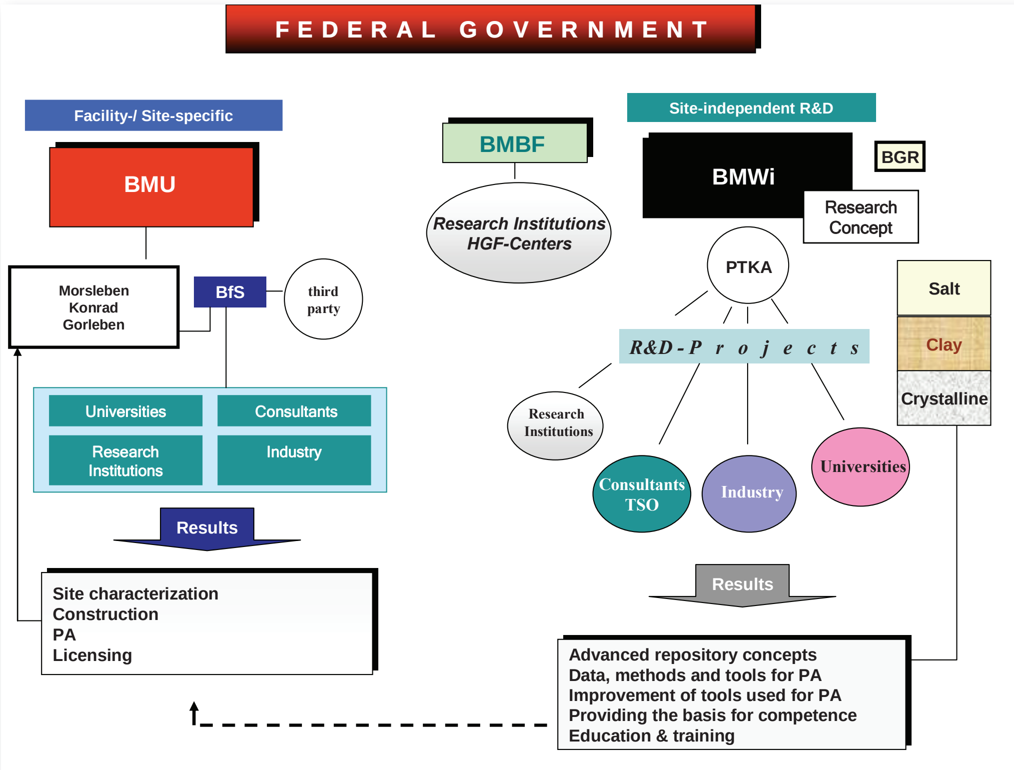
2.3 Responsibilities

The German Federal Government has to ensure the safe disposal of radioactive waste by providing repositories. The organization and responsibilities of involved authorities and institutions is shown in Figure 1.

In Germany basic and applied basic R&D is in the responsibility of the Federal Ministry of Education and Science (BMBF) and the Federal Ministry of Economics and Technology (BMWi).

The Federal Ministry for the Environment, Nature Conservation, and Nuclear Safety (BMU) is the regulator for disposal projects and is responsible for related site-specific R&D.

Site specific R&D is performed on behalf of BMU by the Federal Office for Radiation Protection (BfS), which initiates and co-ordinates R&D activities and uses the expertise of third-party organizations for site-specific tasks. BfS is responsible for



▲ Fig. 1: Responsibilities and organization in the field of radioactive waste disposal

construction and operation of plants and installations for disposing of radioactive waste. Special site-specific R&D tasks are carried out mainly by research centres, scientific consultants, universities, and industrial companies. The results are directly used by BfS for site characterization, performance assessment, and license application.

Non site specific R&D is funded by BMBF and BMWi. Both the environmental research program and the energy research program address the field of waste disposal.

3 R&D Activities

Both in the past century and to date a lot of important and landmark R&D projects have been performed. The results and knowledge gained contributed tremendously to the state of the art available today, especially with regard to all aspects dealing with disposal of hazardous waste in rock salt. The R&D activities are based upon R&D concepts being the guiding principle. After a brief description of the respective R&D concept and the basics for funding, some selected projects are shortly presented

3.1 R&D Concept

In Germany, R&D into disposal of nuclear waste started in the mid 1960s. Waste disposal was and still is a field of scientific-technical research for the national research centres in Jülich, Neuherberg, and Karlsruhe and for the German Geological Survey (BGR) in Hannover. Moreover, the instrument of project funding, a German peculiarity, was used by the responsible authorities to initiate and to finance research activities at universities and in industry.

In the beginning of project funding the responsible departments of the Federal Ministries co-ordinated the scientific projects. In the late 1980s, the Federal Ministry of Research and Technology delegated this coordination task to the project management group “Projektleitung Rahmenplan”, established at the Institute for Underground Disposal of GSF. This group formulated a compilation of major tasks and open R&D problems, which was called “Rahmenplan” – framework plan – repository safety in the post-operational phase [7]. Based on this compilation the necessary R&D-tasks were defined and corresponding projects were funded at several research institutions. This coordination task was continued by the project management group PTE at the Research Centre in Karlsruhe in 1991.

An R&D concept was developed and discussed with relevant authorities and research institutions in the field of nuclear waste disposal. As a result from 1991 to 2001 both research into chemotoxic waste and radioactive waste was covered by an integrated R&D concept – the so called “Förderkonzept Entsorgung”. One of the main advantages was to profit by synergies of both areas. The integration was phased out in 2001, when an conceptual update of radioactive waste disposal research became necessary. Hazardous waste disposal was no longer integrated into the advanced concept, caused by administrative and strategic changes in the respective Federal Ministries [8].

In accordance with the strategic changes in radioactive waste management from 1998 other host rock formations than rock salt were investigated in more detail with respect to their long term isolation potential. Presently, the R&D activities will focus on remaining and urgent questions and problems with regard to rock salt with the aim to come to applicable results within several years.

In 2006 BMWi started to update the R&D concept again for the time period 2007 to 2010. Besides, there are several activities to evaluate the results of R&D projects and to identify further R&D needs, especially for the disposal of heat-generating waste and spent fuel in rock salt.

3.2 Project-Funding

In the 1980s about 90 R&D projects related to radioactive waste disposal were initiated and managed by BMFT, most often realized by the GSF-Institut für Tieflagerung. The project-management group “Projektleitung Rahmenplan” established at this institute was in charge of about 50 further R&D projects, that were performed by other research institutes and universities in the years 1989 until 1992.

On behalf of several Federal Ministries and starting in 1991 a project management group at the Forschungszentrum Karlsruhe is responsible for the project-funding activities [8]. To date about 400 R&D projects were realized on the basis of the R&D concept mentioned above. In addition institutions like BGR and the Research Centres at Karlsruhe and Jülich performed research projects in the field of radioactive waste disposal, acting within the scope of their R&D-programs and institutional budgets.

From 1965 to 1996 about 700 Million Euros have been spent for R&D in the field of radioactive waste disposal. The biggest part of this amount was spent for construction, installation, and operation of research facilities in the Asse research mine (1981-1996). The expenditure of the Federal Government for institutional funding of research activities and project funding is about 37 % each. The remaining 25 % were spent for R&D in the Asse research mine.

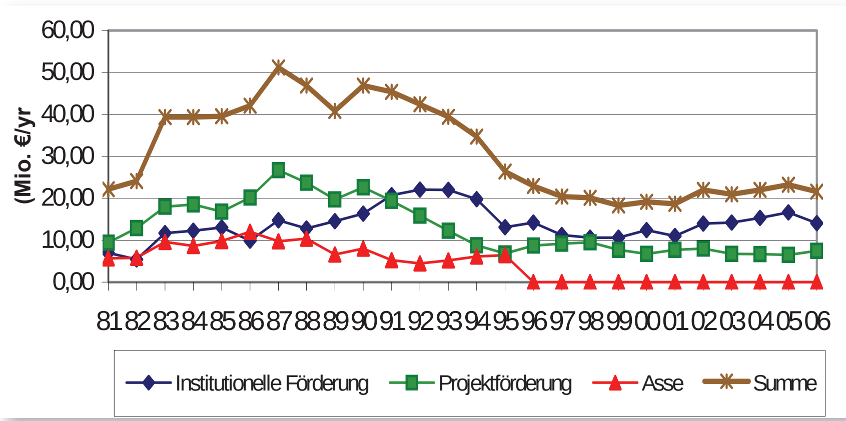


Fig. 2: Funding expenditures in the aforementioned funding areas from 1981 to 2006

Since 1996 the Asse research mine is financed by BMBF because of its responsibility for the closing and backfilling measures.

Presently, BMWi allocates about 9 million Euros per year for R&D projects in the field of radioactive waste disposal, supplemented by still running activities funded by BMBF of about 5 million Euros per year.

3.3 R&D Projects

In the following some examples of R&D projects are described in brief which should give an impression of the present state of science. These sizeable projects were supported and supplemented by several associated R&D projects. These activities led to an advanced state of science and research with regard to underground disposal of hazardous waste especially in rock salt.

R&D projects on disposal of radioactive waste in rock salt were performed on all scales: from laboratory scale to large scale in situ projects, the latter in the Asse research mine, some on a field scale.

From 1985 to 1995 the R&D program Direct Disposal was conducted. It aimed at the development of the disposal concept of direct disposal. All goals of this ambitious project could be reached and the technical feasibility was shown. Details about the full-scale demonstration test will be given later.

The Program Direct Disposal consisted of several large-scale experiments. On of these experiments was the TSDE-Experiment (Thermal Simulation of Drift Emplacement). Unfortunately, two other experiments the “HAW Project” and the drift closure experiment “Dammbau im Salzgebirge” were cancelled in 1992 after detailed preliminary activities shortly after beginning experimental work.

Within the “HAW-project”, conducted from 1982 till 1994, the technology for vitrified high-level waste canister transportation and emplacement was developed and tested [9]. Laboratory tests and an in situ experimental program were conducted. Valuable information on rock salt properties was gained and technology was developed. Unfortunately, the active in situ test had to be given up due to licensing problems and funding uncertainties.

The Dam construction project conducted from 1985 till 1994 was a joint project between Germany, France, Spain, and the EC. It aimed at proving the long-term sealing capacity of an entire dam construction, proving the tightness of the long-term seal experimentally, and to predict the long-term function by model simulations. During the project duration technical components were developed, the necessary design work was performed; measurements on material properties were conducted, and finally, the technical feasibility of the long-term seal could be shown. The large-scale experiment was stopped in 1992; some laboratory experiments were continued till the final stop in 1994 [10].

3.3.1 Shaft Sealing Experiments: The Shaft Sealing Experiment “Schachtverschluss Salzdetfurth II”

In 1992 the mining corporation K+S AG contacted BMBF and offered a proposal for R&D on sealing systems for underground disposal facilities. Especially the long-tem safe closure of shafts was the most interesting subject for both industry and BMBF´s repository research program.

As an operator of several disposal facilities and as Germany’s largest mining company in the field of rock salt and potash mining, K+S AG was interested to establish an approved state of science and technology for shaft seals. Experiences from the closure of abandoned mines were available. However, experience with the stability and the long-term behaviour of seals sometimes was not much advanced. Some former concepts for sealing systems showed poor performance.

BMBF and K+S agreed to conduct a large-scale experiment for the components of a shaft sealing system. A feasibility study was performed under the auspices of GSF – Institute for Underground Disposal. As a result, a basic concept for such sealing systems was developed and large-scale experiments were proposed [11].

In 1996 BMBF started to fund a joint research project of K+S AG and four further research institutions. There were two main research objectives in the project. The sealing systems include a gravel column, which must be stable against subsidence under dry and wet conditions. Gravel columns in mined shafts reach heights of several hundreds of meters.

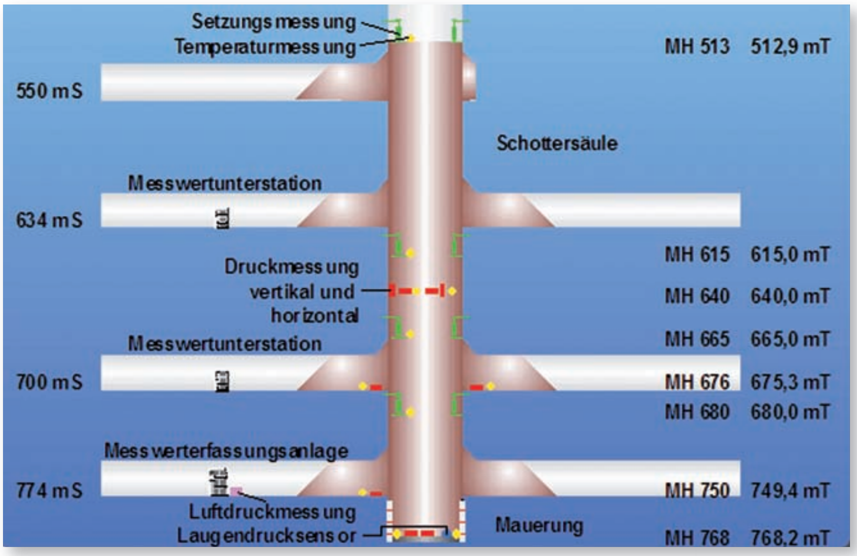


Fig. 3: Experimental set-up of the gravel-column experiment in shaft Salzdetfurth II [12].

The gravel must be self-compacting by applying the conveyance technique, no further settling or jiggling should be necessary. The stabilization of the filling stations depends both on their appropriate geometry and on the preferred complete backfilling of the excavation with gravel.

Figure 3 shows the experimental set-up of the in-situ experiment for measuring subsidence of a gravel column extending over a height of several hundred meters. The instrumentation includes devices for the measurement of subsidence, temperature, pressure (vertical and horizontal) as well as sensors for atmospheric pressure and brine pressure at the bottom of the shaft.

Several tests were performed in two shafts in the mine Salzdetfurth, resulting in basic data suited to model the behaviour of high gravel columns (dry as well as filled with brine in the pore volume). Tests and calculations showed that it is possible to build stable gravel columns that show practically no subsidence. The position of sealing elements could be guaranteed for rather long times with deviation by subsidence in the centimetre-range.

Another large-scale experiment was the permeability test of bentonite seals. The test was realized in the Salzdetfurth salt mine in a boreshaft with a diameter of 2,5 meter and a height of about 25 meters. A static abutment built up from a steel bearing

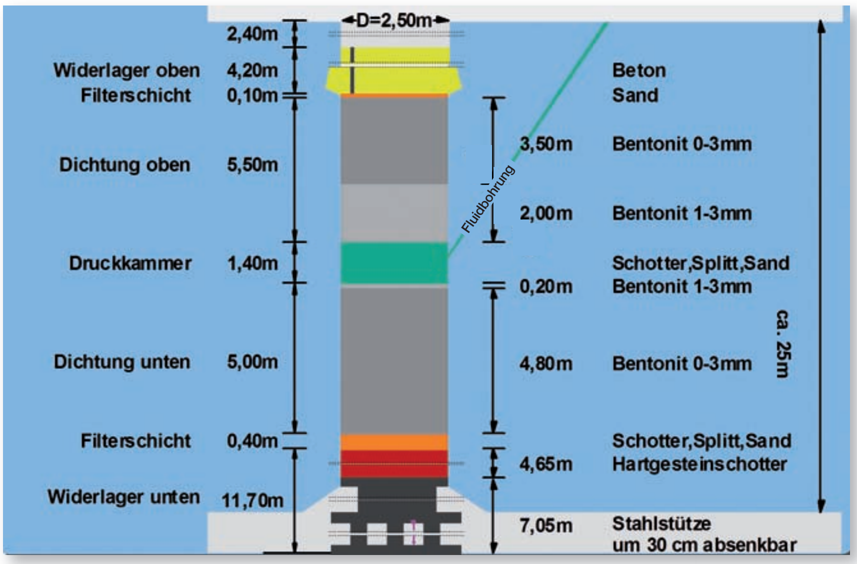


Fig. 4: Principal setup of the bore-shaft experiment in the salt mine Salzdetfurth [12]

construction and a gravel filled column were supporting two bentonite seals separated by the brine injection chamber. A concrete abutment above the upper sealing element stabilizes the experiment and allows brine injection pressures of up to 10 MPa. This abutment mimics the filled shaft above the seals and prevents the uplift of the upper seal. Materials and dimensions were similar to real sealing systems; however, a real mined shaft has a diameter which is larger by a factor of only 2 to 3. Injection of brine was performed in defined injection phases until the experimental pressure limit of 7 MPa was reached. The upper bentonite seal was flooded fast enough to study the processes in saturated bentonite.

The principle set-up of the boreshaft experiment realized in the Salzdetfurth salt mine is shown in Figure 4. The experiment was established in a bored shaft between two drifts in pure rock salt in a region of the mine purely influenced by mining.

Besides the expertise gained, the experiment provided valuable data to understand the behaviour of an embankment made from dry bentonite granules and bentonite pellets that act as a seal in the case of brine influx. This material has the ability to swell when wetted and produces a swelling pressure in the range of more than 1 MPa, depending on material density and brine composition. The hydraulic permeability of the saturated and swollen calcium bentonite is lower than 10^{-10} m/s. Dependent upon the length of the seal the calculated travel time for the brine amounts to several hundreds to thousands of years. The volume of permeating brine is limited to small amounts (several litres up to cubic meters per year). No safety consequences are to be expected.

Sealing systems that combine a subsidence stable gravel column with bentonite sealing elements are suited as shaft seals of sufficient long-term stability. Sealing of shafts, blind shafts and large-diameter boreholes could be build by using materials with long-term stability even when coming into contact with brines. The methods and techniques to produce custommade components are tested. Backfilling methods for the sealing materials exist. After realizing this first proof of suitability, K+S AG started to seal several shafts at the same site by applying the know-how and techniques acquired in the R&D project [12].

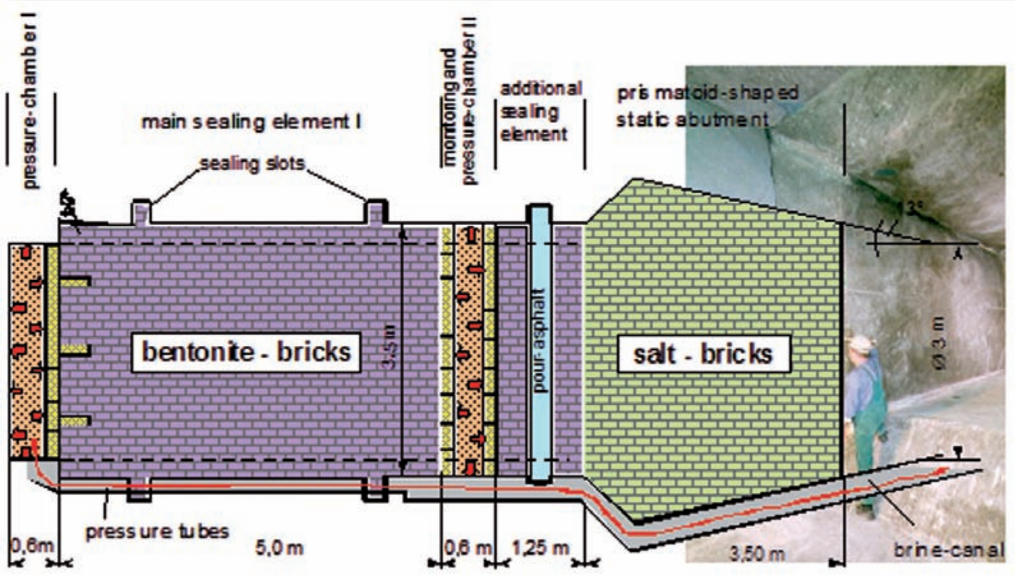
3.3.2 Drift Sealing Experiments: Test Drift Seal Sondershausen and CARLA – The GTS Drift Sealing Project.

To prove the feasibility and efficiency of a drift sealing system made out of highly compacted bentonite bricks, a large-scale sealing construction was planned and built in a drift in the salt mine Sondershausen at a depth of 700 m. The project started in 1997 and was finished in 2002. The in-situ brine permeability test started in September 2000 and was broken off in November 2001 [13].

The experimental set-up of the sealing system consists of the following elements (from the pressure side to the air side), shown in the following Figure 5.

- Pressure chamber I, with a layer of high permeable filter stones
- Main sealing element with highly compacted bentonite bricks, including sealing slots
- Pressure chamber II for the determination of the tightness of the surrounding rock and of the sealing element I as well as for the direct pressure built-up on the static abutment in the final testing stage
- Sealing element II
- A prismatic-shaped static abutment with pressed rock-salt bricks
- A groove in the center of the bottom under the construction with a cross-section of 30 cm x 30 cm with 4 tubes for pressurization, deaeration and control as well as cables to the 80 sensors in three measuring layers for the determination of fluid pressure, contact-pressure, moisture content, swelling pressure and for extensometers

The main requirement for the materials of the sealing system is their long term stability. The proof of long term stability can be based on comparison of their material behaviour with naturally occurring materials like salt clays or basalt (natural analogues). For this reason the existence of natural analogues were used as exclusion criteria for the material selection.



▲ Fig. 5: Experimental setup of the test sealing system in a drift in the mine Sondershausen with an impression of experimental dimensions on the right. The diameter of the drift, which was established with a TBM-device, was widened using a wire saw [13]

Different types of bentonites had been tested with regard to their mineralogical, chemical, mechanical and hydraulic properties. The swelling capability of bentonites is one of the key features for successful sealing against brine or water inflow. To maintain this swelling capacity even under saline conditions, the bentonite has to be built-in with a high dry density without adding water.

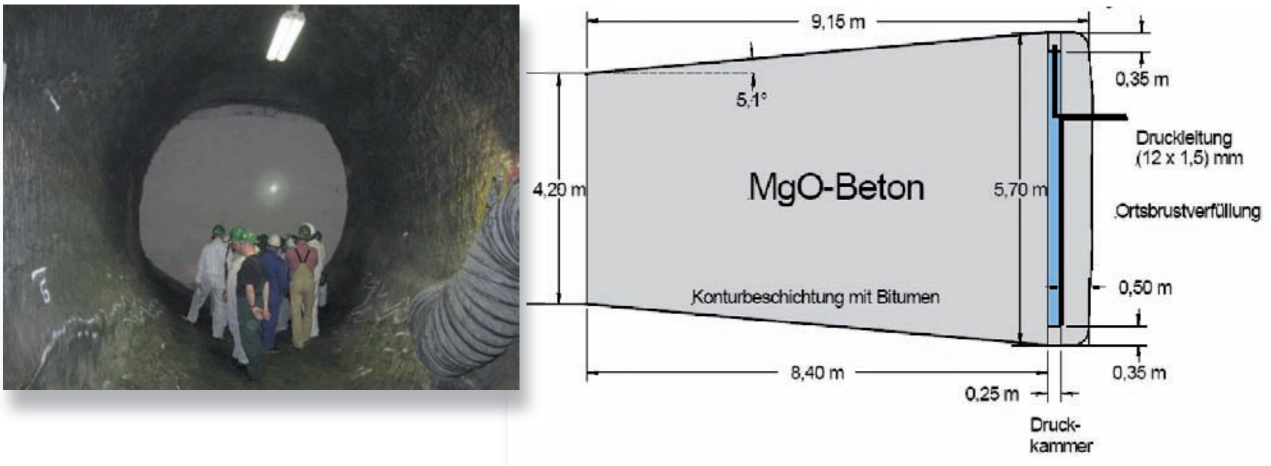
As a result of detailed investigations, bentonite brickwork was recommended for sealing elements in horizontal openings.

For the shaft sealing systems a binary mixture from bentonite compacts (pellets) and granules has been used.

In a saline environment natural bricks from long-term stable materials are an alternative to cement materials. Materials with natural analogues in saline environment are basalt and salt. Laboratory and underground laboratory tests were performed using basalt bricks, sawed natural rock salt blocks and salt bricks, pressed from salt debris. The static abutment of the test drift seal Sondershausen was built with pressed salt bricks.

Type	FS50	FS40
Bentonite content [%]	50	60
Hydraulic conductivity (NaCl-brine) [m/s]	$2 \cdot 10^{-11}$	10^{-12}
Swelling Pressure by constant volume (NaCl-brine) [MPa]	2	4

▲ Table and Fig. 6: Bentonite bricks for drift sealing elements



▲ Fig. 7: This Figure gives a rough impression of the dimension of the large scale test GV-1 in the mine Teutschenthal. The dam built with MgO-concrete has a diameter of about 4 to 6 meters and is about 9 meters long (Source GTS GmbH & Co. KG Teutschenthal)

Some characteristic data for high compacted bentonite bricks are compiled in the upper table. The Figure 6 on the right side shows the building of the first sealing element of the experimental set-up [13].

Bricks with bentonite content from 30 to 70 % were developed by TU Bergakademie Freiberg, Institute of Mining Engineering in cooperation with Preiss-Daimler Industries. Since 1998 some 500 t of such bentonite bricks have been produced and built-in in underground sealing elements in salt mines. The bricks have standard size (250x125x62,5 mm) and can be cut with a saw easily.

In the in-situ test bricks with the natural calcium bentonite “Calcigel” were used. Bricks with Wyoming sodium bentonite (MX 80) were also produced and tested under high fluid pressures in half technical devices with brine and also with water.

Results of the drift sealing experiment were:

- Rate of fluid flow input
- Axial displacements and density changes in the sealing element
- Pressure, in the first place the contact pressure between sealing element and salt contour
- Moisture content distribution in the sealing element.

Long term stability features can be deduced by findings from natural salt clays that may be used as natural analogues. In lab tests the relevant bentonite parameters for saline conditions, like hydraulic conductivity, swelling pressure, and stiffness were intensively studied. Medium-scale tests were performed, showing the suitability of bentonite systems and the tightness of this sealing material. On this scale the bentonite materials used were analyzed and tested prior to in-situ tests.

The in-situ test shows that the function of a sealing system is mainly influenced by technical measures. In a real drift sealing system no cables, tubes, instruments and cemented grooves will disturb and influence the sealing performance. The bentonite sealing element showed the expected behaviour. The technical feasibility to use bentonite bricks for liquid-tight sealing elements was shown.

The sealing materials bentonite bricks, binary mixtures of bentonite compacts and granules as well as bulk mixtures with required properties might also be used as sealing materials for repositories in argillaceous or crystalline rocks.

The project CARLA – the GTS drift sealing experiment – is a still active project which was presented and discussed with an audience from science and administration in October 2005. The scientific basis was established in the research project Development of Basic Concepts for Long-Term Stable Dams for Repositories in high dissolvable salts (Carnallitit) for Underground Repository or Waste Utilisation Mines [14], [15].

All previous R&D activities for drift sealing systems in salt mines were related to locations in rock salt. Such ideal rock salt areas for sealing elements cannot be found in some potash mines. Until today no long term stable sealing system for locations in high-dissolvable salts was developed and tested successfully.

This R&D task has an exceptionally high degree of difficulty because of the especially complex geologic and mineralogical conditions of the Teutschenthal-Mine with Carnallitit, Kieserit and Tachyhydrit as main components of the salt rock. In the opinion of engineers and scientists the technical solutions for the Teutschenthal-Mine should be transferable to all the other locations with high-dissolvable salts as a host rock.

A basic concept and dimensioning rules for long term stable drift sealing systems in Carnallitit locations was developed. This concept is fundamental for the recent research activities at tested sealing components and for dams in Carnallitit Mines.

Main objectives of the work was the study of the excavation damaged zone and possibilities of its sealing, possibilities to take control of the composition of the inflowing salt solution, the selection and test of well-suited building and sealing materials as well as their construction and large scale test.

3.3.3 The TSDE- / BAMBUS-Project

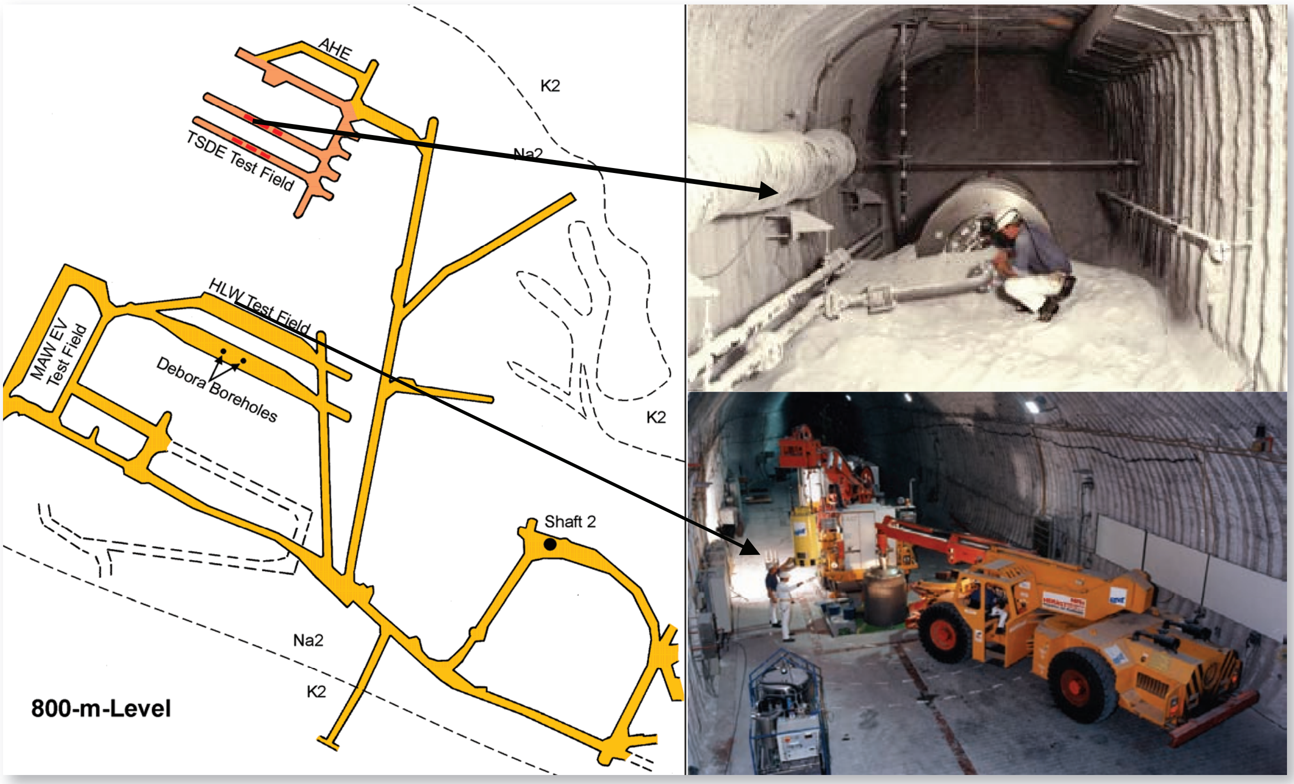
This project was originally started as one of the large scale experiments performed during the execution of the Program Direct Disposal.

The objectives of the project were to improve the understanding of crushed-salt backfill behaviour in a repository for radioactive waste in rock salt, to increase the data base on important phenomena and processes in the repository, and to further develop the computer codes and material laws required for predicting these processes. The project focused on two concepts for emplacement of heat-generating waste in a deep repository in salt: (1) the drift emplacement concept and (2) the borehole emplacement concept. The work was divided into in-situ experiments carried out in the Asse salt mine, laboratory investigations on crushed-salt backfill, benchmarking of material models for crushed-salt backfill, and development of thermo-mechanical models [16].

The results of the project have shown that the performance of repositories in rock salt is determined by continuously evolving processes with rather simple loading histories. These processes are qualitatively well understood and can be simulated over wide ranges. Some parameters were identified that mainly influence these processes and that require further research in order to improve the predictive capability of the thermo-mechanical codes and material laws. In the experimental investigations the data base on relevant parameters was increased and general relationships were obtained for the interdependence of backfill porosity, permeability, and thermal conductivity.

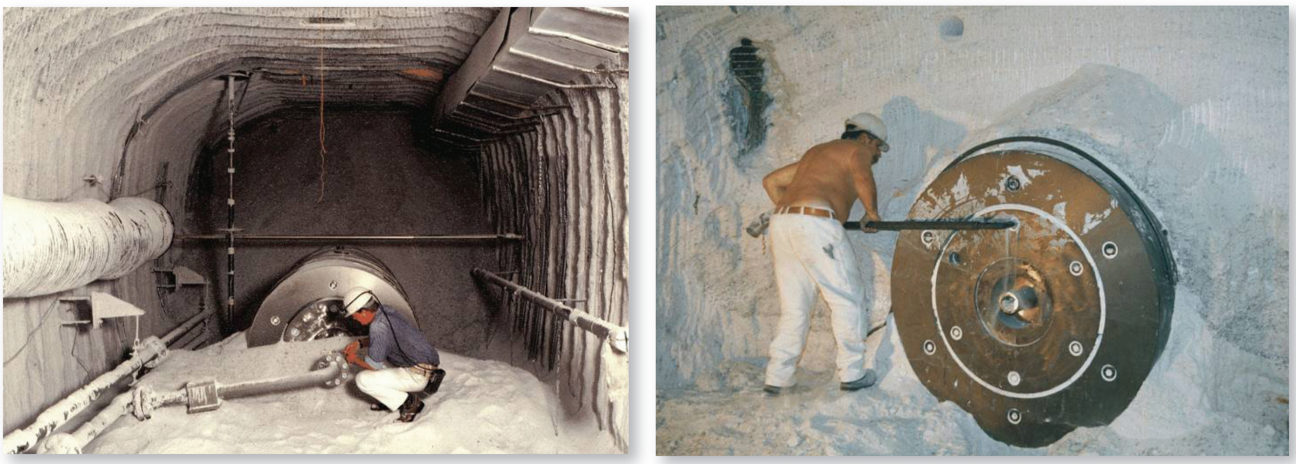
In 1990 the experiment “Thermal Simulation of Drift Emplacement” (TSDE) started. After nearly ten years of heating it was terminated in February 1999. The experiment had been designed in the late 1980s to support the development of the drift emplacement concept described above. Measurements had been made since 1990 in and around two simulated repository drifts in each of which three electrically heated POLLUX dummy casks had been installed. Weight, dimensions, and heat capacity of the containers as well as the drift dimensions correspond to the layout of the drift emplacement concept. Thus, the test field mirrored a part of a typical repository emplacement panel. During the project, temperatures, stresses, backfill density and permeability, rock deformations, gas generation, and gas transport were measured. These data were compared with calculation results of numerical models developed by the modeling groups.

The results of the TSDE-experiment provide a sound basis for the emplacement of casks in drifts in rock salt. They also provide a deep insight into the processes in the backfill material of drifts and the surrounding rock zone. Numerical models and associated codes enabling the simulation of thermo-mechanic behaviour were successfully developed. After a cooling phase the experiment was partly decommissioned. During the post-experimental phase measuring devices were reexamined and recalibrated to check the quality of data and the behaviour of the devices.



▲ Fig. 8: The TSDE- resp. BAMBUS-test field (right, upper part) and the HAW-experimental test field (right, lower part) in the Asse salt mine at the 800-m level are shown. The location of these test fields is shown on the left (Source GRS, Research Project FKZ 02E9783 Preliminary Report)

In 2003 the TSDE-(TSS-) experiment was terminated. At this time it was one of the word-wide longest-lasting heater-experiments performed deep underground under conditions similar to that in an HLW-repository in rock salt. Right now a national benchmark exercise comparing modelling with different constitutional laws for rock salt behaviour is being realized as a joint project from several universities and industry. Preliminary results were presented at the SaltMech6-conference in 2007.

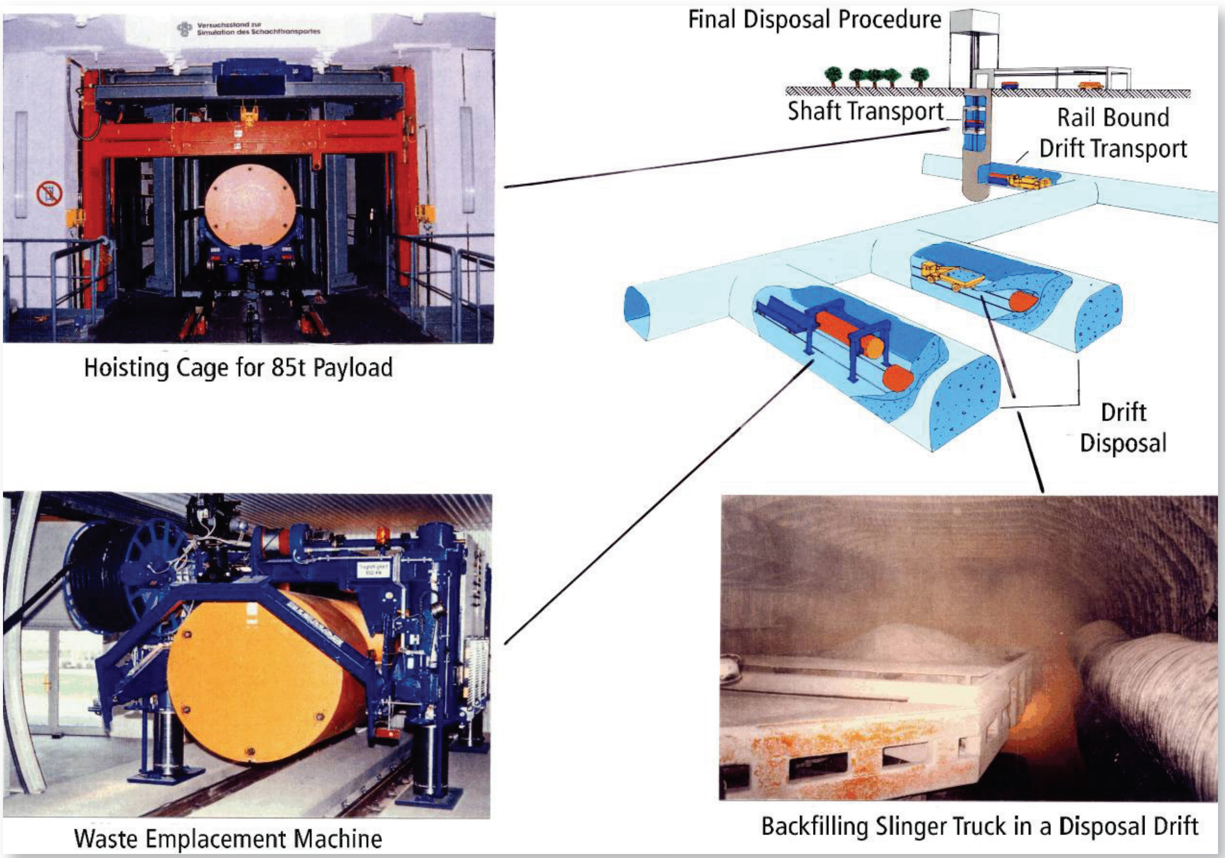


▲ Fig. 9: View into one of the test drifts of the TSDE- (TSS-) experiment during the installation phase (left) and removal (right) of a cask

4 R&D Achievements

One of the most important achievements was the development of the Direct Disposal concept in the framework of an ambitious R&D program starting in 1985 and finished in 1995. This program comprised of the subprograms, a) spent fuel conditioning and cask development (POLLUX casks and canisters), b) demonstration tests (emplacement and handling technologies for heavy payloads, THM behaviour of crushed salt backfill (TSDE and BAM-BUS), c) conceptual design of the disposal systems, and d) laboratory tests. All large-scale tests were successfully executed and could be concluded according to schedule.

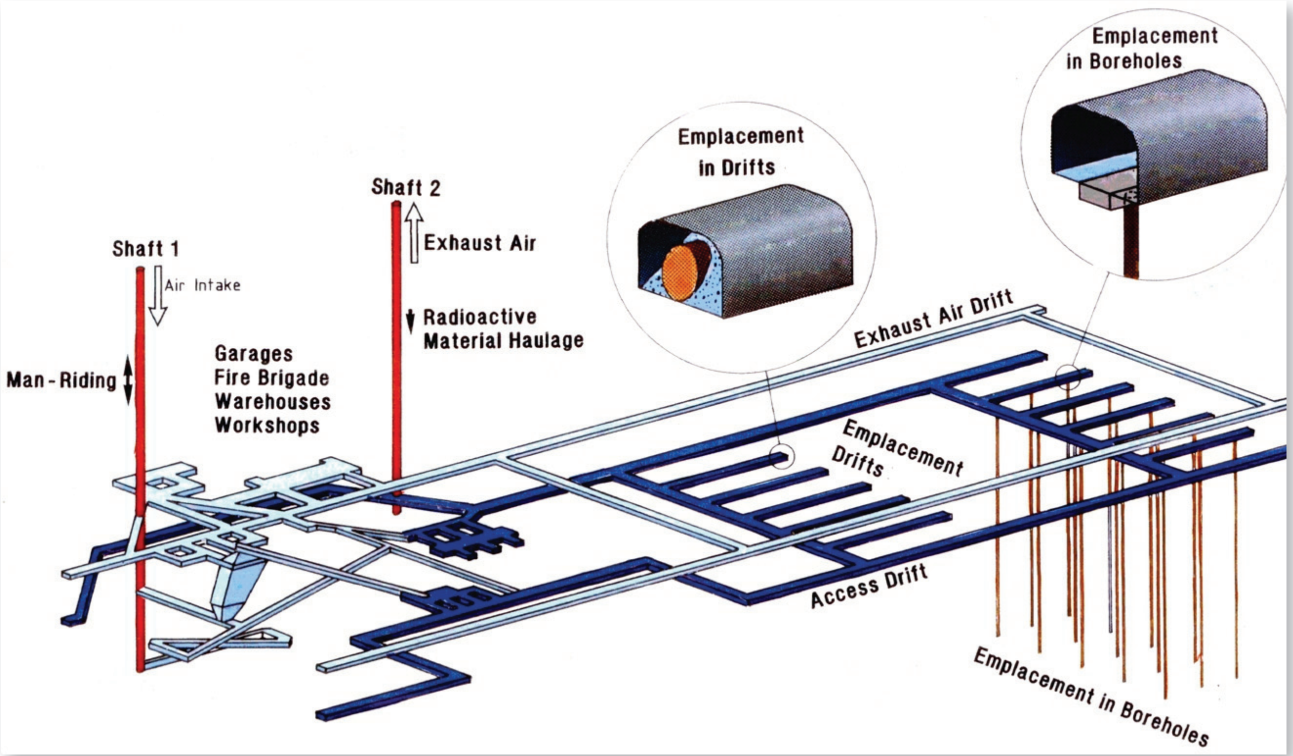
In 1994, as a consequence of this successful R&D projects, the Atomic Energy Act was amended and Direct Disposal of spent fuel became a disposal option legally equal to reprocessing. Today, after an additional amendment of the Atomic Energy Act in 2001, Direct Disposal is the only legally accepted way to dispose of spent fuel and high-level waste [17].



▲ Fig. 10: Some of the components of the German reference concept for high-level waste disposal in rock salt (Source DBE Technology)

Lessons learnt from this and other experiments, which should only be mentioned shortly – “Active Handling Experiment with Neutron Sources” (AHE) [18], “Development of Borehole Seals for Radioactive Waste” (DEBORA) [19] and many small scaled in-situ-experiments and laboratory tests - were that the performance and the reliability of large technical equipment is proven. Material behaviour can be described adequately by experiments and models.

Moreover, it is commonly acknowledged that the use of large-scale or full-scale in-situ demonstration experiments, also performed in underground research laboratories, is indispensable. This is not only essential for scientific reasons but also because of its importance to get public acceptance for safe and secure handling of technology.



▲ Fig. 11: The German reference concept for high-level waste disposal in rock salt (Source DBE Technology)

Besides the engineering work and large-scale experiments, laboratory experiments and research to further develop the instruments for performance assessment have been performed, and knowledge has been accumulated. Progress in research projects is presented periodically under the auspices of the Project Management Agency PTKA-WTE [20].

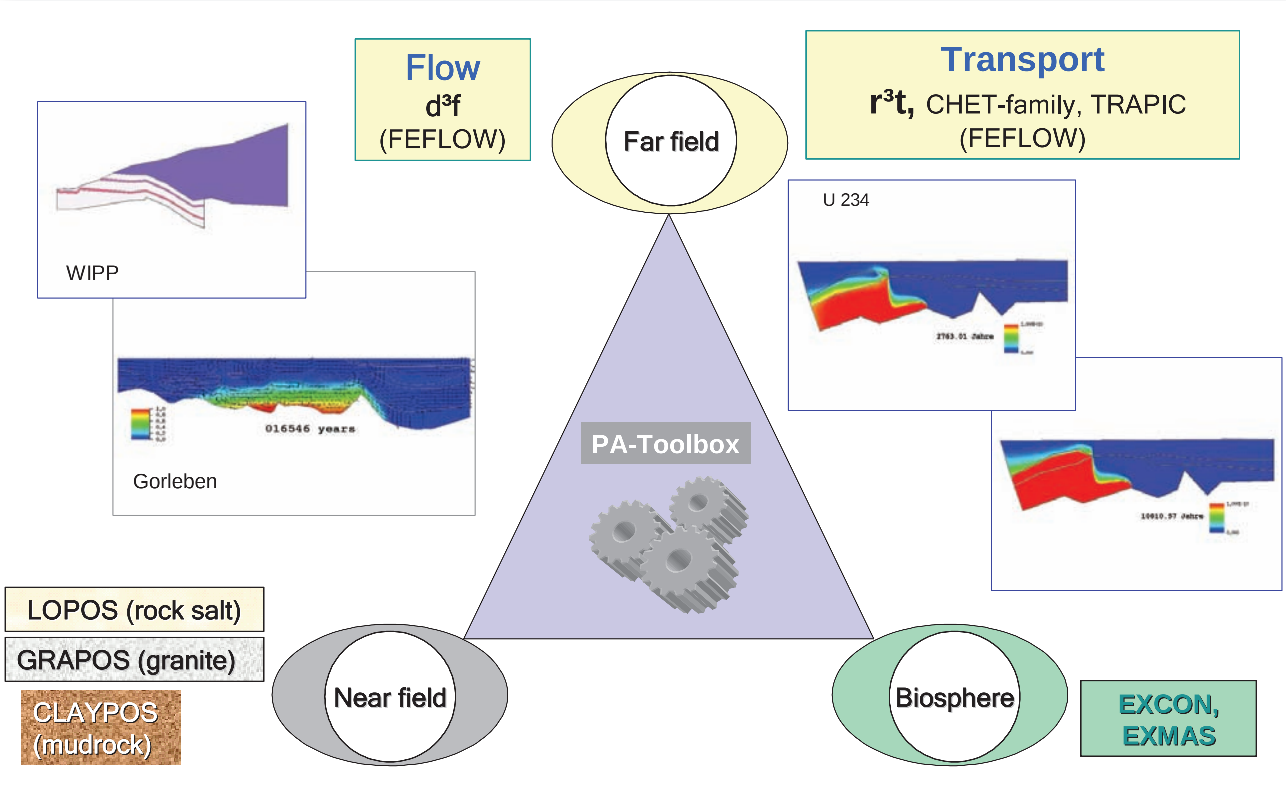
A large step forward was done by developing and testing the tools and instruments to be used in modelling and performance assessment. A good part of this work was done by GRS mbH at the location Braunschweig.

Two main areas in performance assessment are simulation of groundwater movement and radionuclide transport. For the special situation of a repository in rock salt and its scenarios of long-term behaviour, the variable salinity of the groundwater has to be considered. During the last years tools had been developed that allow to perform 3d-modelling of groundwater flow occurring in the overburden of a salt dome. It is now possible to simulate groundwater movement for large (km^2 range) and complex hydrogeological systems in acceptable calculation times of hours. Besides, transport models are also available. This outstanding performance could be achieved by developing sophisticated numerical models and algorithms and using massive parallel computers. These programs can also be applied for problems not related to radioactive waste disposal, e. g., simulating the spatial and temporal evolution of the impact of mining activities on groundwater composition, problems of coastal aquifer management, or transport of heavy metals in the environment [21].

Moreover, up to now there are no indications that rock salt is not suitable to accommodate a repository for heat generating waste. Serious faults and mistakes when developing and modelling disposal systems become more and more improbable.

At the end of the 1990s the Government decided that other host rock types than rock salt should be investigated with respect to their suitability for radioactive waste disposal, too [22].

Although the Gorleben salt dome did not prove unsuitable, the Federal government came to the conclusion, that further exploration of the salt dome cannot clarify questions addressing conceptual and safety-related issues. A moratorium was imposed and



▲ Fig. 12: Components of the “PA-Toolbox” for the safety assessment of underground repositories

several issues, e.g., the problem of gas-pressure, criticality, retrievability and suitability of host rock formations other than rock salt, were studied subsequently. In November 2005 the final report was published by BfS [23].

To reach an adequate level of knowledge of all available German host rock types in due time in order to make a fundamental comparison of different disposal systems is a challenge. Therefore, R&D activities on disposal systems in argillaceous or granitic rock were intensified. The respective projects are focused mainly on argillaceous rock and comprise feasibility studies [24], laboratory experiments and modelling tasks. The activities are carried out to a certain extent in foreign underground laboratories in Switzerland, Sweden, and in France [25, 26]. The results achieved to date have contributed to a certain level of knowledge and expertise.

Now, the R&D activities related to rock salt should be intensified and completed. In Germany only argillaceous rock is another candidate to host a repository for radioactive waste. R&D related to disposal in granitic rock will focus on topics and system components related with engineered barrier systems, which might perhaps assignable to rock salt disposal concepts.

On principal, the feature of applied and basic research is to identify new demands and to perform further detailed investigations to fully understand a defined problem. In parallel the confidence in thoroughly investigated phenomena should increase. Therefore, the mere existence of open scientific questions related to specific details did not put into question a principal decision concerning a technical project.

5 Outlook

BMBF and BMWi are funding non-site specific R&D for underground disposal of hazardous waste as a precautionary measure. Rock salt still is the favourite candidate host rock for disposing of heat-generating nuclear waste in Germany. Chemotoxic waste is disposed of in licensed facilities in rock salt in remarkable dimensions – several 10- to 100-thousands of tons per year of hazardous chemical waste goes underground at, meanwhile, 4 sites. Underground disposal is believed to be the only long term safe way to manage these waste types.

The knowledge gained about disposing of heat-generating waste in rock salt during the past decades has reached a certain level of maturity. It was shown that technological problems can principally be tackled and solved. Techniques for the emplacement of spent fuel and vitrified waste are at hand. A lot of knowledge has been accumulated about the behaviour of the host rock and crushed-salt backfill. Databases and models were permanently improved. Instruments to be applied in safety assessment exercises are available.

Although with regard to HLW disposal in rock salt the status of knowledge is well advanced, some key projects are being carried out that will have an important impact on the salt concept. Their results will be valuable for future decision making. The first one, being part of the EC-ESDRED project, aims at completing and optimizing the Direct Disposal concept by a full-scale demonstration of the emplacement of spent fuel in vertical boreholes. The second project deals with the development of an advanced safety concept for an HLW waste repository. Performance assessment exercises were conducted within the framework of German R&D projects at the end of the 1980s and in the first half of the 1990s. Since then remarkable developments improved the basis for developing an advanced safety case. A joint R&D project is being performed to identify major needs in case of realization an integrated safety case in the future.

Moreover, using the approach of proving the safe enclosure without release for the expected evolution of the repository is considered to be more appropriate for a rock salt repository and takes advantage of its specific properties.

Up to now there are no scientific and technical indications that rock salt is not suitable to host a repository for heat-generating waste. For several reasons German scientists participate in international R&D projects and programs in other rock types than salt on the basis of cooperation agreements. The Federal Government decreed to have alternative host rocks investigated. It is important to get a better understanding concerning the pros and cons of these host rocks. The research activities in foreign underground laboratories in argillaceous or crystalline are necessary, because no German underground laboratory is available [27].

Last but not least, these research activities strengthen the international co-operation and the information exchange and motivate the scientific community to cooperate in EC-Framework Programs [28].

In Germany, the knowledge concerning indurated clay increased during the last years. The results achieved so far from national and international projects, including the so called Clay study [29] of BGR, allow a better and more qualified understanding of the pros and cons of HLW disposal in clay. On behalf of BMWi, the German Geological Survey (BGR) evaluated the potential of alternative host rocks like crystalline and argillaceous rocks. Concerning argillaceous rock, an abridged version of the final report was published in 2006, the detailed scientific report in 2007.

However, to reach a state-of-the art like in rock salt, there are R&D efforts necessary that should deal with issues like conceptual and safety related questions or all problems related to host rock characterization.

The research concept of BMWi, that is the basis for future R&D, is presently being revised and discussed, inter alia, by experts from several German research institutions. This activity is of special importance against the background of streamlining and focusing the research activities taking into account budget constraints and adaptation to future demands, as well as priorities and perspectives. At the end of the phase 2007-2010 there should exist a comprehensive and resilient refurbishment of existing scientific-technical know-how duly to the end of the moratorium at the Gorleben investigation site.

At that time a decision about future German activities in radioactive waste disposal should be possible on the basis of objective, scientifically supported facts.

Details on the research concept will be given by BMWi at this conference in another paper [30]. ■

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2B.02 Site Characterization Methodology for High Level Radioactive Waste Disposal in Beishan

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Abstract
Beishan, located in Gansu province, Northwestern China, has been selected as a potential area for China's high level radioactive waste repository. Granite is the host rock in this area. Since 1999, systematical site characterization technologies have been used, and the performance and effectiveness of methods such as remote sensing technology, geological mapping, surface electromagnetic survey, diamond borehole drilling technology, borehole injection test, borehole acoustic televiewer survey, borehole radar survey, hydrogeochemical logging and geostress measurement have been evaluated. So far, those methods have been proved effective and useful in Beishan, a granite site.

1 Introduction

China started R&D program for high level radioactive waste (HLW) disposal in 1985. The objectives of the program are to establish capability for the final disposal of HLW and to build China's HLW repository in about 2050.

Site selection, started in 1986, has been an important part in China's HLW disposal program. The whole site selection process was divided into 4 stages: nationwide screening, regional screening, area screening and site confirmation. During siting process, the following factors have been considered: social-economic factors and natural factors, including population, economical potential, plant/animal resources, mineral resources, land use, local public attitude, geological/ hydrogeological conditions and engineering conditions.

Since 1990, most of the efforts have been concentrated to Beishan area, Gansu province, which is considered as the most potential site for China's HLW repository. Studies include regional crust stability, tectonic evolution, lithological studies, hydrogeological studies, preliminary geophysical survey, borehole drilling and systematical borehole tests.

Since 2000, systematic site characterization studies have been conducted in Beishan area. Four boreholes were drilled, site characterization methods were studied. The performance and effectiveness of methods such as remote sensing technology, geological mapping, surface electromagnetic survey, diamond borehole drilling technology, borehole acoustic televiewer survey, borehole radar survey, hydrogeochemical logging and geostress measurement have been evaluated, those methods have been proved effective and useful in the granite site.

2 Beishan Area

The Beishan area is located in northwestern China's Gansu province. It is a remote arid Gobi desert area, with very few inhabitants. The precipitation in the area is about 70 mm/a, while evaporation reaches about 3000 mm/a. Tectonically, Beishan is located in the eastern part of the Tianshan-Beishan folded belt in western China. The candidate host rock is granite. The crust in the area has a block structure, with a crust thickness of 47–50 km. The depth contour of the crust strikes nearly EW, with very little variation. The gravity anomaly is approximately -150×10^{-5}–225×10^{-5} m/s². The gravity gradient is less than 0.6 m Gal/km. On the gravity-anomaly map, the gravity-anomaly contour is distributed very sparsely, without obvious step zones, indicating that there are no large faults extending to the depth of the crust. The seismic intensity of the region is less than 6, and no earthquakes with Ms > 4 ¾ have occurred. The topography of the area is characterized by a flat Gobi and small hills, with elevations ranging between 1000 m and 2000 m. Variations in height are usually several tens of meters. Since Tertiary, the region is slowly uplifting without obvious differential movement. Geological characteristics of the Beishan area show that the crust in the area is stable, and it has a great potential for the construction of a HLW repository [1].

The Beishan area is poor in groundwater resources. Pumping tests carried out by local geological teams in the 1980s in the area have shown that, for most of the wells, the outflow rates are less than 50 m^3/d. Beishan groundwater can be divided into three categories: (1) an upland rocky fissured unit; (2) a valley and depression pore-fissure unit; and (3) a basin pore-fissure unit. The upland rocky fissured unit is the most prevalent one in this area, occurring in weathered and structural fractures. Groundwater recharge is primarily from precipitation infiltration, with discharge mostly through evaporation and lateral outflows into the fracture water-bearing zones, intermountain areas, and valley depressions. The present water table in the potential site area is 24-46m below the surface.

In Beishan area, eight granite sections have been selected as potential sites for the future HLW repository. Among them, three sections (Jiujing, Xiangyangshan-Xinchang, and Yemaquan) have been chosen as the sites with the most potential, and detailed work is now concentrated on them.

3 Site Characterization Methodologies

3.1 Remote Sensing Technologies

Beishan area is a Gobi desert area, with very good outcrops of bedrocks and very little vegetation. These features provide good basis for the use remote sensing technology in surface geological mapping and hydrogeological investigation.

Satellite images such as TM (Thematic Mapper) images and SPOT images were used geological investigation in Beishan area. The composed image of TM band 7, 4 and 2 have been successfully used to identify the distribution of different rock types, dykes, faults, lineament and Quaternary basins in Jiujing, Yemaquan and Xiangyangshan-Xinchang sections. SPOT images have high geometric resolution. They are mainly used together with TM images for the interpretation of lineament and faults. With the help of TM and SPOT images, draft geological map can be made before field investigation, this greatly enhance the working proficiency. With the combination of satellite and DEM (digital elevation map), 3D surface site map can be produced.

In 2006, another satellite image, called Quick-Bird, was used in Jijicao block, which is selected as one of the most potential sites in Xinchang section. Quick-Bird has such a resolution of 0.61 m that most lineaments, such as small faults, dykes, fracture zones etc., could be found easily. It has been proved that Quick-Bird satellite image is a necessary tool for detailed geological mapping on a scale of 1/2000 in Beishan area.

3.2 Geological Mapping

In the Jiujing section, the geological mapping (1/50,000) was completed during 200-2001, covering an area of 462 km^2. Based on the detailed field investigations and laboratory work, a geological map has been generated.

In Jiujing section, four granite units are recognized: Jiujing, Bantan, Jiazijing, and Shimenkan. The Jiujing unit is composed of middle-Proterozoic tonalite with an area of 220 km^2. The Bantan unit is composed of porphyritic-monzonitic granite with an area of 53 km^2. The granite in this unit is of good integrity with less deformation and fractures. Thus, it was chosen as the candidate unit for drilling, and borehole BS01 and BS03 was located in the northern part of this unit. It has been proved that geological mapping can give basic geological information for site selection, e.g. the size and history of granite body.

Besides the geological mapping (1/50000), the detailed geological mapping (1/2000) was first conducted in Jijicao block (belong to Xinchang section) with a size of 3km×4km. The detailed geological map can provide detailed information of all faults, fracture zones, and lithology. It can optimize the location design of borehole so that more underground information can be obtained at the same cost.

3.3 Surface Geophysical Survey

Detailed surface geophysical survey was conducted in Beishan area in order to identify the bearing of faults and their depth. Electormagnetic survey was proven the most effective surface geophysical method in identifing faults in granite site. A STRATEGEM resistivity profile measurement system were used in Beishan site, the results show that the fault zone in granite is characterized by low resisitivity (<100 Ωm). Figure 1 shows the resistivity profile of the Shiyuejing Fault, indicating the dipping angle (85°), depth (larger than 2 km) and width (60 meters) of the fault, which is quite consistent with the trenching and drilling observations [2].

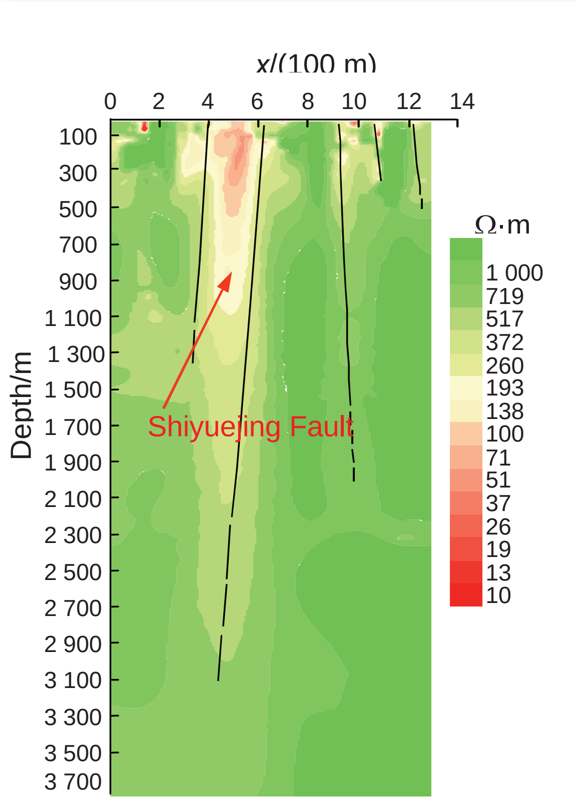

Fig.1: Electromagnetic profile of Shiyuejing fault in Beishan granite site

3.4 Borehole Drilling Technology

The purpose of drilling for site investigation is to get necessary information on the suitability of a site, through progressive compilation and evaluation of data from drilling and subsequent measurement. The drilling and subsequent measurements entail some disturbance. Drilling water, lubrication oils and dirt may be introduced into the rock by drilling. So, it is important to choose proper drilling technology.

The BS01 and BS02 are the first 2 boreholes in the Jiujing section, Beishan area. BS01 is a vertical hole with depth of 700 meters, it is drilled to evaluate the potential granite unit, while BS 02 an inclined one with depth of 500 meters, focused to evaluate the characteristics of the key NE-striking fault: Shiyuejing Fault. The drilling technology with diamond drilling, core-sample taking drilling, pure water as drilling fluid and wire sampling technology for core recovery, was used for the 2 borehole drillings, which has been proved successful. For BS02, pipe casing and mud drilling method were used in order to go through the NE-striking Shiyuejing fault zone. By using this method, the borehole was successfully completed, and perfect core samples for the fault have been obtained, with core recovery over 95 %.

3.5 Borehole Hydrogeological Test—Injection Test

Granite is a fracture media, one of the most important features is the permeability of the rock mass and the fractures. Injection tests can be conducted in selected intervals in completed boreholes in order to obtain the parameters of permeability. Usually there are 2 types of injection tests: transient injection test for intact rock interval and normal injection tests for fractures. In BS01, 6 intervals with fractures were chosen to carry out normal injection tests by using double packer system, while 4 intervals of intact rock were selected for transient injection test. The length of measurement intervals is about 8 meter. The results of normal injection tests show that the permeability for the fractured zone ranges between 1.74×10^{-6} and $<8.0\times10^{-7}$, while for intact rock between 1.85×10^{-9} and 2.66×10^{-8}, this shows the low permeability of the granite media. The experiences in Beishan site shows that injection tests could be the best tests to obtain the permeability of deep granite formation.

3.6 Borehole Acoustic Televiewer Survey

A high resolution of acoustic borehole televiewer system (Fac-40, made by DMT-ILG Company in Germany) was used in BS01, in order to investigate the features of fractures and the integrity of the rock mass along the borehole. The resolution of this system for fractures can be down to 0.1 mm, which is very powerful for the study of fractures. In BS01, the interval between 60 and 550 meter deep was investigated by this system. With the images obtained from borehole televiewer, the directional information of borehole wall, borehole deviation, fracture bearing, core orientation and other important data be obtained. All the data obtained during measurement can be processed by WellCAD software, and the statistics of fractures and the distribution of fractures can be obtained. For example, in an image for BS01, a fracture with its azimuth of 192.5° and tilt of 69° can be clearly seen at the depth of 425.9 meters. The statistic of fractures, resulting from the Fac-40 televiewer survey, reveals that the fractures can be divided into 2 groups: NE-striking and NW-striking. Most of the NE-striking fractures occur below the depth of 230 meter, but with 2 opposite azimuth: NW and SE. Above the depth of 230 meters, most of the fractures are NW-striking. The average fracture density is about 8-12 per 10 meters. In a granite site, acoustic borehole televiewer measurement should be an essential way to study the fractures in borehole.

3.7 Borehole Radar Survey

A RAMAC borehole radar survey system, produced by the Mala GeoScience Company in Sweden, was used in borehole BS01 in Beishan site. It is the first time to use such equipment in China. BS01 is the first borehole for China's site characterization program. During measurement, the dipole reflection mode was conducted by using the following parameters: Single reflection mode, 100 MHz antenna, antenna center separation: 2.9 m; stacks: 32; time window: 1043 ns; sampling frequency: 1007; distance interval: 0.2 m. Results show that the penetration depth of radar wave in the borehole is about 20 meters and there are 22 reflectors cutting the borehole. The results are well consistent with the data obtained by geological logging, borehole television survey and other geophysical survey. The practice has proved that the borehole radar system is an effective tool to understand the extension of fractures and the integrity of rock mass, which is essential to site characterization program for high level radioactive waste disposal.

3.8 Hydrogeochemical Logging

Deep geological environment in granite is vital to evaluate the suitability of the site. By using hydrogeochemical logging along boreholes, the deep environment parameters such as temperature, pressure, redox potential, pH, dissolved oxygen, conductivity and salinity can be obtained. During the measurement in Beishan, a Mont Sopris logging system with probes of temperature, pressure, redox potential, pH, dissolved oxygen, conductivity and salinity were used, and the results have shown such methods are useful to obtain those parameter. In BS01, measurement was conducted in the section of 0 to 500 m, while for BS02, 0 to 300 m deep, BS03, 70-490 m deep. The results have revealed that the deep environment is characterized by reducing, high salinity, neutral features. By analyzing the measurement curves, the water bearing zones and their outflow of water can also be identified.

3.9 Geostress Measurement

Hydrofracturing method was used to measure the in-situ stress at borehole BS01 and BS03 in Beishan site. The major results shows that the maximum lateral principal stress ranges between 7.72 MPa in the shallow and 25.66 MPa in the deep, and has a direction range between N25°E--N45°E. The practices in Beishan show that hydrofracturing method could be the most suitable one to measure geostress.

4 Conclusions

Site characterization practices in Beishan area, Gansu Province, northwestern China, have proved several useful and effective methods for granite site, such as: remote sensing technologies, geological mapping, surface electromagnetic survey, diamond borehole drilling technology, borehole injection test, borehole acoustic televiewer survey, borehole radar survey, hydrogeochemical logging and geostress measurement. By using those effective methods, some key parameters for site evaluation, concept design and performance assessment can be obtained. However, it is still necessary to develop further cost-effective and high-tech methods for site evaluation, in order to ensure the accuracy and the effectiveness to obtain site data. ■

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2B.03 Methods for Assessing the Performance and Safety Relevance of Geotechnical Barrier Systems in a Repository in Rock Salt

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Abstract

In order to understand the contribution of geotechnical barriers to the long-term safety of a repository of high-level radioactive waste and intermediate level waste with considerable heat production in rock salt, model calculations have been performed for the normal evolution scenario and some water intrusion scenarios. The primary objective of this study was to test the ability of existing performance assessment tools. Realistic boundary conditions and parameters were used, but the model calculations have not been performed for any real site. The scenarios have been derived via a top-down approach without systematic scenario evolution. A safety assessment of the results must not be done. The results allow only deriving valuable conclusions with respect to the system behaviour and the relevance of certain events and processes. When the shaft seal performs as intended, small amounts of water may intrude into the infrastructure area. Due to the compaction of the salt backfill the solution will be pressed out again within short time. A reduced effectiveness of either shaft seal or drift seal does not impair the containment function of the repository system directly. Combined effects may result in radionuclide releases. The long-term compaction behaviour of crushed salt as backfill material affects the model results strongly, but some uncertainties exist with respect to this process.

1 Introduction

Owing to its special characteristics, rock salt is being considered in Germany as a potential host rock for a repository of high-level radioactive waste and intermediate level waste with considerable heat production. The safety of the repository has to be evaluated for 1 million years. The safety concept is based on the persistence of the isolating rock zone, which must contain the radioactive waste over very long times. Radionuclide release from the isolating rock zone should be insignificant during this time and should affect the natural conditions at the site outside the isolating rock zone only marginally. The barrier function of the isolating rock zone relies on the properties of the rocks and the performance of the geotechnical barriers.

The elements of the geotechnical barrier system comprise the shaft seal, the drift seals, the borehole seals and the backfilling. The seals contribute significantly to the safety functions containment and retardation by protecting the emplacement areas against water intrusion and by impeding the transport of mobilised radionuclides, respectively. In order to understand the contribution of the geotechnical barrier systems to the long-term safety of a repository of high-level radioactive waste and intermediate level waste with considerable heat production in rock salt, the evolution of the disposal system has to be assessed. Shaft seal and dams in the repository must provide their barrier function until the compaction process of the backfill material has reached the final stage. At a well-characterized site and with a correspondingly adapted disposal concept, geological processes should not result in an impairment of the isolating rock zone during at least one million years. Upon normal evolution of the disposal system, only limited amounts of water stemming from the backfill material and the surrounding rock salt can come into contact with the emplaced wastes over prolonged time

On the basis of a realistic repository concept at a generic site, which resembles the geological conditions at salt domes in Northern Germany, model calculations have been performed for the normal evolution scenario and some conceivable water intrusion scenarios. The primary objective of this study was to test the ability of existing performance assessment tools to perform the necessary consequence analyses[1]. It was neither the intention of this study to perform a safety assessment for a real site nor to contribute to it.

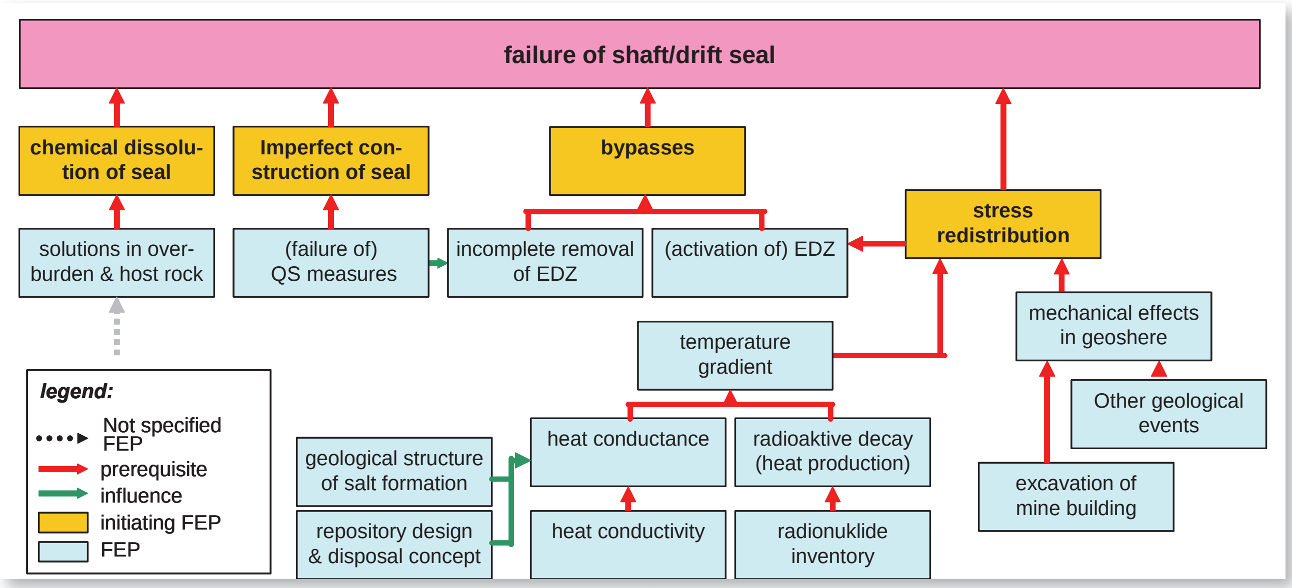
[1]This work was supported by the German Ministry of Economics under the grant no. 02 E 10055.

2 Investigated Scenarios

The normal evolution of the repository system is characterized as follows. The repository is completely backfilled with crushed salt in order to reduce the void volume. The backfill has a low water content of 0.1 per cent and its initial porosity is about 40 %. Apart from the shaft, no hydraulic connection exists or will develop with time to areas outside the isolating rock zone. Also, there are no undetected macroscopic fluid inclusions within the isolating rock zone or, if they exist, they will discharge during the opening of the mine areas. This will be ensured through careful siting and design of the repository area based on in-depth understanding of the geological development of the site as well as respective on-site investigations.

Due to the convergence of the surrounding rock salt, the backfill in the mine openings will be compacted with time. This reduces both the porosity of the backfill and its permeability. The fluid pressure increases accordingly. The long-term hydraulic behaviour of the backfill is described by a phenomenological relationship between porosity and permeability which has been derived on measurements performed at backfill porosities higher than 10 per cent. The shaft is flooded on top of the shaft seal. This solution is connected to an unlimited reservoir. The shaft seals and the drift seals correspond to their specifications and they have a low permeability.

On the basis of a top-down approach possible altered evolution scenarios were identified which may result in a contact of significant amounts of water with the wastes. With this approach key FEP were identified that may seriously affect the system behaviour and that could result in a loss of function of one or several barriers. Also, prerequisites for certain FEP and their resulting FEP can be shown as well as FEP influencing other FEP (Figure 1). Such dependencies contribute to an improved system understanding.



▲ Fig. 1: FEP Failure of shaft seal

The following altered evolution scenarios were investigated:

- lower effectiveness of shaft seal
- lower effectiveness of drift seals
- inflow from brine pockets
- combined effects

No attempt was made to estimate the likelihood of occurrence of the various scenarios investigated. Within the safety assessment for a real site several of these scenarios need only to be considered as what-if scenarios or not at all due to their very low likelihood of occurrence.

3 Investigation of System Performance

3.1 Model System

The model system investigated comprises of a simplified repository structure which is based on the borehole disposal concept that was developed by the DBE. In this concept, all types of waste, i.e. vitrified high level waste, spent fuel, and technological waste are disposed of in boreholes in thin-walled canisters. The different waste canisters are called HAW canisters (CSD-V canisters), BSK-3 canisters and CSD-C canisters, respectively, and have the same diameter of 0.43 m but different length. The various waste types are disposed of in different emplacement fields. The model repository contains one emplacement field for each type of waste canisters. This means, that the total amount of BSK-3 canisters in the model repository corresponds to about one-eights of the amount that need to be disposed of in Germany based on the phase-out agreement (Moratorium). The emplacement fields for the HAW canisters and the CSD-C canisters are sufficiently large for all canisters of this type in Germany.

The system evolution has been modelled with the EMOS software [1]. EMOS is a modular code system comprising of modules modelling the processes in the repository's near field, in the overlying rock strata and in the biosphere. The repository concept is represented in simplified form in a segment structure retaining the relevant features of the repository. The model repository with its various model segments is shown in Figure 2. The shaft is connected to the infrastructure area. The two drift seals separate the infrastructure area from the two main access drifts.

The various segments represent areas of shafts, access drifts, emplacement drifts, and boreholes, respectively. No dedicated temperature field has been calculated for the model repository in this study. The temperature evolution profiles for the various model segments were taken from [2]. These calculations provide temperature values which are realistic enough for the purpose of this study. All relevant physical and chemical processes are taken into account for the model segments, e.g.

- the convergence process of the rock salt, the rate of which depends on the rock pressure and the counter-pressure of both the solid backfill and the fluid in the void volume,
- the reduction of the void volume in the backfill as a result of the convergence of rock salt
- permeation of solution through shaft seals and drift seals according to the seal's permeability
- in case of contact of waste canisters with solution
  - corrosion of the waste canisters with production of hydrogen gas,
  - canister failure,
  - radionuclide mobilisation, and
  - transport of radionuclides due to advective flow of the solution, diffusion and dispersion.

Should solution come into contact with waste canisters, the time evolution of the radiation exposure and the moment of the maximum irradiation exposure are calculated. The radiation exposure resulting from the radionuclide release is calculated as radiotoxicity index (RTI). For this purpose the EMOS farfield module CHETLIN with linear sorption [3] is used in combination with the EMOS biosphere module EXCON [4]. Linear sorption of radionuclides, dilution by groundwater and radioactive decay in the

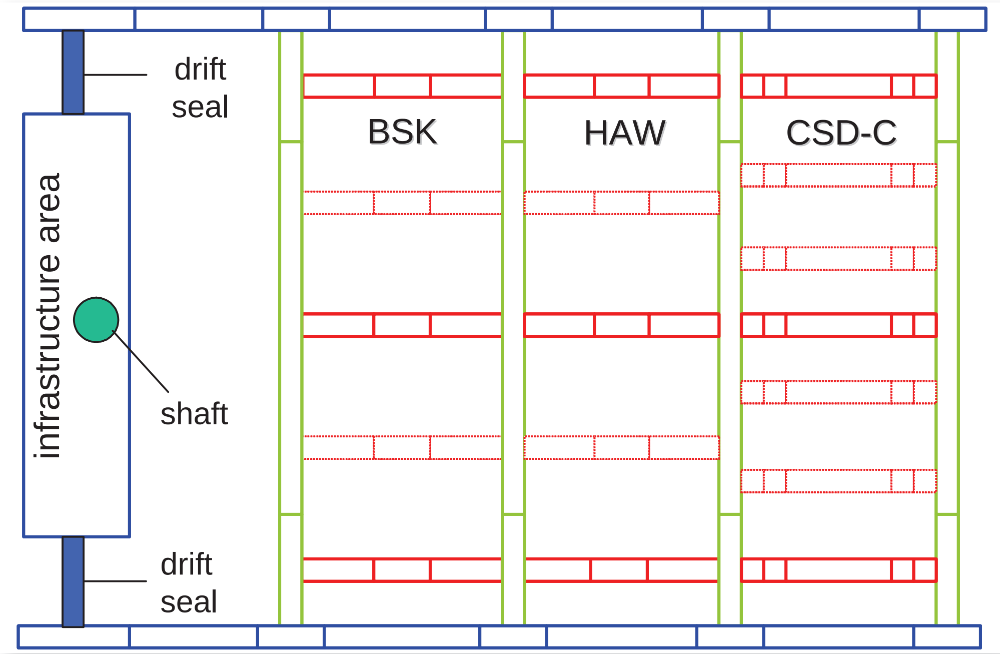


Fig. 2: Simplified model repository structure

overlying rock and in the biosphere are the relevant processes that are taken into account. In the present study the calculated radiation exposures are not considered to be safety indicators. They are rather used as indicators to compare different system evolutions with each other.

3.2 Normal Evolution of Repository System

Numerical calculations have been performed for the normal evolution scenario. For the shaft seal and the drift seals permeability values of 10^{-17} m^2 have been assumed. Under these conditions, which can be regarded as rather high permeability values for such geotechnical barriers, water permeates through the intact shaft seal. This process continues until the void volume in the infrastructure area is completely filled with solution. Depending on the pressure conditions in the infrastructure area and the main access drifts, solution will permeate also through the two drift seals.

The convergence of the rock salt leads to a compaction of the backfilled infrastructure area and access drifts, thus increasing the flow resistance for intruding solution. At the same time, the convergence provides for an advective transport of solution from the mine through the shaft seal. Release of solution from the mine commences after about 900 years. Owing to the final porosity of the backfill small amounts of solution remain in the mine area. Figure 3 shows those areas of the repository in light blue which are filled with solution after 1 million years. The numerical modelling results are given in Table 1. At any time, no solution reaches the emplaced waste in the normal evolution scenario.

3.3 Altered Evolution Scenarios

3.3.1 Lower Effectiveness of the Shaft Seal

The altered evolution scenario with a lower effectiveness of the shaft seal is based on the assumption, that the original shaft seal permeability of 10^{-17} m^2 will increase abruptly after 50 years by a factor of 10, 100, and 1000, respectively. The conditions of all other seals remain as in the normal evolution scenario. The numerical modelling results are summarised in Table 1.

Due to the higher shaft seal permeability solution will permeate more quickly through the shaft seal into the infrastructure area resulting in a faster flooding of this area. Therefore, release of solution from the mine will also commence earlier. The higher

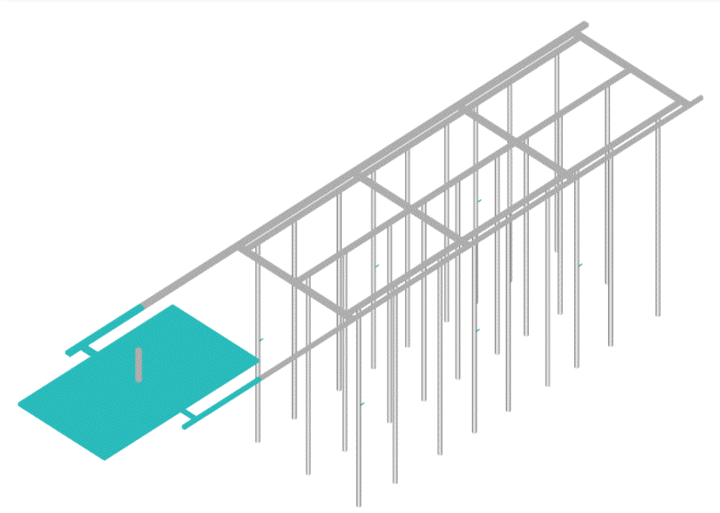


Fig. 3: Solution-filled areas in the mine after one million years in the normal evolution scenario

Table 1: Numerical results for some combined altered evolution scenarios

Scenario	Begin of release of solution from mine [a]	Amount of solution released [m³]	RTI of radionuclides released [Sv]
Normal evolution [K_{shaft} = 10^{-17} m²] [K_{drift} = 10^{-17} m²]	890	990	none
Shaft seal failure [K_{shaft}] = 10^{-16} m² [K_{shaft}] = 10^{-15} m² [K_{shaft}] = 10^{-14} m²	400 150 60	5000 13000 21000	none
Drift seal failure [K_{drift}] = 10^{-16} m² to 10^{-14} m²	860 - 850	990 - 980	none
Inflow from brine pocket six brine pockets	890	1140	$3{,}4\times10^{-6}$

the shaft seal permeability is the more solution will flow into the infrastructure area and the more solution will be pressed out via the convergence process. Some solution will also permeate through the intact drift seal into the access drifts. However, no solution will reach the emplaced waste canisters and no radionuclide mobilisation will occur under these conditions within one million years.

3.3.2 Lower effectiveness of the Drift Seals

In this altered evolution scenario the original permeability of the drift seal of 10^{-17} m² increases abruptly after 50 years by a factor of 10, 100, and 1000, respectively, with all other conditions being the same as in the normal evolution scenario. The modelling results are very similar to those of the normal evolution scenario under these conditions (see Table 1). The moment of release of brine from the mine is hardly affected and no radionuclide release occurs under the conditions investigated.

3.3.3 Inflow from Brine Pockets

This altered evolution scenario is distinguished from the normal evolution scenario by the presence of brine pockets in the vicinity of emplacement boreholes that remained undetected during the mining of the boreholes. This type of scenario is a typical what-if scenario, which may be investigated as part of the safety assessment. The presence of such macroscopic amounts of brine in the vicinity of boreholes is considered to be unlikely in the salt rock strata which are envisaged for the disposal of the waste.

The scenario is characterised by the following sequence of events:

- The system conditions correspond to the normal evolution with additional brine pockets of 100 m^3 being present in the vicinity of six emplacement boreholes, two each at boreholes with BSK-3 canisters, HAW canisters and CSD-C canisters, respectively,
- the brine pockets discharge after closure of the mine into the boreholes resulting in immediate contact between waste canisters and brine,
- the corrosion of the waste canisters occurs with a certain rate and leads to the formation of hydrogen gas, and
- after canister failure, radionuclide mobilisation from the waste matrix occurs with individual rates being dependent on the type of matrix.

This scenario is very different to the other altered evolution scenarios, in which solution permeates into the mine area from outside and needs to cover a long distance to reach the emplaced waste. In this scenario there is an immediate contact between waste canisters and solution which leads to a comparatively fast canister failure. In addition, the increasing fluid pressure decelerates the convergence of the flooded boreholes. Depending on the flow resistance of the borehole seal, the fluid pressures can exceed the corresponding hydrostatic pressure at the depths of the mine. The slower convergence allows an exchange of solution between boreholes and access drifts over a longer period of time.

In this scenario some radionuclides are released from the repository area (see Table 1). The maximum radiation exposure is very low and it occurs very late after one million years. At that time hardly any advective solution flow occurs from the repository area. The calculated irradiation exposures result from diffusion of radionuclides from the mine.

3.3.4 Combined Effects

Some altered evolution scenarios with combined effects have been investigated comprising the common failure of shaft seal and drift seals with a sudden permeability increase by a factor of three orders of magnitude, the failure of the shaft seal with inflow of solution from undetected brine pockets into boreholes, the failure of drift seals with discharge of solution from undetected brine pockets into boreholes, and the common failure of shaft seal and drift seals with discharge of solution from undetected brine pockets into boreholes. The numerical modelling results are summarised in Table 2.

All these scenarios lead to a release of radionuclides from the repository area. However, the likelihood of such scenarios is considered to be very low. Upon a common failure of all geotechnical barriers radiation exposures exceeding the regulatory limit of 0.3 mSv/a are calculated. Radiation exposures below the regulatory limit are calculated, when one of the seals remains intact. The safety relevance of the seals is evident also by comparing the calculated radiation exposures for the scenario inflow from brine pockets (see chapter 3.3.3 and Table 1) and for the combined scenario with additional failure of all seals. The latter scenario leads to radiation exposures which are much higher over the whole time span investigated (up to three orders of magnitude).

In case of scenarios without intact seals, the repository area is completely filled with solution within a couple of years. When the permeability of the shaft seal and the drift seals are very high already, the additional infiltration of solution from undetected brine pockets affects the results only with respect to the calculated radiation exposures.

Table 2: Numerical results for some combined altered evolution scenarios

Scenario	Begin of release of solution from mine [a]	Amount of solution released [m³]	RTI of radionuclides released [Sv]
Brine pockets + shaft seal failure $K_{shaft} = 10^{-14}$ m²	60	21700	8.6×10^{-6}
Common failure of drift and shaft seals $K_{shaft',drift} = 10^{-14}$ m²	84	28800	3.4×10^{-4}
As above + brine pockets $K_{shaft',drift} = 10^{-14}$ m²	82	29800	1.1×10^{-3}

3.4 Borehole Seals

Boreholes seals are made of crushed salt. Due to the compaction their porosity and flow resistance enhances with time, the rate of which depends on the temperature. The time evolution has been investigated using a simplified model. Two emplacement areas are connected to an access drift via seals made of crushed salt. The access drift is sealed against the infrastructure area with a drift seal. Another drift seal is located in the access drift between the two boreholes seals.

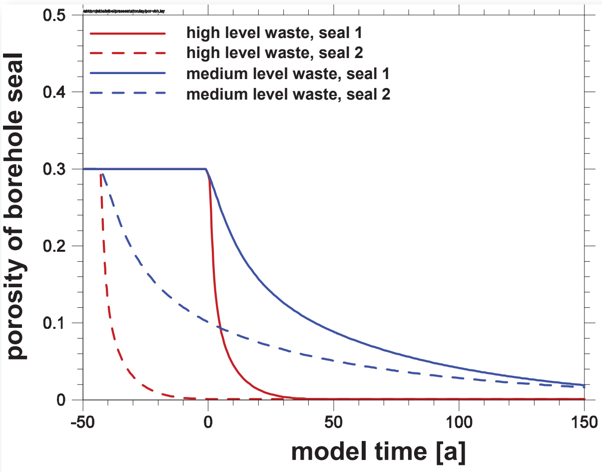
The following assumptions were employed in the model calculations

- intrusion of solution through the shaft occurs at earliest 50 a after emplacement
- length of shaft seal and drift seals: 50 m
- permeability of shaft seal and drift seal: 10^{-18} m²
- reference and initial convergence rate: 0,01 a^{-1}
- temperature in all mine areas is time dependent

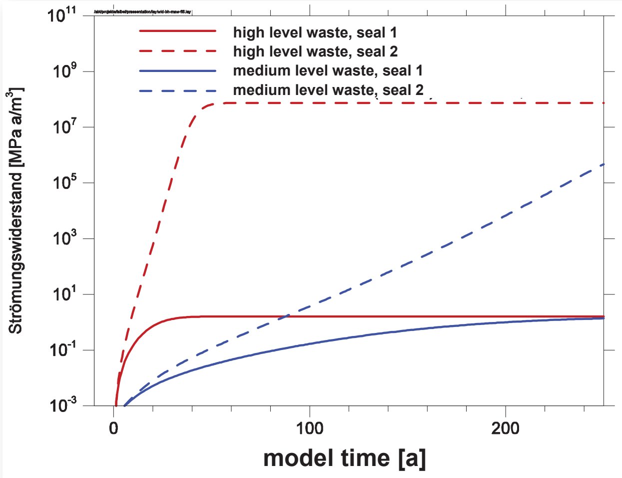
In order to investigate the influence of the temperature on the system behaviour, model calculations were carried out for high-level waste and for medium-level waste, respectively. The modelling results indicate that the seals of the borehole with high level waste are tight after 50 years (Figure 4). The maximum temperature in the seals is 370 K. In contrast, it takes about 500 years for the seal of the boreholes with medium level waste such as CSD-C canisters, to develop such a low porosity, that they are hydraulically tight. In this case, the maximum temperature of the borehole seal is 340 K.

The results for the time evolution of the flow resistance of such seals calculated with the simplified model are shown in Fig. 5. It can be seen that the final porosity and the corresponding permeability after complete compaction of the crushed salt are important parameters. In case of high level waste, the final porosity values are reached within some decades. This is due to the fast convergence of the surrounding rock salt caused by the high thermal output of this waste. In contrast, in case of medium level waste, it takes more than several centuries to reach the final porosity values.

The compaction of crushed salt has been investigated in many laboratory experiments and is has been observed in in-situ experiments. However, some issues are unresolved from the scientific point of view. All of these aspects concern the properties of highly compacted crushed salt, which is compacted slowly with moderate pressures. Such conditions are encountered in a repository. At present the following issues need to be addressed in order to improve the models that are employed in performance assessments to predict the behaviour of backfill and borehole seals over long times:



▲ Fig. 4: Time evolution of porosity of borehole seals for different waste types



▲ Fig. 5: Time evolution of flow resistance of seals made of crushed rock salt

- final porosity of the compacted crushed salt
- porosity-permeability relation at low porosity values, and
- porosity at zero permeability (existence of dead pores).

4 Conclusions

Engineered barrier systems are fundamental to achieve containment of the waste in a salt formation, predominantly via preventing or reducing water intrusion. Model calculations have been performed for the normal evolution scenario and some water intrusion scenarios in order to test the ability of the existing performance assessment tools. Realistic boundary conditions and parameters have been used where appropriate.

The investigated water intrusion scenarios were derived via a top-down approach, but no systematic scenario evolution was performed in this study. Also, combinations of scenarios have been investigated which may have such a low probability that they need not to be considered for a real site. A safety assessment of the results must not be done. The results allow, however, deriving valuable conclusions with respect to the behaviour of such a repository system and the relevance of certain events and processes.

When the shaft seal performs as intended, small amounts of water may intrude into the infrastructure area. The amount depends on the permeability of the intact shaft seal. Due to the compaction of the salt backfill the solution will be pressed out of the mine again through the shaft. A reduced effectiveness of either shaft seal or a drift seal does not impair the containment function of the repository system directly. No radionuclide release was calculated with the boundary conditions used in this study for these scenarios. The long-term compaction behaviour of crushed salt as backfill material affects the model results strongly. However, some uncertainties exist with respect to the duration until the compacted backfill material reaches porosity and permeability values that correspond to the initial values of undisturbed rock.

If, however, both seals do not function according to their specification solution may reach the disposal areas. Due to the convergence process contaminated brine may then be released through the shaft resulting in a potential exposure in the biosphere. The effect is enhanced considerably, if undetected brine pockets with a limited solution volume are present near-by and discharge into the void volume of emplacement boreholes. Thus, significant loss of containment occurs only as a result of a series of events, all of which have limited likelihood of occurrence. Therefore, the likelihood of such a scenario is very low. At any real site this needs to be confirmed by a systematic scenario development. ■

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2B.04 Safety Requirements on Heat-generating Radioactive Waste – Ongoing Development Process of Regulations –

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Summary
The "Safety Criteria for the Disposal of Radioactive Waste in a Mine" as of 1983 have to be revised with respect to the present state of science and technology and to latest international recommendations. The preparation of a draft proposal submitted by GRS in January 2007 was intensively supported by BfS. Main features are the isolation of heat-generating radioactive waste in the isolating rock zone, demonstration of safety for approx. 1 million years, stepwise approach to repository development and realisation as well as continuous safety-related optimisation process. These features were confirmed in a workshop on safety requirements organised by BfS in March 2007. In addition, the GRS draft proposal was examined by BfS on behalf of BMU. Thus, scientific-technical comments and remarks on the proposal as well as recommendations on these safety requirements to be addressed in a legal regulation on "Safety Requirements" envisaged by BMU were given by BfS.

1 Introduction

In the Federal Republic of Germany all types of solid and/or solidified radioactive waste are to be disposed of in deep geological formations. The legal framework is in particular given by the Atomgesetz (Atomic Act) and the Strahlenschutzverordnung (Radiation Protection Ordinance). The basic aspects which must be taken into account to achieve the objective of disposal are compiled in the Sicherheitskriterien für die Endlagerung radioaktiver Abfälle in einem Bergwerk (Safety Criteria for the Disposal of Radioactive Waste in a Mine) issued in 1983 [1]. The Safety Criteria qualitatively specify the measures to be taken in order to achieve the objective of disposal and define the principles by which it must be demonstrated that this objective has been reached. They have to be revised with respect to the present state-of-the-art aiming at the adjustment to international safety-related scientific-technical developments.

2 Revision of the Safety Criteria

On behalf of the Bundesministerium für Umwelt, Naturschutz und Reaktorsicherheit (BMU) the Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH was charged with the update of the 1983 Safety Criteria. According to the licensed Konrad repository for radioactive waste with negligible heat generation, the update should deal with the disposal of high level waste in deep geological formations. Furthermore, restrictions by existing legal regulations should not be taken into account. The update was completed with the report "Safety Requirements on the Disposal of High-radioactive Waste in Deep Geological Formations" [2]. The GRS draft proposal took both the further development of the state-of-the art in science and technology as well as international recommendations published most recently into account. Among these recommendations are especially the IAEA Safety Requirement WS-R-4 [3], the ICRP publication 81 [4] and then the draft report by the ICRP Committee 4 Task Group on Optimisation of Protection [5]. The preparation of the draft proposal submitted by GRS in January 2007 was intensively supported by BfS.

3 GRS Draft Proposal on Safety Requirements

The proposal is divided into protection objectives and safety principles and in requirements on the safety management, on the safety concept and on the proof of safety. Key factors are:

- Limiting the risk for individuals to contract a serious disease due to exposure is selected as protection objective for the phase following the sealing of a repository. Compliance with this protection objective is ensured by waste isolation/confinement in the isolating rock zone.

- The demanded safety management is to improve the safety of the repository in a continuous process and to promote the safety culture. A stepwise approach to demonstrate and to optimise safety under specified boundary conditions (constrained optimisation) is demanded. The embodiment of this step-by-step approach is recommended. For its execution four steps are proposed, i.e., site investigation from above-ground and from underground, planning and construction, operation and closure. In addition, a safety case is to be prepared including multiple lines of evidence to demonstrate safety.

- The safety concept should be developed as a staged system of multiple safety functions which should continuously be improved in the optimisation process. The objective of this process is to ensure a high level of safety for the repository through isolating the waste in a way as permanent, complete and reliable as possible. Reliance on passive safety has to be given rather than on active measures. The period of time for isolating the waste should be in the order of magnitude of 10^6 years.

- The requirements on the proof of safety are focussed on ensuring the isolation of the heat generating radioactive waste in the isolating rock zone of the repository system as well as the geotechnical barriers or sealing dams over this period of about 10^6 years. According to the stepwise approach, proofs have to be furnished at defined decision points and to be documented, respectively (safety case). All arguments and analyses to furnish the proof of safety of the repository must be documented.

4 Requirements on the Safety Concept

The applicant has to compile and realise a safety concept for radioactive waste isolation which ensures adherence to the protection objectives and the safety principles for all phases of the repository system development process. The safety concept has to ensure the complete and reliable isolation and confinement of the waste.

The development of the safety concept should be organised as an optimisation process during planning and construction, operation and the post-closure phase. The continuous improvement of safety is demanded as a step-wise optimisation process under given boundary conditions ("constrained optimisation" in accordance with [3] and [4]). It takes place in an essentially qualitative way under consideration of the protection objectives and especially the scientific-technical bases, planning and management principles. The boundary conditions (constraints) regarding safety in the post-closure phase result from the protection objectives and the requirements on the safety concept, respectively.

The constraints address four issues:

- **Safety concept:** The complete and reliable isolation and confinement of the waste has to be ensured by a repository system where the main emphasis with regard to the assignment of safety functions is placed on the geological barrier, i.e., on an isolating rock zone together with geotechnical sealing components.

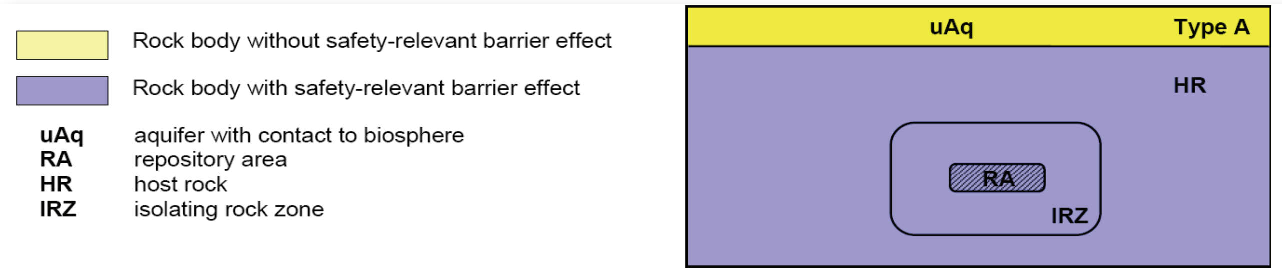
- **Duration of confinement:** The confinement of the waste has to be ensured over at least a period of time in the order of magnitude of 10^6 years.

- **Completeness of confinement:** The confinement of the waste is considered to be complete if, for likely scenarios, only negligible amounts of contaminants will be released from the isolating rock zone, thus causing no harm to man and the environment.

- **Reliability of confinement:** The likelihood of scenarios leading to higher releases than the ones mentioned above -but also to any low-level releases- should be significantly smaller than 1.

The isolating rock zone is defined as part of the geological barrier which - at normal development of the repository and together with the technical and geotechnical barriers - has to ensure the confinement of the waste packages for the envisaged isolation period (Figure 1). According to the Arbeitskreis Auswahlverfahren Endlagerstandorte - AkEnd [6] it is possible to identify sites

- the normal evaluation of which can be predicted over at least 10^6 years,
- where an isolating rock zone can be found which will maintain its crucial properties (e.g., extent, thickness, conductivity) over this time span.

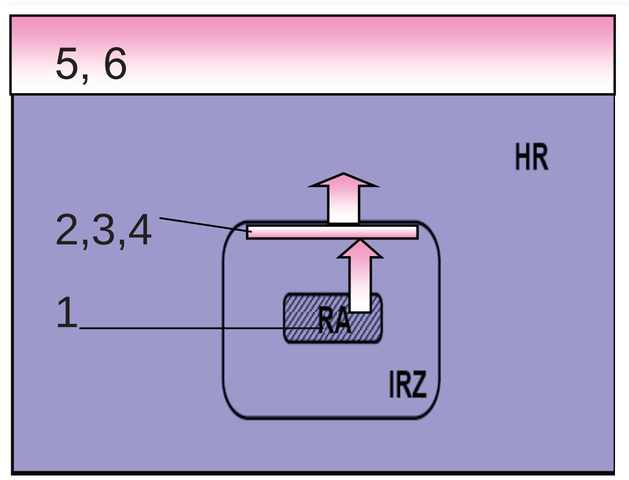


▲ Fig. 1: Scheme of the isolating rock zone within the repository system

Due to the introduction of the isolating rock zone into the safety concept two implications are to be dealt with, i.e. emphasis is given on the geologic barrier and challenges are to be taken up concerning the sealing components.

5 Requirements on the Safety Demonstration

The Safety Requirements contain the requirements on the demonstration or proof of safety for all phases of the repository system development process. The requirement is to prepare site-specific safety cases at well-defined decision points of this process both for the operational phase as well as for the post-closure phase of a repository. The safety case must be comprehensive and transparent. It must contain multiple lines of arguments and analyses for safety justifications of the repository system as well as regarding confidence in the proof furnished. The Safety Requirements put the requirements on the safety case for all phases of the repository development process in concrete terms. The emphasis of the verification is on the proof of the isolation



◄ Fig. 2: Scheme of the GRS proposal to proof isolation and protection of heat-generating radioactive waste in an isolating rock zone in combination with the geotechnical seals. GRS thesis is that by the proof of the isolation, the proof of adherence to the protection objectives is inherently possible. The protection objectives apply for an unlimited period. Verification, however, should be orientated on both the limits of practical reasoning and the possibilities of the optimisation process.

The evaluation of the completeness of the isolation can be done using indicators which are determined by modelling those components of the repository system whose evolution can be prognosticated for the evaluation period. These are in particular the components of the subsystems “isolating rock zone” and “seals of the repository”.

With respect to the proof of safety GRS proposes four indicators to demonstrate isolation and protection of the life base goods environment and groundwater as well as two indicators to demonstrate protection of humans and the environment (Figure 2).

Thus, GRS includes the following indicators in its proposal [2]:

- Retention of the amount of radionuclides in the isolating rock zone (cumulated over the assessment time of 1 million years), e.g. more than 99,99 %.
- Additional concentration of released U and Th in the pore water at the boundary area of the isolating rock zone, e.g. less than 1 µg/l U and 0.1 µg/l Th.
- Contribution to power density in the pore water at the boundary area of the isolating rock zone less than 10 % of natural conditions, e.g. less than 1 MeV/l.
- Index of radiotoxicity of the released radionuclides and of the natural pore water content at the boundary of the isolating rock zone, e.g. less than 1.
- Additional concentration of natural radionuclides low compared to initial concentrations.
- Effective dose to man less than 0.1 mSv/a.

6 BfS Workshop on Safety Requirements

On the occasion of the workshop “Safety Requirements on the Disposal of High-Level Radioactive Waste” which took place at Hannover on March 06 and 07, 2007 the GRS draft proposal was presented and scientifically discussed. On behalf of BMU this workshop was organised by BfS [7]. It served to get to know the ideas of experts regarding disposal of high-level radioactive waste in deep geological formations and to obtain further suggestions. Participants contributes to the evaluation of the GRS draft proposal with respect to the present state-of the-art, to its applicability and to the supplementation, extension and level of detail of single requirements. In two working groups the topics “Protection Objectives/Safety Demonstration/Isolation” and “Stepwise Approach/Optimisation” were discussed in depth. Essential basic elements of the GRS draft proposal such as

- isolation of the high-level radioactive waste in the isolating rock zone,
- safety demonstration for periods of time in the order of magnitude of 10^6 years,
- stepwise approach with well-defined decision points and criteria,
- application of the optimisation process during repository planning, design, construction, operation and closure

were confirmed by the participants. Apart from this, the workshop offered a multitude of further suggestions and references [6].

7 BfS Evaluation and Recommendations

In addition to the workshop, as requested by BMU, BfS examined and evaluated the Safety Requirements in order to develop a statement on the GRS draft. This statement is divided into further extending, partially specifying technical remarks on the GRS proposals and into recommendations for individual safety requirements which should be taken into account in the provisions for a legal regulation on Safety Requirements as envisaged by BMU.

The scientific-technical comments of BfS comprise, inter alia, the following remarks:

- Isolation of waste packages within the repository system (repository and isolating rock zone) is supported.
- Spatial and temporal definition of the isolating rock zone is to be specified more clearly.
- It should be pointed out that isolation describes the efficiency of the repository system, i. e., isolation can not ensure the observance of the protection objectives per se.

Radioactive waste with negligible heat generation (i. e., LLW and ILW) will be disposed of the Konrad repository which is under construction since May 2007. Thus, the scope of the Safety Requirements is to be restricted to heat-generating radioactive waste. The requirements should be applied to repository sites to be selected within a site selection procedure as envisaged by BMU as far as applicable (e .g., comparison of sites) and to a finally selected repository site.

BfS develops its recommendations on the basis on the protection objectives “Long-term protection of man and the environment from harmful effects of ionizing radiation” and “Avoidance of unacceptable burdens and obligations to future generations”. Both objectives are temporally not limited.

The IAEA Safety Principles [8] and the basic radiation protection objectives to avoid radiation exposure and to reduce doses are to be adopted. Thus, with respect to the operational phase of a repository for heat-generating radioactive waste, BfS suggests the following recommendations:

- Avoidance of unnecessary radiation exposure and contamination to man and the environment.
- Radiation exposure and contamination are to be kept as low as possible considering the state of science and technology. A cut-off criterion is not to be introduced.
- Application of existing rules and regulations for nuclear installations in an analogous way.
- Introduction of a four phase safety concept (normal operation, anomalous operation, assumed incidents and events beyond the design basis).
- Performance of site and facility-specific safety assessments.

With respect to the post-closure phase of a repository for heat-generating radioactive waste, BfS suggests the following recommendations:

- Optimisation of repository safety.
- Application of the stepwise approach with respect to planning, construction, operation and closure of a repository.
- Demonstration of safety to be performed for a period of time in the order of magnitude of 10^6 years.
- Consideration of the time period beyond 10^6 years.
- Performance of a risk-based approach.

- Development of guidelines.

The essential element of future obligatory safety requirements is the optimisation of the repository system's safety in the operational phase and in the post-closure phase with respect to the protection objectives and the associated stepwise approach. Thus, a continuous iterative optimisation process is to be performed including the appointment of quantitative and qualitative criteria (e. g., dose, risk, technical feasibility, cost, societal aspects or mining safety). It should be pointed out that optimisation shall be performed in a wide-ranging process and must not result in a minimal dose. The application of the stepwise approach is the prerequisite for the constrained optimisation and demonstration of safety. In accordance with further international recommendations [9], a safety case at well-defined decision points shall be carried out.

BfS considers compliance with the radiological protection objective to be warranted if the risk of an individual to suffer severe damage to its health is limited to 10^{-4} per lifetime. Thus, potential damage due to other – non-radioactive – harmful substances is considered, too.

The safety concept for the post-closure phase relies on the repository safety being guaranteed through a staged and robust system of several safety functions acting passively and free of maintenance. Proof a safety must be furnished for a period of time in the order of magnitude of 10^6 years. For this period the geological development of the isolating rock zone can be predicted [6]. It is suggested to established a site-specific proof of safety for the operational phase of a repository, comparable with any other nuclear facility, and a proof of safety for the post-closure phase. In both cases calculations have to be as close as to reality as possible. The concrete procedure of establishing the proof of long-term safety should be laid down in a guideline. This guideline supports the safety requirements in a more detailed manner and can be continuously adopted to the evolving state-of-the art of science and technology.

8 Way Forward

Based upon the GRS proposal [2], BfS workshop's results [7] and BfS recommendations as of June 2007 as well as the evaluation of the GRS proposal prepared by BMU's advisory bodies Reaktor-Sicherheitskommission and Strahlenschutzkommission BMU carries out the final elaboration of the Safety Requirements. It is intended to present the Safety Requirements in 2008. ■

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2B.05 Radiation Protection Criteria for Long-term Safety Assessments of Nuclear Waste Repositories: Current Issues and Potential Ways Forward

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Abstract
Any long-term safety assessment for nuclear waste disposal in deep repositories usually involves the estimation of potential impact to human health which then has to be evaluated with respect to radiation protection criteria. For this purpose, a wide range of conceptually different standards and regulatory guidelines has been proposed including inter alia constraints or limits of individual doses, constraints of the individual probability of radiation-related death and comparison with concentrations of radionuclides occurring naturally in environmental media. Advantages and limitations of the various approaches are discussed and a general health protection criterion for the disposal of nuclear waste is proposed. It is demonstrated how the fundamental radioprotection principles of dose limitation and optimisation can be easily included in the protection criterion suggested. Modelling the potential impact on human health inevitably involves assumptions on transfer paths and rates of the radionuclides in the environment and on human habits which become increasingly uncertain as predictions extend into the future. It is shown that the recently developed Reference Biosphere Methodology could assist in dealing with this methodological problem by establishing a number of stylised biospheres representing differing climate conditions and focusing on those radionuclide transfer pathways and fundamental human habits which determine impacts on human health. A basic set of such reference biospheres could be developed and applied in an international consensus. Additional scenarios could be added nationally, e.g. if a society decides that some traditional specific habits should be included in these projections.

1 Introduction

According to the Fundamental Safety Principles and more specifically in the principles of radioactive waste management of the IAEA, the disposal of long-lived radioactive waste has to follow specific safety and radiation protection principles [1, 2]. In addition to the principles regarding management safety, three ethical principles related to human and environmental health as well as to sustainable development define the societal and political framework for the development of standards and regulatory guidelines for the disposal of radioactive waste in deep geological repositories. These are (i) radioactive waste should be managed in a way to secure an acceptable level of protection for human health as well as of the environment, (ii) possible effects in the future should not be greater than relevant levels of impact for health and environment that are acceptable today, and (iii) the way of waste disposal should not impose undue burdens on future generations.

The key term of the first two principles is the level of radiation protection that is acceptable today. Any attempt to specify these general principles more precisely, therefore, should discuss what an acceptable level of radiation protection is or might be. The first part of our contribution will be devoted to this subject by evaluating (i) standards for the protection of humans against radioactive and other toxicants, and (ii) actual radiation exposures from nuclear facilities operating today. In a second part, we then discuss potential and limitations of biosphere modelling approaches for demonstrating compliance of the long-term performance of a geologic repository with the chosen protection standards.

2 Acceptable Level of Protection for Human Health

2.1 Radiation Protection Standards

There is international consensus that an adequate level of protection of any individual in planned exposure situations should be the result of the basic principles of application of setting dose limits and constraints and of optimising below those constraints [3]. The principle of optimisation of protection requires that all reasonable means should be applied to keep the likelihood of incurring exposure, the number of people exposed and their individual doses as low as reasonably achievable, taking into account economic and social factors.

Thus, the range of exposures that currently result from the application of these well-established radiation protection principles should indicate a widely accepted level of protection for human health.

In Europe emissions and radiation exposures from operating nuclear facilities are annually documented nationally and supra-nationally, e.g. [4-6]. With the exception of nuclear fuel reprocessing plants in France and UK, for which doses in the range of some 10 µSv per year are reported, values given for the exposures of the person or group receiving the highest individual exposures generally are below 10 µSv per year. It should be noted that releases usually are too low to result in build-up of detectable radionuclide concentrations in environmental media and food and that the doses therefore have to be estimated, sometimes involving conservative modelling approaches [6].

The influence of the radiation protection principle of optimisation becomes obvious by comparing these exposures with dose limits for members of the public, which in most European countries are set to 1 mSv per year [7], and with dose constraints, which e.g. in Germany have been set to 0.3 mSv y^{-1} for exposures caused by emissions of nuclear facilities with air and water, respectively [8]. This comparison also demonstrates that a currently accepted level of radiation protection is not defined solely by dose limits and constraints stated in regulations, but in accordance with the current radiation protection philosophy is reflected in the exposure levels reached after optimisation.

It should be noted that the standards referenced above are for planned exposures, i.e. for facilities and practises under full control of the operator and the regulator who can actively influence operation and therefore emissions. The post closure phase of a nuclear waste repository more closely resembles that after clearance, in which the regulator deliberately gives up control of the source. However, as today's exposures from planned exposures of the public of nuclear facilities are comparable to the internationally accepted exemption level of individual doses of the order of 10 µSv per year [9], further discussion of the exposure situation characterising the post closure phase of a nuclear repository seems not to be needed.

2.2 Groundwater Protection Standards

A key principle of sustainable development is the requirement to protect all life supporting environmental media such as water, soil and air to the extent that self renewing processes are able to preserve their life supporting capabilities. In the potential case of a release of radionuclides from a deep geological nuclear waste repository, the predominant path of exposure is expected to be via drinking water for humans and farm animals and via irrigation water for agricultural and farm land. The use of groundwater to irrigate agricultural and farm land may lead to contamination of the soil. As released radionuclides in general will be long-lived, the build-up of a prolonged component of the exposure as a result of the accumulation of radioactive residues in water, soil and the food chain has to be considered.

As contamination of groundwater constitutes the primary pathway of any potential contamination of the biosphere, existing groundwater and drinking water standards also give information valuable for quantifying the levels of impact for health and environment that are regarded as acceptable today.

The WHO drinking water quality guidelines [10] are accepted internationally and adopted in many standards (e.g. [11]). For various classes of cancerogenic substances permissible concentrations have been derived assuming a fixed water consumption rate and a life-long exposure and considering cancer incidence as the relevant health effect. Thus, the WHO was confronted with the issue of acceptability and comparability of standards for radiological and non-radiological toxicological risks in one system including chemical, biological and physical toxicants from artificial and natural sources, respectively. The approach taken by WHO was to establish a reference level of risk to allow a consistent approach to deal with different health hazards and, most importantly, to achieve a broad equivalence between the levels of protection afforded to toxic chemicals, to radionuclides and to microbial pathogens. This risk based reference level was set to 10^{-6} disability-adjusted life-years per person per year, which is approximately equivalent to a lifetime excess cancer risk of 10^{-5} (i.e., 1 excess case of cancer per 100 000 members of the population ingesting drinking water containing the substance at the guideline value over a life span).

Additionally WHO uses a risk band of 10^{-6} to 10^{-4}, of which the upper limit is considered as adequate for naturally occurring toxicants such as arsenic, the lower limit for some artificial chemical carcinogens. Considering that today's drinking water contamination by radionuclides is almost always of natural origin, the WHO used the 10^{-4} lifetime risk value also for radioactive substances together with an approximately equivalent dose reference level of 0.1 mSv y^{-1}.

Although the assumption regarding radioactive substances obviously is questionable for (artificial) radionuclides released from a final repository, for our discussion we would like to stress that for cancerogenic substances in drinking water a widely accepted level of impact is given as an additional lifetime risk of cancer incidence between $<10^{-5}$ and 10^{-4}. As groundwater often is considered as potential drinking water resource and protected accordingly (e.g. [12]), this range of acceptable risk levels can be generalised.

In case of a release from a deep geological nuclear waste repository, some of the radioisotopes and chemical toxic elements released are also present naturally. For these another principle of groundwater protection could be attended, i.e. that a deliberate contamination of groundwater should not chance significantly the composition of the water compared to its natural composition from its geological layer or region of origin. Therefore in addition to radiation exposure standards, concentration standards (constraints) could be established taking into account the natural composition of ground water in the region of the geological repository site.

2.3 Consequences for a Primary Standard for Geological Nuclear Waste Disposal

A radiation protection standard for a nuclear waste repository is the societally acceptable limit on the repository performance which should not be exceeded at any time if the repository is to be judged safe. There is consensus that it should focus on health and well-being of humans, on the integrity of the basis of life to allow a sustainable development for future generations, and on the protection of ecosystems and biota.

In radiation protection, the long term development of a geological nuclear waste repository is classified as a potential exposure situation, i.e. exposures may or may not occur. A key element in the assessment of potential exposure situations is the evaluation of probabilities at all levels of the assessment and for all features, events and processes considered, although it is recognised that uncertainties will increase for long time scales. Generally, a probability of occurrence may be assigned to all expected and other possible developments of a deep geological nuclear waste repository. Therefore, potential exposure situations resulting from different developments of the waste repository involve consideration of risk which fall outside the general boundaries considered for normal exposures in radiation protection. Recommendations or standards of international bodies or institutions such as ICRP or IAEA, respectively, give no clear guidance about the most convincing way to develop radiation protection standards for potential exposure situations. This fact may have contributed to the – compared to other radiation protection issues – large variation in national standards and approaches apparent from Table 1.

In the case of potential exposures, health risk is the result of two random events. First, the occurrence of the event that causes the exposure, and second, the appearance of a harmful health effect, given the exposure happens. Reference levels for this conditional probability should be derived from the established concepts of individual detriment resulting from planned exposures.

Because of the probabilistic nature of the potential exposure situation it is straightforward to set primary standards in terms of risks, i.e. as risk of attributable health effects for human individuals and as risk of attributable environmental effects, respectively. From these, dose values can be derived as secondary standards using nominal risk coefficients [3].

A convenient starting point for deriving reference levels for human health impact is given by the approach used by WHO in its drinking water quality guidelines [10], as it can be regarded as an internationally accepted risk standard for the exposure pathway most relevant for long-term assessments of nuclear waste repositories. As outlined above, these give an upper reference level of a lifetime risk of cancer incidence of 10^{-4} with a range down to 10^{-6} centred about 10^{-5}.

These values are close to risk figures calculated from the current range of planned exposures resulting from the operation of nuclear facilities summarised above. Annual doses of some 10 µSv as reported for the European fuel reprocessing plants correspond

Table 1: National radiation protection criteria for long-term assessments of geological nuclear waste repositories

Country	Dose Constraint	Risk Constraint
Switzerland [13,14]	If a limited number is exposed, 0.1 mSv y^{-1}; if large groups will be exposed, lower dose constraints should be considered.	10^{-6} y^{-1} in addition to the dose constraint
Belgium [15]	Not yet fixed. With reference to the international discussion the range of 0.1 – 0.3 mSv y^{-1} is mentioned	Not yet fixed. References are to ICRP-46 (recommending a risk constraint below 10^{-5} y^{-1}, to the Netherlands (risk constraint of 10^{-6} y^{-1}) and the UK (risk target of 10^{-6} y^{-1}).
France [16]	0.25 mSv y^{-1} as the main safety indicator	10^{-5} per lifetime for non-radiological carcinogens
United Kingdom [17,18]	0.3 mSv y^{-1} (single source) and 0.5 mSv y^{-1} (complete site); applicable during operational phase of repository only	< 10^{-6} y^{-1} as risk target ('neither a limit nor the boundary between acceptable and unacceptable levels of risk') after closure
Sweden [19,20]	Not given	10^{-6} y^{-1} generally, 10^{-5} y^{-1} if a small group of people is exposed
Finland [21,22]	0.1 mSv y^{-1}, only applicable to expected developments	Only applicable to unexpected or low probability developments: the product of dose and probability of occurrence, respectively, should not exceed the dose constraint; for deterministic radiation effects (doses >0.5 Sv) the probability of occurrence should not exceed 10^{-6} y^{-1}.
Canada [23]	Below dose limit of 1 mSv y^{-1}, reference is made to ICRP recommendation of a design target of 0.3 mSv y^{-1}	Equivalent to dose criterion (using a risk coefficient of 0.073 Sv^{-1}), should be preferred in case of probabilistic safety assessments
USA [24,25]	0.15 mSv y^{-1} for 10^4 y after closure, 3.5 mSv y^{-1} for 10^4-10^6 y after closure	10^{-6} to 10^{-5} y^{-1} recommended by National Academy of Sciences, not adopted by the regulator.

to lifetime cancer incidence risks of about 10^{-4} (assuming this exposure is constant with time), whereas doses around or below 10 µSv per year typical for other today's nuclear installations are equivalent to lifetime risks of about 10^{-5} or even lower.

These data demonstrate that a lifetime risk for cancer incidence in the range of 10^{-6} to 10^{-4} can be considered as an internationally accepted level of human health protection both for radioactive and other cancerogenic substances. We therefore suggest adopting the upper limit of 10^{-4} as a risk constraint for long-term assessments of geologic nuclear waste repositories. For individual projects it then can be expected that, due to the application of the optimisation principle, estimated risks will be considerably below the risk constraint.

There are a number of advantages to prefer a risk over a dose constraint as primary standard. First, conceptually it fits easily into the potential exposure concept. Second, it provides a safety measure to constrain potential health effects from both radioactive and other cancerogenic toxic substances disposed of in a nuclear waste repository. Third, dose limits recommended by ICRP have been modified several times within the last decades , as they are based on risk coefficients derived from epidemiological studies which are modified in a complex judgemental operation by ICRP experts, taking into account radiobiological and radiophysical data and models to derive nominal risk coefficients for different radiation exposure situations (acute vs. chronic, total body vs. partial body irradiation, different radiation sensitivities of various organs and tissues, incidence vs. mortality etc.) and thus are subject to changes with scientific progress. Dose constraints, however, could be derived as a secondary standard, taking into account the conditional probabilities of exposures for the various future repository developments considered, and used together with the primary risk standard both for planning purposes and for communication with the public.

It should be noted that the approach proposed here encompasses much of the range of constraints present in various national standards as summarised in Table 1. Our suggestion to use both risk constraints and derived dose constraints also seems to be in agreement with current approaches in various countries (e.g. Canada, Finland and Switzerland).

3 Assessment Methodologies: The Biosphere

Presently the evaluation of consequences after a potential entry of radioactive substances into the biosphere is integral part of any long term consequence assessment. Although there is interest in the use of various safety indicators [26,27], potential exposures to future human individuals (in terms of dose or health risk) are and can be expected to remain the most important safety indicator both for licensing procedures and for communication with the public. The biosphere, therefore, is integral part of the overall disposal system to be assessed.

However, simulations of radionuclide dynamics in the biosphere and exposure situations to humans in the context of long-term consequence assessments of nuclear waste repositories may be subject to considerable uncertainties. These may primarily arise from (i) potential changes of human habits both individually and of future societies and (ii) from changes of environmental conditions in future times (especially due to climate variations, including both warming and glaciation periods). Additionally, the current debate on the human impact on climate change illustrates that future developments in human habits and environmental conditions can not be considered independently from each other. Due to these uncertainties, there is general agreement that any numerical result estimated for the endpoint of a long-term assessment (usually dose or risk to an exposed hypothetical individual) should be interpreted as a safety indicator [28]. A second, not as transparent result of this perception of uncertainties of future biosphere processes is that in long-term consequences the focus has shifted to simulating radionuclide dynamics and its variability in the geosphere rather than to biosphere processes. In practice this implies that a detailed probabilistic modelling of radionuclide transport in the geosphere taking into account potential future developments may be coupled to a simple deterministic biosphere transport and exposure model based on present day situations. Obviously, such an approach not only poses conceptual difficulties in interpreting the results in a probabilistic context, but also fails to consider the biosphere and its processes adequately as integral part of the overall depository system.

If any assessment has to provide an at least reasonable level of assurance that potential long term consequences will be limited in terms of today accepted levels of dose or risk to humans, flexible but robust assessment methodologies including geosphere and biosphere processes are inevitable. For this purpose, a procedure – often called Reference Biosphere Methodology – has been developed to create stylised biosphere models which are based on general biosphere conditions and human habits [29] and which will be discussed in more detail in the following.

Options of potential future developments exist for which both the probability and nature of their realisation can not be predicted to any reasonable extent (e.g. medical progress in cancer treatment, technical achievements). It may be argued that this generally precludes detailed biosphere modelling for radiation exposure assessments. It should be noted that the category of unpredictable scenarios is not unique to the biosphere, but includes geosphere scenarios also (e.g. deliberate human intrusion). Generally, the existence of such scenarios should not be regarded as an obstacle to evaluate the range of potential consequences of the many predictable scenarios of geosphere and biosphere developments; their results will not only contribute significantly to the safety case, but are also likely to cover the range of potential consequences resulting from at least some of the not seriously predictable scenarios.

The approach to deal with the biosphere as an integral part of the overall disposal system is further supported by the observation that changes of biosphere processes may also influence geosphere processes including the long-term performance of geological barriers. As an example, groundwater recharge and thus radionuclide dilution during geosphere transport may depend on climate driven variations in precipitation and infiltration.

3.1 The Reference Biosphere Methodology

The Reference Biosphere Methodology, which has been developed in the BIOMOVS II project [31,32] and extended and tested by BIOMASS [30,33], provides a tool for the systematic development and justification of assessment biosphere systems within the context of long-term consequence assessments for radioactive waste disposal. Recently, this methodology has been even more refined with emphasis on its capabilities to simulate impacts of climate variations on biosphere processes [34,35] and has been successfully applied to develop biosphere models for five different European areas with contrasting climates [36,35]. As the potential of the Reference Biosphere Methodology seems not been fully exploited yet for long-term consequence assessments of radioactive waste disposal, its major characteristics and potential benefits for assessment credibility are highlighted in the following.

3.1.1 A Consistent Approach

Within the Reference Biosphere Methodology the biosphere system is defined [30] 'as a set of specific characteristics that describe the biotic and abiotic components of the surface environment and their relationships that are relevant to safety assessments of radioactive waste disposals. … Components of the biosphere system are: human activities; climate; topography; location; geographical extent; near-surface lithostratigraphy; and water bodies.' For these major components, a biosphere FEP (features, events and processes) list has been developed [32] from which interaction matrices can be derived for reference biospheres of interest. An example is given in Figure 1. As this procedure closely followed the approach taken for establishing the geosphere FEP list [37], the Reference Biosphere Methodology offers a consistent approach to long-term assessments of repository systems including both the geo- and the biosphere.

3.1.2 A Framework for Considering Climatic Variations

Eventually the most appealing feature of the Reference Biosphere Methodology is its potential to simulate the impact of different climates on ecosystems and to estimate their consequences on the dynamics of radionuclides and the doses to humans and biota. Two approaches are available within the Reference Biosphere Methodology, non-sequential and sequential climate modelling [38,39,34].

The non-sequential approach assumes that ecosystems are in quasi-equilibrium in response to the climate conditions assumed. By comparing the radiological consequences (as individual dose or risk) per unit release into the various ecosystems established for differing climates, the potential future conditions resulting in the highest consequences to man (or biota) can be identified. Modelling is based on the reasonable hypothesis that the major biotic and biotic components and potential human habits including agriculture in future climates correspond to the conditions present today in respective climatic zones. A number of studies successfully applied the non-sequential approach to develop example reference biospheres for various climate conditions (e.g. [35,40]). As a result of general interest, these applications demonstrate that climate driven variations in human exposures can be expected to remain below an order of magnitude, as adoption of ecosystems and human habits (including agricultural production systems) to different climatic conditions only moderately influences radionuclide behaviours.

In the sequential approach sequences of successive climates are simulated including their transition phases. Such sequences may be established from the paleoclimate information available using a Markov model. This more complex approach may be needed to explore the impact of specific transition phase processes on radiation exposures, e.g. a potential pulse input into an ecosystem of radionuclides which have accumulated below an ice sheet during a glaciation period. It should be noted that it may not be necessary to model climate variations over the complete time frame of a long-term assessment, but to limit it to the time needed until a complete climatic cycle has been simulated [38]. Whereas the number of simulations applying the sequential approach is limited [38,39], the methodological development has advanced considerably [34].

3.1.3 A Toolbox for Scenario Development

The Reference Biosphere Methodology enables the construction of stylised biospheres of differing complexity. Thus, it offers a convenient approach to study systematically the effect of increasing ecosystem complexity [41] or of specific processes which are supposed to be important, e.g. climatic variations. Similarly, the methodology can be used to establish a set of generic reference biospheres to which for long-term assessments at a specific site locally important processes can be added if necessary.

Climate/	Weathering	Meteoric erosion	Level changes (e.g. evaporation, storm events) Deposition Freezing	Deposition Erosion Conditioning and moisture content (e.g. evaporation, freezing)	Environmental conditioning	Defines natural climax ecosystem, Affects transpiration rate. Storm damage
Contribution to dust load from eroded surface material, radon and other gas release	**Geology**	Definition of relief	Defines groundwater flow system and chemistry	Contribution to mineral composition (by weathering), gas release to soil/sediment	Availability of mineral resources. Type of geology affects construction practices	Nutrient support for microbial populations
Aspect effect on wind and insolation. Altitude effect on wind speed. Relief effect on boundary layer structure	No effect	**Topography**	Defines geometry (e.g. catchment areas, coastline) and hydrology (including mixing of meteoric and ground water)	Defines geographical extent relative to water bodies. Slope affects erosion rate and drainage characteristics	Relief affects size and type of community (e.g. capacity to develop agricultural systems and other land uses)	Relief affects characteristics of natural systems
Large water bodies influence seasonal temperature and humidity variation, local source/sink for heat and moisture	Erosion and dissolution (e.g. to form karst systems, geochemical reactions, freeze/thaw)	Erosion and dissoluton (e.g. downcutting of river bed, meandering, coastal processes)	**Water Bodies**	Sedimentation, erosion, mixing and suspension. Chemical conditioning by exchange with porewater.	Source, quality and quantity affects sustainability of community, recreational activities. Contribution to trace elements in diet	Source, quality and quantity affects type and behaviour of natural communities
Soil gas exchange with atmosphere	Diagenesis	Soil and sediment deposition influence the geometry (e.g. lake infilling, meandering and delta formation)	Contribution to sediment load. Chemical conditioning of meteoric water. Effect on flow field. Permeability of horizons contributes to geometry	**Soil / Sediments**	Soil moisture characteristics, structure and mineralogy affect potential utilisation. Source of fuel (peat)	Soil moisture characteristics, structure and mineralogy affect development of natural communities
Pollution, Creation of microclimates (buildings etc)	Quarrying and mining activities	Ground levelling	Hydrochemical conditioning (e.g. water treatment, waste water disposal). Extraction. Dam building. Augmentation	Agricultural practices, Dredging. Drainage systems	**Human Community**	Agriculture. Genetic manipulation. Population / ecosystem modification
Atmospheric composition (e.g. pollen and gas release). Microclimate development (e.g. wind break, boundary layer)	Biogeochemical conditioning by microbes	Effects assumed to be insignificant	Conditioning of flow, particulate load and hydrochemistry	Bioturbation. Conditioning, stabilisation & development. Control on moisture content (transpiration).	Natural resources. Disease vector	**Biota**

▲ Fig. 1: Example of an Interaction Matrix to describe the dynamics of a generic biosphere using the Reference Biosphere Methodology; taken from [34]

3.1.4 A Basis for International Cooperation

The Reference Biosphere Methodology has been developed and mainly applied in close international cooperation. As a consequence, a common understanding of biosphere modelling has evolved in many institutions involved in long-term assessments of nuclear waste disposal. In our opinion, it could be attractive to use this experience to explore the potential importance of transition phases between climate states in close international cooperation. Moreover, a set of generic reference biospheres could be developed which represent an international consensus to represent the major potential ecosystem evolutions and, with regard to potential consequences, the most important processes. Such a set of internationally accepted generic reference biospheres could not only be used to compare different disposal options, but could also be applied nationally as starting point for site-specific consequence assessments. Furthermore, such an international consensus could also assist nationally in public acceptance of the long-term assessments.

3.2 Deterministic versus Stochastic

Many long-term assessments of potential consequences of nuclear waste repositories include probabilistic simulations. It should be recognised, however, that these usually are subject to a number of conceptually important limitations. First, the approaches often mix probabilistic simulations with deterministic, e.g. the biosphere transport and transfer of radionuclides to man and the dose coefficients (which deterministically average out the variability in human organ sizes and metabolism and the effect of an exposure). Obviously, such hybrid results are difficult to interpret probabilistically. Second, present knowledge on the probability density distributions of many model parameters and on associations between them is still too limited to give credibility to the calculated probability density distributions – especially to their low probability / high consequence tails – of the end points of the simulations. Third and conceptually most important, probabilistic approaches generally focus on exploring the implications of uncertainty of model parameters, but not of the conceptual model itself.

The Reference Biosphere Methodology may offer a convenient approach to close this conceptual gap. This becomes immediately obvious from its intention to construct various potential evolutions of biospheres present today. The probabilistic nature of the methodology becomes most easily apparent for the sequential approach using a Markov model to simulate the potential future climatic variations. It seems worthwhile to explore the potential of the Reference Biosphere Methodology for quantification of model uncertainty systematically, preferably in international cooperation (see above). Within such an application, estimating the consequences of the individual biosphere evolution scenarios should be done deterministically. A deterministic approach will not only circumvent the difficulties in interpreting results of partially (with regard to parameters) probabilistic simulations, but also benefits from the robustness of parameter means (or another measure of central tendency) and thus its insensitivity to the limitations of our knowledge of probability density distributions discussed above.

4 Conclusions

A proposal is made how to quantify the general safety principles for long-lived radioactive waste disposal that possible effects in the future should not be greater than relevant levels of impact for health and environment that are acceptable today and that the way of waste disposal should not impose undue burdens on future generations. Our approach uses the current exposures from operating nuclear facilities as a guide to establish an impact range which today is generally accepted. Compared to the traditional approach to focus on limits present in national and supra-national legislation and recommendations, our approach has the advantage to take into account both fundamental radiation protection principles of limitation and optimisation.

It is shown that the annual individual exposures from operating nuclear facilities correspond to lifetime risks of radiation induced cancer within a range of 10^{-6} and 10^{-4} which corresponds well with the risk used by the WHO in deriving their internationally accepted drinking water quality guidelines. We, therefore, propose to adopt this range of risk values as an internationally socially accepted figure of health impact from cancerogenic substances in the environment. It is shown that in contrast to a dose standard this approach provides a common health standard for all toxic substances disposed of in a nuclear waste repository including the non-radioactive waste constituents.

The importance of biosphere processes in any long-term assessment to demonstrate compliance with such a risk standard (or any other constraint) is highlighted. It is shown that the Reference Biosphere Methodology is a convenient new approach to overcome some of the limitations in simulating future biosphere processes. This methodology should be applied in an international context to develop a number of internationally accepted generic reference biospheres. Furthermore, its potential to simulate the influence of model uncertainties should be explored. ■

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2B.06 Potential Role of the Safety Case for the Final Disposal Facility in Germany

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Abstract
The application of the safety case is recommended by NEA and IAEA, recently fixed in the Safety Requirements WS-R-4. To implement the safety case in Germany the processes have to fit with the existing legal procedures. The purpose of the contribution is to show the potential role of the safety case under German conditions. It is shown that the implementation of a stepwise safety case in the German system is possible without major legal changes. Only in the case of the early steps before application for a construction permit additional legal work will be helpful. The function of safety case in the German system should be the set of relevant papers for the specific step, which is provided by the operator and finally accepted by the relevant authority. Clear definitions in terms of legal system are necessary for the role and content of the safety case, for the steps and for the roles of entities concerned. Proposals are given. Regarding the potential role of optimization, proposals for the content and handling are derived. The safety case should include adequate answers for different parts of the public; a broader view is given on that issue.

1 Introduction

The safety case is a tool which is highly recommended in international documents regarding both the demonstration of safety and the decision making process for final disposal facilities. A good number of detailed documents on safety case for final disposal are written and published under the auspices of OECD/NEA (e.g. [1]). There exist also activities of IAEA in the field of safety case. The results of these discussions are documented as the IAEA Safety Requirements No. WS-R-4 "Geological Disposal of Radioactive Waste", published in May 2006 [2]. On national level in different countries the implementation of the Safety case as standard tool is in a very early phase. Up to now only a few examples for safety cases exist on national level in some countries and they have very different status in the respective country.

The IAEA Safety Requirements in general have the character of recommendations. WS-R-4 has the character of a strong recommendation to apply the tool of safety case on the respective national level. In German legal terms it has to be accounted as part of the "state of the art and science" ("Stand von Wissenschaft und Technik").

Up to now no formal application of the Safety Case in Germany exists. Actual rules for decisions have not yet implemented this tool. In the following the potential role of the safety case under German legal situation will be figured out. Also some proposals are developed how to deal with the content of the safety case under the specific German situation.

WS-R-4 [2] gives good margins for application and the role of the safety case. Some of these margins are:

- "The safety case for a geological disposal facility addresses both operational safety and post-closure safety."
- "The adequacy of the design and operational features is also evaluated."
- … "includes consideration of the needs of different interested parties for information."
- "Important considerations are justification, traceability and clarity."

2 Implementation in the National System in Germany

The implementation in a national system is not easy. International documents tend to be very general to avoid conflicts with a specific legal system in a specific country. This is also the case with the Safety Case papers. The mentioned international documents provide different and unclear definitions of both the content and the specific role of safety case.

But on national level a clear definition is necessary, because both the legislation and the rules dealing with the Safety Case should give clear advice for the application. The main questions to be solved are the following:

- Implementation in the national legal framework. The definitions have to fit with the legal system; they also have to take into account the practice in related fields.

- Role of the different players. These roles have to be defined in accordance with the German legal system and with the existing roles of authorities and other entities.

- Function of the documentation. A central part of the safety case is the documentation (in a certain view the safety case is identical with its documentation). It has to be fixed, which role the documentation shall play in terms of legally binding documents and in terms of minimum standards which have to be met. Who has to provide the safety case? Who has to check it and to approve it?

- Relation to the necessary decisions and permits. What is the role of the safety case in the process of these decisions and/or permits?

Regarding these premises three different types of steps can taken into account to implement the application of the tool of safety case within the legal framework in Germany:

- Steps within the legal tool “Planfeststellungsbeschluss”

- Steps regarding the periodic safety review

- Steps of the early phases of decision

2.1 Possible Steps within the Legal Tool “Planfeststellungsbeschluss”

Legal framework for final disposal in Germany is based on the legal instrument “Planfeststellungsbeschluss”. The “Planfeststellungsbeschluss” is specific type of license which is given as the final act of the legal procedure “Planfeststellungsverfahren”. It is fixed in the atomic act (“Atomgesetz”) and in the administration procedures act (“Verwaltungsverfahrensgesetz”). It would need difficult political and legal negotiations to amend these acts. Therefore one has to suppose, that these laws shall remain unamended. So the base for the following consideration is the German legal situation as it is.

Taking into account the questions mentioned above, the application of the Safety case must fit with the instrument of “Planfeststellungsverfahren”. In the past the majority of legal interpretation said that within the “Planfeststellungsverfahren” only one step would be possible. This would not be in accordance with a stepwise approach. But an actual decision of the federal administration court regarding the “Planfeststellungsbeschluss” for Schacht Konrad (final disposal facility for non-heat-generating radioactive waste) gives room to a broader legal interpretation of the instrument “Planfeststellungsverfahren”. According to that decision, it will be possible to split the licensing procedure into several parts.

Using the today’s legal interpretation of this instrument some steps can be realized within the instrument of the “Planfeststellungsverfahren”. These steps are:

- License for construction of the disposal facility
- License for operation of the disposal facility
- Closure of the disposal facility

Additional steps are possible in case of an early closure of a section of the disposal mine.

What is the possible role of the Safety case in these three steps? The "Safety case" should be the whole set of documents, which has to be provided by the operator to the licensing authority. The content of the safety case in total must fit with the needs of different interested parties for information. The set of documents should include a general report, detailed supporting scientific documents and documents with information according to the need and level of knowledge of different stakeholders. Some stakeholder have deep technical and scientific interest, other stakeholder want to have general information understandable in every day language, other look for answers to a lot of questions.

The whole set of documents of the safety case should be the base of the license. This is in accordance with today's licensing practice. In the text of a license all documents on which it is based, are listed.

In the existing German legal system a detailed regulation for the documents in the licensing procedure of nuclear power plants is fixed in the ordinance for nuclear licensing procedures ("atomrechtliche Verfahrensverordnung"). A similar regulation for the licensing of final repository would be helpful, describing the standards for the content of the safety case.

2.2 Steps regarding the Periodic Safety Review

The operation period of the final disposal facility for high level waste in Germany will last some 50 to 80 years. Modern view of to ensure a long-term operation according to the standards of the state of the art is to use the instrument of periodic safety reviews.

Periodic safety review (every 10 year) is part of the legal framework for nuclear facilities in Germany (nuclear power plants, interim storage facilities for spent fuel). Usually a complete status report on safety has to be provided by the operator to the relevant authority. After the review by the authority supported by her experts, the relevant authority decides, whether backfitting or amendments of the license will be necessary or whether there is no need of alterations.

Periodic safety reviews would be necessary for final disposal facilities, too. It makes sense to use an actualized safety case as the adequate status report which will be provided by the operator to the relevant authority. Actualization can include for example:

- new insights on geological situation and, if necessary, reaction on that,
- progress in the design and construction of shaft seal and other parts of the closure measures,
- lessons learned for the enhancement of operation of the facility.

So the safety case can be a valuable tool for the repeated periodic safety review whilst operation.

2.3 Steps of the early Phases of Decision

The third time period one has to look for includes the decisions before the license for construction of the disposal facility. Decisions before the "Planfeststellungsbeschluss" are not codified in the German legal system up to now. In the past those decisions had no clear procedures and no fixed legal base.

It is widely seen that a clear process shall be fixed legally. A main argument for that view is to enhance the transparency of the process. The codification of these early steps is now under consideration within the federal ministry of environment (BMU). It possibly shall be part of a future ordinance on safety of final disposal ("Endlagersicherheitsverordnung").

Possible steps within the future procedure are under discussion. A possible candidate for the first step is the agreement of the site selected for further exploration. It can be fixed by a formal decision, which site shall be explored more in detail. Different opinions exist, whether a comparison of several sites is necessary before such an agreement or whether the agreement shall be based on selection results of the 1970ies. In both cases it would be helpful to have a clear formal decision for the future. It has to be fixed, who will decide on the agreement of the site selected for further exploration. Possible candidates are the federal parliament, the federal government or a relevant authority.

A possible second step within the early phase would be the evaluation of the site exploration results. The exploration results shall be documented in a safety case together with all conclusions which can be drawn. Given a positive evaluation it can be decided by the relevant entity that the procedure for the application of the license for construction of the disposal facility ("Planfeststellungsbeschluss") shall be start. Possible candidates for the relevant entity are the federal parliament, the federal government or a relevant authority.

For the early steps the safety case should be the whole set of documents, which give the base for decision making. It has to be provided by the operator to the relevant entity. It must fit with the needs of different interested parties for information. The safety case for an early step shall include preliminary safety assessments and preliminary technical concepts in the necessary level of details. According to the proposed scheme the first practical safety case in Germany could be the set of documents for the agreement of the site selected for further exploration.

2.4 Possible stepwise Approach in the German Legal System

The proposals developed above lead to the following scheme of steps:

- agreement of the site selected for further exploration (example)
- Evaluation of site exploration results (example)
- License for construction of the disposal facility
- License for operation of the disposal facility
- Periodic safety check (every 10 years of operation)
- Closure of a disposal unit (in case)
- Closure of the disposal facility

According to the proposed scheme the first practical safety case in Germany could be the set of documents for the agreement of the site selected for further exploration.

2.5 The Development of the Safety Case over the Steps

The system of steps which are possible in the German system is described above. The safety case is the complete set of documents for the respective step. In general for each step a complete safety case (complete set of documents) is necessary. The operator has to write down the safety case and provide it to the relevant authority. Than the safety case will be deeply reviewed by the relevant authority (resp. other relevant entity).

After the review it is possible that the authority will accept the safety case in total without any alterations. In that case the accepted safety case is identical with the proposed set of documents originally provided by the operator. Than it will be the base of the formal decision to this step.

Depending on the quality of the proposed safety case it is possible that the authority will ask the operator to complete or to alter parts of the safety case regarding questions or problems identified by the review process. The operator than has to provide revised documents for the respective part of the safety case documentation. The final version of the safety case of each step is reached when in the view of the authority the set of documents is finalized. Than it can be the base of the formal decision (license etc.) on the actual step by the authority. This way of revision of documents by the operator is not unusual in German administration procedures, especially those dealing with nuclear facilities.

At each step two different periods exist with different status of the safety case. During the longest part of the administrative procedure the safety case for this step is a proposed set of documents. In the final phase of the decision process it changes its status to a set of documents which are formally accepted by the authority.

Given the scheme of steps proposed above, the first formally accepted safety case exists with the decision regarding the agreement of the site(s) selected for further exploration. With the finalizing of each following step a new accepted safety case will be added.

For each next step a new safety case has to be proposed by the operator. It can and should be based upon the content of the safety case of the preceding step. Documents with no need of alteration can be part of the new safety case without change. Other documents have to be updated (e.g. status of geological knowledge) or to be more detailed (e.g. long term safety analysis) for the new safety case.

3 The possibilities for Optimization

The described series of safety cases give a good opportunity to implement a stepwise enhancement of safety. In the "next" safety case shall be included inter alia:

- actual knowledge of the geological situation at the site
- experience with exploration and operation ("lessons learned") and possible enhancements in different fields
- enhancement of closure methods (e.g. advanced methods of seal construction)
- reflections of actual international state of the art and discussion of possible influences on the further operation of the disposal facility
- evaluation of possible enhancements, additions or change of methods
- discussion of possibilities for optimization

Optimization is a question raised in the international discussion. But the generic demand has obviously to be substantiated in the national regulations. Clear regulations are necessary for the optimization process and the factors to be regarded.

Optimization could not be just a look on a figure of radiation impact calculated in the long term safety analysis and an attempt to make this figure as low as possible. Such a simplistic view of optimization can lead to a decrease of the real safety, when sophisticated tricks are implemented which just influence the result of calculation but not the real safety situation.

The optimization process should take into account factors as

- robustness of assumptions

- geological predictability
- degree of technical difficulties of realization
- economic aspects of realization
- influence on the overall safety situation

Careful balancing of these factors is necessary whilst the optimization process.

4 Some Remarks to the Safety Case and Questions of the Public

An important demand for the safety case is the transparency and the traceability. This includes that the safety case can be understood by the public. But this is a very tricky thing.

The public is very different. Some people just want to have a short version or even a very short version, which can be understood by people without time for reading or without training in understanding special scientific language. Others want to find answers for a lot of questions, which are not so in the focus of a standard natural scientist or engineer. Examples for that kind of questions are:

- We cannot predict for decades, how can we predict safe disposal for a million year?
- Isn’t a continuous control of the waste safer than reliance on geology or technical barriers?
- How can we trust scientific predictions, when the experience shows, that statements of scientists have been wrong?
- Did you really choose the best available method (e.g. host rock, closure concept)?

To get public acceptance, it is necessary to produce a safety case, which can fulfil challenges raised by the public. It is clear that even an outstanding safety case does not necessarily lead by itself to a good public acceptance. But the safety case as the set of documents will be discussed in public anyway. Weaknesses in those documents will deliver good arguments for those who oppose the project. This is a lesson one has to learn from past debates on nuclear and on final disposal in Germany. Weaknesses in this sense can be scientific failures, but also lack of dealing with questions which are raised in public debates.

It is clear that not each single document of the safety case can fulfil these challenges. But the structure of the complete set of documents, which together are the safety case, must include specific documents with a focus on the issues mentioned. Additionally it would be helpful to check other documents in the set whether some of these aspects can be included.

To make a complete safety case regarding these aspects, some hints are important:

- Listen to the public; do not just imagine how the public would think. The thinking of final disposal protagonists on what the public thinks is not very often in accordance with the real discussions and the real questions of the public.
- Include all aspects in the safety case which are discussed in public. The lack of those discussions in the documentation leads to the conclusion that the protagonists of final disposal have no answers to the most important questions.
- Discussions of those aspects in the safety case must consider a holistic view. A reduction to the point of view of just one scientific discipline would be contraproductive.
- No secret parts of the scientific analysis. In the history of final disposal in Germany relevant papers had not been accessible for the public in several cases. A lot of mistrust in public was caused by those cases.

The written safety case must be accompanied by adequate discussions with different sectors of the public. One important feature of these discussions should be that no discrimination of individuals or part of the public happens caused by the proponents of final disposal. One has to realize that the trustworthiness of different people is different in different parts of the public. Aggressiveness or discrimination against specific individuals or groups lead to a natural opposition of those parts of the public who have a good opinion of the respective individuals or groups. Additionally the impression of undemocratic behaviour is caused by that. A good part of the stile of discussion in the history of final disposal in Germany was influenced by those things.

5 Conclusions

The international demand for the implementation of the safety case was analysed on the background of the existing legal administration system in Germany. The analyses show that the implementation of a stepwise safety case in the German system is possible without legal changes. Only in the case of the early steps before the application for a construction permit for the disposal facility additional legal work will be helpful, as it is planed with the “Endlagersicherheitsverordnung”.

The function of safety case in the German system should be the set of relevant papers for the specific step, which is provided by the operator and finally accepted by the relevant authority. Clear definitions in terms of legal system are necessary for the role and content of the set of documents which are the safety case, the steps in which a safety case will be necessary and for the roles on both the side of the operator and the legislator. Proposals for that are given.

A further issue stemming from international discussion is the optimization. The dealing with and the criteria for optimization should be defined clearly, proposals for that have been developed.

Last but not least: the safety case must include adequate answers for different parts of the public. ■

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2B.07 Correlation in Probabilistic Modelling

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Abstract

Monte Carlo simulation is a computer-intensive method of uncertainty propagation through a numerical model. Latin Hypercube Sampling is a Monte Carlo derivative that has found application in the 1996 Compliance Certification Application of the Waste Isolation Pilot Plant. In probabilistic modelling the calculated quantities are not only characterised by the mean values but also by the distributions. The distributions are not only defined by the dimensionality of the simulated problem, but also by the mutual relationships of the input parameters with respect to each others. The main relationships between input parameters are dependencies and correlations. To obtain practical experiences with correlation in probabilistic modelling, the probabilistic speciation code LJUNGSKILE has been implemented with rank order correlation in order to study the requirements and effects of correlation. The practical application of this algorithm, however, is hampered by the lack of adequate correlation information, the different types of correlation and the mathematical requirements of correlation matrices.

1 Introduction

A considerable number of input parameters to long-term safety assessment are not known accurately for various reasons. A stochastic model simulation uses distributions of the parameters in combination with Monte Carlo (MC) methods to evaluate the resulting output distribution of the safety measure. Classical MC method becomes inefficient for a number I of input parameters I > 3. The reason is coverage. Assuming Nomrally distributed parameter values, a MC sample of input parameters will preferentially be composed of values obtained from the center of the distributions.

The tails will be less represented. In performance assessment, the interest is often directed to combinations of parameter values from the tails of the distributions because these combinations often represent also extreme states of the site for which a repository is not designed. If a MC simulation should have at least 99 % chance to include at least one parameter combination simultaneously from the 75 % percentile(s) of all distributions, 553 runs would be necessary for two parameters, over 11.000 runs for three parameters and about $4.5 \cdot 10^6$ runs for four parameters. Here, stratified sampling (importance sampling) is an alternative. Latin Hypercube Sampling (LHS) is a stratified sampling scheme with high efficiency. Each input distribution of n input parameters is divided into m (m ≥ n) strata giving m regions of equal area under the probability density curve.

The efficiency results from the sequence of combining randomly drawn values from the m different strata of n parameters. If each stratum of an input parameter distribution is sequentially numbered (in ascending order), each stratum is characterized by two indices, the first giving the parameter, the second giving the stratum number. An m x n matrix **M** is built holding these stratum indices where no index number may occur twice in any row and any column. Such matrices are known as Latin lattices [1]. The k-th (k = 1 – m) input vector is composed from randomly drawn value of the stratum by the stratum indexec given in the Latin lattice. LHS ensures that each section of a distribution is considered in the m runs of the probabilistic simulations with equal probability and no section of a distribution is considered more often than others. A more detailed description of LHS is available elsewhere [2,3]. Latin Hypercube Sampling has been implemented in the probabilistic speciation code LKUNGSKILE [4].

LHS does not consider correlation between the values of input parameters. This ignorance does not imply that the values are uncorrelated. In contrary: correlation among the input values is a random influence factor in LHS-based probabilistic simulation. If the rank correlations between input parameters are known, a strategy exists to consider this rank correlation in an LHS-based probabilistic simulation. This strategy is distribution-free [5].

2 Introducing Correlation: The Imam-Conover Approach

If a matrix **M** of input vectors for a simulation has been generated according to the LHS principle, this matrix **M** will have an arbitrary correlation matrix. Imam and Conover have proposed to search for another matrix, **S**, which is also compatible with the LHS requirements, but has approximately the required rank order correlation (ROC). S has the same dimensions as **M**, but is otherwise independent from **S**.

The elements of **S** are arbitrary permutations of the so-called van der Waerden values Φ^{-1} (i/m+1), i = 1,...,m. Φ^{-1} is the inverse of the Gauss error function. The matrix **C** may hold the information on the desired rank order correlation structure. Using the Cholesky decomposition, a lower diagonal matrix is calculated with

$$C = P\,P^{T}. \qquad (1)$$

Now, **S** must be rearranged to show the van der Waerden values in an order corresponding to the identity matrix (all diagonal elements equal 1, all off-diagonal elements equal 0). The correlation matrix of **S** is, say, **E**. By Cholesky decomposition,

$$E = Q\,Q^{T} \qquad (2)$$

The matrix **S***

$$S^* = S(P\,Q\text{-}1) \qquad (3)$$

has a correlation structure close to **C**. Finally, the values in matrix **M** are to be rearranged to have the same rank structure as matrix **S***.

The Imam-Conover approach is a “restricted pairing” method. Hence, it selects among the possible LHS input matrices **M** a matrix which a correlation matrix close to **C**. This restricted pairing method is occasionally considered as sufficient to treat correlation in MC simulation satisfactorily [6]. As will be shown below, this assumption seems to be overly optimistic.

3 Correlation(s)

To apply the Imam-Conover approach the modeler has to provide a matrix **C** with desired correlation structure. Therefore, a deeper understanding of the meaning of correlation and its expression is compulsory. “Correlation” is a general term refering to a relation between two variables. This relation is usually not precise. Usually, a tall person weigh more than a short person, but exceptions exist. Correlation does not imply any causal relationship. To give two examples: the number of storks in Northern Germany and the birth rate are positively ciorrelated. But this correlation does not indicate that children are brought by storks.

Likewise, a positive correlation exists between the amount of rum sold in US communities and the number of Methodist ministers. But it would be unwise to expell the ministers in order to reduce alcoholism.

The term correlation is used loosely to denote a general relationship, but in linear relations between variables, its meaning is well-defined – but not uniquely. Correlation is commonly quantified by correlation coefficients. Several correlation coefficients exist. Pearson introduced the correlation coefficient r in 1901. In the univariate case, r is calculated from

$$r = \frac{n\sum_{i=1}^{n}(x_i y_i - (\sum_{i=1}^{n} x_i)(\sum_{i=1}^{n} y_i)}{\sqrt{(n\sum_{i=1}^{n} x_i^2 - (\sum_{i=1}^{n} x_i)^2)(n\sum_{i=1}^{n} y_i^2 - (\sum_{i=1}^{n} y_i)^2)}} \qquad (4)$$

The correlation coefficient r is fully termed as "Pearson's product moment correlation coefficient" and gives the normalized product of the first moments of dependent observations y_i and independent observations x_i. The Pearson product moment correlation coefficient is derived on the assumption of identically and independently distributed (i.i.d.) observations where the residuals are likewise i.i.d. with mean zero and a single standard deviation (homoscedasticity). At Pearson's time, no method existed to test these fundamental requirements. Today, Kolmogorov-Smirnov test is an appropriate method, at least for the residuals.

The second correlation coefficient is the population correlation coefficient ρ. While r (cf. Eq. 4) refers to values of variables, r refers to the correlation between the variables X and Y themselves:

$$\rho = \frac{\mathrm{cov}(X,Y)}{\sqrt{\mathrm{var}(X)\,\mathrm{var}(Y)}}\,. \tag{5}$$

If X and Y are independent, then ρ = 0. But the fact that ρ = 0 does not imply independence. Both the product moment correlation coefficient r and the population correlation coefficient ρ assume a linear relationship between the data and the variables, respectively. These correlation coefficients are comparatively familiar. Less familiar are the distribution-free correlation measures, i.e. Spearman's rho (rank order correlation coefficient) and Kendall's tau. Both can be used alternatively. The rank order correlation coefficient measures how well an arbitrary monotonic function could describe the relationship between two variables.

Spearman's rho does not require the assumption of a linear relationship between variables nor does it require the variables to be measured on interval scales. It can be used for variables measured at the ordinal scale. Spearman's rho is obtained from Eq. 6:

$$rho = 1 - \frac{6\sum_{j=1}^{n} d_j^2}{n(n^2-1)} \tag{6}$$

Figure 1 shows that the various correlation coefficients are not related to each other: the different data sets have the same value of Spearman's rho, but different values of Pearson's ρ. The approach by Imam and Conover [5] to include correlation in probabilistic modelling is based on Spearman's rho. Therefore, the concept of correlation required for including correlation in

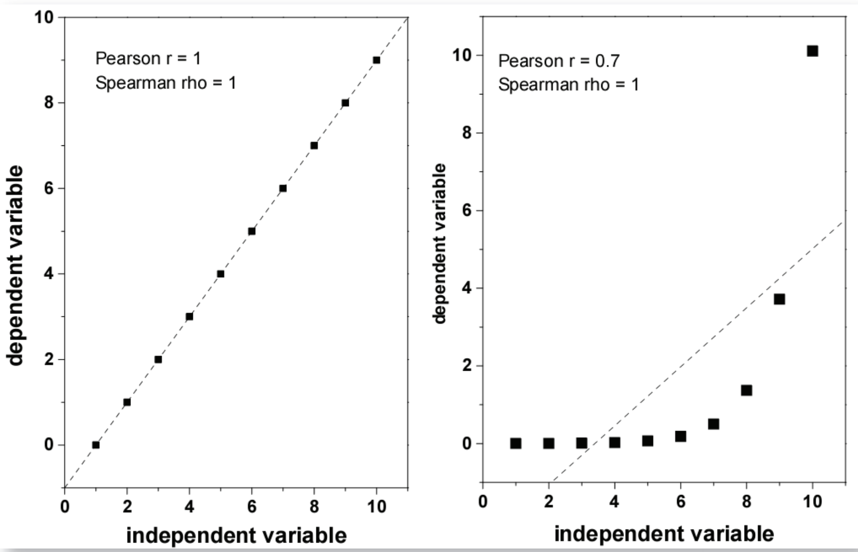


Fig. 1: A comparison of the values for Pearson's r and Spearman's rho for two different data sets. The value of Spearman's rho is the same for both data sets, while the value of Pearson's r varies.

probabilistic modelling is based on a different concept of correlation than the more familiar Pearson correlation coefficient r or the population correlation coefficient ρ.

4 Correlation Matrices

4.1 Properties of Correlation Matrices

Considering rank order correlation (ROC), the one-dimensional relationship between variables can be extended to to multiple variables. The correlation coefficients then become correlation matrices. If a regression model

$$\mathbf{Y} = \mathbf{AX} + \mathbf{B} \tag{7}$$

is given where Y is a p-vector of dependent observations, X is a np-vector of independent observations and A is a p-vector of explanatory variables of the type

$$Y_n = a_1\, x_{n1} + a_2\, x_{n2} + \ldots + a_p\, x_{np} + b_n = (x_{n1}, \ldots., x_{np})\, \mathbf{A} + B_n \tag{8}$$

the least squares estimate for the parameter vector X is

$$\mathbf{X} = (\mathbf{A}^T\, \mathbf{A})^{-1}\, \mathbf{A}^T\, \mathbf{Y} \tag{9}$$

where **X** is the least squares estimate of **X**. The variance-covariance of **X** is

$$\mathrm{Var}(\mathbf{X}) = \sigma^2\, (\mathbf{A}^T\, \mathbf{A})^{-1} \tag{10}$$

where σ^2 is the sample variance.

The correlation matrix is defined in analogy to te univariate case (Eq. 5)

$$\rho_{i,j} = \frac{x_{i,j}}{\sqrt{x_{i,i}\sqrt{x_{j,j}}}} \tag{11}$$

The correlation matrix must be symmetric, Hermitian and positive semi-definite. This requirement implies that the correlation matrix must not have negative eigen/singiular values. The eigen/singular value structure cannot usually be judged by visual inspection but needs to be tested explicitely. Thus not all possible combinations of correlations are allowed. For instance, a variable cannot be strongly negative correlated with each of two other variables that are themselves strongly positively correlated. A correlation matrix is quickly investigated on its appropriateness by Cholesky factorisation. The Cholesky algorithm decomposes a symmetric matrix M into an lower diagonal and upper diagonal matrix **P**:

$$\mathbf{M} = \mathbf{P}\, \mathbf{P}^T \tag{12}$$

Matrix **P** is a root of matrix **M**. The algorithm will fail if **M** is not semi-definite. If correlations from different studies are mixed into a single analysis, or if correlations are based on hypothetical values or best professional judgement, infeasible configurations may be specified. If some correlations are known, the positive semi-definiteness may help to assess the range where the unknown correlations may be.

4.2 A Note on the Generation of Correlation Matrices

Correlation matrices must be symmetric, Hermitian and positive-semidefinite. These conditions have to be tested, otherwise LJUNGSKILE/ROC, the LJUNGSKILE code with rank order correlation, will fail. These conditions cannot be infered on inspection. This is illustrated by Table 1, giving two symmetric, Hermitian matrices with 1's on the diagonal. Only the left matrix is also positive-semidefinite.

Table 1: An appropriate correlation matrix (left) and an inappropriate correlation matrix (right)

1	-0.9	0.6	-0.8	0.75	-0.8		1	-0.9	0.6	-0.8	0.75	-0.8
-0.9	1	-0.7	0.75	-0.7	0.7		-0.9	1	-0.1	0.75	-0.7	0.7
0.6	-0.7	1	-0.3	0.5	-0.55		0.6	-0.1	1	-0.3	0.5	-0.55
-0.8	0.75	-0.3	1	-0.55	0,5		-0.8	0.75	-0.3	1	-0.55	0.5
0.75	-0.7	0.5	-0.55	1	-0.7		0.75	-0.7	0.5	-0.55	1	-0.7
-0.8	0.7	-0.55	0.5	-0.7	1		-0.8	0.7	-0.55	0.5	-0.7	1

There is no objective criterion to identify the positive semi-definite matrix without numerical analysis. Note that the SDV algorithm, a fast algorithm to obtain singular vectors and singular values for positive real-valuesd matrices, is not suitable to analyse a matrix on positive semidefiniteness because a correlation matrix is not necessasrily positive-valued. Hence, the Cholesky decomposition is also used as a detector for non-positive semidefinite matrices.

Methods to generate correlation matrices with desired properties, e.g. distribution of elements, expected values or eigen values etc. have been proposed at various occasions in the statistical literature [7-9]. To construct appropriate correlation matrices with a given mean, the algorithm of Marsaglia and Olkin [9] has been applied to arrive at a correlation matrix R with given eigen values in the interval $(1 - \lambda, 1 + \lambda)$, where λ is the smallest eigenvalue of C (where C is a given correlation matrix). Then C + R – I (where I is the unit matrix) is a random correlation matrix with expected value C. The implementation of the first step of these procedure is iterative and convergence is rapid. A demand to produce a correlation matrix with given mean does only exist if an idea on the required correlation structure is available. Implementing the algorithms is a rather tedious process but saves a try-and-error search for a suitable correlation matrix when running LJUNGSKILE/ROC.

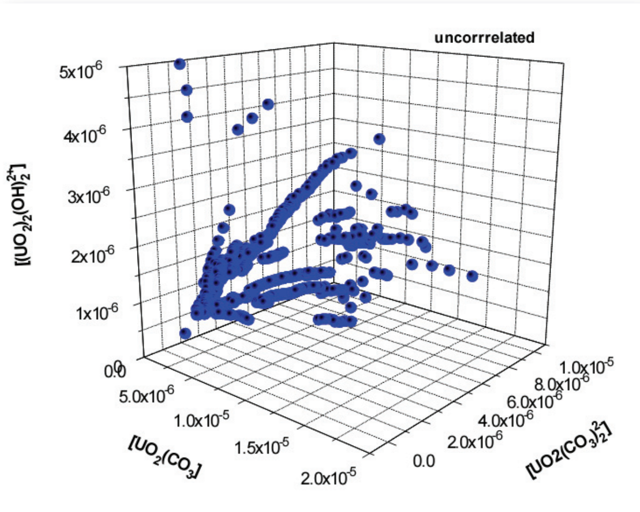
5 A practical Example

The LJUNGSKILE code [4] is based on the public domain speciation code PHREEQC. Repeated PHREEQC runs are performed with user-specified parameter sets. Specification of parameter sets requires specification of species, their formation constants and uncertainties either as standard deviations (normal distribution) or ranges (uniform distribution). The problems and deficits of existing thermodynamic data for aqueous solution species of the actinides have been described elsewhere in detail [10]. In the present context, the almost complete lack of correlation information is of relevance. For some experimental methods the quantity of interest, the thermodynamic datum, must be obtained from data interpretation by non-linear relationships, i.e. pH titrations [11], other techniques allow the evaluation of population correlation coefficients – provided the solubility data is obtained from numerical interpretation (curve fitting) using the L_2 optimisation criterion. However, correlations (or variance-covariance matrices from which the correlationmatrices could be derived) have very rarely been reported. An exception is the variance-covarinace matrix of a Uranium(VI) solubility study by Meinrath and Kimura [12]. The correlation matrix of parameters lg K_s, lg $ß_{101}$, lg $ß_{102}$ and lg $ß_{103}$ (where Ks is the solubility product of solid phase UO_2CO_3(s), and lg $ß_{10i}$ is the formation constant of a uranyl species with i CO_3^{2-} ligands coordinated, $UO_2(CO_3)_i^{(2-2i)}$) is given as Table 2.

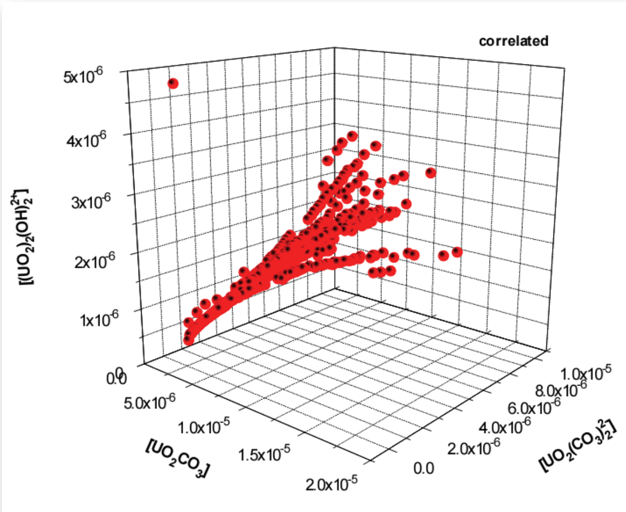
Table 2: Correlation matrix derived from the variance-covariance matrix given in [13]. Only the upper diagonal is given for symmetry reasons

	lg K_s	lg $ß_{101}$	lg $ß_{102}$	lg $ß_{103}$
lg K_s	1	-0.68	0.10	-0.33
lg $ß_{101}$		1	-0.52	0.43
lg $ß_{102}$			1	-0.59
lg $ß_{103}$				1

Due to the fact that these correlations are not Spearman's rho but ρ values they can only give an orientation on the magnitude of correlation to be imposed on the LHS matrix in the LJUNGSKILE/ROC simulations. Two separate simulation experiments are discussed. Common to both simulations is the species model, the thermodynamic data, water composition and the general LHs layout (uncertainties of thermodynamic parameters, distributions and total number of LHS simulations). The difference is the Spearman correlation imposed on the data. In the first run, the off-diagonal elements of the correlationm matrix where set to zero. Hence, correlation within the LHS runs was largely suppressed as far as possible under the LHS constraints. In the second simulation, the correlation matrix given in Table 2 was approximated. The results of both runs is given in Figures 2 (uncorrelated) and 3 (correlation approx. set according to Table 2).



▲ Fig. 2: 3D Scatter plot of concentrations of U(VI) species $UO_2CO_3^{\circ}$, $UO_2(CO_3)_2^{2-}$ and $UO_2(CO_3)_3^{4-}$ calculated with LJUNGSKILE/ROC using thermodynamic data from ref. [Meinrath 02]. The scatter is resulting from 1000 LHS runs with nominally uncorrelated parameters.



▲ Fig. 3: 3D Scatter plot of concentrations of U(VI) species $UO_2CO_3^{\circ}$, $UO_2(CO_3)_2^{2-}$ and $UO_2(CO_3)_3^{4-}$ calculated with LJUNGSKILE/ROC using thermodynamic data from ref. [Meinrath02]. The scatter is resulting from 1000 LHS runs with rank order correlation of the magnitude given in Table 2.

With the exception of one single point in Figure 3, the correlated scatterplot is more homogeneous compared to the uncorrelated data. However, it must be stressed that from several repetitions of the simulation the plot with the visually clearest difference to the uncorrelated plot Figure 2 has been shown. In fact, the scatterplot may differ considerably from run to run – in the correlated case as in the uncorrelated case.

6 Conclusions

A number of issues emerged from the correlation studies using a LJUNGSKILE code augmented by the Imam-Conover rank correlation algorithm. At first, an issue should be addessed which has not mentioned until now: the relationship between correlation and dependence. Here, only the issue related to correlation have been mentioned to keep the discussion focused. The relationship between dependence and correlation is not a prominent topic in probabilistic modelling. From discussions of statistical issues in performance assessment the impression emerged that the difference between correlation and dependence is not well understood and a series of misconceptions exist. In fact, correlation and dependence are not completely independent topics. In general, however, including correlation into a numerical simulation does not imply that dependence is automatically considered, too. It is also inappropriate to ignore correlation and dependence completely.

Correlation has a defined meaning in statistics of linear models, expressed by the population correlation coefficients ρ and the Pearson product moment correlation coefficient r. These more familiar expressions of the magnitude of correlation, however, cannot be applied to non-linear models. Clearly, different correlation coefficients are not directly comparable and express rather different aspects of a data set. The Spearman rank order correlation, however, is usually not available for parameters relevant in, say, reactive transport modelling.

The general impression from many repeated simulations of the same problem with LJUNGSKILE/ROC is a certain similarity when comparing scatterplots with and without correlation. Ferson et al. [14] also describe this observation due to the selection of one special dependence having the desired correlation pattern out of a large number of possible alternatives with varying dependencies. Hence, the effect of correlation alone cannot possibly be studied without simultaneous consideration of the dependencies.

A final (but not the last) issue is the combination of correlation information from different sources. The models investigated here had a very limited number of parameters. A performance assessment code will have several hundred input parameters. Correlation between parameters cannot be specified arbitrarily (cf. Table 1). ■

Acknowledgement

The author is thankful to Scott Ferson (applied Biomathematics, Setauket, USA) for providing a copy of the report 'Myths about correlations and dependencies and their implications for risk analysis' [14] prior to publication.

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2B.08 Data Sufficiency in Repository Site Investigation

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Abstract

An objective, quantitative, and defensible measure of data sufficiency is essential to the planning and management of repository site investigations. This data sufficiency measure is needed to support decisions concerning the scope of investigation activities, and the required data quality and data quantity. This paper presents two data sufficiency measures for this application, and compares and contrasts their relative strengths and weaknesses. The first approach is an empirical "reliability index". This method was developed specifically for the proposed German repository-site selection process. This empirical index is a way to compare the data situation for a given location with that which is typical for a repository at a similar stage in the progression from site selection through to construction. The second approach is based on the use of Bayes theorem to provide an evaluation of the utility of each site characterization activity in terms of its power to reduce uncertainty in a specific performance measure, such as repository safety.

1 Introduction

Questions of data sufficiency arise when planning the investigations for a repository and during the site investigation process itself [1]. How much of a particular type of investigation is necessary in a particular case? These decisions involve balancing of resources and time on one side and the achieved level of knowledge (or reduction of uncertainty) on the other. In the context of a repository programme, data sufficiency can be defined as the level of information necessary to, for example,

- minimize the probability that an inadequate site would be selected over one with appropriate geological conditions
- initiate the next stage of repository development with a level of geological uncertainty consistent with that of other repository projects worldwide at a similar stage of development.
- ensure that the selected site has an acceptable probability of meeting the criteria for licensing

Of these criteria, the first depends on a detailed simulation of the entire site characterization and licensing process for each site. The second criterion requires only the ability to compare the current programme against other, well-documented respository projects. This is the "reliability index" approach developed in Section 2 below. The third criterion requires an optimization of the "post-characterization" data situation based on "pre-characterization" information. This is the "Bayesian" approach developed in Section 3 below.

The reliability index approach was developed considering the staged repository site selection approach proposed by AkEnd [2]. AkEnd's approach makes use of criteria in the early stages – first for exclusion of unsuitable sites and then for making comparisons between apparently suitable sites. The evaluation criteria were formulated by AkEnd to make use of available data to try to ensure that the sites remaining in the process would in the end be able to be shown to be suitable. In the later stages of the process proposed by AkEnd this suitability must actually be demonstrated by using site-specific data to prepare a long-term safety analysis.

The "Bayesian" approach has been used successfully for geological investigations since at least 1972 [3]. The Bayesian approach utilizes "prior" information, based on early limited data, and quantifies how the uncertainty would be reduced for each alternative outcome from the site characterization programme to produce a "pre-posterior" level of uncertainty.

2 Reliability Index Approach

2.1 Background

The reliability index approach was developed by a consortium of companies in order to prepare a report [4] for the German Federal Radiation Protection Office (Bundesamt für Strahlenschutz – BfS). A key question posed by BfS concerned the sufficiency of the data available to a third party for use in a site selection process. One task was to evaluate whether this would be sufficient to enable the suitability criteria proposed by AkEnd to be applied reliably. For this it was necessary to develop a method for evaluating in combination different types of existing investigations of relevant conditions in areas which might be used to contain a repository. The method developed involved the calculation of an index of data sufficiency – which was called the reliability index. This was calculated separately for each of the AkEnd criteria.

2.2 Basis

Two basic data characteristics were used in setting up the index – the quality of the available data and its quantity. Quality was taken to mean the ability of a method to determine precisely the value of a target parameter. Quantity was considered as data density – equivalent in general to the number of observations made within some defined area or volume.

For the purposes of the work for BfS a table was set up which formed a basis for categorizing the available data sets in terms of their quality and their quantity. The category definitions were mainly based on the experience of the members of the project team. Examples of these definitions are given in the following tables.

Table 1: Examples of definitions of categories for data quality

Data type / Categories	1	2	3
Borehole	Cuttings logged	Well logging	Core logging
Geophysics - seismic	Analogue	Digital	3D digital
Determination of in situ hydraulic conductivity	Pump test	Injection test	Multi-stage packer test with downhole cut-off valve
Determination of hydraulic gradient	Derived from packer test	Multi-packer observations	Long-term piezometer observations

It should be borne in mind that in particular in the early stages of a repository site selection programme it is likely to be necessary to make use of data and data evaluations which were collected and prepared for other purposes. Such information may only be available in the form of maps and other types of publications. The approach taken to assigning such data sets to a quality category was based on the source – e.g. a document issued by a public body was ascribed to a higher quality category than a publication by an individual. The underlying idea here was to account for the presumed extent to which the presented information was the product of a completed scientific discussion.

Table 2: Examples of definitions of categories for data quantity

Data type / Categories	1	2	3
Borehole	≤ 1 /100 km²	1 / 10 km² to 1 / 100 km²	≥ 1/10 km²
Geophysics - seismic	≤ 1 profile / 10km²	>1 profile / 10km²	3D digital
Determination of in situ hydraulic conductivity	≤ 1 determination / 50 km³	1 determination / 50 km³ to 1 determination/km³	≥ 1 determination/km³
Determination of hydraulic gradient	≤ 1 determination / 50 km³	1 determination / 50 km³ to 1 determination/km³	≥ 1 determination/km³

2.3 Formulation

In setting up a practical method for calculating the value of an index on the basis of the ascribed quality and quantity categories of the available data sets it was assumed that:

- the reliability which is achievable in using the data depends on both the data quality and on the quantity of data available; and that
- the quality of the data is more significant for the reliability than the quantity of data available – a lot of data points which are known to provide poor estimates of a parameter are worth less than a few points which are expected to provide good estimates.

These ideas were expressed in the following expression for the reliability index (R_i) for each data set i:

$$R_i = C_{Qual} * (C_{Quant})^{Exp_{Quant}} \quad (1)$$

with

C_{Qual} ascribed quality category (1 to 3)

C_{Quant} ascribed quantity category (1 to 3)

Exp_{Quant} an empirical exponent (less than one for the case of quality being more important than quantity)

It can be expected in practice that a specific suitability criterion may be evaluated on the basis of several data sets. For example, AkEnd proposed that there should be a minimum depth of 300 m from the ground surface to the top of the confinement zone, and assessment of compliance with this criterion might reasonably be expected to be based on data from both boreholes and from geophysics. As these may have been ascribed to different quality and quantity categories the following expression was used to calculate the combined reliability index (R_c) for n sets of data:

$$R_c = \frac{(\Sigma_{i=1 \text{ to } n} R_i * C_{Quantn}) * (n)^{ExpSets}}{(\Sigma_{i=1 \text{ to } n} C_{Quantn})} \quad (2)$$

with

C_{Quantn} quantity category (1 to 3) for data set i alone

ExpSets a second empirical exponent (less than one for the case of a reduced weighting for additional data sets)

This expression takes into account the possibly different quantity categories of the data sets which are being combined.The two empirical exponential coefficients, in Expression 1 and in Expression 2, both have the effect of weighting data quantity relative to data quality – the lower the values chosen for the exponents the less is the weight given to data quantity in calculating the value of the reliability index. When making comparisons between different sites or different suitability criteria on the basis of the index it is of course necessary to use the same exponent values in each case. In the work carried out for BfS both were always set to 0.5.

In Figure 1 calculated values of reliability index are shown for cases with the exponents both set to 0.5. Results are shown for combinations of different numbers of data sets with defined quality and quantity categories.

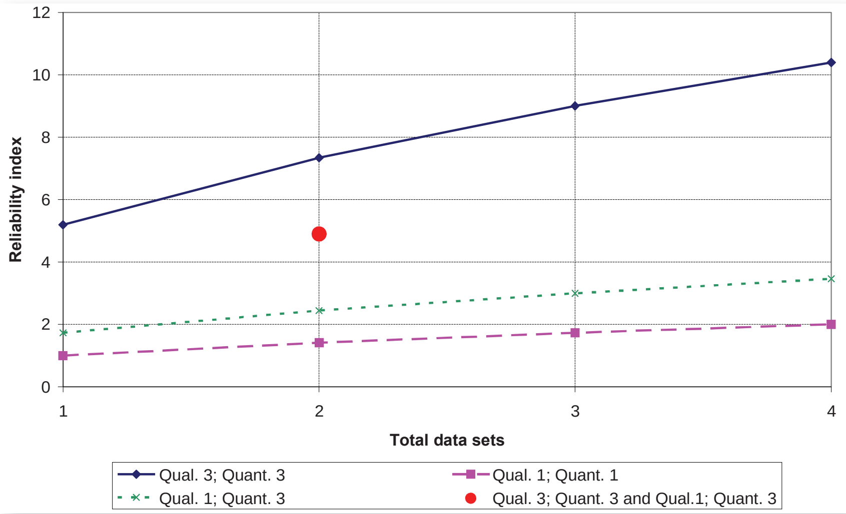

Fig. 1: Calculated reliability index values for various groups of data sets

The consequence of the formulation of this index can be seen from Figure 1. Given the much greater weight assumed for quality over quantity, even with large quantities of low quality data, e.g. quality category 1, the reliability achieved will never match that obtainable with high quality data sets. Another is that adding further data, but with lower individual reliability indices than that for the original data set, can even result in a reduction of the calculated combined reliability.

2.4 Calibration

The reliability index is used to determine how the level of knowledge for the site compares with that for other sites at a comparable stage in the investigation and selection process. This necessitates a type of "calibration" by obtaining the reliability index from the progress of completed projects. In the work carried out for BfS this calibration was carried out against

- a hypothetical project represented by the recommendations of AkEnd, and
- published information from site investigations carried out for NAGRA at Benken in Switzerland [5].

Although it is clearly preferable to use site-specific data when evaluating whether a suitability criterion is satisfied, AkEnd took into account that this may not always be available – for example for the in situ hydraulic conductivity. For the purposes of the proposed selection procedure AkEnd accepted that in cases where no measurements had been made use could be made instead of information on the rock type. For the criterion to be satisfied the rock at the site would have to be expected to have a satisfactory value of the missing parameter. Making the assumption that the relevant data contained in the AkEnd report resulted from a completed scientific discussion (see Section 2.2) these were ascribed to quality category 3. These measurement data are however for other sites and were therefore be ascribed to the lowest quantity category, i.e. 1. Combining these two values in Expression 1 produces the index value of 3 and this was considered to be the lowest limit of data sufficiency for steps 1 and 2 of the repository site selection process proposed by AkEnd.

Prior to the drilling at the Benken site NAGRA had prepared a complex interpretation based on measurements made at other sites in the same geological formation – the Opalinus clay. With a similar logic to that described in the previous paragraph this interpretative work would result in a reliability index of 3 for the Benken site. Following the drilling and testing the area was described as being a first priority repository area (Lagergebiet erster Priorität) and the suitability of the area as a repository for high and medium level radioactive waste decision was formally documented [6]. The additional data on in situ hydraulic conductivity was obtained from packer testing of three 50 m long intervals of the borehole, and this corresponds to category 3 for data quantity. The testing made use of equipment and methods corresponding to quality category 3. Applying Expression 1 we obtain a value of the reliability index of 5.2 for the new data set alone, and combining the two data sets using Expression 2 a value of 6.6 is obtained for the situation after the completion of the hydraulic testing.

2.5 Application

After calibration, the reliability index approach was applied to a set of possible areas for a deep repository in Germany. This application was not entirely satisfactory, since it was frequently not possible to obtain the relevant information. A significant factor in this result is that much of the data on conditions deep underground is privately owned, e.g. by oil and gas companies, and was not made available to the project team. In contrast to this general picture, it was possible to calculate reliability index values for some sites which had already been investigated in the course of the German repository programme.

It is evident that the reliability index described above can be applied to make comparisons between successive stages of a repository programme. It has the advantage that it enables different types of data sets to be considered in combination. In the same way, and with the same advantage, it may be used to make comparisons between different repository programmes. The index, while quantitative, is not a direct measure of repository safety or performance or of site suitability. Nevertheless, because of the comparison with other repository projects, it can provide a defensible basis both for decision-making and for communicating with the public.

3 Bayesian Evaluation

3.1 Strategy-based Approach to assessing Data Sufficiency

The reliability index approach discussed above allows a direct, quantitative comparison to be made between sites, and between investigation programmes. However, it can only be applied as far as there are reference projects available. For the case of deep repositories for radioactive waste this extends as far as the construction and first ten years of operation for a single repository, the “WIPP” site in New Mexico.

The Bayes theorem approach starts from the current state of knowledge of the system under investigation and uses this as the basis for evaluation of the improvement to be expected in that knowledge if a possible site characterization strategy were implemented. This is more quantitative and than the “quality” and “quantity” categories of the reliability index approach. In addition, the results of the Bayesian approach are expressed directly in terms of the reduction of uncertainty obtained from each activity. This facilitates quantitative decision making.

The Bayesian approach provides two results. The first is an evaluation of the probability that a particular characterization strategy for a specific site will be sufficient to achieve a desired level of uncertainty reduction. We can reasonably assume that the calculated performance of a repository at the site will depend strongly on a few key parameters, e.g. the hydraulic gradient in the host rock or the presence of specific geological features. If, following the implementation of a particular strategy, these key parameters will have been sufficiently well defined to allow the safety of the site to be demonstrated the data will then be sufficient, in that respect at least.

The second result from the Bayesian approach is a comparison of competing strategies in terms of their expected effectiveness in reducing specified uncertainties about the investigation site. This information, together with the resources required for each strategy, can be used as an input to an optimization of the investigation procedure. Some more detailed descriptions of the ways in which the approach could be applied to various issues considered to be relevant to the German repository programme were given in a report [7] prepared for BfS.

3.2 Formulation

Bayes theorem enables the existing (conventionally called the a priori) belief about the probability distribution for a specific parameter (P{M}) to be revised on the basis of new observation results (R). After applying this procedure a updated (conventionally called the a posteriori) probability distribution (P'{M|R}) is obtained, given that the observation results R are available. The formula for the calculation of the updated parameter value is as follows:

$$P'\{M_i|R\} = \frac{P\{M_i\}\, P\{R|M_i\}}{\Sigma_i\, P\{M_i\}\, P\{R|M_i\}} \qquad (3)$$

with:

$P\{R|M_i\}$ the probability that the results R will be obtained in the case of the true parameter value M_i

At the initial stage there is information available about the specific characteristics of the methods which may be used for making observations but few, if any, specific site characterization results. For this stage the Bayesian approach can still quantify of the probability of different outcomes from any defined investigation programme. This "preposterior" estimate can be used to evaluate alternative characterization programs and alternative sites, before the major investment is made in characterization fieldwork.

3.3 Application

The example given here is an illustration of the application of Bayes theorem in connection with a strategic choice in the context of a site investigation programme. At an imaginary site there are known to be steeply dipping parallel faults and it is desired to define more precisely what the spacing of the faults is. This could affect the suitability of the site for a repository for a given quantity of waste, or significantly restrict the geometry of an acceptable repository.

Based on the existing data (the a priori condition) two possible descriptions (conceptual models) of the site have been generated and a probability of occurrence has been ascribed to each of them as shown in Table 3 below:

Table 3: Probabilities of occurrence of a priori models

Model	Fault spacing (m)	Probability of occurrence
1	100	0.3
2	300	0.7

A number of investigation methods are under consideration to address this issue, and, as is usually the case, they have different characteristics. Assuming for the purposes of this example that the investigation results can only take one of three values and that one of the conceptual models is the correct one then the expected probabilities of the results are shown in Table 4 for one of the candidate investigation methods, perhaps additional geophysics. This shows, for example, that if Model 1 is actually the correct one then there is a 0.6 chance that the result obtained from the investigation will be 100 m, but only a 0.05 chance that it will be 300 m.

Table 4: Anticipated probabilities of fault spacing results from the investigation

Model	Probability of fault spacing result		
	100 m	200 m	300 m
1	0.6	0.35	0.05
2	0.1	0.1	0.8

Using Expression 3 and the two sets of information in Tables 3 and 4 the calculated updated probabilities for the occurrence of Model 1 or of Model 2 after application of the investigation method are shown in Table 5.

Table 5: Updated (a posteriori) probabilities of Model 1 or Model 2 being correct

Result obtained	100 m	200 m	300 m
Model	Occurrence probabilities		
1	0.72	0.6	0.03
2	0.28	0.4	0.97

Inspecting these values in comparison with the a priori occurrence probabilities of 0.3 for Model 1 and 0.7 for Model 2 we can conclude that this investigation method is a promising one. There is a good chance that a result will be obtained which significantly increases the certainty about the fault spacing which actually applies at the site. If the result obtained from the investigation were actually 300 m this would indicate that there is a 0.97 chance that Model 2 is the correct one.

To compare with other possible methods we can consider the expected value of the updated probability under the assumption that one model or the other is true. With the case considered here the uncertainty in the fault spacing is considerably reduced, which will not necessarily be the case for all types of investigation. One extreme example is a few vertical boreholes, which would result in very little change from the a priori probabilities. In contrast, a single horizontal borehole could provide complete certainty.

As far as data sufficiency is concerned we can consider the situation which would apply if the investigation method is applied and the result obtained is 300 m. The indicated probability of 0.97 that Model 2 is the correct one, i.e. that the actual fault spacing is 300 m, is sufficient for any practical purpose.

4 Conclusions

The issue of data sufficiency is important for the planning and management of site investigations made in connection with repository programmes. Two approaches have been presented here.

The reliability index approach is based on categorizing site investigation activities in terms of data quality and quantity, and the calculated index is calibrated against comparable projects. The power of this approach lies in its ability to rank sites and site characterization programmes against their peers. A limitation of this empirical approach to evaluation of data sufficiency is that it does not evaluate directly the data uncertainty for the site.

The Bayesian approach considers probability explicitly, and takes advantage of knowledge theory to assess data sufficiency. In the Bayesian approach each possible outcome from a site characterization activity is evaluated together with the resulting reduction in uncertainty obtained by carrying out the site characterization programme. The Bayesian approach provides a direct measure of data sufficiency.

The Bayesian approach also relies on comparable projects, as experience with repository site characterization and with each site investigation technique is needed to assess the prior and pre-posterior probabilities which form the core of the approach.

The reliability index and the Bayesian approach have the potential to significantly improve the defensibility of site selection and site characterization decisions. Both methods rely on the use of information from comparable projects. The Bayesian approach has the advantage that it can express directly the probability of "success" and is therefore preferable for cases where explicit goals can be defined. ■

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2B.09 Tectonic Processes, a Key Feature in Selecting Areas for Radioactive Waste Disposal in Russia

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Disposal of high level radioactive wastes (HLRW) in geological formations for a time of 10 — years is, at present, the only possibility of its removal from the biosphere. Therefore, selection of crust sites, in which the ecological safety of HLRW isolation is ensured for its lifetime, is an acute problem in Russia, as weil as in: many developed countries.

Evaluation of geodynamic and tectonic processes velocity with time, as well as their effects on isolation properties of structural tectonic blocks is a principal condition ensuring safety of HLRW disposal.

From practical viewpoint, it is important (area of Krasnoyarsk in Siberia) to predict ernst blocks evolution of Nizhnekansky granitoid massif in Russia where work is being carried out to select a site of underground laboratory construction. At the same time, it is well known that this area belongs to the zone of active orogenesis and its long-term geodynamic stability has not been studied yet.

The methods to predict tectonic blocks evolution when selecting sites HLW disposal sites is based on the following principles:

- Evolution of the Earth‘s ernst is related to the intensity of tectonic processes development in the region. The decisive factor is the initial level of effective tectonic stresses, block structure of the environment and physical and mechanical features of a rock massif.
- Tectonic stress field varies in time and space retaining the inherited tendencies of the preceding period of the region tectonic development. Corresponding indicator is the degree of dislocations, geomorphologie characteristics and other features of the geological environment.
- Modem stress and strain of the geological environment combined with the inherited time-space variation of local fields of tectonic stress determine the development of geomechanical processes of deformation and destruction in each individual region and the possibility of forming new tectonic faults or activating existing faults (zones of weakness and others).

It is planned to develop a dedicated finite-element program complex as a calculating tool of modeling stress and strain variation in rocks with time.

The studies (GPS-technology) of time-space regularities of modern crustal movements and the definition of the correlation between regional and local components of tectonic deformations caused by structural and hierarchical heterogeneity of the geological medium are major tasks in prediction geoecological safety.

The study is supported by RFFI (grant no. 05-05-64975) and ISTC (project no.2764).

2B.10 3D Modelling of Complex Geological Structures and its Relevance for Numerical Groundwater Models – A Case Study

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Abstract

Environmental risk assessment and management requires the integration of disparate information from different geo-data sources. The analysis and prediction of scenarios based on this information should be performed within 3D-modelling tools. Especially the safe disposal of nuclear waste material requires an accurate understanding of geological barriers and should be related to groundwater flow and transport processes in the surrounding of a potential repository. Therefore conceptual geological and hydrogeological models are validated by analyzing site investigation data in 3D space and through time.

An integrated "true" 3D modelling concept within a GIS-based environment is described by the example of a regional groundwater contamination within the heterogeneous aquifer situation of the Bitterfeld mega-site, which is characterized by complex hydrogeological conditions. This regional example in high resolution 3D modelling of geology, groundwater flow and transport provides specific experience in the consecutive integration of 3D modelling results as well as the comparison of the findings of different 3D modelling tools and concepts.

Due to the distinct heterogeneity of the upper aquifer (Quaternary) and the occurrence of large mining dumps, a detailed 3D digital geological model of the subsurface geology was built. The present structural model is the base for a "Spatial Model", covering 65 km² in total and enables investigations on 3D regionalization of contaminants within distinct geological layers. It also provides the basic information for subsequent groundwater flow- and transport modelling with Modflow®/MT3D®, and Feflow®.

1 Introduction

Environmental management and planning requires the integration of disparate information from different sources and the analysis and prediction based on this information with efficient tools for assessment and evaluation. Especially safe disposal of nuclear waste material requires an accurate understanding of geological barriers and an evaluation should be related to hydrogeological prediction in the surrounding of a potential repository. An adequate approach is exemplary performed in the Bitterfeld - Wolfen area (Germany), where structural geological and hydrogeological models are validated by analyzing site investigation data of heterogeneous porous aquifers in 3D space and through time. To provide subsurface geological data for groundwater modelling, the digital 3D modelling of geological structures has markedly increased over the last years, moving from a 2D mapping to a 3D modelling culture related to advanced software and IT capabilities. Recent examples of the mega-site of Bitterfeld-Wolfen, characterized by a regional contaminated groundwater situation, show the growing applicability of this digital 3D modelling concept in practice [1-5].

With a sufficient amount of information and quality of data, the coupling of high resolution 3D subsurface models with numerical groundwater models and simulations become increasingly important and very useful. So, an integrated approach of combining modelling results provides the needed basis for sustainable strategies in environmental risk assessment and management. This paper describes an example of an innovative 3D modelling application for exposure and pathway simulations of contaminants from the former industrial and mining areas, referring to the specific heterogeneity of Quaternary sediments as well as to the artificial morphology of remaining lignite occurrence and related post-mining geology.

2 Setting of the Former Industrial and Mining Area

Large scale groundwater contamination sites, such as the Bitterfeld - Wolfen area, are characterized by different environmental impacts caused by the former chemical industry and extensive landscape devastation by lignite mining over the last 100 years or more [6, 7, 8]. Due to the multi-source regional contamination in the upper and lower aquifers, an integrated assessment regarding the groundwater quality and the local risk areas is needed. Therefore, a regional 3D spatial model of distinct individual environmentally related core modules has been developed for the urban Bitterfeld - Wolfen area.

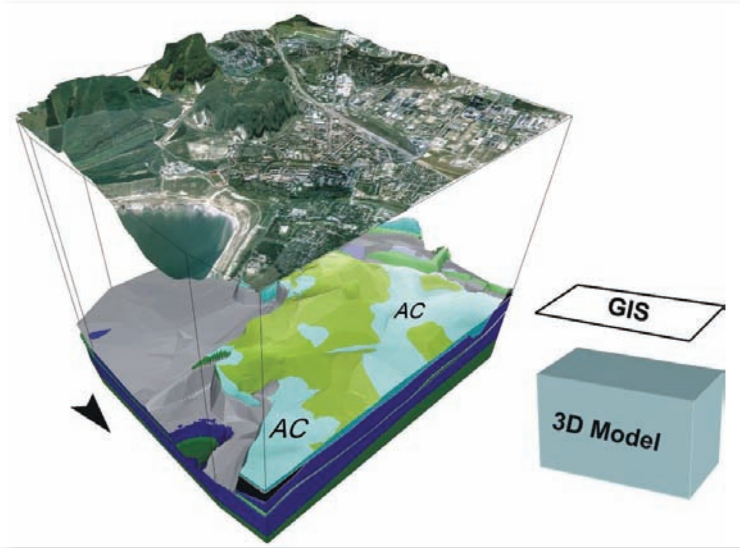


Fig. 1: Geological structures defining aquifer/aquifer contacts (AC) within the Quaternary sediments, as shown here for a close-up of the Bitterfeld area by 4 x 4 km. The concept of the environmental risk assessment is technically based on an integrated approach of "true" 3d subsurface modelling by different methods within a GIS-based environment.

The groundwater situation within the study area has been affected by former chemical production and abandoned lignite mines over an area of more than 60 km², and therefore an intensive monitoring program has been carried out for more than 15 years. About 170 individual organic substances, as well as 30 inorganic substances, have been monitored. The importance of the former industrial area is proved by the more than 40.000 chemical plant workers during GDR time, by 15 industrial toxic landfills, and by more than 5.000 individual chemical substances produced during the nearly 80 years of industrial activity [9].

Due to the multi-source regional contamination of the upper and lower aquifers, risk assessment based investigations of distinct exposure routes of the contaminated ground water have been made. The hydrochemical situation is characterized by a complex mixture of organic compounds comprising a high diversity of individual organic substances as well as a high regional variability of contaminants in the aquifers, respectively [8, 9, 10, 11]. The most frequent substances found in the ground water are Tetrachloroethene (PCE), Trichloroethane (TCE), cis-1,2 Dichloroethane (cis-1,2DCE), Vinylchloride (VC), 1,4-Dichlorobenzene (1,4 DCB), 1,2-Dichlorobenzene (1,2 DCB), Monochlorobenzene (MCB), and Benzene. The regional distribution of contaminants reflects their different and multiple sources and pathway relations. The cities of Bitterfeld and Wolfen, as well as several villages and the rural areas of the alluvial plain affected by high floods of the River Mulde, describe the environmental sensitivity to humans and natural resources [4, 8, 10].

The Bitterfeld area is located at the lower terrace and alluvial plain of the Mulde River and can be described by a generalised hydrogeological and geological cross-section, depicted in figure 2. The hydrogeological situation is characterized by the presence of two porous main aquifers of different heterogeneity which are partially affected by former open-cast lignite mining activities. The upper groundwater aquifer consists of Quaternary sands and gravels. The Quaternary units can be divided into a lower part, represented by lower terrace sediments of the Weichselian Mulde deposits, and overlying sediments, composed of braided river deposits of a smaller tributary stream. Both units are partially separated by a clay layer that provided a hydraulically effective barrier to flow [1, 12, 13].

The upper aquifer (Quaternary) is partially underlain by Lower Miocene lignite seam, acting as a local aquitard. The lignite seam has been intensively mined in the southern part of Bitterfeld. The lower aquifer consists of Lower Miocene and Upper Oligocene micaceous sands of different hydraulic conductivity in its upper and lower parts. The base of this hydrogeological section is represented by Lower Oligocene clays (Rupelian Clay). This clay is considered to be the aquitard of regional scale, hence corresponding to the base of groundwater pollution.

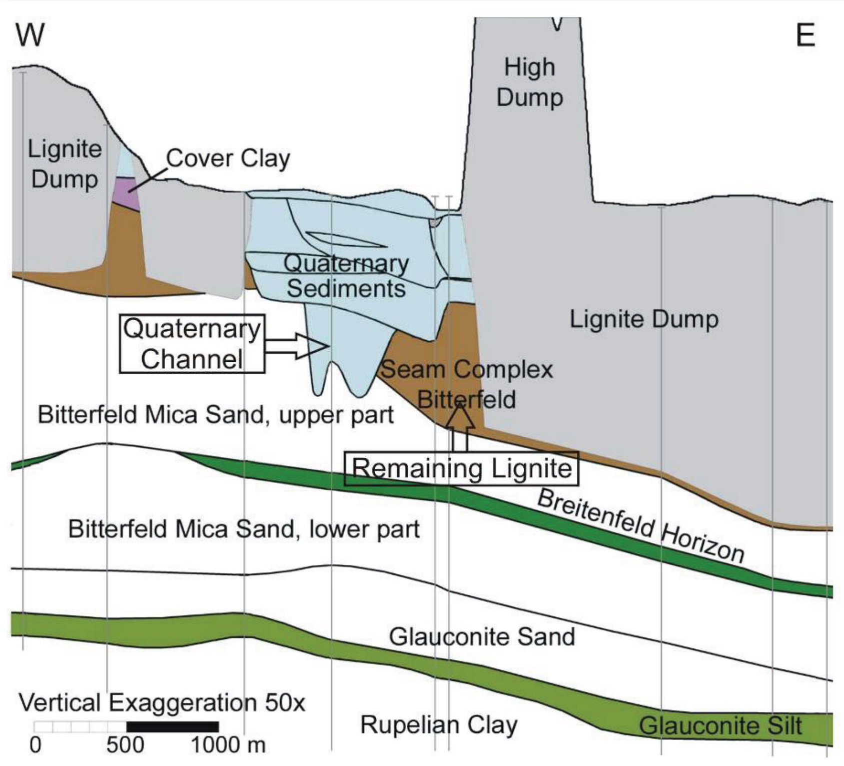


Fig. 2: Generalized cross section of the Bitterfeld area. The bottom layers up to the upper part of the Bitterfeld Mica Sand are smooth marine or lacustrine layers, the Quaternary sediments representing glacial sands and tills are very heterogeneous with deep channels ranging mostly in North-South direction (modified after [1]).

Besides the complex situation of mining-induced aquifer geology, the hydraulic regime of the aquifers has changed completely over the time. During the past mining activities, the groundwater table was lowered tremendously, inducing a regional shift of the groundwater flow directions due to related water exploitation with extended extraction cones. After the re-unification of the two German States in 1989, the lignite open-pit mining and the groundwater extraction came to an end. Subsequently, the groundwater table of the Bitterfeld area was continuously rising and reached the former natural position when an exceptional flooding event of the Mulde River happened in August 2002. The flooding event filled up the remaining open-pit mining Lake Goitzsche within two days and raised the groundwater table of more than 8 m [4, 14].

The dynamic change of hydraulic conditions in a regionally contaminated area with a high complexity of geological structures needs very detailed and accurate subsurface information about the heterogeneous aquifers and their related hydraulic properties for modelling purposes. The simulation and identification of flow pathlines within the regionally contaminated groundwater could only be done by 3D numerical groundwater modelling. Environmental risk and impact assessment requires an appropriate true 3D structural geological model of high resolution for subsequent simulation to minimize the uncertainties of estimated exposure routes of groundwater contaminants within the heterogeneous and complex aquifer situation, particular paying attention to the Quaternary sediments.

3 3D Subsurface-Modelling

Due to the complexity of 3D modelling software tools and the specific situation of regional geology, users might be not aware of the differences in their models or resulting limitations and disadvantages. The availability of subsurface data in their quality and quantity could limit the modelling approach. Apart from this general question, large information gaps often must be bridged during the modelling process, because the regional setting and coverage of subsurface drilling information do not meet the requirements of modelling software for the most applications. 3D modelling software and visualization tools are available using geostatistical algorithms (e.g. EVS®/MVS®, RockWorks®) that work completely different than the TIN-based (triangulated irregular network) interpretation using intersecting cross-sections (GSI3D®). The shortcomings of automatic contouring by statistical and geostatistical algorithms are compensated by the use of constructive intersected cross-sections and mapping information.

To achieve better transitions between "handmade" and expert driven lithostratigraphic correlations and computer processed representations of geological information, it is important:

- to find ways of summarizing, modelling, and visualizing the differences between both scenarios,
- to respond to the increasing quantity of data and the different quality of information, and
- to create simulations of geological phenomena in a digital form coming close to natural scenarios.

"True" 3D subsurface modelling of complex sedimentary sequences is a challenge by itself and is one of the main focuses of the paper, comparing different 3D modelling tools and modelling concepts by using the same geological data set of borehole information and their resulting deviations. The findings have been compared to the influence of the calculated results on subsequent groundwater flow and transport modelling within the environmental risk assessment study.

3.1 Constructive Cross-Section net-based 3D Modelling

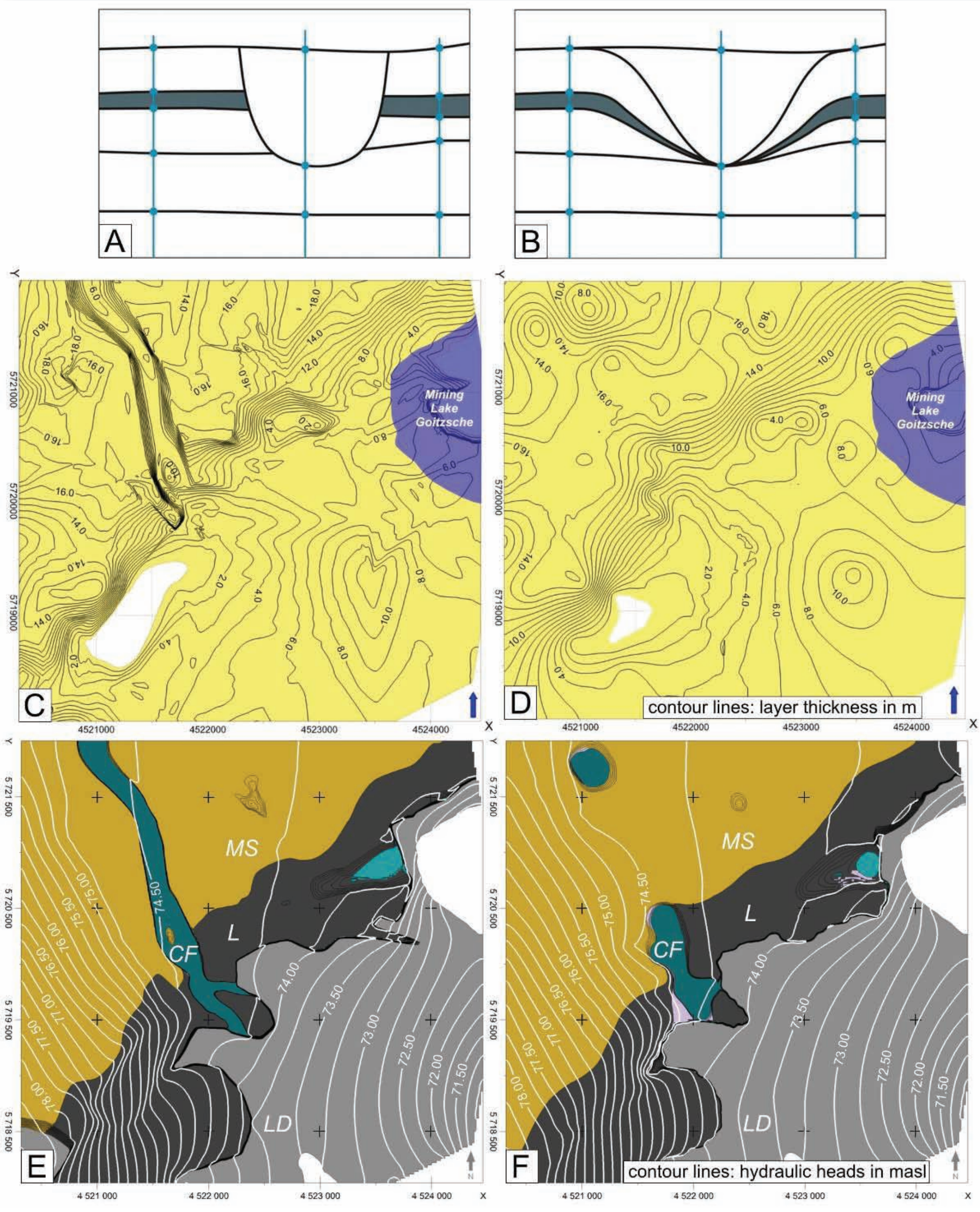
Constructive cross-section net-based interpolation approaches are of advantage when one has unbalanced regional coverage of drilling information describing complex and heterogeneous lithological subsurface conditions. The modelling process with GSI3D® is based on the creation of a series of intersecting user-defined cross-sections. The entire stacking order of all deposits in the study area has to be defined stratigraphically and sedimentologically, and a generalized vertical section has to be created. The lithostratigraphic classification of the sedimentary succession within a consistent regional stratigraphic framework is more helpful for groundwater modelling than a pure lithology-based approach derived from grain-size analysis. Most of the software tools allow the input of 2D geological mapping or surface information, especially from areas with a less dense borehole dataset. GSI3D® allows modelling of the distribution and geometry of sedimentary layers by knowledge-based control of the modeller, which is especially needed for the modelling of heterogeneous aquifer systems and / or artificially formed lithological units. GSI3D® is currently intended for use in the near-surface modelling of Quaternary sediments [15]. The software is used by the British Geological Survey on different mapping scales and in combination with GoCAD® for deep subsurface investigations and geological modelling [16], also for regional investigations of groundwater contaminated mega-sites [1, 3, 4].

3.2 Statistically-based 3D Modelling

Uneven and irregular distribution of geological drilling information is one of the major obstacles in regional subsurface modelling with automatically contoured distributions and thickness of layers in urban areas. In the case of known geology or sufficient coverage of drilling information, the statistically-based interpolation tools provide a less time-consuming modelling approach. As an example, EVS® (Environmental Visualization System; C Tech Development Corp.) provides true 3D volumetric modelling using 2D and 3D kriging algorithms with best fits of variograms to analyze and visualize geoscientific and environmental data. EVS®/MVS® allows the seamless integration with ArcView GIS, as well as with Modflow and MT3D. The 3D modelling of the subsurface geology with EVS® is based exclusively on selected drilling information and the geostatistical interpolation of the individual layer boundaries. In comparison with GSI3D® this procedure can lead to different results particularly in cut-and-fill structures of Quaternary sedimentary channel fills, as well as any artificial structures (e.g. subsurface dumps).

3.3 Resulting Differences in 3D Modelling Concepts

The application of different geological modelling tools and approaches lead often to distinct various outcomes in the resulting 3D geological models. Also the results and as a consequence the predictions based on numerical groundwater flow and transport models are strongly affected by the implemented geometry of aquifers and aquitards based on 3D geological models. In figure 3 the results of a constructive cross-section net-based interpolation (left side) are compared to the results of an automated



▲ Fig. 3: The results of different modelling methods. A & B depicts a schematic cross section through a channel situation (A - Cross-section net-based contouring; B - Automated statistically-based contouring). C & D show adequate examples of the calculated layer thickness of one stratigraphical unit (Upper Bitterfeld Mica Sand) originated from different modelling approaches (C - Cross-section net-based contouring using GSI3D software; D - Automated statistically-based contouring using MVS software). D & F depict the resulting hydrogeological situations at a horizontal cut plane at 60 m above sea level underlain by the geological situation (LD - Lignite Dump, CF - Channel Fill, L - Lignite, MS - Mica Sand). The contour lines represent the calculated hydraulic heads of the two hydrogeological models at this level. For hydrogeological modelling the same boundary conditions were used. So the differences in the modelling results are based completely on the modelled geological situation and the appropriate assignment of hydrogeological parameters.

statistically-based (right side) geological modelling approach at one selected stratigraphical layer (Upper Bitterfeld Mica Sand). Both models have been generated on the base of the same data input comprising the geological information of 114 interpreted borehole data sets. Also the resulting differences in the numerical groundwater models are displayed at a specified level (60 m above sea level) using the distribution of hydraulic heads.

4 High Resolution 3D Models for Groundwater Modelling

To support the investigations of an integrated environmental risk assessment of contaminated mega-sites on a local scale, the high-resolution 3D structural model was enlarged to about an area of 65 km² for the entire region and its downstream areas. The structural model was generated by combining point information of about 250 boreholes that were implemented in 62 cross sections in total. The model allows – beyond visualization purposes – volumetric calculations of partial or distinct sedimentary units, like the remaining lignite, which are relevant for assessing the natural attenuation potential and retardation processes.

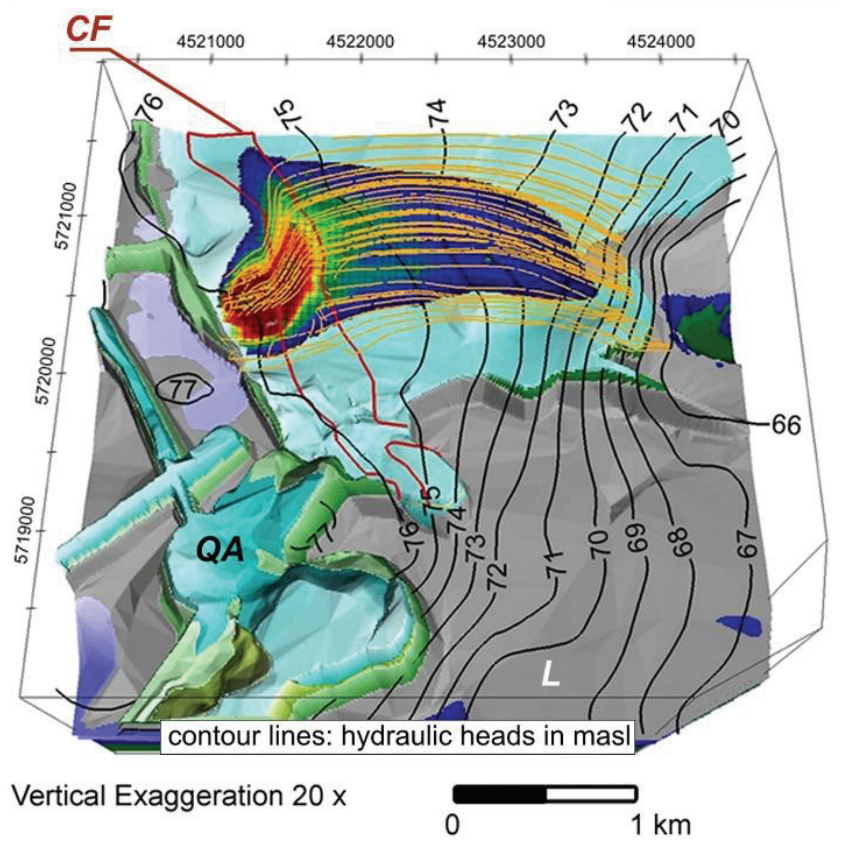
The digital data set of the true 3D structural geology was used with reference to their hydraulic characterization to build up subsequent flow and transport models. The numerical groundwater model was carried out with two objectives: a) Description of the hydrodynamic system and the pathline prediction of the post-mining time, and b) Predictive calculations of the changed hydraulic situation after the flooding of the Goitzsche Lake and the risen groundwater level of app. 8 m after August 2002.

The numerical model consists of two parts: A groundwater flow model and a transport model based on the flow model. The modelling systems Modflow®, ModPath®, and MT3D® with the Visual Modflow® 3.0 pre- and postprocessors were used. The general stratigraphic succession was clustered by condensing geological layers according to their hydrogeological properties: i.e., hydraulic conductivities and porosities. Most important for the numerical hydrogeological model is the completion of geological layers that are fading out. The 31 geological layers were reduced to 10 layers of the numerical groundwater model that have to be sustained across the whole model area. This part of the modelling process could only be done by working with a GIS data base. The hydraulic parameterization and structural setting was described by [1] and [3]. The main structure of the model is composed of Quaternary and Tertiary aquifers. Both distinct aquifers are separated by a clay layer and also by the lignite seam and are subdivided by several less conductive layers. Boundary conditions of the models were taken from the water levels measured in surrounding lakes and piezometers. The values were taken as mean groundwater levels and regionalized for the boundaries of the model area and held also in a GIS structure.

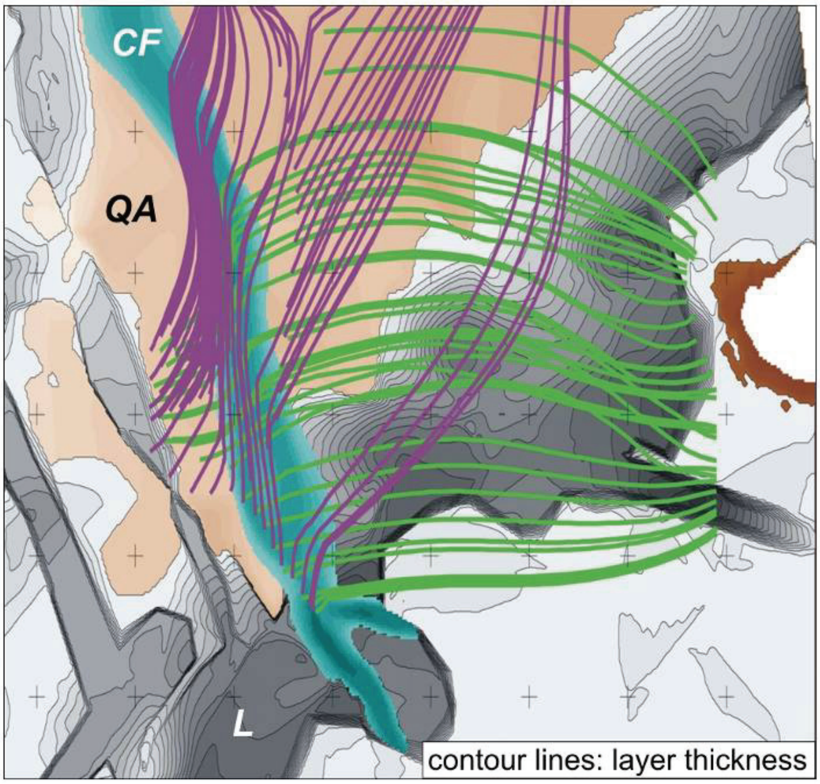
Based on the steady-state flow models, before and after the flooding event of August 2002, two transient transport models were run as study models for ideal, non-reactive tracers. To understand the local hydraulic and transport conditions, only diffusion and dispersion for the simulation were implemented because sufficient sorption and biological degradation data are still not available. Thus, the GIS data structure is important for several stages in numerical groundwater modelling: building the database of hydrogeological parameters and boundary conditions, calibration of the groundwater model by comparison of modelling results with measured data, and visualization of the scenario results. Therefore, it is necessary to realize a safe data exchange between the 3D geological modelling system and the GIS database that also allows further data processing to adapt the geological model to the structure of a numerical groundwater model.

The integration of simulated results from high-resolution 3D geology as well as from flow and transport modelling is depicted in figure 4 and 5. The 3D geology model in figure 4 shows the subsurface topography of the geological strata with removed mining dumps. In addition, the results of the transport model (non-reactive tracers) with a runtime of 30 years and the pathline distribution from flow modelling are integrated by GIS. It is shown very clearly that the predominant hydrogeological impact of the highly conductive Quaternary channel-fill structure is evident and leads to a NE ward spreading of the contaminant plume before August 2002 [4]. As indicated in figure 5 the influence of the same channel-fill is also evident after the flooding event and results in a more focused straight northward groundwater flow direction. The simulation of pathlines is of great importance for any source-receptor related environmental assessment study.

An example of calculated differences in hydrogeological modelling based on different geological modelling concepts is shown in figure 6 (A & B). As described in figure 3 the calculated hydraulic heads at a specified horizontal section at 70 m above sea level, as well as the virtual horizontal cross-section which is responsible to the differences in the hydrogeological models are

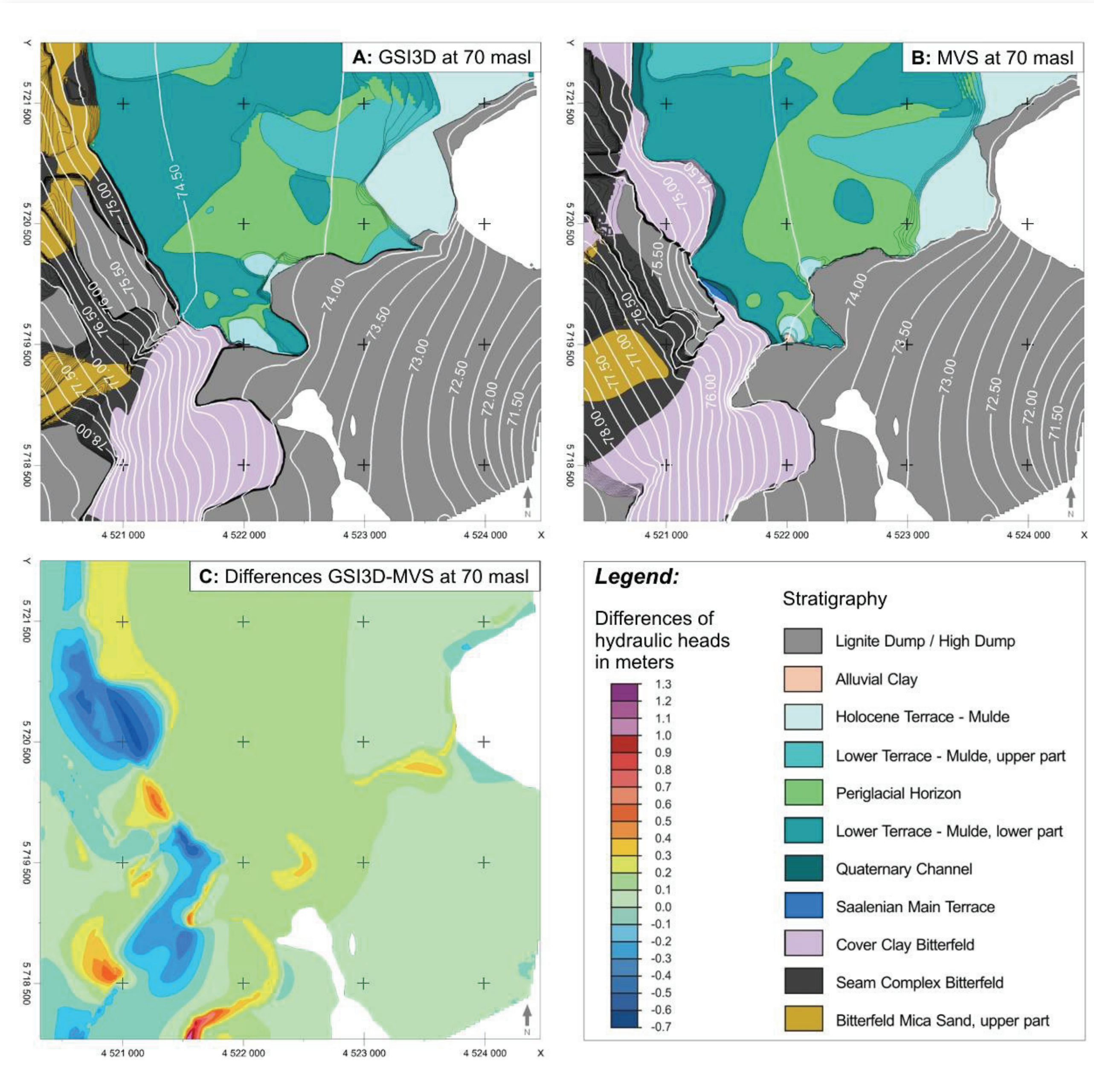


◄ Fig. 4: Groundwater flow system in a complex geological structure of the Bitterfeld area. The groundwater contours do not help in construction of the pathlines. Only groundwater flow and transport models (based on detailed geological models) yield numerically appropriate results that can be used for the assessment of contaminant spreading. The pathlines show the groundwater flow before the exceptional flooding event of the Mulde River in August 2002. Shortcuts: QA - Quaternary Aquifer, CF - Channel Fill, L – Lignite (modified after [4 and 5]).



◄ Fig. 5: Comparison of the calculated pathlines before (green) and after (violet) the exceptional flooding event of the Mulde River in August 2002. Shortcuts: QA - Quaternary Aquifer, CF - Channel Fill, L – Lignite (modified after [4]).

shown. Attention has to be paid to the calculated differences between the calculated hydraulic heads (C) that indicates the obvious relation between the modelled geological situation and the calculated hydraulic heads in the numerical groundwater model.



▲ Fig. 6: Comparison of the results of two numerical groundwater models based on geological models generated by different modelling concepts (A - Cross-section net-based expert driven contouring using GSI3D software; B - Automated geostatistically-based contouring using MVS software). A & B are depicting a horizontal sectional view at 70 m above sea level. The calculated hydraulic heads are displayed as white contour lines in combination with the geology at this elevation level. The most apparent differences are located in the western part of the investigated area caused by the spatial distribution of the "Cover clay" and residual mining land fills (compare to C). C depicts the calculated differences of situation A minus B respectively, of the hydraulic heads in meters.

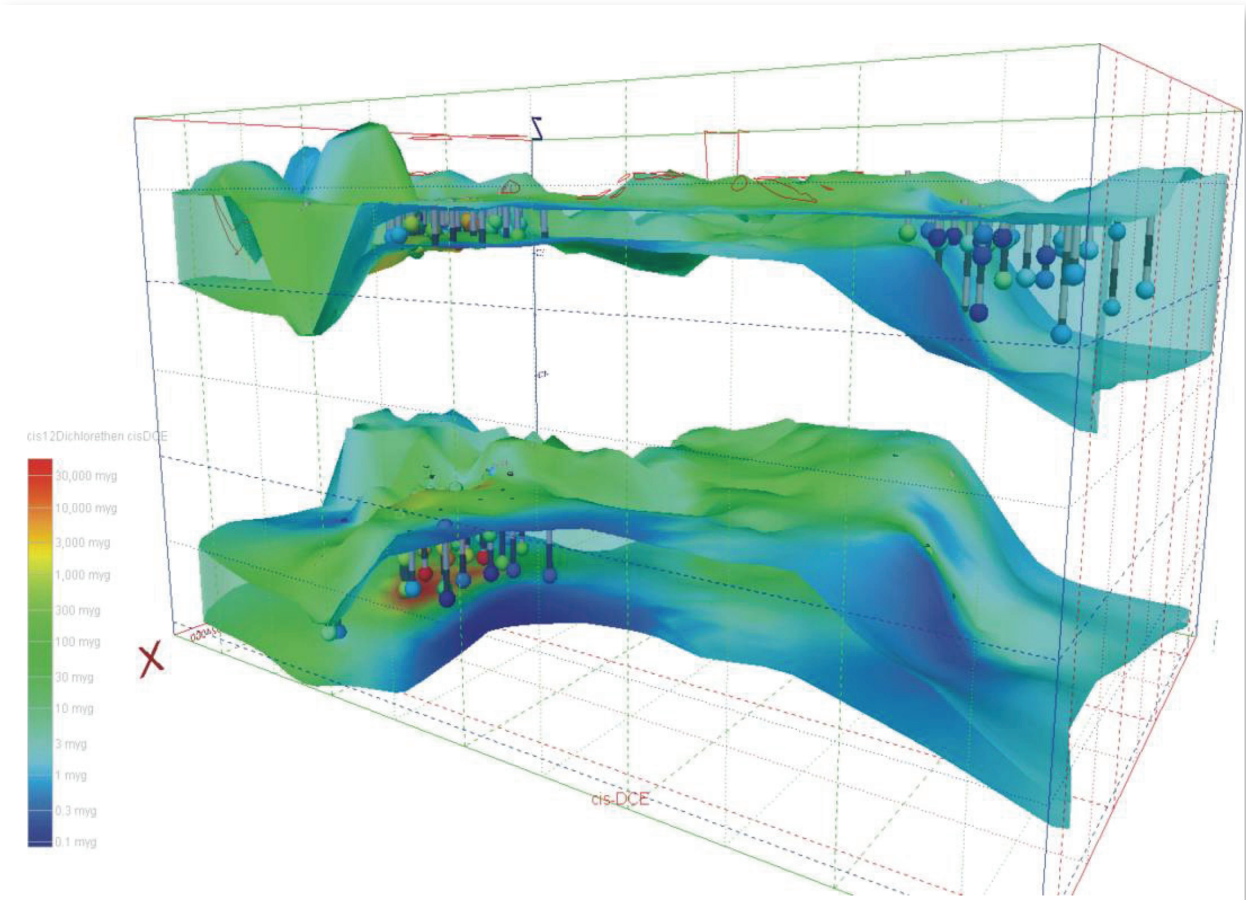
5 Integrated Multi-Source Data Visualisation and Prediction

To assess the complex environmental situation of the Bitterfeld-Wolfen area, a GIS-based spatial model was required that includes the heterogeneous aquifer setting in 3D in as much detail as possible. The subsurface information had to be available for a GIS-based assessment and predictive calculations correlated to surface information of potential receptors.

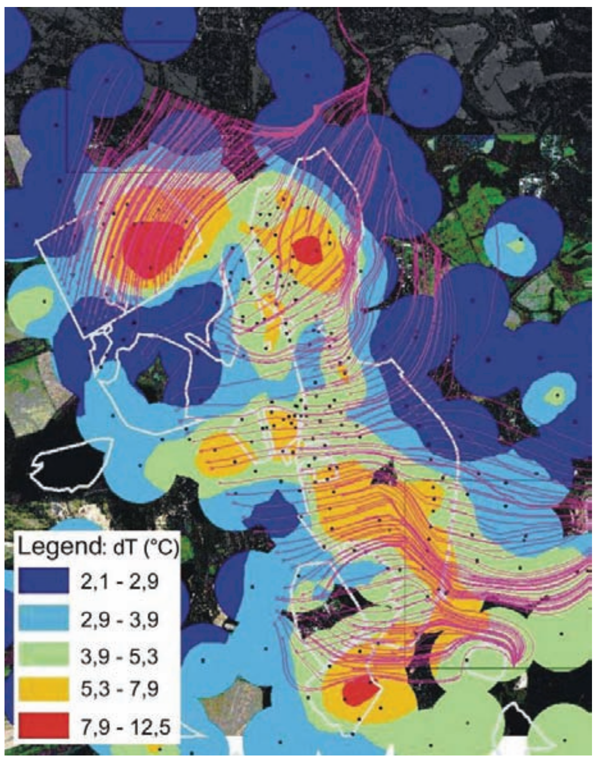
Therefore, the following major information modules have to be integrated into the "spatial model" on a local scale including the specific objectives and used modelling tools: a) 3D model of the subsurface geology, b) groundwater contaminants, c) hydrogeological data and d) land-use classification.

The GIS data management for all hydrogeological and hydrochemical data was done with ArcView® 3.x and ArcView® 8.x (ESRI). The spatial model includes point data such as borehole data (lithology/stratigraphy), hydrochemistry, contaminants monitoring data, etc. (Wycisk et al., 2007). The geological cross-sections, with their vertical 2D structure, were held in a special device for geological 3D models. The geological structures had to be held in a GIS database to obtain an interface to numerical groundwater modelling tools such as Feflow® or Modflow®. These data are stored in GRID formats in ArcView®.

The geological model also serves as a base for the interpolation of monitoring results and for the understanding of hydrochemical processes. According to the outlined aquifers the spatial distribution of substances can be sorted and interpolated separately. The result, shown in figure 7 [17], gives a more reliable 3D picture of distributed contaminats than a pure 2D interpolation just based on a geostatistical method.



▲ Fig. 7: Distribution of cis-Dichloroethene in the upper (Quaternary) aquifer and the lower (Tertiary) aquifer. The high contamination in the lower aquifer results from the higher density of this substance.



◄ Fig. 8: Combination of groundwater monitoring results of ΔT values in °C and pathlines of the groundwater flow model referring to the groundwater situation before August 2002 (after [17]). Measured and interpolated distribution and model results fit well to each other and show the preferred flowpaths in a certain model state.

The integration of groundwater flow direction in combination with regional monitoring data, e.g. differences in groundwater temperature (ΔT in °C), reflecting the exothermic reaction of natural attenuation processes within the aquifer, is shown in figure 8 [17]. The GIS-based data integration depicts the regional pathline distribution as well as a regional kriging interpolation of ΔT distribution. The calculated diameter around individual observation wells represents a 500 m distance. The high temperature areas show a high congruence to the stable and long lasting groundwater flow direction within the lower aquifer. In relation to the sources of the contamination and the calculated pathlines the fate of certain contaminants as well as biodegradation processes can therefore be visualized to a certain extent. Figure 8 shows on the one hand the conformity of monitoring results, interpolated by geostatistical methods with groundwater flow modelling results, here as pathlines, and on the other hand the transformation and reduction of substance downstream of the sources due to the groundwater flow situation before August 2002.

7 Conclusions

To generate an almost realistic scenario of hydraulic processes in heterogeneous aquifer systems, high resolution 3D subsurface models of the aquifer systems corresponding to the real world scenario of the geological subsurface setting enable qualitative and quantitative approaches in predictive modelling. The prediction of groundwater flow as well as regionalisation of contaminants is challenging in most of the Quaternary sediments, comprising a high and to some extent preferential flow direction. The 3D geological model also serves as a future–oriented data base, and provides a consistent data format, as well as an updating capability to improve the predictability of model deliverables in environmental risk assessment. The new approach of digital 3D geo-data management produces a capable upgrading system and forward looking decision support tool. ■

Acknowledgements

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2B.11 Modelling of Solute Transport in Complex Groundwater Systems with Transfer Functions

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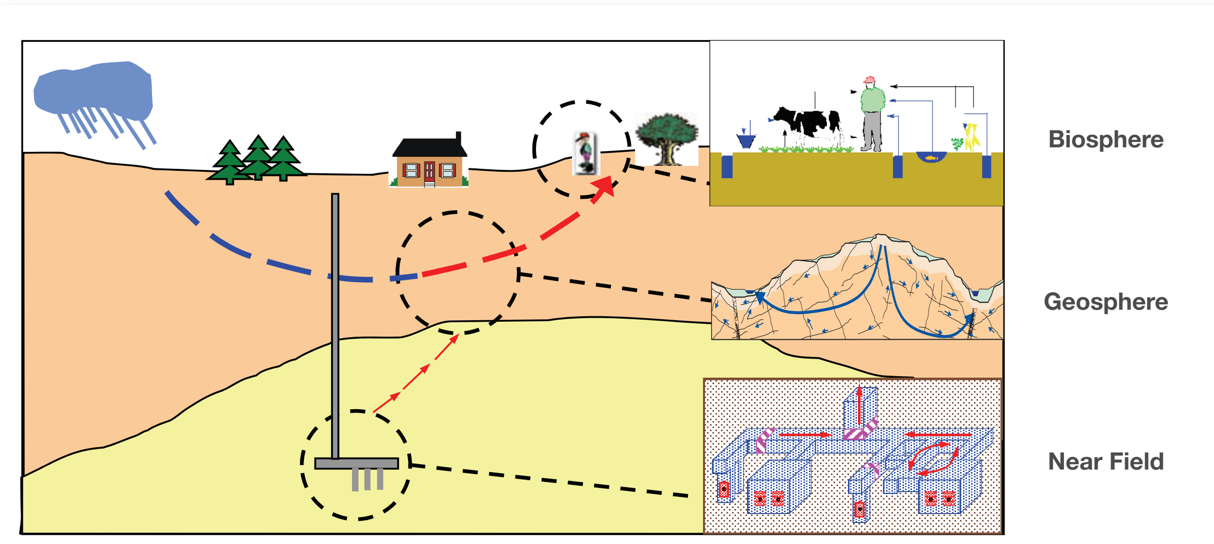
Abstract
Dissolved contaminants are transported from a release point to the groundwater flow system to one or more exfiltration points. In general, such a transport is calculated by application of multi-dimensional computer codes, which means high computational effort concerning parameter variations or uncertainty analyses. By adopting the transfer function method a simplified calculation of such a transport is possible. This method is often used and accepted with respect to the solution of linear differential equations, e.g. for the calculation of heat propagation. In repository systems transfer functions can be applied if the propagation of contaminants has to be computed in a complex heterogeneous overburden with several exfiltration points, which can not be modelled by other kinds of simplification, e.g. a one-dimensional transport. The advantage of transfer functions is especially pronounced if the groundwater flow changes in time by interference with the repository system. Compared to three-dimensional overburden calculations computer run times are reduced, which allows to perform parameter variations related to processes in the repository.

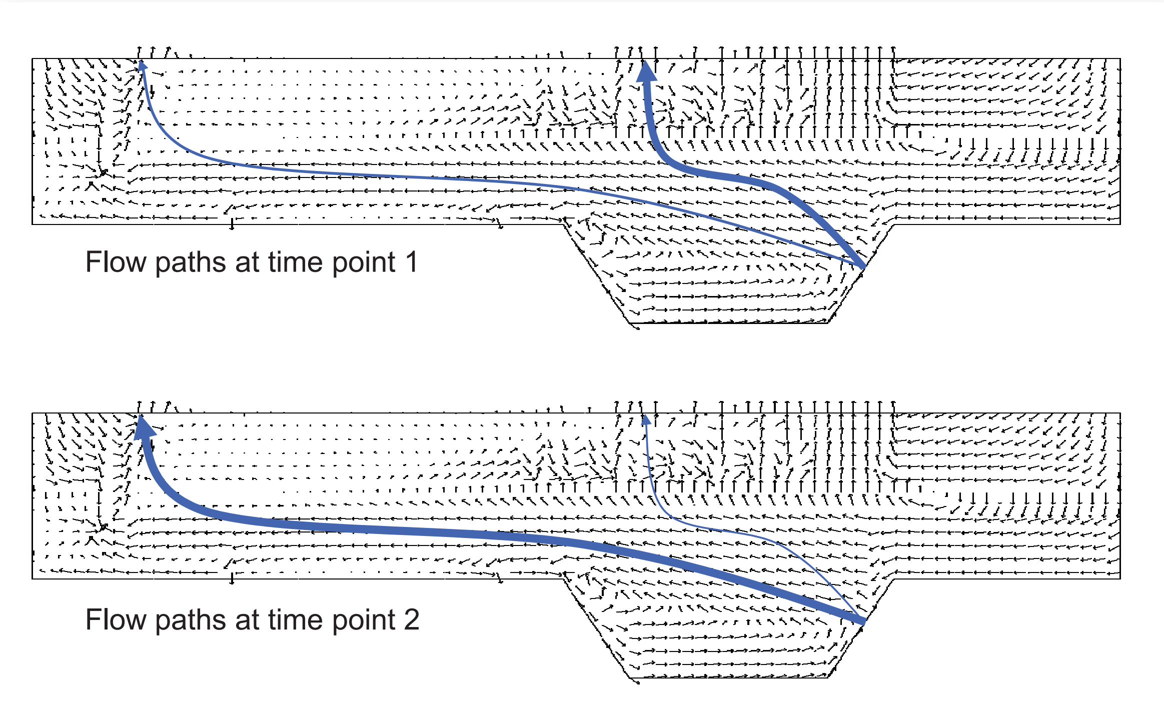
1 Introduction

Long-term storage of contaminants (including radioactive waste) in Germany and many other countries is often planned to be deep underground. A repository system for such long-term storage consists of the near field surrounding the waste deep underground, the far field representing the transport regime in the host rock (if clay or crystalline) and in the overburden, and the biosphere, cf. Figure 1. The following discussion is restricted to transport of contaminants in the overburden.

In contrast to the simplified sketch in Figure 1, the flow scheme of groundwater in the overburden may be complex and time dependent. A particle that is released from the near field to the groundwater can be transported along various flow paths, cf. Figure 2. The relevance of the different flow paths may change with time, as indicated by the varying thicknesses of the blue



▲ Fig. 1: Scheme of repository system



▲ Fig. 2: Example of complex groundwater flow

lines in the two parts of the Figure. Generally, the contaminant transport in such a complex, time-dependent system requires two- or three-dimensional calculations for each considered variant of a reference system. As these calculations consume a lot of computing time, they can not be applied to sets of parameter variations or even uncertainty analyses, e.g. in Monte-Carlo-simulations. For parameter variations in the near field, the proposed method circumvents these disadvantages by application of a simple calculation algorithm based on the transfer functions. The limits of the method are discussed.

The main aims of the new method are thus a reduction of computing effort (reduced run times) and a provision of a suitable tool for application in parameter variations including Monte-Carlo-simulations for the near field.

2 Method of Transfer Functions

2.1 Transfer Function for a Delta Impulse

Generally, the flow and transport system in the overburden can be taken as a black box that transforms an input signal, the contaminant flow rate at the entry point into the flow system $Q_T^{in}(t)$, into the corresponding system response, e.g. the contaminant flow rate or contaminant concentration at one or more exfiltration points. The most informative input signal is the Dirac impulse. It is mathematically described by the delta function δ(x) which is defined by the integral

$$\int_{-\infty}^{+\infty} \delta(\xi - a)\, d\xi = 1 \quad \text{with} \quad \delta(x-a) = 0 \quad for \quad x \neq a \qquad (1)$$

Assuming that a certain amount of a tracer M_T enters the overburden at time $t = t_0$ instantaneously, the referring tracer flow $Q_T^{in}(t)$ can be described by a δ-function:

$$Q_\delta^{in}(t) = M_T\ \delta(t - t_0) \tag{2}$$

The concentration $c_\delta^{in}(t)$ at the entry point thus increases at the time t_0 for an infinitesimal length of time (Figure 3, left). The system response to this input signal is the time dependent concentration $c_\delta^{out}(t)$ at the point of exfiltration $\vec{r} = \vec{r}_{ex}$ (Figure 3, right). For M_T =1 and a steady state flow system

$$f_\delta(\vec{r}_{ex}, t - t_0) = c_\delta^{out}(t) \tag{3}$$

where $f_\delta(\vec{r}_{ex}, t - t_0)$ is called the transfer function for a delta impulse at time t_0.

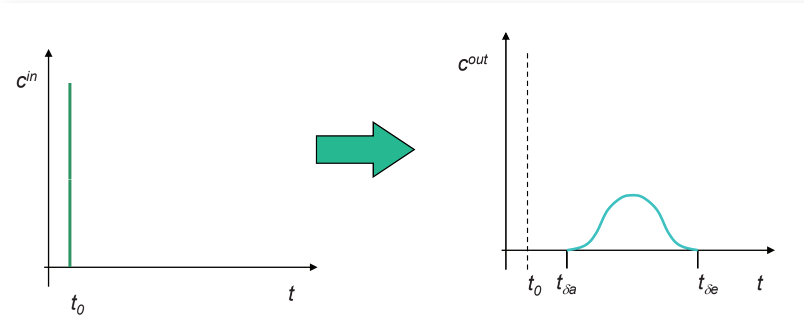


Fig. 3: System response to a delta impulse.

2.2 Transfer Functions for Steady State Geosphere Flow Field

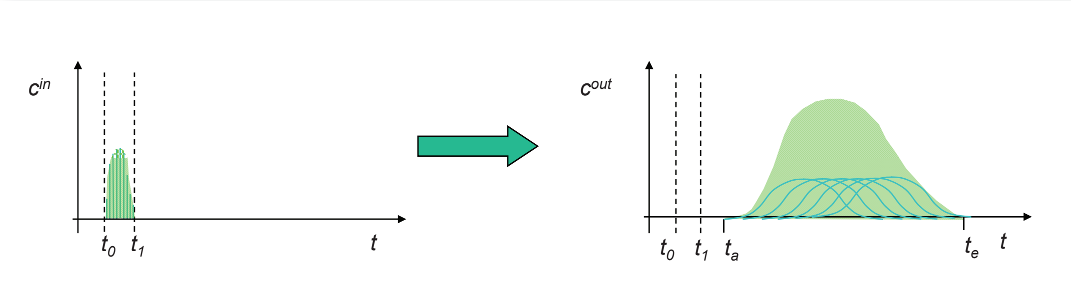
An arbitrary continuous inflow function $Q_a^{in}(t)$ can be written as an integral over time of the product of the tracer inflow value times the Dirac delta at time τ^0 :

$$Q_a^{in}(t) = \int_{\tau=0}^{+\infty} Q_\delta^{in}(\tau)\,\delta(t-\tau)\,d\tau \tag{4}$$

System response to this function can simply be calculated by superposing the response functions $c_\delta^{out}(t)$ due to the linearity of the differential flow and transport equations:

$$c^{out}(\vec{r}_{ex}, t) = \int_{\tau=0}^{t} Q_\delta^{in}(\tau)\, f_\delta(\vec{r}_{ex}, t-\tau)\, d\tau \tag{5}$$

Phase shift of the transfer functions f_δ relative to t_0 as well as the shape of these functions are independent of t_0 provided that steady-state conditions referring to groundwater flow and transport parameters prevail. In other words, in a steady-state flow and transport system just one transfer function has to be found in order to calculate the system response to any kind of tracer inflow function, cf. Figure 4.



▲ Fig. 4: Superposition of the system response as sum of system responses to delta impulses

Assuming a constant tracer inflow Q_c^{in} equation (5) can be transformed into

$$c^{out}(\vec{r}_{ex},t) = Q_c^{in} \int^{t} f_\delta(\vec{r}_{ex},t-\tau)\,d\tau \quad (6)$$

In this case the integral represents a transfer function that calculates the system response from the tracer inflow Q_c^{in}. This is actually less convenient to handle than a function that uses the inflow concentration of the tracer. Since tracer inflow can be interpreted as the product of the solution inflow Q times tracer concentration c

$$Q_T = c \cdot Q \quad (7)$$

equation (6) can be written as

$$c^{out}(\vec{r}_{ex},t) = c^{in} \int_{\tau=0}^{t} Q\, f_\delta(\vec{r}_{ex},t-\tau)\,d\tau \quad (8)$$

If the tracer concentration c^{in} equals "1" the integral in equation (8) defines a transfer function $f_c(\vec{r}_{ex},t)$ for a constant inflow concentration beginning at $t = 0$

$$f_c(\vec{r}_{ex},t) = \int_{\tau=0}^{t} Q\, f_\delta(\vec{r}_{ex},t-\tau)\,d\tau \quad (9)$$

In the case of a temporary tracer inflow during a time interval $[t_0, t_1]$ the lower limit of the integral must be set to t_0 instead of 0, and the upper limit to $\min(t,t_1)$. Note, that the solution inflow Q is constant here but must not be necessarily so.

2.3 Transient Flow and time-dependent Transport Parameters

In case of a transient flow field and/or transient transport parameters the transfer function f_δ is not only a function of the difference $t\text{-}t_0$ but also of t_0. The method of transfer functions thus requires a different transfer function for each point in time, i.e. an extended type of transfer function $f(t-t_0,t_0)$. An exact application of this method is out of question. However, the exact solution can be approximated by slicing time into periods $[t_i, t_{i+1}]$. Here, a set of transfer functions f_{R_i} that are valid only for the related periods of time will be used. Each function f_{R_i} yields the system response to a constant unity input concentration during the interval $[t_i, t_{i+1}]$.

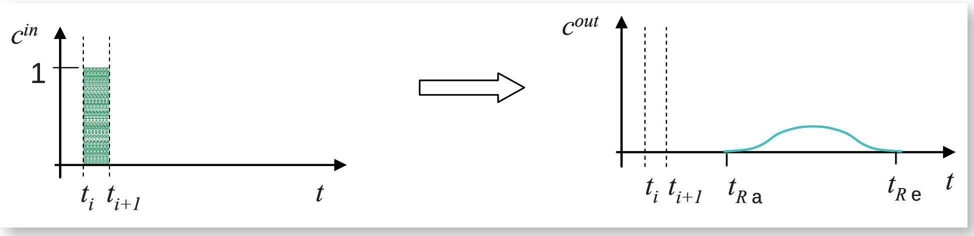


Fig. 5: System response to a constant unity inflow.

In principle, equation (8) can be used in this situation. A time-dependent inflow of solute $Q(t)$ is not in contradiction to this equation, and the transfer function $f_\delta(\vec{r}_{ex}, t-\tau)$ in equation (9) is to be substituted by the three-parameter approach $f_\delta(\vec{r}_{ex}, t, \tau)$[1]. The new transfer functions f_{R_i} then read

$$f_{R_i}(\vec{r}_{ex}, t) = \int_{\tau=t_i}^{t_{i+1}} Q(\tau)\, f_\delta(\vec{r}_{ex}, t, \tau)\, d\tau \qquad (10)$$

However, the previous section has shown that the inflow concentration must be constant in order to separate it from the transfer function i.e. the integral over time. A substitute concentration $\bar{c}_i^{in}$ (cf. Figure 6) can be calculated, though, obeying the condition that the tracer mass entering the system must not be changed:

$$\bar{c}_i^{in} = \frac{\int_{t_i}^{t_{i+1}} c^{in}(\tau) Q(\tau) d\tau}{\int_{t_i}^{t_{i+1}} Q(\tau) d\tau} \qquad (11)$$

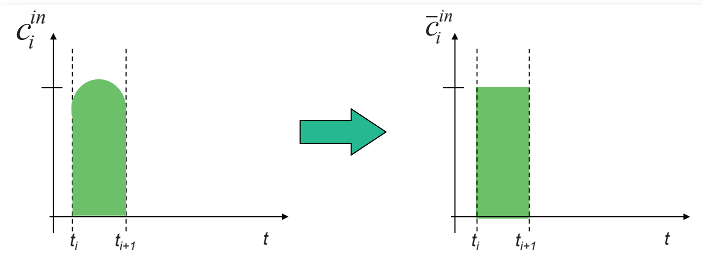


Fig. 6: Approximation of time dependent concentration by piecewise constant concentration

The concentration at the exfiltration point caused by the mass inflow during the m relevant time periods can then be added analogously to equation (5) to get the required approximation $\tilde{c}^{out}$ of the exact solution

$$\tilde{c}^{out}(\vec{r}_{ex}, t) = \sum_{k=1}^{m} \bar{c}_i^{in}\; f_{R_i}(\vec{r}_{ex}, t) \qquad (12)$$

The accuracy of this approximation apparently increases with the number of intervals into which the input signal is divided.

[1]The parameter t-τ is replaced for simplicity by t.

2.4 Radioactive Decay

Contrary to a tracer a radionuclide decreases in mass over time. The mass of a radionuclide *n* contained in the host rock is therefore a function of residence time. A problem arises from the fact that the amount of substance observed at time *t* at the exfiltration point $\vec{r}_{ex}$ consists of fractions of different transit times from inflow to exfiltration point. In order to take the effect of decay exactly into account the transit time of each fraction must be known. Note, that with decay of a parent nuclide the mass of the referring daughter nuclide increases.

A correct age determination for the nuclide is possible if the input signal is described by a sum of delta functions. In this case transit time equals t-τ and thus decay can be calculated correctly for each fraction as

$$c_n^{out}\left(\vec{r}_{ex},t\right)=\int_{t_0}^{t} Q_n^{in}\left(\tau\right) f_\delta\left(\vec{r}_{ex},t,\tau\right) e^{-\lambda_n(t-\tau)}\, d\tau \qquad (13)$$

where $Q_n^{in}(\tau)$ equals the inflow rate of radionuclide *n*. Equation (13) is valid if just one radio-nuclide is to be considered. In case of decay chains Bateman's formula must be used either in terms of particle concentration

$$c_n\left(t\right)=\sum_{i=1}^{i=n}\left[c_i\left(0\right)\left(\prod_{j=1}^{j=n-1}\lambda_j\right)\sum_{j=1}^{j=n}\frac{e^{-\lambda_j t}}{\prod_{\substack{p=i\\p\neq j}}^{p=n}\left(\lambda_p-\lambda_j\right)}\right] \text{ [mol m}^{-3}\text{]} \qquad (14)$$

or in terms of activity concentration

$$A_n\left(t\right)=\sum_{i=1}^{i=n}\left[\frac{A_i\left(0\right)}{\lambda_i}\left(\prod_{j=1}^{j=n}\lambda_j\right)\sum_{j=1}^{j=n}\frac{e^{-\lambda_j t}}{\prod_{\substack{p=i\\p\neq j}}^{p=n}\left(\lambda_p-\lambda_j\right)}\right] \text{ [Bq m}^{-3}\text{]} \qquad (15)$$

which – combining equations (7), (10) and (14) – yields

$$c_n^{out}\left(\vec{r}_{ex},t\right)=\int_{t_0}^{t} Q\left(\tau\right)\cdot f_\delta\left(\vec{r}_{ex},t,\tau\right)\cdot\sum_{i=1}^{n}\left[\left(\prod_{j=1}^{n-1}\lambda_j\right)\sum_{j=1}^{n}\frac{c_i^{in}\left(\tau\right)e^{-\lambda_j(t-\tau)}}{\prod_{\substack{p=i\\p\neq j}}^{n}\left(\lambda_p-\lambda_j\right)}\right]d\tau \qquad (16)$$

where $c_i^{in}(t)$ and $c_n^{out}(\vec{r}_{ex},t)$ are particle concentrations. However, this equation holds only if all nuclides of a decay chain are migrating at the same velocity, i.e. that sorption affects all nuclides in the same way if at all.

Using approximation (12), though, the transit time for each infiltration period [t_i, t_{i+1}] can only be narrowed down to a length of time between t - t_{i+1} and t - t_i. The concentration of a single nuclide is therefore approximately

$$c_{Rn}^{out}\left(\vec{r}_{ex},t\right)=\sum_{i}\bar{c}_{ni}^{in}\cdot f_{Ri}\left(\vec{r}_{ex},t\right)\cdot e^{-\lambda_n\left(t-\tau_i\right)} \tag{17}$$

where τ_i is a fixed but arbitrary point within the interval $[t_i, t_{i+1}]$ at which decay is assumed to begin. In case of decay chains the Bateman formula has to be inserted analogously to equation (16).

3 Application of Transfer Functions

3.1 Transfer Functions

From three-dimensional transport calculations sets of transfer functions have been derived to illustrate the method. Figure 7 shows the results for two different exfiltration points: in the upper part of the figure results are shown for a path with early exfiltration, in the lower part for a path with later exfiltration.

For both exfiltration points a set of transfer functions is shown, each representing a transfer from a specific time interval at the entry point. For instance, the first transfer function (i.e. the first curve starting at about 500 years) represents the transport of tracer entering the flow system during the time interval [200 yr, 500 yr]. Transport starting after about 3000 years from the entry point does not show up in a transfer function within the scale of the upper figure, while in the lower figure transfer functions occur for all time intervals at the entry point. Thus the figure demonstrates the effect of time dependent flow regime.

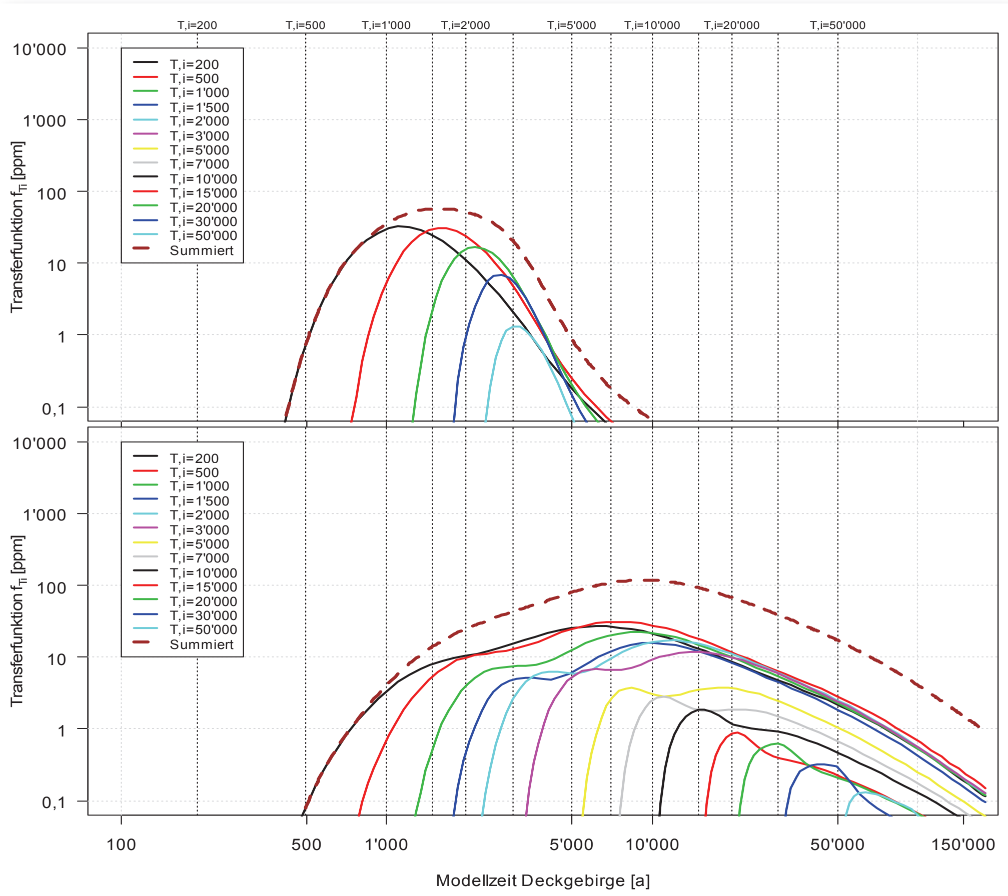


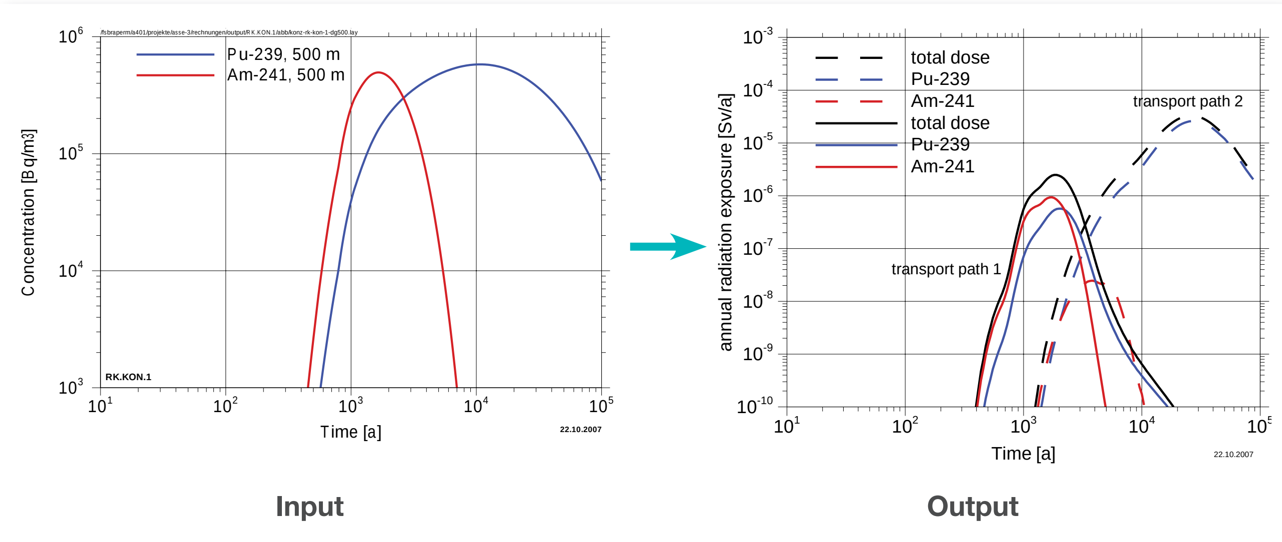
Fig. 7: Examples of transfer functions for time-dependent geosphere flow at two exfiltration points (upper and lower part of the Figure)

3.2 Application to Radionuclide Transport

The adequate tool to calculate transfer functions for a concrete problem is a detailed numerical hydrogeological model of the flow and transport system in the geosphere. Such a model cannot handle δ-functions as an input signal, but only input functions of finite length because solutions are discrete in time as well as in space. Since the response functions must cover time periods in the order of 100,000 years or more only a limited number of transfer functions can be calculated with a reasonable effort and the infiltration periods $[t_i, t_{i+1}]$ inevitably are relatively long.

To illustrate the application of transfer functions, in the following figure an example is given for the calculation of annual radiation exposure resulting from a release of radionuclides from a repository and subsequent transport within the geosphere to the biosphere. The transport in the geosphere is calculated by means of transfer functions shown in Figure 7. The result is given in Figure 8.

At the entry point to the geosphere the concentrations of two radionuclides are given as a function of time, for the short-lived ^{241}Am and the long-lived ^{239}Pu, both with about the same maximum values. These concentrations are used as input values for the transfer functions according to the previous Figure 7. Application of the transfer functions and multiplication by a conversion factor yields the curves of the radiation exposures as shown in Figure 8 (right side). For the transport path 1 both radionuclides dominate the maximum of the radiation exposure to about the same amount, while for the transport path 2 mainly the long-lived radionuclide ^{239}Pu dominates the maximum. This behaviour reflects the importance of time dependence of the transport paths in the geosphere: at paths with early exfiltration maxima an important contribution of short-lived radionuclides occurs, while at late exfiltration maxima mainly long-lived radionuclides are of importance.



▲ Fig. 8: Application of transfer functions to calculation of radiation exposure

4 Uncertainties

When applied to real groundwater systems, most of the uncertainties inherent to the method of transfer functions can only be analysed with great computational effort. They are thus described here qualitatively only. In the following, estimations are described concerning the uncertainty due to inflow functions of finite length in combination with decay of a single nuclide as well as decay in a decay chain. Additionally, the calculated concentrations at the exfiltration point(s) are compared with the results from a detailed transport calculation with the numerical model that provides the transfer functions.

4.1 Tracer Transport

The number of transfer functions is limited due to computational effort. Thus, the time interval of the input signal related to each of the transfer functions can be of considerable length. Nevertheless, these time intervals should be kept as short as possible in order to minimise errors from the transient flow and transport behaviour of the system. The interval length can be adjusted if the significance of the transient effects varies over time. The error arising from the transient behaviour can be assessed a posteriori based on concrete problems by comparison of the results with detailed transport calculations using the hydrogeological model. Such a comparison will be shown in the next chapter for a long-lived radionuclide instead of a stable tracer.

4.2 Transport including Radioactive Decay

4.2.1 Uncertainties for a Single Transfer Function

The uncertainty concerning the transit time of a tracer, which enters the host rock sometime during the time period $t_i \leq t \leq t_{i+1}$ cannot exceed the length of this interval. Nevertheless, the transit time determines the progress of radioactive decay and a certain error in such calculations cannot be avoided.

This error can only be determined for a concrete situation and with considerable effort, thus upper and lower boundary values of the error have been estimated. If the complete nuclide mass entering the host rock during the interval $t_i \leq t \leq t_{i+1}$ is attributed to one point of time the decay can be calculated exactly. Maximum decay happens if the mass enters at ti and minimum decay at t_{i+1}. The corresponding extreme values of the concentration are

$$c_j\left(\vec{r}_{ex},t\right) = c_R^{out}\left(\vec{r}_{ex},t\right)\cdot e^{-\lambda\left(t-t_j\right)} \text{ with } j=i,i+1 \qquad (18)$$

where $c_R^{out}(\vec{r}_{ex},t)$ stands for the concentration at $\vec{r}_{ex}$ and time t without radioactive decay. Maximum error $\Delta c(\vec{r}_{ex},t)$ is then the difference of the two extreme concentrations

$$\Delta c(\vec{r}_{ex},t) = c_{i+1}(\vec{r}_{ex},t) - c_i(\vec{r}_{ex},t) = c_R^{out}\left(\vec{r}_{ex},t\right)e^{-\lambda\cdot(t-t_{i+1})}\cdot\left(1-e^{-\lambda\cdot(t_{i+1}-t_i)}\right) \qquad (19)$$

Equation (19) provides the upper limit of an absolute error. It is dependent of time as well as of concentration $c_R^{out}(\vec{r}_{ex},t)$ which makes it dependent of a specific problem and does therefore not allow assessing the accuracy of the method of transfer functions. However, a meaningful upper limit for a relative error F_{rel} can be defined by

$$F_{rel} = \frac{\Delta c(t)}{c_i(t)} = e^{\lambda(t_{i+1}-t_i)} - 1 \qquad (20)$$

This upper limit of the error becomes large when the time interval t_{i+1}-t_i is large compared to the half-life of a nuclide. If the interval length equals the half-life, the maximum relative error amounts to 100 %. For ^{239}Pu and an interval length of 5000 years for instance the maximum resulting error is 16 %.

The time τ_i when radioactive decay begins (within the interval [t_i, t_{i+1}]) can be described in dimensionless form:

$$\theta = \frac{\tau_i - t_i}{t_{i+1} - t_i} \qquad (21)$$

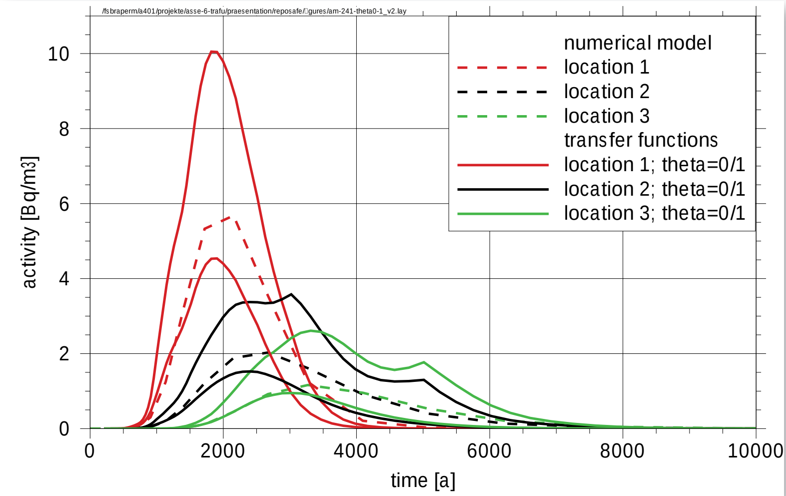


Fig. 9: Calculated concentration of ^{241}Am using a detailed numerical model and transfer functions with θ = 1 as well as θ = 0.

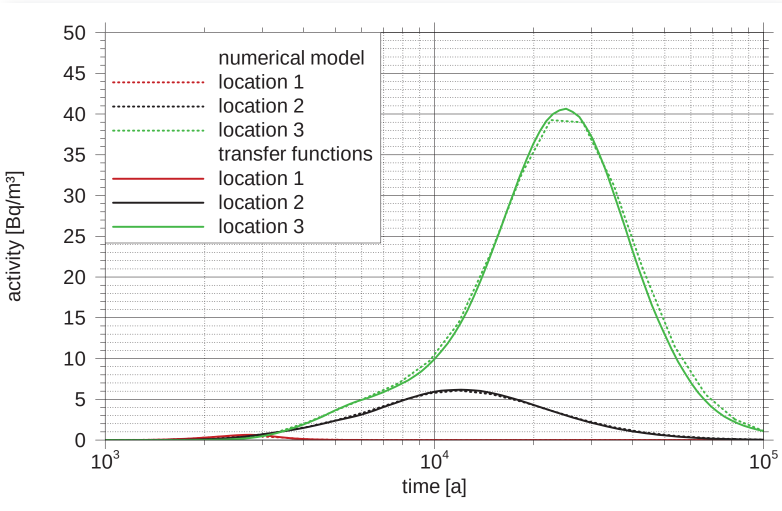


Fig. 10: Calculated concentration of ^{239}Pu using a detailed numerical model and transfer functions with θ = 1.

For a single nuclide the method of transfer functions is always conservative – even using input signals of finite length – if θ = 1 is chosen because θ = 1 means that the radioactive decay begins only at the end of the time interval. On the other hand, for an eventual daughter nuclide θ = 0 would be conservative. For a conservative approach, the choice of θ must represent the most unfavourable case. This can only be evaluated for a specific case and only a posteriori.

An example for an a posteriori analysis is given with respect to two single nuclides (no decay chain), one with a relatively short and one with a long half-life, respectively. ^{241}Am with a half-life of 432 years and ^{239}Pu with a half-life of 24,110 years are chosen for this purpose. 13 transfer functions with increasing interval length are used to cover a model time of 100,000 years (Figure 7). A value of θ = 1 is adopted for the calculations. Figure 9 shows the results of the detailed hydromechanical model as well as results with the method of transfer functions for ^{241}Am at three exfiltration points ("locations"). Figure 10 shows the same for ^{239}Pu.

The concentrations at the exfiltration points are over- or underestimated with the method of transfer functions in comparison to the hydromechanical model. A value of θ = 1 overestimates the concentration, a value of θ = 0 underestimates. As expected, the results of the detailed model are more or less bounded by the two sets of results with transfer functions (for the short-lived nuclide) or are of about the same magnitude (for the long-lived nuclide).

4.2.2 Uncertainties for the Sum of Transfer Functions

Responses to the input signals can simply be added as described in section 2.2; the referring absolute errors can be added as well. The estimation of error for a single transfer function is too complex to derive a general formulation (see preceding section), and the same holds for the total error. The total error can be evaluated and analysed for a specific problem, but not predicted.

5 Summary and Conclusion

The method of transfer functions represents a simplified form of a numerical calculation of radionuclide migration in the host rock. In a concrete case the transfer functions for a tracer, a non-decaying substance, have to be calculated with a detailed numerical model. Further calculations with the detailed model for any inflow of radionuclides can be substituted using these transfer functions. Computing time is thereby reduced to small values.

The application of transfer functions inevitably includes uncertainties. These stem mainly from the fact that the radionuclide concentration at the inflow to the system will not be constant during the time periods [t_i, t_{i+1}] and from the fact that the transport time is considered in a discretized manner only for the calculation of radioactive decay. Theoretically, an infinite number of δ-impulses is necessary to get an exact solution of this type of problem. Using the best numerical approximation to those δ-impulses can result in a very large number of transfer functions that makes the method unattractive. Usage of a limited number of transfer functions each representative for inflow during a certain period of time is therefore proposed instead.

Transfer functions can be successfully applied if the following two conditions are fulfilled:

- The inflow of water at the entry point is of similar size in both calculations, i.e. in the calculation to derive the transfer functions and in the case to which the transfer functions are applied, and

- If sorption processes occur along the transport paths, they must be the same for all radionuclides belonging to the same decay chain.

It must be noted, however, that the transfer functions have to be calculated separately for species (or decay chains) with different sorption parameters. The application of transfer functions has not yet been demonstrated for dissolved contaminant transport with sorption. The authors believe that the method of transfer functions can not be extended in a feasible manner to the transport of decay chains with different sorption properties of its member radionuclides. ■

2B.12 Impacts of Extreme Climatic Scenarios on Groundwater Flow and Radionuclide Transport in the Overburden of a Nuclear Waste Repository in a Salt Dome

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Abstract:
A generic, two-dimensional groundwater flow model is used to simulate the recent groundwater velocity fields and salt concentrations in the overburden of a potential nuclear waste repository in a salt dome. Thereupon, groundwater flow simulations are conducted for the three scenarios (i) constant climatic conditions, (ii) inundation of the area with sea water and (iii) permafrost conditions due a nearby inland ice-sheet. These possible extreme climate scenarios are deduced from the geological past of the area. Based on the stationary velocity fields, transport simulations for selected radionuclides were and are currently performed. The aim of this work is to determine the distribution and the contamination of groundwater and spring water after a potential incident within the repository leading to release of radioactively contaminated brine into the overburden. The programs d³f (distributed density driven flow) and r³t (radionuclides, reaction, retardation, and transport) are used for the numerical modelling.

1 Preface

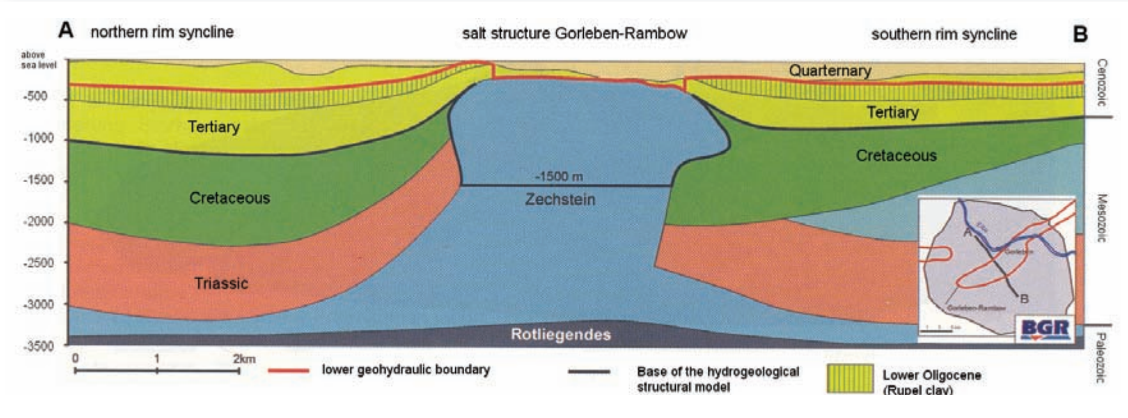
Performing long term safety assessments, one has to consider geological timescales. In Germany, the Committee on a Site Selection Procedure for Repository Sites (AKEnd) postulated a time frame of 10^6 years for the safe isolation of radioactive waste in rock formations [1]. It is obvious that drastic climatic changes can occur during that time. Therefore, numerical modelling is used to analyse groundwater flow and transport of radionuclides after a hypothetical incident within the repository leading to release of radioactively contaminated brine into the overburden.

2 Geology and Hydrogeology

The Gorleben salt dome is located at the border between Lower Saxony, Saxony-Anhalt and Brandenburg in northern Germany. The investigation area selected by the Federal Institute for Geosciences and Natural Resources (BGR) comprises about 475 km² with heights between 13 m and 76 m above sea level [2, 3]. It is located within the glacial valley of the river Elbe. The morphology can be divided into three major parts: the glacial and Holocene lowland of the river Elbe, the Weichsel glacial valley and the Weichsel and Saale Geest.

2.1 Geology

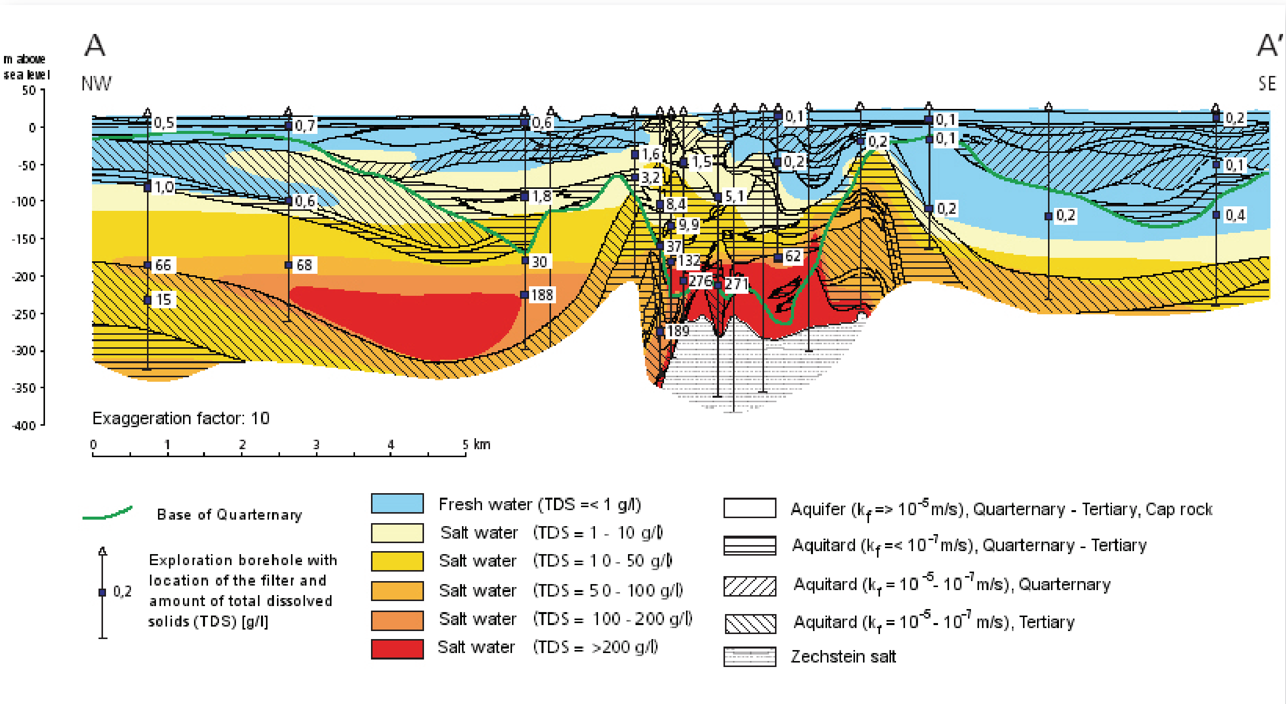
The main geological structure is the Permian Gorleben-Rambow salt dome (Figure 1). From the north and the south, it is accompanied by lateral rim synclines, which are filled with Mesozoic sediments of up to 2,500 m thickness. Notably the northwestern rim syncline can be traced even in the Cenozoic sediments, which gives it a great importance for the hydrogeological system. The Tertiary and Quarternary sediments fill up the rim synclines with up to 1,100 m sediment thickness. Additionally, these sediments are covering the salt dome, where Mesozoic sediments have been eroded almost completely during the Elster cold stage when the Gorleben glacial channel was formed.



▲ Fig. 1: Simplified geologic cross section with location of the model base (red line) after Bornemann (1991) [modified after 2]

2.1 Hydrogeology

The hydrogeological system is a composite of several Cenozoic aquifers and aquitards amounting to about 430 m thickness altogether [4]. Aquifers consist of sands and gravel sands. Intermediate aquitards are composed of boulder marl and limnic clay beds. The layers can be summarised to an upper and a lower aquifer disrupted by an aquitard. The aquitard is penetrated by hydraulic windows, which are the potential locations for salt water upflow. Only one hydraulic window could definitely be located by drillings south of the salt dome, while the other one is suspected to occur above the salt dome or north of the northwestern rim syncline [3]. These two structural elements, the hydraulic window(s) and the northwestern rim syncline, are very important for the understanding of the hydrogeological system, as they cause significant consequences on the groundwater flow.



▲ Fig. 2: Hydrogeochemical cross section [modified after 4]

Groundwater recharge is restricted to the area of the Gorlebener Tannen in the south and the lowland north of the river Elbe (see Figure 3). Recharge rates range from -80 up to 160 mm y^{-1} [3]. The upper aquifer is a fresh water aquifer (see Figure 2: TDS < 1 g l^{-1}) [4]. The water was derived from Holocene recharge. Due to a groundwater inflow from the north [2], the water in the lower aquifer flows southwards. Where the Gorleben glacial channel crosses the salt dome, the groundwater dissolves the salt from the caprock. As a consequence of the higher density, the groundwater sinks down into the Northwestern rim syncline. The surface of the salt water is approximately horizontal, while the salt water/fresh water interface has a distinct morphology [4].

3 The Codes d³f and r³t

Modelling was conducted using the programs d³f (distributed density driven flow) and r³t (radionuclides, reaction, retardation, and transport) which were recently developed lead-managed by GRS. Flow and transport equations are solved based on the UG software from the University Heidelberg [5]. The visualisation is based on the program GRAPE (**GRA**phics **P**rogramming **E**nvironment), which was developed by the Universities of Bonn and Freiburg [6].

3.1 The Code d³f for Groundwater Flow Simulations

The program d³f (distributed density driven flow) was developed during a BMBF-funded project (1995-1999) managed by GRS. It is able to handle density-driven groundwater flow over long periods of time through large, hydrogeologically complex model areas [7], such as the sedimentary overburden of salt domes with variable salt concentrations in the groundwater. Modelling two- and three-dimensional flow through porous media in a fluid-saturated, confined aquifer system is possible, regarding advection, diffusion and dispersion as transport processes. d³f treats the salt concentration as a relative concentration of saturation. It utilises fluid density and viscosity as functions of the salt concentration and the temperature.

3.2 The Code r³t for Transport Simulations

The program r³t (radionuclides, reaction, retardation, and transport) as well was developed lead-managed by GRS in a BMWA-funded project (1999-2004) [8]. It is a program to simulate pollutant transport through porous media considering advection, dispersion and diffusion. The main purpose is to include radioactive decay within decay chains considering sorption, precipitation and diffusion into immobile pore waters. r³t is based on the complex steady-state or transient velocity fields of d³f simulations.

4 The Hydrogeological Model

4.1 Geometry

A generic two-dimensional model was set up generalising the geology and hydrogeology of the Gorleben area as shown in Figure 3.

The model has a length of 16.4 km and a heighth of 400 m (Figure 4). It consists of three layers representing the upper aquifer, the aquitard and the lower aquifer. The lower aquifer is composed of Tertiary sediments, while the upper aquifer consists mainly of Quarternary sediments, less Tertiary sediments. The aquitard layer represents the Tertiary Hamburg Clay and the Quarternary Lauenburg Clay. The aquitard is 50 m thick, while the aquifers have a thickness of 100 m each. Within the area of the northwestern rim syncline the lower aquifer reaches a thickness of 250 m. The location of the first hydraulic window (in the south) is proven by boreholes, while a second (northern) hydraulic window was only presumed interpreting pumping test results [3]. Therefore, two variations of the model were realised, comprising either one or two hydraulic windows. In the following they will be referred to as model 1 (one hydraulic window) and model 2 (two hydraulic windows). Both hydraulic windows are

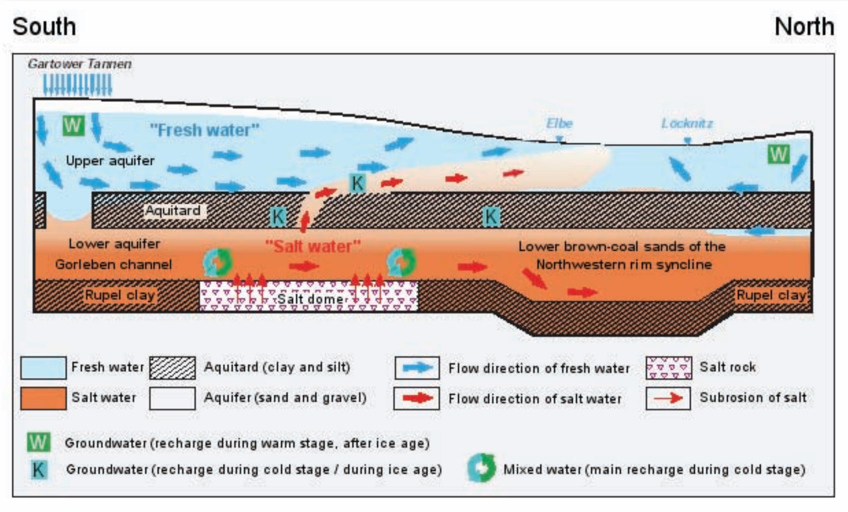


Fig. 3: Schematical cross section of the investigation area [modified after 4]

500 m wide. Utilised hydrogeologic parameters are taken from literature especially from specific papers about the Gorleben area [2] and are given in Tab. 1.

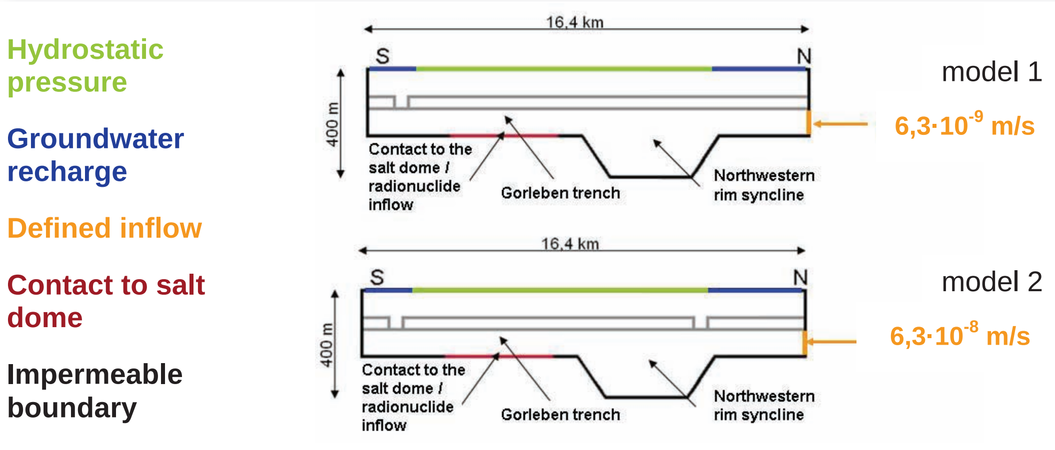
Table 1: Hydrogeological parameters for both models

	Aquifer	Aquitard
Permeability k [m²]	$1{\cdot}10^{-12}$	$1{\cdot}10^{-16}$
Porosity φ [-]	0,2	0,05
Longitudinal dispersion coefficient [m]	10	10
Transversal dispersion coefficient [m]	1	1
Diffusion constant [m² s⁻¹]	$1{\cdot}10^{-9}$	$1{\cdot}10^{-9}$

4.2 Boundary conditions and initial values

The bottom of the model is assumed to be impermeable, since the underlying Oligocene Rupel Clay is the lower boundary of the regional hydrogeological model. As given in [2], the southern boundary as well as the northern boundary of the upper aquifer and the aquitard are impermeable. There is an inflow from the north into the upper aquifer of $1{\cdot}10^6$ m^3 y^{-1}, which is implemented in the model in form of a velocity boundary condition (Figure 4). Hydrostatic pressure is defined for the central part of the surface, where the lowlands of the rivers Elbe and Löcknitz are located. The Gorlebener Tannen in the south and the croplands north of the river Elbe are characterised by groundwater recharge. Values for the recharge rate are ranging from -80 mm y^{-1} up to 200 mm y^{-1} (see 2.1). A recharge rate of 160 mm y^{-1} is implemented in the model as a velocity boundary condition.

Dirichlet conditions for the salt mass transport are set to 0 for the inflow boundaries and to 1 for the contact to the salt dome, which is situated between the southern hydraulic window and the northwestern rim syncline. The salt concentration depends on the direction of groundwater flow in the areas with hydrostatic pressure at the surface. A Dirichlet condition of c_{rel}=0 is set for the inflow of groundwater, while for the outflow the salt concentration equals the concentration of the outflowing groundwater. Diffusion and dispersion across the border is neglected.



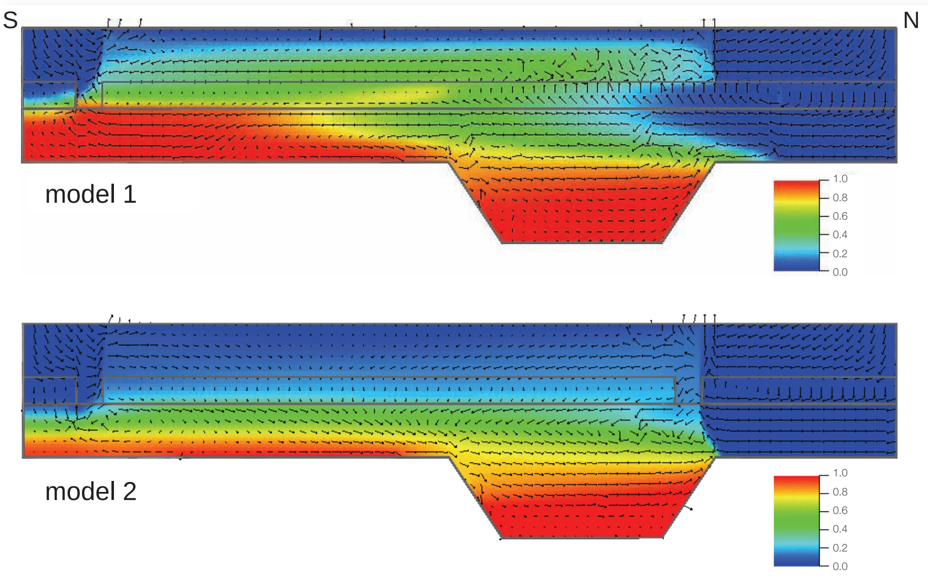
▲ Fig. 4: Geometry and boundary conditions of the two hydrogeological models

Initial salt concentrations are set c_{rel}= 1 for the lower aquifer and the aquitard, c_{rel} is set to 0 for the upper aquifer. Between a depth of 70 and 100 m within the upper aquifer, the salt concentration rises linearly from c_{rel} = 0 to c_{rel} = 1. The temperature is defined to be t = 8 °C at the surface and rises up to 20 °C at the bottom of the model according to the geothermal gradient.

Assuming that today's flow field could have developed since the end of the last glaciation, the start time of the simulation is 11,500 y before present, which accords to the end of the Weichsel cold stage and the beginning of the Holocene.

5 Recent Flow Field and Salt Concentrations

With the start time of 11,500 y before present, todays flow fields and salt concentrations were computed (Figure 5). To obtain similar simulation results, it was necessary to reduce the lateral inflow into the lower aquifer to 10 % of the value given in [2]. The salt concentrations and velocity vectors are depicted in Figure 5.



◄ Fig. 5: Recent salt concentrations and velocity fields for model 1 and 2

Main characteristics of the recent flow field are reproduced correctly in both model variations, such as (i) two domains of groundwater upstream at the surface, (ii) inflow from the north and the areas of groundwater recharge, (iii) southward flow in the upper part of the lower aquifer, and (iv) northward flow in the lower part of the lower aquifer. Due to salt dilution at the contact to the salt dome groundwater with higher salt concentrations is descending into the northwestern rim syncline. Both models result in a horizontal layering of groundwater with different salt concentrations and densities. Model 1 has a distinct fresh water / salt water interface morphology, which reaches the surface in the area of the river Elbe. The northwestern rim syncline is almost saturated with salt water with an almost horizontal surface. Postulating one hydraulic window, these velocity vectors and salt concentrations can only be simulated assuming a reduced lateral inflow of 10 % of the given $1{\cdot}10^6$ m^3a^{-1}. Considering a second hydraulic window and the non-reduced lateral inflow, the simulations prove the position of the hydraulic window to be north of the northwestern rim syncline. Inserting the window above the salt dome or close to the rim syncline reduces the resulting salt concentrations in the syncline up to fresh water conditions depending on the location of the window.

To verify the simulated salt concentrations, three hydrogeological observation wells were selected to compare the recent water density with the model results. One of the diagrams is shown in Figure 6. The selected observation well is situated centrally between the southern hydraulic window and the contact to the salt dome. It is obvious, that model 2 reproduces the actual salt concentration better than model 1. Due to the remarkable simplification of the model geometry, the measured density logs cannot be easily compared with the model results. In Figure 6 the aquitards are marked for both logs. Up to approximately 75 m depth, the measurements of the BGR indicate the presence of freshwater in the upper aquifer and parts of the aquitard. At the bottom of the aquitard high altitude areas of saltwater are situated, underlain by freshwater. The density of the groundwater rises significantly within the lower aquifer. Model 1 shows higher density values than those of the BGR measurements over the entire model depth, with the strongest increase within the aquitard. Model 2 shows only a slight density increase within the upper aquifer. From the middle of the aquitard the density rises almost linearly down to the bottom of the model. Comparisons for the two other observation wells show similar results.

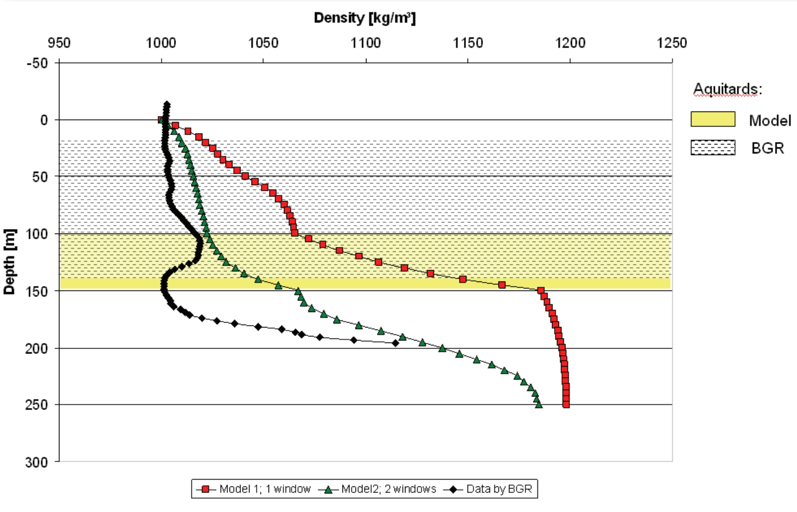


Fig. 6: Density logs for the observation well GoHy193 and the according locations in the two models after 11,500 a model time

In summary, the results of the simulations show that both models compute a similar velocity field. Regarding the salt concentrations, model 2 emulates the recent salt concentration field more realistically.

6 Climate Scenarios

Selected climate scenarios are deduced from the geological past and kept as concise as possible. Modelled climate scenarios are (i) constant climate represented by constant boundary conditions, (ii) the inundation of the area with sea water due to global warming (warm stage) and (iii) permafrost under the influence of a near-by inland ice sheet during a reoccurring ice age.

6.1 Constant Climate

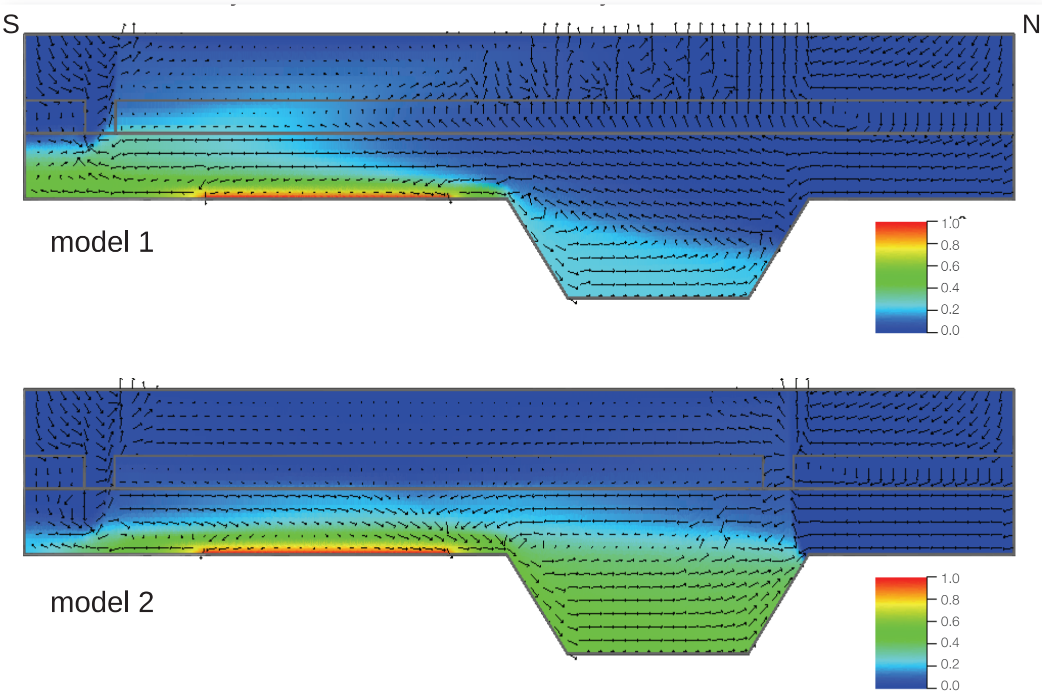
6.1.1 Input Parameters

To compute the salt concentrations and velocity vectors for constant boundary conditions, the models were run up to steady-state conditions after 300,000 years (Figure 7). Input parameters were taken from the previous modelling of the recent velocity fields (see ch. 4.1).

6.1.2 Results

At steady-state conditions, the velocity fields of model 1 and 2 strongly differ from each other, as well as the salt concentrations. Within the lower aquifer, complementary velocity vectors in the two models occur only south of the salt dome. Most relevant is the contrary groundwater flow between the southern hydraulic window and the rim syncline.

In model 1, groundwater flow is targeted southwards, in the second model groundwater flows northwards into the rim syncline. This difference will exert explicitly on the transport modelling (see ch. 7). Another significant difference is the velocity field within the upper aquifer. Comparable to the recent flow fields, there are two areas of groundwater upstream, but this time, in model 1 it reaches almost 4 to 5 km width just above the northwestern rim syncline. In model 2, areas of groundwater upstream are comparable to the recent ones. Altogether, higher flow velocities and larger areas of outflow in the upper aquifer are computed with the first model.



▲ Fig. 7: Steady-state velocity fields for constant bounday conditions after 300,000 a model time

6.2 Sea Water Inundation

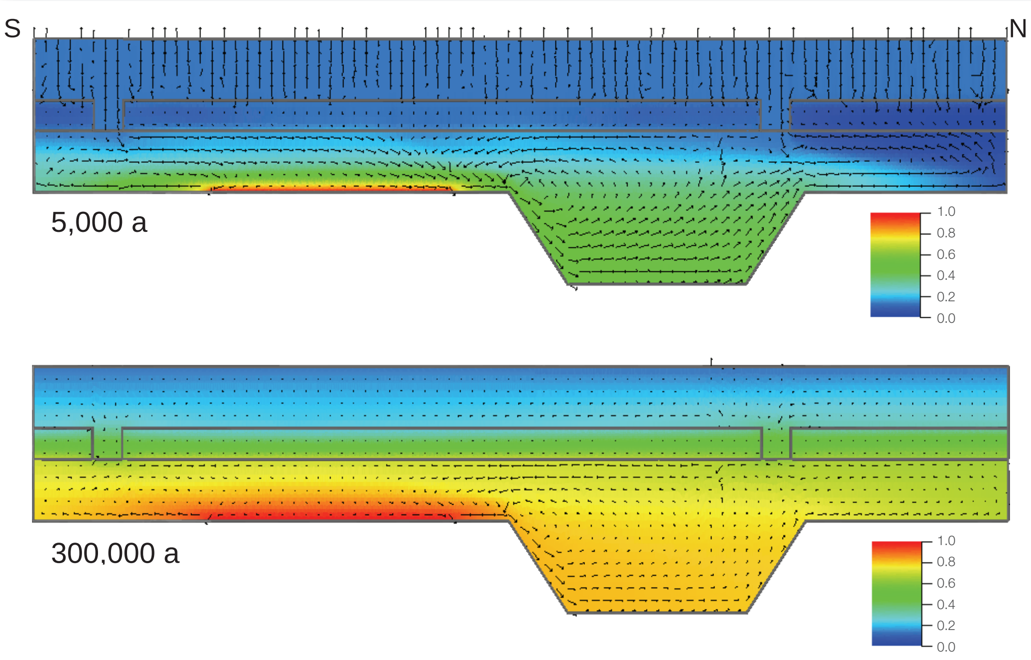
During sea water inundation the recharging sea water has a higher salt concentration than that of the former groundwater recharge. Assuming the sea water intruding the investigation area from the northwest, North Sea salt concentration is implemented in the simulations. According to calculations [9], sea water level will rise up to 75 m in case the entire Antarctic and Greenland ice volume will melt. Simplified, this leads to a maximum sea water column of 50 m in the Gorleben area. As this work is supposed to be a process study of extreme climatic changes, the maximum value is implemented in the modelling.

6.2.1 Input Parameters

For the entire surface a pressure condition equivalent to 50 m sea water column was defined. All other boundaries are assumed to be impermeable. A variation of this simulation is conducted with hydrostatic pressure on the lateral boundaries. This model variation is highly complex and CPU time needed exceeds up to two months for 1,000 a model time. Hence, reducing the simulations to scenarios with impermeable boundaries was necessary. Simulations were conducted starting with recent, transient conditions as well as with the steady-state velocity field.

6.2.2 Results

First impacts on the sea water inundation can be recognised almost immediately. Exemplarily, velocity fields and salt concentrations for model 2, starting with the transient flow field, are shown after another 5,000 and 300,000 a model time in Figure 8. After 1,000 a, the upper aquifer is almost saturated with salt concentrations equivalent to sea water. After 5,000 a (see Figure 8) this applies to most of the aquitard as well, only in the northern part freshwater is still present in the model area. Within the lower aquifer the salt concentrations and velocity vectors still correspond with the steady-state results (300,000 a). During the following simulation up to 600,000 a model time, a horizontal layering results with increasing fluid density from the surface to the bottom of the model. Advection is no longer the dominant process, identifiable from the short velocity vectors, but diffusion processes cause the saturation.



▲ Fig. 8: Velocity fields and salt concentrations for sea water inundation, model 2, after 5,000 a (305,000 a) and 300,000 a (600,000 a) inundation time (model time), based on the steady-state velocity and concentration fields

For long term simulations, it is of almost no importance, which model serves as the input for this simulation. After approximately 10,000 a model time, the salt concentrations and velocity fields are almost identical. However, the start time of the simulation has a noticeable impact on the simulation results during the first 100,000 a model time at least. Due to the higher salt concentrations in the transient fields (see Figure 5), advection almost ceases after 5,000 a already. After 300,000 a model time, the horizontal density-dependant layering is identified is this model variation as well but still shows slightly higher salt concentrations.

6.3 Permafrost

During a cold stage, the area could be covered by an inland ice sheet (comparable to the Elster and Saale cold stages) or affected by a near-by inland ice sheet (comparable to the Weichsel cold stage). This model is not suited to study the impact of a superposing inland ice sheet, because it exceeds over only 16.4 km. A regional model has to be set up to comprise the factors of glaciation, such as the enlarged catchment area and the immense water volume infiltrating from the glacier base [10]. Geometry changes, e. g. wash out of another glacial channel, cannot be implemented in the program d³t.

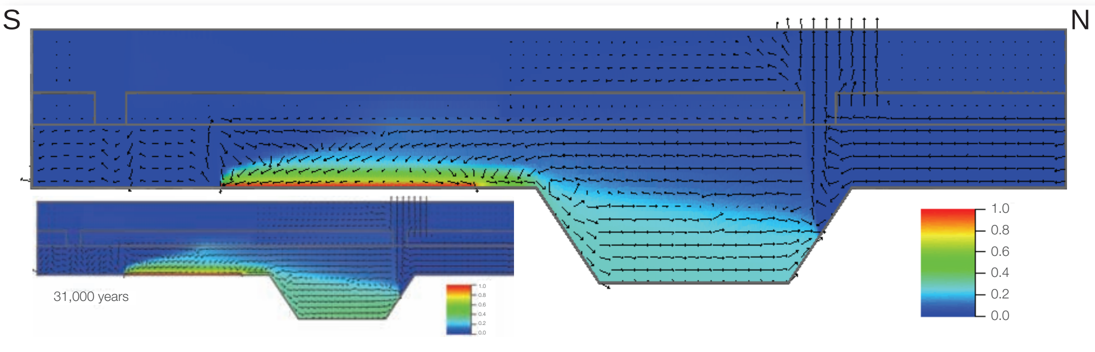
Permafrost causes freezing of fluid-saturated sediments, which prevents groundwater flow. In the Gorleben area, calculated permafrost thicknesses for the past ice ages range from 40 m up to 140 m [3, 9]. To keep the simulation and interpretation as concise as possible, permafrost conditions were applied to the entire upper aquifer. Taliki are unfrozen areas within the permafrost [12]. They may occur, where rivers or lakes have thermal effects on underlying permafrost [11], so that unfrozen areas can form. Assumably, during permafrost conditions an inland ice sheet exists north of the Gorleben area. Melt water can infiltrate into the aquifers, because the weight of the superposing ice sheet prevents the area from permafrost formation. As a consequence, these melt waters can flow into the lower, unfrozen aquifer from the north. Groundwater upflow only takes place through taliki, where the groundwater flow velocity is expected to rise significantly.

6.3.1 Input Parameters

Permafrost conditions were realised reducing the permeability of parts of the upper aquifer to $k = 10^{-20}$ m², representing quasi-impermeable domains. Two taliki are realised within the upper aquifer. One is situated underneath the rivers Elbe and Löcknitz (5 km width), the second is 500 m wide and located 450 m far from the southern border of the model. For the entire surface hydrostatic pressure is defined. Inflow from the north into the lower aquifer is given in form of a gradient $i = 0.1$ [13] and implemented as a velocity boundary condition. As a variation, the southern border of the lower aquifer is assumed to be subject to hydrostatic pressure. The simulations were again started at steady-state conditions as well as with the recent, transient velocity fields.

6.3.2 Results

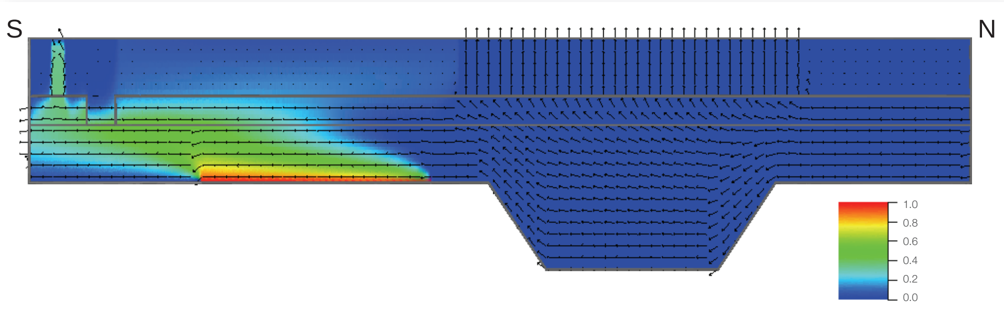
First conclusions can be drawn, although the simulations were still running at time of writing. In contrary to the scenario of sea water inundation, this time the existence of a second hydraulic window is of great importance. After approximately 30,000 a, model 2 has already reached quasi-stationary conditions (Figure 8: 400,000 a). One can see clearly the frozen intervals in the upper aquifer. Only diffusion processes take place here. The melt water inflow causes high velocities close to the northern hydraulic window with maximum values of $1 \cdot 10^{-6}$ ms^{-1}. Most of the inflow is discharged through the talik above the northern hydraulic window. The aquitard prevents a groundwater discharge over the entire length of the southern talik. In the upper part of the lower aquifer, the flow direction is still directed southwards. Above the contact to the salt dome, the flow direction reverses to the north. Groundwater can discharge to the south out of the lower aquifer as well to the surface. The second talik is not used as a pathway for groundwater discharge because it is not located above the hydraulic window.



▲ Fig. 9: Velocity fields and salt concentrations for the permafrost scenario, model 2, after 10,000 a (310,000 a) (below) and 400,000 a (100,000 a) (above) permafrost conditions (model time), based on the steady-state velocity and concentration fields

Comparing this simulation with the results of model 1 (Figure 10) after 10,000 a permafrost conditions, there are significant differences in the salt concentrations and the flow fields. The melt water inflow cannot be discharged through the first talik in the same dimension as in model 2. Most of the water volume is routed southwards rinsing the salt water out of the northwestern rim syncline. This time, the southern talik is a pathway for groundwater flow as well, because of the higher velocities in the southern part of the lower aquifer. Therefore, salt water occurs in the southern talik. In both models salt concentrations in permafrost areas remain unaltered due to the low permeability.

Starting the simulation with the recent flow fields produces comparable velocities. The most important difference is the salt concentration in permafrost areas, which again remain unaltered.



▲ Fig. 10: Velocity fields and salt concentrations for the permafrost scenario, model 1, after 10,000 a (310,000 a) permafrost conditions (model time), based on the steady-state velocity and concentration fields

7 Transport Modelling

After computing the recent velocity fields for both models 1 and 2, initial transport simulations were performed. Considered radionuclides are C-14, I-129, Cs-135 and Zr-93 as well as selected nuclides of the uranium decay chain, which are U-238, including U-234, Th-230 and Ra-226. Examplarily, the simulations for Cs-135 will be presented here.

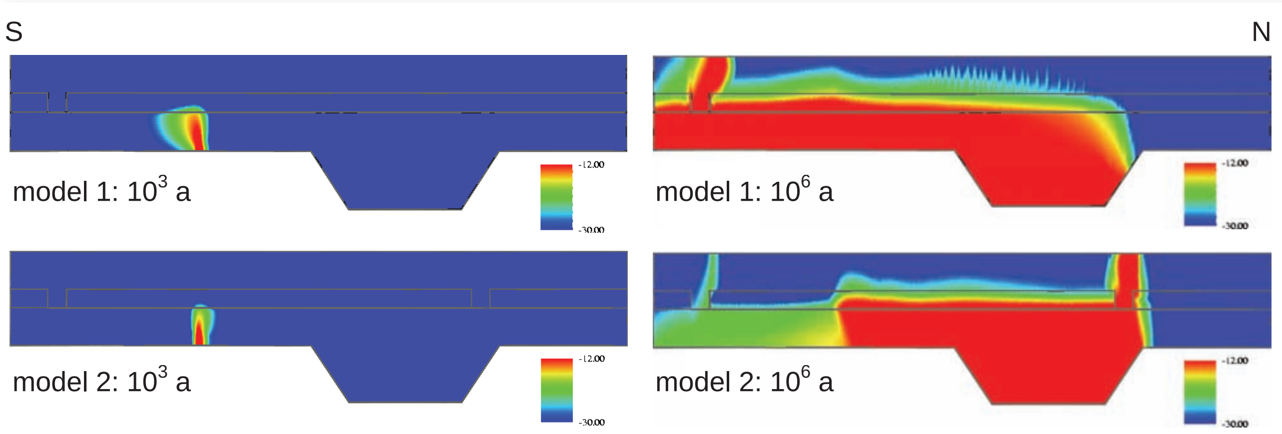
7.1 Model Setup and Input Parameters

The d^3f velocity fields serve as a basis for r^3t calculations. Sorption is dependent on the sediment grain size as well as on the fluid salinity. A specific sorption data set (sorption coefficient: K_d-value) was determined for the Gorleben overburden [14]. For Cs-135, these values are defined as: Aquifer (sand) & freshwater 70 ml/g, aquifer (sand) & salt water 2 ml/g, aquitard (silt & clay) & freshwater 400 ml/g, aquitard (silt & clay) & salt water 70 ml/g. For first simulations, sorption coefficients are defined globally for the different layers. Furthermore, the half time has to be declared (Cs-135: $2.00 \cdot 10^6$ a). The inflow location is selected as a point source centred in the salt contact. Simulations for two different source term periods are performed with (i) a delta impulse of 1 $mola^{-1}$ over a period of one year and (ii) a constant source of 10^{-6} $mola^{-1}$ over the whole simulation period of 10^6 a.

7.2 Results

The distribution of the radionuclide Cs-135 for a delta impulse after 10^3 and 10^6 years are given in Figure 11. Fundamental d^3f velocity fields are the stationary fields of constant boundary conditions (see Figure 11). After 1,000 a, the radionuclides have already reached the bottom of the aquitard. After 10^6 a, the differing flow directions have a strong influence on the radionuclide distribution. While in model 1, the main transport upwards is realised through the southern window, in model 2 the radionuclides are transported through the northern window. K_d-vaules were constituted globally for the three different layers, so within the lower, salt water aquifer, radionuclides are distributed more effectively than whithin the upper, freshwater aquifer. In model 1, the nuclides will leak from the model area into the river Elbe.

So far performed transport simulations show a strong dependence of the radionuclide distribution on sorption coefficients which are mainly controlled by the salt concentration of the groundwater. Radionuclide concentrations mainly depend on their half-life period.



▲ Fig. 11: Transport simulations for Cs-125, model 1 (above) and 2 (below), after 10^3 a (left) and 10^6 a (right) model time, based on the steady-state velocity fields (300,000 a) with constant boundary conditions (Fig. 7)

8 Outlook

In the future, transport simulations will be extended to the scenarios sea water inundation and permafrost. The overall perspectives of the project are (i) to combine the different scenarios and (ii) to calculate the potential exposition to radiation from the time series of the radionuclide concentration in near-surface aquifers. ■

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2B.13 Evaluation of Shale Formations as Potential Host Rocks for Radioactive Waste Disposal in Germany

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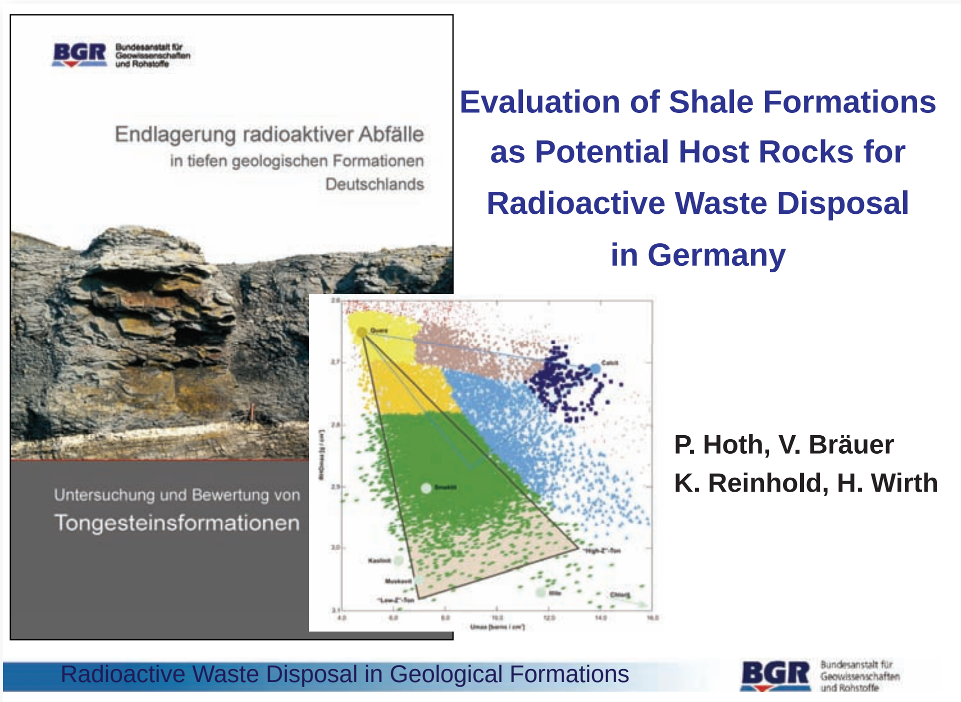

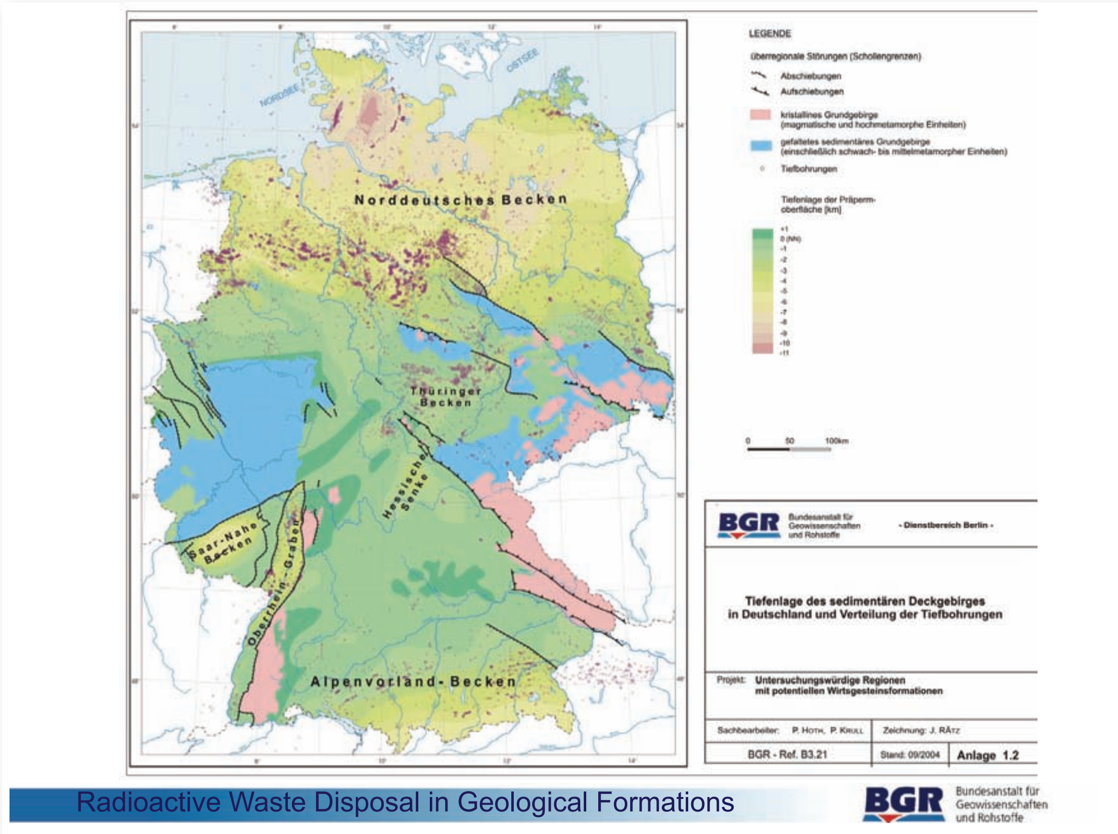
LEGENDE
überregionale Störungen (Schollengrenzen)
Abschiebungen
Aufschiebungen
kristallines Grundgebirge (magmatische und hochmetamorphe Einheiten)
gefaltetes sedimentäres Grundgebirge (einschließlich schwach- bis mittelmetamorpher Einheiten)
Tiefbohrungen
Tiefenlage der Präperm-oberfläche [km]
NORDSEE
OSTSEE
Norddeutsches Becken
Thüringer Becken
Hessische Senke
Saar-Nahe Becken
Oberrhein-Graben
Alpenvorland-Becken
BGR Bundesanstalt für Geowissenschaften und Rohstoffe
- Dienstbereich Berlin -
Tiefenlage des sedimentären Deckgebirges in Deutschland und Verteilung der Tiefbohrungen
Projekt: Untersuchungswürdige Regionen mit potentiellen Wirtsgesteinsformationen
Sachbearbeiter: P. Hoth, P. Krull
Zeichnung: J. Rätz
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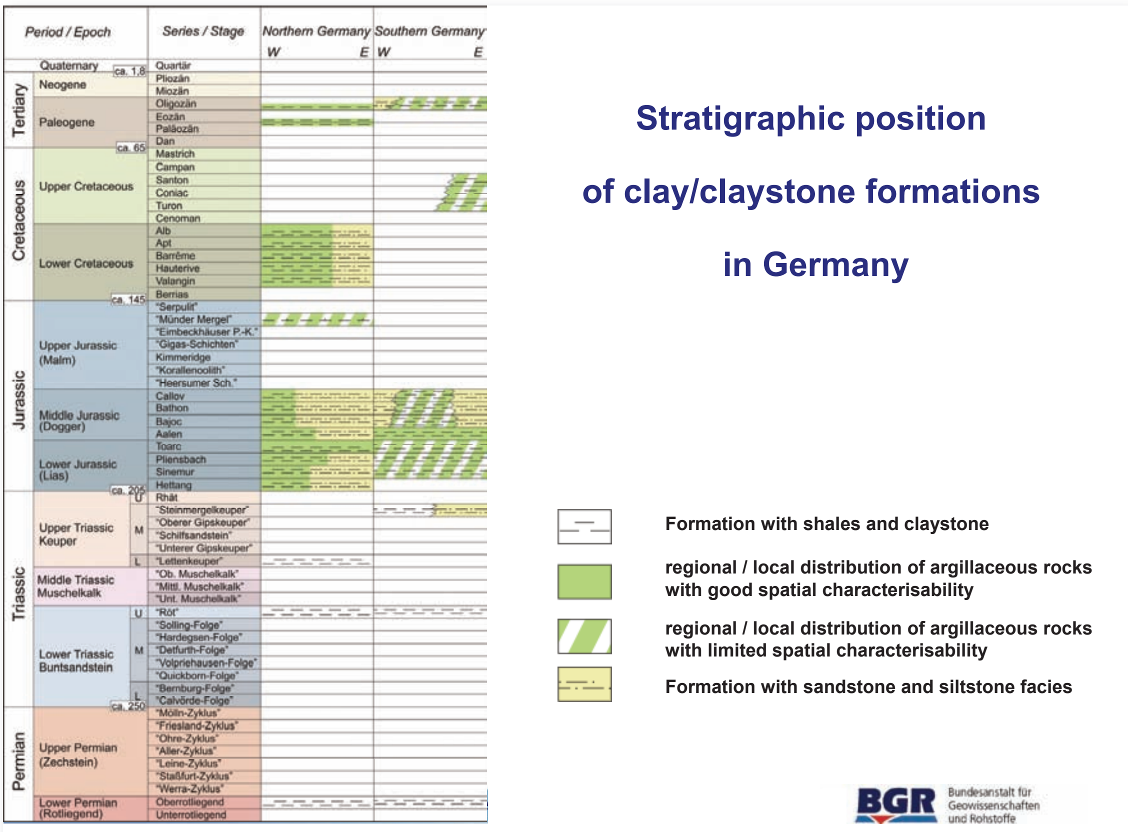
Radioactive Waste Disposal in Geological Formations
BGR Bundesanstalt für Geowissenschaften und Rohstoffe


Period / Epoch
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Triassic
Upper Triassic Keuper
Middle Triassic Muschelkalk
Lower Triassic Buntsandstein
Permian
Upper Permian (Zechstein)
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Stratigraphic position
of clay/claystone formations
in Germany
Formation with shales and claystone
regional / local distribution of argillaceous rocks with good spatial characterisability
regional / local distribution of argillaceous rocks with limited spatial characterisability
Formation with sandstone and siltstone facies
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Target and data base for the clay/shale evaluation study

Mapping of regions which are potentially suitable for further investigations in Germany, no site search !

- Literature
- Drilling records
- Existing maps and compilations
- Borehole logs
- Seismic-Data

- No own in-situ tests and field measurements

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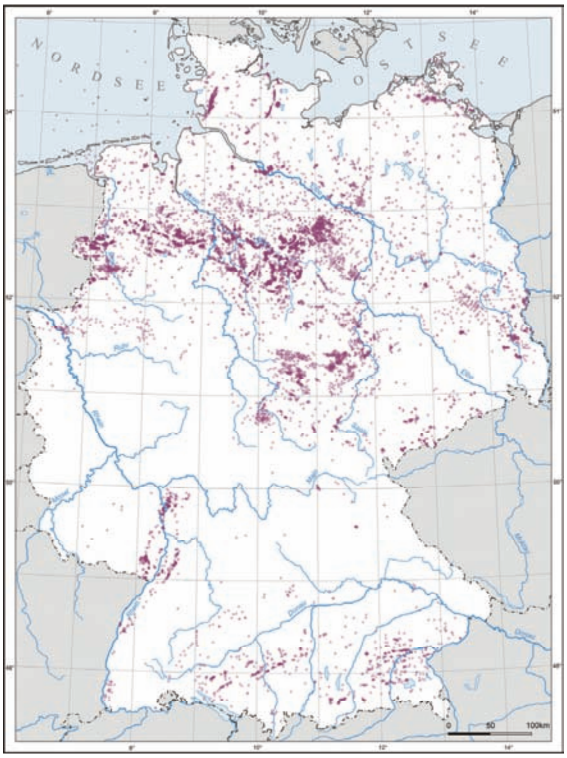

Boreholes in Germany

(approx. 25.000)

Deep boreholes (depth > 300m)

- oil and gas exploration
- oil and gas production
- salt and ore exploration
- geothermal energy
- research wells

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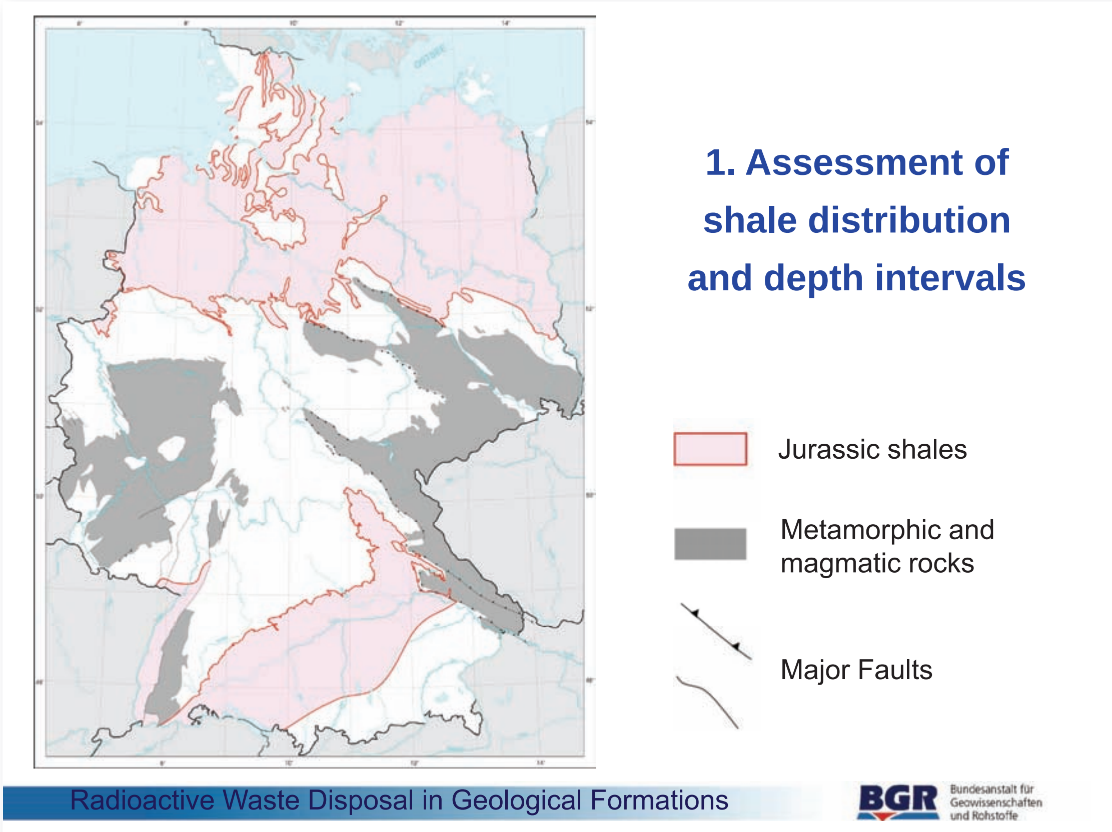
1. Assessment of shale distribution and depth intervals
Jurassic shales
Metamorphic and magmatic rocks
Major Faults
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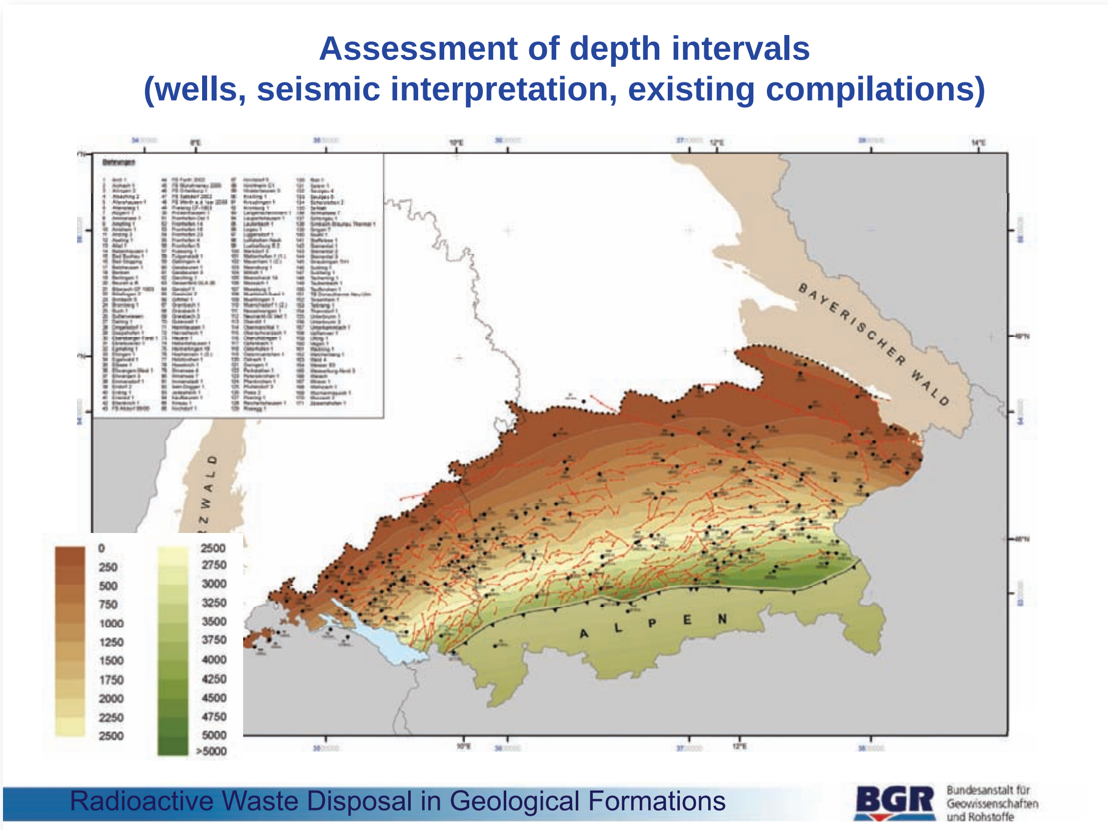
Assessment of depth intervals
(wells, seismic interpretation, existing compilations)
BAYERISCHER WALD
ALPEN
0
250
500
750
1000
1250
1500
1750
2000
2250
2500
2500
2750
3000
3250
3500
3750
4000
4250
4500
4750
5000
>5000
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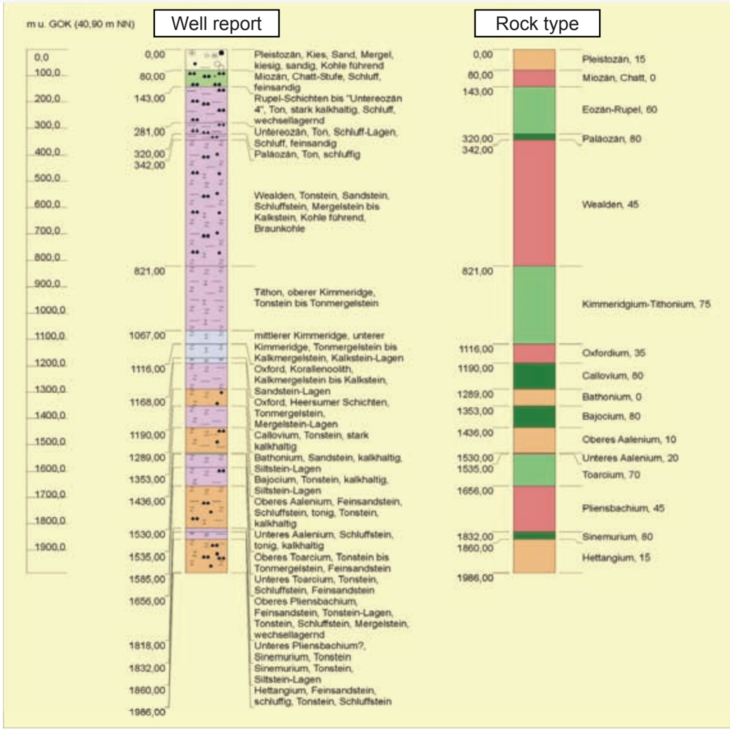

2. Analysis of wells

Rock type classification:

> 80%

60 - 80%

Percentage of rock-type „shale“

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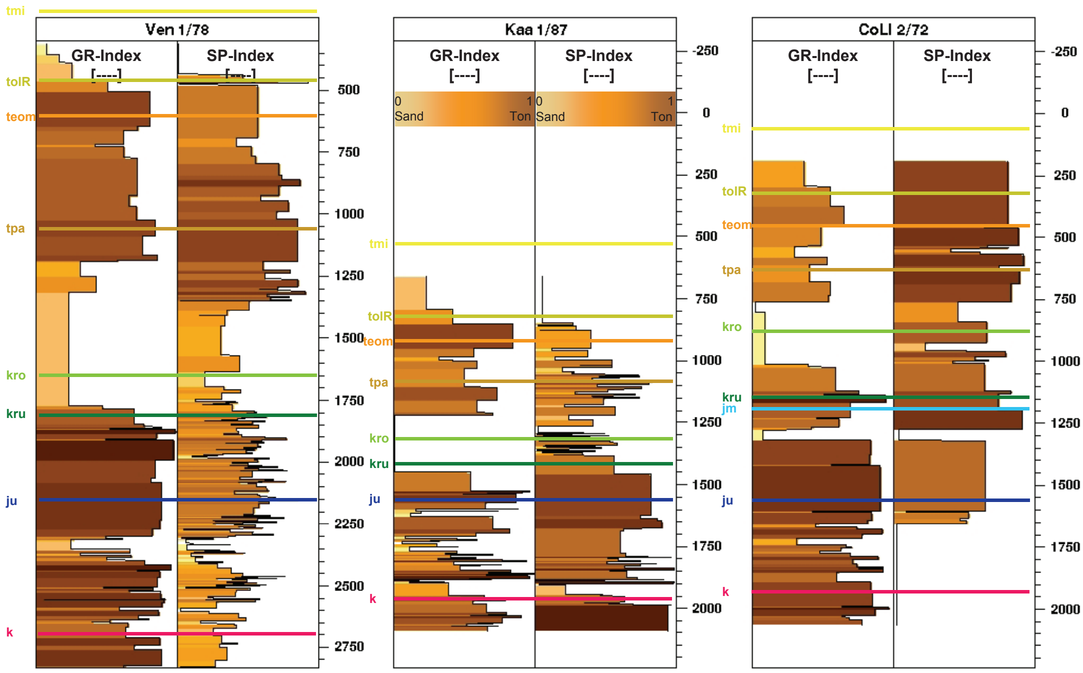
Estimation of the shale content from Gamma- and SP-Logs (shale- and sand- line concept)



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3. Requirements and Criterias

Fundamental requirements on the geological environment of a deep repository (IAEA, Nagra (CH), Andra (F))

- Long-term geological stability
- Favorable host rock properties
- Sufficient extent of host rock body
- Avoidance of, and insensitivity to, detrimental phenomena and perturbations
- Explorability
- Predictability

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3. Requirements and Criterias

Basic exclusion criteria

- Large-areal vertical movements
- Active fault zones
- Seismic activity
- Volcanic activity

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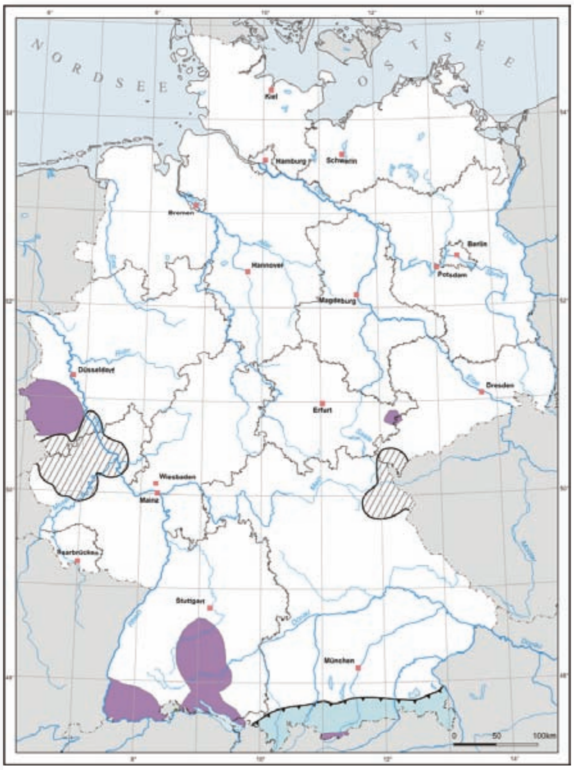

Areas with seismic and volcanic risks in Germany

Seismic zone > 1 (DIN 4149)

Areas with increased risk of volcanism

Both criterias seen as exclusion criterias for Germany by the Site Selection Comm. AkEnd (2002)

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Minimum requirements for disposal sites (1)

- The isolating rock zone must consist of rock types to which a field hydraulic conductivity of less than 10^{-10} m/s can be assigned
- The thickness of the isolating rock zone must be at least 100 m
- The repository mine must lie no deeper than 1,500m
- The isolating rock zone must have an areal extension that permits the realization of a repository (e. g. approximately 3 km² in salt or 10 km² in clay or granite)

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Minimum requirements for disposal sites (2)

- Neither the isolating rock zone nor the host rock must be at risk from rock burst
- There must be no findings or data which give rise to doubts whether the geoscientific minimum requirements regarding field hydraulic conductivity, thickness and extent of the isolating rock zone can be fulfilled over a very long period of time (in the order of magnitude of one million years)

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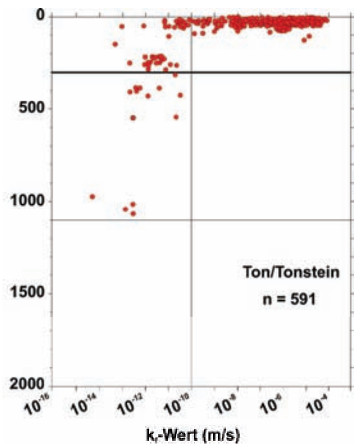

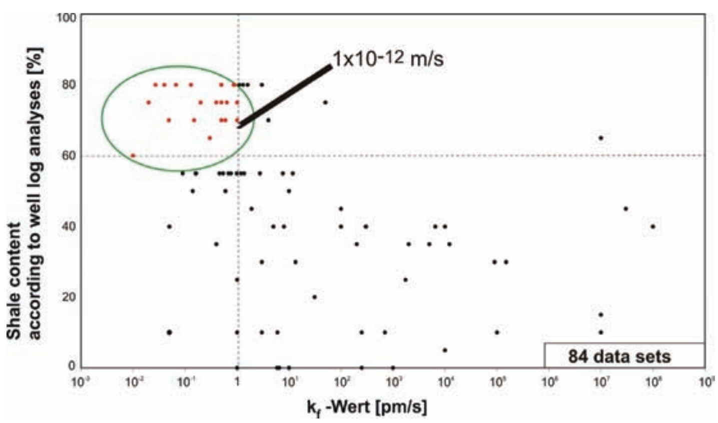

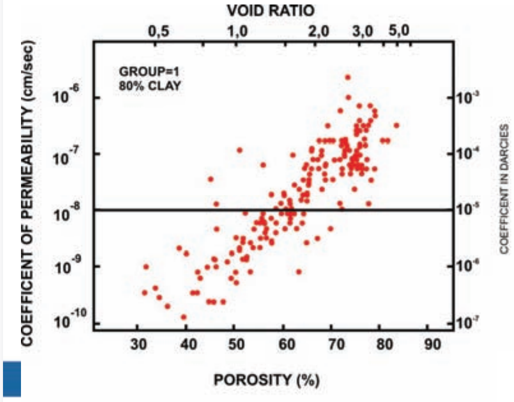

Claystones are often characterized by field hydraulic conductivities of less than 10^{-10} m/s.

Own investigation support results of (Appel & Habler 2001), (Bryant 2003) and others.

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Additional specific criteria in Germany

- Repository must lie no deeper than 1000 m (shales: Temp. and geomechanical prop.)
- Clays with plastic behavior were not the target of the study
- Additional regional restrictions

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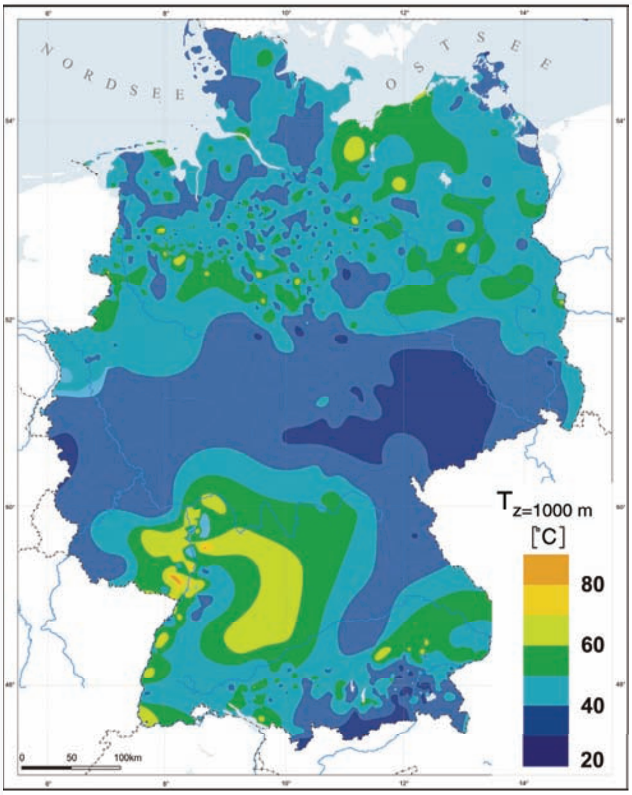

Temperature distribution in Germany

Temperature distribution (depth = 1000m)

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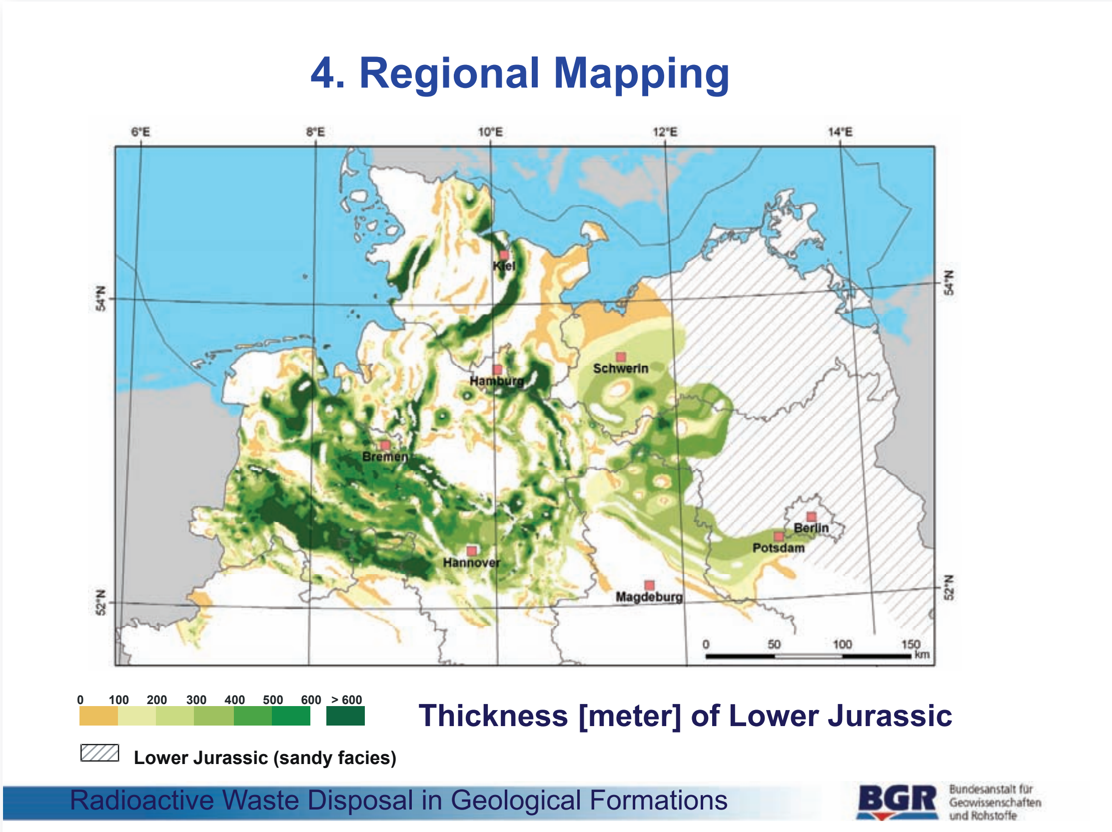
4. Regional Mapping
6°E
8°E
10°E
12°E
14°E
54°N
52°N
Kiel
Hamburg
Schwerin
Bremen
Berlin
Potsdam
Hannover
Magdeburg
0
50
100
150
km
0 100 200 300 400 500 600 > 600
Thickness [meter] of Lower Jurassic
Lower Jurassic (sandy facies)
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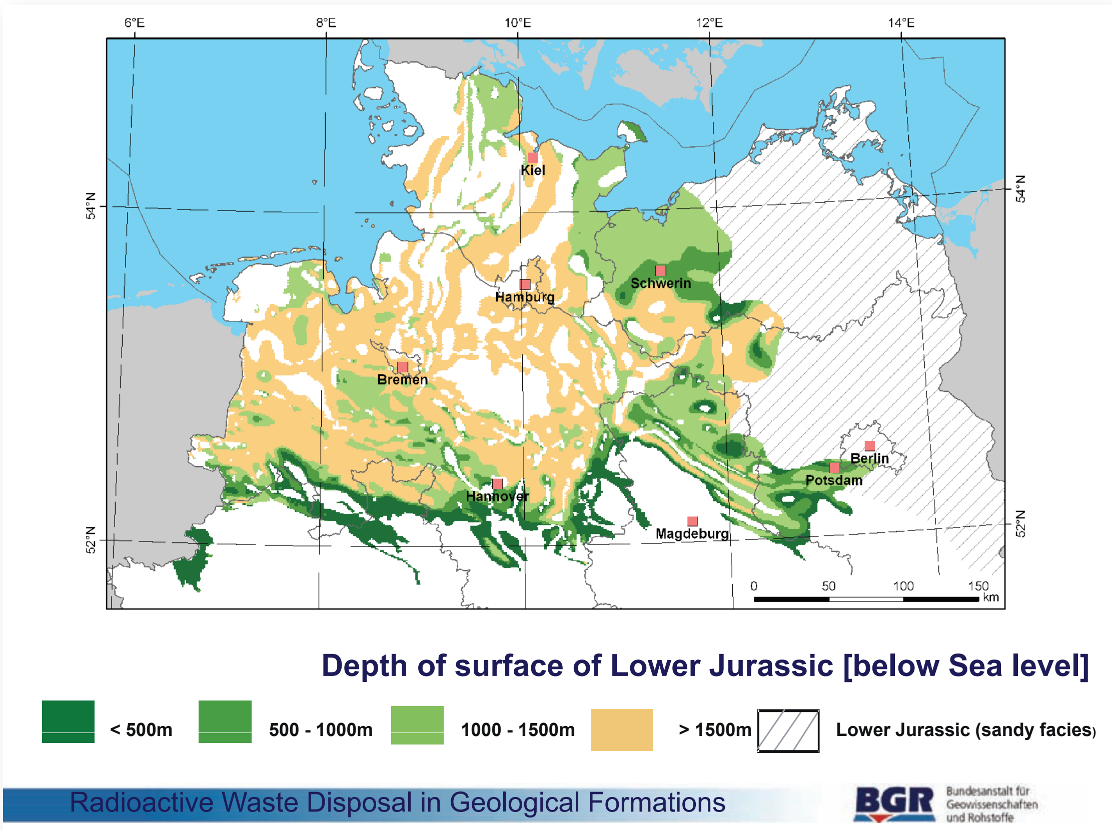
6°E
8°E
10°E
12°E
14°E
54°N
52°N
Kiel
Hamburg
Schwerin
Bremen
Berlin
Potsdam
Hannover
Magdeburg
0
50
100
150
km
Depth of surface of Lower Jurassic [below Sea level]
< 500m
500 - 1000m
1000 - 1500m
> 1500m
Lower Jurassic (sandy facies)
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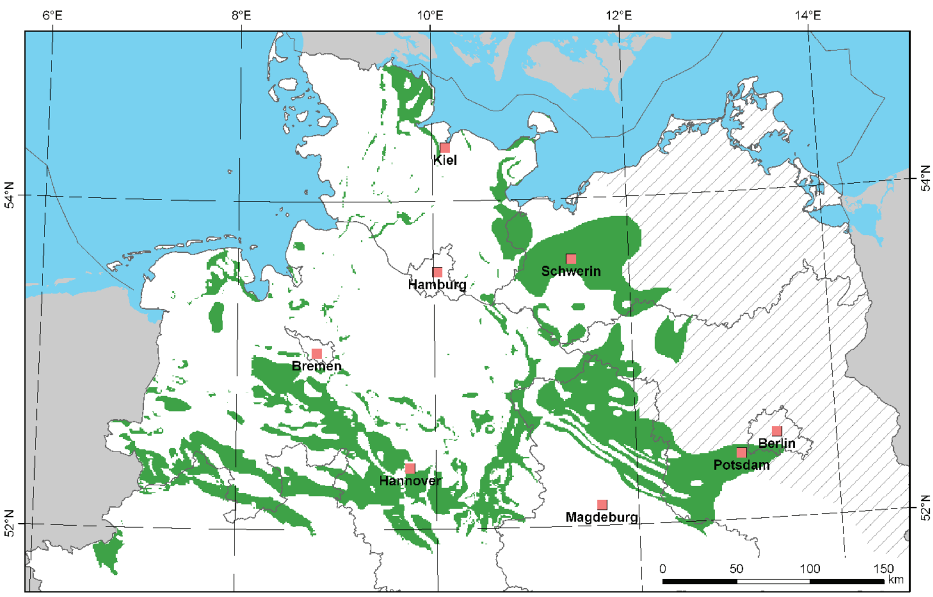

Thickness > 100m
Depth < 1500m

Minimum requirements / Lower Jurassic

Lower Jurassic (sandy facies)

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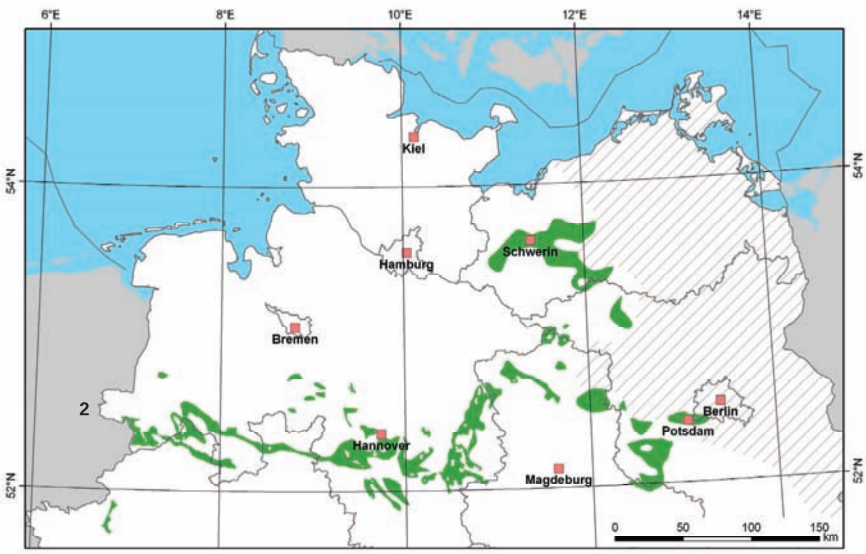

Thickness > 100m
Depth > 300m and < 1000m
(below groundsurface)
area >10km
(with good predictability)

Application of all claystone specific criteria / Lower Jurassic

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5. Test of Mapping Results

Correlation of well logs with CORRELATOR – shale homogeneity and predictability

WELL DATA

- Surface location
- Surface elevation
- Identification
- Symbol for posting

LOG DATA

- Digitization must be at fixed intervals.
- No gaps are allowed.
- Only wells with the same type of logs can be correlated.
- At most two logs are used simultaneously.

Cooperation with:
University of North Carolina
KGS and USGS

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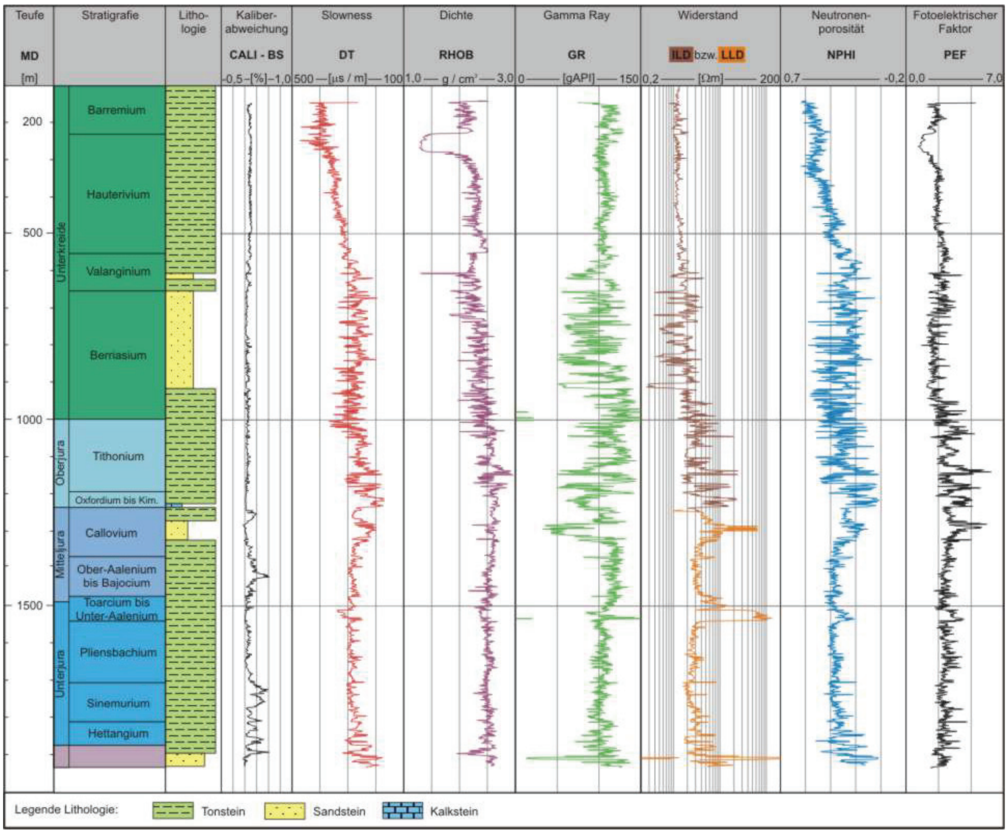
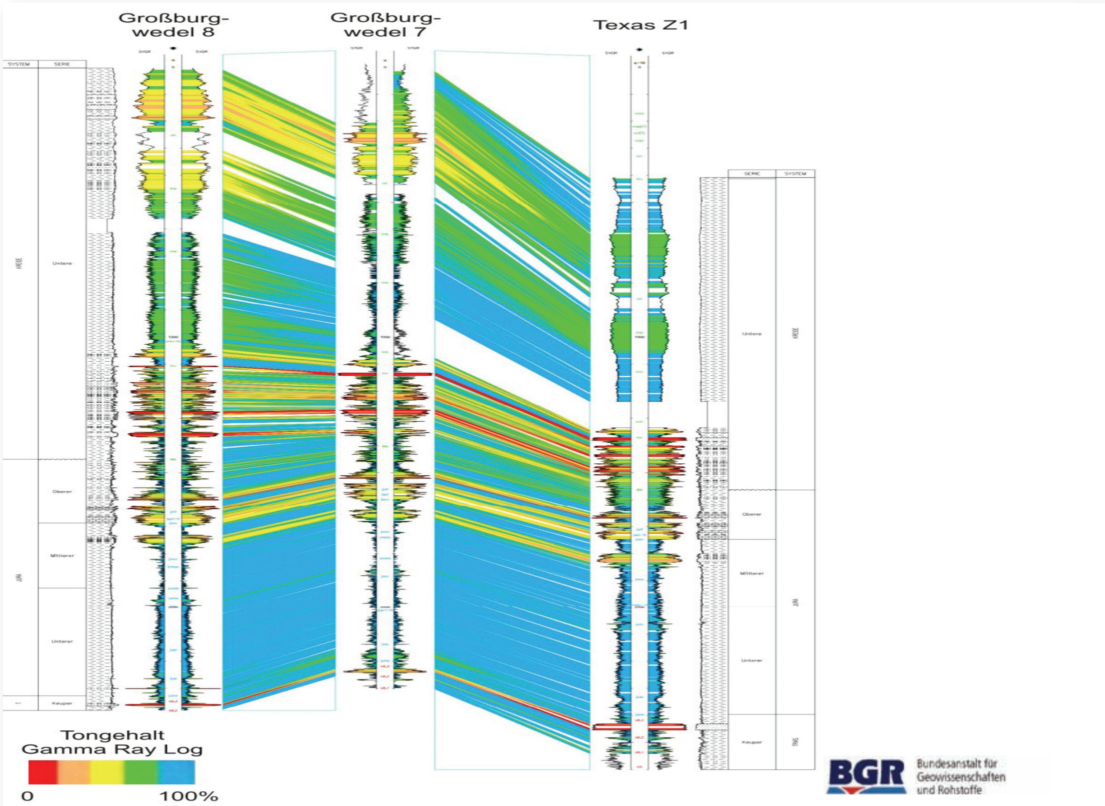
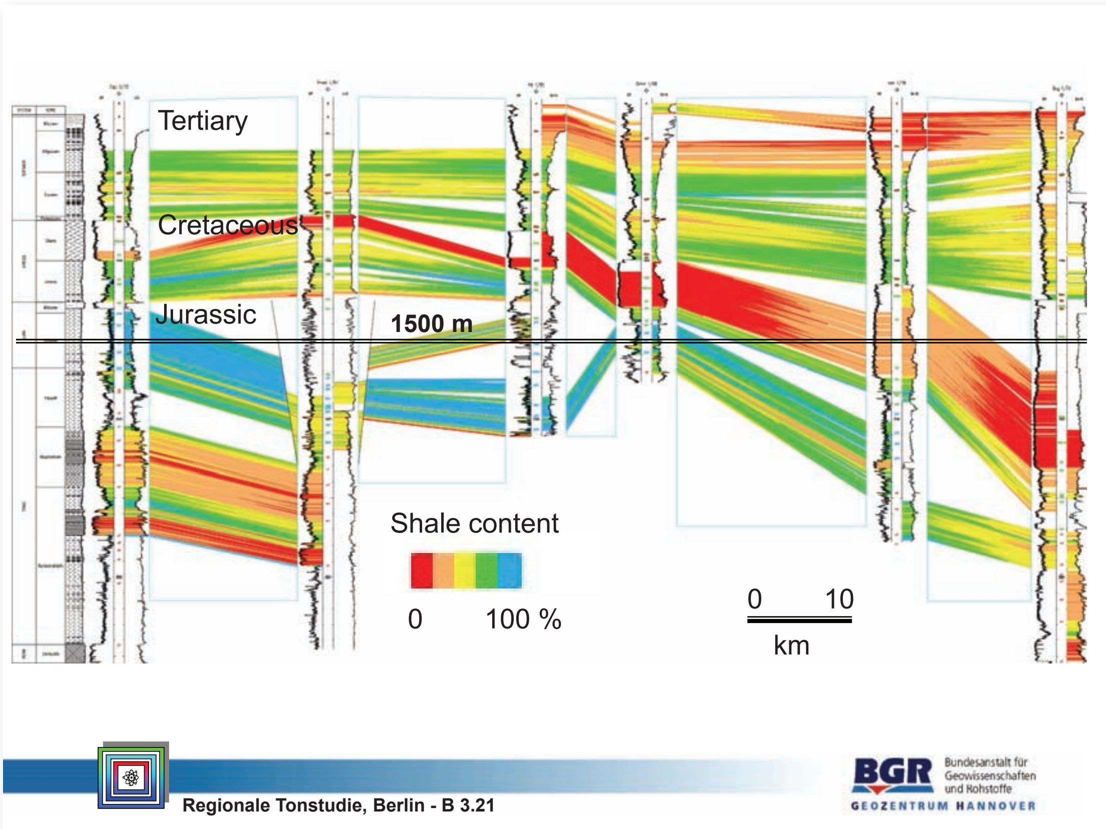
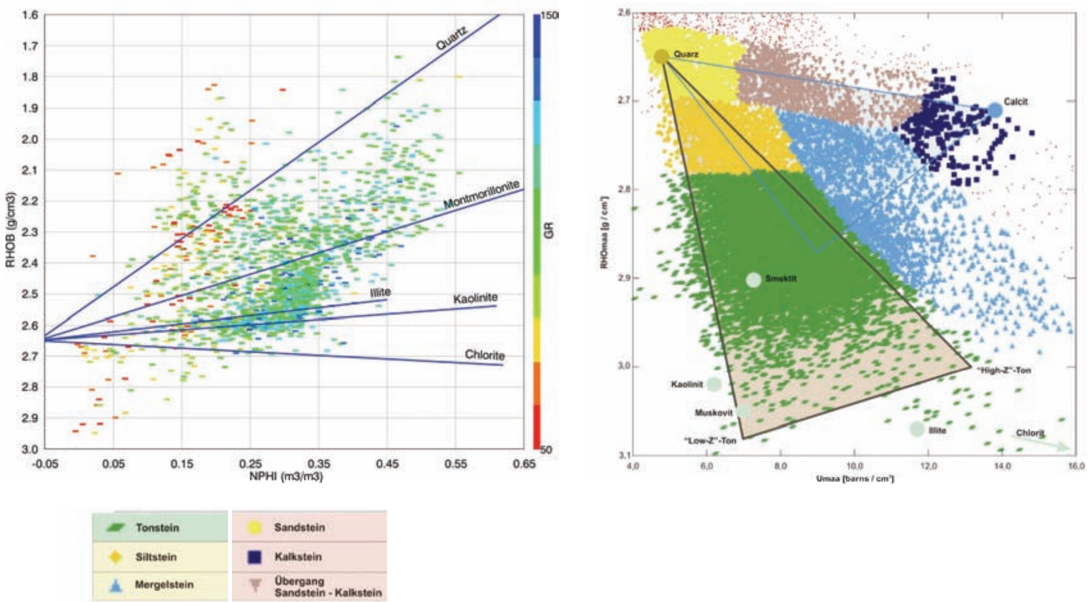



Großburg-
wedel 8
Großburg-
wedel 7
Texas Z1
Tongehalt
Gamma Ray Log
0
100%
BGR Bundesanstalt für Geowissenschaften und Rohstoffe


Tertiary
Cretaceous
Jurassic
1500 m
Shale content
0
100 %
0
10
km
Regionale Tonstudie, Berlin - B 3.21
BGR Bundesanstalt für Geowissenschaften und Rohstoffe
GEOZENTRUM HANNOVER

6. Detailed Characterization of Shales



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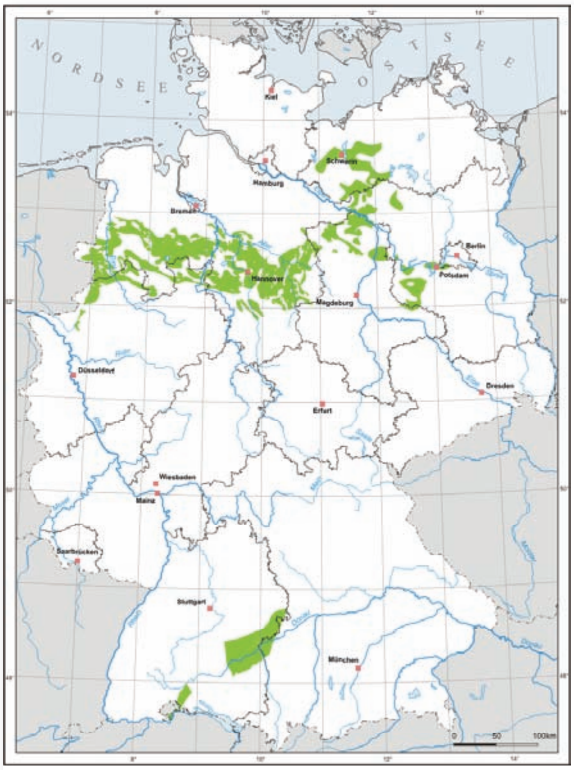

7. Main Evaluation Results

- Regions with Shale formations which are potentially suitable for further investigations in Germany are mapped
- Selection criterias can be easily changed
- Thick and homogeneous claystones and shales exist especially in the Lower Cretaceous and the Middle and Lower Jurassic
- Favorable are shales with a medium diagenetic overprint (coalification values between 0.5 and 1.0 vitrinite reflectance)

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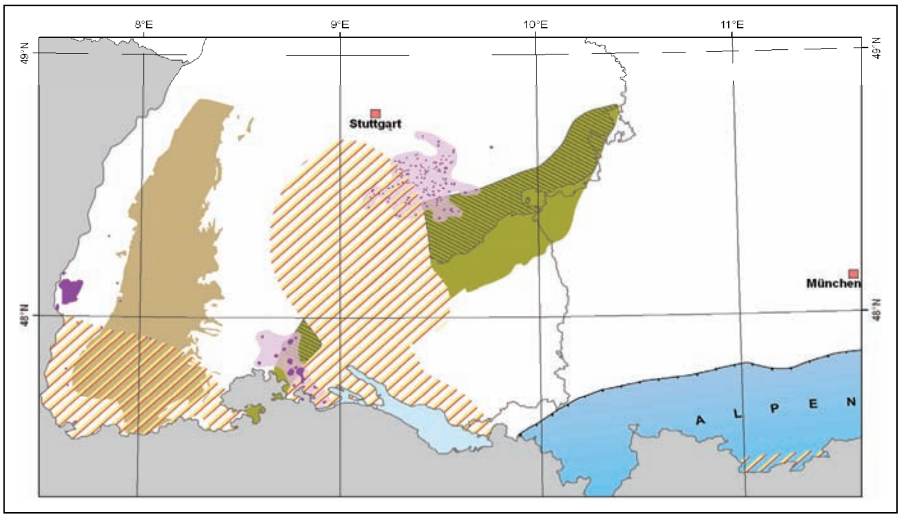
8. Further Restrictions - Examples

- coalification anomalies (vitrinite reflectance > 1,2 %), Tmax shales 30-150°C
- fractured rocks (near volcanic dikes etc.)
- shale gas



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Future Research Activities

R+D	Salt	Claystone	Granite
Charakterisation of potential host rocks	x	**X**	(X)
Geotechnical barriers, Excav. Disturbed Zone	X	X	X
Siting of potential host rocks in Germany	(X)	X	(X)
Safety assessment	**X**	X	--

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Properties	Rock Salt	Clay/Clay Stone	Crystalline Rock (granite)
thermal conductivity	high	low	medium
permeability	almost impermeable	very low to low	very low (not fractured) to permeable (fractured)
stability	medium	low to medium	high
deformation behaviour	viscous (creep)	plastic to brittle	brittle
excavation stability	inherent stability	support required	high (unfractured) low (fractured)
in situ stress	lithostatic isotropic	anisotropic	anisotropic
solution behaviour	high	very low	very low
sorption capacity	very low	very high	medium to high
temperature resistivity	high	low	high

Properties of potential host rocks

favorable

unfavorable

medium

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Components	Rock Salt	Clay/Clay Stone	Crystalline Rock
emplacement level	approx.. 900 m	approx.. 500 m	500 - 1200 m
disposal technique*	drifts and deep boreholes	drifts respectively . short boreholes	boreholes or drifts
design temperature	max. 200° C	max. 100° C	max. 100° C (bentonite backfill)
backfill material*	crushed salt	bentonite	bentonite
duration of interim storage (fuel element container and high active waste container)	min. 15 years	min. 30 - 40 years	min. 30 - 40 years
support	not necessary	necessary, under certain conditions very complex	necessary in fractured areas
waste container concept	existing	to be developed in Germany	to be developed in Germany
mining experience	wide experience (salt mining)	less experience	at lot of experience (ore mining)

* has to be adopted to the host rock

Disposal concepts in different host rocks

favorable

unfavorable

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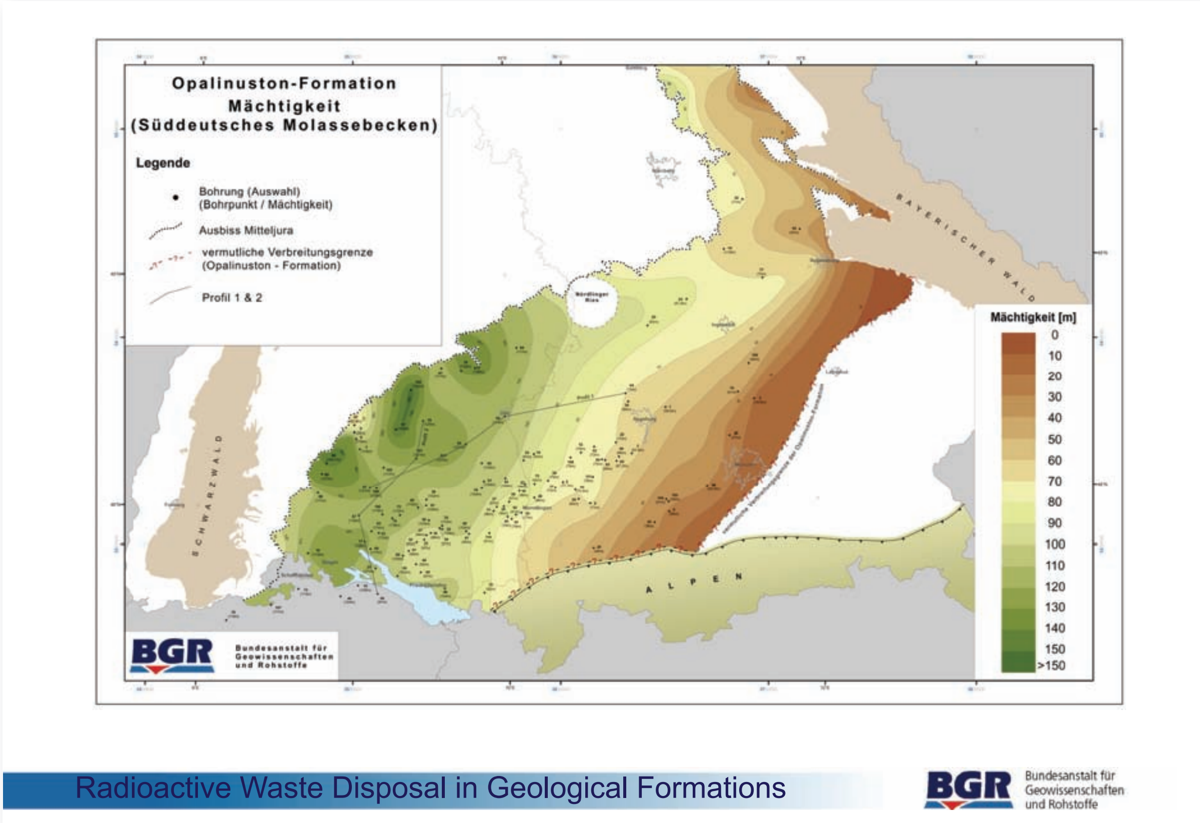
Opalinuston-Formation
Mächtigkeit
(Süddeutsches Molassebecken)
Legende
Bohrung (Auswahl)
(Bohrpunkt / Mächtigkeit)
Ausbiss Mitteljura
vermutliche Verbreitungsgrenze
(Opalinuston - Formation)
Profil 1 & 2
Mächtigkeit [m]
0
10
20
30
40
50
60
70
80
90
100
110
120
130
140
150
>150
BAYERISCHER WALD
SCHWARZWALD
ALPEN
BGR Bundesanstalt für Geowissenschaften und Rohstoffe
Radioactive Waste Disposal in Geological Formations
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2B.14 Demonstrating the Construction and Backfilling Feasibility of the Supercontainer Design for HLW

Hughes Van Humbeeck[1], Chris De Bock[1], Wim Bastiaens[2]
[1]*ONDRAF/NIRAS, Brussels, Belgium*
[2]*EIG EURIDICE, Mol, Belgium*

Abstract

For the final disposal of the vitrified HLW and the spent fuel assemblies in deep Boom Clay layers, ONDRAF/NIRAS is considering the concept of the Supercontainer. In this concept, the waste canisters or the spent fuel assemblies are inserted inside a watertight metallic overpack which is placed inside a thick concrete buffer.

ONDRAF/NIRAS, in collaboration with the EIG EURIDICE, is developing an experimental programme to verify the viability of this concept and to demonstrate the feasibility of constructing the different components of a Supercontainer.

Close to the construction of the concrete buffer, the experimental programme focuses on the filling of the annular gap that remains around the overpack after this has been inserted in the buffer and the emplacement of the concrete lid to close the Supercontainer. The experimental programme includes reduced and large scale tests, planned for 2008 and 2010 respectively.

In parallel, an experimental programme dedicated to the backfilling of disposal galleries is being conducted. A cement-based material has been developed and its emplacement has been tested in a reduced-scale mock-up. The results of this backfill test were encouraging. This test was a preparation for a full-scale test on a 30 m long mock-up that will be conducted in the beginning of 2008.

1 Introduction

ONDRAF/NIRAS, the Belgian Radioactive Waste Management Agency, is responsible of the development of a concept and design for the geological disposal of category B&C waste, which includes long-lived waste (LILW-LL), vitrified high-level waste (HLW) coming from the reprocessing of spent fuels used in nuclear power plants, and spent fuel assemblies as such.

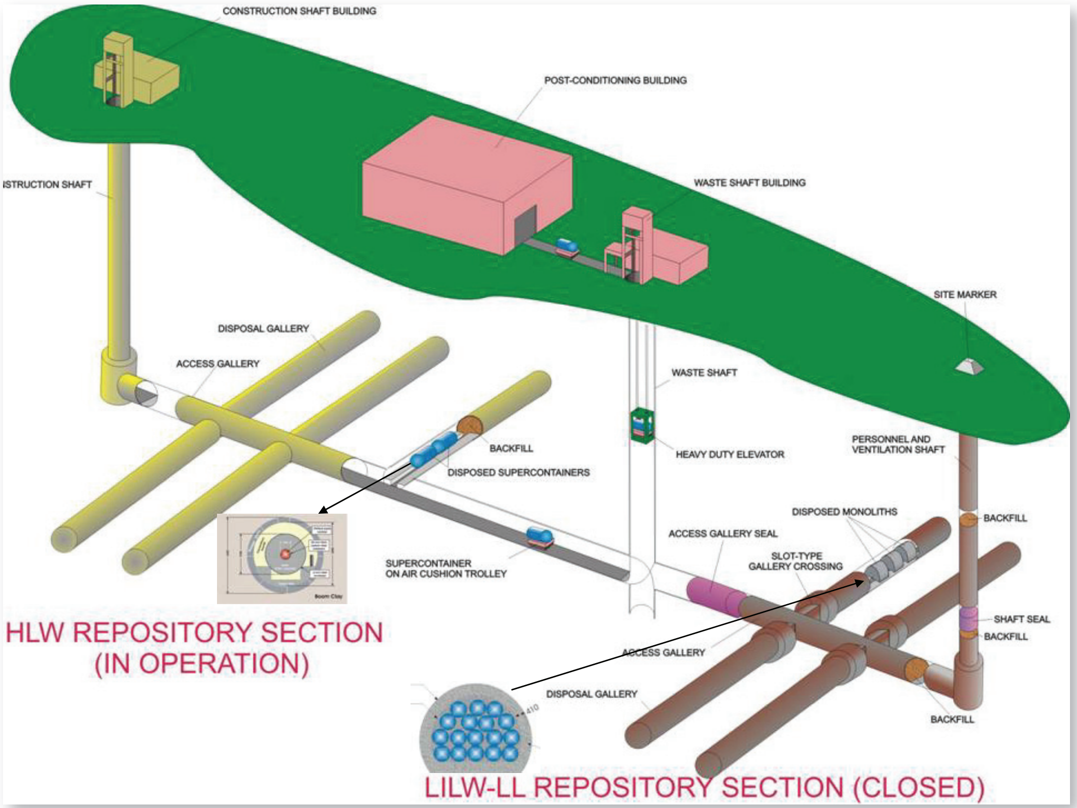
For the disposal of vitrified HLW and spent fuel assemblies, ONDRAF/NIRAS has developed the Supercontainer concept. In this concept, the vitrified HLW canisters or the spent fuel assemblies are inserted into a cylindrical metallic overpack, which is placed inside a thick concrete buffer. Once installed in the disposal galleries, the space between the gallery lining and the supercontainers is backfilled.

In close collaboration with the EIG EURIDICE, an Economic Interest Grouping between ONDRAF/NIRAS and SCK CEN (the Belgian Nuclear Research Centre), experimental programmes are being set up in view of the Safety and Feasibility Case (SFC-1) planned for 2013 [1] to verify the viability of the Supercontainer concept and the feasibility of constructing such a disposal package. In parallel, a testing programme is being conducted to verify the feasibility of backfilling the disposal galleries with a grout and different granular materials.

These two aspects of the Belgian RD&D programme are described in this document. For backfilling aspect, only the backfill tests with grout are described.

2 Description of the Reference HLW Disposal Concept

In the current ONDRAF/NIRAS concept [2], the repository is constructed at depth in an approximately 100 m thick Boom Clay layer, with the overlying sedimentary formations providing the geological coverage. The concept for surface and underground facilities is illustrated in Figure 1, which also shows the emplacement of HLW and LILW-LL in approximately horizontal disposal galleries in spatially-separated sections of the repository. Access to the disposal galleries is through a series of shafts and an access gallery.



▲ Fig. 1: The surface and underground repository facilities

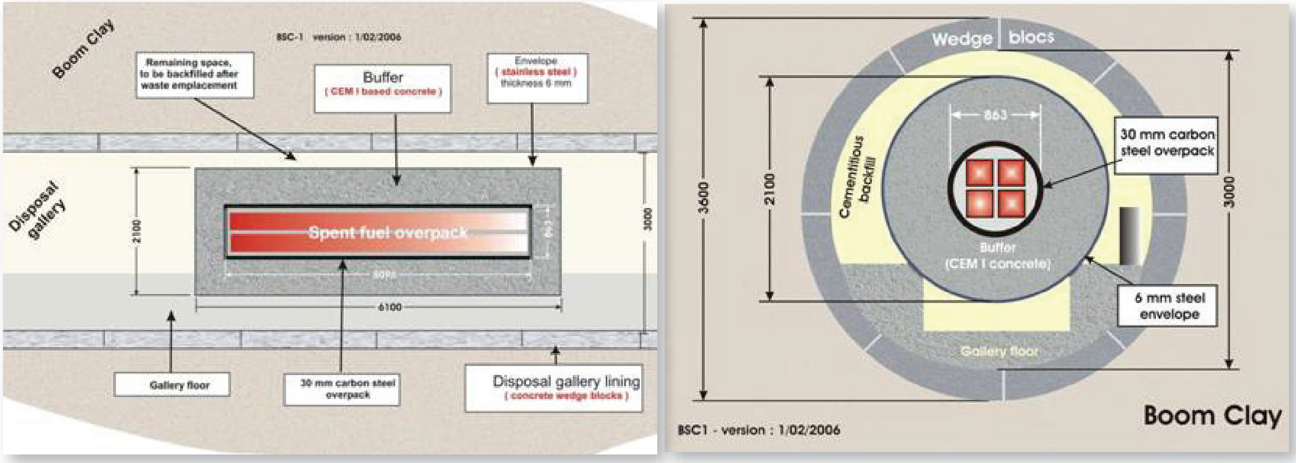
The disposal galleries for HLW and spent fuel assemblies have a length of about one kilometre and an internal diameter of about 3 m. They are lined with concrete wedge blocks to stabilise the excavated galleries against clay convergence.

The design for the disposal of heat emitting waste is based on the Supercontainer concept (Figures 2). In this concept, vitrified waste canisters and spent fuel assemblies are encased in a 30 mm thick carbon steel overpack. Each overpack contains two canisters or four UOx assemblies (but only one MOx assembly). Regarding the long-term safety, the function of the overpack is to provide total containment of the radionuclides during the thermal phase (some hundred years for vitrified waste and some thousand years for spent fuel assemblies) by preventing contact between the waste and water coming from the host formation. The overpack is fitted into a thick Portland concrete buffer, in its turn enveloped by an 8 mm thick stainless steel envelope.

The main roles of the buffer are:

- to create a favorable geochemical environment around the overpack (regarding the corrosion aspect) by maintaining alkaline conditions;
- to provide radiological shielding around the waste allowing human presence during handling and transport of the super-containers without additional radiological protection systems.

With a diameter of about two meters and a length between four meters and six meters, the weight of the supercontainers ranges between 30 and 60 tons (Table 1).



▲ Fig. 2: Supercontainer. Left: Schematic longitudinal view of a supercontainer for spent fuel assemblies. Right: Cross-section of a disposal gallery with such supercontainer.

The waste inside the supercontainers generates a maximum heat output of about 250 W/m.

In the disposal galleries, the supercontainers will be emplaced, one after the other, resting on a concrete floor specifically designed to provide a path for the transportation vehicles (air cushion or rail system) as well as serve as a mechanical support.

After a section of about 30 m is filled, i.e. after emplacement of eight supercontainers for vitrified waste or five supercontainers for spent fuel assemblies, the void remaining void is backfilled. The volume of backfill material to be emplaced is about 80 m³.

Table 1: Supercontainer dimensions (these dimensions are likely to change slightly during further studies)

	Vitrified waste	UOx	MOx
Outer diameter (m)	2.1 m	2.2 m	1.6 m
Outer length (m)	4.1 m	max. 6.2 m	6.1 m
# canister/assemblies per SC	2	4	1
Weight (ton)	30 t	max. 60 t	max. 31 t

3 Experimental Programme to verify the Viability of the Supercontainer Concept and the Construction Feasability

A first study related to the feasibility to construct the supercontainers was carried out by the Belgatom (Belgian research Consultancy Company). In this study it was proposed that a prefab shell would be constructed in normal workshop conditions. This shell is composed of the stainless steel envelope and the concrete buffer. Subsequently, the carbon steel overpack is inserted. Due to the high radiation levels, this action and the following have to be performed under hot-cell conditions using remote control. At this stage, there is an annular gap of ~50 mm wide between the overpack and the buffer; a so-called filler will be applied. Finally, the Supercontainer will be closed using a concrete end piece. The necessity of an additional metallic lid to complete the stainless steel envelope is studied.

A stepwise approach is adopted for the design and feasibility studies of the Supercontainer concept. At first, a desk study was made to define the specifications of the different components of the supercontainer (e.g. buffer, liner, filler). This task was carried out in collaboration with international panels of experts.

3.1. Concrete Buffer Construction: Feasibility Tests

The desk studied resulted in two candidate concrete types for the concrete buffer material: a self-compacting concrete (SCC) and a rheoplastic concrete (RPC). A general composition was proposed for each of these types. Laboratory testing was carried out on several slightly varying mixtures of the buffer concrete to optimise parameters such as slump, workability, compressive strength, etc …

In a next step, two concrete beams were cast at representative scale. One was using SCC, the other RPC. This step was necessary to verify the construction feasibility (risk of cracking …) and to provide material samples for laboratory testing that were produced on industrial scale. A formwork with a height of 6 m and a width of 0.63 m was constructed. The formwork was equipped to measure the temperature during hydration at various locations inside the test beam. The two compositions that were tested are given in Table 2, Figures 3 illustrates the execution of the test. The formwork was filled in three steps (three batches of concrete). The time needed to complete this operation was 65‘ (SCC) and 55‘ (RPC).

Table 2: Composition of the concrete buffer test columns

Type	Material	SCC [kg/m³]	RPC [kg/m³]
Cement	CEM I/42.5N HSR LA (LH)	350.00	350.00
Filler	Calcitec 2001 ME	100.00	50.00
Fine aggregate	Limestone 0/4	840.28	707.86
Coarse aggregate	Limestone 2/6	326.91	413.71
Coarse aggregate	Limestone 6/14	558.55	190.97
Coarse aggregate	Limestone 6/20	0.00	465.17
Plasticizer	Glenium 27/20	14.02	4.41
Water (W/C=0.5)	Tap water	175	175

Fresh concrete samples were taken for immediate testing and testing after hardening. Furthermore, core samples were drilled after hardening of the test beams. Some of the analyses are still ongoing but the main results available are summarised in Table 3.

The results above are the average test results. When looking at individual samples, it is observed that samples taken near the top of the beam show somewhat lower values for density, P-wave velocity, Young‘s modulus and somewhat higher values for permeability. This could indicate to a slightly lower quality of the concrete near the top, possibly due to segregation. However, no variation of compressive and tensile strength was observed. Microscopic analyses are underway to clarify these observations.

▲ Fig. 3: Test column. LEFT: Concrete pumping to fill the formwork. RIGHT: View inside the formwork during the test with SCC, showing temperature sensors.

Table 3: Mean values of test results

Parameter	SCC	RPC	Standard
Bleeding	None	None	NBN B 15-226
Density (at 28 days)	2420 kg/m³	2440 kg/m³	NBN EN 12390-7
Compressive strength	57 MPa	47 MPa	NBN EN 12504-1 NBN EN 12390-3
P-wave velocity	4.69 km/s	4.69 km/s	NBN EN 12504-4
Tensile strength	4.4 MPa	3.3 MPa	NBN B 15-211
Young's modulus	36 Gpa	32 Gpa	NBN B 15-203
Poisson coefficient	0.39[1]	0.16	/
Shrinkage (after 6 months)	420 µm/m	380 µm/m	NBN B 15-216
Thermal dilation (10-60°C)	8.0×10^{-6} °C^{-1}	7.5×10^{-6} °C^{-1}	/
Permeability (water)	2.9 cm³	2.1 cm³	NBN B 15-222

[1]An important amount of dispersion was present on the results, especially those on SCC samples. Moreover, literature values are comprised between 0.15 and 0.20. Consequently, the obtained result should be interpreted with care and another experimental method will have to be applied to confirm or invalidate these results.

3.2 Filler and Concrete End Piece: Small Scale Tests

The prefab shell (concrete buffer) can be constructed in normal workshop conditions because the waste is not yet present. The next step, being the filling of the annular gap between the overpack containing the heat-emitting waste and the concrete buffer, is technically more challenging. Indeed, the inserted waste will generate a significant amount of heat - up to 100°C can be expected at the interface overpack-filler - influencing the installation of the filler.

Two material types are currently considered as filler: a cementitious grout or a powder (e.g. portlandite). To evaluate candidate materials and installation techniques, small scale tests will be conducted under relevant temperature and geometric conditions. For these tests, hot-cell conditions will not be mimicked. The key issues to be studied here are:

- Finding an appropriate combination of material and installation technique;
- Optimisation of the dimensions of the annular gap for this combination;
- Study the interaction between fresh, non-hardened concrete and the heat source (for the grout type only).

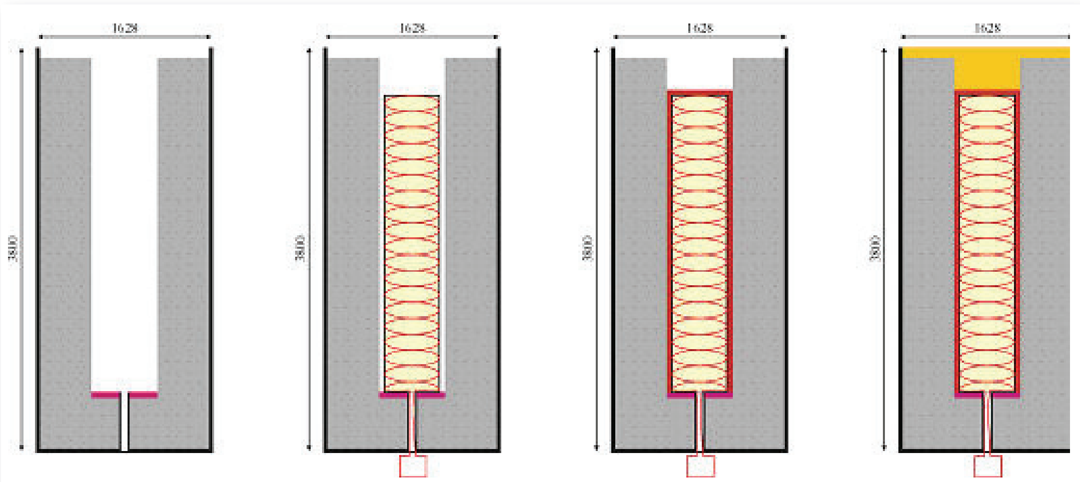
The closure procedure for the Supercontainer construction has not yet been fixed. Various mechanisms will be examined, mainly the choice between using end pieces that are pre-cast or cast immediately in the Supercontainer. Again, the closure of the Supercontainer will have to be performed under hot-cell conditions which impose certain constraints but these will only be investigated at a later stage (cf. section 3.4).

Another item to be tested is the possibility to emplace the filler and the end-piece in one single step. This would simplify operations considerably.

3.3 Filler and Concrete End Piece: Large Scale Heater Test

The small-scale tests and related research will result in a composition for each of the cementitious components: concrete buffer, filler and the end piece. In the next step, a large scale surface test will be conducted. The experimental sequence will be according to the construction planning of an actual supercontainer (see Figure 4):

- Step 1: Construct the outer shell and pour the concrete buffer (for experimental reasons a small opening will be placed at the bottom);



▲ Fig. 4: Overview of the large scale test.

- Step 2: Insert the dummy overpack, equipped with heating elements. The cables of the heating element can be fed through the opening at the bottom;
- Step 3: Switch on heating and place the filler;
- Step 4: Construct and/or install the end piece;
- Step 5: Dismantle and perform post-mortem analyses after several months of heating.

3.4 Prototype Construction

In a last step, a prototype Supercontainer will be constructed. This includes using hot-cell techniques for the installation and positioning of the overpack, placement of the filler, installation of the end piece and (optionally) and closing the Supercontainer with a metallic lid.

4 Experimental Programme to Test the Backfilling of the Disposal Galleries

Two types of material and associated emplacement techniques are currently considered to backfill the HLW disposal galleries:

- Injection of a cement-based material (grout), as a reference solution;
- Projection (dry gunite technique) of a granular material as a potential alternative.

An experimental programme is being conducted to verify the feasibility of these two backfilling techniques. This programme is being carried out in the framework of ESDRED (Engineering Studies and Demonstration of Repository Designs), a technological integrated project within the context of the 6th Framework Program of EURATOM.

4.1 Main Roles and Requirements of the Backfill Materials

The main role assigned to the backfill material is to prevent significant cave-in of the disposal gallery in order to prevent, in case of loss of integrity of the gallery wall, damage to the supercontainer or creep of Boom Clay in the remaining void, destabilizing the host formation. Therefore, the prime operational target of backfilling operations will be to try to achieve a maximal filling of the gap.

The possible backfill techniques and materials are limited by a number of long-term safety and operational feasibility requirements.

For long-term safety, the first requirement is that the backfill should not disturb the corrosion-protective environment of the overpack established by the supercontainer concrete buffer. The backfill should not contain corrosive species, such as reduced sulphur, chlorides, or tend to bring down the high pH of the concrete buffer. Secondly, the backfill should not act as a thermal isolator. Its thermal conductivity should be high enough to prevent the overpack temperature from ever exceeding 100 °C. A minimum value of 1 W/m°C was set. Thirdly, the organic materials (superplasticizers, …) introduced by the backfill must be limited to prevent as much as possible the formation of migration-enhancing complexes between radionuclides and soluble organic compounds. Cellulose-based materials in particular exhibit this problem.

For operational feasibility, the backfill must be emplaceable, that is to say, exhibit the mechanical qualities that will allow it to be pumped or projected into the gap. For a grout, this means it must be fluid enough to be pumped over the length of the drift section to be backfilled and maintain sufficient fluidity during the entire operation. A second requirement is imposed by the needed industrial performance of the process; on average the backfilling must be able to proceed at a certain linear pace in order to limit the total of underground operations to certain duration. For a grout, this means it must become hard within a certain number of days, after which the casing can be removed and the disposal process continued.

For reasons of operational safety, dust generation and water run-back must remain very limited.

A last requirement is related to the option of retrievability. To keep this option open as much as possible, the strength of the backfill component must be limited. A maximum value of 10 MPa was set for the compressive strength. Such value should allow for a removal of the backfill by use of high-pressure beam technology.

4.2 Backfill Tests

4.2.1 Objectives and Methodology

The principal objectives of the experimental programme were to develop a material that met the above mentioned requirements and to verify its emplacement under representative conditions of scale and temperature. The main open questions that the experimental programme has to answer are related to:

- The number of injection tubes needed and their emplacement, which is closely related to the workability of the grout material under thermal load. The use of only one injection tube, installed at the lower part of the volume to be filled (under the supercontainer) constituted the most favorable case.
- The volume and, if present, the longitudinal distribution of the remaining voids;
- The quality of the contact of the backfill material with the supercontainer and the gallery lining (presence of shrinkage …).

From the preparation procedure, more information (equipment performances, presence of dust, volume needed to store the material, possibility to prepare the material in situ or necessity to do it from the surface…) is expected to help to further develop the repository design and operations..

The developed backfill experimental programme is conducted according to the following main stages:

- Development the backfill material carried out between October 2005 and April 2006;
- A backfill test using a reduced-scale mock-up in June 2006 with both main objectives to verify the emplacement of the grout on a length of one Supercontainer and to prepare a full-scale test;
- A backfill test using a full-scale mock-up planned for March 2008. Close to the above mentioned objectives, such a full scale mock-up can be used as communication tool to increase confidence in the feasibility to construct and fit a disposal gallery for HLW.

4.2.2 Composition of the Backfill Material

The backfill material was developed by BASF Chemical Construction (Ham, Belgium). The main challenge was to obtain a material with a sufficient thermal conductivity (and thus a sufficiently high density), an adequate fluidity with as low as possible a quantity of additive.

The result was a material composed by CEM I 52.5 N HSR LA, Limestone powder ($CaCO_3$) from Carmeuse, calibrated river sand 0 - 4 mm, washed and dried and as an additive, superplasticizer Glenium® (polycarboxylate ether based material).

Laboratory tests performed by BASF on the developed material indicated a pH higher than 13.5, and a compressive strength after 28 days of 7.8 MPa. With a density of 2.2, the thermal conductivity was expected (but not measured) to be higher than 1 W/mK.

Normalized gutter tests proved the grout sufficiently fluid to be pumpable over 30 m at a water/cement ratio of between 1.34 and 1.45. The workability period was longer than 3 hours while the hardening time was less than 4 days.

4.2.3 Design of the reduced-scale Mock-up

The reduced-scale mock-up represented a 5 m long and 2 m diameter (2/3 of real life dimensions) section of a disposal gallery. The mock-up was composed as follows (Figures 5):

- The gallery wall is represented by 2 reinforced concrete pipes with an inner diameter of 2 m. Windows on top of the pipes allowed verification of the degree of filling. Both the back end and the front end of the concrete pipes were closed with steel casing;
- The Supercontainer is represented by a 5 m long steel tube, with an external diameter of 1.3 m, placed on a concrete floor cast inside the pipes. The floor had a step-shape similar to the one shown in Figure 2 (right). The central tube was filled with sand to obtain a thermal inertia representative of the concrete buffer of a Supercontainer and heated from the inside by means of an electrical resistance wire. The heat flow was regulated to achieve a temperature of about 50°C at the surface of the tube. This is the calculated temperature at the surface of the Supercontainer during the first month after backfilling of the disposal gallery. During injection, the linear heat power was set at 250 W/m;
- The injection took place from below to avoid segregation of the grout material. In the groove, a steel injection tube (Ø = 3"; L = 3.75 m) was placed through which grout can be pumped at a certain pressure. To allow the grout to rise up into the space of the annular gap, the floor was provided with 2 x 3 elbow holes (Ø = 10 cm). In case of problems with the grout (hardening in the main tube for instance), malfunction to the injection tube, or in case the injection from below proved to be unsuccessful, two reserve injection tubes (Ø = 2.5" ; L = 2.5 m) were mounted on the upper part of the concrete floor. On top of the concrete pipe, as close as possible to the entry point, a venting tube was mounted to allow the air to escape as the gap is gradually filled with grout;
- A Plexiglas pipe was mounted on a reserve injection tube helping to visualize the grout rise during injection;
- The mock-up was thermally isolated with rock wool.

To monitor the evolution of the grout properties, the mock-up was instrumented with temperature sensors (Pt-100), TDR and strain gauges (hardening time), and thermal conductivity sensors.

4.2.4 Grout Preparation and Injection Equipments

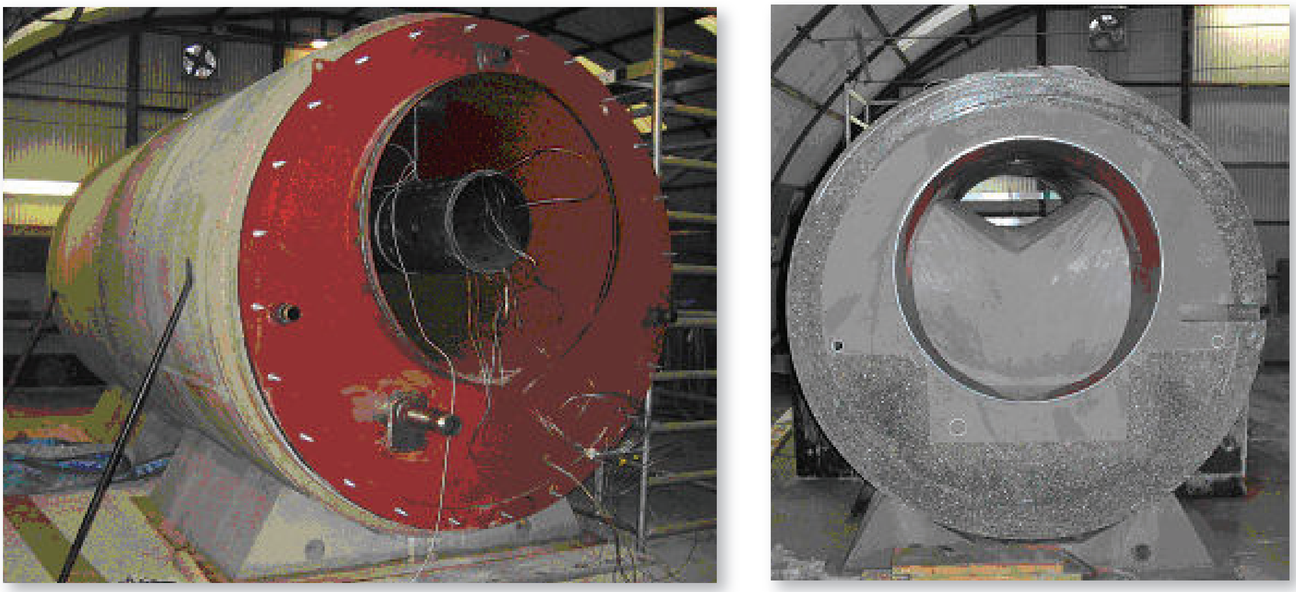
The material was prepared in two 380-liter concrete mixers working simultaneously. The screw pump was mounted with a 500-litre reservoir allowing continuous pumping of the grout. To verify the workability of the grout material over period of at least 2 hours, a maximum flow rate of 3 m^3/h was set.

During the grout preparation, it was decided (BASF engineer expert judgement) to decrease the water-cement ratio from 1.40 to 1.34.

4.2.5 Results and Conclusions of the reduced-scale Test

The results of the reduced-scale test were conclusive. The grout emplacement occurred uninterruptedly. Only the main injection tube installed below the central tube was necessary. It can be concluded that the developed grout had adequate rheological properties, also for an injection under thermal load. The material had sufficiently hardened to remove the casing after four days. Two months after the injection test, a slide was cut (Figure 5) and core bores were taken from the top of the mock-up. It could be observed that a filling rate of 100 % was achieved; the contact of the grout with the central tube was perfect. The distribution of the aggregates of the mortar was very homogeneous. No segregation of the grout occurred.

Compressive strength and the thermal conductivity measured on core bores taken from the mock-up indicated values of respectively 12 MPa and 1.6 W/m°C (for dried samples). The higher than expected compressive strength can be explained by the lower than initially foreseen water-cement ratio decided during the grout preparation. However, the high value of the thermal conductivity associated with the absence of segregation, should allow the target value for the compressive strength



▲ Fig. 5: Mock-up for grout injection test. Left: Mock-up before emplacement of rock wool and sand around the heater. Right: filling obtained after removal of the heater and slicing off.

to be achieved by increasing the water quantity. Consequently, it was decided to maintain the same grout composition for the full-scale test. Water and gas permeability tests are still going on.

The full-scale test is planned for the beginning of 2008. The current preparatory works underline that the most critical issue concerns the grout preparation (mixing time and volume, logistics …). A first indication gained from preparatory steps and having an impact on the design of the repository and the organization of the operations is that it will probably be not possible to prepare the backfill material in the underground installations. The material should be prepared on the surface and injected with pipe though shaft and access gallery into the disposal galleries. ■

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2B.15 Considering Burn-up Credit in the Criticality Safety Analysis for the Final Disposal of Spent Nuclear Fuel

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Abstract

During irradiation of nuclear fuel in the reactor core for power generation, the concentration of fissile material is significantly reduced, and neutron absorbing nuclides (certain actinides, fission products) are generated. Thus, the reactivity of spent fuel is much lower compared to fresh fuel. The amount of the different nuclides resulting from fuel irradiation depends, amongst others, on the type, the initial enrichment and the burn-up of the fuel. However, a certain amount of fissile nuclides remains in the spent fuel, and due to the radioactive decay of both fissile and absorber nuclides with strongly differing decay times, the reactivity of the disposed material does not remain constant within the time frame of 1 million years after disposal. Even an increase of K_{eff} over the time is possible as some investigations have shown [1, 2]. For this reason, a dedicated criticality safety analysis for a final repository is necessary, requiring the consideration of different evolution scenarios of the repository.

Up to now, in many studies on burn-up credit for criticality analyses of spent fuel, the main focus was directed on the determination of the nuclide inventory for a certain burn-up, assuming a uniform burn-up over the whole axial dimension of the fuel rods. But due to non-uniform power and moderator density distributions during fuel irradiation inside the reactor core, this approximation is simplifying and may be not adequate in certain cases. For example in a PWR, at the top and bottom of the fuel the burn-up is significantly lower than in the middle zone, and thus the remaining reactivity is higher there. It can be shown that these lower burnt regions determine the reactivity of the fuel. Due to this fact, this axial burn-up distribution has to be considered appropriately in criticality safety analyses concerning spent nuclear fuel, and thus also has to be investigated for the long term evolution of a final repository.

1 Introduction

Ensuring sub-criticality of the spent nuclear fuel is a basic issue of final disposal. For the operational phase of the repository common approaches of criticality control in transport and storage of nuclear fuel can be applied. For the post-closure phase, long term effects which may change the geometrical configuration due to the hydro-geologic evolution of the disposal site have to be considered. Further on the nuclide composition of the spent fuel and hence the reactivity may change significantly with time due to the radioactive decay. GRS is investigating the problem of ensuring sub-criticality of spent nuclear fuel in a repository since several years [3, 4]. In a current study the issue of taking benefit of the reactivity decrease due to fuel irradiation (burn-up credit) in criticality analysis for a repository is being investigated.

When nuclear fuel is discharged from reactor after power operation, the major part of its fissile material has been consumed for power generation. Although a certain amount of residual fissile material (^{235}U, ^{239}Pu, ^{241}Pu) is still present in the spent fuel bearing the possibility of criticality under certain circumstances, the reactivity of spent fuel is significantly lower compared to fresh fuel. Nevertheless in many cases so far, fresh fuel is assumed in criticality analyses for spent fuel for pursuing an easier applicable conservative approach. But with increasing initial enrichments and fuel burn-up , this approach called “fresh fuel assumption” became overly conservative and too penalizing. Regarding the net reactivity decrease of spent fuel in criticality safety analyses, often referred to as “burn-up credit”, allows for a more realistic approach and thus permits higher fuel densities in storage, packaging and transportation. So the incentive for the use of burn-up credit is a reduced amount of storage space and a decreased number of transportation casks (and thus transports), yielding to financial benefits without decreased criticality safety.

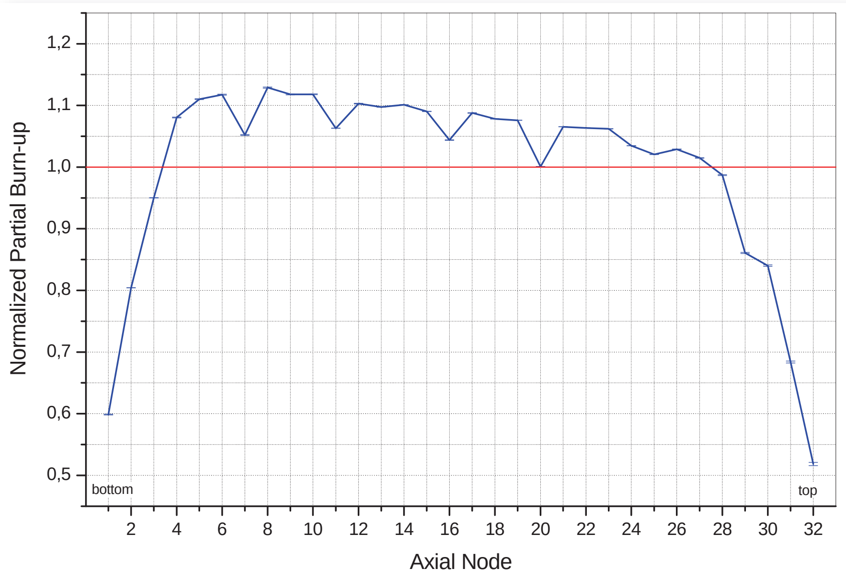
Regarding final disposition of spent nuclear fuel, the incentives for using burn-up credit are similar but nevertheless somewhat different. Besides the arguments mentioned above valid for transportation and interim storage, a more detailed knowledge of the nuclide inventory of the waste is crucial for the overall evolution of the final repository; this matter is not restricted only to criticality considerations of course. During the time frame of 1 million years, due to radioactive decay the concentration of both fissile and neutron absorbing species (some actinides, fission products) changes significantly; to calculate the respective inventories over time, adequate knowledge of the source term, i.e. the inventory at discharge time, as precise as possible at least for the reactivity determining nuclides and/or their mother nuclides within the radioactive decay chains is necessary. This requirement is equal to the tasks involved in common burn-up credit application. Moreover, as real burn-up is not uniformly distributed over the axial length of a fuel assembly, also the inventory changes along the assembly which has to be regarded in the burn-up credit analysis.

Burn-up credit in transportation and interim storage to a various extent is applied in many countries including Germany since several years [5]. To a lesser but nevertheless growing extent, burn-up credit also comes into focus in several countries dealing with the disposition of spent nuclear fuel, e.g. Germany, USA, UK, Sweden and others.

2 Axial Burn-up Profile, End Effect and Calculational Efforts

2.1 Axial Burn-up Profile and End Effect

The burn-up of nuclear fuel depends on the spectra and intensity of the neutron flux distribution inside the reactor core. As this distribution is inhomogeneous, burn-up is a local, non-uniform feature of spent fuel. For each assembly, the burn-up distribution is a unique feature, but in most cases qualitatively very similar. In a typical light water reactor (especially pressurized water reactors), the neutron flux is maximal in the mid plane, decreasing at the top and bottom and at the outer radial zones (rim) of the core due to leakage effects. Relocation of assemblies between cycle downtime reduces the effect due to rim position for the single assemblies, thus remaining lower burn-up especially at top and bottom. The temperature gradient of coolant between inlet and outlet of the core influences the neutron spectra which is hardened at the top of the core due to temperature-induced increase of moderator temperature and thus reduced density. This effect yields in an asymmetry of the burn-up profile. Figure1 shows a typical normalized burn-up profile of a German pressurized water reactor, GKN II (Neckarwestheim).



◄ Fig.1: Typical axial burn-up profile of a pressurized water reactor

Here one can clearly figure out the leakage-induced reduction in burn-up at the ends of the assembly as well as the asymmetry of the profile mentioned above. The dips in the mid-area of the profile are due to the disturbed neutron flux in regions near the spacer grids. Burn-up distribution and extent of asymmetry are dependent on the precise power history, position and the vicinity of each fuel assembly.

The burn-up distribution influences the nuclide inventory distribution and thus the k_{eff} value for every spent fuel assembly. In criticality calculation, one can account for burn-up by using a uniform nuclide distribution resulting from average burn-up, or by using a more realistic modelling accounting for a burn-up distribution. However, the two procedures do not necessarily yield in identical k_{eff} values. The difference between the reactivity of a detailed and a uniform distribution is referred to as "end effect", as it is mainly dominated by the relatively lower burnt assembly ending zones. A positive end effect designates the non-conservativity of assuming uniform burn-up distribution. Depending on the type of fuel (UO_2, MOX), the initial enrichment and the set of nuclides included in the study, the end effect is typically negative for burn-up values lower than a certain problem dependent zero pass around 10 to 20 MWd/kgHM, and becomes positive and in direction increasing for increasing burn-ups [6]. Figure 2 shows the calculated end effect for representative PWR fuel in a generic infinite rod lattice using typical sets of nuclides, actinides and oxygen, and considering ("ACMFP") or omitting ("ACO") fission products (see chapter 4 for details).

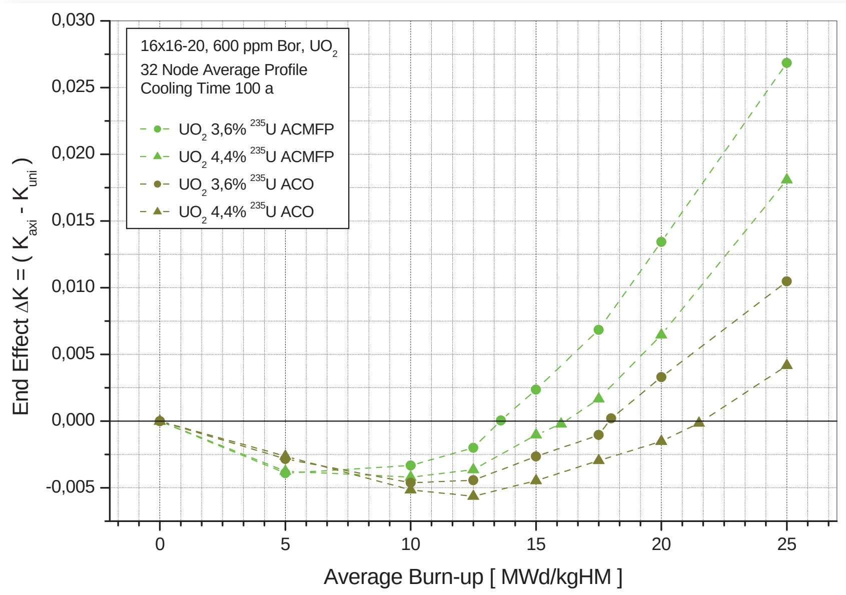


Fig. 2: End effect for typical PWR fuels.

One has to notice that the definition of the end effect requires the fuel rods or fuel assemblies (depending on the storage and/or disposition strategy) to be in an intact configuration. However, when degradation of the waste matrix is proceeding within a final repository, still there are regions in the fuel where the concentration of fissile material is higher than the average. This fact should be accounted for in the criticality safety evaluation of a final repository.

2.2 Calculation Methods

In the "classic" conservative approach for criticality safety analysis for spent nuclear fuel, the fresh fuel assumption is applied as an easy-to-use approach requiring criticality calculations based on the well defined initial composition of fresh fuel, including just the four well known nuclides ^{234}U, ^{235}U, ^{238}U and ^{16}O in a homogeneous distribution; no inventory depletion calculation is needed.

In contrast, applying burn-up credit requires at least one inventory calculation (uniform burn-up distribution along the assembly axis) and criticality calculations including a more or less vast amount of nuclides for which nuclear cross section and decay data are partially barely known and validated. If a burn-up profile is to be applied, the assembly has to be divided into a number of axial zones requiring inventory calculations for each zone. In the international literature, between seven and 32 axial nodes are used practically. These zones bearing different inventories have to be modelled in the criticality model, too.

In the current study, we use 32 axial zones. As inventory calculation code, the GRS home-built reactivity and depletion calculation code OREST 2004 [7] was applied to calculate the zone-by-zone inventory. For criticality calculations, the three-dimensional

Monte Carlo code KENO Va from the SCALE 5 code package [8] was used. An infinite lattice of PWR fuel rods assuming a water/steel reflector at top and bottom of the system was used as generic criticality model to investigate the end effect. KENO Va was also used for any other criticality calculations in the frame of the work at hand.

3 Repository Evolution Scenarios from the Point of View of Criticality

When being disposed in the repository, the spent fuel is in a dry, intact configuration inside a disposal canister. Due to specification of these casks, sub-criticality is ensured at this point in time due to design conditions. However, during the evolution of the repository, degradation of both casks and fuel matrix is expected, especially if ingress of water occurs. In parallel, the nuclide inventory changes due to radioactive decay. Over the time scale of 1 million years after disclosure of the repository, the reactivity of spent nuclear fuel is known to get maximal after a time of about 30 000 years. Up to now, this reactivity increase is not included in the application licence of German transportation and storage casks.

As long as the cask keeps dry inside, criticality is no subject of concern. On the other hand, ingress of water cannot be excluded some time after closure of the repository, and thus has to be evaluated. This situation might lead to a flooded cask with highly increased waste form degradation. Increased moderation also increases the reactivity of the waste. Moreover, in the following nuclide migration even inside the cask might separate fissile nuclides from absorbers and thus locally increase the reactivity of the configuration. At this point the higher fissile content and lower absorber content at the top and bottom zones of the assemblies come into focus. Selective separation of these top/bottom zone materials can yield in a higher k_{eff} than an averaged inventory does. Different credible scenarios have to be taken into account, and have to be analysed if they are connected with a reasonable probability to appear.

To evaluate the potential for the formation of a critical configuration in the time frame of 1 million years after disclosure of a repository, including identification of scenarios leading to a critical configuration of fissile material, combined with the determination of the probability of the respective scenarios, is a sophisticated task which is being discussed within the international community of criticality experts, see for instance various contributions to [9] (Vol. I pp309ff, 326ff, 380ff, .391ff, Vol. II pp168ff, and others).

4 Calculation Results

To demonstrate the importance of the end effect, a generic study was undertaken using infinite rod lattices. Rod parameters as for instance diameter, length, pitch are based on a Biblis-type 16x16 fuel assembly. For this study at top and bottom ends, a 5 cm water plus 50 cm steel reflector are assumed.

Inventory calculations are based on UO_2-fuel with 3.6 and 4.4 wt% initial enrichment and an average burn-up of 40 and 55 MWd/kgHM. The axial burn-up profile shown in figure 1, renormalized to an average of 40 or 55 MWd/kgHM was used, and calculations for uniform nuclide distributions according to this burn-up value were performed. Realistic and conservative assumptions regarding the reactivity of the spent fuel were applied for reactor operation conditions for the inventory calculations.

For the criticality calculations, nuclide sets “actinide only” (ACO) and “actinide plus major fission products” (ACMFP) are regarded in each axial zone. The former consist of the actinide nuclides ^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{237}Np, ^{241}Am, ^{243}Am and ^{16}O, while the latter includes those actinides plus the long-lived fission products ^{95}Mo, ^{99}Tc, ^{101}Ru, ^{103}Rh, ^{109}Ag, ^{133}Cs, ^{143}Nd, ^{145}Nd, ^{147}Sm, ^{149}Sm, ^{150}Sm, ^{151}Sm, ^{152}Sm, ^{151}Eu, ^{153}Eu and ^{155}Gd. Note ^{16}O being neither actinide nor fission product but nevertheless a major component of the fuel matrix thus being included into the analysis.

Figure 3 shows the corresponding multiplication factors K_{eff} within a time frame of 25 to 1 million years of cooling time, while figure 4 shows the respective end effects.

Figure 3 shows a minimum in k_{eff} at about 100 years of cooling time in any case under scope, as well as a maximum of k_{eff} at about 30 000 years after discharge from reactor. The absolute magnitude of k_{eff} in this study is higher in the corresponding

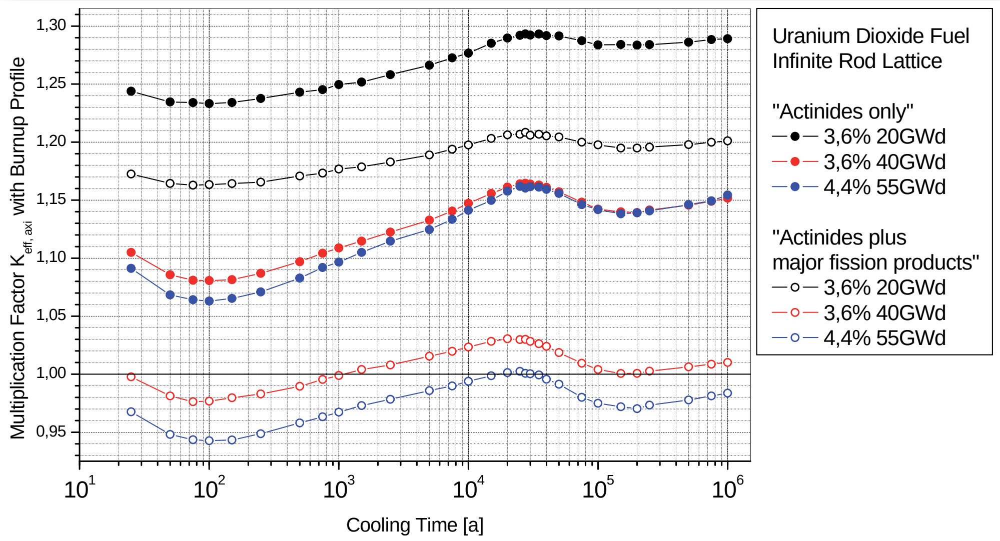


Fig. 3: Multiplication factors keff for infinite rod lattices (see text) in dependence of the cooling time.

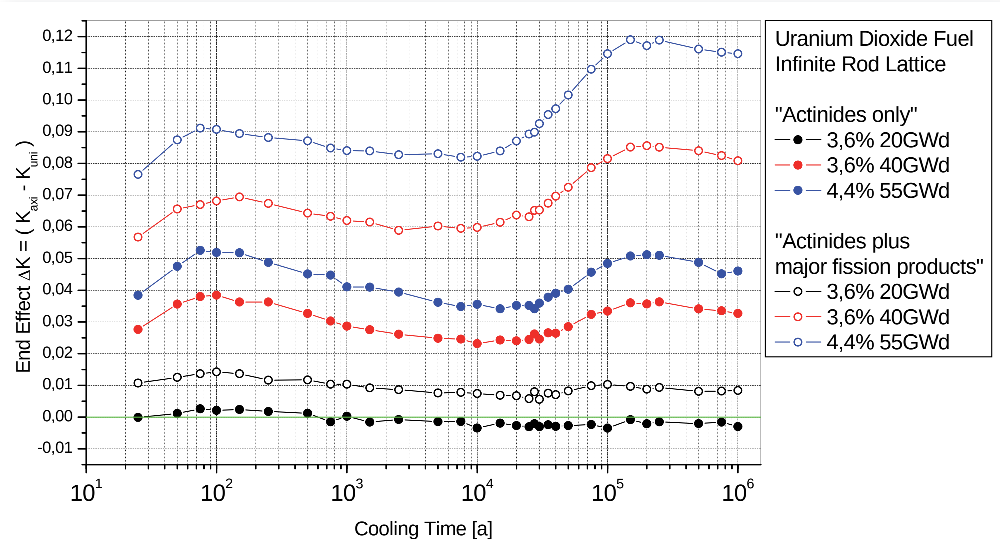


Fig. 4: End Effect Δk for infinite rod lattices (see text) in dependence of the cooling time.

cases for ACO than for ACMFP which is obvious: neglection of absorbing fission products increases the computed reactivity. Moreover, for the 40 and 55 MWd/kgHM cases, k_{eff} is higher when accounting for the axial burn-up profile. In these cases the end effect is positive, also to be figured out in figure 4.2 for the corresponding systems. For a low burn-up of 20 MWd/kgHM, the end effect is much lower than is in the corresponding 40 MWd/kgHM case, which was to be expected following chapter 2.1.

The behaviour of k_{eff} in dependence of cooling time is due to the decay times of the different plutonium isotopes, which give a major contribution to the reactivity of the spent fuel, both increasing (^{239}Pu, ^{241}Pu) and decreasing (^{240}Pu), and ^{241}Am. Note that the decay reaction $^{239}Pu \rightarrow {}^{235}U + \alpha$ leads from one reactive fissile nuclide to another one—however somewhat less reactive.

Comparison from figure 3 to 4 shows the end effect mainly being minimal when k being maximal, and vice versa. This phenomenon is also due to the half lives of the different Pu isotopes, combined with the different initial contents of the zones of different partial axial burn-ups. A zone of higher partial burn-up typically bears a higher Pu content and thus shows a more concise alteration with cooling time. Especially, the reactivity determining top end of the system including axial burn-up profile has a lower burn-up than the homogeneously averaged uniform system, and thus lower Pu content inducing lower change of Pu, contemporary with a higher remaining ^{235}U content (also showing a much longer half live than the Pu isotopes) from initial. Consequently, the top end of the system including the axial burn-up profile shows the higher reactivity but lower change in reactivity in comparison to the higher burn-up average uniform distribution during process of time. This results in an end effect yielding to the opposite direction than the multiplication factor.

This generic study cannot be assigned to realistic cask configurations directly, but nevertheless imposes the necessity to keep in mind and focus the top and bottom ends of the fuel assemblies with their different nuclide inventories and, regarding criticality safety, higher partial reactivity. For certain evolution scenarios of the final repository this matter may be of relevance. The work on this topic is still ongoing.

5 Summary

The regard of the net reduction of the reactivity of spent nuclear fuel compared to fresh fuel, often referenced as burn-up credit, has been a matter of investigation in the framework of criticality safety in transportation and interim storage for almost two decades. Recently, burn-up credit comes also into focus in the field of final disposition of spent nuclear fuel.

Formerly, a uniform homogeneous nuclide distribution along the fuel rods, respective to the average burn-up of the rod, has been considered. Various investigations have this assumption shown to be non conservative for average burn-ups higher than 10 to 20 MWd/kgHM, according to analysis basic conditions, respectively. In configurations with intact rod systems, the so called end effect gives a measure of conservatism of the assumed axial burn-up distribution.

However, consideration of an axial burn-up profile of spent fuel assemblies is not only an important issue when regarding intact transportation and storage casks, but also in the consideration of long term evolution scenarios of a final repository, due to the undeniable existence of fuel rod/assembly parts bearing a distinctly higher residual reactivity than the axially averaged inventory.

With modern calculation tools and some sophistication on the scenarios assumed, this phenomenon can be in principle accounted for in the criticality safety analysis of a final repository for spent nuclear fuel. In view of the difficulty of considering the irradiation history of every single fuel element for the disposal criticality analysis, a conservatively applicable bounding approach has to be developed for this issue. ■

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Session 3

Specific Aspects
of HLW-Disposal

3.01 Applied Basic Research in the Field of Radioactive Waste Disposal in Germany. Key Subjects of Future R&D (2007-2010)

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Geological disposal is accepted as a method of providing for the long-term safety and security of radioactive waste, and it can be implemented safely with current technology. An international framework has been established to promote and assess the safety of radioactive waste management activities, including geological disposal.

In modern societies, however, the implementation of any major new technological project, besides proving its technical merits and safety, must also satisfy societal and policy requirements; this is a particular challenge in Germany. Thus, while the goal of deep geological disposal is widely accepted, the path towards implementation depends on a variety of factors including the national policy and legislative framework, economic conditions, and societal or cultural approach.

Progress towards the implementation of geological disposal is now being made in several countries, but in Germany progress is slower than expected.

There is a practical need to move forward with geological repository projects in many countries; communication of the shared understanding can have a significant impact on progress. On the other hand, policies and projects that lack public understanding or acceptance are liable to fail. Therefore, an open dialogue, which explores the national issues of concern, will promote a shared understanding of challenges and could promote public understanding.

The federal government has the duty to ensure the effective and safe management of radioactive waste and of spent nuclear fuel; this is formalised in the international Joint Convention on the Safety of Spent Fuel Management and the Safety of Radioactive Waste Management administered by the IAEA.

Within these constraints, national or federal governments are responsible for setting policies with respect to nuclear power and radioactive waste management, noting that the need to deal with radioactive waste already produced or committed is independent of future nuclear power programmes.

Today, in modern democracies, controversial scientific developments and major technical projects are valid subjects for public debate, and the decisions on whether and how to implement such developments or projects must take account of societal concerns and opinions.

Today radioactive waste partnerships, involving both community and technical stakeholders, are working together to consider how to build durable relationships. The dialogue between communities and technical stakeholders and decision-makers is an important development that is expected to mature as more countries move to siting and development of geological repositories.

Since it was suggested in the late 1950s, the concept of geological disposal and possible realisations of the concept have been investigated through extensive research, development and assessment programmes in many countries. The overall conclusions of this work are that the concept is scientifically sound, the technologies to site, design, construct, operate and close a geological repository are available, and safety can be provided at all stages of the development, including in the long term after repository closure.

Although the concept and its general feasibility with present-day technology are established, technical issues will arise concerning the implementation within the geological conditions that a particular site may offer. Also, developments in underground construction and sealing, waste technology, material sciences, monitoring etc. may occur over the timescale of a project. And, as information is gained through work in the underground, a more confident and detailed assessment can be made of the long-term performance of the repository system, which in turn may influence design decisions.

Hence research referring with design and construction being adaptable to geological conditions, developments in technology and findings from safety assessments is widely accepted as a technically advantageous approach. The stepwise approach also provides opportunities for stepwise political and legislative decision making, which may involve a wide range of stakeholders and allow ordered resolution of legal, political, social and local community issues.

With its current promotion concept which I will present now, the BMWi would like to contribute to this conviction work.

Element of funding of the application-oriented basic research of the Federal Republic of Germany in the area of the final disposal of high-radioactive wastes is the new BMWi-Foerderkonzept for the years 2007 - 2010. It updates in April 1998 and in November 2001 published promotion concepts for the disposal of radioactive wastes and updates the possible research and development topics. It considers in its adjustment the results of the national and international research and development of many years for the disposal of radioactive wastes as well as institutionally financed activities. The promotion concept is co-ordinated with the institutions working in Germany considerably in this area.

So far were goals of the application-oriented basic research:

- realization of a final repository for heat producing high-radioactive wastes as well as burned down nuclear fuel by the purposeful advancement of the conditions from science and technology.
- creation and advancement of scientific-technical authority and the contribution to their receipt within the range of the nuclear disposal in Germany.

All so far won scientific-technologic knowledge confirms that the realization of a final repository for heat developing wastes is generally possible especially in rock salt. This fundamental knowledge supports the feasibility of the safe disposal in a concrete project. Beyond that the future research work is to contribute regarding the paramount meaning of final disposal security to

- improve the statement security from analyses
- increase the accuracy of in addition necessary parameters
- treat specific question positions with the goal of achieving a higher robustness of the overall system
- examine the results of the computational simulation of complex systems by laboratory and field tests experimentally
- constantly further develop the procedure of the production of the Safety Case for plausibility and systematic
- adopt the conditions of the science practically applicable.

The application-oriented basic research concentrates therefore regarding a brisk and purposeful solution of the final disposal question on the following emphasises:

- development of safety record concepts and creation of the necessary bases for systems analyses, which are conformal with the requirements of a plan statement procedure
- transfer of the technical and other solutions into the state of the art, including the measures for safeguard control, necessary for the establishment, the enterprise and the closing down of a final repository
- support of scientific-technologic authority on high level.

The cooperation in selected projects in European Underground Research Laboratories, aiming to the improvement of the system understanding for the safety evaluation and to the test of technical equipment and procedure, has further high value. Emphasis of the investigations in URL is in clay stone. Work in URL in crystalline rocks (granite) is continued referred on specific questions.

Knowledge conditions actually represent a strong basis for starting a licensing procedure for the establishment of a final repository. At this point the changed emphasis and requirements will lead to a reorientation of the application-oriented basic research according to progress of the final repository project. In the phase from 2007 to 2010 with highest priority R&D work on non-accomplished questions of the final disposal in rock salt will be supported. Parallel to this work the scientific-technologic level of knowledge referring to the disposal of heat developing wastes in clay stones is to be brought on if possible high conditions. Questions for disposal in crystalline rocks are to be worked on supplementing only.

These changed adjustment and objectives will make substantial efforts and the appropriate equipment of the research establishments with resources (personnel and means) necessary. In the phase 2002 to 2006 sixty-five R&D projects have been granted, 21 of which are in the meantime finalized. 28 projects have running times into the year 2008 and beyond that. Before this background it becomes clear, which extent of work in a four years longing phase of the funding concept can be carried out and which results for the practical conversion in final repository projects are available for successful conclusion of the R&D projects presupposed.

Referring to the funding concepts published so far the topics were divided into the following ranges of topics:

- A final disposal concepts and disposal subranges
- B data and instruments for the safety analysis
- C safeguards control

As in Germany no underground rock laboratories in clay stone or crystalline rocks (granite) exist, in-situ investigations are only abroad to be accomplished.

- Clay/tone/clay stone: Mont Terri/Schweiz, URL Mol/Belgium, Tournemire/ France, Meuse Haute-Marne/Frankreich (Bure)
- Granite: Rock laboratory Grimsel/Schweiz, HRL Aespoe/ Schweden.

The underground rock laboratories in clay stone and crystalline rock, which are earliest representative for German conditions, are Mt-Terri, Bure and Aespoe.

A Final Repository Concept and final Disposal Sub-Ranges

A1 Final Waste Disposal Concepts

- Optimization of the final disposal concepts for heat developing wastes in rock salt
- Development of generic final disposal concepts for heat developing wastes in clay stone
- Confrontation and evaluation of generic final disposal concepts on the basis of scientific-technologic, safety analytic and sociological aspects
- Advancement of the conceptual safety record
- Concept testing for storage at a double borehole in the clay stone

A2 Wastes and Bundle (Technical Barriers)

- Advancement of adjustment matrices
- Evaluation of the relevance of chemo+toxic waste components for long-term security

A3 Ultimate Waste Disposal Technology

- Development and testing of a storage concept for un-cut fuel rods in boreholes in particular in rock salt
- Examination of the transferability of the storage handling and monitoring technology of rock salt on clay stone and crystalline rocks
- Development, optimization and testing of procedures for the monitoring of final disposal during the operating phase
- Examination of the effects on vertical storage drillings in salt

A4 Host Rock Characteristics

- Collection, description and evaluation of the final disposal relevant characteristics of rock salt and clay stones in particular regarding the damage and healing behavior as well as regarding the technical feasibility
- Collection, description and evaluation of the final disposal relevant characteristics of rock salt and clay stones regarding microbial effects
- Development and improvement of geophysical procedures for the non destructive host rock investigation with larger range, in particular for the detection and characterisation of inhomogenous ranges in rock salt and clay stone
- Examination and development from analytic methods to the characterisation of inhomogeneities and transferability of effective parameters on larger ranges (Upscaling)

A5 Geotechnical Barrier System

- Conception, building and testing of long-term-safe buildings and/or their components for final disposal in rock salt
- Co-operation at conception, building and testing of long-term-safe buildings and/or their components in clay stone development of concepts for sealing systems within the range of the contact and dispersal zone with proof of the barrier effect
- Engineering proof of the long-term sealing effect of the geotechnical barrier system
- Investigation and selection of suitable filling and buffer materials for final disposal in rock salt and clay stone in particular regarding the behavior in relation to gases and liquids as well as their sorption behaviour
- Selection and investigation of suitable materials for the retention of not or hardly sorbing if necessary dose determining radionuclides in the nearfield of a final repository

A6 Separation and Transformation (P+T)

- Investigations for the effect of P+T scenarios on the final disposal concepts
- Development from procedures to the Pu-disposal
- Advancement of technologies for separation and following transformation of plutonium and actinides as well as evaluation of the transmutations- and safety-potential of the overall system “P+T”

B Data and Instruments for the Safety Analysis

B1 Scenario Development

- Composition and evaluation of FEP‘s (feature, events, processes) for final disposal in the host rocks rock salt and clay stone
- Systematic derivation, classification and evaluation of scenarios for the future development of final disposal in rock salt and clay stone

B2 Geological Barrier

- Investigation and modelling of the geo-mechanical, geochemical and thermal-hydro-mechanical behaviour of rock salt and clay stones
- Investigations for the impairment of the isolation potential of the geological barrier by geo-gene and techno-gene effects, also with consideration of the emergence and propagation of gases
- Investigations and modelling of the radionuclide back attitude with consideration of chemical and microbiological processes

B3 Geochemical-technical Barrier

- Investigation of the chemical, physical and microbiological effects and processes in the near field with effect on the mobility of radionuclides and development from models to the description of the processes happening in rock salt and clay stone
- Security and extension of the thermodynamic database for actinides and fission products on near field conditions

B4 Technical and Geotechnical Barriers

- Investigation of the emergence, the release, the transport and the whereabouts of gases and development of process of describing models as well as investigation of the effects on the behaviour of backfilling or buffer materials
- Effects of gassing on the function of geotechnical barrier buildings
- Investigations for the reciprocal effect of vitreous wastes with clay stone and buffer materials and development process of describing models

- Modelling and quantification of the radionuclide source term on the basis of corrosion investigations at waste products and burned down fuel elements (UO_2, MOX, MTR, HTR)
- Modelling of the long-term behaviour of barrier buildings and/or their components

B5 Methods and Numeric Simulation

- Advancement and actualization of the methods and computational programs for long-term safety analyses, in particular on the basis of defined scenarios, modified disposal conceptions and new realizations from R&D work with consideration of the international development
- Improvement process of describing models, in particular by development of coupled models

B6 Qualification of Models and Reduction of Uncertainties

- Application of existing or development of new methods and procedures for the qualification from models to the description of thermal, geo-mechanical, hydraulic and geochemical processes (THMC coupling)
- Qualification of groundwater flow and transportation models by laboratory and in-situ investigations
- Examination of the transferability of the statements of laboratory tests on technical school and material systems
- Improvement of statement security of long-term safety analyses
- Development of methods to the reduction of uncertainties
- Investigation of relevant processes and/or relevant indicators in natural or historical, anthropogen systems, which contribute to the increase of the confidence in the process understanding or the simulation results

B7 Systems Analyses

- Advancement of the systems analysis for an ultimate waste disposal for heat developing wastes in rock salt
- Orienting systems analysis for final disposal for heat developing wastes in clay stones

C Safeguards Control

An organization into conceptional, technical-methodical and political-institutional ranges requires the special structure and problem of the international safeguard control for execution of necessary research and development programmes. The experiences of the past showed that all these aspects must be united and treated for a successful application with nuclear material controls. Further is important to embed these R&D projects into research networks which have this special authority. Examples for this are available in the support program of the German Federal Government for the IAEA. The employment of the developed techniques, methods and politics with the IAEO or with EURATOM requires a multinational co-operation with the different R&D activities. The following specific points of program should be treated in the next years.

C1 Conceptional Aspects

- advancement of the supplementary protocol
- safeguards for the dumping of burned down fuel elements
- new inspection regime
- consideration of audit principles in Safeguards-approaches
- New Safeguards-Concepts for the nuclear fuel cycle

C2 Technical-methodical Aspects

- monitoring by means of remote sensing
- environmental monitoring
- geophysical measuring procedures
- new verification techniques
- new electronic monitors and long-distance data transmission
- advancement of digital camera and seal systems and mechanism for a tele-call
- Information analysis and data processing for Safe-guards

C3 Political-institutional Aspects

- proliferation-resistance
- multinational plants
- Fissile Material Production Cut off Treaty (FMCT)

3.02 The German Radioactive Waste Disposal Conception in the View of the German Utilities

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Mister Chairman…, Ladies and Gentlemen,

I am delighted to have been given the opportunity to speak at this conference in Braunschweig today about the German conception of radioactive waste disposal, as seen from the perspective of the utilities and GNS – their disposal company responsible. Besides the question of safety, the solution of the final disposal issue – i. e. the final repository – is of crucial importance for the acceptance of nuclear energy for electricity production.

Dr Appel will certainly discuss this in more depth in his presentation and will surely shed light on some interesting aspects. A very decisive point for creating public acceptance is, in my view, the consistent pursuance of a clear conception that is well secured both technically and scientifically, and which is supported by more or less all the relevant (specialist) players.

I personally believe that we in Germany have such a good and clear conception. The problem, as I see it, lies in achieving the widest possible consensus between the experts on the procedure to be followed, and thus to create the basis for public acceptance of the final disposal solution pursued.

I would like to discuss this in more depth in my presentation, and also to make a suggestion of how we can constructively reach a solution. Allow me to start with one remark. In numerous discussions I have also had with opponents of nuclear energy, I have obtained the impression that the feasibility of nuclear waste disposal, such as we in Germany have long aspired to, is not seriously in doubt. In my view, opposition to the planned waste disposal solution is therefore mainly a war by proxy, the subject of which is really the issue of nuclear energy use in general. It should nevertheless be clear to all involved that nuclear waste disposal must be solved, irrespective of the question how we continue with nuclear energy. And the sooner the better.

This is also not a problem of quantity. If the phase out of nuclear power utilisation proceeds in Germany as scheduled, we are talking about a total quantity of **only** around 300,000 cubic metres of “Konrad-grade” waste and around 16,000 tonnes of heat generating waste collected over a time period of approximately 60 years. This is vanishingly small compared to the **annual** amount of conventional problem waste occurring in Germany of around 20 million tonnes, of which 300,000 tonnes have to be deposited underground – partly with an infinite half-life period.

I freely admit, however, that all those responsible for waste disposal in recent years have done much too little to ensure an open exchange of information and discussion with the interested public.

Having made this preliminary statements, I would now like to go into the following aspects:

- A: The German waste disposal conception or Is nuclear energy use in Germany an aeroplane that has taken off without a landing strip?
- B: The “polluters pay principle” and “generation fairness”
- C: Current status and a possible way forward.

A The German Waste Disposal Conception – Is Nuclear Energy Use in Germany an Aeroplane that has taken off without a Landing Strip?

In the Federal Republic of Germany, the issue of nuclear waste disposal was already given due consideration parallel to the development of nuclear energy for industrial use. This was reflected as early as 1957 in a memorandum of the German Atomic Commission, in which attention was drawn to the necessity of safe disposal of radioactive waste. The legislature picked up on this stimulus and laid down specifications in the Atomic Energy Act of December 1959 for the harmless recycling of radioactive materials. The Bundesanstalt für Bodenforschung (Federal Institute for Geological Research), today the Federal Institute for Geosciences and Natural Resources (BGR), recommended final disposal in rock salt formations as early as 1963. The reasons why rock salt was recommended were, amongst other things:

- its good convergence behaviour
- its high specific thermal conductivity
- its geomechanical properties
- the good knowledge/experience with salt mining in Germany
- the availability of numerous unused salt domes in northern Germany, which have remained practically unchanged over the last 240 million years
- international knowledge.

The federal government determined in 1964/1965 that final disposal in Germany should be carried out in deep geological formations. In 1976, the final disposal of radioactive waste was anchored as a task of the state in the Atomic Energy Act. One year later, after an intensive search for sites by the federal government and the state of Lower Saxony, the state government of Lower Saxony named Gorleben as the site for a final repository within the framework of a nuclear disposal centre. This was preceded by a three-stage selection process in which more than 140 sites were considered. The work of investigating the site started in 1979 in consensus between the federal government, the state of Lower Saxony and the concerned municipalities.

At this point I should also like to point out that the project, at least in the initial phase, was accompanied by what at the time was an unusually high degree of public involvement. I am thinking here amongst other things of the Gorleben Commission, the Gorleben Hearing of 1979 and various informative events that were conducted in the 80‘s. I also consider it important to remember that the investigation of the salt dome in Gorleben as a radioactive final repository continues to be accepted by the local community up to this day. Since the start of the moratorium in October 2000, they have repeatedly and publicly called for the investigation to be concluded as soon as possible, in order to be clear on the suitability or unsuitability of the site.

It is undisputed that so far there are no discoveries that speak against the suitability of Gorleben and the technical feasibility of the final disposal. This was even confirmed by the nuclear-critical Red-Green federal government in the agreement of 2001 with the utilities. Let me make some remarks on the Gorleben site selection process and the request to start a new one:

I believe it is no exaggeration to say that the German procedure for the selection of Gorleben is the basis for many of the final repository site selection procedures running today.

However, it is no surprise that today, around 30 years after the beginning of the search for a final repository site in Germany, the involvement of the public takes place in accordance with a different procedure, a more modern procedure from today‘s point of view. But this does not disqualify a suitable location – irrespective of how it was found. Against this background, the demand repeatedly voiced in Germany in recent years, that the site selection should begin afresh with the aim to find the best possible site by means of comparison, is not only irrational but also serves no purpose.

I know of no other country that is conducting a comparative exploration of sites with the aim to find the best one.

The **point** is rather always to find a **suitable site**! Apart from this I belive that it is scientifically impossible to find the objectively best possible site by means of comparison, since

- a final repository represents a structure that cannot be standardised;
- many site-specific influencing factors must be taken into consideration;
- safety is dependent on the interrelations of technical, geotechnical and geological concepts;
- within the framework of a long-term safety analysis, in the end small differences would have to be compared connected with large uncertainties.

Therefore the decision in Germany so far was and is correct to first finish exploring an apparently suitable site (here: Gorleben), and only to resort to alternatives in the improbable event of actual proved unsuitability.

In this context – and because this issue was raised by Minister Gabriel during this opening speech on Thuesday – I would like to print out that the underground exploration in Gorleben started only after safety criteria for final disposal in deep geological formations have been published by the BMI, the federal minister in charge at that time.

Coming back to the German waste disposal concept. Parallel to the exploration of the salt dome in Gorleben, the former iron ore mine shaft known as Konrad was examined regarding its suitability as a final repository for non-heatgenerating waste. After the positive conclusion of preliminary examinations in 1982, the PTB (precursor of the Federal Office for Radiological Protection, BfS) applied for a licence as final repository. This took place in consideration of the planned extension programme for nuclear energy in Germany and of the large quantities of waste expected in this regard. Over time, it also transpired that Konrad could be realised earlier than a final repository for high radioactive waste in Gorleben. To summarize this point.

Conclusion

Parallel to the utilisation of nuclear energy, a plausible waste disposal concept was developed in Germany, the realisation of which was begun early. The realisation faltered, however, when the further utilisation of nuclear energy was called into question (➛ war by proxy). This means that the landing strip is feasible but politically delayed!

B The “Polluters Pay Principle” and “Generation Fairness“

Nuclear energy was a politically desirable energy option in the 60‘s, just as renewable energy is today. It was considered to provide the solution to all energy problems, the way out of the increasing scarcity of resources expected, and the improvement of the economic situation and living standards of people in Germany. This was the object of political and social consensus.

Against this background, the federal government of the time intensively urged the industry to develop nuclear energy. The tasks of large-scale technical realisation through the development of high-performance nuclear power plants and the realisation of waste treatment and disposal technologies (e.g. reprocessing plants etc.) fell to industry.

In turn, the federal government took over the task of fundamental research, and also the realisation of the final repository. The latter was first fixed in writing in the Atomic Energy Act in 1976 (article 9a paragraph 3). **But** from the very beginning the owners of the waste have reimbursed the federal government all costs associated with the final repository projects.The basis for this is the Final Disposal Advance Payment Directive (EndlagerVlV). Thus far, about 2.4 billion Euros have been paid to the federal government in this way for the realisation of the final repository.

Moreover, the German utilities have made their contribution to the development of direct disposal of spent fuel assemblies through the construction of the pilot conditioning plant and the development of the POLLUX® concept, and have invested for these project about half a billion Euros in addition.

The polluters pay principle has thus always been applied in Germany – contrary to the assertions made by some politicians as for example Mr Trittin, former Minister for the Environment.

One further observation on the subject of “generation fairness”. In my view it is totally undisputed that the generation that has benefited from the use and advantages of nuclear energy is also responsible for the safe and environmentally-friendly disposal of the associated waste, and that they must not inflict this “residual waste” on future generations. This is only possible, however, if we continue consistently down the path we have begun and do not believe we can afford the luxury of throwing overboard the results of the work carried out in the last 30 years and starting afresh. To summarize this point.

Conclusion
The polluters pay principle is in effect for final disposal in Germany. But the principle of generation fairness can only be realised if the concept that has been pursued for 30 years is continued to be promptly realised.

C Current Status and the Way forward

Their fundamentally different understanding of the utilisation of nuclear energy notwithstanding, the German utilities have respected the political decision of the former federal government to phase out the utilisation of nuclear energy, and have conclude this in an respective agreement with the federal government.

It is also laid down in this agreement that the licensing process for the Konrad repository will be concluded in accordance with the provisions of the law. At the same time, the application for immediate execution of the repository licence has been withdrawn in order to allow a court inspection of the licence.

As expected, multiple plaintiffs objected to the licence that was passed in May 2002. In the four-year legal battle, the Lüneburg Higher Administrative Court (OVG) rejected all suits and did not approve a review. In March 2007, the Federal Administrative Court confirmed the ruling of the OVG. The Konrad licence is therefore now incontestable and can be carried out.

With regard to the quantity of Konrad grade waste already existing and expected in the next years to come the German utilities, expressly welcome the fact that the acting Federal Minister for the Environment, Mr Gabriel, has so promptly and consistently conformed with the agreement and commissioned the conversion of the Konrad pit facility to a final repository. Our hosts today, the Federal Office for Radiological Protection, already deployed a project group in May 2007, and subsequently called on the German Company for Construction and Operation of Waste Repositories (DBE) to start with the preparatory work for the conversion of the pit facility. I am therefore confident that the Konrad final repository will become operational in the year 2013, just as announced by Minister Gabriel. Unnecessarily long interim storage of waste can thus be avoided. The first part of the waste disposal strategy from the 90‘s, the final disposal of low and medium radioactive waste, is hence well on its way – albeit after a delay of more than a decade.

However - the situation with the exploration of the salt dome at Gorleben is quite different. In the already mentioned agreement between the federal government and the German utilities the federal government confirmed that the exploration results already gathered do not oppose the presumable suitability of the salt dome. Nevertheless, concept and safety related questions were raised by the BMU. It was agreed to suspend the exploration of the Gorleben site until these issues could be clarified. This moratorium should last at least for three years, however, but for not longer than 10 years.

By the middle of 2001, the (BMU) Federal Ministry for the Environment, Nature Conversation and Nuclear Safety defined a total of 13 questions to be answered. The German utilities naturally also took a very great interest in the clarification of these questions. GNS therefore commissioned a group of international experts to tackle the individual issues. The questions were processed in the course of 2001 and the experts came to the following conclusions:

- Most of the issues had already been discussed in the international community.
- None of the issues was new. All of the issues had already been described in publications and discussed for years, in part accompanied by controversy. German scientists were also directly involved in the discussions.

- None of the questions specified by the federal government justified the suspension of the exploration.

- The experts recommended evaluating all site-specific results of the investigation in a complete long-term safety analysis (Total System Performance Assessment, TSPA) after the completion of the exploration in Gorleben.

The Federal Office for Radiological Protection (BfS), which was given the task by the Federal Ministry for the Environment of processing the issues, presented its report at the end of 2005. In its concluding statement, the Federal Office for Radiological Protection comes to a very similar conclusion to that of the international experts. I quote: *"The studies concluded that no significant gaps in our knowledge can be identified at a generic level. The open questions identified can either be clarified in a regulatory manner, or are not considered so relevant that they must necessarily be clarified before any further decisions on the procedure for final disposal are made." Source: Concluding report of the Federal Office for Radiological Protection, "Host rocks in comparison", page 149, 4th paragraph.*

In other words: The questions thrown up by the federal government have been clarified to the greatest extent possible on a generic level, and the exploration could therefore have been resumed immediately after the presentation of the report from the Federal Office for Radiological Protection, as set out in the agreement. And in my view, it should have been resumed!

Under the terms of the 2001 agreement, the conditions for the lifting of the moratorium and the resumption of the exploration are therefore fulfilled. You all know, however, that the exploration at Gorleben is at a standstill as before. The Federal Minister for the Environment, Mr Gabriel, has made the continuation of the work in Gorleben conditional on other sites with various host rocks being explored parallel to Gorleben, with the aim of finding the "best possible" site. In my view, the following observations can be made regarding this connection:

- In the afore mentioned agreement between the government and utilities, there is no mention at any point of exploring other sites. This is, in particular, not a condition for the resumption of the exploration in Gorleben.

- The Atomic Energy Act demands a safe location, and for good reason does not demand the "best possible" site. I refer here expressly to the grounds given for the ruling of the Higher Administrative Court Lüneburg in the Konrad administrative dispute case.

- To find the objectively best site in scientifically impossible – as stated before. In this context I refer here to the statement of the International Committee on Nuclear Technology (ILK) from July of this year, and to a noteworthy article by Dr Jan Weber of the Federal Institute for Geosciences and Natural Resources (BGR), printed in the October edition of the specialist periodical "atw".

This notwithstanding, a series of measures are reasonable and even necessary, which can be implemented **parallel** to the resumption of the exploration and which would thus contribute to the transparency of and confidence in the final storage solution being pursued:

- Performance of an international peer review for the evaluation of the exploration results hus far, regarding the presumable suitability of the salt dome in Gorleben, or any analyses that should still be conducted.

- Performance of an initial safety analysis (TSPA) with the aim of making reliable statements on the feasibility of safe final disposal in the Gorleben salt dome, and obtaining information on any continuing gaps in our knowledge and possibilities for optimisation.

- Analysis of existing data on alternative sites – especially clay sites – and whether these assert themselves as better alternatives than Gorleben. This would also be wise in order to have a prioritised option for action, in case Gorleben should objectively prove to be unsuitable at the end of the exploration – which to us seems to be unlikely from todays point of view.

- Presentation and discussion of the results of the above points in an international workshop at the end of 2009/start of 2010.

I believe this would be a good way forward, which would nevertheless allow a reasonable solution to be found in a suitable time frame.

To conclude my presentation, I would like to briefly sum up my key points:

- The planning and work for the final disposal of the radioactive waste began in Germany at practically the same time as the development of nuclear energy.
- The waste disposal concept that applied until the end of the 90's was and is satisfactory for the requirements, and was implemented by those responsible in an appropriate timeframe, until 1998.
- After the political decision was made to phase out nuclear energy in Germany, meanwhile the Konrad repository is licensed and will be operational by 2013. The disposal of low and medium radioactive waste is thus secured.
- With regard to a final repository specifically for heat generating high radioactive waste, no progress has been achieved since 1998. On rational consideration, the German utilities consider it necessary to conclude the exploration of the salt dome without delay.The waste can then be placed in final storage starting between 2025 and 2030, still by this generation.
- Parallel to the resumption of the exploration, measures should be taken that lead on the one hand to a conclusive evaluation of the exploration results so far and the measures still necessary, and on the other hand also to more transparency and acceptance.

Thank you for your attention. ■

3.03 Requirements on Decision Making Processes in Radioactive Waste Management - Expectations of COWAM 2 Stakeholders

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Abstract
Originally, decision making in Radioactive Waste Management, particularly siting of repositories for radioactive waste, was understood as a technical issue. Stakeholders were "involved" in such processes - if at all - only in the very last phase. By local actors such processes were often perceived as intransparent and unfair, the results were often offended as disregarding the stakeholders' concerns and interests. Responding to resulting drawbacks and even failure of decision making processes, some national and international institutions started to discuss the ethical and social aspects of decision making and to involve stakeholders. From approximately 2000, the European Commission launched several research projects explicitly dealing with the concerns and interests of local stakeholders and including them into research. Within the Commission's COWAM 2 project (2004-2006), 17 recommendations for decision making processes were developed by participating stakeholders describing the state of the art of decision making in Radioactive Waste Management with regard to stakeholders concerns. The recommendations are presented and briefly explained. They are exemplarily applied to the decision making process resulting in the designation of the Gorleben salt-dome as the potential German repository site for radioactive waste. The procedure, obviously, does not fit with the presented recommendations.

1 Background

From the beginning of non-military use of nuclear power decision making in Radioactive Waste Management (RWM) was understood as a pure technical issue - admittedly with an ethical connotation regarding the protection of human and environment against radiation damages. This was also, and particularly, true for the first systematic siting attempts for the disposal of radioactive wastes undertaken in several countries, e.g. in Germany, Sweden and Switzerland, during the 1970ies and early 1980ies. At that time, regional and local actors were „involved" in such Decision Making Processes (DMPs) - if at all - rather in an informal way than by participation and normally only in the last phase of a process, when most of the relevant details of the respective decision were already fixed. In several cases the decisions became offended, because they were perceived by local actors as being intransparent and even unfair and as disregarding stakeholder concerns and the interests of future generations. Consequently, some countries made efforts to regain confidence by modifying ongoing DMPs or even launching new processes with (increased) consideration of ethical and social aspects and with participation of stakeholders.

Today, also on the international level, ethical and social aspects as well as stakeholder involvement in decision making are seen as important issues not only of RWM, but also in the broader scope of radioactive waste governance. Since approximately 2000 international organisations address themselves to the problem of stakeholder integration in RWM decision making and developed research programmes for confidence building. Particularly to be mentioned here are the OECD/NEA's Forum on Stakeholder Confidence (FSC) and several EURATOM projects on risk perception and communication and on participation of (local) stakeholders in RWM, e.g. RISCOM[1], ARGONA[2], COWAM[3] and CARL[4].

In the year 2000, the European Commission started a series of meanwhile three COWAM projects. COWAM means **CO**mmunity **WA**ste **M**anagement, expressing the projects' intention to deal systematically with the expectations of regional

[1]final report: http://www.valdoc.org/FR.pdf

[2]http://www.argonaproject.eu

[3]http://www.cowam.com

[4]http://www.carl-research.org

and local actors regarding decision making in RWM and to integrate them into the respective research. Particularly within the COWAM 2 project the requirements on DMP in RWM as derived from the expectations of local and regional actors were identified and discussed. The results of the respective working group of the COWAM 2 project are presented in the following sections.

2 The COWAM Projects

2.1 COWAM 1

A major objective of the first COWAM project (2000 - 2003, today COWAM 1) was to establish an European network of municipalities and other local and regional organisations concerned about RWM issues. This network served as a platform for dialog among these actors, but also with other groups of relevant stakeholders, e. g. regulators, implementers and NGOs. More than 200 representatives of stakeholder groups from 10 countries were forming the network, the majority of them coming from Belgium, France, Sweden, Switzerland, Spain and the United Kingdom. Main topics of discussion were the implementation of local democracy, the access of non-experts to expertise in the DMP, the influence of local people on the national nuclear waste management framework, sustainable development in regions hosting facilities, and the quality of the DMP. Despite of their different roles in national RWM, there was a broad agreement among the participants about the general requirements on DMPs and the necessity to improve the participation of local stakeholders, and they identified the following topics as being essential for future strategic research in this field [1]: implementation of local democracy, influence of local actors on the national DMP, quality of decision making, and long-term governance of radioactive waste management.

2.2 COWAM 2

The subtitle of the COWAM 2 project (2004 - 2006) "Improving the Governance of Nuclear Waste Management and Disposal in Europe" reflects in a general way the intentions of the 6th EURATOM Framework Programme as stated in [2]: The *"absence of a broadly agreed approach to waste management and disposal is one of the main impediments to the continued and future use of nuclear energy"* resulting in the need to

- develop decision processes that are perceived as fair and equitable by the stakeholders involved,
- enhance the understanding what influences public acceptance and
- develop guidance for improved governance of geological waste disposal.

Therefore, COWAM 2 was designed to contribute to the actual improvement of the governance of RWM by better addressing and understanding societal expectations, needs and concerns, notably at local and regional levels, and particularly regarding DMPs.

The project was organized as a research partnership between research contractors and stakeholders including representatives of stakeholder groups from 13 countries. Four Work Packages were established to deal with the research topics as proposed by COWAM 1. To assure the influence of the stakeholders the research directions within the individual Work Packages was controlled by so called stakeholder reference groups and stakeholder participation in the steering of the project (details about organisation, methodology and results in [3]).

2.3 COWAM in Practice - CIP

Within the 7th EURATOM Framework Program (2007 - 2009) the series of COWAM projects is carried forward by COWAM in Practice. This project is designed to contribute to actual progress in the governance of RWM, to follow up and analyse ongoing national DMPs of RWM governance (Spain, United Kingdom, Romania, Slovenia, and France), to support stakeholders, particularly local communities, in their engagement, and to capture the learning from that experience for the other members of the European Union.

3 Recommendations of COWAM 2 Stakeholders

3.1 Framework

3.1.1 Work Package 3 - Quality of Decision Making Process

Within the COWAM 2 project the experiences and expectations of stakeholders regarding decision making were particularly discussed in Work Package 3 “Quality of Decision Making Process”. The Work Package had approximately 35 members representing stakeholder groups with different roles and background from Belgium, Germany, France, Hungary, Romania, Slovenia, Spain, Sweden, Switzerland, and the United Kingdom (the majority from Belgium, France, Slovenia and Spain). The results of the Work Package are presented in its key document “Insights and Recommendations” [4] comprising 17 requirements on DMPs, which are presented in the form of recommendations (see table 1 and chapter 3.2). These recommendations were passed in consensus documenting that there are similar expectations on the quality of decision making in RWM across the different groups of stakeholders.

3.1.2 Input for the Development of Recommendations

The major input for the development of the recommendations came from analyses of planned, ongoing or finished DMPs in the participating countries. They were described by means of a questionnaire on the national decision making situations and are, thus, reflecting the empirical experience of the participants with regard to DMPs in their countries. Most of the evaluated processes refer to site selection for repositories. The results of the questionnaires were weighed against methodological fundamentals of decision making and societal and ethical expectations of the stakeholders with respect to a fair and equitable process. Based on the integral analysis of these processes the participants

- identified the key characteristics of a fair and equitable process and its procedural elements and
- explored the conditions of an improvement of DMPs and of practical ways to involve stakeholders during all phases of DMPs.

The main focus of the derived recommendations is on potential benefits for local and regional stakeholders.

3.1.3 What makes the Quality of a Decision making Process?

Simply speaking, a DMP is a course of acting leading to a decision. The following phases may be distinguished within such a process:

- problem identification (situation analysis, problem recognition, goal definition),
- development and design of options,
- option selection (evaluation, choice, bargaining),
- implementation of the selected option.

Complex decisions, as in RWM, normally consist of a chain of decisions, where all but the last decision are made to decide whether and in what direction the DMP should go on. Regarding the quality of the process all individual decisions of the overall process have to meet the same requirements.

Due to their long-lasting character most DMPs in RWM are highly vulnerable against influences from outside the frame of the process. Therefore, a politically and socially open minded “climate” (or context) is essential for the well-regulated continuation of the process over time according to the rules as agreed upon before start. Besides that it is not enough to have a methodologically "good" process, because such a process does not necessarily result in a good decision. "Good process" refers also to the defined goals of the process by in fact solving the identified problem in due time and in an "agreeable" way and being resistant against influences preventing a (good) solution. A DMP achieving these requirements is referred to as "robust". The conditions sine qua non for such a robust process are the compliance with requirements formulated to ensure the quality of the DMP according to the state of the art of decision making and fundamental principles of the civil society (democratic, fair, …), and a political and social environment facilitating the sustainability of the process.

3.2 Recommendations

Considering the representation of a broad variety of relevant stakeholder groups within Work Package 3, the achieved consensus about the presented recommendations (see chapter 3.1) assigns a high degree of credibility to them. Thus, the recommendations can be regarded as representing the state of the art of decision making with respect to the needs of stakeholders.

Table 1: 17 Recommendations at a glance

A	Define goals
B	Always provide alternatives
C	Ensure weighing and balancing of values and interests
D	Be comprehensive
E	Proceed stepwise
F	Ensure flexibility
G	Be transparent and open
H	Allow sufficient time
I	Stick to the "rules of the game"
J	Define roles and responsibilities
K	Ensure early and inclusive participation
L	Establish control of the process
M	Adapt formats to tasks
N	Allocate adequate resources
O	Ensure continuity of structure and awareness
P	Secure influence of participants
Q	Enhance well-being
The arrangement of the recommendations does not mean ranking with regard to relevance!	

Enforcedly however, the recommendations are rather generic. This is because none of the examined DMPs could be used as a “template” for decision making in other countries or for every purpose. Too different are the frameworks of decision making in different countries according to the legal situation and the “national culture” of decision making. However, the recommendations can easily be adapted to a given national context and to the purpose of concern. Having been designed for the development of a DMP, they are also capable for the evaluation and potential amendment of an existing DMP.

The following explanations of the recommendations can only highlight selected aspects. The complete description of the recommendations is presented in [4].

A Define goals

The identification of the problem and the definition of the goal(s) of the DMP are fundamental issues at the very beginning of a DMP. If the problem is badly identified or differently described by different actors, it is difficult - if not impossible - to agree upon the goals and to reach them. Without a minimum degree of agreement on the problem, the main goals of the DMP and on the rules of the DMP there will be no successful process. Additionally, “good” decisions are only “good” with respect to the predefined goals. Therefore, DMPs have to be designed according to their purposes.

B Always provide alternatives

Alternatives are a methodological “must” for decision making. Since failure of the proposal is a possible outcome of the procedure and decisions need a choice of options, alternatives have also to be considered as contingencies. Additionally, the elaboration and presentation of alternatives provide chances to improve the confidence between the different involved actors: With alternatives, stakeholders will have the feeling to have a choice and, thus, to really participate in a DMP - provided they have real influence on the process.

C Ensure weighing and balancing of values and interests

Weighing and balancing the (expected) properties of different technical options is a common task in technical decision making. Generally, standards and instruments to compare the options with regard to specific aspects are available or can be developed for the respective purpose, e.g. different types of criteria. Long-term management of radioactive waste, however, compasses several relevant, particularly ethical and societal, issues, that can only be dealt with sufficiently by considering, weighing and balancing individual and group specific values and interests.

D Be comprehensive

Many decisions in the field of RWM are safety related. Therefore, from the safety point of view, it is essential to identify those factors that are relevant for the decision to be taken. The basis for this is a fundamental understanding of the features and processes that might affect them. However, safety is the most important but by no means the only aspect to be considered: Also relevant non-technical aspects have to be identified and dealt with in an appropriate way. Potential implications and side effects of an option have also to be considered.

E Proceed stepwise

A phased or stepwise approach keeps options open, is more traceable, enables inclusive reviewing as well as better control, enhances the chance for technical revision and political back-up, and may facilitate the insertion of emergent issues that were previously left out of account. Options may successively be “closed down” by interim decisions.

F Ensure flexibility

The long-term character of the issue of concern, both in “objective” and institutional terms, calls for procedural flexibility. The process must allow iterations, with opportunities for recourse (and mutual learning ...). Decisions may be reversible to a certain extent (in some countries this may include retrievability of waste). Individual decisions within a DMP may be liable to errors caused by, e. g., insufficient or even wrong information. This may occur particularly with siting, where individual decisions in early steps of the overall process are strongly based on paper works.

G Be transparent and open

Transparency of the process and openness of decision making are fundamental to achieving understanding, confidence and trust. In a complex, long-term and contentious issue such as RWM major arguments and steps must be understood and supported by the key stakeholders and the public.

H Allow sufficient time

Robust decisions must be based on careful elaboration of options to decide from and the systematic discussion and evaluation of these options. It is an inefficient use of resources and creates frustration if there is not sufficient time for this work and if too many goals in a complex issue are to be pressed in too short a period of decision making.

I Stick to the "rules of the game"

The rules and criteria of a DMP, e.g. a site selection procedure, have to be consented to before the start and adhered to during the process. Revisions should undergo a careful review and be consented to. Unambiguous rules add to reliability, accountability, continuity, and, finally trust in the "decision making system". If there is consent among participants about the lead time, later modifications of it need consent also.

J Define roles and responsibilities

Continuity and accountability can only be guaranteed if there is a sound political and legal basis with a corresponding institutional framework. Particularly for responsible institutions with a driving role in the process, such as the oversight body, the regulator and the implementer, it is essential to have a clear mandate as well as a clear and unambiguous responsibility. All actors have to know their own and their partners roles and responsibilities in the DMP. Decisive is the control of resources and the process itself.

K Ensure early and inclusive participation

To achieve sustained decisions among individuals, groups and organisations there is a need for "informed consent". This, in turn, requires an explicit elaboration of different possible ways and consequences of courses of actions. Inclusive participation increases the chance that all relevant perspectives on the issue are being raised. Who "should" and will participate depends on the aim and stage of the DMP, because issues and interests are depending on the purpose of the DMP and participants vary during the course of the process. Regarding siting processes, it has to be avoided that a stepwise introduction of stakeholders prevents regional and local actors from expressing their views, because they might be introduced only when important determinations within the DMP already have been done.

L Establish control of the process

Due to the objective and institutional long-term dimension of RWM, special consideration has to be given to the "owner of the process". For the control of the process it might be pertinent to establish an independent "guardian of the process" to see to it that the programme is on target. Inclusive monitoring, reviewing, reporting and evaluation have to take place. Furthermore, to ensure continuity, the DMP might be integrated it into existing formalised and legal procedures (such as SEA, EIA).

M Adapt formats to tasks

The procedural tools to be applied in the DMP have to be matched with the phase of the DMP and the goals to be achieved.

N Allocate adequate resources

Good decisions are characterised by a careful processing of several alternatives. In order to assemble material for an option analysis, one has to search for information, to design the proposed project or facility, and to evaluate the specific advantages and disadvantages. Resources have to be provided accordingly. They have to be sufficient to favour the capacity of stakeholders in using pluralistic expertise.

O Ensure continuity of structure and awareness

Due to the long-term dimension of DMP in RWM these processes are associated with intrinsic uncertainties and external influences. The challenge is to ensure that thoroughly discussed and broadly consented goals still can be understood, agreed upon and followed by generations to come. The process should have adequate procedural directness, but also sufficient flexibility to allow for technological and societal learning. There must be an institution to monitor the process with regard to the continuity of the intentions of the process agreed upon at the very beginning and its structure and to arrange for the adequate consideration of the values and interests of future generations.

P Secure influence of participants

The kind and extension of stakeholders‘ influence on a DMP varies from country to country, because both depend on the national culture and legal framework of participation. In some countries it is concluded from the fundamental idea of equal rights, that there should be the right to accept or withdraw. A good DMP should give local communities real power so that they can contribute and negotiate on equal terms. This is seen as an essential condition in order to improve confidence within the whole process. At any rate, the proof of real participation will lie in whether and how inputs are considered or not, whether and how actors are respected and indeed do influence the process.

Q Enhance well-being

Nuclear facilities, particularly repositories for radioactive waste, are perceived by many people as dangerous and a threat to their and their offspring well-being. Therefore, it is not only essential to demonstrate why the facility has been sited "just here" but also to overcome the fear of discredit and of a loss of quality of life. Thus, participating local communities must benefit from their participation - there must be local benefit as well as national. Preferably, such a benefit should emerge from measures to improve regional development with regard to sustainability rather than short-term compensation.

4 Application of the Recommendations (e.g. Gorleben Case)

From the COWAM stakeholders' point of view, it is evident that a DMP not complying with the recommendations presented in chapter 3 needs to be amended accordingly or to be replaced by a feasible new process. The experiences of COWAM 2 stakeholders from their national DMPs was the major source of input for the Work Package 3 recommendations but the application of the recommendations to ongoing or planned DMPs in the represented countries was out of scope of the Work Package. Due to the inevitably generic character of the recommendations the Work Package abstained from the design of rules for the adaption of the recommendations to specific applications and for their application.

However, to demonstrate briefly such an application the set of recommendations is applied here in a rough way to the set of consecutive decisions resulting in the designation of the Gorleben salt dome as the German repository site for all kinds of radioactive waste and the start of its investigation [5]. The quality of the process of the Gorleben designation and its result were and are highly contested among German stakeholders, including the author, as are the results of its surface bound investigation. To reduce the influence of personal interpretation the application of the COWAM 2 recommendations refers only to those procedural aspects that can be evaluated alone from the historical record of the decisions and their obvious consequences. When concentrating on these aspects, the following phases may be distinguished for the evaluation (see also table 2):

- Phase A: Development of the national waste management strategy and selection of host rock for final disposal
- Phase B: Siting process for the national waste management centre and start of surface bound site investigations at three selected sites
- Phase C: Abort of the investigations and the overall first siting attempt and start of new siting process

It has to be mentioned that the regulatory and procedural responsibility for the described siting process rests with the Federal Government. Consequently, at the beginning of the described process the Federal Government was the driving force of the process. The siting process was carried out on behalf of the Federal Government with application of criteria that were designed regarding the needs of the surface installations of the nuclear waste management centre. They were not sufficiently adapted to the necessity of long-term-safety of the planned repository. As regards the definition of goals and development and application of siting criteria, the Government of Lower Saxony took over the leadership in Phase C. Both, goals and criteria were certainly different from the first phases. The goals of the Lower Saxony Government, the quality of alternatives to decide from and the applied criteria have been object to rumours rather than knowledge over years, and even today they are heavily disputed in the public.

During none of the three phases there was any participation of stakeholders. However, the strong resistance to the site investigations at the end of phase B caused a reaction of the responsible institutions that might be misinterpreted as seizing the concerns of local stakeholders. Instead, some witnesses to history argue that the Gorleben decision was made and accepted on the basis of pure political interests [e.g. 6].

Table 2: Historical Record of the Gorleben Case

Phase A	1960ies to early 1970ies	determination of host rock: rock-salt in salt-domes
		development of waste management strategy: nuclear waste management centre with facilities for storage, conditioning, reprocessing and final disposal of all types of waste
Phase B	1973	start of siting process for nuclear waste management centre; criteria developed with emphasis on surface facilities
	1975	result of siting process: highest ranking salt-domes Wahn, Lutterloh, Lichtenhorst in the state of Lower Saxony to be compared for final identification of the site of the nuclear waste management centre
	1976	June: start of investigations at three sites causes immediately strong local opposition
		August: interruption of investigations
		November: Lower Saxony asks for stop of investigations to present its own site
Phase C	1977	February: Lower Saxony designates Gorleben salt-dome
		July: Federal Government accepts Gorleben
		November: waste management report of the Federal Government declares Gorleben at least suitable for LLW and ILW
	1979	March/April: “semi-public” Hearing about the nuclear waste management centre and final disposal in the Gorleben salt-dome
		April: start of surface bound Gorleben investigation

Already this cursory evaluation demonstrates that the Gorleben DMP clearly does not meet the recommendations of COWAM 2 stakeholders. To a certain extent the reasons lie in the temporal discrepancy between the start of the Gorleben process and the development of DMPs with regard to methodological quality, fairness and equity. On the other hand, the process was characterized by severe mistakes and deficiencies that were recognised and expressed by several actors inside and outside the responsible institutions - already at that time.

These mistakes and deficiencies could and should have been resolved. However, the amendment of the process would have to be comprehensive and start with steps at its very beginning, i.e. ~30 years ago. Therefore, launching a new process according to the actual state of the art is a more promising option. The siting process proposed by the German Committee on Site Selection Procedure [7] complies with the majority, if not all, of the recommendations. It could easily be implemented or form the core of an adapted process. However, a successful future process needs a climate of mutual trust among authorities and stakeholders and confidence in the DMP. Such a climate has not yet been accomplished - not least due to the reluctance of major industrial and political stakeholder groups.

5 Summary and Conclusions

Decision making in the field of radioactive waste management, particularly siting of repositories, was originally seen as a pure technical issue. Drawbacks and even failure of national siting approaches resulted in the integration of non-technical aspects and the improvement of stakeholder participation to re-gain confidence in decision makers and the outcome of DMPs. Today, the integration of non-technical, such as ethical and social aspects and the participation of stakeholders are widely seen as pre-conditions for successful DMPs with robust decisions.

In recent years, not only the establishment of methodological principles and the application of tools for the participation but also the identification of stakeholder requirements helped to improve the quality of DMPs. The COWAM projects are examples for stakeholder influenced bottom-up methodological research. Their results, thus, reflect a kind of state of the art as defined by the experiences and knowledge of participants representing a broad spectrum of stakeholders and researchers from different countries.

Requirements on DMPs as developed by the COWAM 2 stakeholders provide general standards to develop DMPs or to assess existing DMPs that could be perceived as fair and equitable. Due to their general character they have to and can be adapted to the specific context and purpose as defined by the goal of the respective DMP, the national legislation and DMP culture resp.

An indispensable prerequisite for successful decision making is a climate of mutual trust and confidence among all stakeholders. It cannot be generated by the fulfilment of such requirements alone, but needs systematic efforts of the responsible institutions to gain confidence and trust. For the situation in Germany the Committee on Site Selection Procedure has a presented a proposal that has found broad agreement but has not yet been transferred to a new DMP due to lack of willingness of major industrial and political stakeholders to participate in such a process. ■

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3.04 Demands on the Governance of High Active Waste Disposal – International Developments and Impacts on the German Disposal Process

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Abstract

Current developments of nuclear and non-nuclear projects reveal that the "traditional" approval process that focuses on technical aspects and allows only few opportunities for public participation has not been successful in realising potentially hazardous and/or publicly disputed installations like a repository for high active waste.

The implementation of an enhanced governance process that follows the principles of legitimacy, fairness, transparency and quality of decision making is a prerequisite of a successful siting and approval procedure.

- *The consideration and integration of the national and the local levels with their specific roles and requirements,*
- *assuring appropriate resources and*
- *implementing an adequate organisational framework*

are further key factors of good governance that support a successful process for high active waste disposal.

Implementing an adequate procedural framework that considers the national and regional level and realising participatory elements requires a turn in decision making from the traditional approval process to an enhanced transparent and stepwise procedure. International experience reveals that taking these steps is a promising approach.

1 Introduction

Enhancing transparency and legitimacy of decision making is nowadays widely regarded as an important factor in the process of planning and licensing of potentially hazardous and/or publicly disputed installations. This development is also reflected in the international discussion on the "governance" of site selection and approval procedures for high active waste repositories.

The Institute on Governance defines "governance" as *"the process whereby societies or organizations make important decisions, determine whom they involve and how they render account"*. According to this definition it is obvious that stakeholder involvement and public participation are vital but not the only features of good governance. They have to be embedded in a suitable framework with regard to the decision making process.

Certainly, there is no one-fits-all strategy as every governance process has to be matched carefully to the respective national and project specific conditions. But a closer look at existing participatory processes in the nuclear and non-nuclear field and at the lessons learnt from international discussion, research projects and initiatives in radioactive waste management, can reveal key elements of good governance. They encompass procedural as well as participatory aspects. Some of the most relevant aspects will be presented in the following paper and discussed in the context of the needs in the site selection and approval process of a high active waste (HAW) repository in Germany.

2 Short History of European Governance Projects

Before we look at the development of governance projects at the European level a quick glance at the Forum on Stakeholder Confidence (FSC), a working group at the OECD/NEA, may serve as a guidepost for conceptual and practical efforts in governance of radioactive waste management (RWM):

"In the radioactive waste management context, a set of specific action goals should be targeted. …In particular, in order to identify and implement solutions that are widely regarded as legitimate, it will be important: ... to develop a broad understanding that the status quo is unacceptable and that an important problem needs to be solved; ..." [1]

On the European level the first important projects on the governance of RWM and other potentially hazardous industrial activities were funded in the late 1990s. Broad support was given to the TRUSTNET [2] and COWAM [3] initiatives, both funded as three consecutive projects since 1997 or 2000 respectively. Networking as well as inter- and intra-disciplinary exchange have been important parts in both of them. RISCOM II which followed a Swedish pilot project and RISKGOV are further examples of European governance projects, both funded within the 5th Framework Programme of the European Commission.

The requirements of good Governance are summarised as results of these projects in various models and frameworks, e. g.:

- The "Mutual Trust Paradigm" and the process structure of "Inclusive Governance" (TRUSTNET)
- "Local democracy" as the centre of a model of improved governance (COWAM)
- The "RISCOM Model" with "three roots of communicative action" (RISCOM)
- A framework for evaluation of risk governance processes by RISKGOV.

More details of these and other European governance projects are summarised in [4].

The specific Euratom programme for 2005-06 emphasises the need for guidance for the application of new and emerging approaches in the radioactive waste sector and for cooperation and dialogue among different social and technical actors. National differences (e.g., culture, history, legal and administrative regimes) have to be taken into account.

The three projects, as can be seen in table 1, have been selected by the European Commission on the issue of governance of radioactive waste management for the period 2006-2009.

These projects have set up a joint web-portal on invitation of the European Commission www.radwastegovernance.eu to inform the public of their respective activities [6]. To support cooperation among the projects a liaison committee has been set up and a newsletter is published to provide information of the progress and liaison activities.

3 Key Factors of Governance

Within current research projects a wide range of experience with existing participatory processes in the nuclear and non nuclear field and with results from research projects and initiatives has been evaluated:

- One project, funded by the German Federal Office for Radiation Protection (BfS) deals with demands on public participation in the different stages of the planning process for a repository [7].
- "OBRA", funded under the 6th FP EURATOM aims at establishing an Observatory to promote appropriate forms of interaction between stakeholders, mainly local and regional communities and experts [8].

Table 1: FP6 - Management of Radioactive Waste - Geological Disposal: Current

Project	Instrument	Coordinator/ partners	EC contribution / total cost	Start date & duration
ARGONA – Arenas for risk governance in radioactive waste management	STREP	SKI, SE 13	1.2 M€/2.29 M€	01/11/06 3 years
CIP – New governance approaches in waste management (‘COWAM In Practice’)	STREP	MUTADIS, FR 11	0.8 M€/ 2.28 M€	01/01/07 3 years
OBRA – European observatory for long-term governance on radioactive waste management	CA	ENVIROS, ES 10	0.3 M€/ 0.46 M€	01/11/06 2 years

Some basic elements prove to be of general importance for successful decision making processes in complex and publicly disputed projects. From these basic elements a set of key factors has been derived which is not necessarily exhaustive but covers some central requirements that will be further discussed in the following:

- The vital principles: legitimacy of decision making, fairness, transparency and quality of the process of decision making and of the decisions reached;
- Considering the national and the regional level by establishing adapted processes that integrate the specific stakeholders and address the relevant topics and questions;
- Assuring appropriate resources which facilitate competence building of the actors involved, adequate organisational conditions and regional development;
- Implementing an adequate organisational framework.

3.1 Legitimacy, Fairness, Transparency and Quality

Following Lennartz and Mussel [9] a legitimate process is a process that follows criteria such as transparency, representativeness, reliability, competence, fairness and efficiency. According to the Institute on Governance (IoG) [10] “legitimacy and voice” – one of five principles of good governance – means:

- *“Everyone who needs to be, is at the table;*
- *There are forums for bringing the partners together;*
- *The forums are managed so that the various voices are listened to and the dialogue is genuine and respectful; and,*
- *There is a consensus orientation among all those at the table.”*

The IoG [10] links this to the principles of “participation” and “consensus orientation” which were formulated in 1997 in the context of the United Nations Development Program [11] and were defined as follows:

“All men and women should have a voice in decision-making, either directly or through legitimate intermediate institutions that represent their interests. Such broad participation is built on freedom of association and speech, as well as capacities to

participate constructively." and *"Good governance mediates differing interests to reach a broad consensus on what is in the best interest of the group and, where possible, on policies and procedures."* [11]

These definitions show that legitimacy, fairness and transparency are closely linked and interrelated. A process will only be considered legitimate in the eyes of those involved in or affected by a project, if a transparent procedure enables them to understand and follow the development and if they have a fair chance to articulate their views.

Consequently, a process that is oriented towards the principle of legitimacy supports high quality decisions which are founded on a broad range of views and competencies.

An evaluation of various disposal processes shows that the "traditional" approval process that focuses on technical aspects and allows only few opportunities for public participation has not been successful in implementing a HAW repository in any country.

Respecting legitimacy, fairness, transparency and quality does not guarantee a successful realisation of a siting and approval procedure - but definitely theses principles are a prerequisite of success.

3.2 The National and Regional Levels

Disposal of radioactive waste is a national task that needs local implementation when it reaches an advanced stage. In most countries this process takes place in a politically and socially controversial arena: while the solution of the challenge is of high national interest the local level is highly affected. This constellation imposes special requirements on the layout of the process and the integration of the two levels.

Different scientific discussions on risk communication and evaluation of national site selection processes show that in the field of radioactive waste management an agreed national frame- and agenda-setting should be established beforehand (cf. e.g. [7], [12], [13]). A national consensus or any kind of general understanding of the necessity of a suitable site and of its technical and geological standards as well as the consideration of socio-economic selection criteria will support the regional debate and decision making process [7]. According to COWAM [14], a national framework for nuclear waste management is essential for decision-making processes at the local level. The FSC [15] points out that moving from the national to the local dimension (e.g. for site characterizations) requires the existence of a decision-making process that is widely supported and adhered to by all actors. Such a decision making process needs to assure that regional processes meet national demands and to keep regional objectives compatible with the general process on the national level.

Experience shows that including regions with potential sites in a national selection process can cause strong opposition in those areas. Fixing key features on the national level at an early stage can help a timely start of the regional governance process in order to pick up potential conflicts [12].

With regard to the German situation in HAW disposal various important corner stones of the process are still open or discussed among stakeholders, e.g.

- The extent and features of the site selection procedure
- The protection targets
- Criteria for comparison and assessment of sites (but the latter currently under discussion)
- Organisation and (binding) character of public/stakeholder involvement
- Regional development
- Financing.

Activities to find agreement on these questions were initiated by the German Government with the implementation of the AkEnd (Committee on a Site Selection Procedure for Repository Sites). Since important stakeholders could not agree on a procedure to put the recommendations of the AkEnd's final report published in 2002 [12] into practice the disposal process has come to rest in the recent years.

The current discussion of safety criteria lead by BMU (Federal Ministry of the Environment) is an important step towards enhancing the exchange between the different institutions and experts involved. Other highly contentious questions, esp. on site selection, roles/responsibilities and financing, seem to require a "high-level decision" on the general direction as a kick-off for the further consensus process.

With regard to the principle of legitimacy the definition of important corner stones of the German disposal process on the national level should be reached in a participatory consensus process. This process should include i.a. the affected national authorities, waste producers, expert commissions, non governmental organisations (NGO) and representatives of the federal states.

As long as the candidate site(s) has not been determined the integration of those who will potentially be affected is not possible. But the representation of regional interests should be considered in the consensus process. This will help an adequate adaptation of the process on the regional level at an advanced stage.

3.3 Appropriate Resources

Appropriate financial and personal resources have to be guaranteed in order to realise the necessary governance actions and competence building of the actors involved. Sufficient resources can be regared as prerequisites of an appropriate governance process of a HAW repository [4].

There are three main areas which have to be considered with regard to resources:

3.1.1 Access to Expert Knowledge

In [14] the COWAM network emphasises the relevance of access to expert knowledge and of competence building among those involved in the process or potentially affected by a planned repository. This fosters independence and self-confidence of local actors and supports the multidisciplinary and pluralistic approach and the quality of decision making.

Opportunities for competence building should cover technical questions as well as the ecological and socio-economic impact that may be relevant in the context of high level waste disposal.

Providing the necessary resources and organisational structures for knowledge acquisition, allows for a process that strives for equitable conditions of all actors and thus supports acceptance of the procedure and its results.

3.1.2 Appropriate organisational Conditions

A governance process for complex projects that is oriented towards the principles under 3.1 will afford participatory measures that go beyond the activities foreseen within the formal application procedure.

Implementation, support, management and realisation of those measures that support the governance process on the national and local levels require a clear organisational structure, support by professional staff and equipment.

The extension of the Frankfurt/Main airport is an example of a participatory stakeholder process that aims at developing the conditions under which the region will accept the extension. The management of the entire complex process, including a regional stakeholder forum as a basic element, is in the hands of professional external experts who act as a management agency and scientific board. They are responsible for all organisational matters, communication and integration of results. The process is chaired by a university-president who is known to have a neutral position and good reputation [7].

The necessary framework and organisational support has to be set up for each individual case. But appropriate organisational conditions that support target-oriented progress can only be established if adequate resources are provided.

3.1.3 Regional Development

Potentially hazardous and/or publicly disputed installations and infrastructure planning are often accompanied by measures taken to support a positive economical, ecological and social development of the affected region. With regard to long-term sustainability of regional development the applied measures should support specific development targets. These should be derived considering the regional situation and possible impacts. Repository planning in Belgium and Finland for instance demonstrates that a close integration of regional development with the planned installation has a positive influence and helps the local public to identify or at least come to terms with the project.

Financial resources are necessary to allow for an adequate handling of supportive measures for the regional development of the affected region. The three aspects stated above are essential features of governance processes that strive for legitimacy, fairness and transparency. An enhanced siting and approval process therefore has to ensure adequate resources to implement theses features and to reach the desired progress. The necessary budget should be calculated in advance and should be provided considering the “polluter pays”-principle. Furthermore it has to be considered that the financial flow needs independent administration and control.

3.4 Adequate Organisational Framework

With regard to the organisational framework the following principles are of vital importance:

- clear targets and an agreed understanding of the perspective and goals of the process and
- the integration of informal measures of the governance process into the formal decision-making procedure.

It is obvious that these targets can only be achieved by institutionalised measures that follow defined rules. Thus participatory governance measures have to take place in a regular working context with clear structures and accountabilities.

The complexity of the disposal process poses different requirements at different stages. Hence such structures cannot be implemented once for the entire procedure but need to be adapted with regard to the respective step, e.g.

- the national consensus phase before entering into the site selection phase (see 3.2),
- the site selection phase that can include more than one candidate region and requires close interaction of the regional and the national levels,
- an assessment phase at a chosen site,
- the planning and approval stage.

It is therefore necessary to implement a comprehensible stepwise approach with defined goals, conclusions and decisions.

Such a stepwise approach has not (yet) been defined in Germany. Examples like the Belgian case for disposal of low and medium active waste or the current developments in the UK to find a national consensus on the site selection procedure for a HAW repository reveal that it is neither possible nor reasonable to formally define the concise procedure in a legally binding way in advance. Various actions such as the interaction of the potential host regions and the national level or the involvement of stakeholders on the national and regional levels need flexible adaptation to actual needs. But there has to be a clear concept as to how those “informal” activities as agreed parts of the process (even if not legally fixed) will be integrated into the formal decision making.

This means that defining the framework for the site selection and the approval process must include agreements on the bindingness of results elaborated e.g. by national or regional stakeholder fora.

The Canadian host community at Port Hope for example where a final solution for the nuclear legacy had to be found reached a contractual agreement with the Government on the conditions under which people were willing to accept a repository [1]. This approach helped to define the objectives of the regional stakeholder process, to determine how results would be fixed and strengthened the position of the region in the site selection process.

It is also possible that the bindingness of results might vary depending on the issue that is dealt with. Discussion on the regional level for instance that deal with safety relevant technical questions of the facility might result in recommendations. Discussions on regional planning and land use planning as well as measures for regional development might result in binding agreements between the affected region and the competent authority [7].

Both examples show that those governance measures not formally fixed in a legally binding procedure are integrated in the decision making process and that they have clear perspectives and targets.

4 Conclusions

Traditional formal approaches for approval of nuclear facilities have proved insufficient for the site selection and implementation of HAW repositories in several cases. In many countries like Switzerland, Belgium and the UK to name but a few this lead to a rupture in the disposal process. They were followed by alternative approaches that consider lessons learnt from former set backs and are aimed at the criteria of transparency and legitimacy.

Practical experience in the nuclear and non-nuclear field as well as results from research clearly indicate, that enhanced processes are necessary to avoid the failure of the process due to lack of support on the national and especially the regional level.

An adequate governance process that meets the complex requirements of high active waste disposal must integrate formal decision-making in a stepwise approach and informal measures to assure

- a clear framework and national policy,
- stakeholder integration on the local and regional levels,
- interaction between national level and the regions, and
- regional development.

Such a process might appear extensive and costly beforehand. But these efforts have to be weighed against the danger that time, costs and uncertainties arising from legal proceedings against the plans can be considerable. Certainly, a good governance process is no guarantee that a repository site will be found and that the facility will be accepted and can be realised in the selected region. But international experience and lessons learnt from non-nuclear projects should encourage the (politically) responsible institutions to take the necessary steps. Together with the relevant stakeholders they should define an adequate framework and implement a governance process for a HAW repository that follows the criteria of legitimacy, transparency and fairness. ■

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3.05 Research and Development – The Basis for Safety Assessment and Confidence Building in Radioactive Waste Disposal – Conclusion of the Conference –

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1 Introduction

Radioactive waste disposal research and development has been carried out for more than forty years mainly in those countries which use nuclear power as a source for electricity production. In the beginning the development of feasible disposal concepts dealing with appropriate handling and emplacement techniques as well as with environmental issues and the long-term safety of mankind was of great importance. Today worldwide many repositories are in operation for the final disposal of low and intermediate level waste. For this very reason research as well as site characterisation and qualification are mainly focused on the disposal of high-level waste (HLW) and of spent nuclear fuel (SNF). In view of the status of different national programs structure and content of the so-called safety case with the underlying scientific and technical facts and arguments are of utmost importance for further works.

The guidelines for the planning and licensing of radioactive waste repositories are established by a framework of international recommendations which are reflected in national laws, ordinances and regulations. The safety standards laid down by the International Atomic Energy Agency (IAEA) provide a good orientation also for the planning of appropriate research and development programs. The Organisation for Economic Co-operation and Development (OECD) and its Nuclear Energy Agency (NEA) provide a forum for detailed information exchange and the elaboration of co-ordinated expert views. In particular the activities of the Radioactive Waste Management Committee (RWMC) and its working groups are intended to sharpen visibility and transparency of the different safety approaches and research results. The European Commission (EC) supports actively multinational research projects in underground rock laboratories and in the field of performance assessment.

In the course of the REPOSAFE conference 47 presentations highlighted the most recent developments in radioactive waste management in Europe and abroad giving an insight into some important aspects of waste disposal research and technical development (RTD). A wide range of experimental and analytical research in rock salt and argillaceous formations was covered including the backfilling and sealing of repositories in deep geological formations.

Important questions today are:

- What has been achieved so far and what are the future challenges in RTD?
- What specific know-how and expertise is needed for solving the problem of HLW / SNF disposal?
- In what way does RTD contribute to the generation of confidence?

Before giving an answer it has to be acknowledged that the various national programs in waste management have partly different objectives and have reached different stages. In addition, there are a number of aspects which allow not for a straight forward and rational evaluation since they got a political and/or societal background. For these reasons the concept of RTD is used in different ways depending on individual and/or national positions and interests.

2 What do we understand under Research and Technical Development (RTD)?

Concerning the long-term safety of HLW/SNF-disposal RTD is directly coupled with the relevant safety criteria laid down in international recommendations and national regulations. It covers a vast area of different disciplines, approaches and motivations. This kind of multidisciplinary research is driven by conceptual constraints and by safety criteria.

Considering the advanced technologies and safety achievements in underground mining as well as the actual state-of-the-art in radioactive waste disposal technologies future technical work will be devoted mainly to the demonstration and optimisation of emplacement techniques including the adaptation to site-specific conditions (Figure 1).

Applied research (implementation orientated) as well as site-specific research is required for the qualification of a HLW/SNF disposal site. Both are making up the very basis for the "so-called" safety case.

Objectives		
➢ safe disposal operation	radiation protection	safety standards
➢ safe repositories	long-term safety !	safety criteria !?
Various areas of research		
➢ generic or basic research	constitutive models	
➢ interdisciplinary research	coupled processes	
➢ applied research	disposal concept and criteria driven !	
➢ site investigation	advanced methods	
➢ site qualification	safety case !	
Challenges of radwaste disposal research		
➢ complex and interacting systems		
➢ long time frames/remaining uncertainties?		
➢ well founded and coherent "safety case"		

▲ Fig. 1: Research in radioactive waste disposal – objectives and criteria, scientific areas and challenges (Marked in red: Most important issues related to implementation and licensing of repositories)

Important prerequisites for safety research activities (operational safety, long-term safety) are well founded, sustainable and broadly accepted safety criteria and safety standards aiming at the protection of mankind and the environment for the relevant time-spans.

The IAEA safety standards represent the current state-of-the-art. Two important documents are in preparation: DS 354 (Safety requirements on disposal of radioactive waste) and DS 334 (Safety guide on geological disposal). Experts from many countries have put in their expertise and viewpoints. In this context the IAEA Joint Convention is very important being something like the internationally agreed backbone of safety strategy and safety culture in Spent Fuel and Radioactive Waste Management. So far 42 countries have signed this convention. In addition, IAEA is running the Waste Management Assessment and Technical Review Program (WATRP) which, by the time, has developed into a well established tool for reviewing national radioactive waste management programs and disposal concepts.

While the IAEA activities are more or less overarching making up the "Safety Culture" with its basic principles and criteria, the NEA activities are devoted mainly to the use of nuclear energy and on this line to the implementation of geological disposal. The RWMC provides a platform for implementers, regulators, administrators and researchers from the OECD member states. IAEA and EC participate in the annual meetings. Various expert groups and forums are doing the technical work. In view of RTD for radioactive waste disposal the work of the "Integrated Group of the Safety Case" (IGSC) and the "Forum of Stakeholder's Confidence" (FSC) are essential, not only in fulfilling the Agency's mandate but also for common understanding and sharing knowledge. The so-called "Clay Club" under the auspices of the IGSC is dealing with scientific aspects which are important for the characterisation of argillaceous rocks and the impact of heat generating radioactive waste on such rock formations. In general, IAEA and NEA and their various committees, expert groups etc. are not only fostering international communication and co-operation but also reflecting the state-of-the-art in radioactive waste management and underpinning the awareness of safety related RTD.

3 Where do we stand today in Geological Disposal of High-Level Radioactive Waste and Spent Nuclear Fuel?

Worldwide the management and disposal of radioactive waste is tackled in different ways – depending on the type of waste – and in different time frames – depending on the amount of waste and the national socio-political environment. While the disposal of low and intermediate level waste is successfully performed in near surface repositories in many countries, the management and disposal concepts for high level and long-lived waste vary from country to country.

In the course of the REPOSAFE conference presentations were given on the actual achievements in some selected countries. Regarding the disposal in granitic rocks Sweden and Finland are making good progress. In Sweden a site selection procedure is under way while in Finland a selected site is presently investigated by underground exploration, and a tough time schedule for licensing and construction of a SNF repository has been implemented. In the Russian Federation first steps have been taken in order to qualify a crystalline rock massive northeast of the Siberian town of Krasnojarsk for underground emplacement of radioactive material. Close to the contact zone between the central and east Siberian shield the seismicity of the area is an important issue. In the Peoples Republic of China granitic rocks are being investigated in the vicinity of Beishan in the Gobi Desert. By deep core drillings some sections have been identified as sparsely fractured and low in water content.

A number of presentations dealt with argillaceous rock formations and with research in the different underground rock laboratories (URL). After qualification of the Jurassic Opalinus Clay by the exploration drilling at locality of Benken a site selection process has been initiated in Switzerland. Parallel to that all relevant research work is performed in co-operation with international partners in the Mt. Terri URL. In France interest is with the underground laboratory at Meuse/Hautemarne in Loraine where, at a depth of about 480 m, the properties and rock mass parameters of the Callovo-Oxfordian clay formation are investigated. The objective is to compile all relevant data and to develop the disposal concept needed for the selection of a repository site in that region. In Belgium the focus is on the Tertiary Boom Clay and the URL Hades at Mol with research on specific aspects of plastic clay as host formation.

The concept of underground radioactive waste disposal in rock salt, and of chemotoxic non-radioactive waste as well, is mainly pursued in Germany with two abandoned production mines (Asse, Morsleben) used for non-heat generating waste disposal and the salt dome at Gorleben investigated for its suitability to host a repository for all types of radioactive waste, in particular heat-generating waste. In the Asse mine which was used as URL subsequent to a preceding waste emplacement phase a complete transport and handling system for borehole emplacement was tested by GSF and her partners. The system passed all technical checks including the compliance tests with the limiting radiation dose set by the radiation protection ordinance. For drift emplacement of SNF a 1:1 scale heating experiment with dummy canisters and electrical heaters was run for almost nine years proving the distinct reduction of backfill porosity by heat impact.

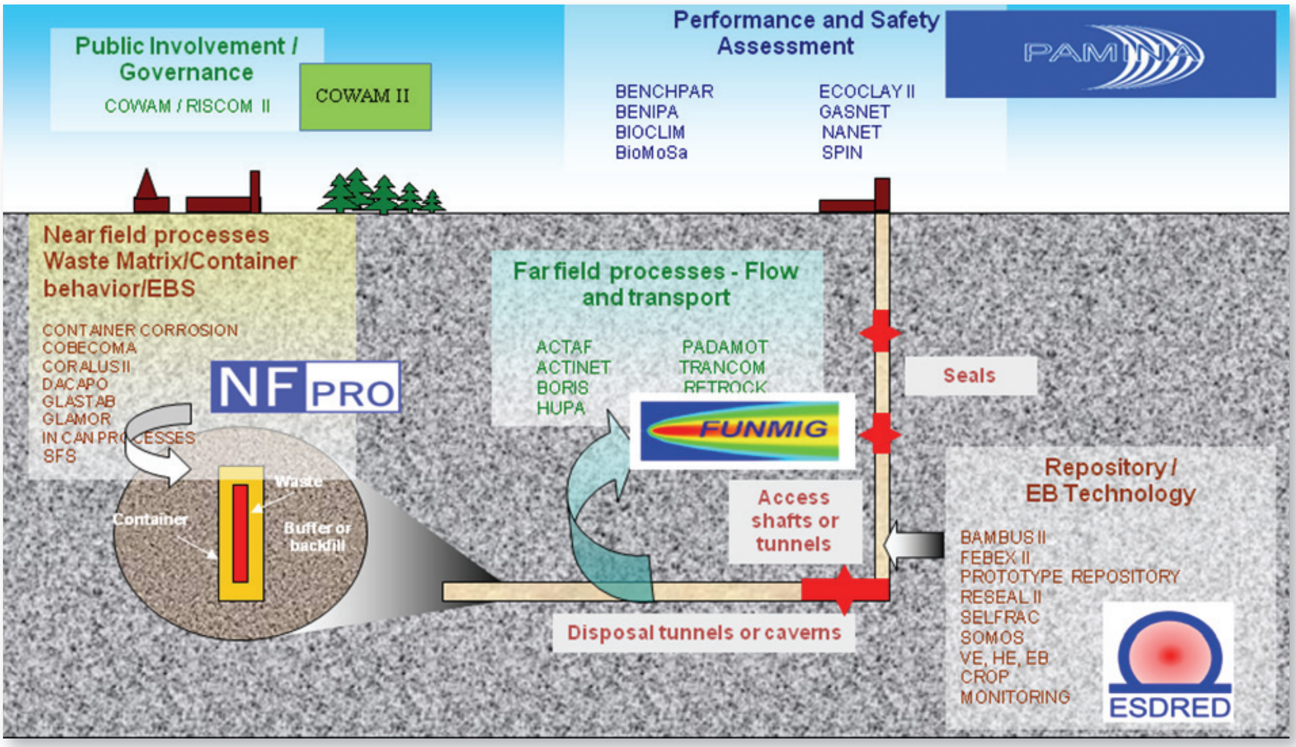
In the State of New Mexico/USA the Waste Isolation Pilot Plant (WIPP) is in operation since 1999. Transuranic (non heat generating) waste from the defence program is emplaced in underground vaults on the 650 m level. In 2006 the license was recertified.

Referring to the concept of deep geological disposal it has to be stated that in Germany the construction of the Konrad repository for short and long-lived radioactive waste with negligible heat generation to be emplaced at the 800 m level has been started in May 2007.

Research and technical development in radioactive waste disposal is promoted and to some extent co-financed by the EC. In the near future the 6th frame program is coming to an end while the 7th has already been started. The CARD Project, which stands for Concerted Action on Research and Development, is still to be finished. The objective is to outline possibilities and ways towards the establishment of a Technology Platform in the field of radioactive waste disposal. So far, views and opinions from authorities and stakeholders in different countries were collected and a concept for setting up and running a Strategic Research Agenda was drafted. A workshop presenting all findings and proposals to implementers, researchers and other stakeholders will be due in March 2008. The idea is to use the strategic research agenda as a tool for prioritisation and decision making in the course of the 7th frame program.

The EU frame program reaches from partitioning and transmutation (P&T) to radioactive waste characterisation and covers all aspects of geological disposal. With respect to P&T it should be made clear that this ‘future technology’ does not solve the problem of highly toxic and long-lived radioactive waste. The EU project REDIMPACT has revealed that P&T may result in the reduction of the high-level waste (HLW) volume and in a change of the radionuclide inventory. However, the remaining waste is still long-lived, and there is a surplus of non-high-level waste which is generated by the P&T process. The state-of-the-art in radio-active waste disposal research becomes evident by a detailed analysis of the ‘integrated projects’ which have been performed by a number of partners in the EU 6th frame program and, thematically related, in earlier research work (Figure 2).

- Project NF-Pro deals with all aspects of the so-called near field; with buffer, backfill and the excavation damage zone, just to name some of them.
- Project ESDRED is aimed mainly at the development (or better further development) of disposal techniques. A final work-shop will be held next year in Prague, presenting concepts and achievements to the ‘disposal community’ which includes explicitly organisations from new member states.
- Project FUNMIG covers flow and radionuclide transport processes in the far field, and project PAMINA is devoted to all aspects of performance and safety assessment.



▲ Fig. 2: Schematic overview on the ‘integrated projects’ in radioactive waste disposal performed in the 6^{th} EU frame program with research projects in previous EU-programs (Source: EU).

Most projects are structured in such a way that all three disposal options (granite, argillaceous rocks and rock salt) are addressed, as far as it is relevant and applicable. Many research issues have been addressed in the REPOSAFE conference. In the following some selected and specific issues are highlighted in order to outline the scope of safety-related research.

4 What Research has a Strong Impact on the Long-term Safety of Geological Repositories, in particular for High-level Radioactive Waste and Spent Nuclear Fuel Disposal?

In recent years the discussion has widened on the subject of future research and what is really needed in order to implement geological disposal as soon as possible. As already mentioned there are different views and standpoints, however, when it comes to the safety issue the position should be fairly clear:

- The disposal concepts including the emplacement technologies are safety-orientated and have been developed to an almost mature state. Future activities should be mainly focused on demonstration and adaptation to waste and/or site-specific demands.

- The so-called "repository near field" contributes largely to the safety of a geological repository. In depth characterisation of the physical and chemical parameters provides greater knowledge on the coupled long-term processes (THM-C). Key issue for research is still a more detailed assessment of the safety potential of the 'near field' including the reduction of existing uncertainties. From this point of view URL experiments form an indispensable part of the research agenda.

- The "repository far field" is to some extent part of the environment with its close to surface soils, geological formations and groundwater systems. Detailed survey of these conditions is mainly task of site characterisation and monitoring. Site-independent research may contribute to a better understanding of the physical changes applied by climatic changes and other natural events. Transport and retention of radionuclides (naturally occurring and/or released) are important research topics. Data and models of the repository far field are needed for making up what is called 'Total System Performance Analysis' (TSPA). Beside their importance for understanding present and future environmental conditions and processes, this kind of knowledge contributes particularly to confidence building with the public.

- All research topics provide input to what is called the "SAFETY CASE" for licensing and implementing a radioactive waste repository. This term is used as a synonym for many things, many things beyond safety alone. In France it is called "Dossier Argile" and in Germany it is called "Plan" which is based on all information required for the licensing of the disposal project. This includes in particular data and results from research used for the demonstration of long-term safety. As a matter of fact the safety case can provide guidance also for research and technology development, and can specifically put focus

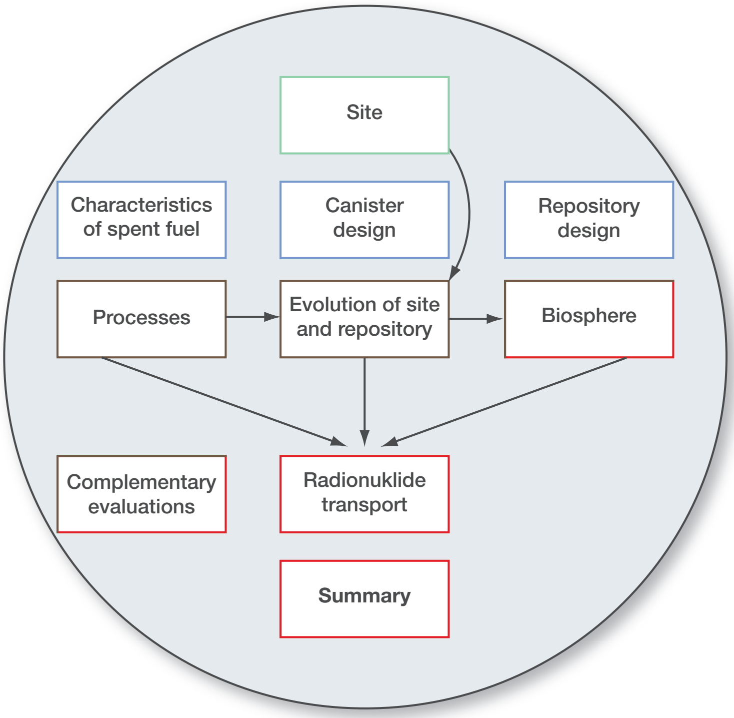


Fig 3: POSIVA Safety Case Portfolio (by M. Palmu, to be updated in 2008)

on aspects which are important for public awareness. Summing up, the safety case incorporates all findings and results which directly or indirectly are related to the repository project and conflates arguments and facts which contribute to the understanding of the repository's safety in the operational and post-closure phase.

A good example is given by the Finish waste disposal program for spent nuclear fuel which is proceeding as planned. The safety case is structured as a "PORTFOLIO" (Figure 3) containing compartments with the required issues for the design, the planning and the licensing of the underground repository. In this conference we elaborated on a number of these compartments of which each is a world on its own. It is quite a task to compile all know-how and expertise available under these headlines and to put everything into relation of a disposal concept and / or a repository site. However, only data and characteristics of a real site can provide the basis needed for both, the evaluation and the use of research results in preparing and presenting the safety case.

5 Which Key Issues of Safety-related Research were addressed in the Course of this Conference?

The international organisations IAEA, OECD-NEA and EU outlined their working programs as well as the state-of-the-art in waste management. Results and strategies are laid down in overarching position papers and safety guides as well as in technical documents and reports on research projects. One can say that all these products are setting the boundary conditions for any national waste disposal research program. Agencies from six different countries reported on their progress in geological disposal and in particular about the steps towards the realisation of a deep underground repository. Another four research organisations / experts from outside Germany gave insights into specific aspects of their research work which, to some extent, is governed by the individual national concepts and policies.

About 30 scientific papers presented were mainly focussed on geochemical issues in the near- and far field of the repository, on safety-related aspects including the safety case as well as on research preferably performed in underground laboratories. In the following some examples are highlighted in order to underline their importance in the scope of the entire repository system.

5.1 Spent Nuclear Fuel (SNF) Corrosion and Radionuclide (RN) Mobilisation

Long-term safety begins with the radioactive waste / SNF properties and the chemical conditions in the very near field of the repository. In contact with water or brines corrosion processes may trigger enhanced RN mobilisation.

By changing from reducing to oxidising conditions the RN release from SNF increases by orders of magnitudes. Laboratory experiments have provided a fairly good data base. Observations in nature have indicated the sensitivity of the alteration process from Uraninite U (IV) to secondary U (VI) minerals. Due to the buffering capacity in most deep repositories this process might be of minor relevance.

Considering this process in integrated performance assessment studies it becomes obvious that in specific cases faster corrosion can lead to higher radiation exposure shortly after shut down of the repository. In the Research Centre Karlsruhe (FZK) corrosion experiments on HLW glass and SNF have in particular been executed under laboratory conditions (see D. Bosbach, A. Loida, P. Panak (FZK) and E. Valcke (SCK/CEN)).

5.2 Radionuclide (RN) Transport and Retardation in fractured Granite

An outstanding feature of the Grimsel Underground Rock Laboratory (GTS) is the possibility to perform – in fractured granite – tracer tests with radioisotopes. The specific hydrogeological condition together with appropriate technical means allow for an almost 100 % recovery of the tracers from the surrounding rock mass.

In contrary to laboratory experiments in situ tests are run in a real geo-environment. The advantage is that test-conditions are provided by nature which is difficult to simulate in a laboratory. A disadvantage is the complexity of the fracture system which has to be characterised by specific exploration and monitoring tools.

According to T. Schäfer (FZK) a series of experiments was focused on radionuclide transport in the presence of bentonite buffer. At the interface with granite bentonite colloids might be formed. They act as “carriers”. This is can be of importance since high flow velocity proved that by colloidal transport actinides migrate without retardation in fractures.

5.3 Natural Analogue Studies for better Understanding of Radionuclide Migration in Geological Time Frames

The geochemical behaviour of radionuclides in the near-surface environment is governed by the characteristics of the sediments as well as by hydrogeological and climatic conditions. An important feature is the geochemical situation with changing oxidising / reducing conditions.

As presented by V. Havlova (NRI) and T. Brasser (GRS) such a complex system has been studied at Ruprechtov / Czech Republic. The local and surrounding granite provides a source for uranium. Part of the granite has been weathered forming kaolin. The bedrock is covered by Tertiary volcanic and fluvial sediments. The U-enrichment occurs in the direct vicinity of lignite seams. It was proved that to some extent migration of U (VI) and fixation of U (IV) took place. Microbial degradation of organic matter and formation of PO_4 played an important role in this entire process like the reduction of U (VI) on arsenopyrite and the presence of sulphate reducing bacteria. From measurements of U series disequilibria and ^{234}U / ^{238}U isotope ratios in the U (IV) phases it was derived that the U (IV) phases remain stable for long periods of time.

5.4 Repository Backfill for Stabilisation and Sealing of Underground Cavities and Disposal Rooms, a Barrier against Groundwater and Brine Intrusion and RN Transport

Each disposal concept provides for the insertion of buffer and backfill in the very near field of the waste canisters. In rock salt the use of crushed salt is most appropriate since the material properties are the same as of the host rock. The question is: How does crushed rock salt react under heat load and how long does it take to minimise its hydraulic conductivity?

In one of the largest heating test performed worldwide, second after the test in Yucca Mountain, compaction and porosity were measured for more than eight years in the Asse URL. At the same time a total of 2,8 Mio kW hs was applied in order to simulate SNF disposal conditions. The initial porosity dropped from 35% to about 20 %. This promising result indicates that in relatively short time (approx. 500 to 1000 years) salt backfill will be compacted completely forming an almost impermeable geotechnical barrier. For a more concrete and/or repository-specific determination of the endpoint or better of the long-term evolution of backfill compaction specific lab tests as well as the employment of adapted constitutive laws for simulation are appropriate tools. H.-J. Herbert (GRS) reported about tests with specific salt based materials for demonstrating their pronounced self sealing potential.

5.5 Diffusion Parameters in Argillaceous Rocks

Diffusion parameters characterise to some extent the isolation potential of argillaceous rocks. It is a fact that in laboratory tests technical problems with the conditions of testing plugs and samples are almost inherent. Under specific technical and geological conditions in situ tests can produce more representative data to be used in large scale simulation models.

Due to the great tightness of argillaceous rocks in situ diffusion tests will take a considerable amount of time. However, knowledge about pore water transport and distribution can also be gained from monitoring of the pore water pressure under strain conditions. As learnt from P. Bossart (Swisstopo) a heater experiment was performed in the Mt. Terri underground laboratory with monitoring the pore water pressure build up and successive relief. Using specific mini-packers designed by GRS it was possible to gain large and consistent data sets. According to these findings heating up the test field to about 95 °C did not cause any distinct damage to the adjoining rocks of the Opalinus Clay Formation. The pore water pressure in the monitoring boreholes increased slowly to steady state conditions. After shutdown of the heaters the pressure resettled gradually till constant (initial) state conditions were reached. The benchmarking with provisional modelling showed adequate agreement.

5.6 Gas Pressure build up in Rock Salt

Presently, a great issue of experimental and analytical research is the generation of gas in the repository as well as gas migration and possible pressure build up. At the beginning of 7th frame program two research proposals were filed to the EC requesting co-financing. They were so comprehensive that the Commission returned them and asked to combine them for the next call.

With respect to heat-generating HLW – emplaced in 300 m deep boreholes in rock salt – the gas generation from canister corrosion is in the range of some litres per year. This results from the liberation of very small brine inclusions in the salt and from a corrosion rate of about 1 µm per year. As T. Popp (IfG) explained in his presentation this process may lead to a pressure build up of some millibars per year in a sealed borehole. Field tests and laboratory experiments have proved that once the effective lithostatic pressure is reached gas migrates into the surrounding rock mass. This kind of “flow” is dilatancy controlled and causes no fracturing. This underpins the fact that in rock salt the gas issue is a non issue, at least for the borehole disposal concept.

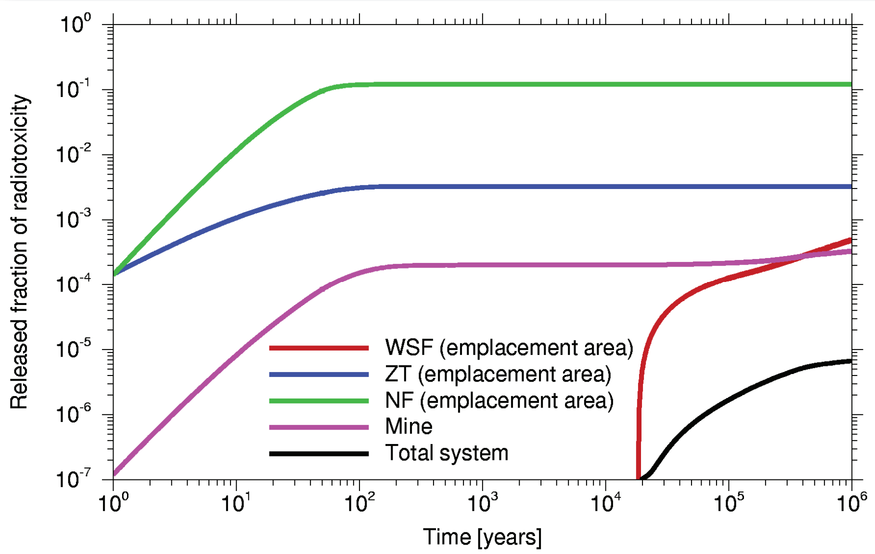
5.7 Performance Assessment (PA) Modelling for Detailed System Understanding and System Optimisation

As mentioned before, a repository for heat-generating and long-lived radioactive waste in deep geological formations is a complex system of technical and geological components which are subject to “short-term” and interacting processes as well as to “long-term” geological developments. In order to assess the overall performance of the repository system specific constitutive models, coupled processes models as well as conceptual repository design and site specific models are being developed. The combination of these models makes up what is understood as total system performance analysis (TSPA) which is a key “portfolio” (see above) of the safety case.

The work of BfS and GRS on the Morsleben repository for low and intermediate-level radioactive waste can be referred to as a concrete example for PA modelling. A primary aim was to check the efficiency and sustainability of the different geotechnical measures for stabilisation and sealing off the different disposal sections as well as parts of the access galleries. The analytical findings gave a good indication about the long-term performance of the individual sub-systems and of the total system. They have been also used for looking into the details of safety-relevant construction elements, site conditions and RN transport processes.

As mentioned by P. Brennecke (BfS) the calculation of the “release fraction of radiotoxicity”-radiotoxicity is the sum of the RN inventory multiplied by the nuclide specific dose factors – was used for modelling of the near field processes in the different disposal sections (WSF – west and south field, ZT – central section, NF – north field) taking into account different waste types as well as different backfill and sealing measures (Figure 4).

Extending this calculation to the integral release of all sections of the repository near field (purple line) it shows a total of about 0,3 per mille which is reached 100 years after shut down. The calculated release from the entire repository including far field with the covering rock formations (black line) starts at about 10,000 years after shut down and makes up 0,01 per mill of the total inventory after 1 million years.



▲ Fig. 4: Results from PA modelling showing the release fraction of radiotoxicity as a function of time for the different disposal sections (green, blue, red) as well as for the mine (without overburden), marked black and the total system including the covering rock formations of the so-called far field, marked purple (BfS).

6 What is the Role of Research in Confidence Building for HLW/SNF Emplacement in Geological Repositories?

As demonstrated by these few examples as well as by the different presentations in the course of this conference a geological repository for high-level and long-lived radioactive waste is a complex system. What makes things difficult are the long half-lives of parts of the radionuclide inventory, the long timeframes of decay to stable isotopes and the 'dynamics' of the geological system with its inherent 'uncertainties'. In context with modelling of such complex repository systems the following questions often arise:

- Is it possible to develop something like **one** realistic model of a site / repository?
- Is there a model at all we can have confidence in from a scientific point of view?

I guess there will never be just one answer to these questions – yes or no. It depends very much on the individual view point and possibly on each ones experience and skills. It depends also on the details and the degree of accuracy asked for. And there is still the matter of timescales. Do we talk about a model of today's situation or is it a model of future times when geological events like ice ages and volcanism etc. have changed our world? We even don't know if our species will still exist on this planet in one million years of time or so.

Presuming there is nothing like a realistic (long-term) model a rational approach has to be developed based on all data and scientific / technical know-how available. However, only data and characteristics of a real site can provide the rationale needed for both, the evaluation of research results and their use for the safety case. Some years ago T. Äikäs (Posiva) proposed how to deal with modelling of a complex system like a radioactive waste repository in deep geological formations – in a pragmatic way (Figure 5).

From the geological point of view one has to consider that some rock formations can provide more representative geo-information than others for "close-to-reality-modelling". However, this challenging task requires research with all its different disciplines and facets, today and in future. Developing multiple lines of arguments can underpin the key processes of the base case model.

In principle "no" – one realistic model not possible to develop

- ➢ Study more detailed the base case uncertainitiers
- ➢ Site ananlysis for studying the performance of the system
- ➢ "Horror" scenarios to cover uncertainties

Optimisation of technical concept

- ➢ Use geosphere data by best possible manner
- ➢ Design criteria and parameters
- ➢ Final selection of depth

Tasks for site characterisation

- ➢ Develop set of models consistent with data
- ➢ Building confidence in predictions regarding behaviour of geosphere
- ➢ Develop understandig of influences caused by construction

▲ Fig. 5: Example for the development of a PA model on a radioactive waste repository in granite, and the interrelation with data needs, R&D and confidence building (drafted by T. Äikäs, Posiva, personal information).

This takes us directly to the issue of confidence building. Confidence in the safety of radioactive waste disposal goes far beyond the confidence a scientist gets in his work and in his own results. Touching the present situation of HLW/SNF disposal one has to acknowledge that in some national programs, a selection was presented at this conference, current activities are in the stage of site identification and qualification. After managing successfully the final "Entsorgung" of low and intermediate-level waste (LLW, ILW) in well designed near-surface and underground repositories for more than 15 years confidence has grown in the feasibility of solving the problem of HLW and SNF disposal in the foreseeable future, at least in some countries. Despite of all technical and scientific capabilities building up confidence requires a well established culture of openness and communication. This is a multi-stakeholder/multi–parameter task which needs fairness und good will from each side (Figure 6).

What builds up confidence?

- ➢ A well balanced socio-political system
- ➢ Trust in the integrity of acting people
- ➢ Trust in an open and fair dialogue
- ➢ Trust in the skills of scientists and engineers
- ➢ Experiences shared with other countries and international organisations

Who needs confidence?

- ➢ Decision makers
- ➢ Implementers
- ➢ The Public, and
- ➢ Stakeholders in general,
  also scientists and researchers

Who can provide confidence?

- ➢ Politicians
- ➢ Implementers
- ➢ Scientists and engineers
- ➢ A Journalism based on facts

How can confidence be achieved?

- ➢ Well established safety criteria
- ➢ High quality science and engineering
- ➢ Stepwise approach in waste management
- ➢ Transparent and reliable licensing procedure
- ➢ Unbiased public information
- ➢ Political leadership (Bern Conference, 2007)

▲ Fig. 6: Parameters and stakeholders in the field of confidence building in the geological disposal of highly radioactive waste. From a scientist's viewpoint advanced research and shared experiences form an important platform for communication and decision making.

From the viewpoint of a Technical Safety Organisation there is a clear rationale behind all that: Research in the field of HLW disposal is not a matter of site-specific work alone. The subject is very much broader. Main objectives are:

- Incorporation or adaptation of the latest findings in the different fields of science and technical development,
- Optimisation of the disposal systems,
- Training of those who will have to license, construct and operate the repositories,
- Confidence building in waste management concepts and the long-term safety of geological repositories.

Stakeholder's confidence requires, among others, high quality in science and engineering. This can be achieved only by research on concrete projects with real perspectives. It is an obligation for those who are responsible for solving the HLW/SNF disposal issue to provide favourable conditions for research and for the research teams. However, all this makes sense only if there is a clear intention to solve the problem.

7 Conclusions and Closing Remarks

Besides the various research topics and the latest results on radioactive waste disposal the REPOSAFE conference provided an excellent overview on the development in countries like France, Sweden, Finland and the USA which are making a strong move towards site identification / qualification for a HLW /SNF repository.

As an overall assessment of the ongoing research activities in radioactive waste disposal and to some extent also as an outcome of this conference the following conclusions should be drawn:

- Disposal techniques for underground emplacement of high-level radioactive waste and spent nuclear fuel have been largely developed and tested, mostly under mock-up conditions. In order to confirm the mature state several research projects within the EU frame program are presently being executed. The future challenge is to test these techniques under 'hot' conditions and adapt / apply them to real waste repositories.
- Engineered barriers contribute substantially to the long-term safety of repositories in deep geological formations. There are no doubts that, in addition to various conceptual plans and material investigations, the demonstration of the long-term performance has still to be tackled by large scale in situ experiments – as far as possible – under realistic conditions.
- Safety assessment of HLW/SNF repositories is, in general, still a main task of scientific research. The development and qualification of applicable computer models require an in-depth system understanding which again needs a multidisciplinary data base reflecting most of the long-term processes and material parameters. This kind of research has still to be carried on and will cover, among others, subjects like "natural analoga" in order to validate computer models and to gain greater confidence in long-term predictions.

However, as scientists we have to check our work and ask ourselves:

Are we doing the right things in order to gain that know-how and expertise which is needed for solving the problem of HLW disposal? In what way can we contribute to the generation of confidence?

My answer on that is very simple: Let's start with smaller problems or better with the manageable problems first. By that approach one has a chance to learn about the repository system from basics, and by the time one gains that kind of know-how and experience which is needed for tackling the next problem. In the real world this is learning by doing or in a more technical language we call it stepwise approach. This builds up confidence in a practical and most convincing way.

Reflecting the German situation I only can welcome the final decision on the Konrad repository. The abandoned deep iron ore mine which was investigated on its suitability from 1976 to 1982 and which was licensed in 2002 after an extensive process will become Germany's repository for all types of short-lived and long-lived radioactive waste with negligible heat generation. In this context I wish to thank Minister Sigmar Gabriel (Federal Minister for the Environment, Nature Conservation and Nuclear Safety) for his clear words at the beginning of this conference.

The REPOSAFE conference is not over yet, and many of you will take the opportunity to visit the German projects, but let me take the chance to thank you all for coming to this conference, the first international conference on radioactive waste disposal in Germany for three years. Specific thanks to all participants from European countries and from abroad. I hope you enjoyed your stay in Braunschweig. And last but not least thanks from the organisers to all presenters for their high quality papers and posters. Despite all interesting and challenging scientific tasks, 30 years of radioactive waste disposal research gave me also most fruitful ideas and enjoyable contacts with scientists, engineers and managers from Germany and abroad. Thank you all! ■

3.06 100 Years Salt-Mine Asse II

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Abstract
Salt Mine Asse II, driven in 1906, was used as a salt-mine for nearly six decades. In 1964 the exploitation ended. Asse II was used for research and development in disposal of radioactive waste. From 1967 to 1978 low- and medium-active waste was stored in the mine. Series of different technologies to store radioactive waste were carried out. Tests for the disposal of high-active waste were performed, though this type of waste never entered the mine. In total 125.000 drums with low-active waste and 1.300 drums containing medium active waste were stored in the mine. Since 1979 Asse II was used for purposes of research in the safe long-term storage of radioactive waste. This work ended in 1993, since that time Asse II is prepared for closing-down according Federal Mining Law. Long-term safety shall be guaranteed by refilling the workings with salt and gravel to stabilize the mine, closing the pore space with protection fluid against the disintegration of carnallitite and building of flow barriers which prevent that liquid will flow through the disposal chambers and dissolve radionuclides in the post operational phase. Without further delay the mine will be closed-down completely in 2017.

1 Introduction

The salt-mine Asse II looks back on an eventful history of more than 100 years. Let me introduce you to the most important events during this century.

The first part of my presentation is about the exploitation of potash salt and rock salt. The second part is about the disposal of radioactive waste. Afterwards Asse II was dedicated to the research of safe long-term disposal of radioactive waste.

The last and most extensive part deals with the closing-down of Asse II. If all necessary work is done in accordance with the closure operation plan, this mine will be the first one worldwide which is closed with stored radioactive waste inside.

1.1 History, Geologic Facts, Responsibilities of Asse II

The Asse is a range of hills of about 8 km (5 miles) length which extends from west-north-west to east-south-east in the northern foreland of the Harz mountains.

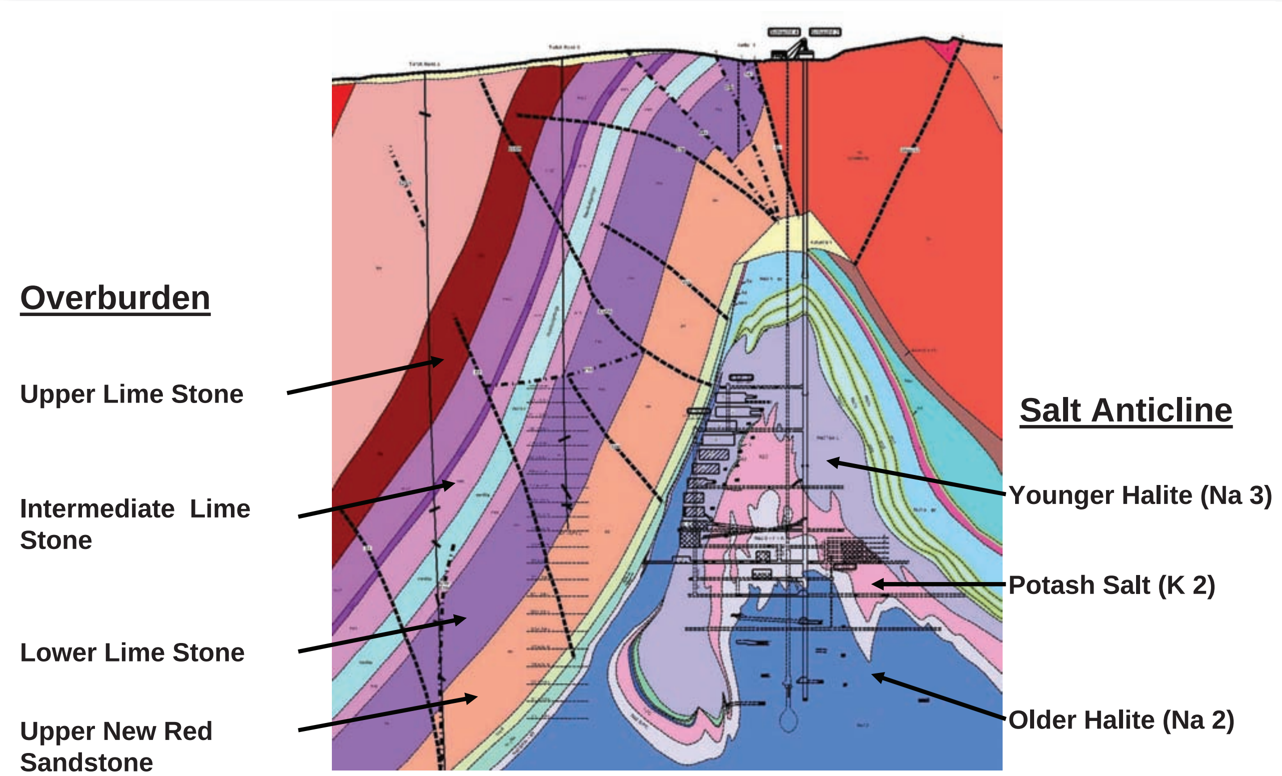
The Asse Research Mine belongs to the GSF – National Research Centre for Environment and Health, Munich. It is under the responsibility of the Federal Ministry of Education and Research which is also financially responsible.

A cross section through the anticline is shown in the figure below (Figure 1).

The salt rocks which can be seen in the core of the anticline were formed by evaporation of the ocean 250 to 220 million years ago in the geological period of Upper Permian. 110 million years ago, during the period of Upper Cretaceous, the formerly flat layers were tectonically folded to the Anticline we know today.

The salt anticline mainly consists of salt rocks of the Staßfurt-Series (Na 2 and K 2) and of the Leine-Series (Na 3). Its basis is in about 2.200 m (7.200 ft) below the surface.

The overburden at the South Flank consists of Upper New Red Sandstone and Lower, Intermediate and Upper Limestone.



▲ Fig. 1: Cross section through the anticline

Inexpert mining of potash salt caused a flood of the workings in shaft Asse I. This is why another shaft, Asse II, was sunk from September 1906 to November 1908 to a depth of 765 m.

In the North Flank of the salt deposit potash salt was mined from 1909 until 1925. All of these rooms were refilled with tailings of the milling process.

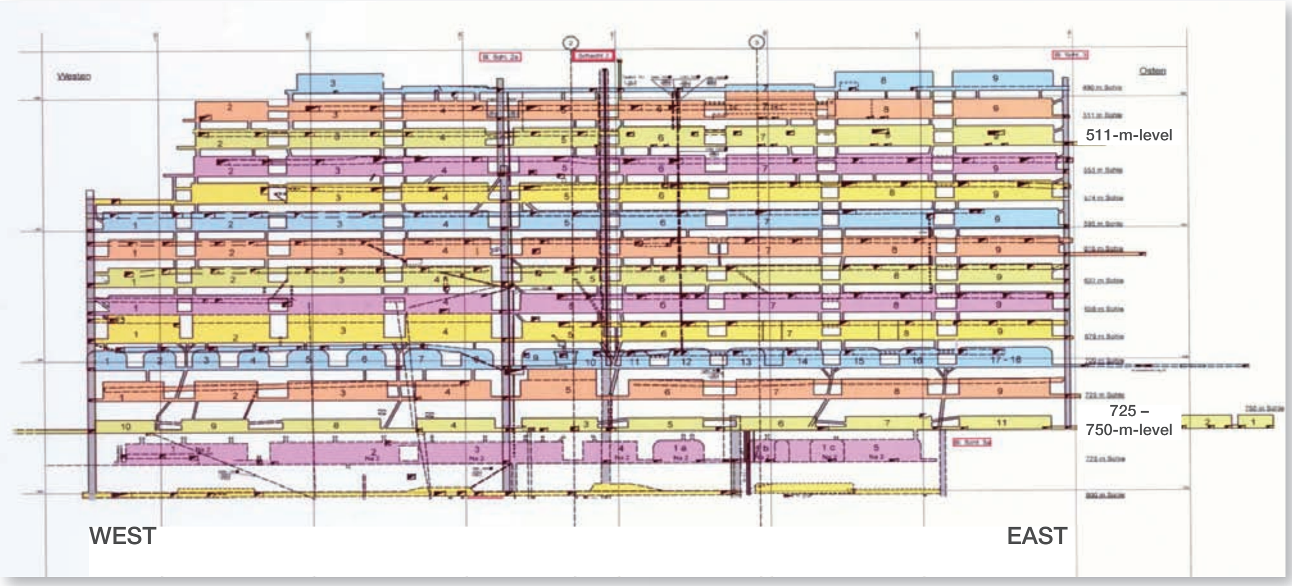
In 1916 extraction of rock salt was started on the 750-m-level in the Younger Halite and 11 years later, in 1927, as well in the Older Halite. This exploitation was stopped on March 31st, 1964 for economical reasons.

Turning the cross section through 90° the next figure shows the longitudinal section of the mining cavities in the South Flank of the Asse salt-mine.

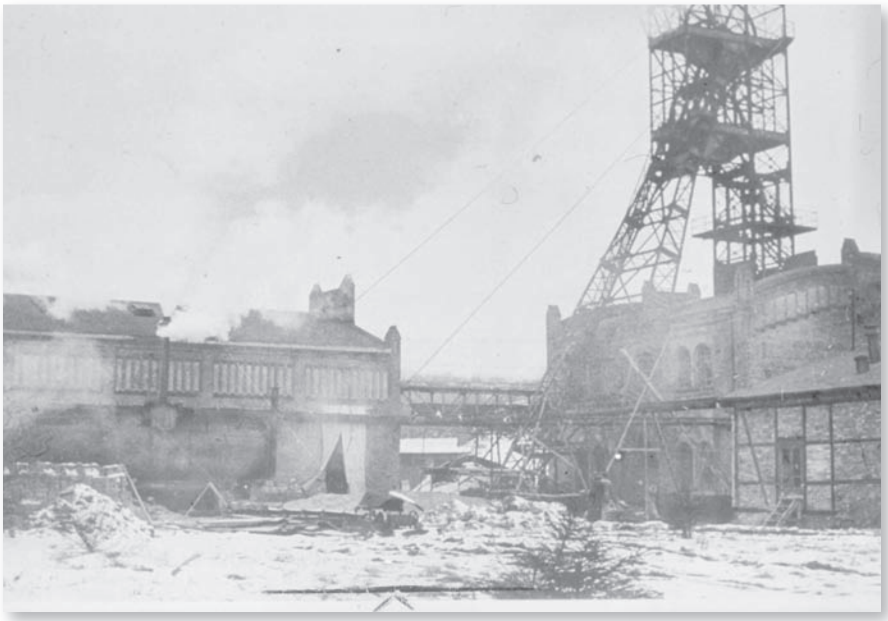
When the mine was closed, a total of 131 rooms with a volume of 3.5 million m³ had been developed on 13 different levels to a depth between 750 m and 490 m.

A normal room is 60 m long, 40 m wide, 15 m high and has a volume of approximately 36.000 m³.

Between rooms on the same level pillars were left with a width of 12 m and between rooms situated on top of each other horizontal pillars were left with a thickness of 6 m.



▲ Fig. 2: Longitudinal Section of Salt Mine Asse II – The Cavaties in the South Flank after Stopping the Mining Activities in 1964



◄ Fig. 3: This photo shows the condition of the aboveground buildings in the year 1964 after the exploitation of the mine was stopped. On the right side you can see the hoist frame on the left side the building for the hoist engine. Both buildings are now listed monuments.

2 Disposal of Radioactive Waste

In 1965 the GSF purchased the former salt-mine Asse II, to perform a research- and development-program for the disposal of radioactive waste.

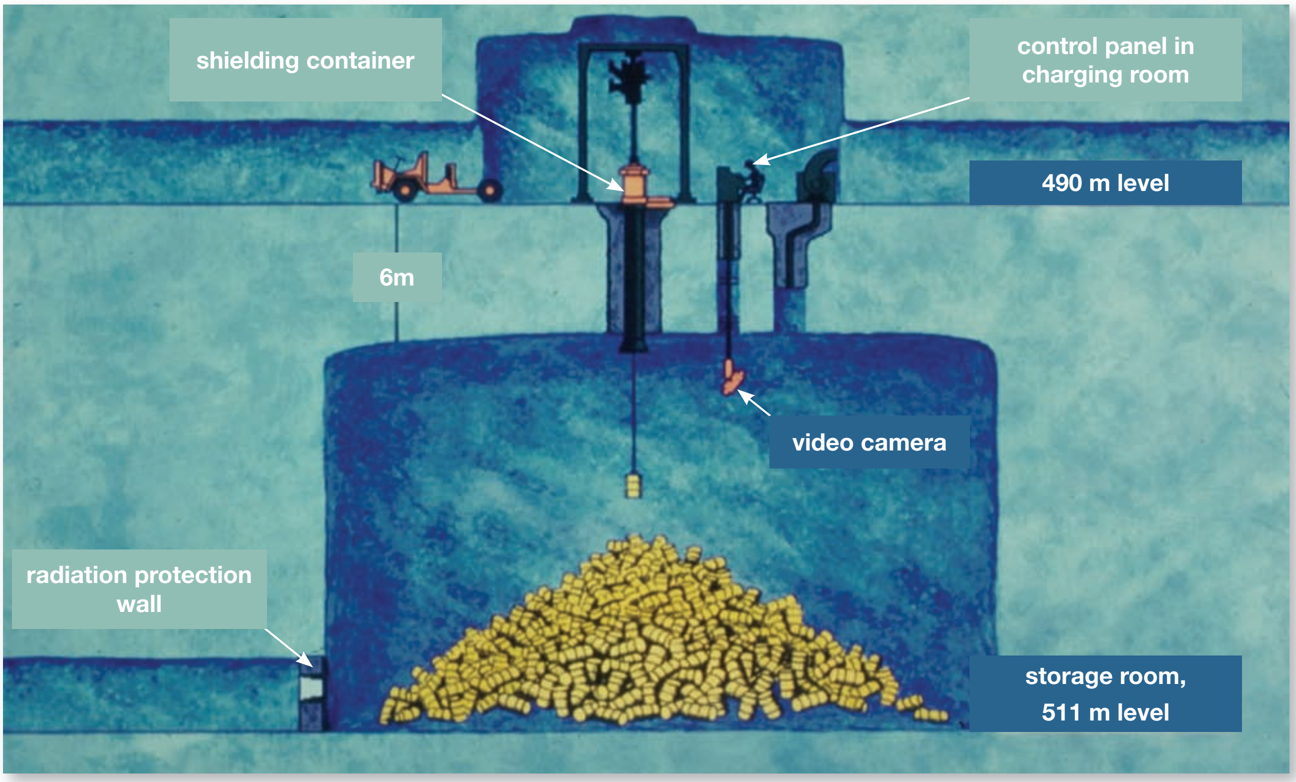
The disposal of low-active waste was carried out from April 4th, 1967 to December 31st, 1978 on the 750-m- and the 725-m-level in ten old chambers on the South Flank in the Younger Halite and in two old chambers in the core of the salt anticline.

The disposal of medium-active waste was carried out from August 31st, 1972 to January 17th, 1977 on the 511-m-level in one old chamber on the South Flank in the Younger Halite.

◀ Fig. 4: Disposal of low-active waste (horizontal stack)

The first barrels of low-active waste were stacked one on top the other by a forklift which was equipped with a barrel clamp.

The disposal of low-active waste was continued in chambers where the drums were horizontally piled in layers of (up to) ten drums. In 1974 a new technology was applied. The chambers were filled from above with drums which contained low-active waste. The figure shows the scheme of the disposal of medium-active waste.



▲ Fig. 5: Since 1974 medium-active waste was stored on the 511-m-level. The inside of the storage room was supervised through lead glass window in the radio protection wall and by video camera.

The barrels were tipped over the embankment by front end loaders. Loose salt was spread in layers over the barrels at intervals. These shielding layers made the roadway for the front end loader.

The time the staff had to stay in the chamber was reduced to a minimum by this technology.

Because of its higher radioactivity and the correspondingly higher dose rates, this waste could only be transported and manipulated within a shield. For the same reasons it was impossible to enter a room in which medium-active waste is stored.

Therefore, another technique was developed for the disposal of medium-active waste and tested in the Asse salt-mine.

First of all a former room on the 511-m-level was prepared as a storage room. The only connection between this room and the underground workings was closed by a radiation protection wall of concrete with a thickness of 80 cm (2.6 ft). A lead glass window was inserted into this wall for purposes of control.

Above the storage room a stope was turned on the 490-m-level as a charging room, where all technical equipment was installed. The two rooms are separated by a horizontal pillar of 6 m thickness.

The barrels containing medium-active waste were delivered and brought in the rooms in shields. The barrels entered the storage room without their shields by crane. Everything was monitored by video camera and could also be supervised through the lead glass window. The entire operation was steered by a main control desk.

In August 1976 the 4th amending to the Atomic Energy Act became effective including new juridical basic requirements. It regulated the responsibility for the long-term disposal of radioactive waste. For building and operating long-term repository sites it was necessary to pass a plan approval procedure.

The procedure was not carried out in Asse II, as the last licenses ran out at the end of 1978 and could not be prolonged.

Till then about 125.000 barrels containing LAW were stored on the 750-m- and 725-m-level and about 1.300 barrels containing MAW were stored on the 511-m-level.

3 Research and Development Work for the Long-term Disposal of Radioactive Waste

In 1979 the mine faced new challenges as the storage of radioactive waste had ended. In negotiations between representatives of the Federal Republic of Germany and Lower Saxony there was made the decision that shaft Asse II should be for the exclusive use in purposes of research and development work for the long-term disposal of radioactive waste.

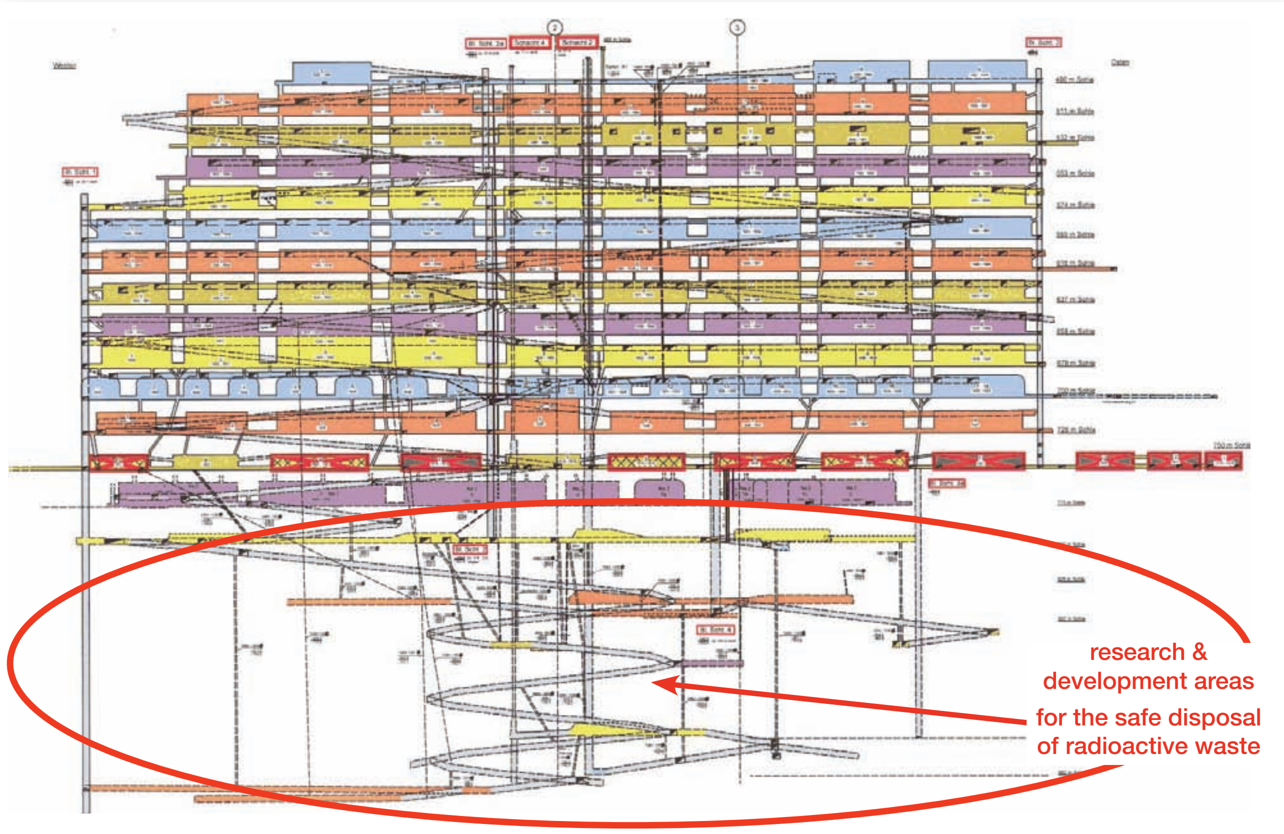
Therefore in the 1980's below the existing workings a new one had been drifted in the depth between 800 and 975 m. In this area scientific experiments for the planned repository site in Gorleben should be carried out.

For this purpose the shaft 2 was also extended from the original level of 775 m to a depth of 950 m. At the same time the spiral chute was extended along the periphery of the safety pillars by the shafts 2 and 4 down to the 975-m-level.

In this area six galleries with a length of about 180 m each were also driven to store the brine inflow into the mine. Two blind shafts located at the sides of the workings were extended to ensure the mine ventilation. Test fields were mainly prepared on the 800-m-, 875-m- and 950-m-level.

The photo below Figure 6 shows an example of the planned experiments for the final disposal of high radioactive waste tested with dummies.

In the foreground you can see the transport vehicle which puts the single transport container for the test-drums on the storage drill hole. In the background you can see the storage machine. An experiment with high-active waste did not take place.



▲ Fig. 6: Research and development works: Extension of the mine

◄ Fig. 7: Demonstration of the disposal of high-active waste (test with empty drums)

4 Closing-down of the Mine

4.1 Technologies

In 1993 all experiments were cancelled for political and financial reasons. As there was no further use for the shaft Asse II, the closing of the mine is prepared according to the Federal Mining Law.

The first step to stabilize the mine was refilling the chambers in the Younger Halite in the South Flank with tailings of the former potash mine Ronnenberg near Hannover.

Between August 1995 and the end of 2003 daily approximately 1.200 t of salt were mined from the dump by hydraulic excavators. The salt was sieved, dried, loaded in freight carriages and transported by railroad to the Asse mine.

Here the carriages were unloaded in a special bulk storage unit. Afterwards the backfill was transported by conveyor belt to a pneumatic stower, then by pipeline from the surface directly into the underground cavities which had to be filled.

The maximum haul was about 1.200 m. On the average capacities of 130 t per hour were reached under normal conditions.

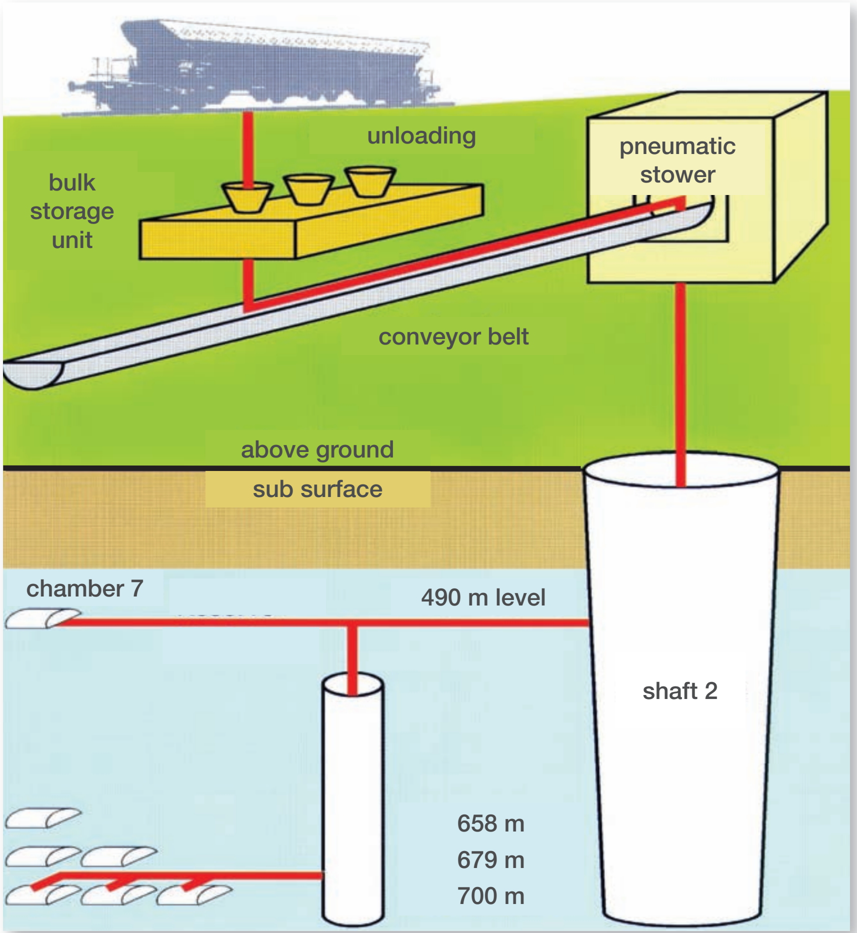
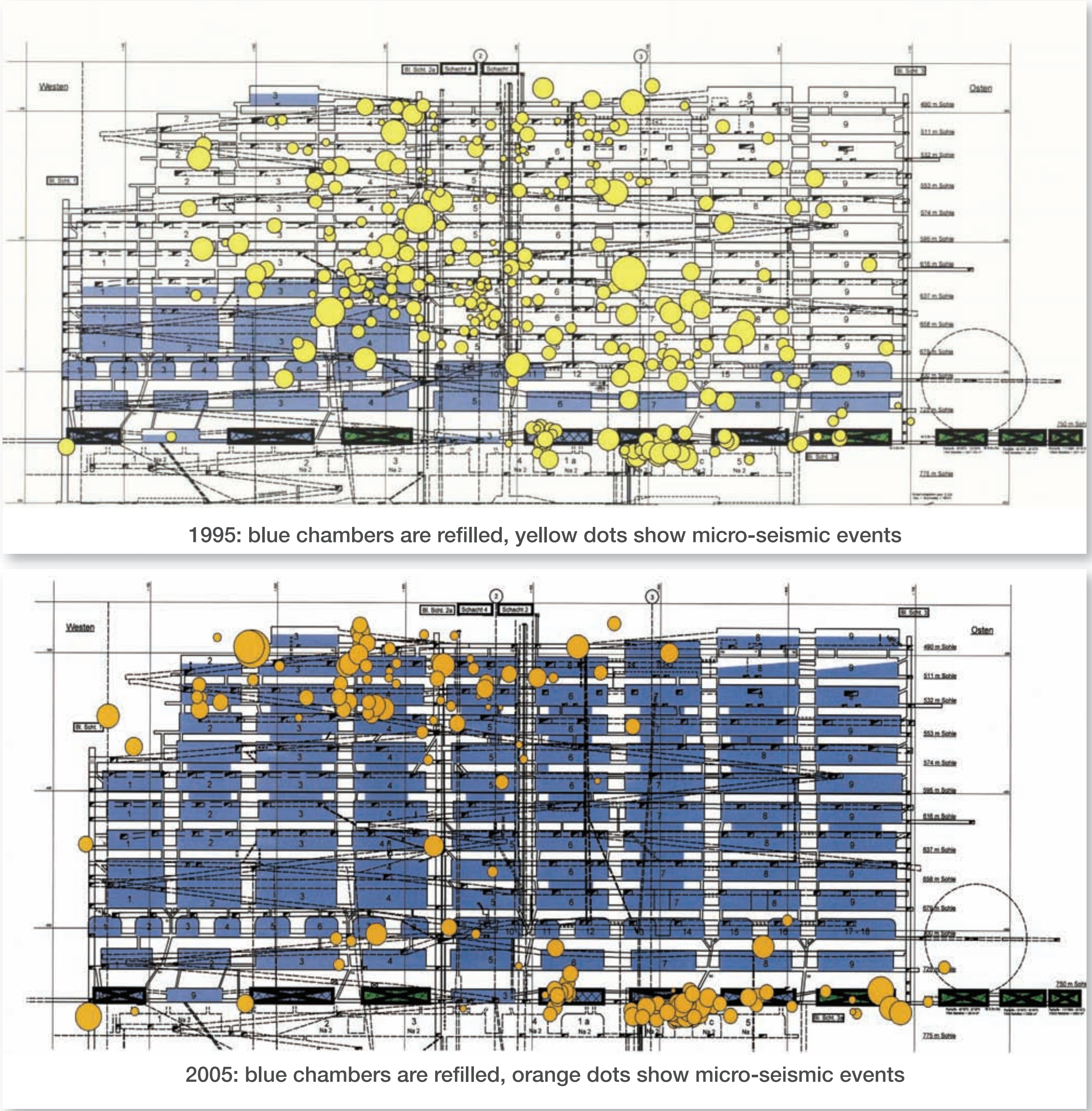


Fig. 8: Diagram of refilling the mining in the south flank with tailings from the dump in Ronnenberg

At the beginning of 2004 the backfill delivered from the dump in Ronnenberg amounted altogether to 2.2 million tons of salt. To determine the effect of the refilling activities, various measuring instruments were inserted in the horizontal and vertical pillars



1995: blue chambers are refilled, yellow dots show micro-seismic events

2005: blue chambers are refilled, orange dots show micro-seismic events

▲ Fig. 9: Microseismic activities in 1995 and 2005 in the South flank

before the operation started. Among others strain gauges (extensometer, inclinometer), stress-meters and geophones are concerned. In the pillars a strong acceleration of the strain rates occurred since the middle of the 80's.

After the refilling started a tendency to slower increasing strain rates is ascertained nearly everywhere, which is an effect of the refilling activities.

The microseismic activity in the area of the chambers of the South Flank in 1995 is shown in the upper diagram of Figure 9. Each yellow point represents a microseismic event. Its strength is marked by its diameter.

Clearly visible are the widespread places where these events were located, they occurred everywhere in the not yet refilled area of the workings.

In 2005 the seismic events – here figured as orange points – have decreased very much both in number and intensity in the meanwhile refilled areas.

We recognised in the areas which have been refilled for a longer time a definite decline of microseismic activities whereas in the part of the workings which was refilled only a short time ago still several clearly noticeable microseismic activities were located.

On the whole we are able to state that the backfill in the South Flank did already help to stabilize the workings.

4.2 Legal Conditions

The closure of the mine was already planned during the refilling of the chambers in the South Flank.

Presupposed is a closure operation plan based on the Federal Mining Law and licensed by the mining authority. The closure operation plan contains a general description of the planned realization of the closure operations.

An important appendix to the closure operation plan is a safety report. This safety report contains a long-term safety assessment to demonstrate that the regulatory safety requirements will be fulfilled during the post-operational period.

4.3 General Set-up and Safety Requirements

The shaft Asse II was neither planned nor advanced for the disposal of radioactive waste. From 1909 to 1964 it was used for extracting minerals as its one purpose. Only when the mining had been given up, the storage of radioactive waste was carried out. Because of this historical situation there is a general set-up of three basic conditions which essentially influence the closure concept.

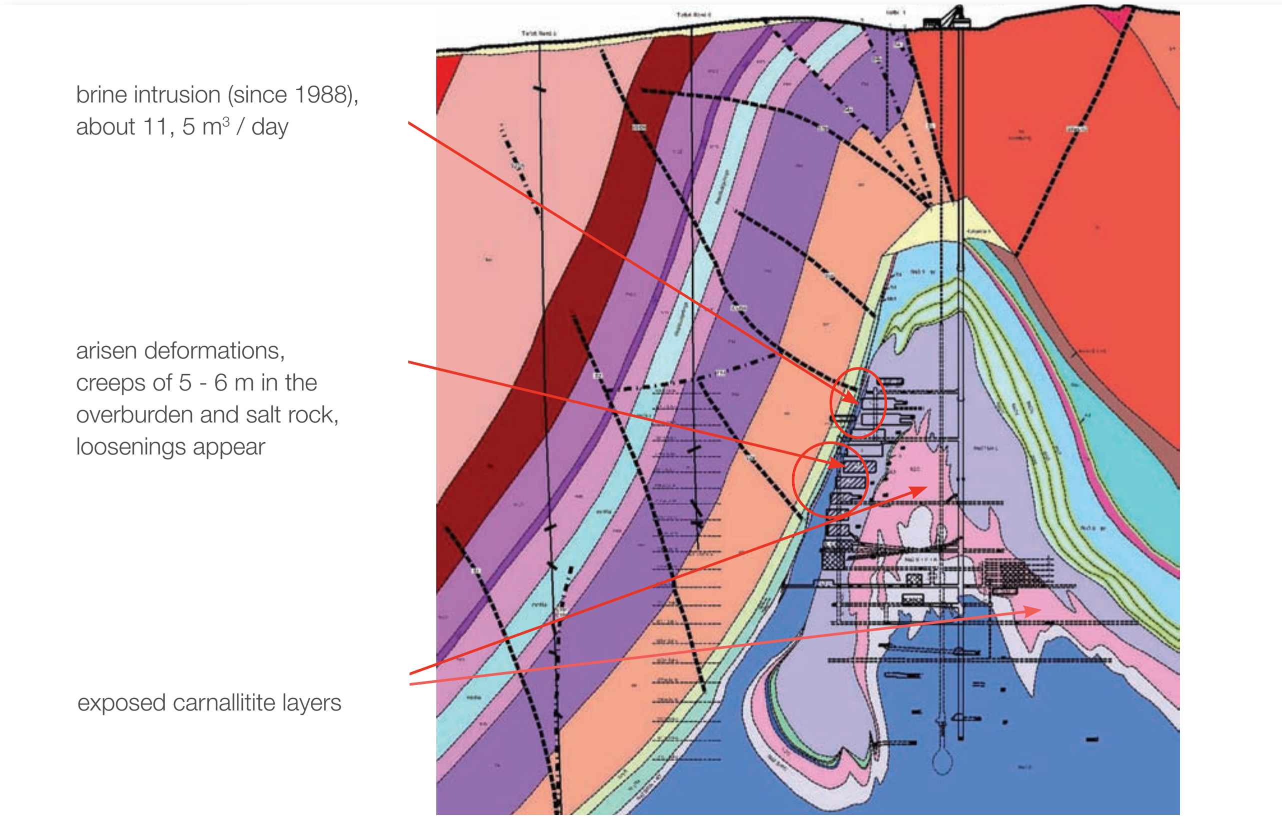
The first basic condition is the long cut-off of the mine and the resulting geomechanical situation. The second basic condition is a brine intrusion into the mine openings of shaft Asse II that occurred 20 years ago and never vanished. This inflow is connected with the third basic condition, the exposed carnallitite everywhere in the mine. The underground facilities in the mine openings of shaft Asse II were designed for the mineral production. The mining chambers lie in the upper area of the workings in the South Flank very close to the edge of the salt anticline. Therefore, the rock salt barrier is extremely thin there.

As a result of the intensive mining and the long cut-off the supporting system of pillars has become elastic throughout the decades. This is why deformations in the workings, as well as in the overburden arose. The salt rock and the overburden sag, creeps of 5 -6 m were ascertained in the area of the chambers. Because of these movements of the overburden, zones of loosenings form. The detachment of the rock salt and of the adjacent overburden will proceed.

The second basic condition is the brine intrusion in the south Flank. This brine inflow occurred for the first time in August 1988. It is a saturated solution of NaCl (sodium chloride) which cannot dissolve rock salt. The quantity of brine which is gathered in 24 hours sums up to approximately 11.5 m^3 (about 400 cubic feet). Its chemical composition, temperature and density are nearly stable up to now.

The third basic condition is the exposed carnallitite layers nearly everywhere in the workings. They are inseparable connected with the brine inflow as it can destroy the carnallitite.

It is to be supposed that in the post operational period the inflow of brine will continue undiminished and fill the pore space in the backfill material. Contacting the carnallitite, these layers will be disintegrated by the brine. Especially above the 750-m-level the disintegration of carnallitite rocks – if it ever occurred – would have considerable consequences for the stability of the workings in the post operational period.



▲ Fig.10: Basic conditions in the mine

In that case a destabilization of the workings would occur as a result of the fragmentation. Raised strain rates in the overburden could also be expected. Fracturings up to the surface could not be precluded. The general set-up in Asse II leads to the following consequences for the closure concept:

- On account of the existing brine intrusion a dry long-term storage of the radioactive waste is impossible.
- The overburden and the salt barrier make a limited contribution to the isolation of the waste.

Additional technical measures of closure are unavoidable to fulfil safety goals.

The tasks shown in the green fields are already finished (Figure 11). These are the refilling of the rock salt chambers in the south Flank and the refilling of the potash chambers in the North Flank.

At the moment the labours shown in the yellow fields are carried out. To protect the exposed carnallitite layers in the post operational period against fragmentation by brine intrusion, the remaining cavities are filled with salt or gravel and a protection liquid (saturated magnesium chloride solution). Near the storage chambers numerous flow barriers made of a special concrete (Sorelbeton) are built. They shall reduce the fluid circulation in the surroundings of the disposal chambers.

The labours shown in the red fields did not yet begin. The refilling of the remaining pore spaces in the disposal chambers with brucite ($Mg(OH)_2$) will be carried out to reduce the mobilization of radionuclides from the disposal chambers.

In the end the shafts Asse II and IV will be filled and sealed to limit the release of liquids via the shafts.

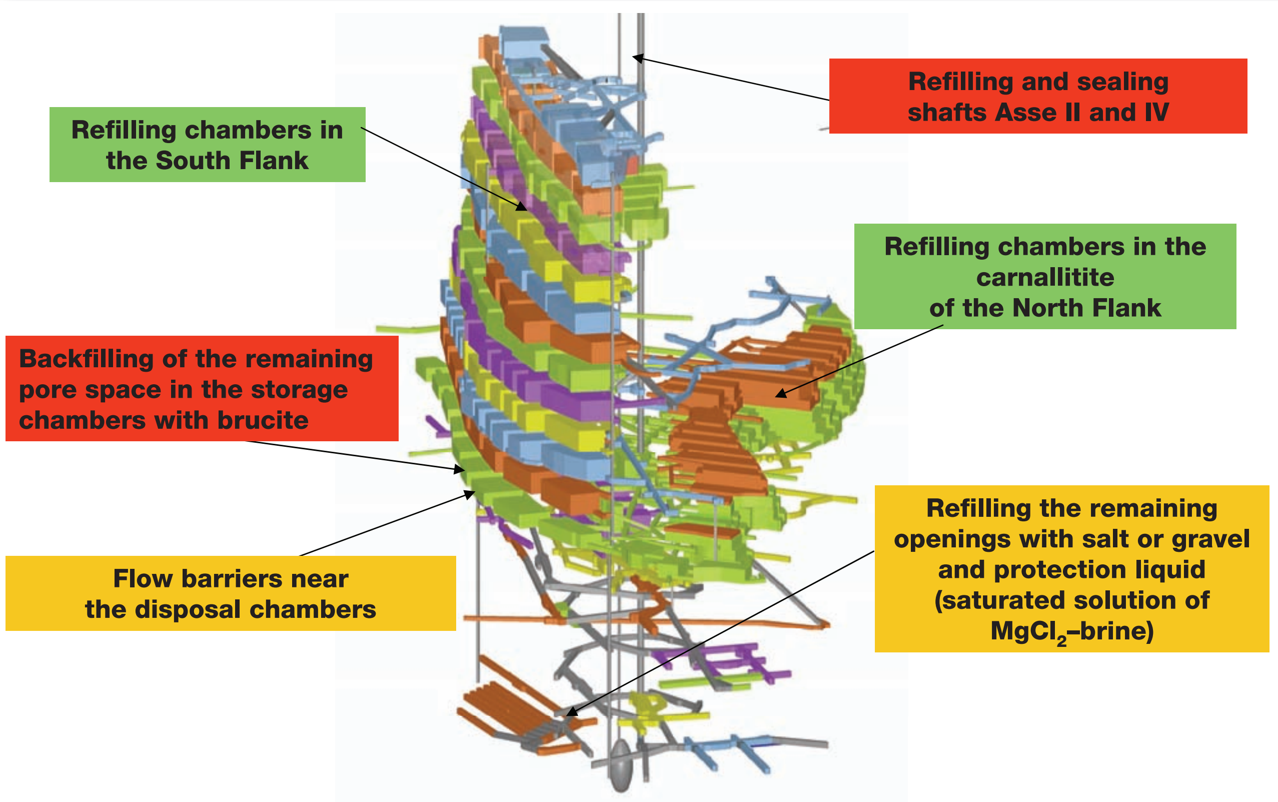


Fig. 11: Safety Modules

4.4 Present Situation of Work for the Closing-down

The shafts Asse II and IV were withdrawn and retreated to a level of above 800 m (starting in a depth of 925 m). To enter the mine from surface is possible but to the 490-m- and the 750-m-level.

The cavern below shaft IV is filled with shotter and protection fluid, its entrance is filled with Sorel concrete. The blind shaft 1 at the western ending of the working is filled with gravel.

The workings between the level of 975 and 925 m – mostly galleries – are filled with salt and protection fluid.

In the end of July 2007 about 74.000 tons of rock salt, 22.600 tons of shotter, as well as 12.000 m^3 of protection fluid and 6.000 m^3 of Sorel concrete will be filled in the deepest parts of the workings. The ongoing filling of the depth will be realized with salt from shaft Asse which is obtained as debris in cutting processes.

In the beginning of 2007 another important part of the closure concept was begun with. It is the construction of flow barriers in the surroundings of the disposal chambers. The flow barriers are built to prevent a flow of protection fluid, which is located in the pore space of the debris, through the disposal chambers when it will be forced up by convergence from the lower parts of the mine in the post operational phase. The way the flow barriers work is shown in Figure 12. Without them protection fluid may flow through the disposal chambers, dissolve radionuclides and carry them across the mining. The flow barriers prevent these possible movements. Flow barriers are simple and robust buildings, of about 30 m length in average and the diameter of a normal gallery. In a first step supports are built at both ends of the barrier. The flow barriers are made of Sorel concrete. This concrete is named by its inventor and used successfully for more than a century for sealings (e.g. in drilling) in salt and potash mining.

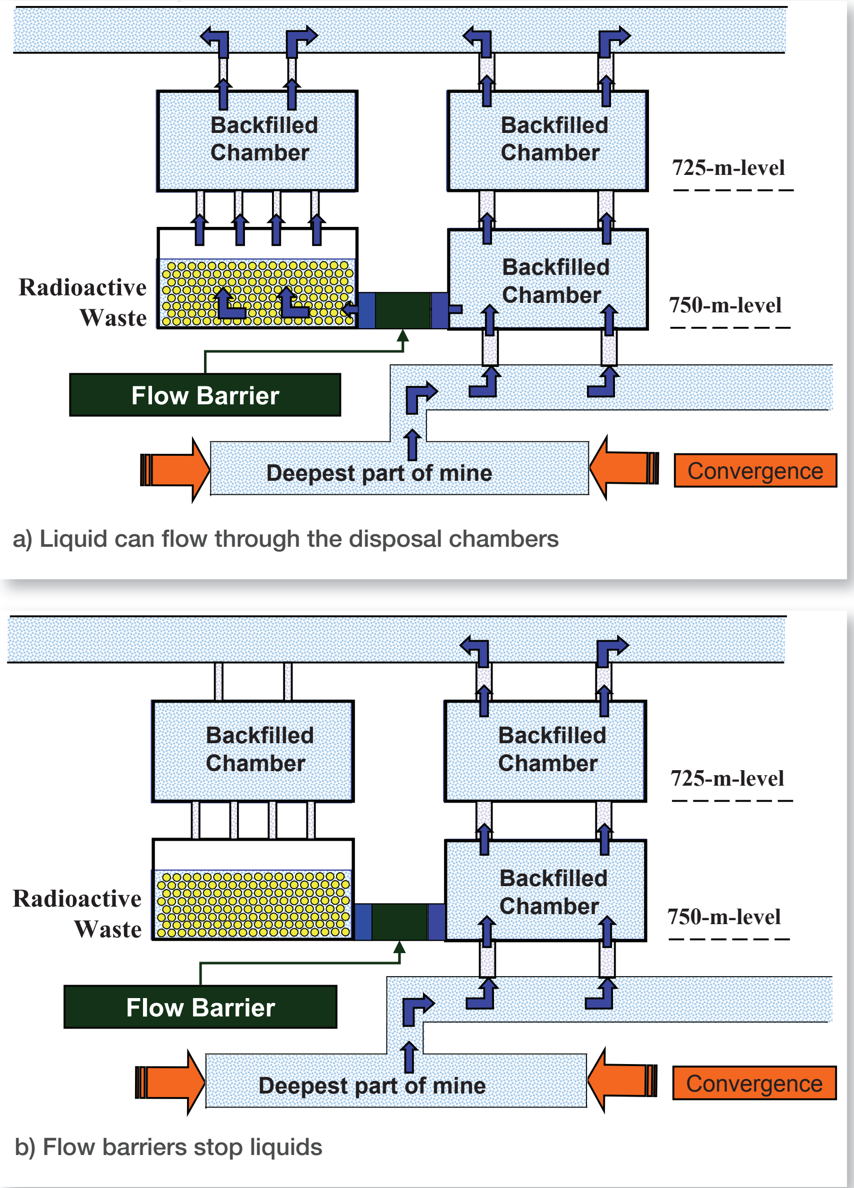


Fig. 12: Flow barriers and how they work

There are differences in composition between Sorel concrete and conventional concrete which is e.g. used in house building. In Sorel concrete the liquid is Magnesium chloride solution, the additional is rock salt with grains of up to 4 mm, and the bonding agent is Magnesium oxide. As the protection fluid which is filled into the mine is also a saturated solution of magnesium chloride, Sorel concrete is long-term permanent. Flow barriers are built in four steps:

- First boreholes for the filling and ventilation are made, which is usually done from the galleries above.
- Afterwards the entire walls have to be cut from loosenings by part-face heading machines.
- Then at both ends shutters are built in which the supports are poured.
- Only then the flow barrier can be built and the further refill of the mine continues. To avoid the occurrence of loosenings after the cutting, the flow barrier has to be completed within three month.

Fig. 13: Sorel concrete is poured in the prepared gallery on this photo. The concrete extends very evenly.

Loosening zones are extinguished by a part-face heading machine. According to the degree of loosening, the walls are cut to a depth of 1 m. To build shutters for the supports which are poured between them, fundaments are poured, then shutters can be made in two ways: a) They are build of lime-cemented sandstone b) Recyclable walls of wood are raised.

Above ground level, on the area of Asse II, a bulk storage unit was prepared. It consists of three silo-like buildings of 26 m height with a capacity of 600 m^3.

Material can be delivered by freight carriages or trucks to the Asse and be unloaded into the storage units. It is going by blow pipes down to the level of 700 m.

Fig. 14: This photo shows a part-face heading machine ready to cut loosenings in a gallery which is prepared for building a flow barrier.

▲ Fig. 15: This photo shows where rock salt and bonding agent (MgO) are blended. The unit is on the 700-m-level.

Down at the 700-m-level, broken and sieved rock salt (additional) is prepared and mixed with the delivered bonding agent. This is the so-called “pre-product”. The material is pneumatically transported to the 750-m-level. In a mobile unit it is blended with liquid ($MgCl_2$-solution) and pumped to the place where the flow barrier is poured.

Altogether in the closer surroundings of the disposal chambers 65 vertical and horizontal flow barriers will be built. Adding the volume of the supports and additional areas of upholding backfill till 2014 about 500.000 m³ will be manufactured.

The transport of the magnesium depot into the depository chambers is being done soon. It is planned to fill the rest of the pore space in the backfill above the 700-m-level with protection fluid from 2014 on. Both shafts, Asse II and IV, will be closed in 2017. ■

3.07 Survey on the German Repository Projects

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Under the custody of the BfS are the Morsleben repository for low and intermediate level radioactive waste, located in the federal state of Sachsen-Anhalt, the mine for the exploration of the Gorleben salt dome and the Konrad repository, both located in the federal state of Niedersachsen.

▲ Gorleben – Mine for the exploration of the Gorleben salt dome

▲ Salzgitter – Konrad repository (in construction)

▲ Morsleben – Repository for low and intermediate level radioactive waste (ERAM)

1 Morsleben Repository

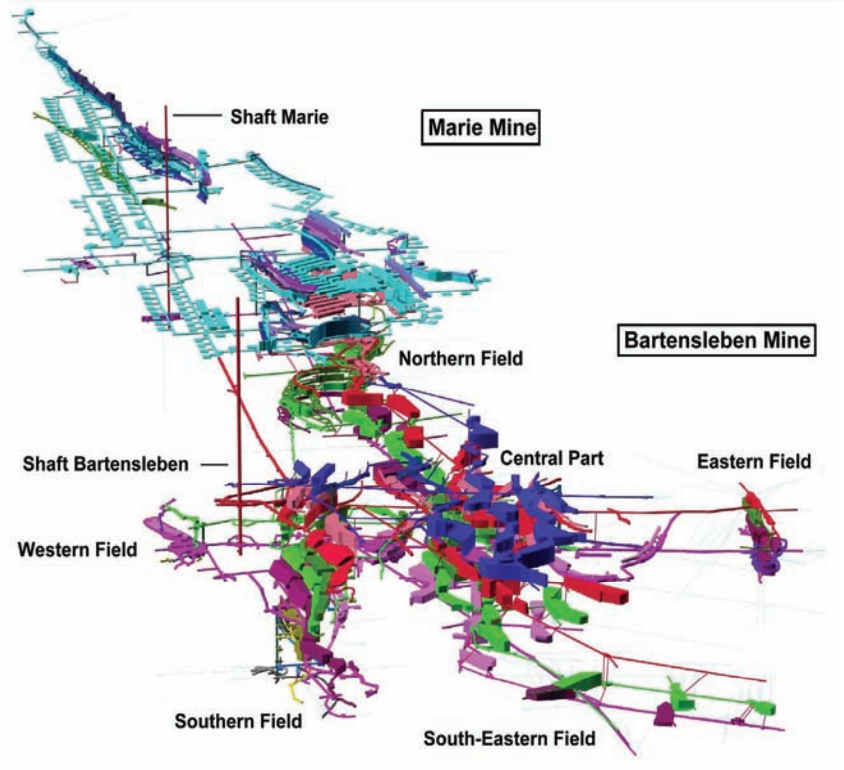
The Morsleben repository (ERAM = Endlager für radioactive Abfälle Morsleben) was licensed, constructed and operated in former German Democratic Republic (GDR). Radioactive waste management work in GDR was initiated in the 1960s. Like Western Germany GDR radioactive waste disposal policy has been based on the decision that radioactive waste is to be disposed of in deep geological formations.

For in GDR spent fuel was taken back by the Soviet Union for reprocessing, waste management had to deal with LLW and ILW radioactive waste originating from the operation of nuclear power plants and from the production and application of radioisotopes.

The favoured option for a repository in the GDR was the construction and operation of a central disposal facility in an abandoned mine. In 1970, the abandoned Bartensleben salt mine situated in the Aller valley was selected from ten salt mines considered as potential repositories.

Mining operation in the Aller valley started in 1897 with the sinking of shaft Marie. Shaft Bartensleben was sunk in 1912. Both shafts are connected by two drifts for ventilation and escape route purposes. Until 1969 the twin mine had produced potash and rock salt by room and pillar mining. Mining was performed down to a depth of about 500 m. The resulting mine openings have a maximum length of about 150 m and a maximum width and height of about 35 m, respectively. The ERAM twin mine has an overall length of 5.6 km and an overall width of 1.7 km (Figure 1). In total, a volume of about 8.7 mio. m^3 has been excavated. During mine operation about 3 mio. m^3 have been backfilled with crushed salt. The present void volume amounts to 5.5 mio. m^3.

The Morsleben repository was licensed in former GDR in a stepwise licensing procedure lasting several years, which comprises the site licensing, the license for the construction, test operation and the unlimited operation. The last step would have been the



◄ Fig. 1: Underground facilities of the ERAM twin mine

license for the closure of the facility. In 1990 due to German reunification, the Morsleben repository became a federal repository and the license was limited up to the year 2000. BfS became responsible for the operation. According to German Atomic law a plan approval procedure with an environmental impact assessment and thus an involvement of the public is necessary for further operation as well as for the closure of the ERAM. In 1992 BfS applied for of a plan approval procedure for the operation of the ERAM beyond the year 2000 and in 1997 this application was limited to the closure of the ERAM.

In 2005 BfS filed the plan “Closure of the ERAM” and the documents required for the involvement of the public to the licensing authority, the Ministry for Environment of the federal state of Sachsen-Anhalt. In total about 250 licensing documents regarding the geoscientific site investigation, the feasibility of the closure concept and the long term safety assessment have been filed to the licensing authority.

From 1971 to 1998 radioactive waste from the operation of nuclear power plants, the decommissioning of nuclear facilities, the nuclear industry, research institutions, collecting facilities of the federal states or directly from small waste producers, and other users of radioactive materials has been disposed of in the ERAM.

Waste disposal has been performed on the 4^{th} level in a depth of about 500 m below surface using existing mine openings, resulting from former mining activities. Solid low level waste, packaged in drums and cylindrical concrete containers was stacked in emplacement rooms of the western field and the eastern field. Stacking was accomplished by a forklift. Low and medium level solid radioactive waste and spent sealed radiation sources were dumped into non-accessible cavities in the southern field by means of remote controlled equipment. From 1981 to 1990 also liquid low level radioactive waste was in-situ solidified in the emplacement rooms in the southern field by using lignite filter ash as binding agent.

In addition to the radioactive waste disposed of, sealed cobalt and caesium radiation sources and small quantities of solid europium waste in special steel containers and one 280 litre drum containing radium-226 waste are intermediately stored in boreholes at the ERAM facility. Within the scope of the licensing procedure for the closure of the ERAM, an application was submitted to final disposal of this intermediately stored waste.

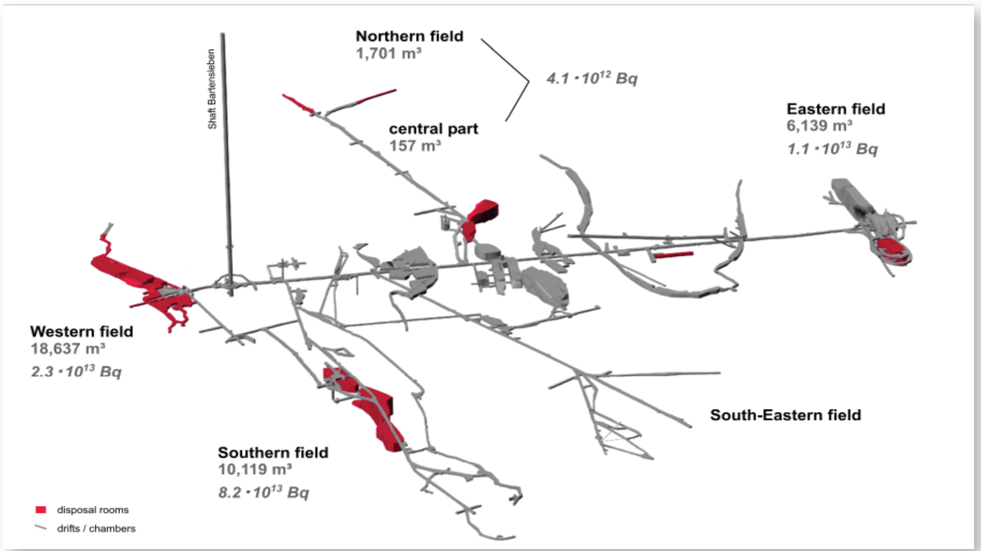


Fig. 2: ERAM emplacement fields on the 4th level (activity as of 2005)

In total, nearly 37,000 m³ of radioactive waste and about 6,600 sealed radiation sources were emplaced in the repository (see also Figure 2). The corresponding activity amounts to in total approximately 5.2 x 10^{14} Bq (activity related to June 2005). The activity of alpha-emitters amounts in total to 7.5 x 10^{11} Bq. The most significant radionuclides are cobalt-60 and caesium-137.

The main characteristics regarding the ERAM closure concept are:

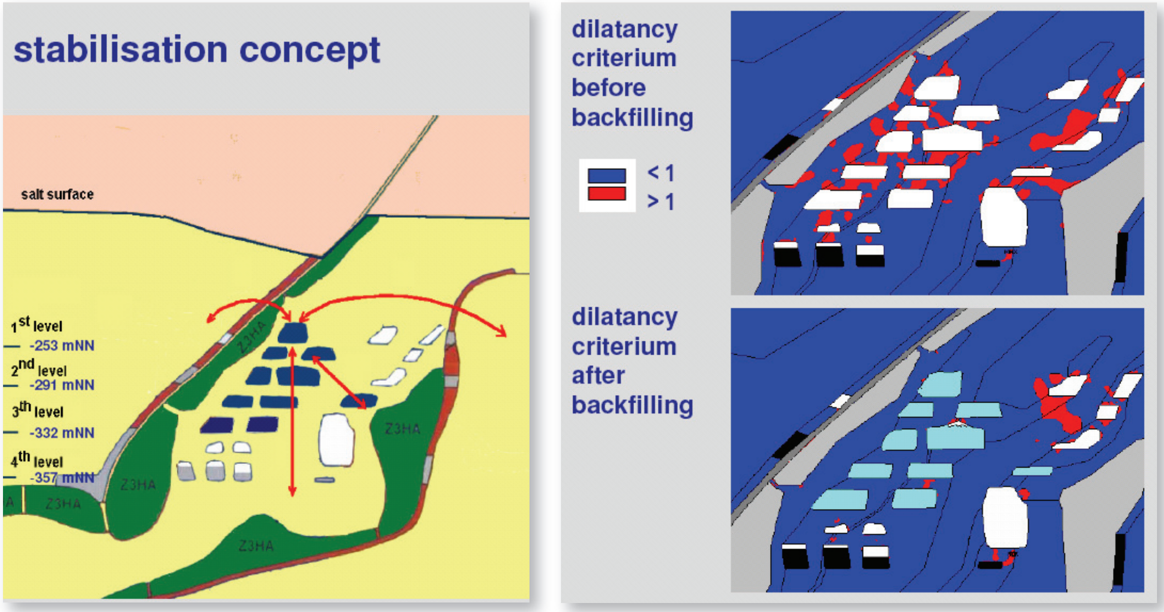
- high excavation rate
- old age of the mine openings (up to 100 years)
- large total void volume of the underground openings compared with the volume of the emplaced radioactive waste
- very low convergence rates
- existence of potash seams, partly excavated and backfilled

In particular the central part of the Morsleben repository is mined out to a large extent with a void volume of about 2.8 mio. m³. Due to the long period of time since excavation (50-75 years) as well as the short distance to the salt table the central part appears to be mechanically highly stressed.

Numerical calculations, carried out starting from the primary state of stress and covering the excavation phase showed, that the risk of a progressive failure in the central part of the Morsleben repository could not be excluded. These results were confirmed by the results of convergence measurements, by the geotechnical surveillance and in-situ observations. In 2001, a significant roof fall occurred in a restricted chamber on the second level. In Figure 3 the red colour marks areas of stressed or disturbed rock salt according to the calculations. In the case of a successive failure, the realisation of the planned closure concept could be endangered.

To improve stability and to guarantee a regular plan-approval procedure for the closure, stabilisation measures are necessary. The planning work had been started in 1997. The idea of the concept is to stabilize the central part of the mine by backfilling selected chambers with a salt concrete. The backfilling of these chambers and the surrounding rock salt act together as a system of arches and pillars supporting the overburden and minimizing deformations of the main anhydrite.

The backfill process was simulated by numerical calculations and further calculations covering a period of 1000 years after backfilling, which show a significant improvement of the geomechanical situation in the central part of the ERAM (Figure 3). Geotechnical surveillance measures give first but strong hints of the success of the stabilisations measures.



▲ Fig. 3: Concept for the stabilisation of the central part Bartensleben and stress regimes (dilatancy boundary 1.0) before and after backfilling of mine excavations in the central part. The grey areas mark the occurrence of main anhydrite and the red areas mark a violation of the dilatancy criterion of rock salt. Backfilled openings are marked light blue (backfilled with salt concrete) or black (backfilled with crushed salts).

The stabilization of the central part started in 2003 with the necessary mining activities for the infrastructure and by backfilling the first chamber. Salt concrete is used as backfilling material, consisting of crushed salt, cement and fly ash as supplementary constituents. The salt concrete is produced in a production unit near to the ERAM site and transported by a stationary pumping unit via pipeline. The underground pipeline system covers about 400 m vertical and up to 1200 m horizontal distance.

Up to November 2007 15 of 24 chambers have been backfilled. With regard to the 270,000 m³ of total volume of the chambers to be backfilled and assuming a daily backfilling rate of 500 m³ of salt concrete, the stabilization of the central part of the Morsleben repository will be accomplished by 2010.

The closure concept of the ERAM, sketched in Figure 4, is based on an extensive backfilling of the salt mine with a salt concrete mixture, the sealing of the disposal areas by drift sealings and shaft sealings for the closure of the repository.

The disposal areas, that means the mine workings used for the disposal of radioactive waste and their wider surroundings will be hydraulically isolated from the rest of the mine workings by sealing the drifts with salt concrete. The measures within the closure concept aim at stabilizing the mine works and to isolate the radioactive wastes in such a way that the protection targets are fully met.

The emplacement areas will be closed by drift sealings. A complete drift sealing consists of several sealing segments one behind the other, constructed with salt concrete. The required permeability is equal to or less than 10^{-18} m². The function of such a drift sealing is to impede in the long term the access of solutions into the disposal areas and the spread of radionuclides from the disposal areas.

The entire mine openings shall be backfilled as far as possible with salt concrete in order to reduce the mine openings accessible to solutions, to stabilize geomechanically the closed repository system including the implemented technical barriers and to minimize extraction processes at soluble layers of potash salt by water access. The remaining openings will store gas volumes and prevent undue extreme high gas pressures due to the production of gases within the repository by processes like corrosion in the after closure phase.

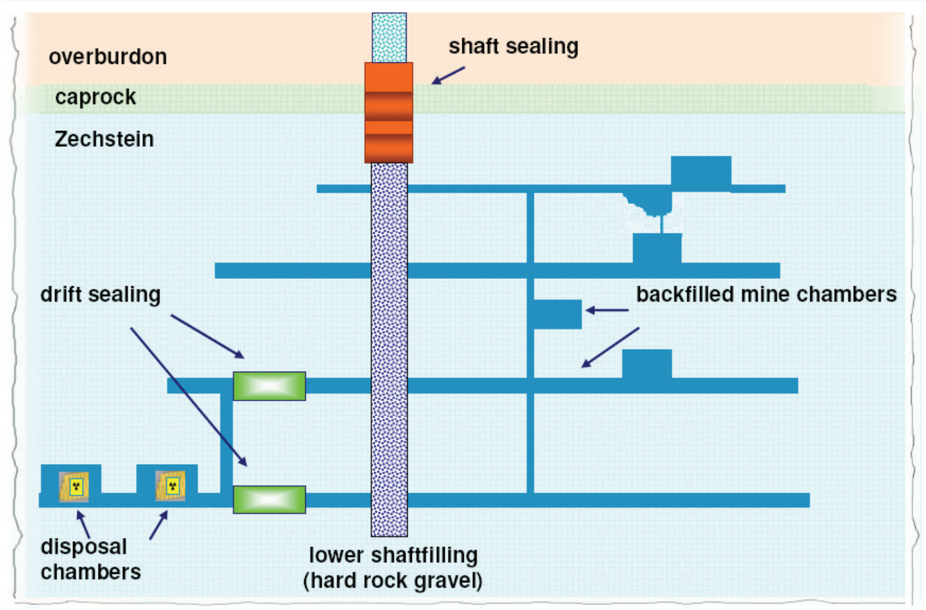


Fig. 4: Essential elements of the ERAM closure concept

Both shafts of the ERAM will be closed by systems of sealing elements of various materials with low permeability in order to minimize the inflow of groundwater from the overlying strata into the mine and to minimize the discharge of radionuclides in solution from the mine into the overlying strata.

The safety and performance assessments for the selected closure concept were carried out by two teams with different models and computer codes for the same closure concept and the same basic parameters but independent conceptual approaches: The used codes are EMOS and PROSA. Both groups have performed deterministic and probabilistic dose calculations and investigated the same suite of scenarios. The estimated normal evolution of the repository in the post-operational phase with no disruptive event and no significant brine intrusion into the sealed disposal areas of the repository doesn't lead to any release of radionuclides. In the reference scenario it is assumed that brine intrudes into the mine and acts as a transport medium for radionuclides. Within this scenario a reference case has been defined based on likely or conservative assumptions regarding a release of radionuclides. The result of the reference case calculations is that the radiological protection objective is clearly achieved and that additional safety indicators show large safety margins.

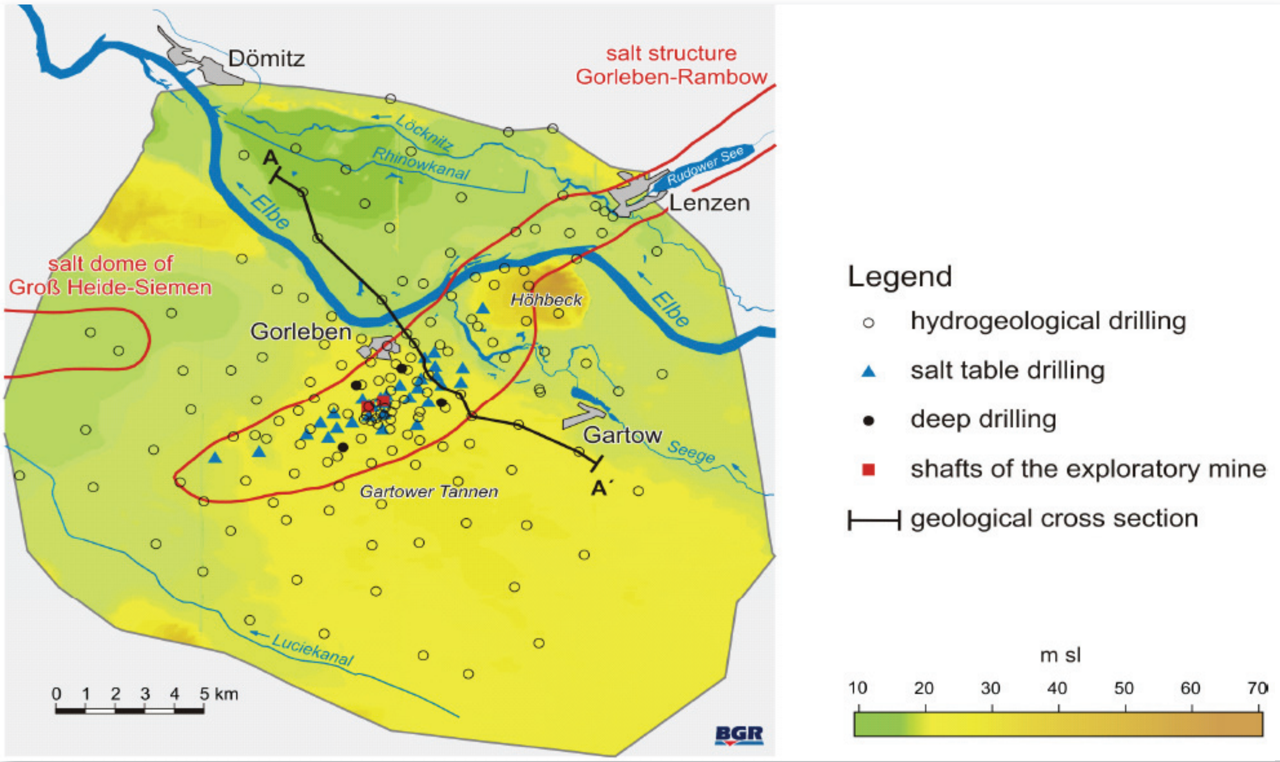
Present the documents for the closure of the ERAM including the closure concept and the safety assessment are examined by experts on behalf of the licensing authority. Findings are intensively discussed, and as a result of these discussions some of these documents have to be supplemented or to be reviewed. The initiation of the public involvement is to be expected by 2008/2009.

2 Gorleben Exploration Mine

The Gorleben salt dome in the north-east of the federal state Niedersachsen was being investigated for its suitability to host a repository for all types of radioactive waste. Site decision had been taken in 1977 on the basis of a proposal of the Government of the federal state of Niedersachsen for the Gorleben site.

Objectives for the exploration were:

- proof of sufficient volume and suitable homogenous salt rock sections to excavate the required emplacement rooms
- proof of the safety of the repository against brine influx in the operational and the post-operational phase



▲ Fig. 5: Geological and hydrogeological investigation area

- proof of the stability of the repository for the post-operational phase especially regarding the impact of the planned emplacement of heat generating waste.

In 1979 a comprehensive site-exploratory programme has been started in Gorleben to investigate the salt dome as well as the cover deposits and the surrounding rocks in the frame of a comprehensive geological and hydrogeological site exploration programme.

Due to the border to the former GDR the investigations were restricted to the 300-km²-wide area in Niedersachsen in the south and the west of the Elbe river (see Figure 5). From 1996 to 1998 comparable investigations have been carried out in the 175-km²-wide area of Mecklenburg-Vorpommern and Brandenburg.

In the period until 1985 the investigations were at first carried out in an area whose northern border was marked by the river Elbe.

Exploration works in the salt dome at the site of Gorleben commenced in 1986. After the shafts and the required infrastructure areas had been developed, the largest part of the exploration area 1 of a planned five exploration areas was opened. The planned emplacement area is situated at a depth of 880 m. Up to the present 7.1 km of drifts, more than 11,000 m of boreholes and approximately 230,000 m³ of rock salt has been excavated underground.

Significant results for the first exploration area are:

- Complicated geological structures and intensively folded strata in the infrastructure area nearby the shafts.

- No continuous Main Anhydrite (divided in blocks), which is important regarding the long term safety assessment.

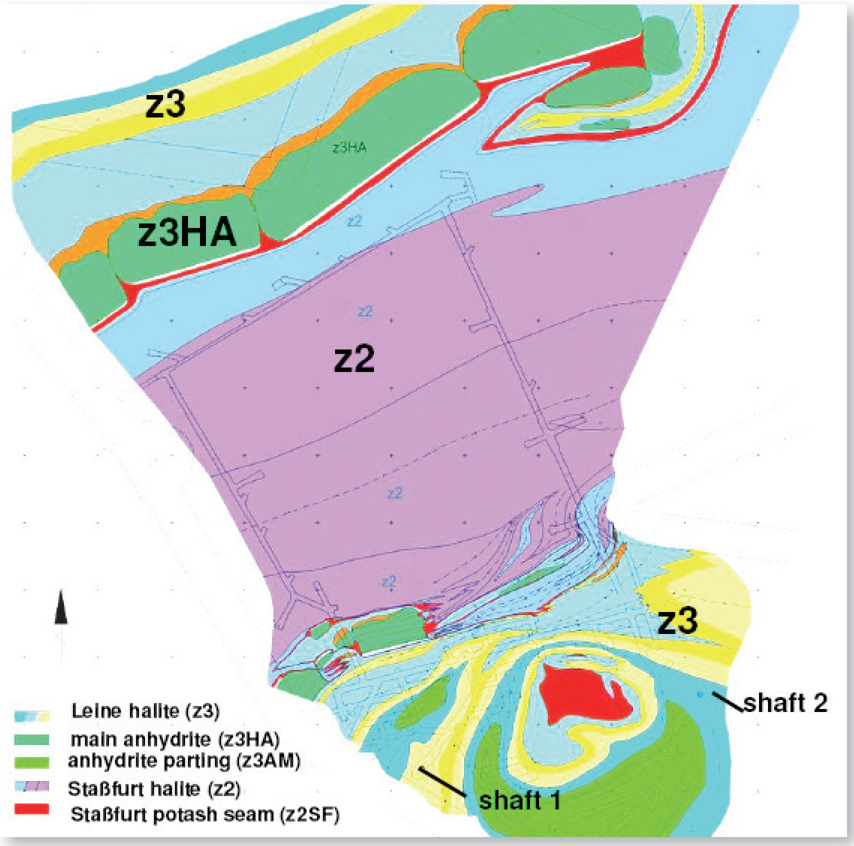


Fig. 6: 840 m Level of the Gorleben exploration mine

- Exploration area 1 features large, homogeneous salt rock areas planned to host the emplacement areas for the disposal of high level radioactive waste.
- Encountered solutions are residuals of altered former Zechstein sea water.

The current status of the Gorleben investigation programme is:

- Surface investigation programme is completed. The results are published by BGR in 2007.
- Subsurface exploration is nearly completed for the first of five planned exploration areas.
- Exploration work has been stopped due to the Gorleben moratorium and the results for the exploration area 1 are documented (as of 2000).

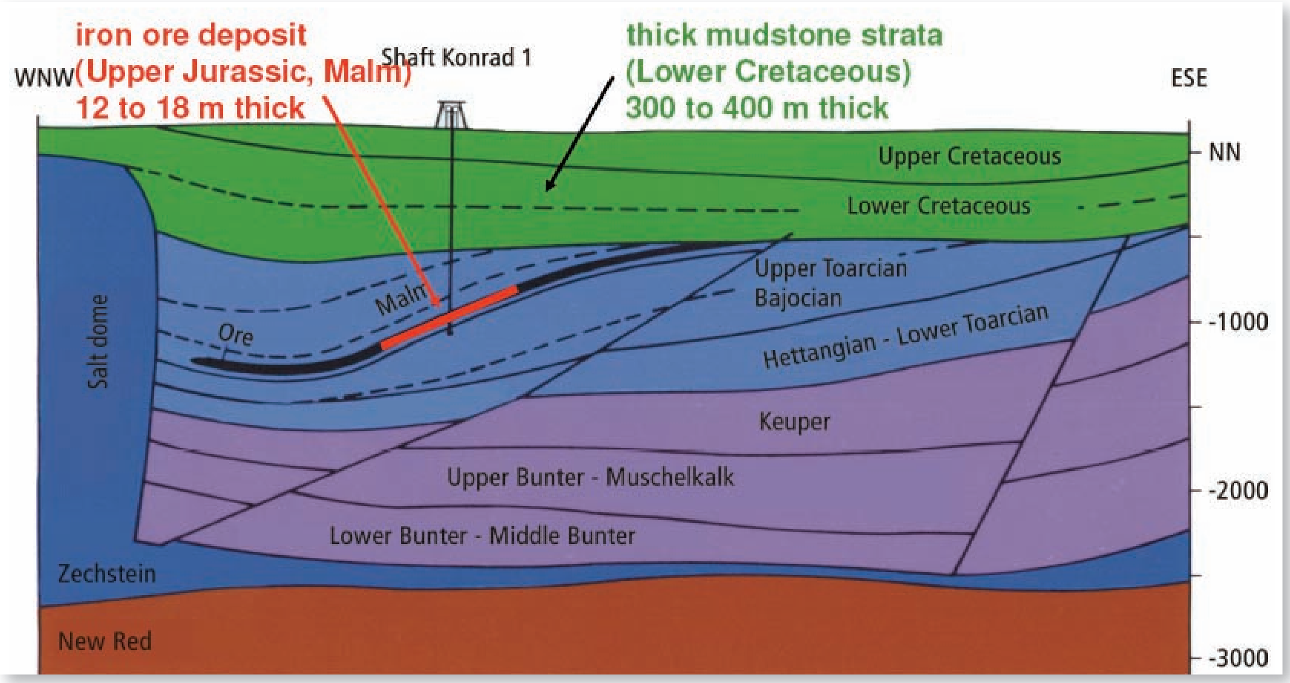
3 Konrad Repository

The Konrad mine is a former iron ore mine located in the south-eastern part of Niedersachsen within the limits of Salzgitter.

The two shafts have been sunk from 1957 to 1962. Until 1976 the iron ore deposit, formed about 150 million years ago, has been mined in a depth between 800 m and 1200 m. Ore mining has been abandoned because of economical reasons and because of the inappropriate content of phosphate in the ore.

The history of the Konrad repository began in 1975 with first preliminary site investigations for hosting a repository for low and intermediate level radioactive waste with negligible heat generation. The impetus for the reuse of the Konrad mine came from the mine workers committee, for the workers were worried about their jobs facing the threatening shut down of the Konrad iron ore mine.

As the geological profile shows, this ore deposit does not reach the surface at any point and is exposed by the Konrad mine in a depth of between about 800 m and 1,300 m (see Figure 7). The iron ore deposit, planned to be used as the host rock for the waste disposal is about 12 m to 18 m thick and covered by thick mudstone strata i.e. 300 m to 400 m of Lower Cretaceous marl and clay stone performing the geological barrier.



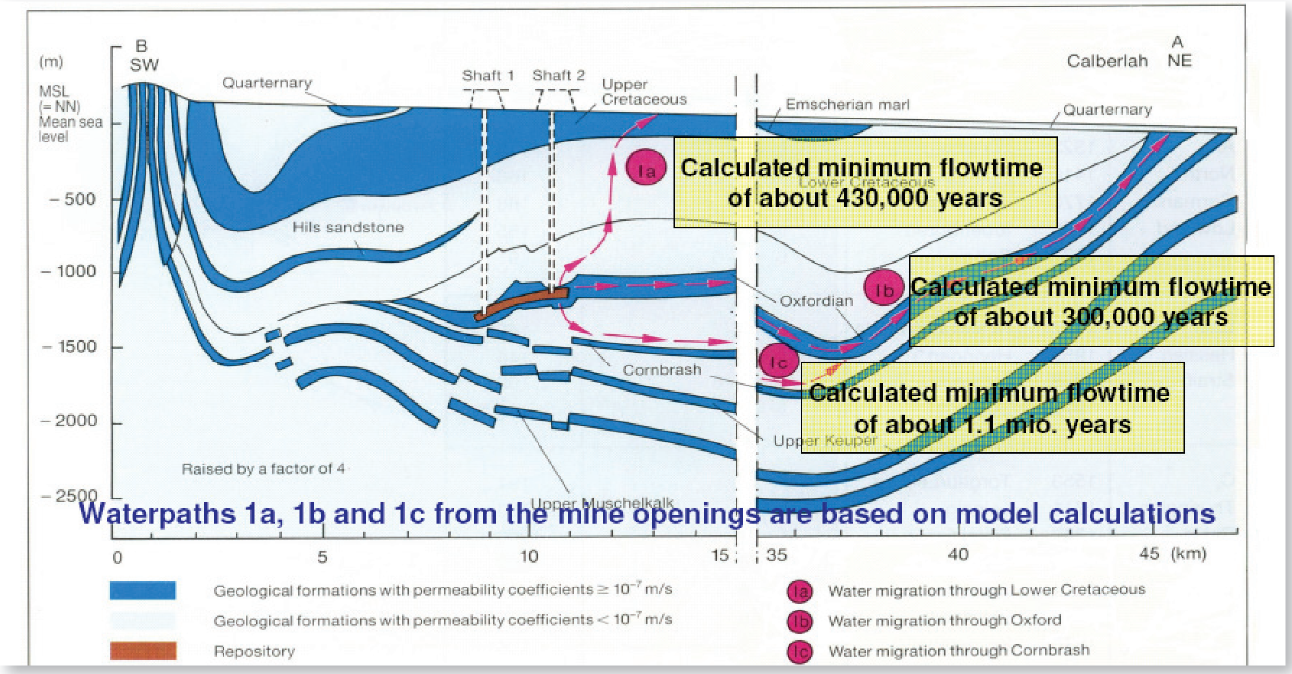
▲ Fig 7: Cross section of the southern part of the "Gifhorner Trog"

Main features of the Konrad mine favourable for the re-use as a repository for radioactive waste are the depth of the iron ore deposit, the planned disposal horizon, the extreme low permeability of the overburden (thick mudstone strata) and the geomechanical stability of the iron ore horizon.

The hydrogeological situation is characterized by an alternate bedding of more or less permeable and impermeable strata. The system is bordered from above by Lower Cretaceous clay and from below by salt layers of the Middle Muschelkalk; the lateral hydraulic borders are formed by salt domes. In the post-operational phase the remaining voids of the repository will gradually fill up with deep groundwater within at least 2000 years. With regard to potential transport of radionuclides out of the repository area into the biosphere ground water movements have been calculated on the basis of freshwater simulation models. The model area extends for more than 40 km from Salzgitter in the south to Gifhorn in the north. Parameter studies were used to calculate the movement of deep groundwater and the water paths leading from the repository to the biosphere.

Figure 8 shows the "Lower Cretaceous Pathway" (1a), the "Oxford Pathway" (1b) and the "Cornbrash Pathway" (1c) and the appropriate flow times for each pathway.

In 1982 PTB, the predecessor of the BfS applied for a license to convert the Konrad mine into a repository. In 1990 the final license application with a reviewed "Plan Konrad" was filed to the licensing authority, the Ministry for Environment of the Federal



▲ Fig. 8: Cross section through the Konrad fresh water model area

State of Niedersachsen. In 1992/1993 the longest in German history, 75 days lasting public hearing, took place in the case of the plan approval procedure for the Konrad repository, dealing with about 290.000 objections from all over Germany. The plan-approval procedure was concluded in 2002 with the plan-approval decision permitting the disposal of 303,000 m³ of radioactive waste with negligible heat generation. After in total about five years lasting legal procedures the Konrad licence becomes definitely valid in 2007. As a legal and unappealable plan-approval decision for the Konrad repository is now available, the Konrad mine will be converted to a repository. This conversion affects the surface facilities of shaft Konrad 1 (Figure 9) and shaft Konrad 2 (Figure 10) as well as the subsurface facilities. Large parts of the installations at the site existing today will be

▲ Fig. 9: Planned surface facilities at shaft Konrad 1 for the future functions as the downcast ventilation shaft, the man-riding shaft and the transportation shaft for debris and material.

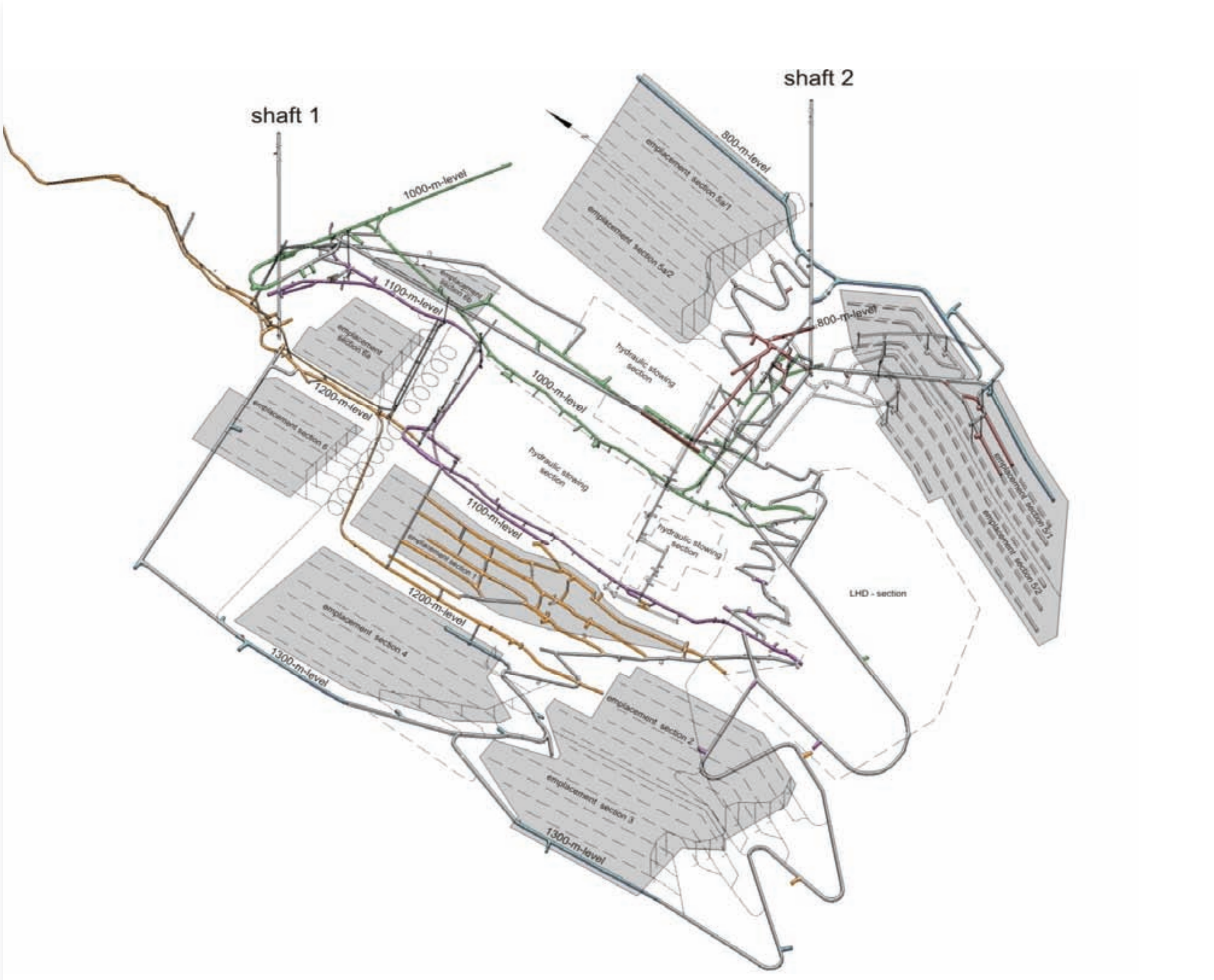
▲ Fig. 10: Planned surface facilities at shaft Konrad 2 for the future functions as the upcast ventilation shaft and the transportation shaft for the radioactive waste to be disposed of.

dismantled and new buildings necessary for repository operation will be erected. The shafts must be rebuilt and redeveloped; the listed pit frame of Konrad 1 must be adapted to the applicable requirements according to Mining Law. In the mine itself the cavities will be set up where the radioactive waste will be emplaced. Currently there are only the galleries and cavities worked during former commercial ore mining and during the exploration of the mine.

Figure 11 shows the mine openings with exploratory drifts (800-m-level, 1,300-m-level) and the drifts and ramps constructed for operational purposes (e.g. South Ramp, North ramp, East ramp). Grey marked are the planned emplacement fields. As the Konrad licence restricts the emplaceable waste volume up to 303.000 m^3 the emplacement fields adjacent to shaft 2 provide sufficient emplacement room.

An overview of the planned and licensed emplacement operation is given in Figure 12.

The waste packages are delivered on wagons or lorries. In separate package reloading areas in the reloading hall, the transport units will be reloaded onto shaft conveyance loading cars using bridge cranes. A rail system will transport each loading car to the intake control and, following release through the buffer tunnel to the shaft. At the pit bottom the waste packages will be transferred on an underground transporter by a rail-mounted straddle carrier and then transported to the emplacement room and there it will be stacked by a fork lift. The emplacement capacity amounts up to 17 transport units per day and about 4000 transport units per year in one-shift operation. A transport unit consist of a container or a pool pallet with up to two cylindrical waste packages.

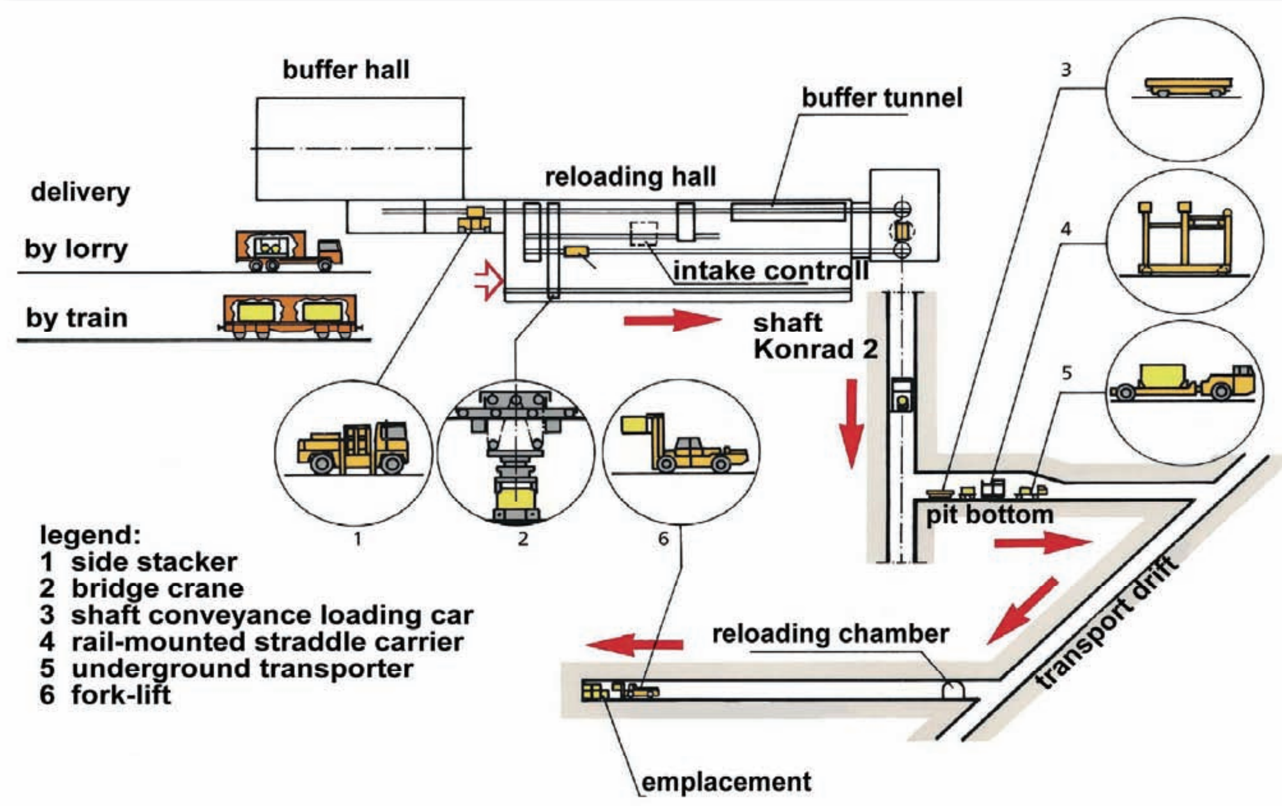


▲ Fig. 11: Mine layout and planned emplacement fields

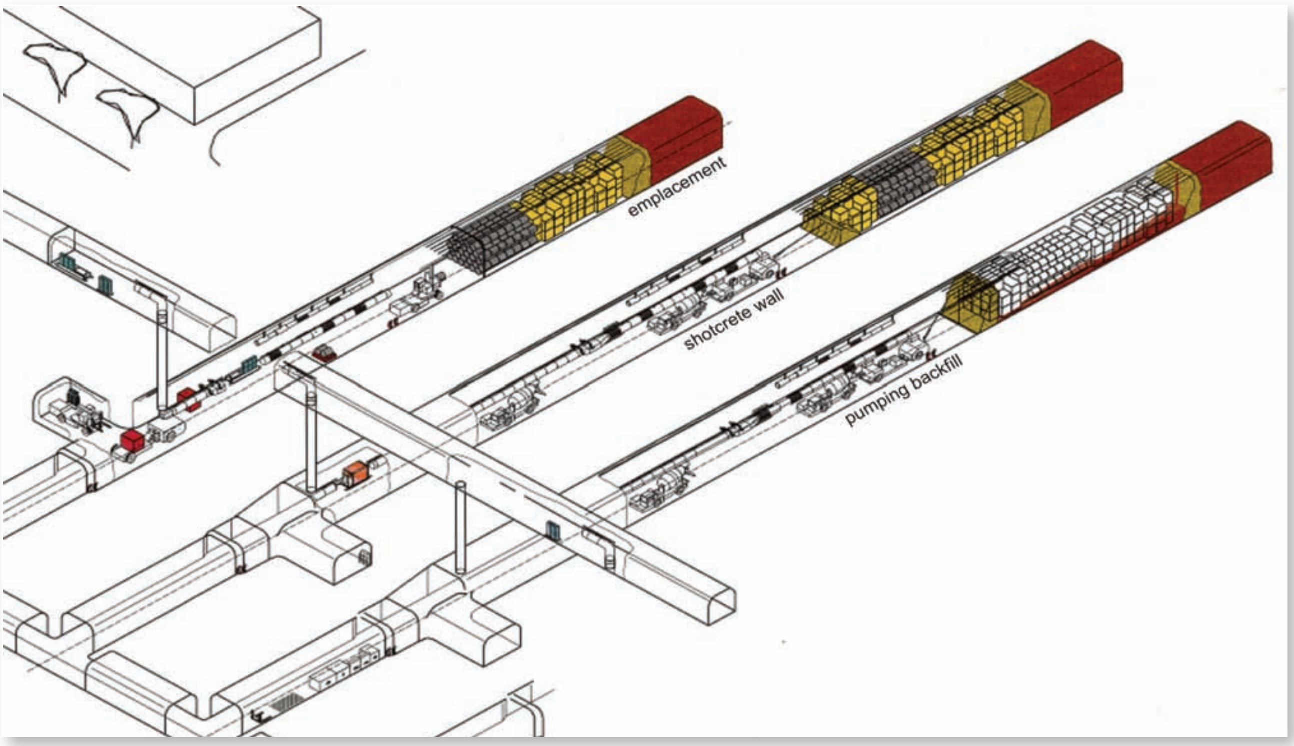
Figure 13 gives a view in a Konrad emplacement field and shows the planned emplacement of box-type containers and cylindrical waste packages in an emplacement room by fork lift, the closure of an emplacement section in an emplacement room by a shotcrete wall and the backfilling of the remaining voids in an emplacement section with a pumpable mortar. Filled emplacement rooms are closed and backfilled to minimize exposure of operating personnel to released gaseous radionuclides.

The plan-approval (licensing) procedure according to nuclear law that had been initiated in 1982 was concluded on June 5, 2002 with the plan-approval decision permitting the disposal of 303,000 m^3 of radioactive waste with negligible heat generation, which is about 90 % of the total amount of waste arising in the Federal Republic of Germany. Up to present there are 88,515 m^3 conditioned radioactive waste with negligible heat generation stored in interim storage facilities in order to be disposed of in the Konrad repository. It is currently estimated that approximately 277,000 m^3 of radioactive waste with negligible heat generation will have been produced by 2040.

With the decisions announced by the Federal Administrative Court on April 3, 2007, with which the complaints of the city of Salzgitter, the communities of Lengede and Vechelde, and of a farmer from Salzgitter against the non-admission of revision in the judgments of the Lüneburg Superior Administrative Court against the plan-approval of the Konrad mine were dismissed, the judgments of the Lüneburg Superior Administrative Court of March 8, 2006 have become final. Thus the plan-approval



▲ Fig. 12: Scheme of planned emplacement procedure



▲ Fig. 13: Planned Konrad emplacement rooms and the planned emplacement and backfilling procedure decision has also become final and executable. As a legal and unappealable plan-approval decision for the Konrad repository is now available, BfS is going to set up the infrastructure required for the conversion. This preparation (planning) phase will last about two years. The actual conversion of the Konrad mine into a repository will take about four years. Thus, after a period of about six years for the preparation and conversion of the Konrad mine into a repository, the emplacement of radioactive waste in the Konrad repository could start in the year 2013. ■

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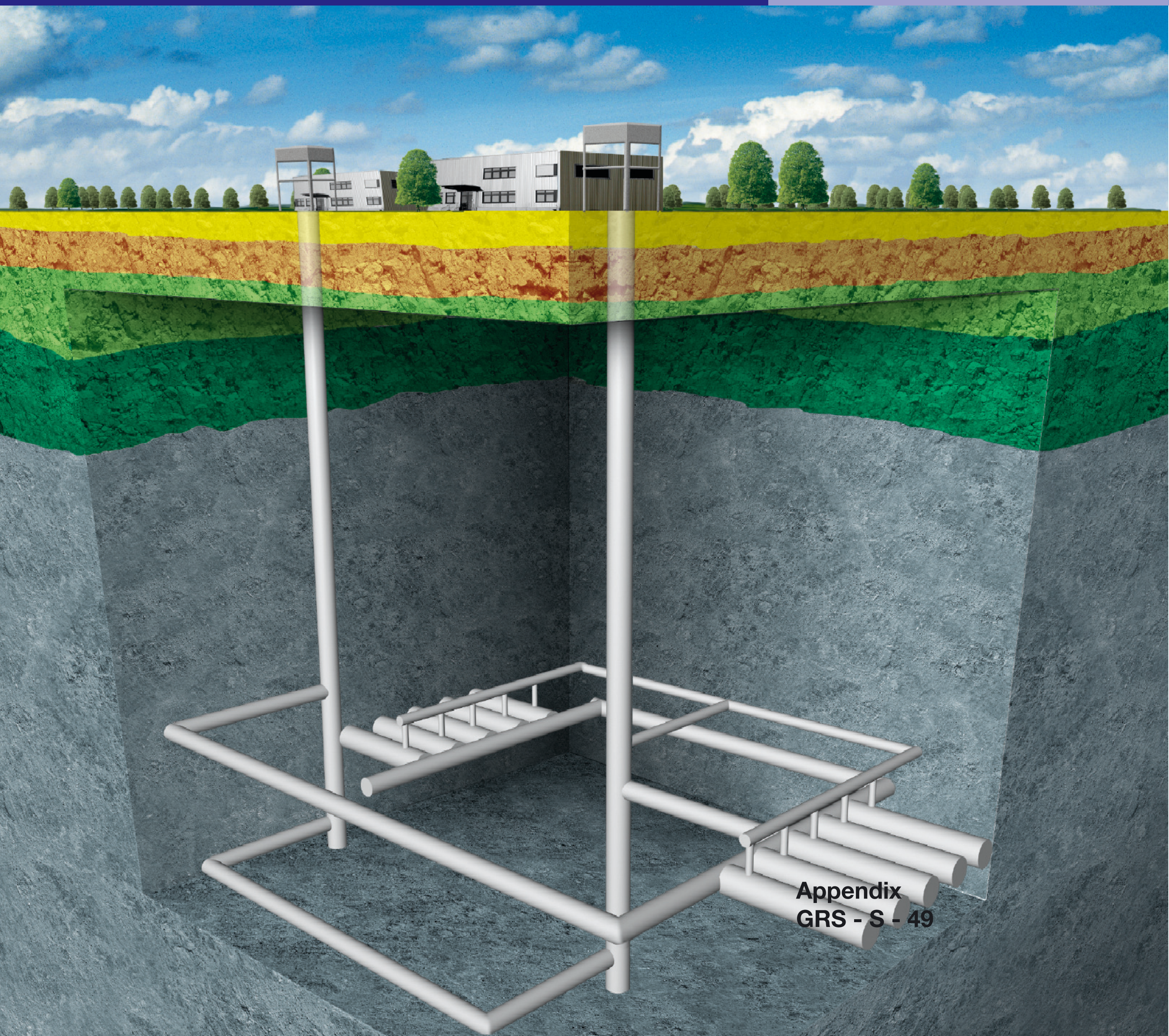

Appendix
GRS - S - 49

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Appendix
GRS - S - 49
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Contents

A Realistic Approach for Assessing the Long-Term Release of C 14 from a Closed Final Repository for Low-Level Radioactive Waste in a Salt Mine

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Abstract
The contribution of C-14 to radiation exposure in the biosphere can be significant as compared to that of other radionuclides disposed in a repository for low-level radioactive waste. The release pathways of C-14 and processes relevant to its release from a closed final repository for low-level radioactive waste are discussed. Because a conservative approach may lead to undue overestimation of the potential radiation exposure, a more realistic approach is outlined. At the present level of refinement, our realistic approach provides a sufficient safety margin to German federal limits for radiation exposure to demonstrate compliance with the ALARA (as low as reasonably achievable) principle and can thus facilitates licence approval.

1 Introduction

The proof of long-term safety of a final repository for low-level radioactive waste requires an assessment of the potential radiation exposure from gaseous radionuclides. C-14 is the only radionuclide that can contribute significantly to radiation exposure via gas pathway as all other gaseous radionuclides can be neglected due to their short half-lives, low inventories or low radiological relevance.

Sources and environmental behaviour of C-14 have been summarized previously [7].

Hitherto only conservative approaches have been used for estimating the mobilisation, migration and release of C-14 via gas and groundwater pathways from a repository. The results of these approaches generally comply with German federal limits for radiation exposure.

Despite compliance with the limits for radiation exposure, these approaches may provide insufficient safety margins to demonstrate compliance with the ALARA (as low as reasonably achievable) principle. Overestimation of the potential radiation exposure may therefore lead to false conclusions and ineffective measures for minimising radiation exposure.

The following paper outlines a realistic approach for an estimation of long-term radiation exposure by C-14. The approach is based on a model final repository for low-level radioactive wastes in a salt mine. A realistic approach provides an additional safety margin to radiation exposure limits and can thus facilitate licence approval where compliance with the ALARA principle has to be demonstrated.

2 General Background

2.1 Scenario

The scenario for a model final repository in salt rock includes a hydraulic connection between the mine building, the overlying rock and the biosphere. Driven by rock convergence, gas and brine will be squeezed out of the repository and penetrate into the overlying rock. This process takes place on a geological time scale. There is no (or only a negligible) leakage of gas via the mine shafts. Radionuclides are retained in the near-field by limited dissolution and sorption.

The detailed scenario includes further features, events and processes such as:

- retarded exchange of fluids between the emplacement chambers due to engineered barriers
- time dependence of geochemistry in the emplacement chambers

- mobilisation of radionuclides
- retardation of radionuclides by sorption and precipitation
- gas generation and accumulation in emplacement chambers
- squeezing-out of fluids due to convergence and gas generation
- transport of radionuclides by density-driven convection of fluids
- release of gaseous radionuclides
- migration of gas into the overlying rock
- preferential fluid transport along fracture zones in the overlying rock
- transport retardation and dynamic storage of contaminated fluid
- dilution of the fluid by ground and surface water with lower salt content.

2.2 Geochemistry

The geochemical environment (i.e. the concentration of major and minor constituents of the brine, the prevailing solids, and the gases of an emplacement chamber) results from the interaction of the brine with the waste matrix materials (cement/concrete), the waste packages (concrete, iron, corrosion products), the backfill materials and the degradation products of organic matter of the waste itself. The conversion of organic matter to CO_2 and CH_4, the degradation of cement and the corrosion of metals are time-dependent processes that determine the composition of the brine. Although precise degradation rates of cement and conversion rates of organic matter to CO_2 under the potential repository conditions can not be known in sufficient detail, CO_2 will predominantly precipitate as carbonate. As a result, a significant change of the initial pH dominated by cement in the waste matrix, in the waste container and in the backfill of emplacement chambers is not expected to occur.

Equilibrium constants are known for alkali and alkaline-earth carbonates, carbonate complexes and mixed solid phases. A nearly complete data set of Pitzer constants is available. This data is used for the geochemical modelling of brines. The main carbonate compounds dissolved in brine are $MgCO_3$(aq), $CaCO_3$(aq) and CO_3^{2-}/HCO_3^- . Precipitation of solid phases such as magnesite, calcite and dolomite limit the concentration of dissolved inorganic carbon.

The typical range of concentrations of carbon expected in brine in emplacement chambers, assuming a closed system, is shown in Table 1. The concentration that would be obtained by dissolution of the complete inventory is shown for comparison. The dissolved fraction of inorganic carbon represents only a small amount of the total carbon inventory and varies. The major amount of carbonates is fixed in the solid phase.

Table 1: Typical concentration range of dissolved carbon and carbonates in brine in the emplacement chambers

Speciation	Concentration (10^{-5} mol/(kg H_2O))
C (inventory, hypothetically dissolved)	200 000 – 3 000 000
C (total dissolved inorganic C)	0.8 – 40
CO_3^{2-}	0.04 – 1
HCO_3^-	0.000001 – 3

Isotopic effects on chemical reaction rates are ignored, i.e. compounds containing radioactive C-14 are assumed to have identical reaction rates for equilibration, precipitation, sorption and gas generation as compounds having non radioactive C.

Isotopic dilution occurs via exchange processes. The isotopic exchange of C-14 in carbonates and bicarbonates takes place rapidly in brine via hydrogen exchange.

$$^{14}CO_3^{2-} + HCO_3^- \leftrightarrow CO_3^{2-} + H^{14}CO_3^-$$

Isotopic equilibrium between the gas phase and the fluid phase is also rapidly achieved.

$$^{14}CO_2\,(g) + HCO_3^- \leftrightarrow CO_2\,(g) + H^{14}CO_3^-$$

The isotopic equilibration of solid phases and brines may be slow, as the kinetics of dissolution and precipitation are controlled by accessible surfaces. The isotopic equilibration of other compounds such as hydrocarbons, fatty acids, or alcohols is controlled kinetically and may be slow, too.

2.3 Inventory of C-14

The inventories of C-14 in emplacement chambers are fixed on completion of the waste emplacement. Additional C-14 does not build up in low-level radioactive waste (e.g. via decay chains or nuclear reactions). The inventory is estimated to cover all uncertainties. Care is thereby taken not to produce an overly conservative estimate.

The inventory of C-14 and its concentration in the emplacement chamber may vary within several orders of magnitude between different emplacement chambers. Table 2 depicts a range of inventories and concentrations of C-14 of a repository in salt rock. The C-14 inventory and concentration vary typically by a factor of 10 at most. However, the inventories and concentrations in some emplacement chambers are outside of this range. To assign the maximum concentration to all emplacement chambers (as some models do) would be overly conservative.

Table 2: Inventory and concentration of C-14 in emplacement chambers

Emplacement chamber	C-14 inventory (10^{11} Bq)	C-14 concentration (10^{8} Bq/Mg)
lowest	0,02	0,05
typical range	0,5 – 5	1-10
highest	10	35

The typical isotopic ratio of C-14 to total carbon in the low-level waste equals 10^{-8}-10^{-9}. To a first approximation, C-14 can be assumed to be homogeneously distributed with a similar isotopic ratio in all carbon species in the brine due to isotopic exchange.

2.4 Speciation of C-14

The waste can be characterised in terms of organic / inorganic / metallic and non-metallic fractions as these waste fractions are documented. The organic fraction typically accounts for 10-30 % of the waste. However, the assignment of the C-14 inventory to each of these fractions is unavailable from the waste documentation. Therefore, a working assumption has had to be made.

It is generally assumed that the C-14 inventory is bound to organic matter. This assumption ignores that C-14 can be bound to solid phases e.g. as carbonates in concrete or carbides in ashes or activated impurities in metals and on metal surfaces. If C-14 were homogeneously distributed in the waste, a significant amount (70-90 %) would not be contained in organic matter.

The speciation of organic C-14 may differ from that of the organic fraction, but will underlie similar degradation processes. As organic matter does not completely convert under saline conditions [2], a large fraction of C-14 will thus remain within the deposited waste and will undergo radioactive decay. This means that the release of C-14 will be significantly reduced compared with the predictions of conservative models that assume a complete degradation. For a better quantitative assessment the chemical speciation of C-14 should be known in detail.

In summary, the assumption that the C-14 inventory is bound to organic matter rather than homogeneously distributed in the waste is conservative.

3 Release of C-14 from the Waste

The following time-dependent conditions govern the release of C-14 from the deposited waste:

- waste and waste packages have been exposed to air since emplacement (initial condition)
- oxidising conditions favour degradation of organic constituents (transient condition)
- waste packages and waste products are exposed to anaerobic, cement-conditioned brine (prevalent long-term condition)

The release of C-14 under the initial and the transient conditions is minor as compared to that under the long-term condition. Thus the emphasis of further considerations will be on the latter.

The release of C-14 from deposited radioactive waste depends on its chemical speciation. The important processes are:

- Degradation, dissolution or desorption of organic compounds; Under anaerobic conditions microbial degradation is the major release mechanism of C-14; However, according to some studies [2], the microbial yield will be limited; The degradation of organic matter produces intermediates; These intermediates are comparable to substances from standard waste dumps under anaerobic conditions and consist mainly of glucose, amino acids, fatty acids, hydroxycarbonic acids, isosaccharinic acids etc. [3]; The final products are carbon-bearing gaseous compounds (such as CH_4, CO_2, hydrocarbons and other volatile compounds) and aqueous compounds (such as $CO_2/HCO_3^- /CO_3^{2-}$ and carbonate complexes);
- Dissolution of surface contamination (e.g. crud); The final products are aqueous compounds;
- Release via corrosion of metals; The final products are some carbon-bearing gaseous compounds and aqueous compounds; However, the progress of corrosion is slow as compared to microbial degradation;
- Lixiviation (leaching) from metals; this process was studied on Zircaloy hulls and activated core parts of LWRs [5]. However, activated core parts are hardly present in a repository for low-level waste, and if any, release irrelevant amounts of gaseous C-14.
- Hydrolysis of carbides (e.g. Al_4C_3 contained in ashes) on contact with water instantly releases hydrocarbons:

$$Al_4{}^{14}C_3 + 12\ H_2O \leftrightarrow 4\ Al(OH)_3 + 3\ {}^{14}CH_4$$

- Methanogenesis:

$${}^{14}CO_2 + 4\ H_2 \leftrightarrow {}^{14}CH_4 + 2\ H_2O$$

Hydrogen needed for this process is generated by anaerobic corrosion of metals, whereas CO_2 comes from aqueous compounds; Methanogenesis is supported by microbial activity and catalysts; Due to the consumption of hydrogen, the total amount of gas is reduced by a factor of 4 as a side-effect; In the case of significant methanogenesis, this has a profound influence on the transportation of fluids; However, due to its low enthalpy, methanogenesis is not favoured over other reactions.

Radioactive waste repositories during their operational phase show a continuous release of C-14 by mine ventilation [6]. Radiation exposure is of limited concern, but released C-14 from deposited waste accounts already for a fraction of at least 2 % within 30 years of operation. This provides evidence of ongoing reactions in the waste. According to some studies, 75 % to 90 % of the C-14 is ${}^{14}CO_2$. The remaining fraction is dominated by ${}^{14}CH_4$. The domination of ${}^{14}CO_2$ proves aerobic conditions in the waste that prevail during the operational phase (initial condition, cf. above).

4 Transport of C-14 to the Biosphere

C-14 is transported from the emplacement chambers to the biosphere via gas and brine. CH_4 is poorly soluble in brine. In contrast, CO_2 dissolves readily at the prevailing conditions in the repository (pH, pressure, temperature) to form aqueous compounds. The concentration of aqueous $CO_2/HCO_3^- /CO_3^{2-}$ and complexes is determined by precipitation of carbonates and sorption on solid backfill [4]. Saturation of the brine with CO_2 resulting in release of gaseous CO_2 is not expected to occur.

As a result, C-14 is transported via the brine pathway predominantly as aqueous ${}^{14}CO_2/H^{14}CO_3^- /{}^{14}CO_3^{2-}$ and carbonate complexes, whereas via the gas pathway it is transported predominantly as ${}^{14}CH_4$. Gas and brine undergo chemical and mechanical interactions with each other and with the solid phase that they encounter during transport. The interactions determine the release rate of C-14 to the biosphere.

Isotopic dilution occurs when the C-14-bearing brine is transported out of the emplacement chamber and equilibrates with carbon containing solid phases [1] that are free of C-14. Due to the composition and the relative amount of brine and mineral phases, most C-14 will be fixed as carbonates in the solid phase. Some C-14 dissolved in the brine may be bound to compounds other than aqueous $CO_2/HCO_3^-/CO_3^{2-}$ and carbonate complexes and thus undergo different processes. The amount of such compounds is not expected to be significant.

Gaseous C-14 is mostly in the form of CH_4. Gaseous CO_2, if any, will undergo isotopic dilution that lowers its content of C-14. The release of C-14 from the mine via gaseous compounds may be hindered by their oxidation to aqueous $CO_2/HCO_3^-/CO_3^{2-}$ when transported through sulphate-containing backfill and rock (e.g. gypsum overlying a salt rock formation) in presence of water. Most of this aqueous $CO_2/HCO_3^-/CO_3^{2-}$ will precipitate as carbonate. The retardation of gas transport in the mine by technical and natural barriers gives more time for these processes and the radioactive decay to take place and thus lowers the released fraction of C-14 to the biosphere.

5 Conservative Approach

Hitherto the following scenarios have been considered in more or less conservative approaches, which release the inventory of C-14 to the biosphere:

1) The long-term assessment of a potential radiation exposure by $^{14}CO_2$ yielded an insignificant dose for both pathways (brine and gas). The dose limits are easily met with a large safety margin. Therefore, the exposure by $^{14}CO_2$ need not be further discussed.
2) The contribution of non-oxidised $^{14}CH_4$ reaching the biosphere and causing potential radiation exposure as $^{14}CH_4$ is also insignificant (see Fig. 1a).
3) The contribution of $^{14}CH_4$ being oxidised prior to reaching the biosphere and causing potential radiation exposure is also insignificant (see Fig. 1b).
4) Provided $^{14}CH_4$ is released via the gas phase directly to the overlying rock, subsequently reaches the biosphere and is oxidised there (in a pond, marsh, etc.), a potential radiation exposure by ingestion of fish may become relevant (see Fig. 1c), but only if the complete C-14 inventory is released as $^{14}CH_4$.

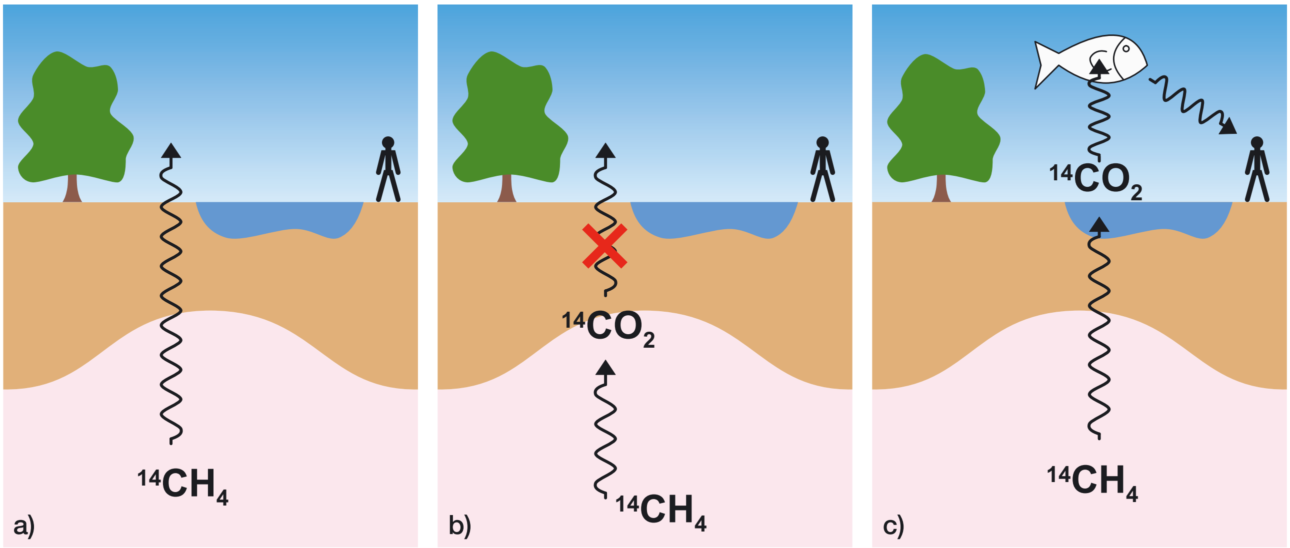

Fig. 1: Scenarios for radiation exposure by $^{14}CH_4$

The release of the complete C-14 inventory as $^{14}CH_4$ is an unnecessarily conservative assumption. Consequently, the above approach should be reconsidered and a more realistic approach applied.

6 Realistic Approach

A realistic approach considers the processes outlined before. Priorities are set in the order of decreasing importance. At the present level of refinement only the most important processes are considered.

The assumptions and simplifications underlying this approach are given below:

1) C-14 is mainly bound to organic materials. Within these it is homogeneously distributed. Occurrences of easily purgeable C-14 in e.g. carbides or non-purgeable C-14 in inorganic compounds and already released fractions of C-14 were not considered. They would lower the total release, but were neglected at the present level of refinement.
2) CO_2 is formed mostly by oxidation under aerobic conditions prior to closure and during a transient period after closure.
3) A realistic gas generation rate which includes methanogenesis for CH_4 and $^{14}CH_4$ is applied.
4) Isotopic dilution was considered only for $^{14}CO_2$. Isotopic dilution of $^{14}CH_4$ is slow and was therefore neglected at the present stage.
5) Oxidation of CH_4 during transport that would result in the precipitation of C-14 is insignificant in absence of oxidising backfill, rock or other materials. It was therefore neglected.
6) The equilibration of carbonates (solid, solution, gas) is assumed to be instantaneous.
7) C-14 in transient intermediates of chemical processes was disregarded because of the currently insufficient knowledge.
8) C-14 released prior to mine closure does not contribute to the post-closure exposure. It lowers the total release after closure but is disregarded, because of its relatively small amount.
9) C-14 in carbonates is mainly precipitated and remains fixed to the solid phase.

A computational model is not available to cover all these aspects quantitatively in detail. Therefore, the distribution of C-14 was estimated using a semi-quantitative model adjusted to the conditions in a final repository in salt (see Fig. 2).

Experience from landfills, natural analogues and laboratory studies show that approx. 48 % of the total carbon inventory is not degradable in the long-term under conditions comparable to those in a repository in salt.

All oxidants available in an emplacement chamber can oxidize on the average only 46 % of the carbon inventory to CO_2.

Approx. 6 % of the carbon inventory is reduced to CH_4.

Approx. 44 % of CO_2 will be fixed in the solid phase as carbonates. Less than 1 % of CO_2 will dissolve as CO_3^{2-} /HCO_3^-/CO_2 and much less than 1 % will be present as gas after geochemical equilibration processes.

5,7 % of CH_4 will be present as gas, whereas 0,3 % will dissolve. A negligible amount of CH_4 will be fixed or sorbed on solid phases.

C-14 is assumed to be uniformly mixed with inactive carbon.

Under the assumed conditions the semi-quantitative model shows that less than 8 % of the total C-14 is released to the biosphere. Less than 1 % of C-14 as CO_3^{2-} /HCO_3^- /CO_2 and approx. 0,3 % as CH_4 is released in solution, approx. 5,7 % of C-14 as CH_4 and less than 1 % as CO_2 is released as gas. Only CH_4 can contribute to the potential radiation exposure in the biosphere, provided that it is oxidised in surface waters (see Fig. 2).

A deeper level of refinement would lower further the predicted release of C-14 to the biosphere and thus the potential radiation exposure. This would include:

1) Amount of release of C-14 by mine ventilation already before closure.
2) Oxidation of CH_4 and other non-CO_2-compounds to CO_2 during passage through sulphate-containing backfill or rock;
3) Equilibration with other phases resulting in precipitation and sorption of $^{14}CO_2$;
4) Isotopic dilution of C-14 during transport of CO_2 through non-contaminated backfill and rock.

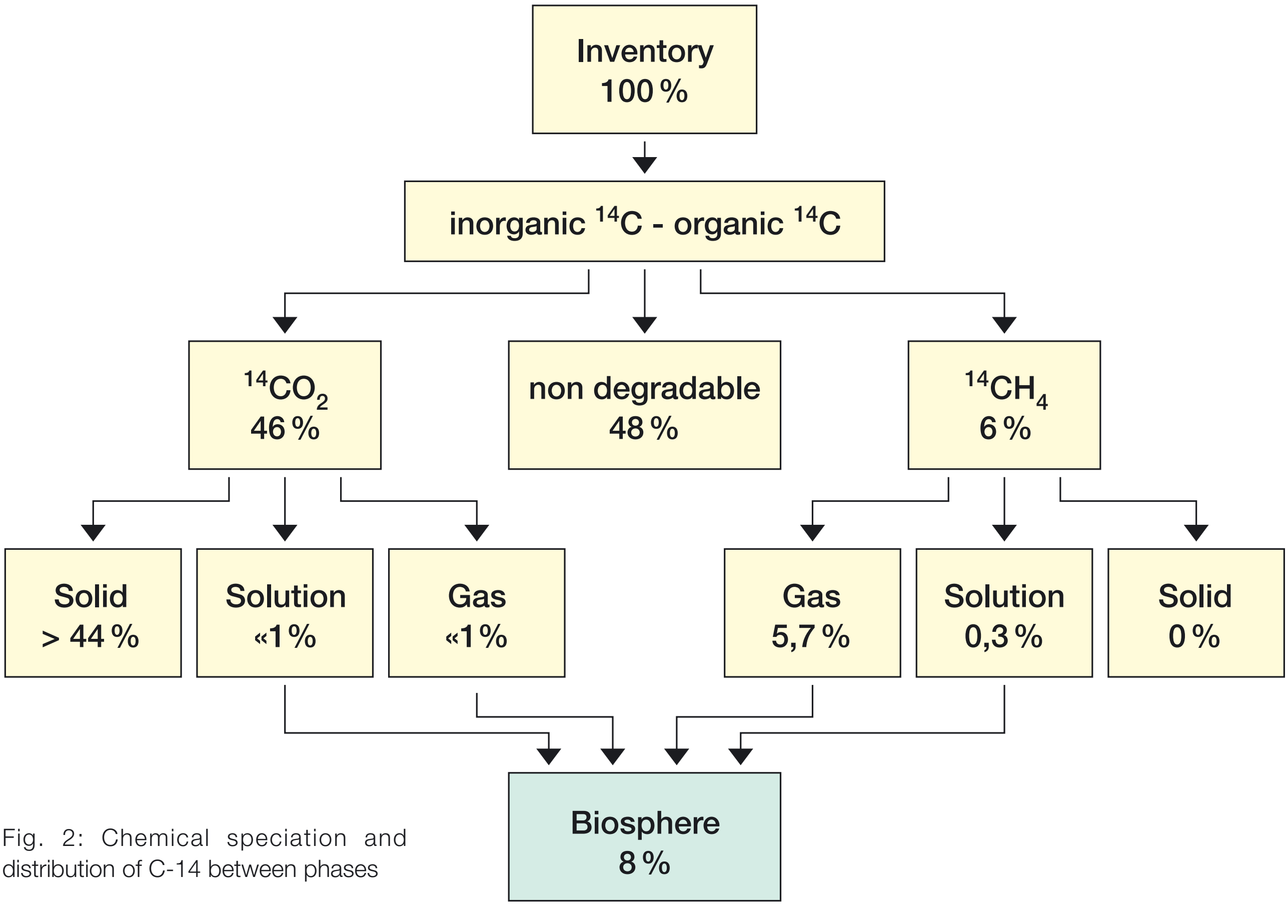

Fig. 2: Chemical speciation and distribution of C-14 between phases

7 Summary and Conclusions

A realistic approach for estimating the release of C-14 after closure of a final repository for low-level radioactive waste in a mine shows a significantly lower release of C-14 than that predicted by previously used conservative approaches. As a result, a significantly lower potential radiation exposure would be assessed, hence increasing the safety margins to federal radiation exposure limits.

Consequently, with a realistic approach it is possible to demonstrate that a repository in salt complies with the ALARA (as low as reasonably achievable) principle, thus facilitating licence approval. Conservative approaches may face difficulties to demonstrate a well-balanced radiological design as requested by the ALARA principle

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Migration and retention properties of the Czech reference granitic samples

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Abstract

Czech deep disposal concept is based on granitic host rock. Deep knowledge about transport mechanisms occurring in the rock is therefore crucial to achieve reliable safety assessment of deep geological repository. Migration behaviour of radionuclides and other contaminants in crystalline rocks is strongly associated with sorptive and diffusive properties of the rock matrix. Retardation caused by sorption onto minerals is of main importance for sorbing radionuclides (Cs, Sr), diffusion into the intact rock matrix is of particular importance for those radionuclides which exhibit little or no sorption (^3H, I). Diffusion into the rock matrix is dependent on diffusion coefficient, D_a and on those material properties as transport porosity ε, tortuozity τ, pore constrictivity σ and the rock capacity factor α, considering total porosity ε_t, sorption coefficient K_d and bulk density of rock samples ρ.

Batch sorption on crushed samples, sorption on coupons and diffusion experiments with Cs, Sr, Eu with Czech reference granitic samples and artificial granitic groundwater were performed, although the experiments extended into long period. K_ds were determined for both crushed samples and coupons. Diffusion coefficients for both sorbing (Cs, Sr, Eu) and non-sorbing radionuclides (I, ^3H) were determined, altogether with rock capacity, formation and geometric factors.

The first attempt to model Cs diffusion using activity decrease in the input reservoir was performed using invented code, developed in GoldSim environment.

Complementary experiments for studying Cs sorption and diffusion into the rock matrix were accomplished using Rutheford Backscattering Spectroscopy (RBS). Two types of diffusion pathways seem to be present within homogenous granitic rock: intergranular pores and mineral grain microcracks.

1 Introduction

According to The Concept of Radioactive Waste and Spent Nuclear Fuel Management in Czech Republic the disposal of high-level waste and spent nuclear fuel into a deep geological repository is the most realistic option for disposal. It is expected that a deep geological repository in the Czech Republic will be built in granitic rocks (RAWRA, www.rawra.cz). Deep knowledge about transport mechanisms occurring in the rock is therefore crucial to achieve reliable safety assessment of deep geological repository.

Therefore, the research has been focused on description and quantification of retention and migration processes of PA relevant radionuclides in crystalline rock of granitic type. The different approaches were used: batch sorption experiments on crushed samples and rock coupons, diffusion experiments, RBS study with non-sorbing radionuclides, and modelling. The results of the laboratory experimental programme can be also interconnected with lab experiments held on samples drilled off during long-term in-situ diffusion experiment, accomplished in Grimsel in the frame of Long Term Diffusion (LTD, www.grimsel.com).

2 Theory

Deep geological repositories are planned for disposal of high-level wastes over long time periods. In case of repository disruption, dissolved radionuclides will be transported by groundwater along fractures. Retardation for most of radionuclides is expected, including sorption on the fracture infill/walls and diffusion into the rock matrix, opening fresh surfaces for additional sorption. Retardation of radionuclides and other contaminants in granitic rock is therefore strongly associated with diffusive and sorptive properties of the fissure infill and rock matrix. In presented work the attention was paid to homogenous rock matrix properties.

The diffusion process is governed by the Fick´s first and second law, reported elsewhere [1, 2, 3, 4, 5]. The rate of change of concentration at the point in one dimension is described according to Fick´s second law by Eq. (1)

$$\frac{\partial C_p}{\partial t} = \frac{D_p}{R_p}\frac{\partial^2 C_p}{\partial z^2} \qquad (1)$$

where C_p *is the concentration in the pore water (mol m*$^{-3}$*),* D_p *the pore diffusion coefficient (diffusivity, m*^{2}*s*$^{-1}$*), R is the retardation factor in the rock matrix (m*$^{-3}$*·kg*$^{-1}$*)*

Radionuclide moves in the pore water. Tortuosity of the pores increases the diffusion path, and constrictivity reflects the changing size of the pores. Therefore the pore diffusion coefficient should be different form that in unconfined water [1]:

$$D_p = D_w \frac{\partial_D}{\tau^2} \qquad (2)$$

where δ_D *is the constrictivity,* τ^2 *the tortuosity,* D_w *the diffusivity in unconfined water and* D_p *the pore diffusivity in pores. The ratio* δ_D/τ^2 *is called geometric factor G.*

Total porosity ε is made up of *a transport porosity*, ε_t and a *storage porosity*, ε_d

$$\varepsilon = \varepsilon_t + \varepsilon_d$$

The transport (pore) porosity ε_t*, the tortuosity and the constrictivity* can be united into one parameter called *formation factor F*

$$F = \varepsilon_t \frac{\delta_D}{\tau^2} \qquad (3)$$

The *effective diffusion coefficient* D_e can than be expressed by Eq. 4, where

$$D_e = \varepsilon_t D_p = \varepsilon_t D_w \frac{\delta_D}{\tau^2} = F D_w \qquad (4)$$

Sorption causes retardation of the radionuclide in the rock matrix. Sorption, determined using batch sorption experiments is then often described by the mass based *sorption coefficient* K_d *(m*^{3}*kg*$^{-1}$*)*:

$$K_D = \frac{C_{rock}}{C_{solution}} \qquad (5)$$

where C_{rock} *is the concentration of nuclides per solid mass (mol.kg*$^{-1}$*) and* $C_{solution}$ *is the concentration of nuclides per solid mass (mol.m*$^{-3}$*).*

Retardation coefficient in the rock pores water is then defined as (6)

where K_d *is mass-based distribution coefficient (m*^{3}*kg*$^{-1}$*) and r is the bulk rock density (kg.m*$^{-3}$*).*

$$R = 1 + \rho \frac{(1-\varepsilon_t)K_d}{\varepsilon_t} \qquad (6)$$

The alternative way of examining diffusion is using apparent *diffusion coefficient* (D_a) and *rock capacity factor (α)* as in (7)

$$D_a = \frac{D_p}{R} = \frac{D_e}{\alpha} = \frac{D_e}{\varepsilon_t + \rho K_d} \qquad (7)$$

Rock capacity factor α is dependent on transport porosity, accessed by tracer (ε_t) and on the amount of tracer sorbed by the rock sample. For non-sorbing neutral species $\alpha = \varepsilon_t$; for cations (sorbing diffusants) $\alpha > \varepsilon_t$; and for anions $\alpha < \varepsilon_t$, treating anion exclusion formally as negative sorption.

3 Material and solution used

Considering potential granitic host rock in the Czech Massive, homogenous fine-grained non-fractured granite samples from Pribram region (Central Bohemia, Central Moldanubian Pluton) were chosen as reference samples for the study. The silicate analyses and mineralogical composition are shown in Table 1.

Table 1: Silicate and mineralogical composition of Pribram granite.

Silicate analyses	Wt. %	Mineral composition	Vol. %
SiO_2	70,72	Quartz	21
TiO_2	0,22	K feldspar	37
Al_2O_3	14,31	Plagioclase	30
Fe_2O_3 total	1,45	Biotite	3
FeO	1,21	Chlorite	5
MnO	0,088	Hornblende	4
MgO	0,76	Epidote	
CaO	2,64		
Na_2O	3,54	Porosity	0,06 – 0,2
K_2O	3,48		
P_2O_3	0,065		
H_20	1,1		
CO_2			

Several samples with different mafic mineral content were also used to determine influence of the rock constituents on radionuclide sorption (tonalite, diorite, gabro). Its composition is given elsewhere [6]. Rock samples were processed in the way shown on Fig. 1

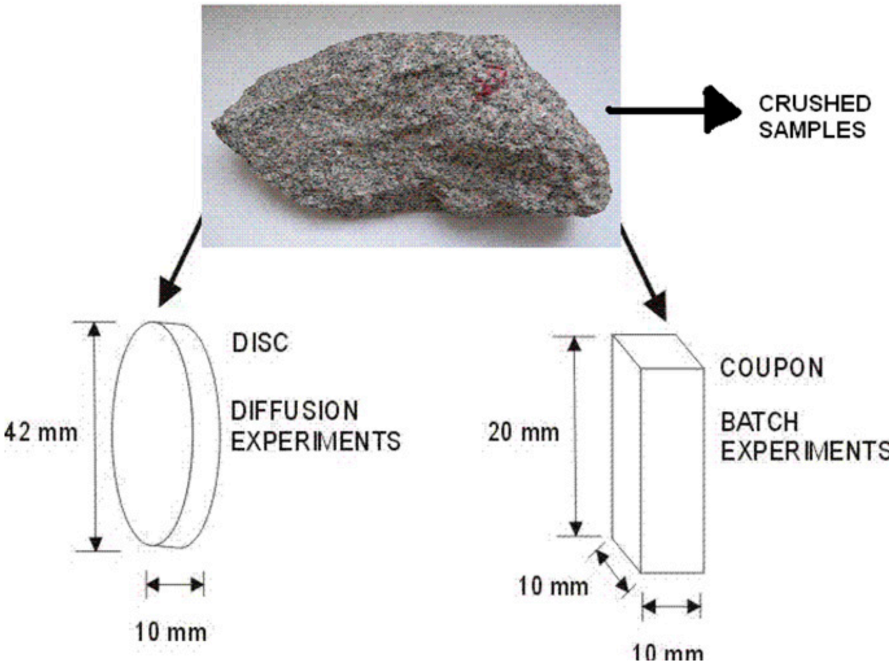


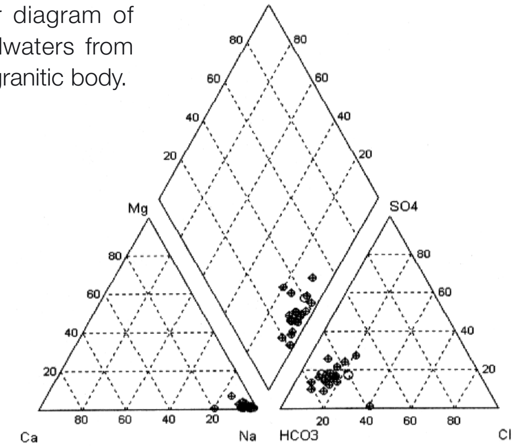
Fig. 1: The scheme of sample processing for different experimental methods.

Artificial granitic water, prepared as artificial equivalent of granitic deep groundwater from the locality, was used as a solute. The artificial groundwater composition is listed in Table 2 and shown on Fig. 2.:

Table 2: Composition of artificial groundwater used in laboratory retardation experiments.

Component	Concentration (mg/l)
Ca	3,4
Na	74,1
Mg	0,6
Cl	11,9
S(6)	19,3
pH	9,1

Fig. 2: Piper diagram of deep groundwaters from the Pribram granitic body.



4 Experimental

Radionuclide retardation within granitic rocks matrix was studied using different laboratory method. The experimental methods were following:

- batch sorption experiments with crushed samples
- batch sorption experiments with rock coupons
- static through-diffusion experiments
- trace element diffusion into rock, measured by RBS

Batch, and diffusion experiments using ^{3}H, ^{125}I (non-sorbing tracers) and ^{137}Cs, ^{85}Sr and ^{154}Eu (sorbing tracers) were performed on crushed samples, coupons and discs respectively. Synthetic granitic water simulating real groundwater from crystalline rock massif was used as a solute. Sample porosity was determined using by Hg porosimetry and water saturation method (0,2 - 0,09 %) [7, 8].

4.1 Batch experiments

Static batch sorption has been the standard method for studying the interaction of radionuclides and crystalline rock since early 80ties. Hereby, the experiments were performed crushed granitic samples with different grain size (<0,063 mm, 0,063 - 0,25 mm, 0,25 - 0,8 mm, > 0,8 mm), using ^{125}I, ^{137}Cs, ^{85}Sr and ^{152}Eu radionuclides to derive distribution coefficient R_d, used instead of K_d if steady state is not reached. The results are summarised in the Table 3 (R_d max – min, m^3kg^{-1}).

Table 3: Radionuclide sorption range onto crushed samples and rock coupons, expressed in terms of distribution coefficient R_d (m^3kg^{-1}).

Radionuclide	Rd (m^3kg^{-1}) – crushed samples	Rd (m^3kg^{-1}) – coupons
^{137}Cs	0,6 – 0.015	0,031 – 0, 044
$^{152,154}Eu$	0,578 – 0.15	0,086 – 0,0003
^{85}Sr	0,187 – 0,01	0,004 – 0,0001
^{125}I	0,0086 – 0	0,0002 – 0

Sorption of Cs and Sr was found to be dependent on grain size: coarser fraction and coupons exhibit lower sorption than finer rock fraction. The most efficient sorbent of Cs and Eu was found to be diorite (biotite 11%, silica 15%). Sorption of Eu was observed to be independent on grain size and retention of iodine was almost negligible. Therefore, it could be again concluded that crushing could generally lead to overestimation of cation sorption in comparison with experiments in situ, observed in many works.

Sorption of Cs onto granite coupons continued even after 2000 hours (see Fig. 3). The first phase (up to 400 hours) could be assigned as fast sorption on outer available surface, on the other hand the second phase can be described as radionuclide binding on the inner sorption sites in sheet silicate interlayers, and/or as diffusion [9, 10]. According to the data from similar type of Czech granite, fraction of Cs sorbed on outer sorption sites, on the edges of sheet silicate interlayer and on the sites within the interlayer could reach up to 10,5%, 49,1% and 40,4% [672 hours of sorption, unpublished results]. It can be assumed that the fraction of Cs sorbed onto inner interlayer sites will increase with time as trapped cations move from edges into the direction of distant sites with more stabile structure. This Cs fraction stay immobilised within the rock and cannot be desorbed.

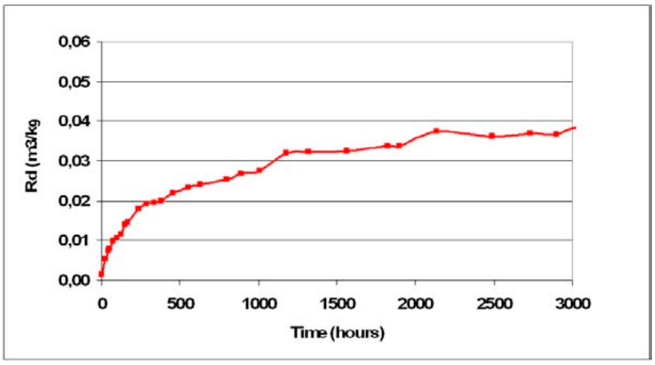


Fig. 3: Long time span of ^{137}Cs sorption onto granitic rock coupon.

4.2 Diffusion experiments

Selected granite rock disc samples were used for through-diffusion experiments with ^{3}H, ^{125}I, ^{134}Cs, ^{85}Sr and ^{154}Eu. The methodology was reported elsewhere [2, 4, 5, 10]. The activities in both input and output reservoirs were monitored with the aim of determine the shape of the breakthrough curves. Non-sorbing radionuclide (^{3}H, ^{125}I) experiments revealed radionuclide breakthrough curves that were evaluated using a time-lag method (see Fig. 4) to determine apparent diffusivity coefficient. D_a (m^2s^{-1}). The mathematic solution is given in e.g. in [11, 12]. This method of determination of D_a can be used only in cases in which activity A_1 in the injection cell is constant with time and A_2 is negligible compared to A_1 in all times, no bulk flow (advection) occurs and rock sample is homogenous.

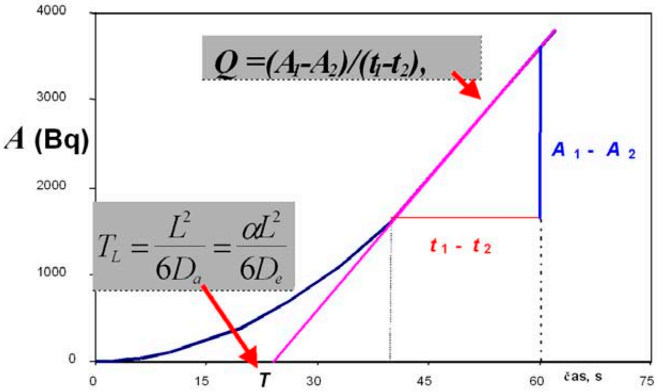


Fig. 4: Time-lag method: D_a determination using activity vers. time dependence.

Rock capacity factor, formation factor and geometric factor for non-sorbing species were calculated using equations mentioned above for given rock types using porosity values, measured by water immersion method and Hg porosimetry (0,08 – 0,22%). The conservative Kd values were used for calculations. Only for Cs both, conservative and realistic values are mentioned. Summarisation is given in the following Table 4.

Table 4: Rock capacity factor, apparent diffusivity Da ($^{m2s-1}$), formation factor F_f and geometric factor G calculated for diffusion experiments. Representative values are listed.

Radio-nuclide	Rock type	Kd used (m^3kg^{-1})	Porosity (%)	Rock capacity factor α	D_a (m^2-s^{-1})	D_e (m^2s^{-1})	Formation factor F_f	Geom. factor G
^3H	Tonalite	0	0,3	8.10^{-4}	$6,7.10^{-11}$	$5,36.10^{-14}$	$2,23.10^{-5}$	0,01
	Gabro	0	0,18	$1,8.10^{-3}$	$8,3.10^{-11}$	$1,49.10^{-13}$	$6,23.10^{-5}$	0,035
^{125}I	Granite	0	0,08	8.10^{-4}	$9,0.10^{-11}$	$7,2.10^{-14}$	3.10^{-5}	0,0375
	Diorite	0	0,2	2.10^{-3}	$3.\ 10^{-12}$	3.10^{-14}	$2,5.10^{-6}$	0,00125
^{125}Cs	Granite	0,032	0,08	83				
Modelled	Granite	0,06	0,08	160	$5,3.10^{-14}$	$8,49.10^{-12}$	$4,5.10^{-3}$	2,34
$^{152,154}Eu$	Tonalite	0,00856	0,3		No breakthrough			
^{85}Sr	Gabro	0,011	0,18					
	Tonalite	0,0001	0,3	0,27	No breakthrough			

However, a set of additional experiments using different cells and rock samples with granitic type rocks (gabbro, diorite, tonalite) showed a variation of D_a/D_e and rock capacity factor value. Even within one rock core the differences were found. However, formation factors for ^3H and ^{125}I did not vary as much (see Table 4). Therefore we can assume that the difference in between D_a for non-sorbing nuclides differs only due to differences in D_w, not due to material properties. The formation and geometric factors calculated for sorbing and non-sorbing radionuclides varied as well. As the experimental data set was not as wide as for example in [9], further analyses of diffusivity variability will be accomplished within LTD project [13].

2 Modelling

Eu, Sr and Cs as sorbing radionuclides have not breakthrough the granitic samples so far even after 2000 hours of experiment continuation. Eu and Sr radionuclide experiments have been still continuing (see Table 5). Therefore, diffusivity D_a for Cs diffusion into granitic samples was calculated using Cs decrease in the inlet reservoir and rock properties (see following Table 5). Radionuclide activity decrease considering sorption and diffusion was modelled using GoldSim diffusion module, considering linear sorption and no advection transport. Three possible porosity values were used for calculation. Realistic K_d value was used for modelling. Modelled and experimental line fit is presented on Fig. 5.

Table 5: Rock properties used for Cs diffusivity calculation.

		Porosity	Calculated Da $x10^{-14}$ (m^2/s^{-1})
D_w (m^2/s^{-1})	$2,06 \times 10^{-9}$		
Tortuosity	0,6	0,008 (Pribram granite)	5,3
Kd (m^3/kg)	0,06	0,0065	4,3
Density (kg/m^3)	2,8	0,005	3,7

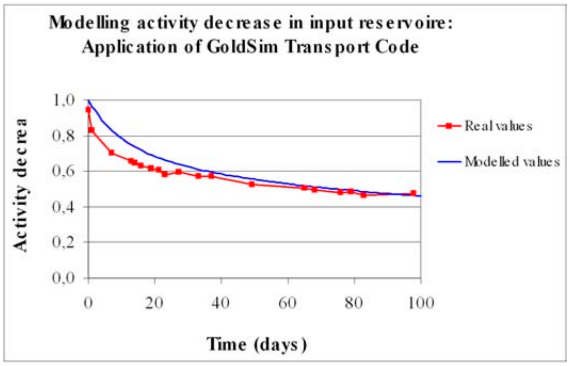


Fig.5: ^{137}Cs activity decrease in the inlet reservoir of diffusion cell – real and modelled data. GoldSim diffusion dashboard (NRI/CTU)

The values obtained using GoldSim diffusion module revealed the data in consistency with other reported e.g. [10]. This attempt showed the module could be used as a useful tool for predicting diffusivity values for long-term experiments with sorbing radionuclides.

The model refinement will follow in the next step altogether with Sr and Eu experiment modelling.

2 Cs Diffusion: RBS Study

Further complementary method for studying of radionuclide diffusion into the rock was used: Rutherford Back-Scattering spectrometry (RBS). The first attempt in the Czech Republic was undertaken in cooperation with the Institute of Nuclear Physics of the Czech Academy of Science.

Rutherford Back-Scattering spectrometry (RBS) is a non-destructive nuclear method for elemental depth analysis of nm-to-µm thick films. It comprises measurement of the number and energy distribution of energetic ions (usually MeV light ions such He^+) back scattered from atoms within the near-surface region of solid targets. The detection limits vary from 10^{12} -10^{15} atoms/cm^2. The mass resolution should be improved using heavy ion projectiles down to one mass unit [14].

Two identical homogenous granitic samples were submerged into non-active 0,1M (Sample 1) and 0,01M Cs (Sample 2) solutions for 14 days. Samples were withdrawn, dried with tissue and used for the measurement. The methodology can be referred to [15, 16]

RBS measurements were performed afterwards using ion beam 2.53 MeV, He^+ at the scattering angle 170°. The sample was scanned laterally to get information from different granite grains (mica, feldspar, quartz). Elemental depth profiles were deduced for Cs, Si, Ca (K) and Cs were detected on different sites and correlated with increasing depth of the measurement. The He^+ beam spot used was 1 mm^2.

Cs retention can be positively correlated with Fe and Ca(K) content (see Fig.6 and 7).

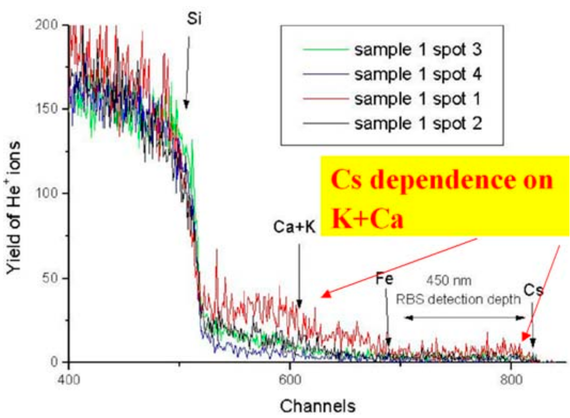


Fig. 6: Influence of Ca (K) and Fe on Cs binding on feldspar grain (0,01M Cs, SAMPLE 1, spot 1 – Feldspar)

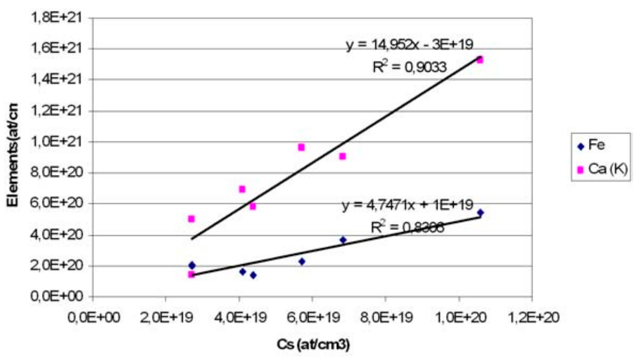


Fig. 7: Dependence of Cs sorption on Ca (K) and Fe content in granitic samples (all measured spots for both samples)

Cs sorption/migration on the granitic rock surface can follow different pattern according to mineral grain distribution. Cs was detected to be purely sorbed on quartz (see Fig. 8 c,d), sorbed on the surface of mica (atom amount was stabile for a definite depth and then decreased rapidly – Fig. 8b) and diffusing into feldspar mineral grains. The last case can be documented by diffusion profile on Fig. 8a: Cs concentration decreased in gradual diffusion profile. However, diffusion coefficient was not possible to calculate.

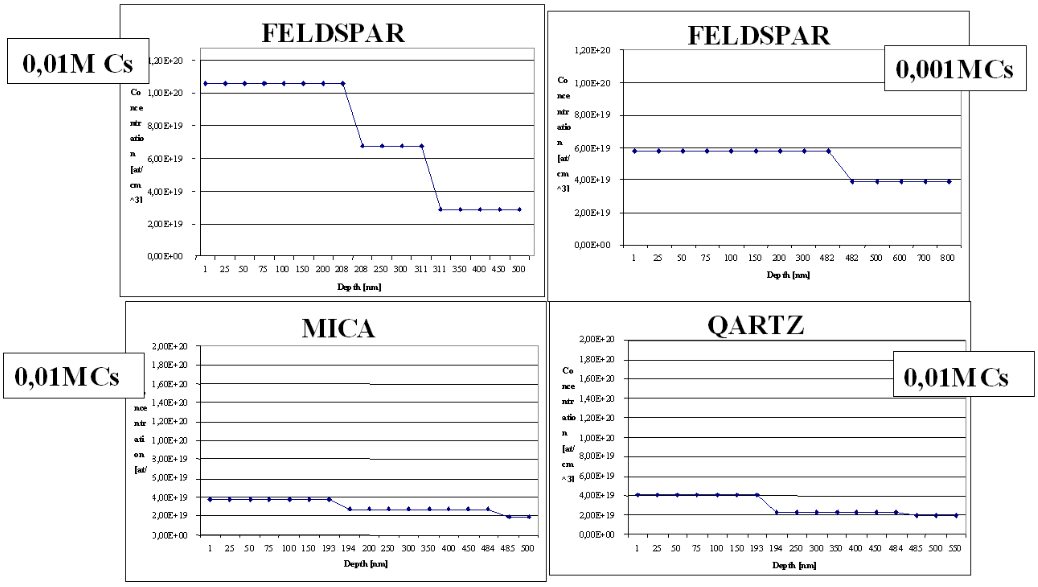


Fig. 8: RBS profiles of Cs sorption onto different mineral grains of granitic sample. From upper left a) feldspar, b) mica, lower left c) + d) quartz.

According to the results of RBS measurements, two types of diffusion pathways seem to be present: intergranular pores and mineral grain microcracks. Feldspar is known to contain microcracks and opened cleavage planes enabling additional diffusion pathway to intergranular space [10, 17]. This piece of knowledge should be seriously considered in evaluation of granitic migration properties, relevant to PA, and in particular during the process of diffusion modeling: not only porosity/interganular connected pore space could play a role in the potential radionuclide migration within the host rock. Mineral grain composition and conditions could play also a role as a migration parameter. The phenomenon is also being studied within [13].

2 Conclusion

The aim of the presented paper is to give a short overview of multimethod approach to study retention parameters of granitic rocks, considered in The Concept of Radioactive Waste and Spent Nuclear Fuel Management in Czech Republic as a potential host rock for geologic repository of HLW. The sorption distribution coefficients were measured for crushed samples and coupons, diffusion experiments performed for sorbing and non-sorbing species. The results of laboratory experiments showed that sorption and diffusivity are much more complex to be described by a simple diffusion coefficient and diffusivity.

Sorbing species retention is generally dependent on particle size: sorption decreases with increasing particle size. This generally could cause overestimation of the results due to the higher surface area of the crushed samples. Moreover, the sorption continuation with contact time could be the evidence for inner sorbing sites available for sorption in performance assessment time scale. RBS measurements traced two diffusion pathways for possible migration of radionuclide within the rock: inter-granular space and mineral grain microcracks. This, according to some authors [17], should be solved including two separate diffusion coefficients into PA models, one for intergranular process and one for microcracks,.

Diffusion of non-sorbing species can be determined within simple through diffusion experiments as those radionuclides can breakthrough the sample. However, sorbing species do not necessarily get through and determination of diffusivities could be accompanied with difficulties. The invented GoldSim diffusion module proved to be an efficient tool to model long-term diffusion experiments even with sorbing radionuclides (Cs).

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The design of a spatial information system for the Morsleben repository

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Abstract
The contribution of C-14 to radiation exposure in the biosphere can be significant as compared to that of other radionuclides disposed in a repository for low-level radioactive waste. The release pathways of C-14 and processes relevant to its release from a closed final repository for low-level radioactive waste are discussed. Because a conservative approach may lead to undue overestimation of the potential radiation exposure, a more realistic approach is outlined. At the present level of refinement, our realistic approach provides a sufficient safety margin to German federal limits for radiation exposure to demonstrate compliance with the ALARA (as low as reasonably achievable) principle and can thus facilitates licence approval.

1 Introduction

The backfilling activities in the Morsleben radioactive waste repository (ERAM) began in 2003. They are necessary to stabilize the intensely mined central part of the repository. An extensive measurement and monitoring system was designed and installed to plan and monitor all activities associated with the closure of the repository [1]. This particular part of the repository has a very complex structure, which means that construction conditions might be critical for certain structural elements during the planned backfilling activities. Still, thanks to geotechnical monitoring it is possible to ensure the local structural stability and the mandatory operational safety [2,3]. The use of an integrated spatial information system right from an early stage makes it possible to locate the great number of continuously acquired measurement results. The interactive visualization proved to greatly facilitate the interpretation of the various types of information [4].

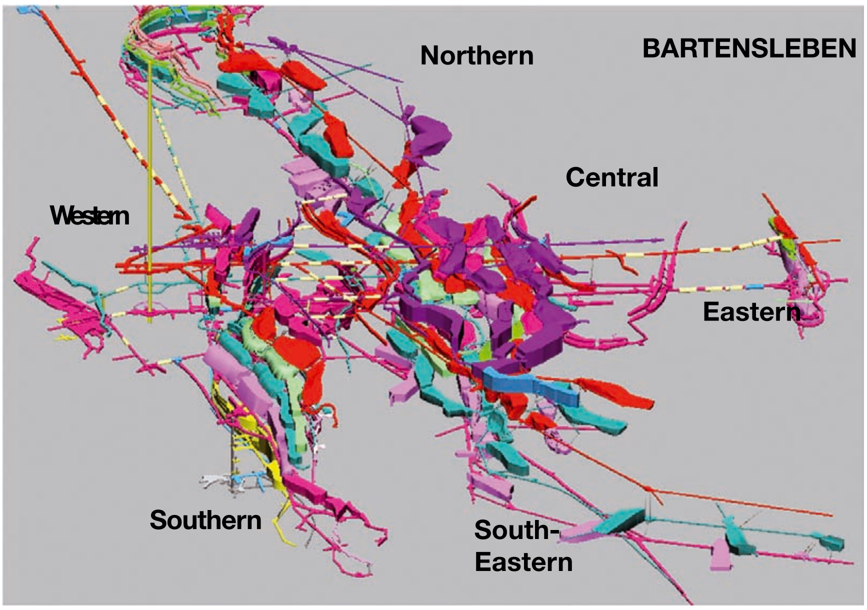

Fig. 1: 3D illustration of the halls of the ERAM

2 ERAM-SIS: Design of a spatial information system

The interpretation of geotechnical data requires a good understanding of the spatial context. A detailed 3D representation is most helpful – together with an interactive visualization tool. The backfilling activities in Morsleben rely on a comprehensive 3D model of the ERAM repository, created during the planning phase. The highly complex shapes were modeled in AutoCAD by digitizing hundreds of paper maps and sections [5] and were later refined by laser-scanning the most relevant halls [6]. In addition, a table describing the attributes (ID, name, volume, zone, altitude, level etc.) of all the cavities was set up as a relational database.

It became evident, that – to tap their full potential – the two sources of information had to be integrated into a single application. The Toporobot approach [7], originally developed for modeling and visualizing natural caves, offered a solution. It had been used before to model and document prehistoric mines (Bergbaumuseum Bochum) [8], to create an inventory of underground features in the cave of Milandre, and to build a web based spatial information system of the Mont Terri Rock Laboratory [9].

Such an information system combines 3D visualization with a database (Fig. 1 and Fig. 2). The ERAM-SIS displays the complex model of all mine passages and halls in real time and offers a fluid navigation in the 3D scene. In addition, all attributes are shown tabulated in spreadsheets, grouped by type of objects. As the views are bi-directionally linked, the user can either select from the spreadsheet and have the corresponding objects highlighted in 3D or pick objects in 3D to see the associated attributes in the spreadsheet. The database table allows to sort and search, to hide/show and color the objects as well as to structure the scene.

The information system offers specific functionality to assist with the analysis of geotechnical data. It calculates the volume of passages or halls, displays the distance between two selected points, and finds the smallest distance between two halls. It displays time series of measured data or observations and it animates the model to show the partially filled volumes. Maps and profiles schematically summarizing the geological situation can be shown or hidden interactively.

In addition to the integrated attributes and geo-referenced data, the program can handle links to further information such as reports, photos, maps, and profiles. Conversely, the information system can be used as an add-on application for a web browser. If this function is enabled, links contained in external documents (e.g. HTML, PDF, SVG) can launch the information system, point to a specific selection and/or have the camera fly to a particular view.

Initially, ERAM-SIS was designed to handle mostly static geometrical objects. But it became soon apparent that it would be particularly useful to display dynamical content such as the backfilling activities either in the planning stage or during its process. Thus, the information system was enhanced to display spatial phenomena in time. This allowed for instance to observe geophysical events in space such as clouds of microacoustical epicenters (Fig. 3). Visual feedback helped then to discover geophysical changes triggered by the backfilling activities.

During the planning stage, the system was used to control whether all predefined constraints (e.g. cavity A needs to be filled before cavity B) were fulfilled and to assist in the optimization of the operations.

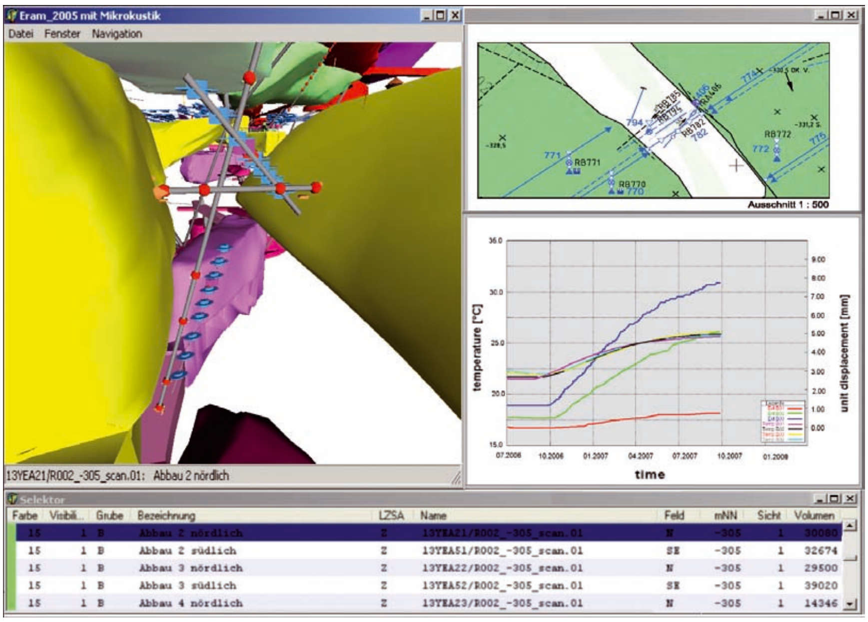

Fig. 2: ERAM-SIS 3D display of filtered and linked information

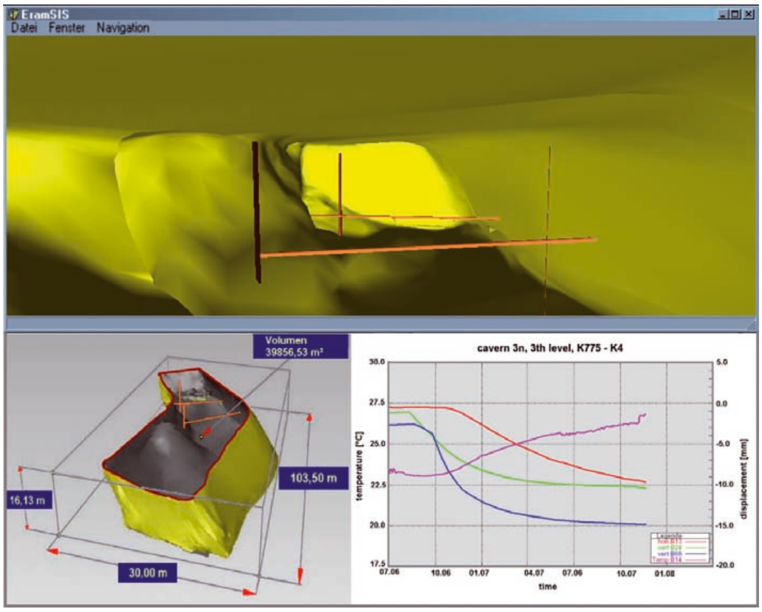

Fig. 3: 3D illustration of the halls and results from measuring systems

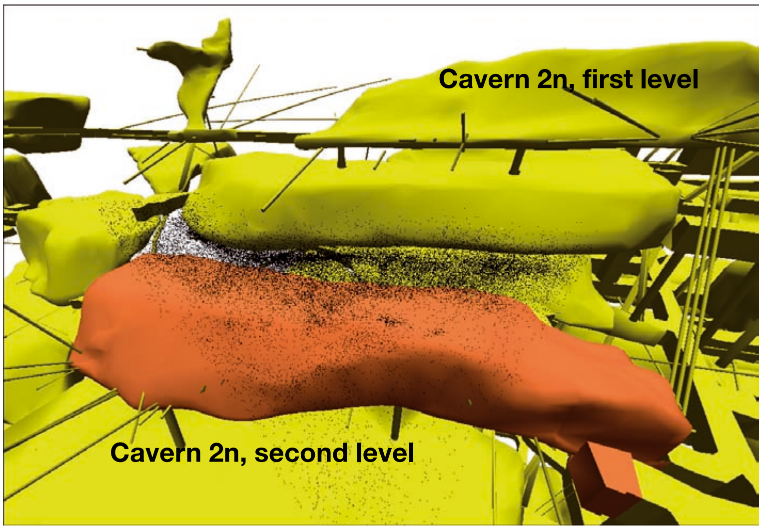

Fig. 4: Micro-acoustic signals above the material store

3 Experiences Gained in Using ERAM-SIS for the Observation Method

The geotechnical observation method has been developed to control rock behavior before, during and after backfilling activities in the areas of geomechanically exposed supporting elements [3].

There is a large quantity of automatically produced measurement results which have to be handled reliably and in a short time. To attain this ERAM-SIS is a suitable aid. It is possible to manage all data in a database which is connected with spatial information. By this it is much easier to interpret measurement results which are located in the rock.

The reliability of this system has already been proven. While backfilling a cavern on the 3rd level an unexpected acceleration of deformations was detected by the extensometers above the material store cavern on the 2nd level.

If the automatic evaluation determines unusual measurements it is important to identify the location and analyze the reliability of the result. In this case the ERAM-SIS is used. With this instrument it can easily be checked if other measuring systems nearby can confirm the first result. Especially for this purpose it was possible to include the micro-acoustic measurements in the interpretation. This system detected clear and strong signals in the determined area as well.

As an immediate measure the material store cavern has been evacuated and closed. The ongoing measures showed that the deformation progress did not stop. So it was decided to supplementary backfill this cavern, because it was unpredictable when or if this deformation progress would stop.

This is one example of how the program ERAM-SIS assists in making decisions which support the safe closure of the repository.

Additionally the application of the observation method to the ERAM requires some adaptations. E.g. new information gained during the process of backfilling can change the utilization of some caverns. This leads to modifications in the backfilling process and the measurement conception, too. For these aspects the ERAM-SIS is also a useful tool. By defining attributes that have to be kept while backfilling caverns it is possible to make a check-up during the whole backfilling progress which determines the locations where the attributes are not complied.

4 Conclusions

The spatial information system ERAM-SIS was designed to visualize the complex shapes of the cavities of ERAM and to integrate different types of information sources, e.g. measurements, geological information, photographs, inspection reports etc. The tool is meant to be used by a wide spectrum of users and offers sophisticated features in an intuitive interface.

Visual inspection and examination has proved to greatly assist with the interpretation of geotechnical data. In addition, it facilitates interaction between interdisciplinary co-workers.

Using this integrated visual approach, considerable insight has been gained already, while at the same time saving time and effort.

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Geomechanical integrity of waste disposal areas in the Morsleben repository

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Abstract

The Morsleben repository has been established in the old Bartensleben salt mine for the disposal of radioactive waste. Several parts of the mine, e.g. the southern, western, and eastern parts, were used requiring the analysis of the integrity of the salt barrier. To this end, numerous geomechanical finite-element calculations have been performed taking into account the specific geological situation and mining geometry as well as location-specific material parameters. The barrier integrity at each location was evaluated considering two criteria. The first criterion is related to dilatancy of rock salt; the integrity is guaranteed if rock stresses do not exceed the dilatancy boundary. The second criterion is related to fluid pressure; the integrity is guaranteed if the hydrostatic pressure of an assumed fluid column extending to the surface does not exceed the minimum principal stress at a certain location in the salt rock. The calculations show that dilatancy of the salt rock is restricted to the near vicinity of the rooms and hypothetical fluid pressure does not exceed the minimum principal rock stress at the outer contour of the salt barrier. Thus, the barrier integrity is given for each of the considered parts of the mine.

1 Introduction

For about two decades the Morsleben repository was used for the disposal of low and medium level radioactive wastes. The repository was established in the old Bartensleben mine, a former salt and potash mine consisting of several mining parts. Especially, the southern, the western, and the eastern parts which are located at the periphery of the mine were used for waste disposal. To assess the geomechanical stability of these structures as well as the integrity of the salt barrier geotechnical safety analyses are necessary. These analyses are based on geological and engineering-geological studies of the site, on laboratory tests and in-situ measurements, and on geomechanical model calculations. Model calculations are the most important part of the geotechnical safety assessment and comprise the geomechanical modelling of the host rock to simulate as closely as possible the conditions of the site and the behaviour of the rock, e.g. geology, repository or mine geometry, initial rock stress, as well as constitutive models and parameters (Langer, 1999).

While actual investigations are focused on the central part which is not used for disposal, but is the most critical area of the Bartensleben mine with the most considerable degree and unfavourable configuration of excavation (Büttner & Heusermann, 2004, Heusermann & Fahland, 2005, Fahland et al., 2007), this paper deals with the safety analysis of the real disposal areas of the Morsleben repository. To this aim, numerous two-dimensional finite-element calculations on characteristic cross sections of the several disposal areas (southern, western, and eastern parts) have been carried out. The results of numerical modelling have been used to analyse the integrity of the salt barrier in those parts mine from a geomechanical point of view.

2 Geological and Mining Situation

The Morsleben repository is located in the fault structure „Allertalzone“ (Fig. 1). The top of the salt structure is at approximately 140 m below mean sea level, respectively about 270 m below ground surface. The thickness of the salt structure varies between 380 m and 500 m. The exploration and modelling of the geological structure of the salt rock and the overburden in several characteristic cross sections of the different parts of the mine is based on the geological mapping of drifts, rooms, and numerous drillcores of the site as well as on ground-penetrating radar measurements (Behlau & Mingerzahn, 2001).

The salt rock is characterised by a distinct folding of the salt layers and a high amount of main anhydrite layers (z3HA) of the Leine-sequence. The structure of the salt rock in the central part includes the main stratigraphic units of the Zechstein strata (salt layers z2HS, z2SF, z3LS, z3OS, z3BK/BD, z3AM/SS, and anhydrite layers z3HA). The structure of the overburden and country rock includes the caprock, Bunter, Muschelkalk, Keuper, Jurassic, Cretaceous, and Quaternary layers.

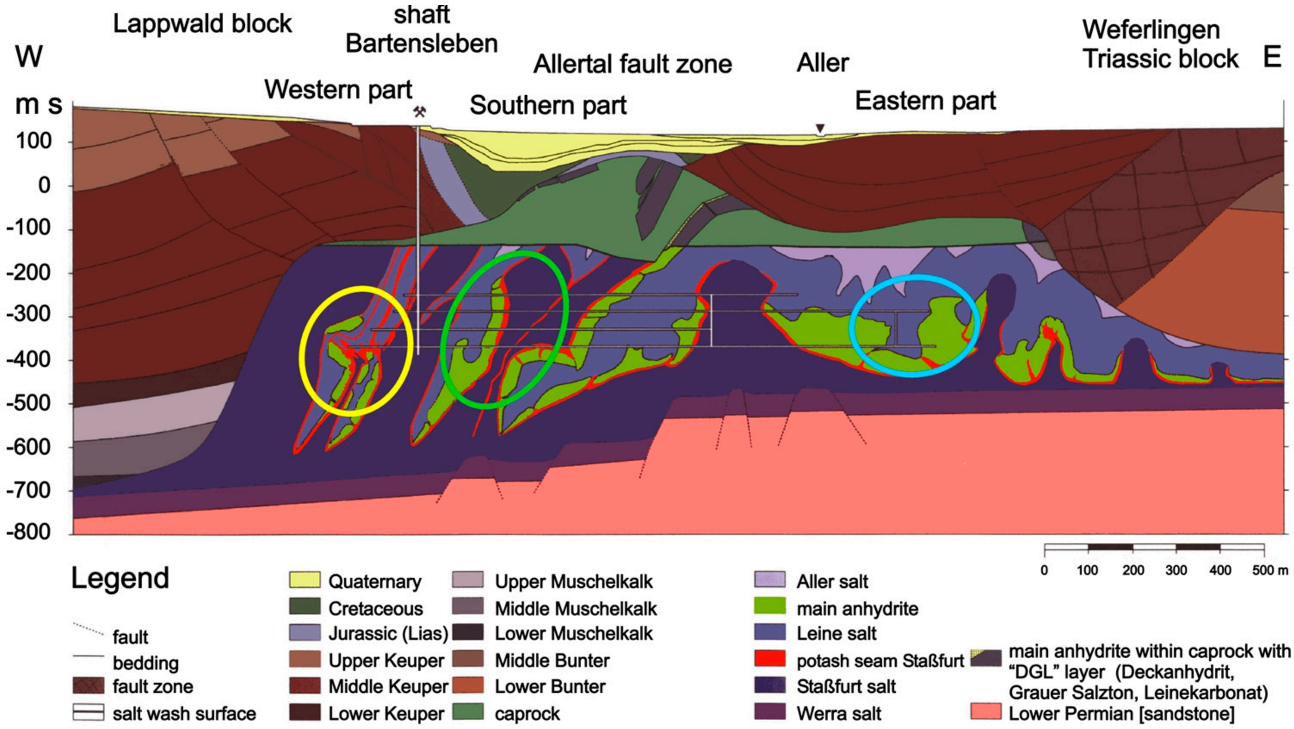

Fig. 1: Geological structure in the Morsleben repository (adopted from BfS).

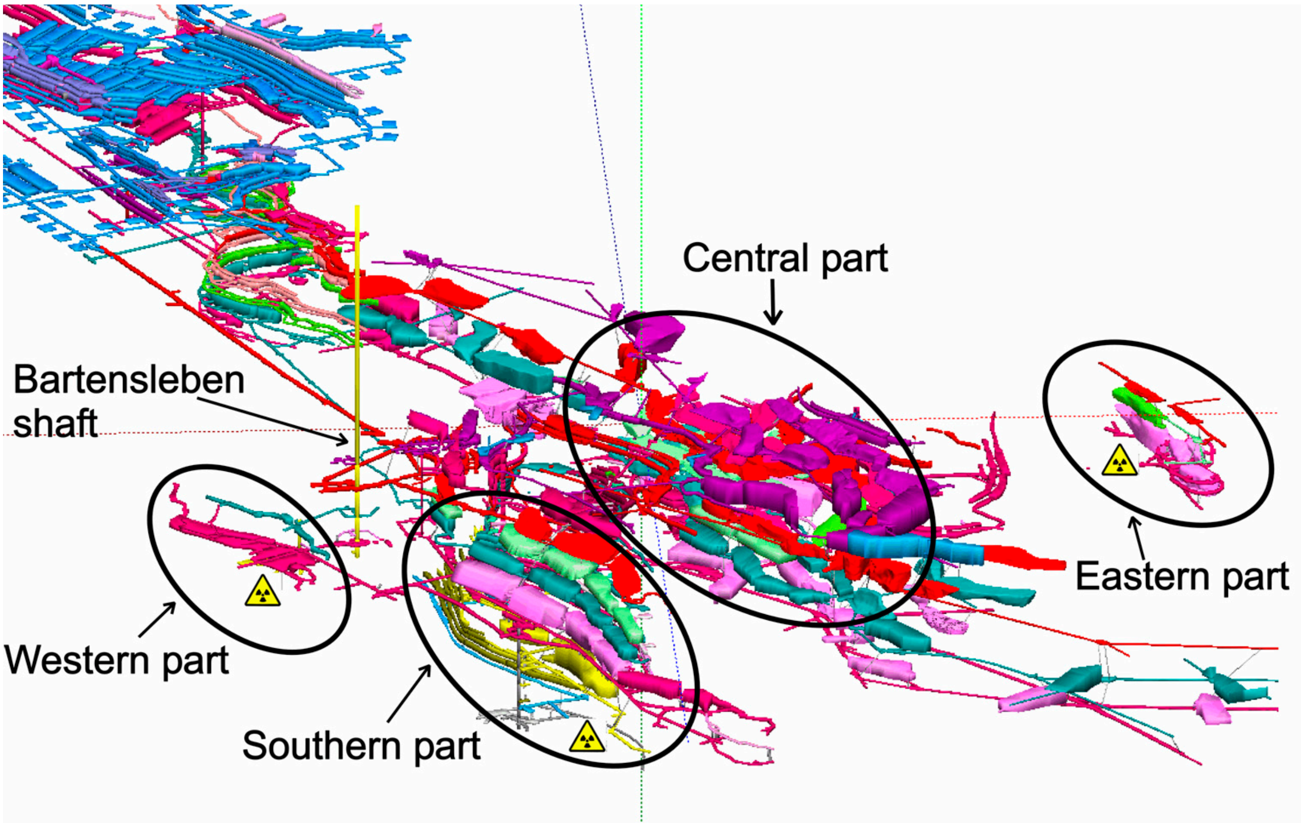

Fig. 2: Mining situation in the Morsleben repository: mapping and scanning data by DBE, model generated with ERAM-SIS (see Heller et al., 2004)

The Morsleben repository consists of the Marie mine and the Bartensleben mine, both connected by drifts on the first and third level. The Bartensleben mine includes several mining parts with four main levels: the northern, western, southern, southeastern, eastern, and central part. Most of the wastes are disposed of in the southern, western, and eastern parts. Fig. 2 shows an overview of the mining situation and old mining rooms excavated a couple of decades ago and located at different levels of the mine. From a geomechanical point of view, the central part which is not used for waste disposal shows the most unfavourable number and configuration of rooms with respect to size, shape, and arrangement in steep rows due to the strong inclination of salt layers. The southern part is characterized by a similar, to a smaller extent unfavourable configuration forming several pillars and roofs between the rooms.

3 Geomechanical and Numerical Modelling

To develop a geomechanical model as a basis of the numerical modelling, the geological structure of the salt rock and the overburden of the several disposal parts as well as the geometry of the rooms must be idealized and simplified (Plischke, 2007). Due to the two-dimensional modelling characteristic geological cross sections have been taken into account oriented more or less perpendicular to the axis of the geological structure and of the mining rooms. For the modelling it was assumed that the rooms were instantaneously excavated in the year 1940. Thus, up to now a time elapse of about 67 years must be regarded to analyse the recent stress and deformation state. The total time elapse considered in all models amounted 100 years up to the year 2040.

3.1 Eastern part

For the eastern part, the main units of the Zechstein strata (salt layers z2, z2SF, z3AM, z3SS, and anhydrite layers z3HA) and composites of the main units (z3LS-OS-BK/BD) are considered. The structure of the overburden was idealized taking into account the main layers caprock cr and Middle Keuper km (Fig. 3, left).

Finite-element modelling of the eastern part was performed on the basis of the idealized structure of the geological layers and the geometry of the old mining rooms. Fig. 3 (right) depicts a plot of a part of the 2-D model, 485 m in height and 650 m wide. The upper boundary of the model has a distance of about 130 m to the ground surface.

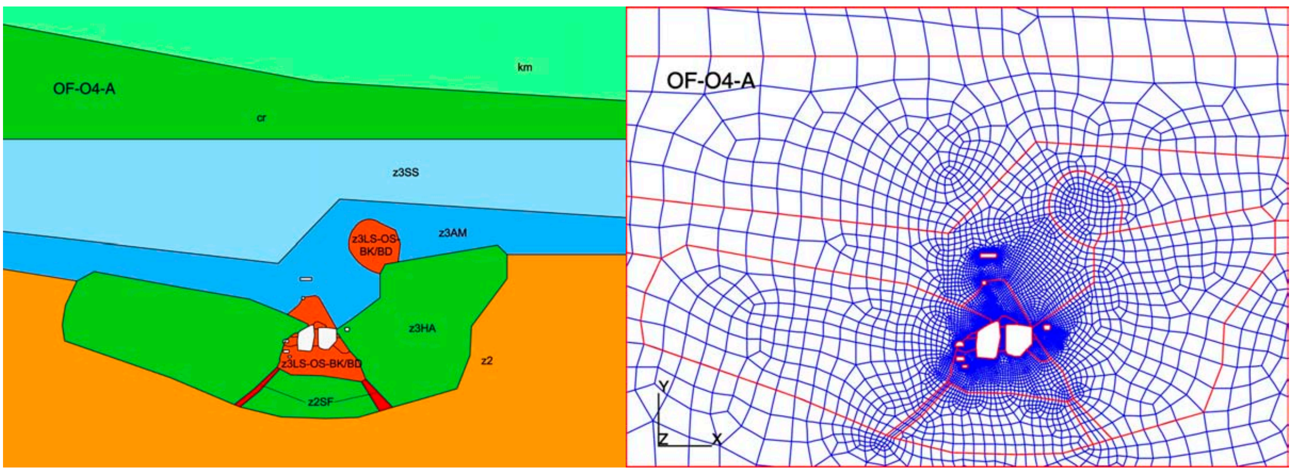

Fig. 3: Eastern part: Idealized geomechanical model (left) and part of FE mesh (right)

3.2 Western part

For the western part, the main units of the Zechstein strata (salt layers z2HS, z2SF, z2SF-UE, and anhydrite layers z3HA) and composites of the main units (z3-z4) are considered. The structure of the overburden was idealized taking into account the main layers caprock cr and Middle Keuper km (Fig. 4, left).

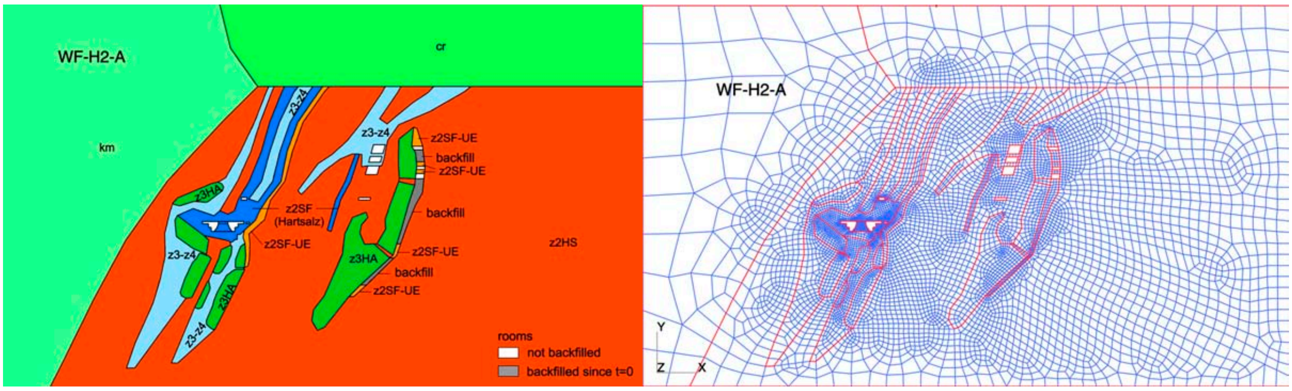

Fig. 4: Western part: Idealized geomechanical model (left) and entire FE mesh (right)

Finite-element modelling of the western part included the idealized structure of the geological layers and the geometry of the old mining rooms considering the two disposal rooms (left part of the structure) and the rooms of Lager B (right part of the structure). Fig. 4 (right) shows a plot of the entire 2-D model, 650 m in height and 1100 m wide. The upper boundary of the model has a distance of about 130 m to the ground surface.

3.3 Southern part

For the southern part, the main units of the Zechstein strata (salt layers z2HSO, z2HSB, z2HSW, z2W, z2SF, z3O, z3LS, z3AM, and anhydrite layers z1WA, z3HA) and composites of the main units (z3OS-BK/BD, z3-z4) are considered. The structure of the overburden was idealized regarding the main layers caprock cr, Keuper-Jurassic k-j, and Quaternary q (Fig. 4, left).

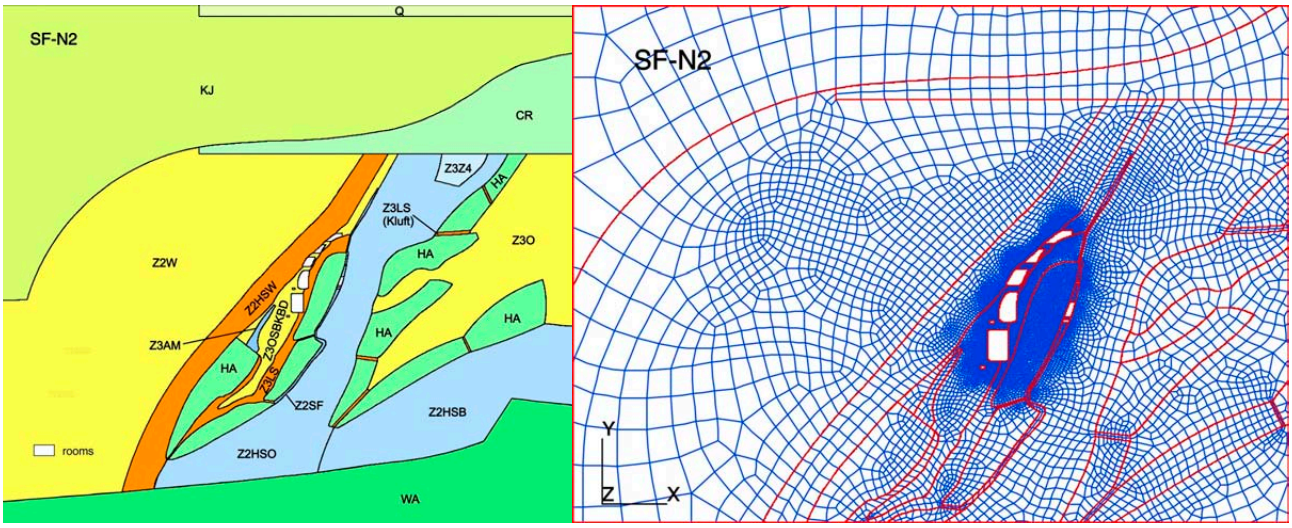

Fig. 5: Southern part: Idealized geomechanical model (left), part of FE mesh (right)

Fig. 5 (right) depicts a part of the finite-element model of the southern part. The idealized structure of the geological layers and the geometry of the old mining rooms were considered. The entire 2-D model is 800 m in height and 1000 m wide. The upper boundary of the model has a distance of about 120 m to the ground surface.

4 Material Behaviour

4.1 Creep

The idealized geological layers were classified with regard to their material behaviour. Anhydrite layers and the overburden were assumed to be elastic, the ductile rock salt layers were classified with respect to creep behaviour in terms of the steady-state creep rate. For the anhydrite layers, the modulus of elasticity was determined in laboratory tests from the post-failure stage to take the reduced stiffness of the jointed anhydrite blocks into account. Thus, a modulus of 30 GPa was used. The deformation behaviour of the ductile rock salt layers was described by a constitutive equation including both elastic and steady-state creep deformation. Additionally, the dilatant behaviour of rock salt was considered by a viscoplastic constitutive model. According to Hunsche & Schulze (1994), the effective steady-state creep rate can be calculated using:

$$\dot{\varepsilon}_{eff}^{cr} = A_{cr} \cdot e^{-\frac{Q}{RT}} \cdot \left(\frac{\sigma_{eff}}{\sigma^*}\right)^n \qquad (1)$$

with R = universal gas constant (8,3143 · 10^{-3} kJ/mol · K), T = temperature (K), σ_{eff} = effective stress (MPa), σ^* = reference stress (1,0 MPa) and the material parameters A_{cr} = structural factor (1/d), n = stress exponent (-), Q = activation energy (54,0 kJ/mol). The several types of rock salt layers mainly differ with respect to the structural creep factor A_{cr}. Thus, the creep capability of the layers can be taken into account using a factor A* related to the reference value A_0 = 0,18 1/d:

$$A_{cr} = A^* \cdot A_0 \qquad (2)$$

4.2 Dilatancy

A viscoplastic constitutive model was used to calculate dilatant deformations of the rock salt layers. Viscoplastic flow must be considered if the stress state exceeds the dilatancy boundary. Here, the dilatancy boundary is assumed to coincide with the yield function F which is described by a modified Drucker-Prager yield criterion:

$$F = \alpha \cdot J_1 + \sqrt{J_2^D} - k \qquad (3)$$

with J_1 = 1. invariant of stress tensor (MPa), J_2^D = 2. invariant of stress deviator (MPa^2), α = fictive angle of friction (-), k = fictive cohesion (MPa). The parameters α and k were determined to α = 0,2887 and k = 0. To determine the viscoplastic strain an associated flow rule is used:

$$\dot{\varepsilon}_{ij}^{vp} = \frac{1}{\eta} \cdot \langle F \rangle \cdot \frac{\partial \tilde{Q}}{\partial \sigma_{ij}}; \quad (\tilde{Q}=F) \qquad (4)$$

with η = viscosity [MPa·d], $\tilde{Q}$ = stress potential [MPa], and

$$\langle F \rangle = \begin{cases} 0, & \text{if } F < 0 \\ F, & \text{if } F \geq 0 \end{cases} \quad [MPa]$$

5 Model Calculations

The finite-element models of the several disposal parts of the Morsleben repository were established on the basis of the idealized structure of the geological layers (Fig. 1) and the geometry of the old mining rooms (Fig. 2). For reasons of simplification, two-dimensional plane-strain models were used in spite of the distinct three-dimensional development of stresses and strains around the rooms. This is valid since the rooms, especially in the southern and eastern parts, have a considerable length in the normal direction and the stress and deformation state calculated in a 2-D model is more unfavourable and „conservative“ with respect to the assessment of the stability of the structure and the integrity of the salt barrier. Generally, reference models taking into account best-estimate parameters for the salt rock were used for all disposal parts. Additionally, a couple of model variations were calculated to consider different material behaviour or structural configurations.

Calculations were performed using the well proven and released finite-element code ANSALT developed by BGR (NIPP, 1991). Pre- and postprocessing of the data was carried out with the INCA/PATRAN tool. To analyse the stability and integrity of the structure the required data were calculated, e.g. deviatoric stresses, tensile stresses, effective strains, and displacements.

5.1 Eastern part

Fig. 6 (left) shows the distribution of deviatoric stresses around the rooms of the eastern part at time t = 100 years after excavation. Medium to low deviatoric stress values occur in the salt rock around and above the rooms. For the far field of the salt layers, low to negligible stress values occur. The highest stress values are obtained in wide parts of the surrounding anhydrite layers. This is caused by the creep of the salt structure, stress redistribution and stress relaxation around the rooms as well as subsequent accumulation of stress in the anhydrite.

Considering the calculated effective strain (Fig. 6, right), wide parts of the salt layers are subjected to very low to negligible strain values up to 0.25%. As expected, highest strain values up to about 3% were obtained for the near field around the rooms, especially for the pillar between the two disposal rooms.

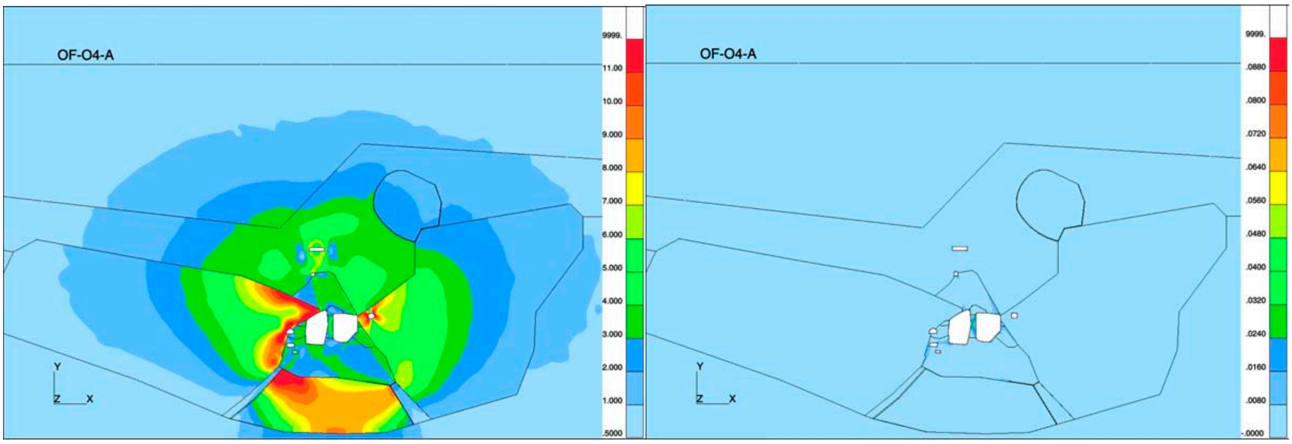

Fig. 6: Eastern part: Calculated deviatoric stress (left) and effective strain (right)

5.2 Western part

Fig. 7 (left) depicts the distribution of calculated deviatoric stresses around the rooms of the western part at time t = 100 years after excavation. Stress redistribution around the rooms takes place in wide areas of the salt rock caused by different creep properties of the several salt layers and by several stiff anhydrite layers distributed around the rooms. Stress redistribution reaches up to the overburden and country rock, but show only low deviatoric stress values of up to 5 MPa. Similar to the results obtained for the eastern part, the highest stress values are obtained in wide parts of the anhydrite layers. This is caused again by the creep of the salt structure, stress redistribution and stress relaxation around the rooms as well as subsequent accumulation of stress in the anhydrite. The increase of stress values in the anhydrite layers is uncritical with respect to the high degree of strength of the anhydrite.

Considering the calculated effective strain (Fig. 7, right), wide parts of the salt layers are sub-jected to very low to negligible strain values. Highest strain values of about 3% were calculated for the near field around the two disposal rooms and the rooms of Lager B.

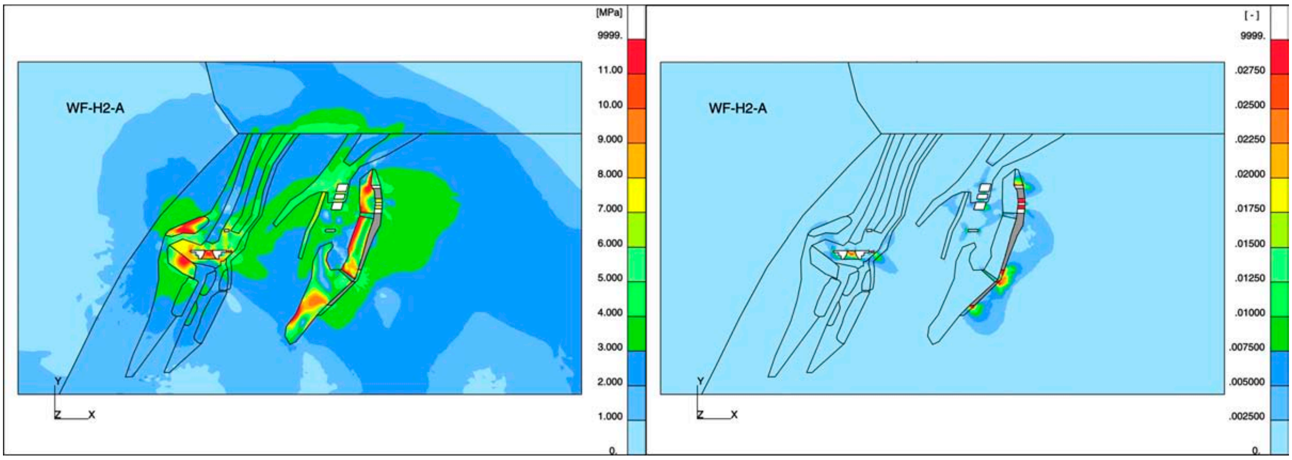

Fig. 7: Western part: Calculated deviatoric stress (left) and effective strain (right)

5.3 Southern part

The mining situation of the southern part is marked by several rooms located in at different levels of the mine. According to the pronounced inclination of the salt layers the excavation rows are arranged in a steep configuration forming several roofs between the rooms. Since these roofs are subjected to large deformations with subsequent fracturing, modelling of the mining situation covered two different cases. In a first step, a reference model was established assuming intact roofs. A second model took a total loss of bearing capacity of the roofs into account.

5.3.1 Reference model

In Fig. 8 (left), the distribution of calculated deviatoric stresses around the rooms of the southern part at time t = 100 years after excavation is plotted. Stress redistribution around the rooms occurs in wide areas of the salt rock caused by the geometrical configuration of the rooms and by several stiff anhydrite layers distributed around the rooms. This redistribution reaches up to the overburden (caprock) showing deviatoric stress values of up to 7 MPa. The highest stress values are obtained in wide parts of the anhydrite layers (up to about 16 MPa) and in the roofs between the rooms (up to 10 MPa). This is caused again by the creep of the salt structure and stress redistribution around the rooms as well as subsequent accumulation of stress in the anhydrite.

Considering the calculated effective strain (Fig. 8, right), wide parts of the salt layers are subjected to very low to negligible strain values of 0.8%. Highest strain values were calculated for the roofs between the rooms reaching a considerable amount of about 8%.

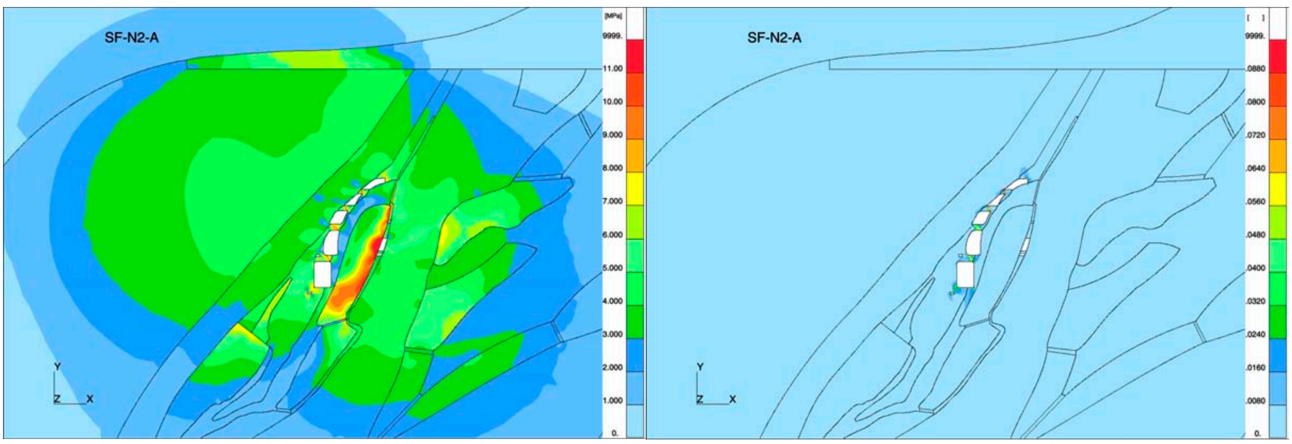

Fig. 8: Southern part: Calculated deviatoric stress (left) and effective strain (right)

5.3.2 Neglect of roof bearing capacity

Neglecting the bearing capacity of the roofs between the rooms, a reasonable increase of calculated stress and strain is obtained. Fig. 9 depicts the distribution of deviatoric stresses (left) and effective strains (right) at time t = 100 years after excavation. As expected, the stress redistribution in the salt structure is very similar to the results obtained with the reference model.

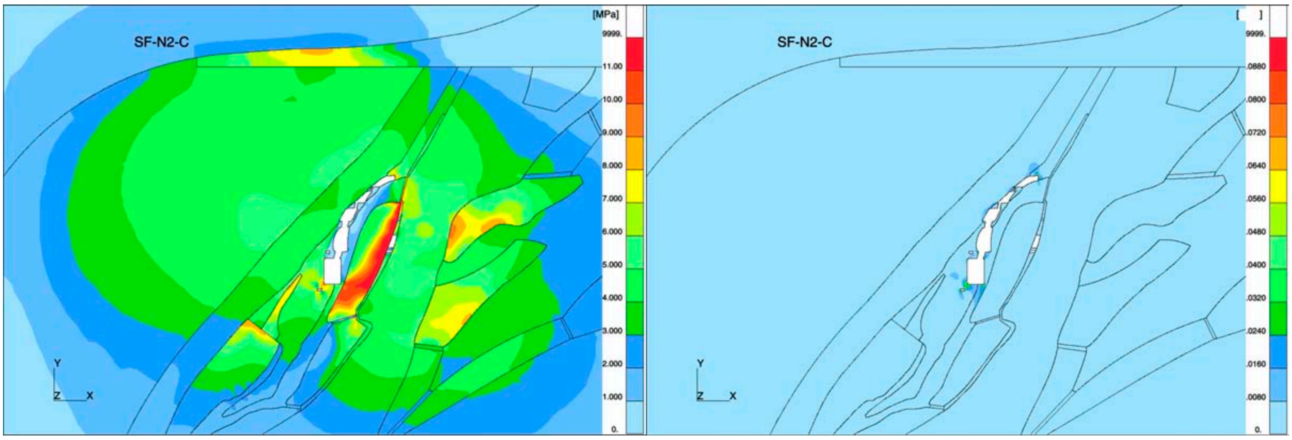

Fig. 9: Southern part without roof bearing capacity: Calculated deviatoric stress (left) and effective strain (right)

6 Analysis of Salt Barrier Integrity

6.1 Criteria

To analyse the integrity of the salt rock barrier from a geomechanical point of view, the following criteria were used (Langer & Heusermann, 2001):

- Dilatancy criterion (Fig. 10): The geomechanical integrity of the barrier is guaranteed if rock stresses do not exceed the dilatancy boundary; if this boundary is exceeded, micro-cracks will form and will cause progressive damage and increasing permeability of the salt rock.
- Hydraulic criterion (Fig. 11): The geomechanical integrity of the barrier is guaranteed if the hydrostatic pressure of an assumed column of brine extending to the ground surface does not exceed the minimum principal rock stress at the considered location of the salt body contour (e.g. top of the salt structure, contact area between salt and anhydrite blocks connected hydraulically to the overburden).

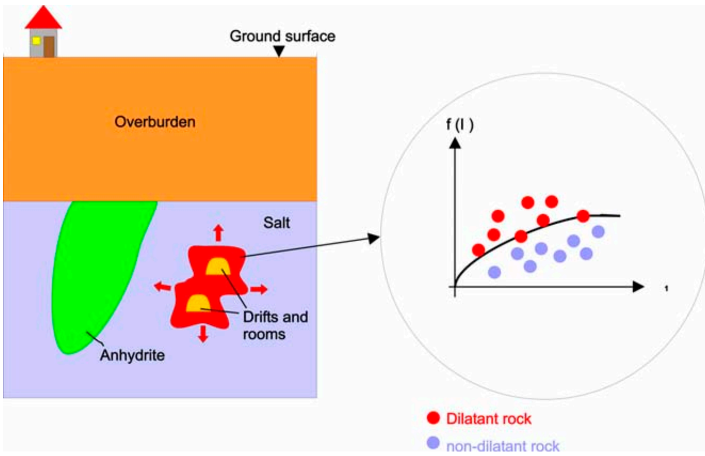

Fig. 10: Illustration of the dilatancy criterion

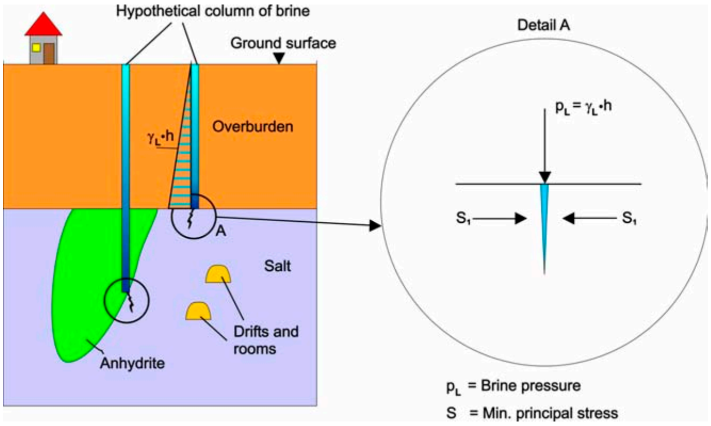

Fig. 11: Illustration of the hydraulic criterion

6.2 Eastern part

The dilatant rock zones of the eastern part calculated for a time elapse of 100 years after excavation are plotted in Fig. 12 (left). Excavation effects and creep of the salt rock cause the development of dilatancy around all rooms, as expected. Wide parts of the salt barrier between the disposal rooms and the top of the salt dome show no dilatancy. The development of dilatant rock zones with time is very slow.

Regarding the hydraulic criterion, the calculations yielded zones around the rooms of the eastern part with a distinct reduction of the minimum principal stress. From a very hypothetical point of view, the stress conditions appear to be unfavourable with respect to a fictive brine pressure which exceeds the minimum principal stress in the salt rock between the rooms and the anhydrite blocks (Fig. 12, right, yellow to red coloured areas). Since no hydraulic connection between the anhydrite and the overburden exists and wide parts of the salt barrier show neither dilatancy nor hypothetical exposure to brine induced fracturing, the integrity of the salt barrier in the eastern part is guaranteed.

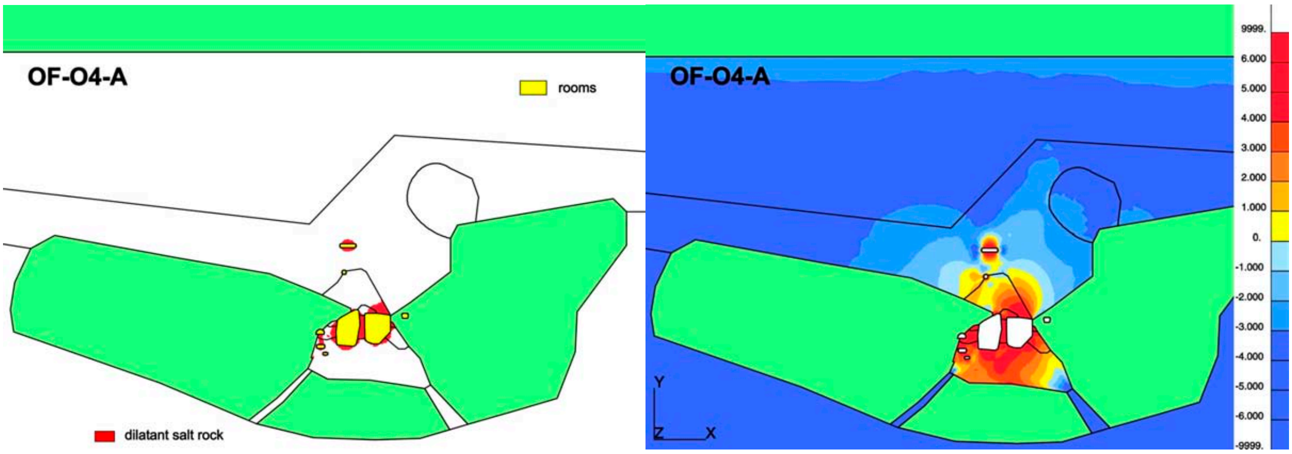

Fig. 12: Eastern part: Dilatant rock zones (left) and hypothetical rock zones affected hydraulically (right)

6.3 Western part

Fig. 13 (left) shows the dilatant rock zones of the western part calculated for a time elapse of 100 years after excavation. Excavation effects and creep of the salt rock cause the development of dilatancy around all rooms, as expected. On the opposite, wide parts of the salt barrier between the rooms and the top of the salt dome as well as the country rock show no dilatancy.

Fig. 13 (right) depicts the zones around the rooms of the western part with a distinct reduction of the minimum principal stress indicating a hypothetical exposure to brine induced fracturing (yellow to red coloured areas). Since no hydraulic connection between the anhydrite and the overburden exists and wide parts of the salt barrier show neither dilatancy nor hypothetical exposure to brine induced fracturing, the integrity of the salt barrier in the western part is guaranteed.

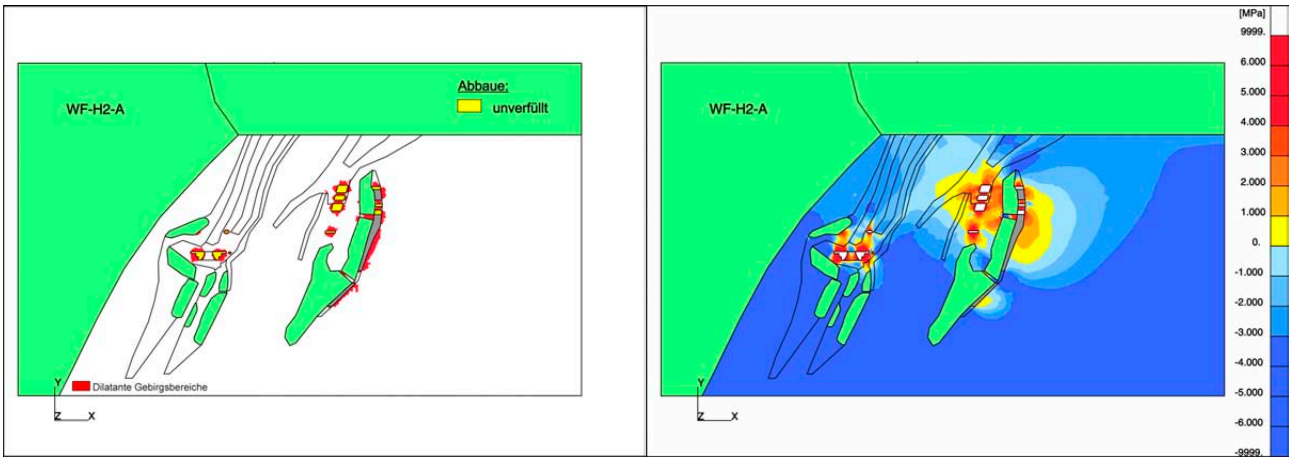

Fig. 13: Western part: Dilatant rock zones (left) and hypothetical rock zones affected hydraulically (right)

6.4 Southern part

The dilatant rock zones of the southern part calculated for a time elapse of 100 years after excavation are plotted in Fig. 14 (left). Excavation effects and creep of the salt rock cause the development of dilatancy around all rooms, as expected. Wide parts of the salt barrier between the disposal rooms and the top of the salt dome show no dilatancy.

Regarding the hydraulic criterion, zones around the rooms of the southern part with a distinct reduction of the minimum principal stress are obtained (yellow to red coloured areas). From a very hypothetical point of view, the stress conditions appear to be unfavourable with respect to a fictive brine pressure which exceeds the minimum principal stress in the salt rock around the rooms as well as between the anhydrite layers and the rooms. Since no hydraulic connection between the anhydrite and the overburden exists and wide parts of the salt barrier show neither dilatancy nor hypothetical exposure to brine induced fracturing, the integrity of the salt barrier in the southern part is guaranteed.

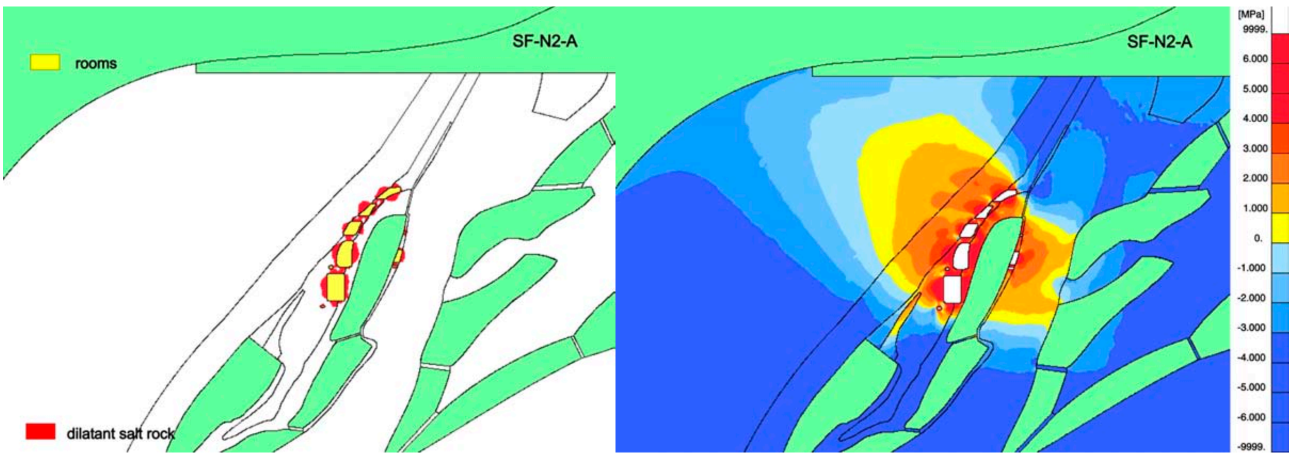

Fig. 14: Southern part: Dilatant rock zones (left) and hypothetical rock zones affected hydraulically (right)

If the bearing capacity of all roofs between the disposal rooms is neglected, wider parts of the salt rock show dilatancy and hypothetical exposure to brine induced fracturing (Fig. 15). Nevertheless, a sufficient area of the salt barrier, especially at the top of the salt dome, shows again no dilatancy and no critical stresses with regard to the hydraulic criterion. Thus, the integrity of the salt barrier in the southern part is guaranteed for this conservative case too.

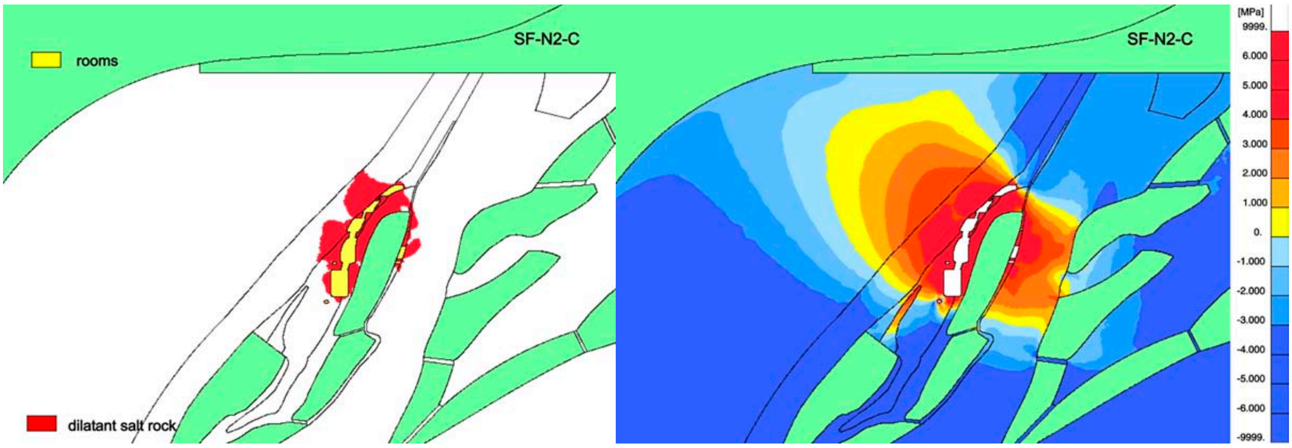

Fig. 15: Southern part without roof bearing capacity: Dilatant rock zones (left) and hypo-thetical rock zones affected hydraulically (right)

7 Conclusions

To assess the integrity of the salt barrier in the disposal areas of the Morsleben repository (i.e. eastern, western, and southern part of the Bartensleben mine), geomechanical model calculations have been carried out. Two-dimensional models of characteristic cross section of these mine parts were established based on idealized models of the geological structure. The calculations comprised the analysis of the recent stress and strain state of the salt barrier and the evaluation of the barrier integrity taking a dilatancy criterion and a hydraulic criterion into account.

The following results could be obtained by means of finite-element modelling:

- As expected, dilatant rock zones occur in the near field around the rooms of all disposal areas. Additionally, the salt rock between the rooms and several anhydrite layers nearby is subjected to dilatancy. Sufficient areas of the salt barrier show no dilatancy.
- Regarding the hydraulic criterion, zones around the rooms of all disposal areas with a distinct reduction of the minimum principal stress were obtained indicating a hypothetical exposure to fracturing induced by brine from the overburden.

Since wide parts of the salt barrier, especially at the top of the barrier, in the eastern, western, and southern part of the repository show no dilatancy and no hypothetical exposure to brine induced fracturing and no hydraulic connection between the anhydrite layers and the overburden exists, the geomechanical integrity of all disposal areas is guaranteed.

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Three Dimensional Analysis of Combined Gas, Heat and Nuclide Transport in a Repository in Rock Salt Considering Coupled Thermo-Hydro-Geomechanical Processes

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Abstract
To study the coupled thermo-hydrologic-geomechanical processes and their influence on gas and nuclide transport in a two phase flow configuration in a porous medium, a coupling of the thermo-hydrodynamic code TOUGH2 and the thermo-mechanic code FLAC3D is described and applied to analyze three dimensional gas, heat and nuclide transport in a repository for heat generating nuclear waste in rock salt. According to stress dependent hydrological properties, like, porosity, permeability and capillary pressure, the influence of coupled processes on a two phase flow can be relevant. The present procedure can be applied to quantify safety margin related to hydro-fracturing and dilatancy due to fluid pressure build-up and to define bounding analyses, if the hydrological properties, such as porosity, permeability, and capillary pressure, depending on mean effective normal stress are used.

1 Introduction

To assess the long term safety of a repository for heat generating nuclear waste in a deep geological rock formation, often groundwater flow into the repository is postulated. The water can react with the radioactive waste or with its containers and can gradually disassemble them. The radioactive substances after being dissolved in liquid phase can be transported out of the repository and subsequently can be released into the geosphere. The heat and fluid transport can be enhanced significantly by gas generation, mainly hydrogen due to corrosion of metallic materials in the repository. The two phase flow, pressure build-up due to gas generation and heat transport can influence substantially the mechanical behavior of the filling and sealing materials and host rock. The transient stress situation, especially, if the fluid pressure approaches the lithostatic pressure, can reactivate an unfavorably oriented fault or can lead to new fracturing. This can in turn affect the hydrological properties, like porosity, permeability and capillary pressure.

To study the coupled thermo-hydrologic-mechanical (THM) processes and their influence on gas and nuclide transport in a two phase flow configuration, the thermo-hydrological code TOUGH2 and the thermo-mechanical code FLAC3D are coupled. In GRS, TOUGH2 has been modified to investigate gas, heat and nuclide transport under various conditions. For instance, in [1], three dimensional gas, heat and nuclide transport in a repository in rock salt is analyzed with TOUGH2 considering porosity and permeability of crushed salt depending on pressure, temperature and rock convergence without considering mechanical effects. FLAC3D is developed for rock and soil mechanics and can treat some THM features for a single phase flow but not for two phase flow situation [2]. In [3], TOUGH2 and FLAC3D are coupled linearly to study gas and nuclide transport in a three dimensional isothermal system. In this paper, which extends the analyses of [3], the two codes are coupled sequentially to analyze a three dimensional non-isothermal system as proposed in [4].

2 Thermo-Hydrological Analysis with TOUGH2

To perform analysis with TOUGH2, which is a recognized tool to study three dimensional coupled fluid and heat transport of two-phase multi-component fluid mixtures in porous media, one of the several equation-of-state (EOS) modules must be linked with the main code of TOUGH2. The number and the properties of the fluid components are determined by the EOS module selected. The mass balance equation of each fluid component with contributions from gas and liquid phase includes advection and dispersion in both phases. In the heat balance equation, convection and conduction are considered. The velocities of gas and liquid phase are determined by multiphase extension of Darcy's law considering relative permeability of each phase depending upon liquid saturation [5]. Usually, the first fluid component is groundwater. In the equation of state module EOS7, which is used here, the second fluid component can be solute or denser salt water [6]. The third fluid component is a soluble gas (air or hydrogen). The complete steam table is used to calculate the liquid properties as a function of pressure, temperature and mass fraction of solute or salt water. The gas phase is treated as a mixture of ideal gases such as air or hydrogen and vapor.

For numerical simulation the region to be modeled is discretized into volume elements. The conservation equations are solved simultaneously with the integral finite difference method using Newton-Raphson iteration scheme. Time is discretized fully implicitly as a first order backward finite difference. This together with full upstream weighting of fluxes at the element interfaces can avoid time step limitations, mainly when a phase appears or disappears. The scalar quantities like pressure and temperature are determined at the center of the elements and the vector quantities such as velocities, mass and heat fluxes at the element interfaces.

3 Mechanical Analysis with FLAC3D

FLAC3D is an established three dimensional explicit finite difference code for engineering mechanics and can describe the behavior of structures built of soil, rock or materials that undergo plastic flow when their yield limits are reached. Materials are represented by polyhedral elements within a three dimensional grid that is adjusted by the user to fit the shape of the object to be modeled. Each element behaves according to a prescribed linear or non linear stress/strain law in response to applied forces or boundary restrains. To analyze coupled processes, fluid pressure and temperature can be prescribed by invoking the fluid and thermal configuration and the mean effective normal stress can be calculated as [4; 8]:

$\sigma_{mean} = (1/3)(\sigma_1 + \sigma_2 + \sigma_3)$, $\sigma_{mean,eff} = \sigma_{mean} + (p - \psi p_{cap})$.

The Bishop factor ψ, which depends on soil properties and should be derived from site specific data, approaches unity for saturated soil and zero for dry soil. Here, to simplify the analysis, ø is replaced by liquid saturation ($\psi = S_{Liquid}$). To study THM processes, it is to be noted that, total stresses, pressure and temperature are computed at the grid points (corner nodes of a zone) and principal stresses and average pressure at the center of a zone.

4 Sequential Coupling of TOUGH2 and FLAC3D

The original version of TOUGH2 with EOS7 is modified to couple FLAC3D sequentially. An identical numerical grid with equal numbers of FLAC3D zones and TOUGH2 elements is required for the coupling (Fig. 1). In the sequential procedure, the code TOUGH2 is executed as a 'main program' to perform thermo-hydrological analysis and the FLAC3D as a 'subroutine' to conduct a quasi static mechanical analysis (Fig. 2). It can be expedient to run the codes in an environment of the same operating system to ease the data transfer between the codes. In the beginning, the initial fluid pressure and temperature distributions are computed by TOUGH2 and transferred to FLAC3D. Since the TOUGH2 mesh uses one centre point within an element to determine pressure and temperature in element and since the FLAC3D grid points for temperature and pressure are located at the corners of the elements, data have to be interpolated from the mid-element of TOUGH2 to the corner grid points of FLAC3D. The volume averaged pressure and temperature at all grid points of FLAC3D can be determined as:

$p_n = [\Sigma p_k V_k] / \Sigma V_k$, $T_n = [\Sigma T_k V_k] / \Sigma V_k$.

This is a one possible scheme to compute the pressure and temperature at the grid points. Depending on problem, other schemes should also be considered. Subsequently, the initial stress distribution is calculated and a restart file is created. During the transient analysis, at the end of each time step, TOUGH2 calls FLAC3D and the current fluid and temperature distributions are transferred to FLAC3D. In FLAC3D, the new strain and stress distribution are computed by employing the previous restart file and the current fluid and temperature distributions of TOUGH2. At the end of the FLAC3D run, a new restart file is created and the current stress distributions are transferred back to TOUGH2. Since the stresses are determined at the centre of a zone, they can directly be allocated to the corresponding elements of TOUGH2. To continue the two phase flow and heat and nuclide transport analysis with TOUGH2, the hydrological properties, such as porosity, permeability and capillary pressure for all elements are determined basically as a function of mean effective stress at each time step. Usually, these coupling functions can be non-linear and should be determined by using site specific data. For instance, following functions are adopted:

$\Delta\sigma_{mean,eff}(t) = \sigma_{mean,eff}(t) - \sigma_{mean,eff}(t=0)$, $\varphi_0 = \varphi(t = 0)$, $\varphi_{aux} = \varphi_0 \exp(a\Delta\sigma_{mean,eff})$.

Since porosity is described as a function of stress, which is in turn a function of pressure, the compressibility (TOUGH2 parameter) of the porous medium can be determined:

$\beta = (1/\varphi_{aux})[\Delta\varphi_{aux}(t)/\Delta p(t)]$.

This varying compressibility is then inserted in the conservation equations of the hydrological analysis to compute the porosity as a function of pressure and subsequently other properties:

$\Delta\varphi(t) = \beta\varphi(t)\Delta p(t)$, $k_0 = k(t = 0)$, $k = c\varphi^b$, $c \sim k_0$,

$p_{cap} = p_{Gas} - p_{Liquid} = p_{cap}(S_{Liquid})[(k_0\varphi)/(k\varphi_0)]^{1/2}$,   a, b, c: constant parameters.

Thus, the main state variables, pressure, temperature and stress are exchanged between the codes at each time step. The permeability and the compressibility of the porous media are kept constant during a time step. Depending on problem, other coupling functions can also be considered. This sequential method, in which the two codes are coupled at the end of each time step, can be improved and verified by an implicit iterative sequential coupling, in which the two codes are coupled at every Newtonian (physical) iteration within a time step. But, the iterative sequential coupling can increase the computational effort significantly.

5 Thermo-Hydro-Mechanical (THM) Analysis

A three dimensional simplified model of a non-isothermal repository system is considered to demonstrate the sequential coupling of TOUGH2 and FLAC3D (Fig. 3 and Table 1). The model with reasonable parameters consists of four material domains: upper 400 m as a barrier rock, lower 200 m rock salt as a host rock, a 10 m high repository within the host rock at the bottom and an excavation damaged zone (EDZ) around the repository. Initially, the complete system, except the repository, is flooded with groundwater. A transient temperature boundary condition at the bottom surface of the repository is introduced to simulate decay heat generation. To simplify the analysis, the radioactive substances in the repository are simulated by a single non decaying solute. The uniform gas generation in the repository is represented by hydrogen formation rate in three time segments [3]:

$0 \leqslant t \leqslant 1000$ years: linear increase from 0 to Q_{Gas},

$1000 \leqslant t \leqslant 5000$ years: $Q_{Gas} = 45$ kg/year,

$5000 \leqslant t \leqslant 6000$ years: linear decrease from Q_{Gas} to 0.

The fluid consists of three components: ground water, solute in liquid phase and hydrogen. The solubility of hydrogen in liquid phase is given by:

$X_{Gas\ in\ Liquid} = m_{Gas}/m_{Liquid} = (p/C_{Henry})(M_{Gas}/M_{Liquid})$.

To determine the relative permeabilities and the capillary pressure, the Van Genuchten functions are applied [5]:

$S_{Liq,Eff} = (S_{Liquid} - S_{Liq,\ Res}) / (1 - S_{Liq,Res})$, $S_{Gas,Res} = 0$, $k_{Liq} = kk_{Liq,Rel}$, $k_{Gas} = kk_{Gas,Rel}$,

$k_{Liq,Rel} = (S_{Liq,Eff})^{1/2}\{1 - [1 - (S_{Liq,Eff})^{1/\lambda}]^{\lambda}\}^2$, $k_{Gas,Rel} = (1 - S_{Liq,Eff})^{1/3}[1 - (S_{Liq,Eff})^{1/\lambda}]^{2\lambda}$,

$p_{cap} = p_b\,[(S_{Liq,Eff})^{-1/\lambda} - 1]^{(1-\lambda)}$, $\lambda = 0.77$, $p_b = 0.56\,(k_0)^{-0.346}$, k in m^2, p_b in Pa.

To determine the spatial stress conditions with FLAC3D, an isotropic elastic material for the cap rock (barrier rock) and repository is assumed. The rock salt (host rock) and EDZ are considered to be elastic visco-plastic materials and their behaviour is described by neglecting primary but including secondary creep rate:

$(d\varepsilon/dt)_{secondary} = D \exp[-A/\theta(t)]\,(\sigma^{dev} / \sigma_{ref})^5$, $\sigma^{dev} = [(3/2)\,\sigma_{ij,dev}\,\sigma_{ij,dev}]^{1/2}$, $\sigma_{ref} = 1$ MPa,

$\sigma_{ij,dev}$: deviatoric part of total stress σ_{ij}, D = 0.18 1/day, A = 6495 K.

Case THS31: In this reference case for the hydrological analysis with TOUGH2, the hydrological properties do not depend on stress ($\beta = 0$, φ = constant, k = constant).

Case THMS31: Sequential coupling: Using basically the same input data of case THS31, TOUGH2 and FLAC3D are executed sequentially at each time step considering porosity and permeability depending upon stress (for repository: $\beta = 0$, φ = constant, k = constant):

$\varphi_{Barrier,aux} = 0.05 \exp[5 \cdot 10^{-8} \text{ (1/Pa) } \Delta\sigma_{mean,eff}]$, $k_{Barrier} = (1.6 \cdot 10^{-14} \text{ m}^2)\ (\varphi_{Barrier})^4$,

$\varphi_{Host,aux} = 0.01 \exp[5 \cdot 10^{-8} \text{ (1/Pa) } \Delta\sigma_{mean,eff}]$, $k_{Host} = (1.0 \cdot 10^{-11} \text{ m}^2)\ (\varphi_{Host})^4$.

$\varphi_{EDZ,aux} = 0.02 \exp[5 \cdot 10^{-8} \text{ (1/Pa) } \Delta\sigma_{mean,eff}]$, $k_{EDZ} = (6.25 \cdot 10^{-11} \text{ m}^2)\ (\varphi_{EDZ})^4$.

To avoid numerical problems due to unrealistic values of the rock compressibility, especially, when $\Delta p(t)$ is too small between the consecutive time steps, the rock compressibility is not allowed to exceed a certain reasonable value, for instance, $\beta \leq 1{\cdot}10^{-7}$ 1/ Pa.

Case THS33: This bounding hydrological case is same as THS31, but now a constant rock compressibility $\beta = 2{\cdot}10^{-8}$ 1/Pa for the entire model, except repository, is applied, which is derived as a maximum value from the case THMS31. This implies $\varphi = \varphi(p)$ and $k \sim \varphi^4$.

These three cases are computed with the modified version of TOUGH2/EOS7 up to the problem time of 10^4 years with a maximum time step of $8{\cdot}10^8$ s. The temperature at the centre of the repository for the case THS31 with constant hydrological properties in Fig. 4 indicates that the maximum temperature of 148 C is reached at t = 240 years. Fig. 5 shows the pressure distribution of case THS31 in the boundary plane (y = 5 m) at t = 1000 years, around which the maximum value occurs. The gas generation and the decay heat influence the pressure distribution significantly leading to the maximum value of 14.7 MPa in the repository, about 2.7 MPa above the lithostatic level. The temperature distributions in two boundary planes (y = 5 m and y = 45 m) at t = 1000 years are not very different, as the width in the y-direction is not very large (Fig. 6 and 7). The distributions of pressure, temperature, gas saturation and solute mass fraction in liquid phase at t = 10^4 years are depicted in Fig. 8 to 11. At t = 10^4 years the pressure in the repository is still around 7.8 MPa, about 1.8 MPa higher than the hydrostatic level (Fig. 8). Due to large capillary pressure difference between the rock salt, the excavation damaged zone and the repository and due to increasing gas solubility with increasing pressure, more gas is released at the ends than at the centre of the repository. Due to relatively low permeability of the host rock, the gas saturation remains below 1 % beyond 250 m from the repository in first 10^4 years (Fig. 9). The decreasing heat generation and the increasing heat transport, mainly via heat conduction, away from the repository reduce the temperature level in the host rock significantly causing a maximum temperature of 34 C in the repository (Fig. 10). The solute (nuclide) with initial mass fraction of 0.3125 in liquid phase in the repository does not migrate vertically beyond 200 m from the repository within 10^4 years (Fig. 11).

Assuming that the hydro-fracturing due to pressure build-up in the rock can occur, if the fluid pressure reaches the minimum compressive principal stress, the factor of safety related to hydro-fracturing can be defined as a ratio:

F_{Frac} = |min. compressive principal stress| / p; $F_{Frac} = 0$, if any principal stress > 0 or $\sigma_{mean} > 0$.

In Fig. 12, the factor of safety related to hydro-fracturing in the boundary plane y = 5 m at t = 100 years is depicted, around which maximum pressure occurs in case THMS31. In the upper area of the model, the safety factor is little smaller than in the lower area, as the difference between the fluid pressure and the lithostatic pressure increases with depth. Postulating that the risk of hydro-fracturing is given for a safety factor below 1.3, mainly the region in the rock salt right above the repository is affected; in major part of the model, the hydro-fracturing is not expected. Additionally, to characterize the mechanical stability of the rock salt, the factor of safety related to dilatancy (increase of volume due to opening or widening of cracks) can be defined as a ratio of the dilatancy boundary, which is derived from the experimental observations in [7], to octahedral stress:

$F_{dil} = \tau_{dil} / \tau, \quad \tau = (1/3)\ [(\sigma_1 - \sigma_2)^2 + (\sigma_1 - \sigma_3)^2 + (\sigma_2 - \sigma_3)^2]^{1/2}$,

$\tau_{dil} = 0.8996|\sigma_{mean}| - 0.01697|\sigma_{mean}|^2$, ($\tau_{dil}$ and σ_{mean} in MPa),

$F_{dil} = 0$, if any principal stress > 0 or $\sigma_{mean} > 0$; $F_{dil} < 1$: Mechanical stability is affected.

As the factor of safety related to dilatancy lies far beyond 1, the mechanical stability of the entire rock salt can be expected (Fig. 13). The criterion for the hydro-fracturing is clearly stricter than the dilatancy criterion, since for the dilatancy criterion all principal stresses should only be negative but for the hydro-fracturing criterion all principal stresses should be sufficiently negative. For an integral comparison, the pressure in the repository and the nuclide migration from the repository are presented in Fig. 14 and 15. The pressure in the case THMS31 with coupled mechanical effects is substantially lower than in the case THS31 without mechanical effects. In all three cases, the nuclide migration from the repository is nearly same, around 66 % of the initial mass of 10^7 kg in 10^4 years, as the net effect of the driving pressure difference and the effective permeability is not very different. Yet, the higher permeability in the cases THMS31 and THS33 leads to a little higher release than in the case THS31. The net effect of the driving pressure difference and the effective permeability on the nuclide migration is difficult to estimate without detailed numerical analyses. According to the postulated stress-porosity-permeability relationship, the impact of the coupled processes is noticeable, which can be enveloped reasonably well by the limiting hydrological cases THS31 and THS33. In other situations with a more sensitive stress-porosity-permeability relationship, the impact of the coupled processes can be substantial.

6 Conclusions

To study the coupled thermo-hydro-mechanical (THM) processes and their influence on gas and nuclide transport in a two phase flow configuration in porous media, a sequential coupling of the thermo-hydrodynamic code TOUGH2 and the thermo-mechanic code FLAC3D is described and applied to study three dimensional gas, heat and nuclide transport in a repository for heat generating radioactive waste in rock salt. The scoping coupled THM analyses show that the transport behaviour of the contaminated two phase fluid is noticeably influenced by the transient stress conditions. The present method can be applied to quantify safety margin related to hydro-fracturing and dilatancy due to fluid pressure build-up and can help to define bounding analyses, if the hydrological properties, like porosity, permeability, and capillary pressure, depending on mean effective normal stress are employed.

Symbols

A: normalized activation energy [K]; k: permeability [m^2]; M: molecular weight [g/mol]; m: mass [kg]; p: pressure [Pa]; p_b: bubble entry pressure [Pa]; Q: mass flow [kg/s]; S: phase saturation; t: time [s]; T: temperature [C]; V: volume [m^3]; X: mass fraction; β: rock compressibility [1/Pa]; ε: strain; φ: porosity; θ: temperature [K]; σ: stress (tensile: > 0, compressive: < 0) [Pa]; σ_j: principal stress (FLAC3D: $\sigma_1 \leqslant \sigma_2 \leqslant \sigma_3$) [Pa]; τ: octahedral shear stress [Pa]; ψ: Bishop factor.

Subscripts: aux: auxiliary; i: index of a TOUGH2 element; k: index of a FLAC3D connected zone; n: index of a FLAC3D grid point; s: time step.

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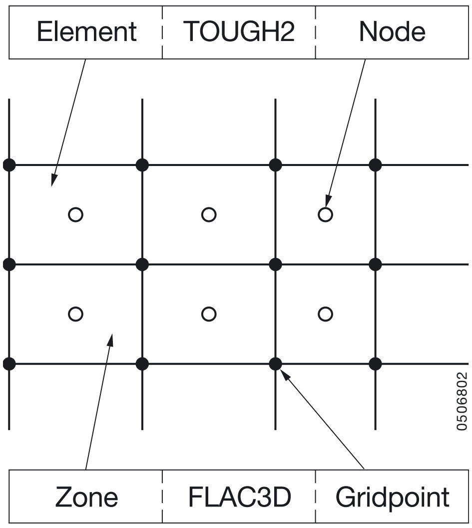

Fig. 1: Identical mesh for TOUGH2 and FLAC3D.

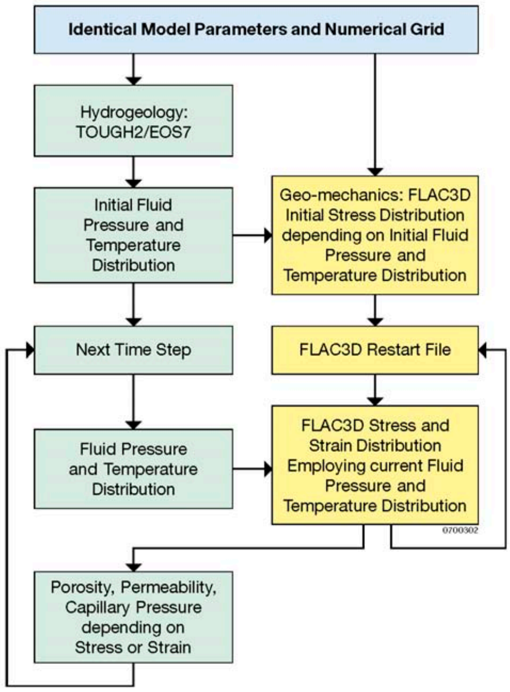

Fig. 2: Sequential coupling of TOUGH2 and FLAC3D.

Volume of repository	1E5 m^3
Density of liquid phase	$\rho_{Water}(p,T)$
Dynamic viscosity of liquid phase	$\mu_{Water}(p,T)$
Dynamic viscosity of hydrogen	8.95 E-6 Pas
Gas constant of hydrogen	4124 J/(kgK)
Molecular weight of liquid phase	18 g/mol
Molecular weight of hydrogen	2 g/mol
Henry constant for hydrogen, C_{Henry}	7.31E9 Pa
Mol. diffusion coefficient in liquid	5E-11 m^2/s
Porosity of barrier rock (cap rock)	0.05
Porosity of rock salt (host rock)	0.01
Permeability of barrier and host rock	1E-19 m^2
Porosity of repository	0.4
Permeability of repository	1E-12 m^2
Residual liquid saturation, $S_{Liq,Res}$	0.2
Residual gas saturation, $S_{Gas,Res}$	0.0
Compressibility of rock, $\beta = (1/\varphi)(\partial\varphi/\partial p)$	0
Thermal conductivity of total model	2 W/(mK)
Specific heat of total model	1000 J/(kgK)
Thermal expansion coeff. of total model	4E-5 1/K
Dry rock density of total model	2000 kg/m^3
Elastic bulk modulus of barrier rock	2E9 Pa
Elastic shear modulus of barrier rock	1.2E9 Pa
Elastic bulk modulus of repository	30E6 Pa
Elastic shear modulus of repository	2E6 Pa
Elastic bulk modulus of rock salt	18.12E9 Pa
Elastic shear modulus of rock salt	9.843E9 Pa

Table 1: Reference model parameters.

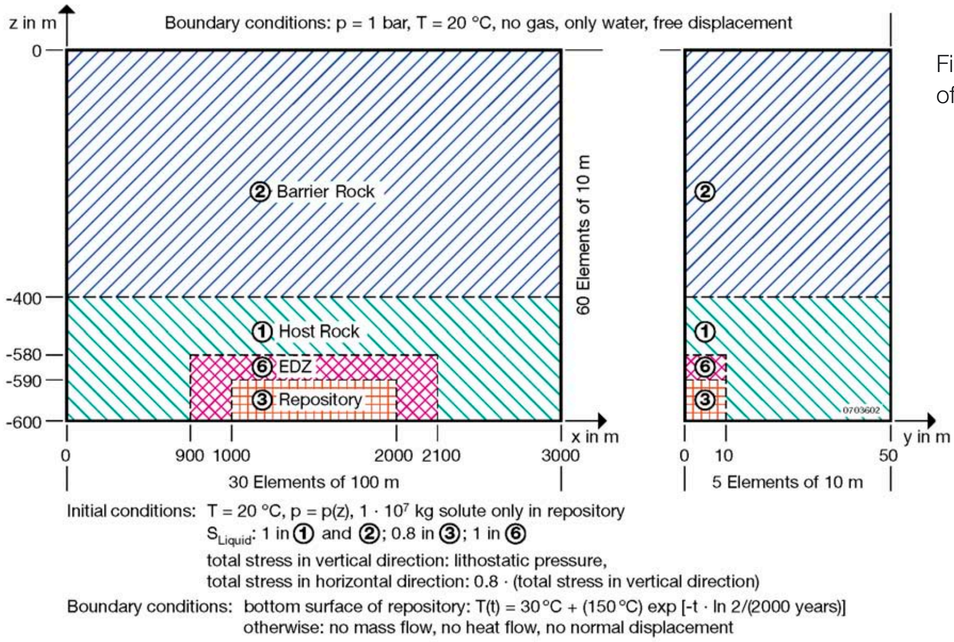

Fig. 3: Three dimensional model of a repository in rock salt.

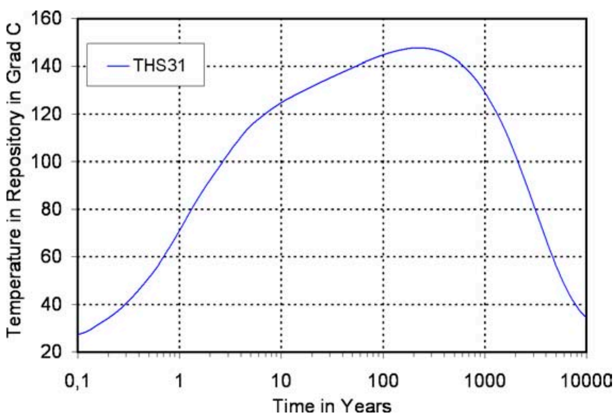

Fig. 4: Temperature in the centre of the repository in case THS31.

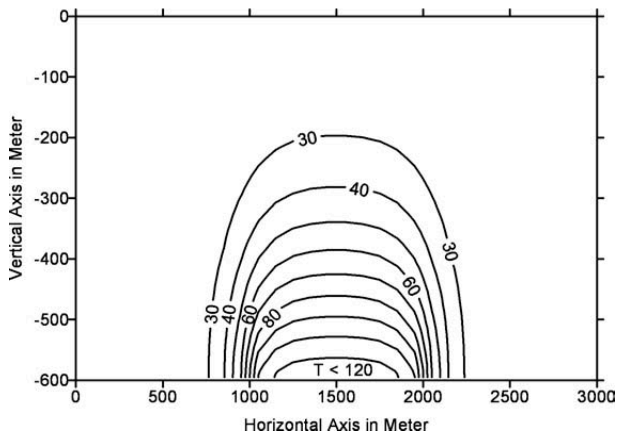

Fig. 7: Temperature (grad C) at y = 45 m and t = 1000 years in case THS31.

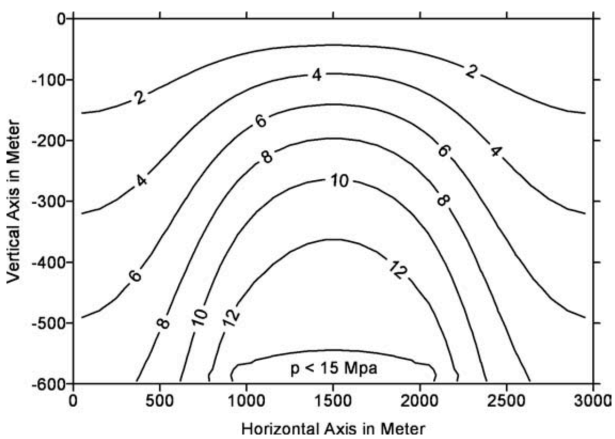

Fig. 5: Pressure (MPa) at y = 5 m and t = 1000 years in case THS31.

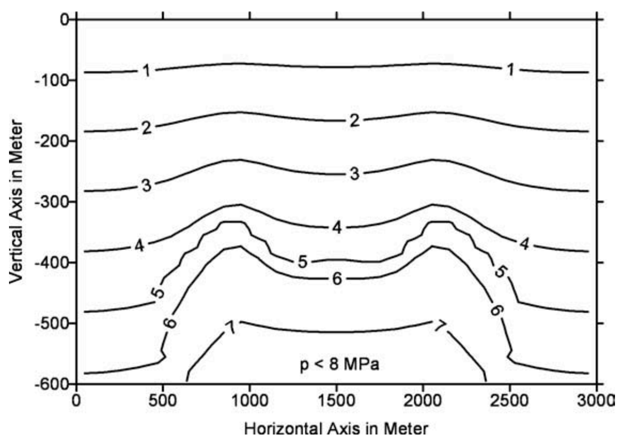

Fig. 8: Pressure (MPa) at y = 5 m and t = 10000 years in case THS31.

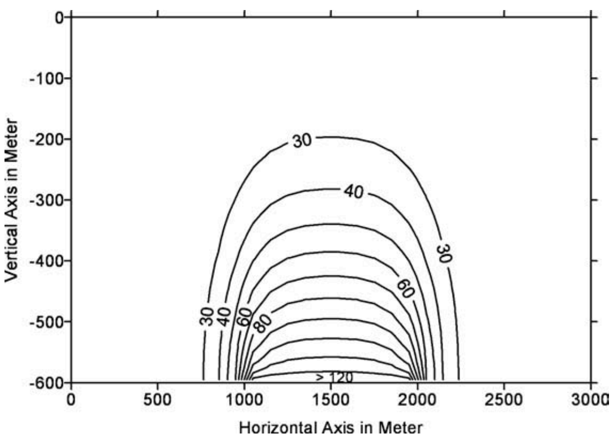

Fig. 6: Temperature (grad C) at y = 5 m and t = 1000 years in case THS31.

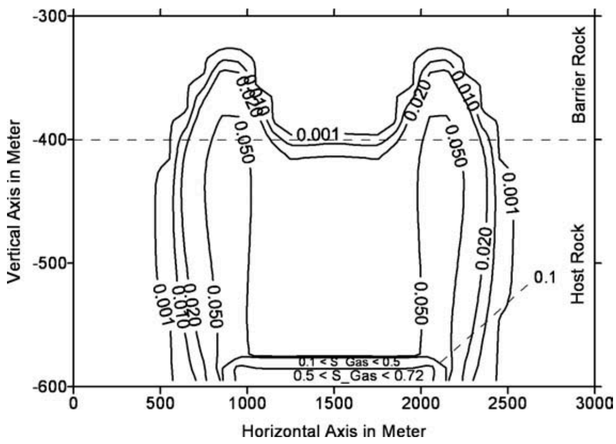

Fig. 9: Gas saturation at y = 5 m and t = 10000 years in case THS31.

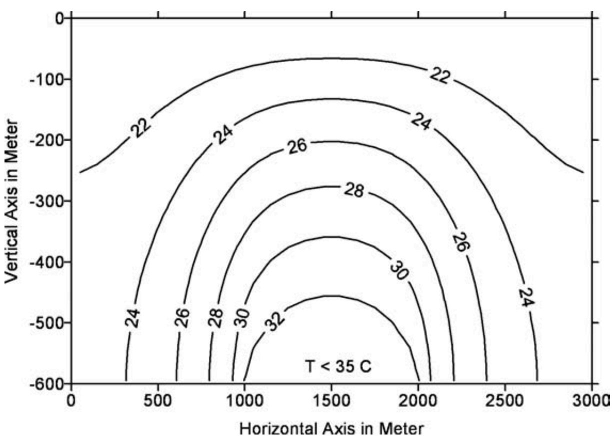

Fig. 10: Temperature (grad C) at y = 5 m and t = 10000 years in case THS31.

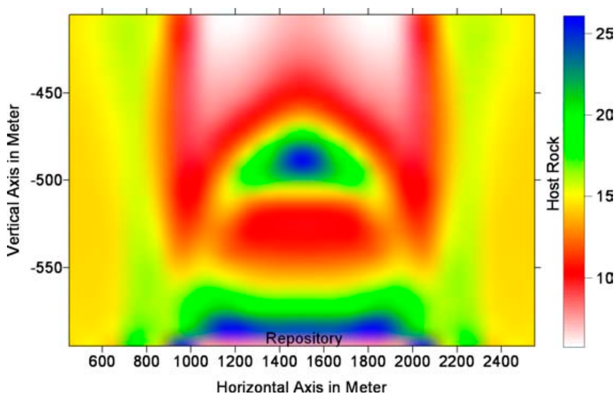

Fig. 13: Factor of safety regarding dilatancy at y = 5 m and t = 100 years case THMS31.

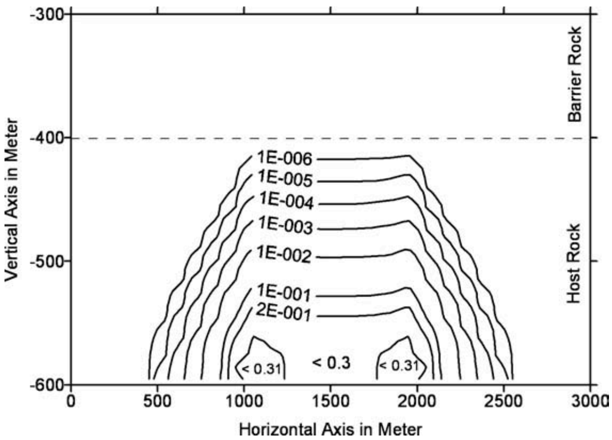

Fig. 11: Nuclide mass fraction at y = 5 m and t = 10000 years in case THS31.

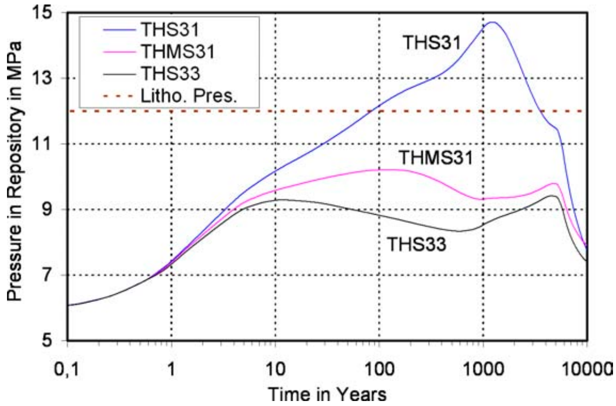

Fig. 14: Pressure at the centre of repository.

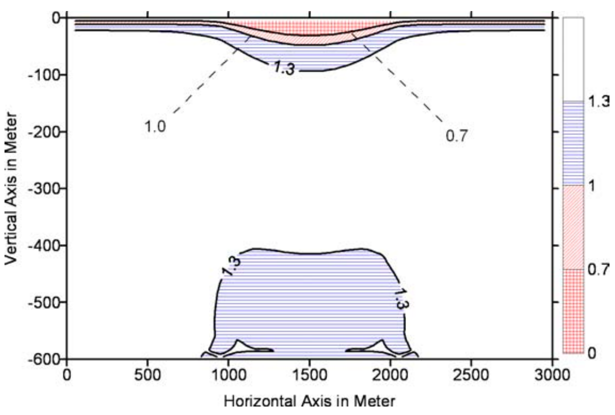

Fig.12: Factor of safety regarding Hydrofracturing at y = 5 m and t = 100 years in case THMS31.

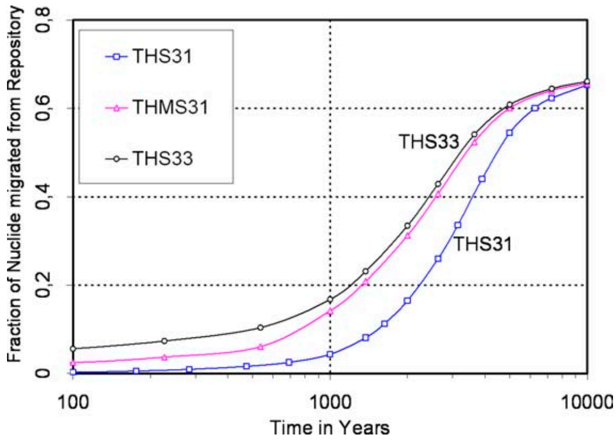

Fig. 15: Fraction of nuclide migrated from repository.

Application of the dilatancy concept to ascertain the damage and healing behaviour of rock salt

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Abstract

Besides other host rocks, rock salt formations are considered for the long-term storage of radioactive waste to exclude a threat to the biosphere. This means that the host rock's integrity has to be guaranteed during construction, operation and in the post-closure phase of a repository. Consequently, the contribution of the geological barrier to the safety of a repository has to be assessed by studying its natural characteristics and the main processes which make the transport of radionuclides possible or which can prevent the transport into the biosphere. In this context, the impacts of disturbance induced by the excavation of the underground facilities and long-term effects during re-compaction of the EDZ are the most important items. Therefore, understanding of the reciprocal action of damage respectively healing in rock salt is of vital importance for performing long-term safety analysis.

The integration of the relevant processes into the constitutive models requires consistent experimental data sets. The purpose of this paper is to illustrate the progress of experimental work performed in the last decade focusing on this issue. Firstly, the actual knowledge state of the dilatancy concept will be introduced, which provides the criterion to decide whether creep deformation without volume increase or dilatant deformation with propagating damage will occur. Then, the experimentally well documented deformation aspects, damage and the corresponding evolution of permeability, will be broadly discussed on the basis of field and laboratory tests. Also the actual state of numerical modelling describing damage will be briefly summarized. However, the description of the mechanical behaviour of salt is not complete if its favourable capacity for self-sealing of damage is not considered. Here some progress in the experimental work will be reported. The creation and evolution of the EDZ as well as of the hydraulic properties could be modelled quite well, however, it has to be stressed that self-sealing and healing are still open issues which require further investigation.

1 Introduction

Understanding the transport properties of rock salt and their relationship to its mechanical behaviour is of vital importance for the design and safety analysis of underground cavities, in particular with respect to the long-term storage of heat-generating radioactive waste and the storage of oil or hydrocarbon gas in salt caverns. The integrity of the geological barrier requires a sufficient tightness against fluids and gases which has to be guaranteed during construction, operation and in the post-closure phase of a repository which are schematically depicted in Fig. 1-1. Therefore, the understanding of the competing processes of damage respectively compaction and healing is of vital importance for the predictability of damage-related near-field processes and long-term effects, i.e. recovery of hydraulic integrity and subsequent gas generation:

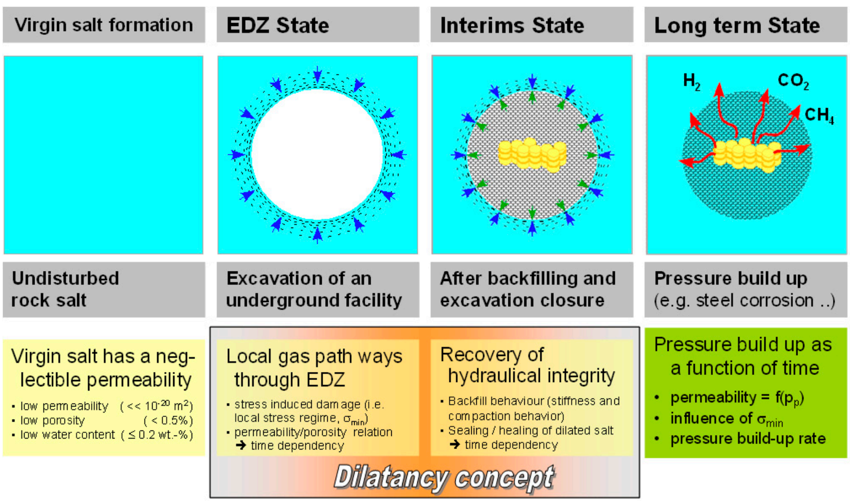

Fig. 1-1: Gas transport issues in a salt repository related to the dilatancy concept.

(1) Rock salt in undisturbed state is attributed to be impermeable for gases and fluids due to its low porosity and low permeability. During construction of underground openings in a rock salt formation, the change of stress state in the vicinity of these openings will affect the mechanical and hydraulic integrity of the surrounding rock salt by initiating local damage.

(2) The creation of the EDZ, and thus, the development of potential hydraulic pathways are closely related to stress dependent property changes, i.e. onset of dilatancy, as it was demonstrated through permeability measurements in field tests and under laboratory conditions. Depending on the order of permeability increase the EDZ has a high potential for gas transport in a repository, where gas will be produced by corrosion of metals or degradation of organic matter.

The EDZ in rock salt has been under systematic investigation for the last fifteen years as summarized during an European Commission Cluster conference and workshop held in Luxembourg in November 2003 [1]. In addition, the international conference „Saltmech6" in May 2007 in Hannover (D) gave a comprehensive overview about the actual knowledge regarding the mechanical behaviour of salt rocks and the coupled gas transport properties [2].

(3) In the post-closure phase (i.e. Interims state) of the repository, when the loading conditions will change to non-dilatant, subsequent healing (probably due to fluid assisted compaction creep) will take place restoring at least the initial gas tightness of the salt.

(4) In the long term state it is usually assumed that due to the subsequent gas production a time dependent pressure build-up with consequences of concern may occur. This aspect is separately discussed in the paper [3].

The deformation behaviour of rock salt depends on different micro-mechanical processes. For the modelling of these processes by constitutive equations and for the prediction of the long-term behaviour it is very important to distinguish between processes without dilatancy and those which are coupled with the evolution of dilatancy and damage. The principal behaviour of salt damage referred to the stress state is usually described on the basis of the so-called „dilatancy concept" which has been evaluated as a reliable basis for a prognosis of EDZ [4].

Based on this fundamental concept and referring to the general requirements of performance assessment (PA) studies the existing knowledge regarding the gas transport properties in rock salt will be briefly summarized related to the various phases of EDZ-evolution and its subsequent healing restoring salt integrity. Special account is given to the coupling between dilatancy and permeability as a base for a prognosis of the gas transport properties using numerical modelling of the mechanical behaviour of rock salt. Finally, conclusions are given about the state of knowledge obtained so far and remaining deficiencies.

2 The dilatancy concept

In contrast to crystalline rocks like granite, rock salt deformation is strongly influenced by pronounced visco-plastic behaviour. It reacts to the change in stress state by creep, which reduces the stress differences caused by the excavation. Creep takes place at constant volume and without damage of the salt. Thus, the risk of macro-fracturing is substantially mitigated. On the other hand, a micro-fractured EDZ with typical extents of several decimetres up to one or two metres develops around openings. Representing the relevant stress states, the dilatancy concept facilitates to distinguish between processes without dilatancy and those which are coupled with the evolution of dilatancy and damage [4].

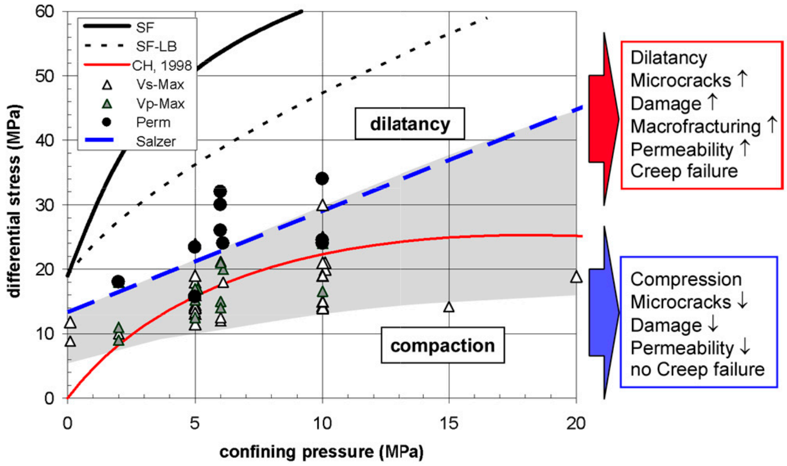

Fig. 2-1: The „dilatancy concept" - current understanding of EDZ nature and properties referring to the mechanical behaviour of salt in the stress space [4]. Experimental results from deformation tests are indicated where various micro-cracking sensitive physical parameters (Vp, Vs and permeability) were measured [6]. Short-term failure strength (compression) for rock salt (origin: BGR): SF respectively SF-LB (lower bound). Dilatancy boundaries: CH, 1998 – [4]; Salzer – [7].

In Fig. 2-1 the axes of the diagram represent the normal (σ) and the deviatoric (τ) stress, respectively. The stress space below the failure boundary is separated by the dilatancy boundary in the two domains, compaction and dilatancy. As long as the state of stresses remains in the non-dilatant compaction domain, the ductile rock salt deforms without any crack formation and without dilatant crack propagation. The transition in the state of stresses, where the deformation behaviour changes from ductile (no volume increase, $\Delta V \leqslant 0$) to dilatant deformation ($\Delta V > 0$) associated with permeability increase, corresponds to the dilatancy boundary.

Important property changes associated with dilatancy are additionally indicated. The dilatant domain is characterized by micro-cracking, causing accumulation of damage. Permeability and probability of creep failure are accordingly increasing. Air-humidity in the mine or fluids from inherent brine inclusions can permeate through the dilating salt, causing a humidity-assisted increase in ductility [8]. In the non-dilatant domain, rock salt is „compressible": micro-cracks are compacted, closed or even healed, and further micro-cracking is suppressed. Accordingly, permeability decreases and no failure will occur even during long-term deformation.

Although the existence of a dilatancy boundary is an experimental fact, it has to be mentioned that this boundary is more a transitional field than a distinct line, because the detection of onset of micro-cracking depends obviously on the sensitivity of the measured parameter. High-resolution ultrasonic velocity measurements (e.g. [9]) give clear hints of local onset of micro-cracking at lower stresses, whereby in axial compression tests Vs decreases sooner than V_p (the reverse is true under extensional conditions). Since a good agreement with the onset of humidity-induced creep acceleration was observed [8] this stress level has to be understood as the „lower damage boundary" which corresponds roughly to the older formula given by [4].

Measurements of the volume change during deformation proved opening of micro-cracks (respectively onset of dilatancy), primary at significant higher stress levels resulting in a dilatancy boundary as described for instance by [7]. Importantly, only at onset of dilatancy a simultaneous increase of permeability is observed (see Fig. 2-1).

Referring to the in situ case, by excavation of an opening in rock salt, the stress state in the vicinity is strongly disturbed. High deviatoric stresses occur, and the stress state is shifted into the dilatancy field. Damage evolution is controlled by the prevailing stress field, the creep properties of the salt, and the local lithologic heterogeneity. The closer the stress state is to the failure boundary, the faster damage evolves. The stress field is influenced by the geometry of the excavation and by technical means (e.g., lining or backfill in the excavation). Salt creep rate is a function of effective stress and temperature, but is also influenced by factors such as humidity and dilatancy [10]. Dilatancy leads to a reduction of strength of the salt. The deformation by creep and dilatancy results in a redistribution of stresses and a zone of low stress in the vicinity of the excavation.

When a supporting backfill or sealing structure is emplaced, creep leads the stress state to return below the dilatancy boundary, where only compression can occur, thus promoting healing and decreasing porosity and permeability.

3 EDZ formation – Damage and permeability evolution

3.1 Micro-mechanical model of salt deformation

The deformation behaviour of rock salt generally depends on the basic micro-mechanical processes which are active in correspondence with the dilatancy concept. This is illustrated in Fig. 3-1. Therefore, this concept is also the base for the so-called Composite Dilatancy Model (CDM), for details see [11].

In rock salt, the non-dilatant creep deformation is controlled by the trans-crystalline movement of dislocations in distinct glide systems [12]. For creep without volume change the Orowan equation is used to relate the macroscopic deformation rate to the dominant micro-mechanical processes

$$\dot{\varepsilon}_{cr} \sim b \cdot \rho \cdot v \qquad (3\text{-}1)$$

where $\dot{\varepsilon}_{cr}$ is the macroscopic strain rate, b the Burgers vector of a gliding dislocation, $\rho = 1/r^2$ the density of gliding dislocations with mean distance r, and v their mean velocity. The velocity v itself depends on stress, temperature and the parameters of the micro-substructure, Q is the activation energy for dislocations, M the Taylor-factor for cubic crystal symmetry (fcc), Δa the activation area for moving dislocations and σ* the effective stress acting on these dislocations:

$$\dot{\varepsilon}_{cr} = \frac{b}{M}\frac{1}{r^2}\nu_0 \exp\left(-\frac{Q}{R\,T}\right)\sinh\left(\frac{b\,\Delta a\,\sigma^*}{M\,k_B\,T}\right) \qquad (3\text{-}2)$$

Dislocations may propagate on {110}-gliding planes in [110]-directions, where the unavoid-able interaction of the dislocations will cause dislocation pile-ups and the build-up of internal stress concentrations. It is generally accepted, that the evolution of damage is strongly coupled with the stress concentration at dislocation pile-ups during creep deformation (e.g. [13]).

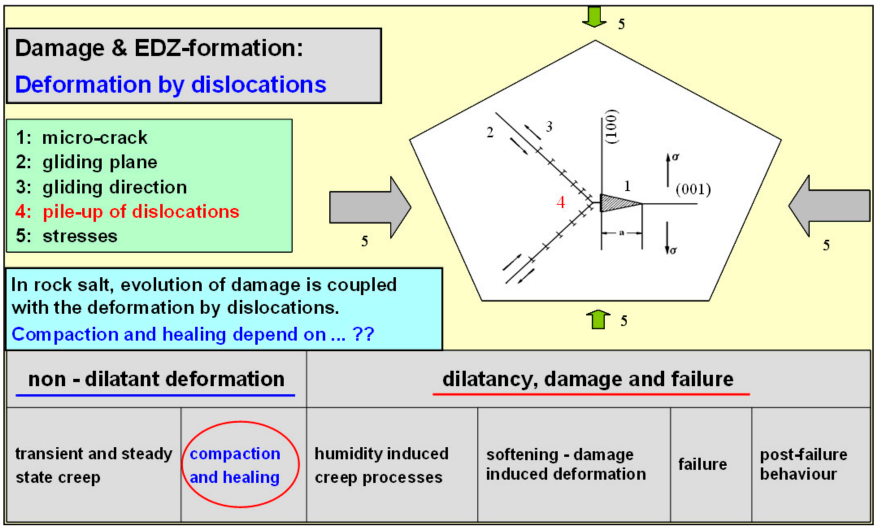

Fig. 3-1: Micro-mechanical model of non-dilatant deformation in rock salt on basis of rate controlling dislocation mechanisms and for their coupling with dilatancy related processes.

After the transient creep phase, where the micro-structure evolves as represented by Equation (3-1), the steady-state creep behaviour is achieved as soon as the generation of dislocations and the deformation hardening by dislocation pile-ups is compensated by thermally activated recovery processes - however, the steady-state creep behaviour will only hold as long as the stress generates no trans-crystalline micro-cracks at a pile-up of dislocations. This will happen in the dilatant stress domain.

Microstructural investigations demonstrate that at sufficiently low effective mean stress, dilatancy can occur, which involves different crack mechanisms. Fig. 3-4 shows both, oriented intragranular crack opening or tensile cracks (Mode 1) and, in addition, diffuse dilatancy, i.e. opened grain boundaries, producing significant permeability (see chapter 3.3)

Since rock salt exhibits no brittle failure behaviour, but even a ductile post-failure behaviour, also this process is described by CDM as a function of the creep rate. Incorporating humidity induced creep, which takes place only in dilated rock salt, the total strain rate $\dot{\varepsilon}_{tot}$ is expressed by the creep rate $\dot{\varepsilon}_{cr}$ where the impact of the humidity induced creep on ductility is denoted by F_h, that of the damage (i.e. damage induced weakening/softening) by δ_{dam}, and that of the post-failure behaviour by P_F

$$\dot{\varepsilon}_{tot} \sim P_F \cdot \delta_{dam} \cdot F_h \cdot \dot{\varepsilon}_{cr} \qquad (3\text{-}3)$$

In case of the deformation in the non-dilatant stress domain, these additional impact factors have the value of unity.

The damage function δ_{dam} depends on the irreversible volume change energy d_{dam} (briefly: damage energy) which evolves during the deformation under the conditions of the dilatant stress domain

$$\delta_{dam}\,(d_{dam}, \sigma_{min}) = \exp\,[\delta_1 \cdot (d_{dam}/\sigma_u)^{\delta_2 \cdot \sigma_{min}}]. \qquad (3\text{-}4)$$

The determination of the damage and the affected microstructural parameters, as described in [11] for the CDM - for example, is not a simple task, but it has to be pointed out that the various stages of damage can be convincingly simulated until failure. Alternative concepts are summarized in chapter 3.4.

3.2 In situ findings

The occurrence of the EDZ, and thus, the development of potential hydraulic pathways, is closely related to stress dependent property changes. This was demonstrated through permeability measurements in field tests at various sites since the beginning of the eighties.

It has been generally confirmed that the extent of the EDZ relies on the stress state and the geometry around the opening. Typically, it ranges between a few decimetres up to 1 to 2 m. Permeability measurements at the 800 m level of the Asse salt mine [14] showed, that the usual cross section of a drift with a flat floor and a domed roof leads to a larger extent of the EDZ below the floor compared to the walls and roof - corresponding to the state of stresses around a drift of that shape. Another well investigated example is the dam building project in the salt mine Sondershausen. Using conventional packer tests IBEWA determined the depth distribution of permeability in various temporal stages, as depicted in Fig. 3-2.

After 30 years lifetime of the original circular drift (Ø = 3 m) the extent of the EDZ is in the order of around 1 m, as indicated by decreasing permeability ranging from a maximum permeability of around 10^{-16} m^2 close to the drift opening to around 10^{-21} m^2 corresponding to the undisturbed rock. As a prerequisite for installing a bentonite-based sealing element, the drift was altered to a rectangular cross section. The low permeability measured one month after alteration demonstrates that the cut-off of dilated contour parts is a useful method to improve self-sealing effectiveness when constructing technical barriers. Repetition of the measurements after two years showed that the initial permeability profile is nearly restored.

In addition, it has to be mentioned that anisotropy effects corresponding to the acting stress field have always to be considered. Recently, a field test performed by GRS in the Asse salt mine, as measured by a near-drift system, nicely documents that the corresponding permeability behaves strongly anisotropic in the rock contour (e.g. [15]). In general - due to the highly anisotropic stress field in the drift contour - gas flow occurs preferentially in a plane parallel to the excavation surface whereas perpendicular gas mobilisation is inhibited.

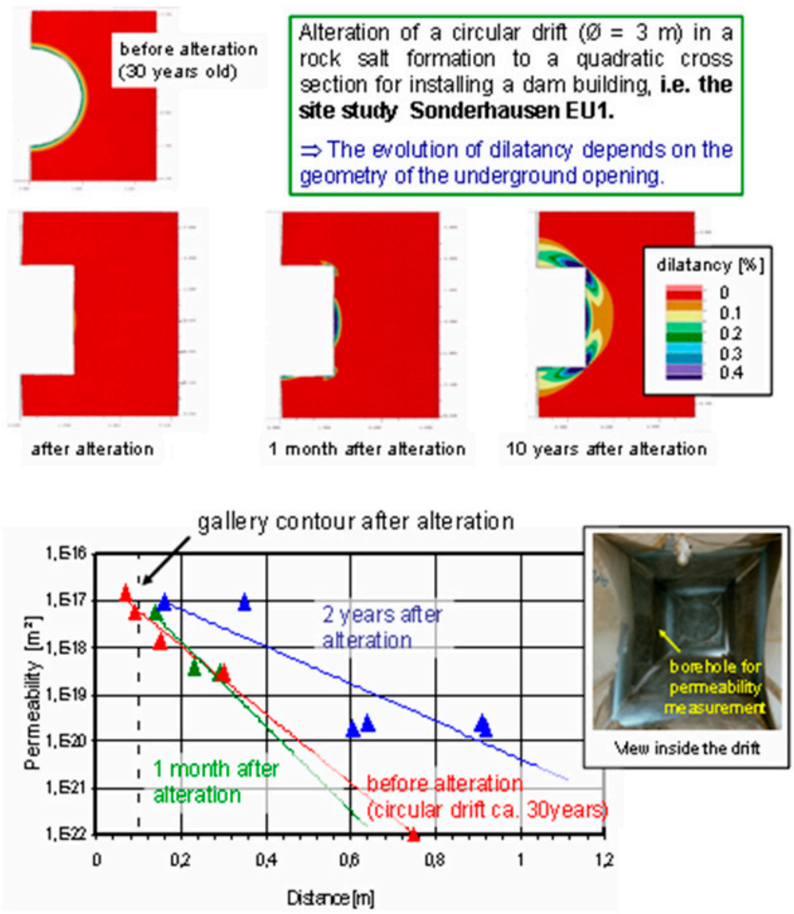

Fig. 3-2: EDZ-evolution in the rock contour of a 30 years old circular drift before and after alteration to a quadratic cross section. (top) Comparison of numerical calculations simulating the various drift states with time. (bottom) Results of repeated permeability measurements in the rock contour of the drift related to various phases of alteration (before and after – 1 month respectively two years, after [16]).

3.3 Outcome of laboratory investigations

During the last two decades, substantial progress has been made in the understanding of the effect of deformational conditions and parameters on the salt permeability on the base of laboratory tests on core samples at well defined conditions. Syn-deformational monitoring of various crack sensitive parameters (i.e. volumetric strain, permeability and ultrasonic wave velocities) facilitates to discriminate the actual state of damage during progressive deformation as discussed in chapter 2. Complete data sets are depicted in the Figs. 3-3 and 3-4.

Permeability under deviatoric stress conditions („dilatancy domain") evolves in different stages corresponding to stress, strain and time. In deformation experiments on natural rock salt (e.g. [17; 6]) and synthetic fine-grained halite [18] the onset of dilatancy is found to be accompanied by a drastic permeability enhancement of up to 5 orders of magnitude after the pore space had dilated by a small amount (< 1%), followed by a period of plus/minus constant permeability during strain hardening up to 10% axial strain or even more (compare Fig. 3-3).

Remarkably, as can be seen from Fig. 3-6, the order of permeability increase depends significantly on the acting minimal stress (for details see [5, 6]). This suggests that the evolution of permeability is not only a function of dilatancy but also of microcrack linkage as proposed by [18].

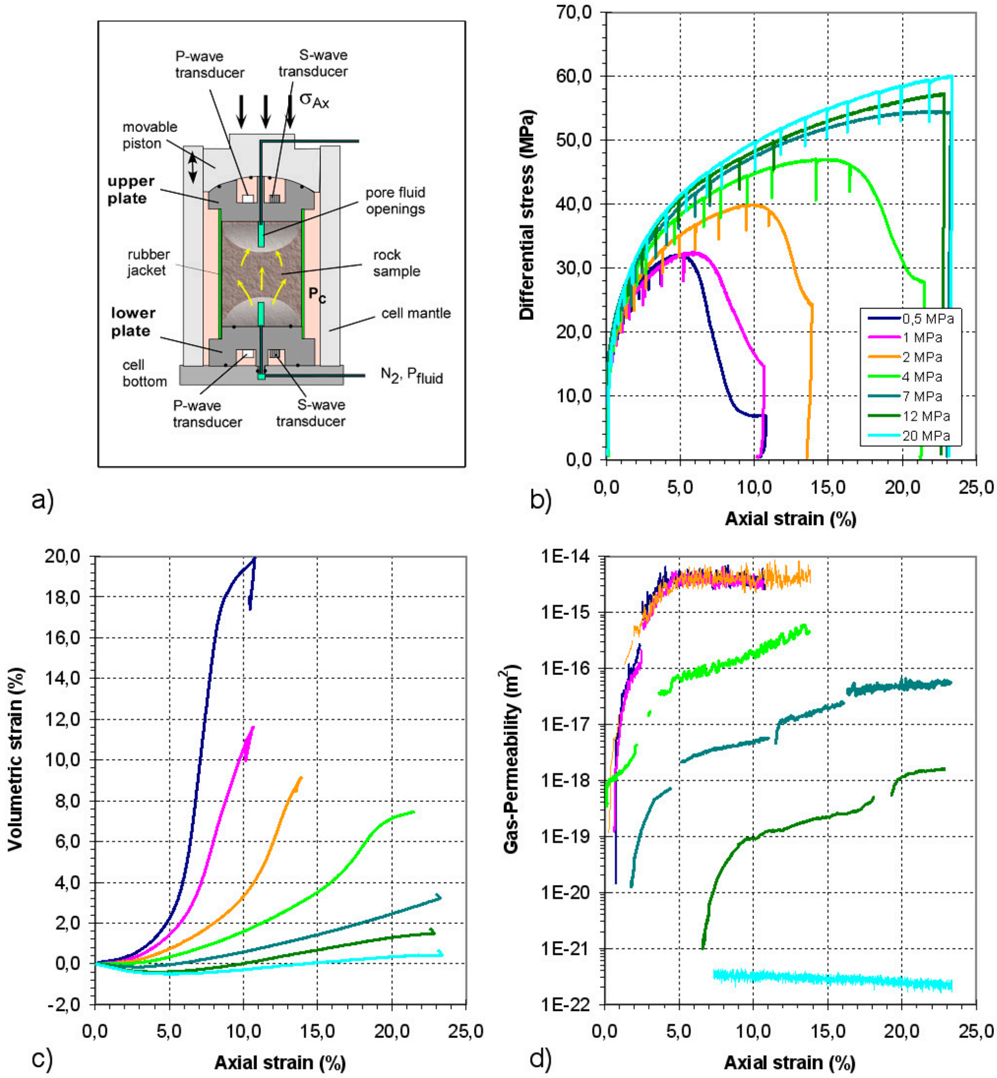

Fig. 3-3: Monitoring of damage during triaxial deformation – Experimental series of short term strength tests on Leine rock salt from the salt mine Teutschenthal/ Angersdorf. a) Experimental set up. Experimental results plotted vs. axial strain: b) Strength; c) volumetric strain and d) gas-permeability.

Also, the influence of the stress field affecting the permeability has been demonstrated [6]. As schematically depicted in Fig. 3-4, depending on the load geometry in the triaxial test, the micro-fractures are oriented preferably perpendicular to the minimum principal stress, i.e. relating to in situ conditions parallel to the excavation surface. Remarkably, also the relationship of the variation of the seismic velocities V_p and V_s depends on the stress state which again demonstrates the influence of crack geometry. However, it has to be mentioned, that a direct correlation of measured elastic wave velocities with transport parameters, i.e. permeability or porosity is difficult (compare Fig. 3-5).

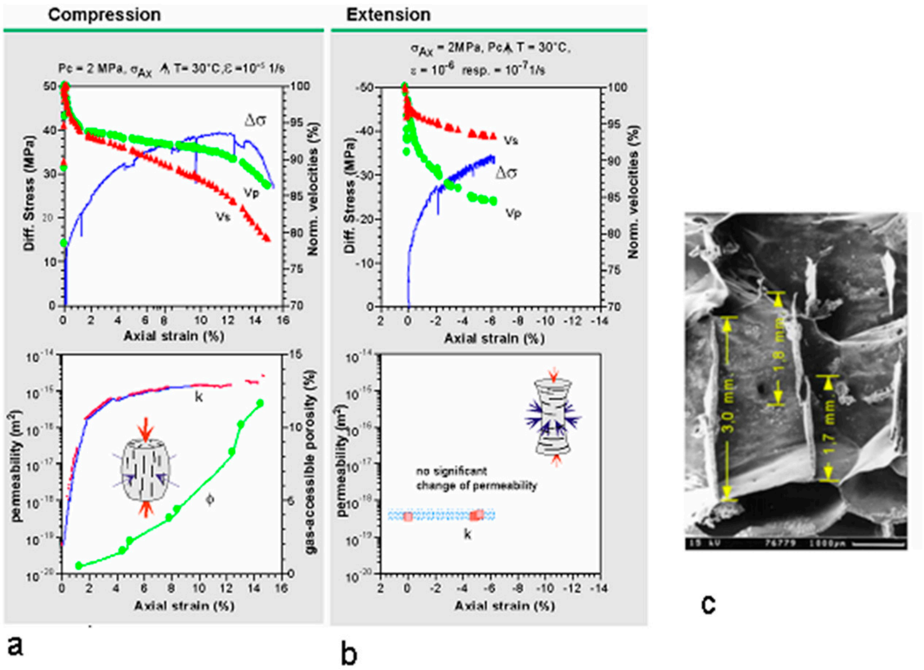

Fig. 3-4: Investigation of dilatancy, permeability, and seismic velocities under triaxial deformation (after [6]). a) compression: $\sigma_1 > \sigma_3$; b) extension: $\sigma_1 < \sigma_3$; c) REM-microstructure of dilated rock salt (pore space skeleton after impregnating the sample with epoxy and dissolving the soluble salt).

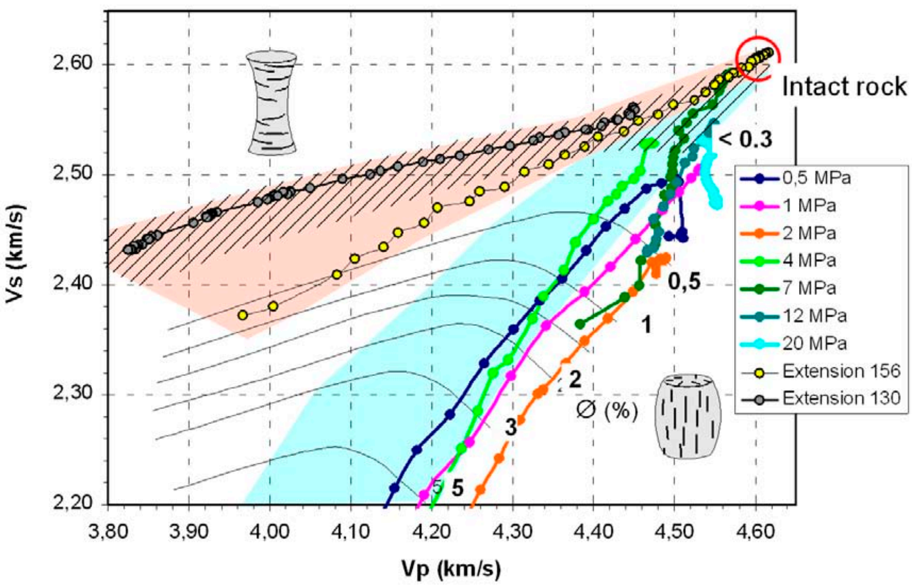

Fig. 3-5: Variation field of Vs vs. Vp together with isolines of porosity for rock salt. Data sets include the triaxial test series, depicted in Fig. 3-3 and older results of [6]. Note that the maximum Vp- and Vs-values correspond to the intact rock state without dilatancy.

3.4 Permeability-porosity relationships

The numerical modelling of fluid transport properties of dilated rock salt can be performed on the base of two approaches, either by a phenomenological description of experimentally derived relationships between the relevant parameters or simulating the fracture geometry and network properties based on textural investigations of damaged samples. Although the capability of the latter physical funded procedure seems to be obvious, the available micro-structural data base representing in situ conditions of the low-porous rock salt undergoing deformation might not be sufficient for the definition of the equation parameters (e.g. [19]). Therefore, the summary is constrained on the first approach.

In Kozeny-Carman's and other classic models permeability is described as proportional to simple integer powers of the relevant pore geometry parameters, i.e., porosity, hydraulic radius, tortuosity and/or specific surface area. For a given process, these parameters are usually assumed to be related to each other through power-law relationships, therefore leading to a power-law dependence of permeability on dilatancy respectively porosity (ϕ), possibly with a non-integer exponent:

$$k = k_0 \, \phi^n \qquad (3\text{-}5)$$

Extensive laboratory testing aiming on deformation induced permeability changes were performed covering a wide range of loading conditions (see Fig. 3-6). Although there is a significant data scatter, it is obvious that a single power-law exponent does not always hold as porosity changes. One possible approach is to keep the power-law representation but with a variable exponent representing at least two parts of permeability evolution ([6]): (1) an initial steep increase due to progressive development of micro-cracks, and (2) beyond a certain threshold boundary a saturation state with moderate increase due to widening of created pathways. In addition, it is important to note that the threshold until reaching the saturation level in region (2) is obviously a function of σ_{min}.

Based on this concept, an improved description of the dilatancy induced permeability increase in rock salt under consideration of the minimal stress was recently presented in [20] (note the various model curves in Fig. 3-6):

$$k = \frac{k_{tp}}{\left(\left(\frac{\phi}{\phi_{tp}}\right)^{-n_1} + \left(\frac{\phi}{\phi_{tp}}\right)^{-n_2}\right)} \qquad (3\text{-}6)$$

n_1 and n_2 are constant inclination values according to the two relevant porosity/permeability slopes in the double logarithmic diagram. The other parameters are depending on the minimal principal stress σ_{min} (for parameters of the constitutive equations see Table 3-1):

$$k_{tp} = a_k \cdot \exp(-b_k \cdot \sigma_{min}) \qquad (3\text{-}7)$$

$$\phi_{tp} = a_\phi \cdot \exp(-b_\phi \cdot \sigma_{min}) \qquad (3\text{-}8)$$

Table 3-1: Parameters for describing permeability as function of porosity depending on the minimal stress σ_{min} (after [20]).

Parameter	Value	Parameter	Value
a_k	4.27E-14 m^2	a_ϕ	0.0263
b_k	1.26 MPa^{-1}	b_ϕ	0.3093 MPa^{-1}
n_1	4	n_2	1.07

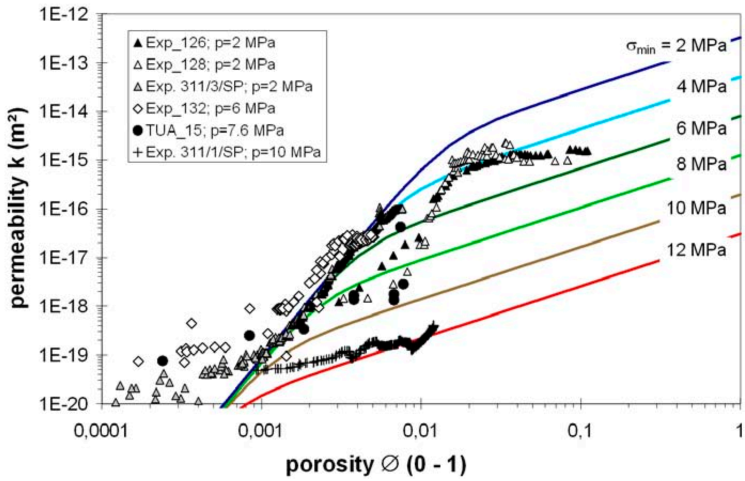

Fig. 3-6: Measured and calculated permeability/porosity relationships depending on the minimal stress (modified after [5]). The various modelling curves result from the relationship of [20].

Although the modelled permeability data seem to be slightly overestimated in region 2 the new concept of stress and porosity dependent permeability evolution has been successfully proved by [20] in the BAMBUS II-project.

3.5 Rock mechanical modelling of EDZ and associated property-changes

For the prediction of the mechanical behaviour of rock salt respectively the evolution of the EDZ and its coupled hydraulic behaviour, all those processes which contribute substantially to the time-dependent and spatial evolution of stress and strain in the material have to be taken into account (e.g. [21]). They comprise non-dilatant creep as well dilatancy and damage affected deformation processes. In general, each constitutive model describes a phenomenon (i.e. single deformation process) by one or a set of appropriate equations. Then, the combination of these modules reflects the overall coupled deformation processes, which allows beyond others the quantification of damage resp. porosity which can be connected to the hydraulic properties via a porosity/permeability relation of the form described in the foregoing chapter. Implementation of the model into various numerical (commercial or scientific) codes facilitates its universal application. Actual constitutive models and the used numerical codes are summarized in Table 3-2.

Exemplarily, Fig. 3-2 demonstrates the capability of numerical modelling by recalculation of the spatial and time dependent evolution of dilatancy around a drift in a salt mine where the cross section was changed from circular to quadratic. Note the induced stress concentration in the edges whereby most of dilatancy evolving with time is concentrated on the centre of the walls respectively the floor and roof. For comparison, permeability measurements were done at various depths in a borehole in the centre of the left wall. Repetition of the measurement nicely demonstrates the time and geometry dependent evolution of permeability.

Table 3-2: Actual constitutive models and numerical codes [21]. For the various abbreviations and details of the codes see the mentioned paper. The different codes are distinguished between finite element method (FEM), the finite different method (FDM) and the distinct element method (DE).

Institution	Constitutive model	Code
Hampel	Composite dilatancy Model (CDM)	FLAC (FDM)
BGR	CDM	JIFE (FEM)
IfG	Minkley	FLAC (FDM), UDEC (DE)
IfG	Günther/Salzer	FLAC (FDM), UDEC (DE)
INE-Pudewills	FZK-model	ADINA, MAUS (FEM)
TU Clausthal	Hou/Lux	MISES3 (FEM), FLAC (FDM)
IUB Hannover	Multimechanism Deformations Coupled Fracture (MDCF-IUB)	UT2D (FEM)
Uni Barcelona	COupled DEformation, BRIne, Gas and Heat Transport	CODE_BRIGHT (FEM)

However, a general problem has to be mentioned. The permeability will be an isotropic feature as long as the anisotropic crack formation is not explicitly taken into account. To the authors' knowledge, only models, which are based on damage mechanics as proposed by [22], incorporate anisotropy into a stress dependent porosity-permeability relation.

4 Post-closure phase – Recovery of hydraulical integrity

In the post-closure phase, after the end of excavation and backfilling the shear stress in the rock salt is continuously decreasing by creep until the isostatic state of stress is reached. Therefore, the state of stresses in the EDZ will consequently move from the dilatant into the non-dilatant domain. At last, this causes the re-compaction and the related decrease of the permeability in the rock salt of the EDZ.

Reestablishment of hydraulical integrity of the EDZ at in situ conditions, at least partial, has been recently confirmed by in situ investigations at a unique test site existing on the 700-m level of the Asse salt mine [14]. There, a cast steel liner of about 20 m length had been installed in a drift and backfilled with concrete as early as in 1914. Permeability measurements demonstrate that under the floor of the open drift, a typical EDZ with 1.5 m extent and permeability up to above 10^{-16} m^2 had evolved. Around the lined drift, however, the permeability had diminished to values between 10^{-20} m^2 and 10^{-19} m^2.

Although a considerable reduction was reached, the permeability of the undisturbed salt is attributed to significantly lower values, which may indicate that no real healing is obtained but self sealing due to compressive crack closure. It has to be stated that in this work the term „compaction" is preferred to describe such processes of decrease whether porosity or permeability. This term does not distinguish mechanically induced crack closure (due to hydrostatic or shear-enhanced compaction) from true healing (due to mass transfer by chemical processes like solution and reprecipitation, recrystallization etc., e.g. [18]). The latter will accomplish a recovery of the cohesion between crack planes. With respect to this definition the evolution of the temporal compaction and the permanent healing of rock salt have to be investigated with long-term tests using pre-damaged samples, i.e. after triaxial strength testing as depicted in Fig. 3-3.

The compaction is performed at T = 30 °C by the application of an isostatic pressure which is increased stepwise up to p_{iso} = 12 MPa and then decreased in steps again (whole testing time: ~ 70 d; for details of the performed experimental work see [23]). Isostatic loading of the dilated specimen results in a spontaneous but rather small decrease of permeability respectively porosity, as can be seen in Fig. 4-1 and Fig. 4-2. During constant loading sections, the porosity and the permeability are further decreasing, where a short transient behaviour with a more rapid decrease in the first stage of a section is detected until a more or less stationary decrease is obtained. The time dependence is plotted in Fig. 4-2.

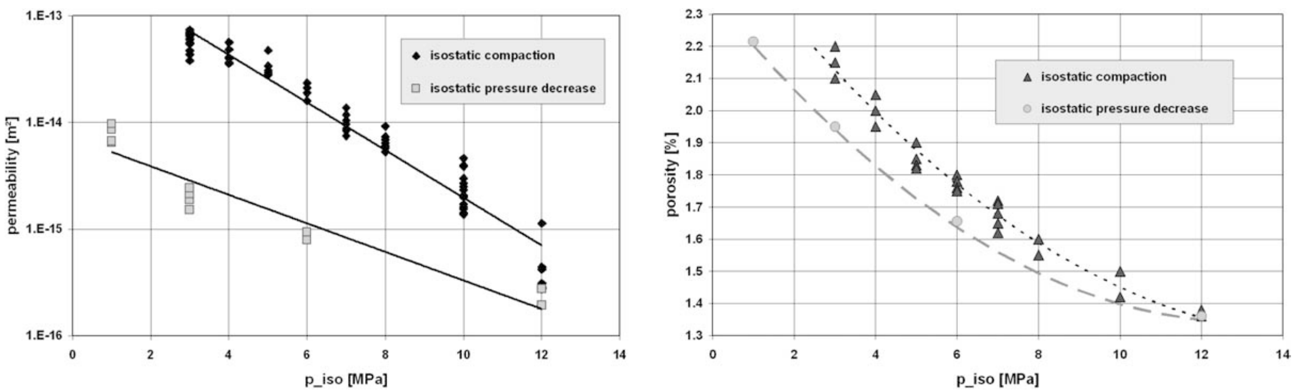

Fig. 4-1: Evolution of the general trends of the porosity (right part) and the permeability (left part) in dependence on the stepped isostatic pressure, where the time dependence is dominant (Fig. 4-2). Permanent decrease of porosity and of permeability (i.e. healing) is obvious.

However, stepwise unloading of the sample results in a spontaneous increase of both, permeability and porosity, which partially restores the initial state of dilatancy. Therefore, both figures clearly demonstrate that the permanent volume decrease (i.e. real healing) remains small. The permanent decrease in permeability achieves one order of magnitude, but decreases in between about three orders of magnitude during the stepwise increased isostatic pressure.

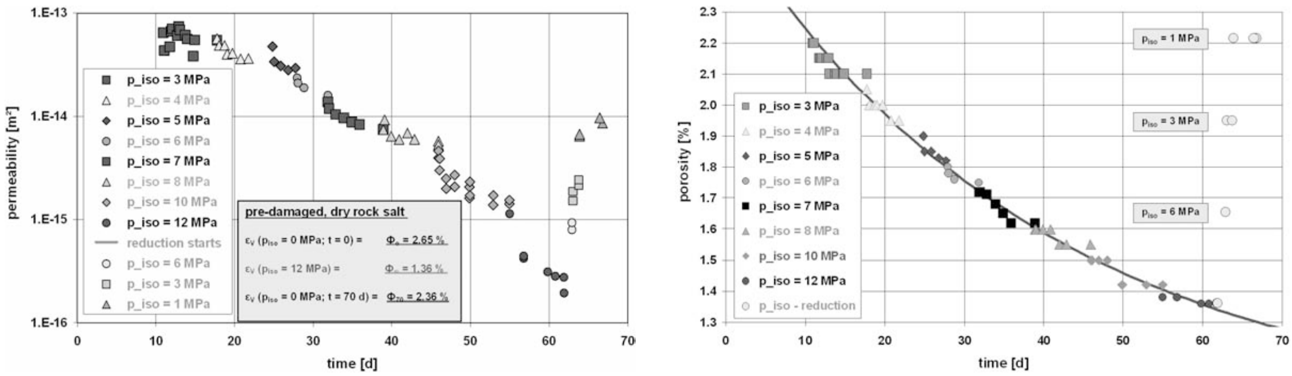

Fig. 4-2: Evolution of porosity (right part) and permeability (left part) in dependence on time during stepped isostatic loading with a time dependent transient compaction and decrease of the permeability, but a spontaneous de-compaction and permeability increase after 61 days of compaction.

The variation of the consecutive measured porosity and permeability during the compaction respectively decompaction cycle is compiled in Fig. 4-3. Remarkably in the double-logarithmic plot the relations between permeability k and porosity ϕ of both, compaction and decompaction cycle, follow straight lines but both slopes are much steeper than the relationships, as observed in the dilatant stage according to the formalism of [22].

However, it becomes obvious that the acting mechanisms for rock salt undergoing dilatant deformation with porosity production and the reverse, i.e. compaction of predilated or granular salt with porosity-destroying processes are different as usual (e.g. [24]). Although it seems clear that humidity effects may promote healing in dilated salt experimental data for quantifying the effect are not available. Nevertheless, in a first attempt, the hysteresis between the k-ϕ-trend lines is attributed to permanent compaction (i.e. healing).

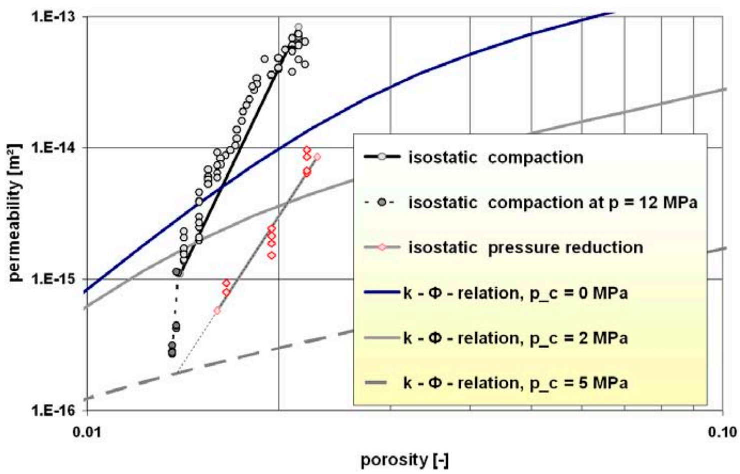

Fig. 4-3: Evolution of permeability in dependence on porosity during isostatic compaction. Dark solid line represents measurements during the stepwise increase of the isostatic pressure $p_{iso} \leqslant 10$ MPa, dark dashed line those during the section with $p_{iso} = 12$ MPa, and light solid line those during the reverse cycle, i.e. the stepwise reduction of p_{iso}. For comparison, the evolution of permeability and porosity during loading in the dilatant stress domain is plotted after [20].

Summarizing the results of long-term compaction experiments with durations of several months performed by BGR and IfG demonstrate that permeability of dry salt at a given hydrostatic pressure decreases exponentially with time:

$$k = k_0 \cdot e^{(-a \cdot t)} \quad \text{with } a = \text{compaction coefficient} \quad \text{and} \quad t = \text{time (d)} \qquad (4\text{-}1)$$

As shown in Fig. 4-4 the compaction coefficient *a* varies by a factor of 10 depending on pressure, whereas temperature seems to be of minor importance.

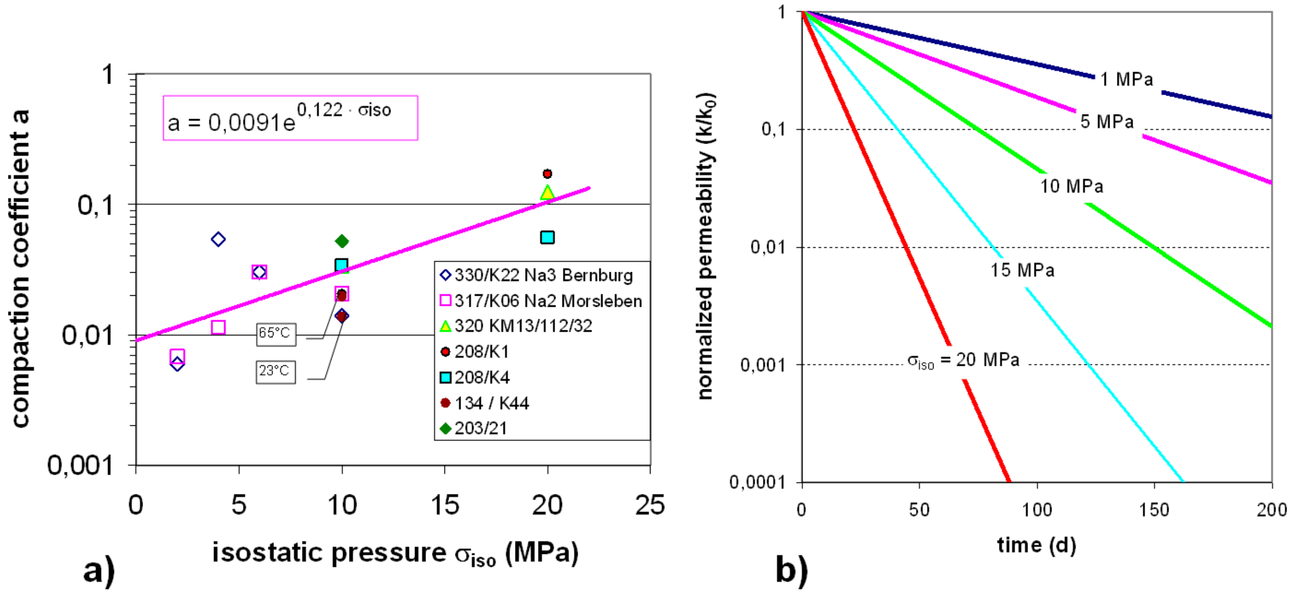

Fig. 4-4: Pressure induced permeability decrease with time. a) Evaluation of isostatic long-term compaction tests with continuous permeability monitoring (not shown here) using a simple exponential approach. b) Development of the normalized permeability with time at various pressure stages according to the respective compaction coefficients a as determined before.

5 Concluding summary and recommendations

Summarizing the previous sections, the following statements regarding the actual knowledge about damage and healing in rock salt can be made. They are based on the dilatancy concept and incorporate the relation with the gas transport properties of rock salt:

- ***EDZ-formation:***

Gas flow through rock salt is mainly restricted to the excavation damaged zone (EDZ). Well proven techniques for quantifying the extent and spatial geometry of the EDZ in salt formations are available (e.g. permeability, electrical, seismic....). During the excavation of an underground opening, dilatancy and EDZ evolution start to take place without delay:

- induced by deformation induced stress re-distribution due to the excavation
- EDZ evolves with time, depending on geometry and size of excavation, rock properties, excavation technique, mechanical support (e.g. backfilling, lining).

The extent of the EDZ is limited, i.e. no interactions between various zones. From a geomechanical point of view the EDZ is well understood:

- Based on sophisticated laboratory investigations with syn-deformational monitoring of various physical crack-sensitive parameters (e.g. permeability, volumetric strain, ultrasonic wave velocities) a comprehensive experimental data base is available.
- Verified rock mechanical models for predicting its initiation and evolution exist.
- Permeability-porosity relationships depending on σ_{min} are available.

However, the knowledge about hydro-mechanical coupling is not sufficient. Healing capacity of salt was proven but its time dependence and the relevant influence parameters (i.e. humidity) are not well understood.

- ***Recovery of hydraulic integrity - Sealing resp. healing***

Due to experimental difficulties it has to be considered that the experimental database is not sufficient, i.e. long-term compaction experiments with simultaneous permeability measurements are missing. The remaining challenges are:

- Understanding of physical processes which control the efficiency of healing in dilated rock salt with respect to humidity effects respectively pore pressures.
- Transition from dilated rock to natural rock conditions.
- Development of generally agreed constitutive models for the compaction of dilated rock salt.

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Prioritizing R&D for seven radioactive waste disposal options — an independent, interactive approach

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Abstract
A panel of university geoscience students has created a ranked list of research and development priorities concerning the disposal of radioactive waste. This prioritization is the result of a two month process during which the students were introduced to the disposal problem, submitted independent literature reviews, and developed selection criteria. Seven aspects regarding disposal were considered; geologic disposal in granite, in tuff/basalt, in salt, and in clay; sub-seafloor disposal; on-site disposal of mine tailings; and mineral encapsulation of radionuclides. The diversity of options provides the potential for dramatically different balances of geological, technological, political, and economic criteria in creating priorities. An emphasis on research and development forces the panellists to consider the role of near-term choices in changing the nature of long-term options. The prioritization process resulted in a method and ranking that has various similarities with the strategies and conclusions of other panels considering radioactive waste disposal options. Since the student panel was independent and not affiliated with stakeholders in the field, the similarity provides evidence for the robustness and objectivity of existing decision making processes. Furthermore, the unusual composition and focus of the student panel resulted in a noteworthy emphasis on the development of prompt and international solutions.

1 Introduction

Nuclear energy is a proven technology for future electricity generation. In 2002 there were 441 operational nuclear power reactors in the world, and another 32 under construction [17]. The main problem with nuclear energy is the safe management and disposal of the resulting radioactive waste [1]. On the other hand, despite public concerns about nuclear waste disposal, the technology still remains the largest proven carbon free generation option [26, 37], accounting for about one-fifth of the electricity produced annually in the US and about one-quarter of that produced in the UK [37]. Thus the future of nuclear power depends on how the safety issues regarding nuclear waste disposal are solved [26].

Global radioactive waste production statistics are not publicly available, but the USA (104 reactors) alone has some 160,000 spent fuel assemblies (45,000 metric tons) in storage, and it produces another 7,800 (2,000 metric tons) every year [38]. Mine tailings by volume far exceed the SF and HLW production. Of all the waste produced, some 5 per cent needs to be disposed of in a geological repository [4]. Fifty years ago, the US National Academy of Sciences already recognised the potential of disposing radioactive waste in deep geological formations [34], and it is now generally seen as the preferred means of disposal for high level and long-lived radioactive wastes [20]. Furthermore, the geological disposal of nuclear waste has been sanctioned by the International Atomic Energy Agency (IAEA) as the safest form of storage of high level waste away from human exposure.

Although there has been prolonged agreement as to the appropriateness of underground disposal, there is yet to be an implementation of proposed programs [20]. This is due to the fact that the discussion regarding the best overall substrate for geological disposal and whether to solve the waste problem nationally or internationally is still on-going. At the same time, discussions focusing on the immobilization and isolation of radionuclides in the underground repositories by adequate packaging in specific materials are directly linked to the choice for a particular repository substrate.

The following report is only supplementary to these debates, yet it offers a collective opinion from a distinctive perspective. It is based on an earth science related-project carried out by a panel of seven students enrolled in a Liberal Arts and Sciences college with no prior knowledge of radioactive waste disposal methods. This unconventional approach could be an interesting comparison against proverbial conclusions repeatedly drawn in the past, by panels consisting of highly-experienced, well-informed specialists. As Ch. McCombie (Arius, Switzerland) stated; the panels involved in drawing up site selection criteria often do this for the simple reason of their expertise. The student panel approach allows for a much more free-thinking discussion about the various disposal options and priorities for R&D.

The report discusses the criteria-based ranking of seven given disposal methods as priorities for research and development.

This includes the added options of the safe disposal of mine tailings and the possible improvement of containers for packaging of waste via incorporation in ceramics. Above-surface storage and transmutation research were not included due to the limited relation to the earth sciences; new mining techniques (ISL) were not included to keep the project focussed on the already existing waste.

It is hoped that this report will contribute to gain a broader perception; one which incorporates opinions reflective of a subsequent generation, on policies governing waste disposal and the use of nuclear energy in general. More importantly, it is expected that the approach may offer a small, if any, contribution to the scientific goal of the prolonged deliberation process.

2. Methods

2.1 Prioritization procedure

The ten week procedure for arriving at the final ranking for research and development priorities regarding seven cases concerning the disposal of radioactive waste included 8 stages. The procedure is summarized in Fig. 1.

1. Seven research and development options for radioactive waste disposal were selected. Only a single short list of options was derived, because of the small panel size (7 students) and the limited time allowed for the project. Four options (disposal in clay, salt, granite, and tuff) were provided in order to allow a wide range of siting issues to be considered. Sub-seabed disposal was included to gauge the panel's response to the environmental issues inherent to that option. Disposal of radioactive mine tailings allows the consideration of an understudied and more immediate class of environmental, health and socio-economic threats, a waste form with a different characteristic and volume, yet directly linked to the production of the other waste types through the resource cycle. The final option, encapsulation of radionuclides in minerals introduces the consideration of new technologies that can change the nature of the disposal problem by enhancing engineered barriers.

1. Selection of 7 options for R&D prioritization related to the disposal of radioactive waste
2. Basic theory regarding related geological, physical, chemical, hydrological, logistical, and technical aspects is presented and discussed with the student panel
3. Each student panel member writes a document with arguments for and against the assigned option by doing literature research and using an extended list of criteria (table1)
4. During a panel meeting the option evaluation documents are reviewed and discussed
5. Using a two-step ranking procedure, a ranking was agreed upon. The two steps were based on the overall balance of the individual option rankings and the subsequent voting on a priority ranking from high to low after discussing the results for the overall balance
6. A reduced list of criteria was produced (table 2) by analyzing the arguments for or against the various options as based on the extended list of criteria (table 1)
7. Interactive poster presentation of the prioritization rankings and criteria to experts in the field of radioactive waste processing, storage and disposal
8. Final ranking based on traditional decision matrix which also takes into account the weight of the comments of the experts on a certain criteria and on specific R&D criteria

Fig. 1: Prioritization procedure for 7 R&D research options concerning radioactive waste disposal.

2. The panellists were given an introduction to the nuclear fuel cycle and radioactive waste disposal problems. This included an overview of the properties of nuclear fission, radioactive decay, radiation properties, a description of state of the art above-ground storage, and lectures on relevant topics in geology, hydrology, geochemistry, seismology, and tectonics. This stage

provided the panellists with the background knowledge needed to interpret relevant literature.

3. Each panellist was assigned one of the seven options to explore and evaluate using the information provided in stage 2 and by consulting literature. A month time was given to write a literature review and describe the strengths and weaknesses of the assigned option as a research priority. Panellists were also asked to present the strongest possible arguments for supporting their particular option. This encouraged them to take on some of the characteristics of a typical panel having a diversity of areas of expertise and of personal biases. An important element in this process was the use of an extended list (table 1) of possible criteria to consider when arguing for the pursuit of the assigned disposal option. This list is fairly typical of those considered by nuclear waste policy panels, although less exhaustive than most.

4. The information from the seven literature reviews was shared and debated for two hours during a panel meeting. Written review reports prepared by the panellists were distributed to all students and studied. The merits of the various options were discussed as a panel, allowing equal time for each option with several rounds of questions and answers. .

5. A consensus was reached on the ranking of the options following a two-step ranking procedure in a subsequent panel meeting that lasted 2 hours. Each panellist was asked to submit a simple ordered ranking of the seven options, and these rankings were then summed to create an initial draft of the panel's priority list. Subsequently, each panellist stated the strengths and weaknesses for each option based on the most weighed criteria in their individual ranking process. After a discussion of the arguments the panellists voted again on the ranking. In this case, single votes (cast blind) from each member for a favourite option were gathered first. After the most popular 1st choice was identified, a single vote for 2nd choice from the remaining options was used to identify the panel's 2nd choice before the remaining options were considered in the same way to create the rest of the ranking. This two-stage ranking procedure explored the effects of differences in panel voting methods and of the influence of sharing information during ranking discussions.

6. In the same panel meeting a reduced list of prioritization criteria was developed Each of the strengths and weaknesses cited during the ranking process was classified according to the original list of criteria (table 1) presented to the panellists. Those criteria not used were discarded from the original list. The remaining criteria were combined into broader categories in such a way as to create a short list of criteria felt to have roughly equal weight in the ranking process. The original lists of strengths and weaknesses were then classified using the new criteria (table 2). This post-hoc characterization of selection criteria and weighting resembles a simple and practical realization of logistic regression analysis applied to an interactive multiple criteria decision process [32].

7. The criteria for each option and the R&D rankings and assessments were summarized and submitted for review by experts in the field by means of an interactive conference poster (presented at Reposafe, November 6-9, 2007). The poster allowed reviewers to submit their own rankings and argue for or against a certain R&D option. This provided the panellists with access to expert testimony and information to which they had not previously been exposed.

8. The expert comments were evaluated and discussed by all panellists after the conference in a 2 hour panel meeting session and used to create a final ranking for R&D options. First a general eligibility ranking for the various options for radioactive waste disposal was agreed upon. This involved the making of a more traditional decision matrix using the reduced list (table 2) of criteria. Next, the implications for this ranking using criteria explicitly related to research and development were used to determine the final ranking.

Table 1: Extended list of criteria

Initial criteria list for the evaluation of the 7 R&D options for RW disposal
• (human) exposure risk - (IM, IA) • terrorism - (IA) • transportation - (TF, AV) • retrievability - (IA, TF) • geological abundance of substrate - (AV) • availability requirements - (AV, PF) • environmental impact - (IM, AV) • catastrophe risks - (IM, IA, AV, PF, TF, FF) • monitoring and maintenance options - (TF, FF) • radionuclide mobility - (IM) • realization, operation, maintenance/closure cost - (TF, FF) • political acceptability - (PF) • geological hazards - (IM, IA, AV, PF, TF, FF) • rock stability for repository creation - (AV, TF) • groundwater table and flow - (IM, AV, TF, FF) • stability of governments - (PF) • communication to present and future generations - (PF,TF) • required engineered barriers - (IM, IA, TF, FF) • governmental stability - (PF)

Table 2: Reduced list of criteria

Reduced criteria list for the evaluation of the 7 R&D options for RW disposal	
• Immobility	(IM)
• Inaccessibility	(IA)
• Availability	(AV)
• Political feasibility	(PF)
• Technical feasibility	(TF)
• Financial feasibility	(FF)

2.2 Selection of criteria

The six main criteria categories to evaluate the characteristics of the various RW disposal options encompass one or more criteria in the extended list (table 1 and 2). Since the criteria of the long list fall in several of the six main criteria categories (as indicated between parentheses next to each criterion in the long list), the approach can be considered a fuzzy analysis of the various RW disposal research options. The analysis does not include the weighing of the suitability for R&D efforts, but rather, serves as input for R&D considerations. The six categories of criteria can be described as follows:

The immobility criteria consider the general containment of radionuclides in geological nuclear waste disposal, taking into account all possible pathways to exposure to the biosphere, humans in particular. These pathways include fluid transport mechanisms such as transport in water, oil or other media that could bring the waste in contact with people or increase background radiation to a level that exceeds the allowed dose. The geochemistry of the rock composition of a repository is a second factor that is discussed in its contribution to the mobility of waste. Additionally important is the proneness of a particular stratum or site to seismic hazards or alternatively, those features of the host material that will promise minimal transport of the waste in a seismic event.

The inaccessibility criteria include the protection against unwanted invaders of the repository and against the formation of unorthodox entry ways to the RW resulting from (geological) hazards. At the same time, inaccesability is also a factor hindering monitoring, maintenance, and retrieval and re-use. So depending on the choice of whether or not to allow accessibility, this criterion will have a high or a low score. The student panel is convinced that an unnoticeable closure and sealing of a deep repository is favorable for guaranteeing proper safety in the near and far future.

The availability criteria in the context of nuclear waste disposal refer to the global and local abundance of current disposal options. Also considered is the availability of a particular disposal option allowed by governments and the (longer-term) availability of other resources required for implementation.

The category of political feasibility criteria encompasses the likelihood that all stakeholders (e.g. governments, peoples, environmental movements and businesses) involved are in agreement and adhere to international or national regulations. The chance of regulations being modified or created to allow certain RW disposal solutions is generally unpredictable.

The issue of technical feasibility is often given a high weight in decision making processes. It focuses on physical realization as well as on available techniques. Also considered is the required technology to prevent malfunction or accident to such a degree that it is guaranteed that no major environmental disasters can occur. Technical feasibility is highly interconnected with the issue of financial feasibility, since most of the technical challenges are surmountable, as long as money is not an issue.

The financial feasibility criteria incorporates issues such as development, maintenance and closure costs, marginal cost, relative environmental cost, and the economics of scale. Usually financial feasibility is of high importance for the choice of a repository option, but not always in the right way. Often the tendency is to choose the cheaper option over the safest one. However, long-term safety is certainly worth the money.

3 Results and Discussion

3.1 Individual evaluation of the seven RW disposal options

After the panel was introduced to the theory and science of the various aspects involved in the disposal of RW, each member evaluated and characterized the suitability of their assigned disposal option in an individual report. The results were then summarized and can be found in the next paragraphs.

3.1.1 Mineral barriers

Containment in minerals, which is defined as the vitrification of nuclear waste in durable accessory minerals, can potentially achieve the goal of immobilisation of long-lived radionuclides that are present in nuclear waste. Accessory minerals are mainly ceramics such as zircon ($ZrSiO_4$), apatite ($Ca_5(PO_4)_3(F,Cl,OH)$) and monazite ($LnPO_4$), which are present in host rock in trace amounts. Accessory minerals will serve as a natural alternative to Synthetic rock (Synroc) or the already commercially used borosilicate glass, in packaging of Spent Nuclear Fuel prior to deep geological disposal. It is conceived that the combination of radioisotopes with minerals will create a more stable form of containment for the waste.

Most accessory minerals of interest are found in nature, commonly associated with a considerable amount of radioisotopes. Accessory minerals have a comparatively higher thermodynamic stability than borosilicate glass (a material currently used) and are durable under high amounts of irradiation, minimising radionuclide mobility [19, 28]. Additionally, minerals prove quite cost-effective in the long-term as a containment option as they are without a shelf-life and may provide a solution to the accumulation of mine tailings, which may serve as a ready source of accessory minerals. Moreover, the possibility of using extremophilic bacteria to incorporate nuclear waste in products of their metabolism that are analogous to mineral candidates, would prove an effective alternative to the mining of these minerals if the use of mine tailings as ores is unfavorable. It is thought that certain bacteria may be able to indirectly make use of the radioactivity of the nuclides as an energy source for their metabolism or alternatively, reduce isotopes to insoluble states [1, 3, 23].

There are a number of factors that pose a challenge to implementation. There still is a sustained need for deep geological disposal of radioactive vitrified waste and thus there will also be the need for repositories that provide suitable host rocks for the minerals; these will likely be rock types with which the minerals are naturally associated, potentially demanding an international site if suitable locations are not available nationally. An added challenge concerns the potentiality that several minerals or mineral aggregates will have to be employed as they differ in radionuclides with which they are naturally associated. Finally, most transuranic elements rarely occur naturally, so it is only possible to establish empirically their behaviour in accessory minerals, a process which is time consuming and expensive in the short-term.

3.1.2 Salt

Deep disposal in rock-salt is one of the more feasible and reasonable options. Isolation, costs, and availability are the main advantages. Thermal properties and economical exploitation of the salt are the main obstacles to realization.

In more detail, the advantages of using rock salt as a host rock for a geological repository are to following. Firstly, due to its high plasticity [7], isolation of the nuclear waste is relatively high. The speed of deformation is low enough to allow filling the site before it seals itself. Monitoring and maintenance of the site will therefore only be necessary for the first couple of centuries. Secondly, salt has a high thermal conductivity, reducing the risk of container overheating. Thirdly, the potential size of a repository in salt is to be large enough to offer an international solution, making it interesting for the national economy of the host country as well.

There are, however some challenges to overcome. Firstly, research has shown that in some cases the salt composition can be altered under the influence of radiation and heat. The altered composition can become explosive, thus posing a big threat on the safety of the site. This deserves high research priority, because it can make or break the case for geologic disposal in salt. Secondly, salt deposits have historically been exploited, meaning they have multiple access shafts/ramps. This poses a threat to both inaccessibility and environmental isolation. Thirdly, if water enters the salt dome, corrosion rates of containers, as well as radionuclide mobility, are high.

3.1.3 Volcanic Tuff

Extensive research has been conducted to investigate possible storage of radioactive waste in volcanic tuff. Most of this research was carried out in the USA for their proposed storage site at Yucca Mountain. However, many scientists still doubt whether the Yucca Mountain site is suitable as a RW repository and also the efficiency of tuff in general as a host rock for such a repository.

The arguments in favour of the Yucca Mountain repository mainly concern the possible aridity of the site. Because it is a mountain, the repository can be build above the ground water table. For a geological repository it is important that radionuclides do not reach the ground water table and vice versa; therefore this is a definite plus. Furthermore, the region also experiences very little precipitation, which adds to the argument that repository can be a very arid repository. Another argument is the possibility of a 'hot repository' in volcanic tuff. This means that the repository will not be cooled and will thus stay hot enough to vaporize water for several thousands of years. Finally Yucca Mountain is isolated from major population centres and for that reason there is no strong NIMBY response [8].

Critics of volcanic tuff as a host rock and of the Yucca Mountain site claim that because the number of years during which a repository has to function is so large, there is no way of guaranteeing the region will stay as arid as it is now. A possible increase in precipitation (which is not unlikely according to future climate models) could be fatal, for water can travel through the faults and cracks present in tuff [10, 33]. Therefore, the engineered barriers for such a repository have to be able to withstand thousands of years of corrosion. There is no way to test this in a laboratory. Last but not least, there is the threat of a volcanic eruption. While the chances of such an eruption appear to be small [8], the potential damage it could cause is tremendous.

3.1.4 Sub-seabed

Sub seabed disposal appears to be a good option for nuclear waste disposal, since the distance from mankind is large. When buried within oceanic sediments, the material would not be easily accessible for the so-called „dangerous“ parties. The waste would be laid to rest beneath 5000 meters of waters and several hundreds of meters of clay-rich mud. The sediments of the chosen sites have characteristics that are favorable for the burying of radioactive materials. The thick muds generally have very fine particles and therefore have a low permeability. In case a canister breaks open, the chances are small that the radionuclides will be brought up to the ocean floor. The adsorption capacity of the sediments for radionuclides is high [14], lowering even further the chance of their release. The ocean functions as a potent dilutor.

In theory, the technology that is needed for sub seabed disposal of nuclear waste is already available. Deep-sea drilling techniques could be used to place the containers hundreds of meters underneath the sea floor [38]. Radioactive waste that is disposed under the seabed must be packaged in corrosion-resistant containers or glass. Sub-seabed disposal utilizes sites in the ocean that are most stable and predictable, which usually means on the abyssal plains. The selected locations are far from any plate boundaries and would minimize chances of disruption by turbidites, volcanoes, earthquakes and any other seismic activity.

Future research would have to focus on in situ testing, for it is important to get a detailed overview of the properties of the site as well as the behaviour of the radioactive waste. More studies need to be done on the method of placing (and retrieving) the containers. It is very important that the canisters are not damaged during their placement (and possible retrieval) and that the material for the canisters can withstand the pressure difference between the surface and the ocean floor.

The idea of sub-seabed disposal might be easier to sell to society than land-based disposal, due to the „not-in-my-backyard" syndrome. For seabed disposal, technical matters are not the main source of concern. Several technical solutions can be found for the problems that come with sea bed disposal. International laws that impose general obligations upon states to protect the marine environment form a bigger challenge. The most important restrictions derive from the London Dumping Convention (LDC) which states that „any deliberate disposal at sea of wastes or other matter from vessels, aircraft, platforms or other man-made structures at sea" [23] is prohibited. The challenge is to come to a general agreement with all nations to allow sub-seabed disposal.

3.1.5 Granite

Next to sub-seabed, granite is the most abundant option for a geological repository. Since it is not abundant in every country [32], international cooperation would be required if granite is to be the only repository option in use. One of the reasons why granite is a promising host rock for disposal of RW is the low mobility of the radionuclides due to the low porosity and permeability. Weathering usually concerns the top 50 meters, while a repository is located much deeper. Faults in the rock can be monitored and thus a site with as few faults as possible can be selected. All major cracks are formed during the forming of the granite and those sites are the most dangerous when it comes to further cracking. Sites with only small cracks have a much lower risk of cracking further [5].

The accessibility of granite is low, since it is a hard rock in which drilling is difficult and expensive. This makes it expensive to build a repository in granite, yet the high safety of the repository can outweigh the high costs. In view of terrorism, the low accessibility is a positive feature.

3.1.6 Mine Tailings

The residues from mining and milling operations are called mine tailings [25]. Uranium mine tailings contain hazardous elements as radium, radon, thorium, arsenic, selenium and molybdenum [1], which will leak into the environment if the mine tailings are not properly disposed of. In high concentrations elements like Se, As, Mo are toxic.

Mine tailings are disposed of on-site, in the vicinity of the mine, for a number of reasons. Firstly, they have a large volume, rendering transportation infeasible. For example, the Mary Kathleen uranium mine (Australia) produced a total of 31 Mt of material from open pit mining [2]. The tailings slurry from acid leaching of uranium from the ore was pumped into a retention area of about 30 ha, 1.5 km from the mill. Secondly, there are regulations, governing the rehabilitation of mining sites. For example, the multiple layer concepts [20] are applied when tailings are to be disposed of in tailings impoundments.

On the other hand, on-site disposal of mine tailings poses a few problems. Firstly, in areas where the mine is located near a river, a slurry pipeline has to be built to relocate the tailings to another site deemed suitable. For example, the Moab (Utah) uranium mill tailings pile located adjacent to the Colorado River, released contaminants by seepage into the river and the feasibility of relocating the tailings to a more qualified disposal site using a slurry pipeline was explored [16]. Secondly, bringing back tailings to the dug out mine (for storage) necessitates their treatment, otherwise radon gas emission will continue and the risk of groundwater contamination will be enhanced [39].

3.1.7 Clay

Many natural analogues suggest that clay layers are a suitable substrate for building a repository for RW. Examples are Alligator River (Australia) [15], the Cigar Lake Uranium ore deposits (Canada) and the Oklo natural reactors (Gabon) [2, 9], and Pocos de Caldas (Brazil) [2], with regard to long-term U migration; Loch Lomond lakebed (Scotland) [30], with regard to middle-term ^{129}I migration; clays at Orciatico (Italy) [11], with regard to thermal alteration; preserved wood at Dunarroba (Italy) [35], with regard to clay's isolation capacity and biodegradation processes; and the Littleham Cove native copper deposits (England) [31], with regard to the potential stability of copper canisters.

There are several reasons why clays deserve research preference for disposal of RW. First of all, clay has the ability to self-heal, which gives it great advantages over non-plastic rocks: in case of an earthquake it can re-seal the repository; stress from geologic deformation is buffered; it can heal the Excavation Disturbed Zone. Compared to all plastic rocks and sediments, it can do so the quickest due to high creep rates [18]. Secondly, modelling of processes in clay like flow, radionuclide diffusion and retardation, and mechanical stress distribution, is more straightforward than in other media and thus requires less research. This means that less exploration effort may need to be done than for some of the other options, and that possible failure of the Engineered Barrier System under the strongly buffering conditions of a clay layer is very accurately predictable. Therefore, the spread of radionuclides after Engineered Barrier System failure in a water-saturated plastic medium is more homogeneous than in fractured systems or non-saturated plastic media. Thirdly, the retardation in clays due to sorption is more efficient than in rocks. The anoxic conditions in clay are perfect for preserving canisters and for immobilizing most important nuclides.

There are also some challenges which still need to be met. Some anionic radionuclide species (^{129}I, ^{36}Cl) do not interact with the clay as a Natural Barrier [18]. Therefore, the Engineered Barrier System needs to be designed to retain these radionuclides, and possibly also some other cationic radionuclides, because their sorbed mobility may be quite high (as already known of Ce and Sr) [4]. Also, more study on the behaviour of different types of clay under repository pressure and temperature is needed. Furthermore, knowledge in areas like hydraulic conductivity, creep rate, gas transport pathways and mechanical strength, is far from complete. The bigger challenge while this research is on-going, is to develop a uniform international reference library of Safety and Performance Analyses for clay disposal sites.

3.2 Initial Ranking and Review

The composition of the panel members was quite diverse; the seven students follow –besides geosciences - different programmes within the science curriculum of the Liberal Arts and Sciences College and take various courses in social sciences (e.g. law, argumentation and analysis) and arts and humanities (e.g. music) as well. The panel can therefore be characterized as broad minded and free-thinking.

Since the wide range of issues and theory to be addressed had to be introduced in a consecutive manner, it took about 6 of the total 10 weeks prioritization procedure before the panel was up to speed with analysing and evaluating their cases. The decision making part for the ranking of the options was relatively quick, especially after the criteria had been summarized into six main categories. The long list of criteria (table 1) was not workable for extensive comparison, especially since the process was not automated.

During the first ranking in which a two-step procedure was followed the effects of differences in panel voting methods were explored, one being the summation of individual rankings of the seven options and one being the democratic vote on the ranking of the options after a debate of the summation results. As it turned out there was no change in the rankings. The initial ranking was as follows: 1. Sub-seabed, 2. Salt, 3. Clay, 4. Mine tailings, 5. Granite, 6. Mineral barrier, 7. Tuff.

This ranking was submitted for external review in order to give the inexpert panelists an opportunity to react to expert commentary in reconsidering their choices. The review process took the form of an interactive poster where experts could give their ranking based on the six criteria (Reposafe, November 6-9, 2007). The poster presented the condensed versions of the option summaries given above together with an invitation to deliver any comments to the panelists who monitored the poster. A fair number (>13) of experts participated in this process. The most notable result was that the R&D prioritization for sub-seabed and mine tailing varied significantly among reviewers. Sub-seabed disposal was either supported as a top priority (in agreement with the panel's first ranking) or recommended for demotion to lowest priority, a strong indication of the difficulty in achieving large scale consensus on that option. Reviewers dismissed panel concerns about possible explosive effects of strong heat sources on salt. Disposal in granite received strong support because of the advanced state of the development of this option. More surprising support was received for a high prioritization of mine tailings disposal because of the possible near-term instability of many storage sites.

There were also important comments about the general nature of the prioritization process. Several reviewers remarked upon the consistency of the criteria and ranking presented with those that had resulted from much more extensive processes (e.g. CoRWM). The panel's support for an international solution was considered noteworthy. Finally, it became clear that a more explicit distinction should be drawn between the criteria used to rank disposal options and those used to prioritize options for research and development.

3. 3 Revisions and final ranking

The panel reconvened a week after the external review session in order to discuss the expert comments and determine a revised and final ranking. At this meeting a simple decision matrix was developed (Fig. 2), which can be considered an additional step to the two-step method in stage 5 of the prioritization process. In constructing this matrix, the panel agreed to apply the criteria to the ranking of the options as international disposal alternatives. Subsequently, the panel then defined R&D criteria that could be used to perform a post-hoc adjustment of the disposal option ranking to produce a ranking of research and development options. These criteria for R&D prioritization were agreed upon by all panel members.

Although R&D criteria could be added as seventh criteria category and be weighed equally together with the other six main criteria, the panel chose instead to evaluate the balance of the six main criteria and the weakest and strongest criterion for each option therein as one of the considerations for R&D. Other criteria for prioritization of the options for R&D are the time-frame in which a solution needs to be found. With existing plans to build another 32 nuclear plants, solving the waste problem associated with the nuclear resource cycle is an urgent matter. Therefore, if a relatively small research effort is needed towards realization or improvement of an option, that option could have R&D priority over an option with a higher ranking based on the six main criteria, but requiring extensive research. The present situation of the waste is also a consideration. Does the waste pose an immediate threat, or can we buy time with intermediate solutions while constructing permanent solutions? Obviously, if political consensus (international or national) is conceived to be a time-consuming process for a certain option, than despite its qualities, the option may be ranked lower than based on its suitability alone. Another consideration is that a more generic solution is preferred, so, options that are considered good but are scarcely available or not uniform in character, should be considered later than options that are more abundant. Finally, resources (energy and materials) for R&D and implementation should not outweigh the energy produced by a nuclear power plant. The amount of money required for R&D is given a low weight in the prioritization of R&D options, as the panel states that safety is far more important.

	Clay	Granite	Minerals	Salt	Seabed	Tailings	Tuff
Immobility	+++	++	+++	+++	+++	+	-
Inaccessibility	+	+	0	-	++	- -	0
Availability	+	+++	++	+	+++	na	- -
Political Feas	0	+	+	+	- - -	++	-
Tech. Feas	+	+	-	++	+	+++	++
Finan. Feas	0	-	-	++	- -	++	++
R&D ranking	**3**	**1**	**4**	**2**	**5**	**1**	**6**

Fig. 2: Criteria matrix for the 7 RW disposal options; +++ = very high, 0= neutral, - - - = very low, na = not applicable. Subsequent R&D ranking is based on considerations explained in the text

Together with the incorporation of the external comments about individual options, the clear distinction drawn between the criteria used to rank disposal options and those used to prioritize options for R&D resulted in a different ranking from the initial ranking (Fig 2). Most notably is that the sub-seabed disposal dropped in ranking significantly, since the panel became much more pessimistic about reaching political consensus within a short time-frame about the choice of ocean sites, the disposal methods and operational procedures including those issues surrounding RW transport over longer distances. If retrievability becomes a requirement in the future should transmutation be a possible and desired option, the high inaccessibility score may be a negative instead of a positive feature. The geologic disposal in granite and salt rose in ranking, because these options appear to be already well and successfully developed (granite in Sweden and Finland, salt in Germany). Granite received preference over salt due to its wider abundance and much more limited accessibility. In addition, this option could become a strong international solution with further development. It was decided that mine-tailing disposal would have to be considered separately from the ranking of the other options. Mine tailings pose an immediate problem to the environment and are not contained. Since the problems of disposal at both ends of the fuel cycle would be significantly increased by a switch from fossil fuels to nuclear power, taxing the limits of the temporary storage mechanisms now in place, rapid development of solutions to both problems seems a necessary condition for the consideration of this particular carbon alternative. This problem needs to be solved simultaneously with the search for repositories and barriers and is therefore also ranked on top. Due to the relative limited abundance of clay and its lack of uniform composition, and due to the vast amount of research that would need to be carried out and doubts about its technical feasibility to use minerals as an engineered barrier, these options received intermediate rankings. Tuff was again ranked last of all, because of the limited availability and because of climate change concerns.

4. Conclusions

The general recommendations of the student panel concerning R&D for RW disposal are perhaps best summarized by two short imperatives: act now; act internationally.

The first imperative refers both to the importance given to addressing immediate concerns (mine tailings) and to the ranking of the options. The panel noted that climate change adds to their sense of urgency. The panelists sense of urgency was further enhanced by the overall consistency of their rankings with those expressed by poster viewers and in the findings of much larger committees such as CoVRM. This was seen as evidence that consensus support for practical options is achievable in the short term.

The second imperative refers to an emphasis on the development of options that provide solutions for most or all countries dealing with RW disposal. Mine tailings and granite disposal again serve as examples. The former are present in a number of presently less developed nations (e.g. Gabon). The latter is available in many parts of the world so that development of the technology could benefit many nations (and could allow for a rapid expansion of capacity if nuclear energy use grows). At the same time, granite is an example of a solution that would require international cooperation for nations such as The Netherlands, whose only internal geologic disposal options were viewed as less safe by several panel members.

Additional observations concern the role of technical information in the process of building consensus among an inexpert panel. Good access to the latest research proved to be a deciding factor in prioritizing options. The updating of the knowledge base of the panel via interaction with expert reviewers proved to have a significant impact and seems to have made it easier to achieve consensus. This was reassuring, in that there had been some concern that the injection of new information would confuse the panelists. It was also clear that too much focus on studies to determine the absolutely optimal solution was not considered helpful. Once a class of options was shown to be similar with respect to issues of safety, the focus of the panel shifted away from technical details in the literature and toward progress reports that helped identify the option that was best developed. (parallel to geologic over borehole listing in CoRWM chapter 11)

In conclusion, given the political simplicity of the principle of urgent and international responsibilities; and given the successful use of technical literature by an inexpert panel in forming a ranking consistent with that of larger efforts such as CoRWM; it seems likely that development priorities like those given here can be implemented with consensus support in the near future, as long as efforts are made to provide broad access to the relevant information.

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Environmental Impact Assessment of a Final Dis-po-sal (High Level Active Waste)

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Abstract

An Environmental Impact Assessment (EIA) of a final disposal site of High Level Active Waste is in Germany a legal obligation. The results of the environmental impact assessment are important for the acceptance of a final disposal site by the public. For transparency in a public discussion, a common understanding about the investigation area and assessment procedure of the environmental impact assessment is required.
During the construction phase of a final disposal site relevant environmental impacts are expected, caused by land use as well as the noise and air pollution caused by transport vehicles. Environmental impacts during the operational phase of the final disposal site are relatively low, especially of non-radiological effects, and can be compared to big facilities for interim storage of radioactive material. Thus an EIA for a final disposal site is feasible.
To minimize environmental impacts, especially non-radiological impacts during the construction phase, it is important that comparable alternative sites, alternative routes for transportation and alternative technical methods exist.
This article shows only an extract of the Öko-Institute's investigations on EIA of final disposals, a topic in which the Öko-Institut gained a lot of experience by reviewing many EIA-reports of nuclear facilities for different German Authorities.

Definition of the area around a final disposal for the investigation of the environmental impact

The area, in which the environmental impact of a final disposal site has to be investigated, is the area in which the emissions of different activities of the final disposal can be measured as an immission. The area of investigation of the environmental impact is a spaciotemporal function.

The spatial dimension of investigation for environmental impact is an area in which immission exceeds a de-minimis-level. This de-minimis-level has to be defined for every kind of immission. In an ideal case it is a spherical circle around the source of emission (e.g. irradiation of a single source without shielding). Immissions in the underground do not necessarily have an impact on subjects of protection such as man, the health of man, animals, plants, biodiversity, soils, water, air, climate or landscape. A harmful effect on subjects of protection only occurs when the source of emissions is not buried deep enough in the underground (e.g. noise) or when contaminated matter is able to reach the biosphere e.g. via groundwater.

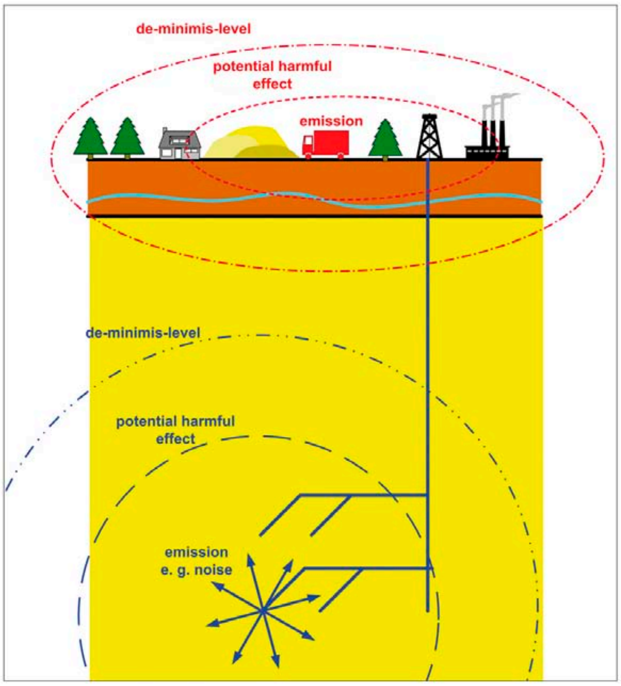

Fig. 1: Examples for different spatial dimensions of different sources of emission, different de-minimis-level and different potential harmful effect

The temporal dimension which has to be considered for the investigation of the environmental impact is the time span during which immissions, exceeding the de-minimis-level, impact subjects of protection.

Significant harmful effects on subjects of protection depend on the intensity and the lifespan of the source, on the existence of the subjects of protection in the area of immission and on the sensitivity of the subjects of protection.

Short-term impacts with low intensity are not as significantly harmful as long-term impacts with nearly the same intensity (e.g. drawdown of groundwater level for short or longer periods). Impacts with extra high intensity within short periods can have more harmful effects than impacts of the same kind but with lower intensity because of dissemination of intensity over a longer period (e.g. high intensity of impact because of much more transport vehicles over a short period versus the same amount of transport figures disseminated over a longer period).

The environmental impact of a final disposal site depends on the different activities as well as the intensity and the duration of activities during different phases of a final disposal site. A final disposal site can be subdivided into six phases:

Phases of a final disposal	Duration
Investigation	ca. 8 years
Construction	ca. 8 years
Operation	ca.50 years
Sealing	ca. 8 years
Post closure	up to 1 Million years

Examples of potential environmental impacts which differ from phase to phase of a final disposal

Land use

The potential negative effect of land use depends on the value of land for animals and plants, especially protected species. Land use during the aboveground investigation depends on the amount of drilling investigation and the necessity of new infrastructure. To drill more than 100 m deep in the underground a land use of 2.000 m^2 is necessary per deep borehole /Thiele 2005/. The amount of land use during aboveground investigation can be estimated to be around 10.000 m^2.

During the underground investigation phase land use is higher because buildings for a mine are needed aboveground and the infrastructure has to be extended. The land use for buildings and infrastructure is about 40.000 to 70.000 m^2. Additional land use for the stockpiling of excavated bedrock is the same amount during the underground investigation phase. In summary during die phase of underground investigation die land use is about 110.000 m^2.

The construction phase of final disposal site causes the highest land use because buildings and infrastructure will be extended for handling of radioactive waste and the area for stockpiling of excavated bedrock will increase. The additional land use is estimated at 200.000 m^2.

To minimize the land use of valuable country side during the construction phase, investigations of alternative sites for the final disposal are necessary to compare the impact of land use with respect to the value of the local nature.

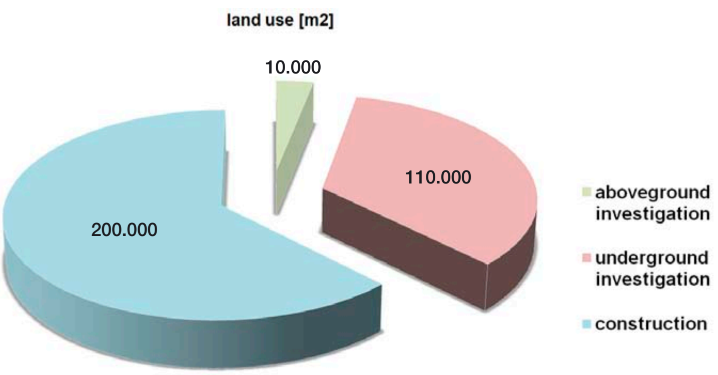

Fig. 2: Estimated land use during different phases of a final disposal site

Air pollutants, dust and noise caused by vehicles and construction site equipment

Vehicles, especially lorries and construction site equipment, cause air pollutants like nitrogen oxides, sulfuric oxides and carbon particulate emission in the vicinity of the final disposal site and along the transport routes. These can result in impacts on the surroundings and nature. The significance of the effects basically depends on the intensity of the impact and the distance between the source of emission and the subjects of protection.

Air pollution, dust and noise are very relevant during the phase of underground investigation and the construction of the final disposal site, because during this relative short time period the greatest amount of bedrock will be excavated and transported from the mine to a stockpile. During the operation phase the amount of transports per week is not as high as during the construction phase because the backfilling of bedrock depends on the time for the encapsulation of radioactive waste into the bedrock. For the encapsulation of canisters into the bedrock additional time is needed for radiation protection measures for workers, for the building of technical barriers around the canisters and for the quality control.

Assuming that 60.000 to 110.000 m^3 bedrock have to be transported within two years necessary for the underground investigation, 20 to 26 lorries per week will be needed for the transport. During the construction of the final disposal site for a time period over 8 years, 800.000 m^3 bedrock und 150.000 m^3 concrete must be transported /Nagra 2002/. Under these conditions transport figures of 37 to 58 lorries per day are expected.

During the operational phase of a final disposal site the necessary amount of transports is the same as during the construction phase but the period of transportation lasts some decades without peaks of traffic.

Environmental impacts by immissions caused by transport vehicles can be minimized by comparing alternative sites for a final disposal and stockpiling, alternative transport routes and alternative transport facilities. This approach was taken in Finland where four different final disposal sites and alternative transport routes were taken into consideration. For the environmental impact assessment of final disposal in Finland four sites for final disposal and additional different transportation routes for every site had been considered /Posiva 1999/.

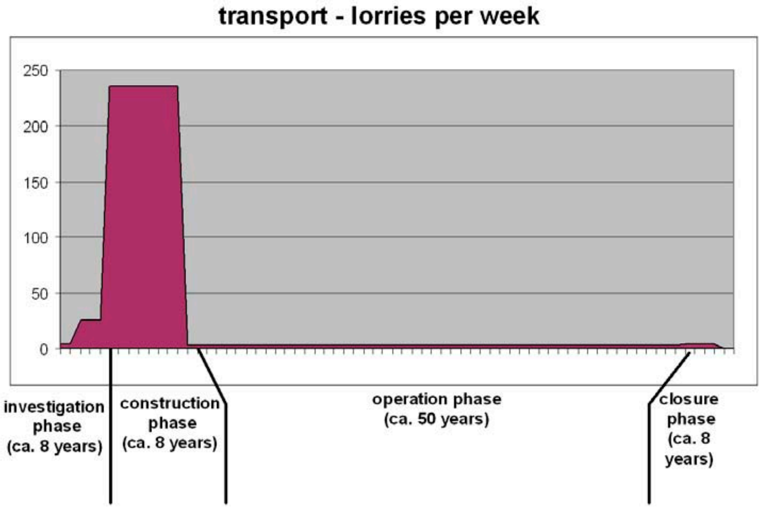

Fig. 3: Transport figures, lorries per week during different phases of a final disposal

Conclusion

The intensity of environmental impact differs during the several phases of a final disposal. Especially during the construction phase non-radiological environmental impacts as transport emissions and land use are significant. The result of an environmental impact assessment depends on the intensity and duration of emissions as well as on the existence and sensitivity of subjects of protection in the area of investigation. An Environmental Impact Assessment of a final disposal site is feasible but a minimization of impacts is requested and is possible by comparing alternative sites for the final disposal, alternative routes for transportation (especially of bedrock) and alternative facilities/vehicles for buildings and transportation.

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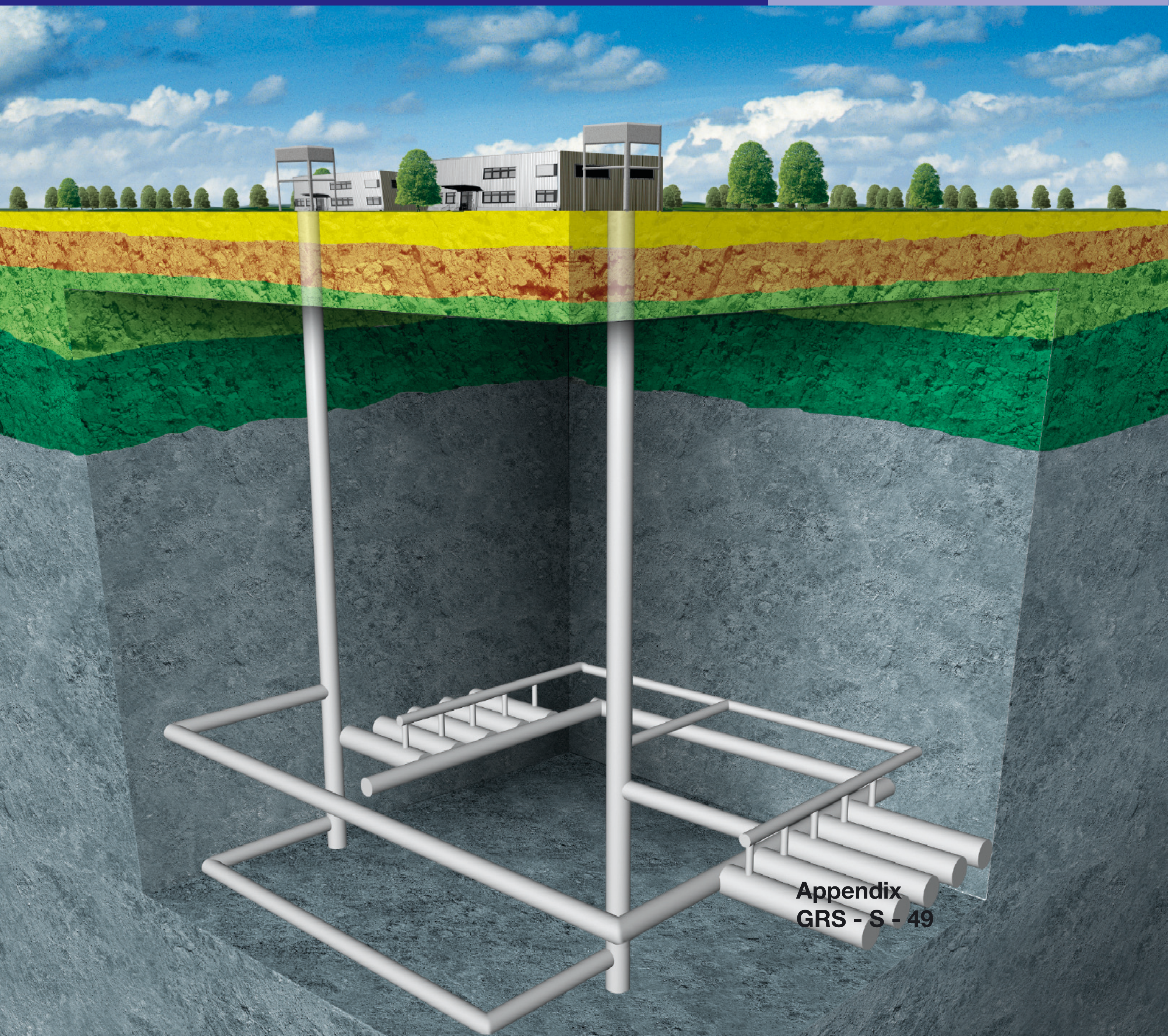

Appendix
GRS - S - 49

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Appendix
GRS - S - 49

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Contents

A Realistic Approach for Assessing the Long-Term Release of C 14 from a Closed Final Repository for Low-Level Radioactive Waste in a Salt Mine

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Abstract
The contribution of C-14 to radiation exposure in the biosphere can be significant as compared to that of other radionuclides disposed in a repository for low-level radioactive waste. The release pathways of C-14 and processes relevant to its release from a closed final repository for low-level radioactive waste are discussed. Because a conservative approach may lead to undue overestimation of the potential radiation exposure, a more realistic approach is outlined. At the present level of refinement, our realistic approach provides a sufficient safety margin to German federal limits for radiation exposure to demonstrate compliance with the ALARA (as low as reasonably achievable) principle and can thus facilitates licence approval.

1 Introduction

The proof of long-term safety of a final repository for low-level radioactive waste requires an assessment of the potential radiation exposure from gaseous radionuclides. C-14 is the only radionuclide that can contribute significantly to radiation exposure via gas pathway as all other gaseous radionuclides can be neglected due to their short half-lives, low inventories or low radiological relevance.

Sources and environmental behaviour of C-14 have been summarized previously [7].

Hitherto only conservative approaches have been used for estimating the mobilisation, migration and release of C-14 via gas and groundwater pathways from a repository. The results of these approaches generally comply with German federal limits for radiation exposure.

Despite compliance with the limits for radiation exposure, these approaches may provide insufficient safety margins to demonstrate compliance with the ALARA (as low as reasonably achievable) principle. Overestimation of the potential radiation exposure may therefore lead to false conclusions and ineffective measures for minimising radiation exposure.

The following paper outlines a realistic approach for an estimation of long-term radiation exposure by C-14. The approach is based on a model final repository for low-level radioactive wastes in a salt mine. A realistic approach provides an additional safety margin to radiation exposure limits and can thus facilitate licence approval where compliance with the ALARA principle has to be demonstrated.

2 General Background

2.1 Scenario

The scenario for a model final repository in salt rock includes a hydraulic connection between the mine building, the overlying rock and the biosphere. Driven by rock convergence, gas and brine will be squeezed out of the repository and penetrate into the overlying rock. This process takes place on a geological time scale. There is no (or only a negligible) leakage of gas via the mine shafts. Radionuclides are retained in the near-field by limited dissolution and sorption.

The detailed scenario includes further features, events and processes such as:

- retarded exchange of fluids between the emplacement chambers due to engineered barriers
- time dependence of geochemistry in the emplacement chambers

- mobilisation of radionuclides
- retardation of radionuclides by sorption and precipitation
- gas generation and accumulation in emplacement chambers
- squeezing-out of fluids due to convergence and gas generation
- transport of radionuclides by density-driven convection of fluids
- release of gaseous radionuclides
- migration of gas into the overlying rock
- preferential fluid transport along fracture zones in the overlying rock
- transport retardation and dynamic storage of contaminated fluid
- dilution of the fluid by ground and surface water with lower salt content.

2.2 Geochemistry

The geochemical environment (i.e. the concentration of major and minor constituents of the brine, the prevailing solids, and the gases of an emplacement chamber) results from the interaction of the brine with the waste matrix materials (cement/concrete), the waste packages (concrete, iron, corrosion products), the backfill materials and the degradation products of organic matter of the waste itself. The conversion of organic matter to CO_2 and CH_4, the degradation of cement and the corrosion of metals are time-dependent processes that determine the composition of the brine. Although precise degradation rates of cement and conversion rates of organic matter to CO_2 under the potential repository conditions can not be known in sufficient detail, CO_2 will predominantly precipitate as carbonate. As a result, a significant change of the initial pH dominated by cement in the waste matrix, in the waste container and in the backfill of emplacement chambers is not expected to occur.

Equilibrium constants are known for alkali and alkaline-earth carbonates, carbonate complexes and mixed solid phases. A nearly complete data set of Pitzer constants is available. This data is used for the geochemical modelling of brines. The main carbonate compounds dissolved in brine are $MgCO_3$(aq), $CaCO_3$(aq) and CO_3^{2-}/HCO_3^- . Precipitation of solid phases such as magnesite, calcite and dolomite limit the concentration of dissolved inorganic carbon.

The typical range of concentrations of carbon expected in brine in emplacement chambers, assuming a closed system, is shown in Table 1. The concentration that would be obtained by dissolution of the complete inventory is shown for comparison. The dissolved fraction of inorganic carbon represents only a small amount of the total carbon inventory and varies. The major amount of carbonates is fixed in the solid phase.

Table 1: Typical concentration range of dissolved carbon and carbonates in brine in the emplacement chambers

Speciation	Concentration (10^{-5} mol/(kg H_2O))
C (inventory, hypothetically dissolved)	200 000 – 3 000 000
C (total dissolved inorganic C)	0.8 – 40
CO_3^{2-}	0.04 – 1
HCO_3^-	0.000001 – 3

Isotopic effects on chemical reaction rates are ignored, i.e. compounds containing radioactive C-14 are assumed to have identical reaction rates for equilibration, precipitation, sorption and gas generation as compounds having non radioactive C.

Isotopic dilution occurs via exchange processes. The isotopic exchange of C-14 in carbonates and bicarbonates takes place rapidly in brine via hydrogen exchange.

$$^{14}CO_3^{2-} + HCO_3^- \leftrightarrow CO_3^{2-} + H^{14}CO_3^-$$

Isotopic equilibrium between the gas phase and the fluid phase is also rapidly achieved.

$$^{14}CO_2\ (g) + HCO_3^- \leftrightarrow CO_2\ (g) + H^{14}CO_3^-$$

The isotopic equilibration of solid phases and brines may be slow, as the kinetics of dissolution and precipitation are controlled by accessible surfaces. The isotopic equilibration of other compounds such as hydrocarbons, fatty acids, or alcohols is controlled kinetically and may be slow, too.

2.3 Inventory of C-14

The inventories of C-14 in emplacement chambers are fixed on completion of the waste emplacement. Additional C-14 does not build up in low-level radioactive waste (e.g. via decay chains or nuclear reactions). The inventory is estimated to cover all uncertainties. Care is thereby taken not to produce an overly conservative estimate.

The inventory of C-14 and its concentration in the emplacement chamber may vary within several orders of magnitude between different emplacement chambers. Table 2 depicts a range of inventories and concentrations of C-14 of a repository in salt rock. The C-14 inventory and concentration vary typically by a factor of 10 at most. However, the inventories and concentrations in some emplacement chambers are outside of this range. To assign the maximum concentration to all emplacement chambers (as some models do) would be overly conservative.

Table 2: Inventory and concentration of C-14 in emplacement chambers

Emplacement chamber	C-14 inventory (10^{11} Bq)	C-14 concentration (10^{8} Bq/Mg)
lowest	0,02	0,05
typical range	0,5 – 5	1-10
highest	10	35

The typical isotopic ratio of C-14 to total carbon in the low-level waste equals 10^{-8}-10^{-9}. To a first approximation, C-14 can be assumed to be homogeneously distributed with a similar isotopic ratio in all carbon species in the brine due to isotopic exchange.

2.4 Speciation of C-14

The waste can be characterised in terms of organic / inorganic / metallic and non-metallic fractions as these waste fractions are documented. The organic fraction typically accounts for 10-30 % of the waste. However, the assignment of the C-14 inventory to each of these fractions is unavailable from the waste documentation. Therefore, a working assumption has had to be made.

It is generally assumed that the C-14 inventory is bound to organic matter. This assumption ignores that C-14 can be bound to solid phases e.g. as carbonates in concrete or carbides in ashes or activated impurities in metals and on metal surfaces. If C-14 were homogeneously distributed in the waste, a significant amount (70-90 %) would not be contained in organic matter.

The speciation of organic C-14 may differ from that of the organic fraction, but will underlie similar degradation processes. As organic matter does not completely convert under saline conditions [2], a large fraction of C-14 will thus remain within the deposited waste and will undergo radioactive decay. This means that the release of C-14 will be significantly reduced compared with the predictions of conservative models that assume a complete degradation. For a better quantitative assessment the chemical speciation of C-14 should be known in detail.

In summary, the assumption that the C-14 inventory is bound to organic matter rather than homogeneously distributed in the waste is conservative.

3 Release of C-14 from the Waste

The following time-dependent conditions govern the release of C-14 from the deposited waste:

- waste and waste packages have been exposed to air since emplacement (initial condition)
- oxidising conditions favour degradation of organic constituents (transient condition)
- waste packages and waste products are exposed to anaerobic, cement-conditioned brine (prevalent long-term condition)

The release of C-14 under the initial and the transient conditions is minor as compared to that under the long-term condition. Thus the emphasis of further considerations will be on the latter.

The release of C-14 from deposited radioactive waste depends on its chemical speciation. The important processes are:

- Degradation, dissolution or desorption of organic compounds; Under anaerobic conditions microbial degradation is the major release mechanism of C-14; However, according to some studies [2], the microbial yield will be limited; The degradation of organic matter produces intermediates; These intermediates are comparable to substances from standard waste dumps under anaerobic conditions and consist mainly of glucose, amino acids, fatty acids, hydroxycarbonic acids, isosaccharinic acids etc. [3]; The final products are carbon-bearing gaseous compounds (such as CH_4, CO_2, hydrocarbons and other volatile compounds) and aqueous compounds (such as $CO_2/HCO_3^- /CO_3^{2-}$ and carbonate complexes);
- Dissolution of surface contamination (e.g. crud); The final products are aqueous compounds;
- Release via corrosion of metals; The final products are some carbon-bearing gaseous compounds and aqueous compounds; However, the progress of corrosion is slow as compared to microbial degradation;
- Lixiviation (leaching) from metals; this process was studied on Zircaloy hulls and activated core parts of LWRs [5]. However, activated core parts are hardly present in a repository for low-level waste, and if any, release irrelevant amounts of gaseous C-14.
- Hydrolysis of carbides (e.g. Al_4C_3 contained in ashes) on contact with water instantly releases hydrocarbons:

$$Al_4{}^{14}C_3 + 12\ H_2O \leftrightarrow 4\ Al(OH)_3 + 3\ {}^{14}CH_4$$

- Methanogenesis:

$${}^{14}CO_2 + 4\ H_2 \leftrightarrow {}^{14}CH_4 + 2\ H_2O$$

Hydrogen needed for this process is generated by anaerobic corrosion of metals, whereas CO_2 comes from aqueous compounds; Methanogenesis is supported by microbial activity and catalysts; Due to the consumption of hydrogen, the total amount of gas is reduced by a factor of 4 as a side-effect; In the case of significant methanogenesis, this has a profound influence on the transportation of fluids; However, due to its low enthalpy, methanogenesis is not favoured over other reactions.

Radioactive waste repositories during their operational phase show a continuous release of C-14 by mine ventilation [6]. Radiation exposure is of limited concern, but released C-14 from deposited waste accounts already for a fraction of at least 2 % within 30 years of operation. This provides evidence of ongoing reactions in the waste. According to some studies, 75 % to 90 % of the C-14 is ${}^{14}CO_2$. The remaining fraction is dominated by ${}^{14}CH_4$. The domination of ${}^{14}CO_2$ proves aerobic conditions in the waste that prevail during the operational phase (initial condition, cf. above).

4 Transport of C-14 to the Biosphere

C-14 is transported from the emplacement chambers to the biosphere via gas and brine. CH_4 is poorly soluble in brine. In contrast, CO_2 dissolves readily at the prevailing conditions in the repository (pH, pressure, temperature) to form aqueous compounds. The concentration of aqueous $CO_2/HCO_3^- /CO_3^{2-}$ and complexes is determined by precipitation of carbonates and sorption on solid backfill [4]. Saturation of the brine with CO_2 resulting in release of gaseous CO_2 is not expected to occur.

As a result, C-14 is transported via the brine pathway predominantly as aqueous ${}^{14}CO_2/H^{14}CO_3^- /{}^{14}CO_3^{2-}$ and carbonate complexes, whereas via the gas pathway it is transported predominantly as ${}^{14}CH_4$. Gas and brine undergo chemical and mechanical interactions with each other and with the solid phase that they encounter during transport. The interactions determine the release rate of C-14 to the biosphere.

Isotopic dilution occurs when the C-14-bearing brine is transported out of the emplacement chamber and equilibrates with carbon containing solid phases [1] that are free of C-14. Due to the composition and the relative amount of brine and mineral phases, most C-14 will be fixed as carbonates in the solid phase. Some C-14 dissolved in the brine may be bound to compounds other than aqueous $CO_2/HCO_3^- /CO_3^{2-}$ and carbonate complexes and thus undergo different processes. The amount of such compounds is not expected to be significant.

Gaseous C-14 is mostly in the form of CH_4. Gaseous CO_2, if any, will undergo isotopic dilution that lowers its content of C-14. The release of C-14 from the mine via gaseous compounds may be hindered by their oxidation to aqueous $CO_2/HCO_3^- /CO_3^{2-}$ when transported through sulphate-containing backfill and rock (e.g. gypsum overlying a salt rock formation) in presence of water. Most of this aqueous $CO_2/HCO_3^- /CO_3^{2-}$ will precipitate as carbonate. The retardation of gas transport in the mine by technical and natural barriers gives more time for these processes and the radioactive decay to take place and thus lowers the released fraction of C-14 to the biosphere.

5 Conservative Approach

Hitherto the following scenarios have been considered in more or less conservative approaches, which release the inventory of C-14 to the biosphere:

1) The long-term assessment of a potential radiation exposure by $^{14}CO_2$ yielded an insignificant dose for both pathways (brine and gas). The dose limits are easily met with a large safety margin. Therefore, the exposure by $^{14}CO_2$ need not be further discussed.
2) The contribution of non-oxidised $^{14}CH_4$ reaching the biosphere and causing potential radiation exposure as $^{14}CH_4$ is also insignificant (see Fig. 1a).
3) The contribution of $^{14}CH_4$ being oxidised prior to reaching the biosphere and causing potential radiation exposure is also insignificant (see Fig. 1b).
4) Provided $^{14}CH_4$ is released via the gas phase directly to the overlying rock, subsequently reaches the biosphere and is oxidised there (in a pond, marsh, etc.), a potential radiation exposure by ingestion of fish may become relevant (see Fig. 1c), but only if the complete C-14 inventory is released as $^{14}CH_4$.

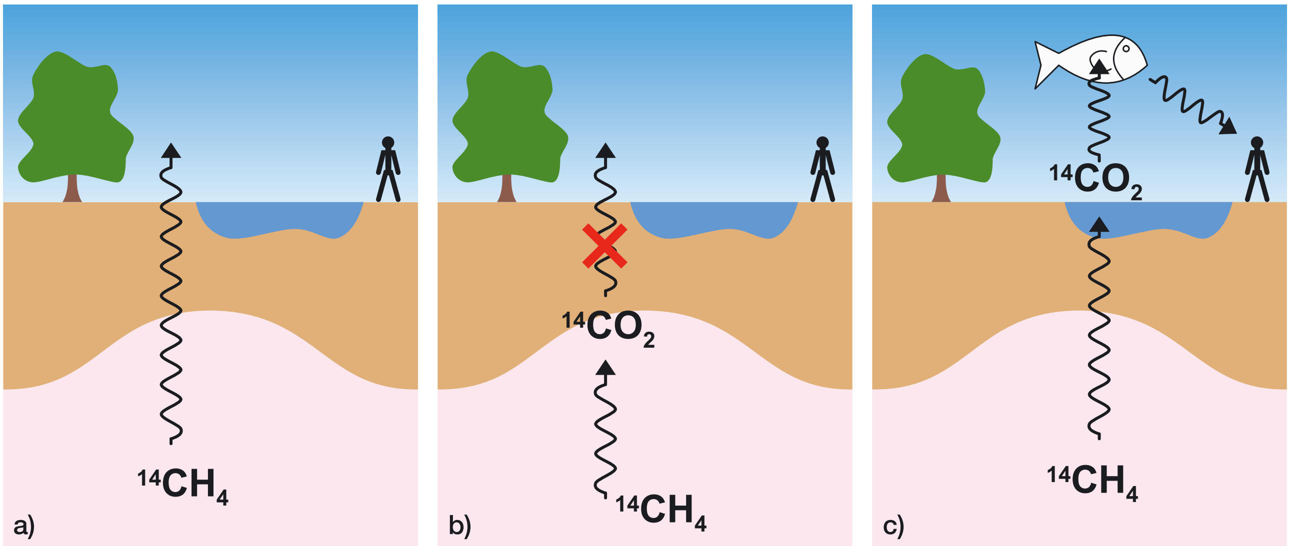

Fig. 1: Scenarios for radiation exposure by $^{14}CH_4$

The release of the complete C-14 inventory as $^{14}CH_4$ is an unnecessarily conservative assumption. Consequently, the above approach should be reconsidered and a more realistic approach applied.

6 Realistic Approach

A realistic approach considers the processes outlined before. Priorities are set in the order of decreasing importance. At the present level of refinement only the most important processes are considered.

The assumptions and simplifications underlying this approach are given below:

1) C-14 is mainly bound to organic materials. Within these it is homogeneously distributed. Occurrences of easily purgeable C-14 in e.g. carbides or non-purgeable C-14 in inorganic compounds and already released fractions of C-14 were not considered. They would lower the total release, but were neglected at the present level of refinement.
2) CO_2 is formed mostly by oxidation under aerobic conditions prior to closure and during a transient period after closure.
3) A realistic gas generation rate which includes methanogenesis for CH_4 and $^{14}CH_4$ is applied.
4) Isotopic dilution was considered only for $^{14}CO_2$. Isotopic dilution of $^{14}CH_4$ is slow and was therefore neglected at the present stage.
5) Oxidation of CH_4 during transport that would result in the precipitation of C-14 is insignificant in absence of oxidising backfill, rock or other materials. It was therefore neglected.
6) The equilibration of carbonates (solid, solution, gas) is assumed to be instantaneous.
7) C-14 in transient intermediates of chemical processes was disregarded because of the currently insufficient knowledge.
8) C-14 released prior to mine closure does not contribute to the post-closure exposure. It lowers the total release after closure but is disregarded, because of its relatively small amount.
9) C-14 in carbonates is mainly precipitated and remains fixed to the solid phase.

A computational model is not available to cover all these aspects quantitatively in detail. Therefore, the distribution of C-14 was estimated using a semi-quantitative model adjusted to the conditions in a final repository in salt (see Fig. 2).

Experience from landfills, natural analogues and laboratory studies show that approx. 48 % of the total carbon inventory is not degradable in the long-term under conditions comparable to those in a repository in salt.

All oxidants available in an emplacement chamber can oxidize on the average only 46 % of the carbon inventory to CO_2.

Approx. 6 % of the carbon inventory is reduced to CH_4.

Approx. 44 % of CO_2 will be fixed in the solid phase as carbonates. Less than 1 % of CO_2 will dissolve as CO_3^{2-} /HCO_3^-/CO_2 and much less than 1 % will be present as gas after geochemical equilibration processes.

5,7 % of CH_4 will be present as gas, whereas 0,3 % will dissolve. A negligible amount of CH_4 will be fixed or sorbed on solid phases.

C-14 is assumed to be uniformly mixed with inactive carbon.

Under the assumed conditions the semi-quantitative model shows that less than 8 % of the total C-14 is released to the biosphere. Less than 1 % of C-14 as CO_3^{2-} /HCO_3^- /CO_2 and approx. 0,3 % as CH_4 is released in solution, approx. 5,7 % of C-14 as CH_4 and less than 1 % as CO_2 is released as gas. Only CH_4 can contribute to the potential radiation exposure in the biosphere, provided that it is oxidised in surface waters (see Fig. 2).

A deeper level of refinement would lower further the predicted release of C-14 to the biosphere and thus the potential radiation exposure. This would include:

1) Amount of release of C-14 by mine ventilation already before closure.
2) Oxidation of CH_4 and other non-CO_2-compounds to CO_2 during passage through sulphate-containing backfill or rock;
3) Equilibration with other phases resulting in precipitation and sorption of $^{14}CO_2$;
4) Isotopic dilution of C-14 during transport of CO_2 through non-contaminated backfill and rock.

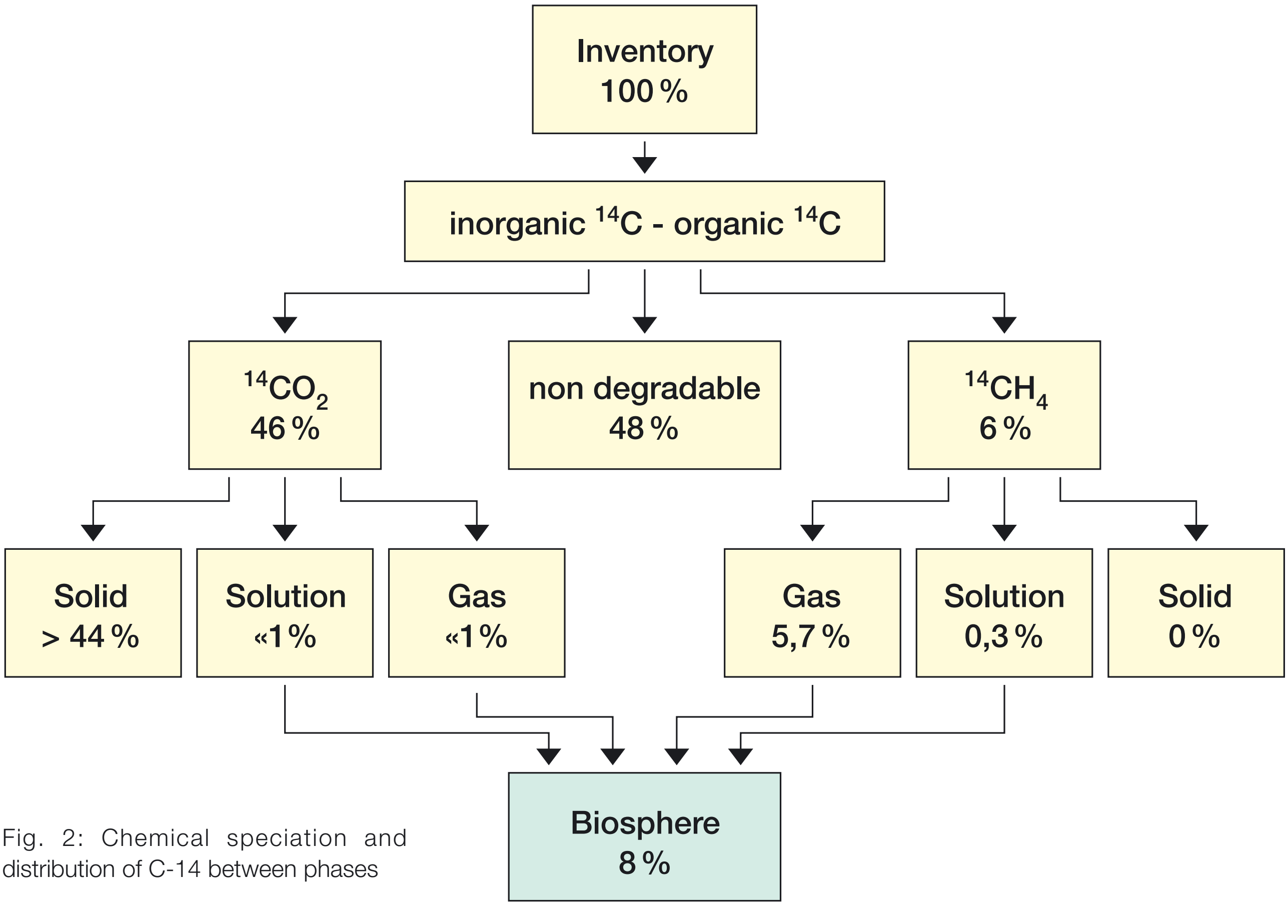

Fig. 2: Chemical speciation and distribution of C-14 between phases

7 Summary and Conclusions

A realistic approach for estimating the release of C-14 after closure of a final repository for low-level radioactive waste in a mine shows a significantly lower release of C-14 than that predicted by previously used conservative approaches. As a result, a significantly lower potential radiation exposure would be assessed, hence increasing the safety margins to federal radiation exposure limits.

Consequently, with a realistic approach it is possible to demonstrate that a repository in salt complies with the ALARA (as low as reasonably achievable) principle, thus facilitating licence approval. Conservative approaches may face difficulties to demonstrate a well-balanced radiological design as requested by the ALARA principle

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Migration and retention properties of the Czech reference granitic samples

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Abstract

Czech deep disposal concept is based on granitic host rock. Deep knowledge about transport mechanisms occurring in the rock is therefore crucial to achieve reliable safety assessment of deep geological repository. Migration behaviour of radionuclides and other contaminants in crystalline rocks is strongly associated with sorptive and diffusive properties of the rock matrix. Retardation caused by sorption onto minerals is of main importance for sorbing radionuclides (Cs, Sr), diffusion into the intact rock matrix is of particular importance for those radionuclides which exhibit little or no sorption (^{3}H, I). Diffusion into the rock matrix is dependent on diffusion coefficient, D_a and on those material properties as transport porosity ε, tortuozity τ, pore constrictivity σ and the rock capacity factor α, considering total porosity ε_t, sorption coefficient K_d and bulk density of rock samples ρ.

Batch sorption on crushed samples, sorption on coupons and diffusion experiments with Cs, Sr, Eu with Czech reference granitic samples and artificial granitic groundwater were performed, although the experiments extended into long period. K_ds were determined for both crushed samples and coupons. Diffusion coefficients for both sorbing (Cs, Sr, Eu) and non-sorbing radionuclides (I, ^{3}H) were determined, altogether with rock capacity, formation and geometric factors.

The first attempt to model Cs diffusion using activity decrease in the input reservoir was performed using invented code, developed in GoldSim environment.

Complementary experiments for studying Cs sorption and diffusion into the rock matrix were accomplished using Rutheford Backscattering Spectroscopy (RBS). Two types of diffusion pathways seem to be present within homogenous granitic rock: intergranular pores and mineral grain microcracks.

1 Introduction

According to The Concept of Radioactive Waste and Spent Nuclear Fuel Management in Czech Republic the disposal of high-level waste and spent nuclear fuel into a deep geological repository is the most realistic option for disposal. It is expected that a deep geological repository in the Czech Republic will be built in granitic rocks (RAWRA, www.rawra.cz). Deep knowledge about transport mechanisms occurring in the rock is therefore crucial to achieve reliable safety assessment of deep geological repository.

Therefore, the research has been focused on description and quantification of retention and migration processes of PA relevant radionuclides in crystalline rock of granitic type. The different approaches were used: batch sorption experiments on crushed samples and rock coupons, diffusion experiments, RBS study with non-sorbing radionuclides, and modelling. The results of the laboratory experimental programme can be also interconnected with lab experiments held on samples drilled off during long-term in-situ diffusion experiment, accomplished in Grimsel in the frame of Long Term Diffusion (LTD, www.grimsel.com).

2 Theory

Deep geological repositories are planned for disposal of high-level wastes over long time periods. In case of repository disruption, dissolved radionuclides will be transported by groundwater along fractures. Retardation for most of radionuclides is expected, including sorption on the fracture infill/walls and diffusion into the rock matrix, opening fresh surfaces for additional sorption. Retardation of radionuclides and other contaminants in granitic rock is therefore strongly associated with diffusive and sorptive properties of the fissure infill and rock matrix. In presented work the attention was paid to homogenous rock matrix properties.

The diffusion process is governed by the Fick´s first and second law, reported elsewhere [1, 2, 3, 4, 5]. The rate of change of concentration at the point in one dimension is described according to Fick´s second law by Eq. (1)

$$\frac{\partial C_p}{\partial t} = \frac{D_p}{R_p}\frac{\partial^2 C_p}{\partial z^2} \qquad (1)$$

where C_p *is the concentration in the pore water (mol m*$^{-3}$*),* D_p *the pore diffusion coefficient (diffusivity, m*^{2}*s*$^{-1}$*), R is the retardation factor in the rock matrix (m*$^{-3}$*·kg*$^{-1}$*)*

Radionuclide moves in the pore water. Tortuosity of the pores increases the diffusion path, and constrictivity reflects the changing size of the pores. Therefore the pore diffusion coefficient should be different form that in unconfined water [1]:

$$D_p = D_w \frac{\partial_D}{\tau^2} \qquad (2)$$

where δ_D *is the constrictivity,* τ^2 *the tortuosity,* D_w *the diffusivity in unconfined water and* D_p *the pore diffusivity in pores. The ratio* δ_D/τ^2 *is called geometric factor G.*

Total porosity ε is made up of *a transport porosity*, ε_t and a *storage porosity*, ε_d

$$\varepsilon = \varepsilon_t + \varepsilon_d$$

The transport (pore) porosity ε_t*, the tortuosity and the constrictivity* can be united into one parameter called *formation factor F*

$$F = \varepsilon_t \frac{\delta_D}{\tau^2} \qquad (3)$$

The *effective diffusion coefficient* D_e can than be expressed by Eq. 4, where

$$D_e = \varepsilon_t D_p = \varepsilon_t D_w \frac{\delta_D}{\tau^2} = F D_w \qquad (4)$$

Sorption causes retardation of the radionuclide in the rock matrix. Sorption, determined using batch sorption experiments is then often described by the mass based *sorption coefficient* K_d *(m*^{3}*kg*$^{-1}$*)*:

$$K_D = \frac{C_{rock}}{C_{solution}} \qquad (5)$$

where C_{rock} *is the concentration of nuclides per solid mass (mol.kg*$^{-1}$*) and* $C_{solution}$ *is the concentration of nuclides per solid mass (mol.m*$^{-3}$*).*

Retardation coefficient in the rock pores water is then defined as (6)

where K_d *is mass-based distribution coefficient (m*^{3}*kg*$^{-1}$*) and r is the bulk rock density (kg.m*$^{-3}$*).*

$$R = 1 + \rho \frac{(1-\varepsilon_t)K_d}{\varepsilon_t} \qquad (6)$$

The alternative way of examining diffusion is using apparent *diffusion coefficient (*D_a*)* and *rock capacity factor (α)* as in (7)

$$D_a = \frac{D_p}{R} = \frac{D_e}{\alpha} = \frac{D_e}{\varepsilon_t + \rho K_d} \qquad (7)$$

Rock capacity factor α is dependent on transport porosity, accessed by tracer (ε_t) and on the amount of tracer sorbed by the rock sample. For non-sorbing neutral species $\alpha = \varepsilon_t$; for cations (sorbing diffusants) $\alpha > \varepsilon_t$; and for anions $\alpha < \varepsilon_t$, treating anion exclusion formally as negative sorption.

3 Material and solution used

Considering potential granitic host rock in the Czech Massive, homogenous fine-grained non-fractured granite samples from Pribram region (Central Bohemia, Central Moldanubian Pluton) were chosen as reference samples for the study. The silicate analyses and mineralogical composition are shown in Table 1.

Table 1: Silicate and mineralogical composition of Pribram granite.

Silicate analyses	Wt. %	Mineral composition	Vol. %
SiO_2	70,72	Quartz	21
TiO_2	0,22	K feldspar	37
Al_2O_3	14,31	Plagioclase	30
Fe_2O_3 total	1,45	Biotite	3
FeO	1,21	Chlorite	5
MnO	0,088	Hornblende	4
MgO	0,76	Epidote	
CaO	2,64		
Na_2O	3,54	Porosity	0,06 – 0,2
K_2O	3,48		
P_2O_3	0,065		
H_20	1,1		
CO_2			

Several samples with different mafic mineral content were also used to determine influence of the rock constituents on radionuclide sorption (tonalite, diorite, gabro). Its composition is given elsewhere [6]. Rock samples were processed in the way shown on Fig. 1

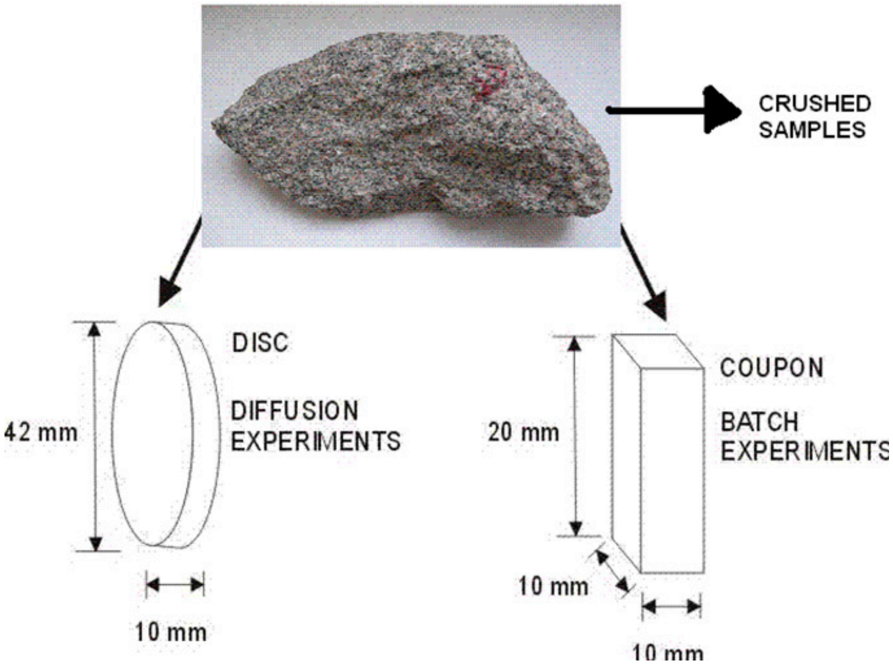


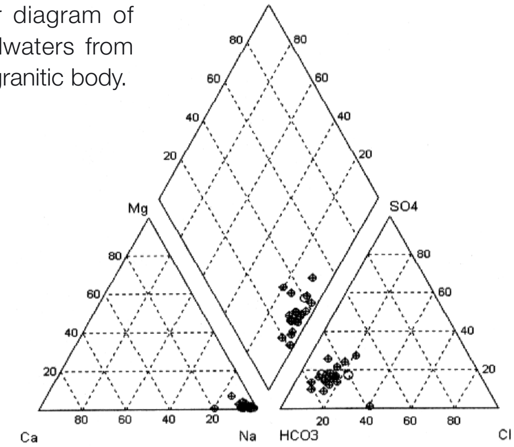
Fig. 1: The scheme of sample processing for different experimental methods.

Artificial granitic water, prepared as artificial equivalent of granitic deep groundwater from the locality, was used as a solute. The artificial groundwater composition is listed in Table 2 and shown on Fig. 2.:

Table 2: Composition of artificial groundwater used in laboratory retardation experiments.

Component	Concentration (mg/l)
Ca	3,4
Na	74,1
Mg	0,6
Cl	11,9
S(6)	19,3
pH	9,1

Fig. 2: Piper diagram of deep groundwaters from the Pribram granitic body.



4 Experimental

Radionuclide retardation within granitic rocks matrix was studied using different laboratory method. The experimental methods were following:

- batch sorption experiments with crushed samples
- batch sorption experiments with rock coupons
- static through-diffusion experiments
- trace element diffusion into rock, measured by RBS

Batch, and diffusion experiments using ^{3}H, ^{125}I (non-sorbing tracers) and ^{137}Cs, ^{85}Sr and ^{154}Eu (sorbing tracers) were performed on crushed samples, coupons and discs respectively. Synthetic granitic water simulating real groundwater from crystalline rock massif was used as a solute. Sample porosity was determined using by Hg porosimetry and water saturation method (0,2 - 0,09 %) [7, 8].

4.1 Batch experiments

Static batch sorption has been the standard method for studying the interaction of radionuclides and crystalline rock since early 80ties. Hereby, the experiments were performed crushed granitic samples with different grain size (<0,063 mm, 0,063 - 0,25 mm, 0,25 - 0,8 mm, > 0,8 mm), using ^{125}I, ^{137}Cs, ^{85}Sr and ^{152}Eu radionuclides to derive distribution coefficient R_d, used instead of K_d if steady state is not reached. The results are summarised in the Table 3 (R_d max – min, m^3kg^{-1}).

Table 3: Radionuclide sorption range onto crushed samples and rock coupons, expressed in terms of distribution coefficient R_d (m^3kg^{-1}).

Radionuclide	Rd (m^3kg^{-1}) – crushed samples	Rd (m^3kg^{-1}) – coupons
^{137}Cs	0,6 – 0.015	0,031 – 0, 044
$^{152,154}Eu$	0,578 – 0.15	0,086 – 0,0003
^{85}Sr	0,187 – 0,01	0,004 – 0,0001
^{125}I	0,0086 – 0	0,0002 – 0

Sorption of Cs and Sr was found to be dependent on grain size: coarser fraction and coupons exhibit lower sorption than finer rock fraction. The most efficient sorbent of Cs and Eu was found to be diorite (biotite 11%, silica 15%). Sorption of Eu was observed to be independent on grain size and retention of iodine was almost negligible. Therefore, it could be again concluded that crushing could generally lead to overestimation of cation sorption in comparison with experiments in situ, observed in many works.

Sorption of Cs onto granite coupons continued even after 2000 hours (see Fig. 3). The first phase (up to 400 hours) could be assigned as fast sorption on outer available surface, on the other hand the second phase can be described as radionuclide binding on the inner sorption sites in sheet silicate interlayers, and/or as diffusion [9, 10]. According to the data from similar type of Czech granite, fraction of Cs sorbed on outer sorption sites, on the edges of sheet silicate interlayer and on the sites within the interlayer could reach up to 10,5%, 49,1% and 40,4% [672 hours of sorption, unpublished results]. It can be assumed that the fraction of Cs sorbed onto inner interlayer sites will increase with time as trapped cations move from edges into the direction of distant sites with more stabile structure. This Cs fraction stay immobilised within the rock and cannot be desorbed.

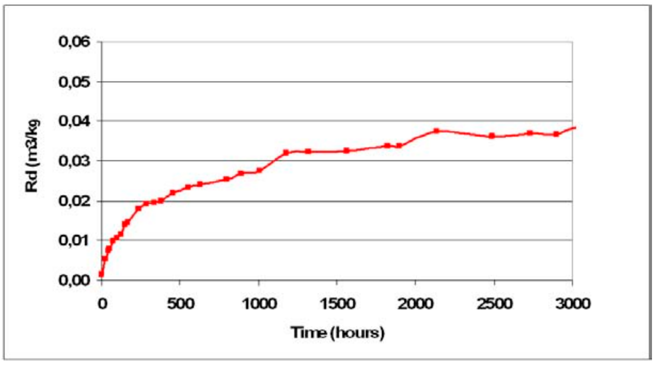


Fig. 3: Long time span of ^{137}Cs sorption onto granitic rock coupon.

4.2 Diffusion experiments

Selected granite rock disc samples were used for through-diffusion experiments with ^{3}H, ^{125}I, ^{134}Cs, ^{85}Sr and ^{154}Eu. The methodology was reported elsewhere [2, 4, 5, 10]. The activities in both input and output reservoirs were monitored with the aim of determine the shape of the breakthrough curves. Non-sorbing radionuclide (^{3}H, ^{125}I) experiments revealed radionuclide breakthrough curves that were evaluated using a time-lag method (see Fig. 4) to determine apparent diffusivity coefficient. D_a (m^2s^{-1}). The mathematic solution is given in e.g. in [11, 12]. This method of determination of D_a can be used only in cases in which activity A_1 in the injection cell is constant with time and A_2 is negligible compared to A_1 in all times, no bulk flow (advection) occurs and rock sample is homogenous.

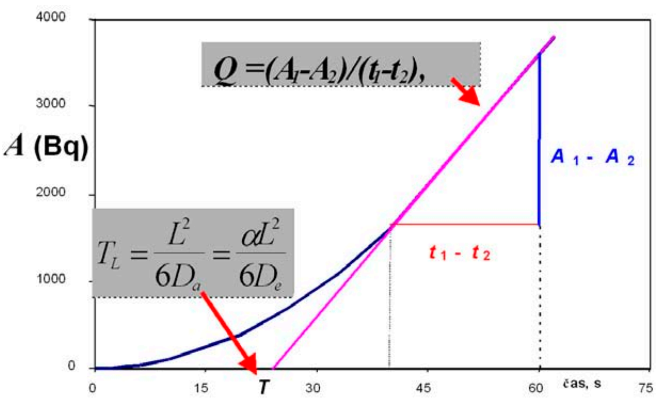


Fig. 4: Time-lag method: D_a determination using activity vers. time dependence.

Rock capacity factor, formation factor and geometric factor for non-sorbing species were calculated using equations mentioned above for given rock types using porosity values, measured by water immersion method and Hg porosimetry (0,08 – 0,22%). The conservative Kd values were used for calculations. Only for Cs both, conservative and realistic values are mentioned. Summarisation is given in the following Table 4.

Table 4: Rock capacity factor, apparent diffusivity Da ($^{m2s-1}$), formation factor F_f and geometric factor G calculated for diffusion experiments. Representative values are listed.

Radio-nuclide	Rock type	Kd used (m^3kg^{-1})	Porosity (%)	Rock capacity factor α	D_a (m^2-s^{-1})	D_e (m^2s^{-1})	Formation factor F_f	Geom. factor G
^3H	Tonalite	0	0,3	8.10^{-4}	$6,7.10^{-11}$	$5,36.10^{-14}$	$2,23.10^{-5}$	0,01
	Gabro	0	0,18	$1,8.10^{-3}$	$8,3.10^{-11}$	$1,49.10^{-13}$	$6,23.10^{-5}$	0,035
^{125}I	Granite	0	0,08	8.10^{-4}	$9,0.10^{-11}$	$7,2.10^{-14}$	3.10^{-5}	0,0375
	Diorite	0	0,2	2.10^{-3}	$3. 10^{-12}$	3.10^{-14}	$2,5.10^{-6}$	0,00125
^{125}Cs	Granite	0,032	0,08	83				
Modelled	Granite	0,06	0,08	160	$5,3.10^{-14}$	$8,49.10^{-12}$	$4,5.10^{-3}$	2,34
$^{152,154}Eu$	Tonalite	0,00856	0,3		No breakthrough			
^{85}Sr	Gabro	0,011	0,18					
	Tonalite	0,0001	0,3	0,27	No breakthrough			

However, a set of additional experiments using different cells and rock samples with granitic type rocks (gabbro, diorite, tonalite) showed a variation of D_a/D_e and rock capacity factor value. Even within one rock core the differences were found. However, formation factors for ^3H and ^{125}I did not vary as much (see Table 4). Therefore we can assume that the difference in between D_a for non-sorbing nuclides differs only due to differences in D_w, not due to material properties. The formation and geometric factors calculated for sorbing and non-sorbing radionuclides varied as well. As the experimental data set was not as wide as for example in [9], further analyses of diffusivity variability will be accomplished within LTD project [13].

2 Modelling

Eu, Sr and Cs as sorbing radionuclides have not breakthrough the granitic samples so far even after 2000 hours of experiment continuation. Eu and Sr radionuclide experiments have been still continuing (see Table 5). Therefore, diffusivity D_a for Cs diffusion into granitic samples was calculated using Cs decrease in the inlet reservoir and rock properties (see following Table 5). Radionuclide activity decrease considering sorption and diffusion was modelled using GoldSim diffusion module, considering linear sorption and no advection transport. Three possible porosity values were used for calculation. Realistic K_d value was used for modelling. Modelled and experimental line fit is presented on Fig. 5.

Table 5: Rock properties used for Cs diffusivity calculation.

		Porosity	Calculated Da $x10^{-14}$ (m^2/s^{-1})
D_w (m^2/s^{-1})	2,06 x 10^{-9}		
Tortuosity	0,6	0,008 (Pribram granite)	5,3
Kd (m^3/kg)	0,06	0,0065	4,3
Density (kg/m^3)	2,8	0,005	3,7

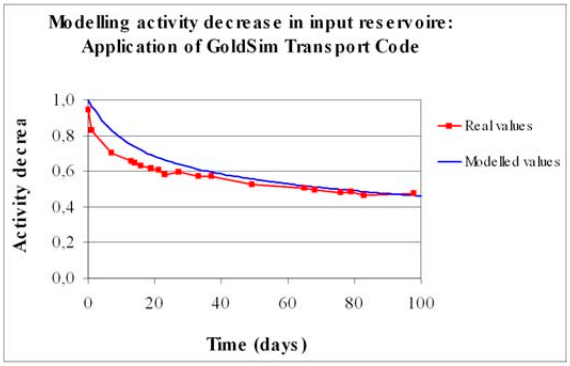


Fig.5: ^{137}Cs activity decrease in the inlet reservoir of diffusion cell – real and modelled data. GoldSim diffusion dashboard (NRI/CTU)

The values obtained using GoldSim diffusion module revealed the data in consistency with other reported e.g. [10]. This attempt showed the module could be used as a useful tool for predicting diffusivity values for long-term experiments with sorbing radionuclides.

The model refinement will follow in the next step altogether with Sr and Eu experiment modelling.

2 Cs Diffusion: RBS Study

Further complementary method for studying of radionuclide diffusion into the rock was used: Rutherford Back-Scattering spectrometry (RBS). The first attempt in the Czech Republic was undertaken in cooperation with the Institute of Nuclear Physics of the Czech Academy of Science.

Rutherford Back-Scattering spectrometry (RBS) is a non-destructive nuclear method for elemental depth analysis of nm-to-µm thick films. It comprises measurement of the number and energy distribution of energetic ions (usually MeV light ions such He^+) back scattered from atoms within the near-surface region of solid targets. The detection limits vary from 10^{12} -10^{15} atoms/cm^2. The mass resolution should be improved using heavy ion projectiles down to one mass unit [14].

Two identical homogenous granitic samples were submerged into non-active 0,1M (Sample 1) and 0,01M Cs (Sample 2) solutions for 14 days. Samples were withdrawn, dried with tissue and used for the measurement. The methodology can be referred to [15, 16]

RBS measurements were performed afterwards using ion beam 2.53 MeV, He^+ at the scattering angle 170°. The sample was scanned laterally to get information from different granite grains (mica, feldspar, quartz). Elemental depth profiles were deduced for Cs, Si, Ca (K) and Cs were detected on different sites and correlated with increasing depth of the measurement. The He^+ beam spot used was 1 mm^2.

Cs retention can be positively correlated with Fe and Ca(K) content (see Fig.6 and 7).

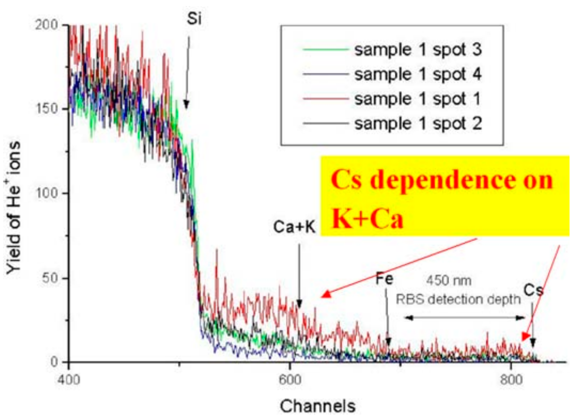


Fig. 6: Influence of Ca (K) and Fe on Cs binding on feldspar grain (0,01M Cs, SAMPLE 1, spot 1 – Feldspar)

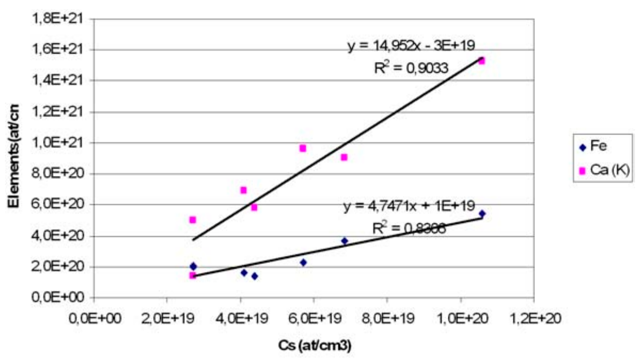


Fig. 7: Dependence of Cs sorption on Ca (K) and Fe content in granitic samples (all measured spots for both samples)

Cs sorption/migration on the granitic rock surface can follow different pattern according to mineral grain distribution. Cs was detected to be purely sorbed on quartz (see Fig. 8 c,d), sorbed on the surface of mica (atom amount was stabile for a definite depth and then decreased rapidly – Fig. 8b) and diffusing into feldspar mineral grains. The last case can be documented by diffusion profile on Fig. 8a: Cs concentration decreased in gradual diffusion profile. However, diffusion coefficient was not possible to calculate.

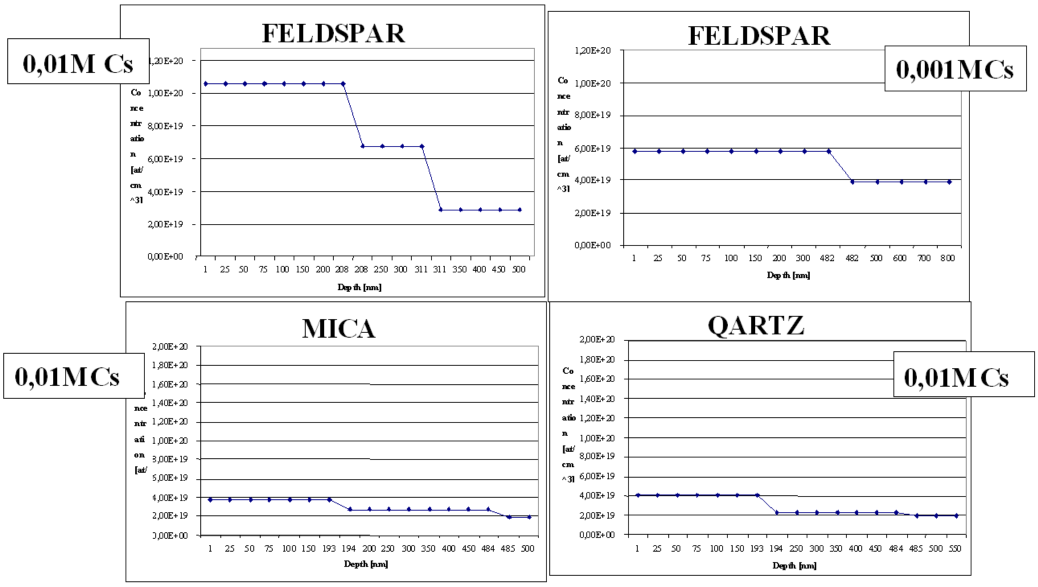


Fig. 8: RBS profiles of Cs sorption onto different mineral grains of granitic sample. From upper left a) feldspar, b) mica, lower left c) + d) quartz.

According to the results of RBS measurements, two types of diffusion pathways seem to be present: intergranular pores and mineral grain microcracks. Feldspar is known to contain microcracks and opened cleavage planes enabling additional diffusion pathway to intergranular space [10, 17]. This piece of knowledge should be seriously considered in evaluation of granitic migration properties, relevant to PA, and in particular during the process of diffusion modeling: not only porosity/interganular connected pore space could play a role in the potential radionuclide migration within the host rock. Mineral grain composition and conditions could play also a role as a migration parameter. The phenomenon is also being studied within [13].

2 Conclusion

The aim of the presented paper is to give a short overview of multimethod approach to study retention parameters of granitic rocks, considered in The Concept of Radioactive Waste and Spent Nuclear Fuel Management in Czech Republic as a potential host rock for geologic repository of HLW. The sorption distribution coefficients were measured for crushed samples and coupons, diffusion experiments performed for sorbing and non-sorbing species. The results of laboratory experiments showed that sorption and diffusivity are much more complex to be described by a simple diffusion coefficient and diffusivity.

Sorbing species retention is generally dependent on particle size: sorption decreases with increasing particle size. This generally could cause overestimation of the results due to the higher surface area of the crushed samples. Moreover, the sorption continuation with contact time could be the evidence for inner sorbing sites available for sorption in performance assessment time scale. RBS measurements traced two diffusion pathways for possible migration of radionuclide within the rock: inter-granular space and mineral grain microcracks. This, according to some authors [17], should be solved including two separate diffusion coefficients into PA models, one for intergranular process and one for microcracks,.

Diffusion of non-sorbing species can be determined within simple through diffusion experiments as those radionuclides can breakthrough the sample. However, sorbing species do not necessarily get through and determination of diffusivities could be accompanied with difficulties. The invented GoldSim diffusion module proved to be an efficient tool to model long-term diffusion experiments even with sorbing radionuclides (Cs).

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The design of a spatial information system for the Morsleben repository

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Abstract

The contribution of C-14 to radiation exposure in the biosphere can be significant as compared to that of other radionuclides disposed in a repository for low-level radioactive waste. The release pathways of C-14 and processes relevant to its release from a closed final repository for low-level radioactive waste are discussed. Because a conservative approach may lead to undue overestimation of the potential radiation exposure, a more realistic approach is outlined. At the present level of refinement, our realistic approach provides a sufficient safety margin to German federal limits for radiation exposure to demonstrate compliance with the ALARA (as low as reasonably achievable) principle and can thus facilitates licence approval.

1 Introduction

The backfilling activities in the Morsleben radioactive waste repository (ERAM) began in 2003. They are necessary to stabilize the intensely mined central part of the repository. An extensive measurement and monitoring system was designed and installed to plan and monitor all activities associated with the closure of the repository [1]. This particular part of the repository has a very complex structure, which means that construction conditions might be critical for certain structural elements during the planned backfilling activities. Still, thanks to geotechnical monitoring it is possible to ensure the local structural stability and the mandatory operational safety [2,3]. The use of an integrated spatial information system right from an early stage makes it possible to locate the great number of continuously acquired measurement results. The interactive visualization proved to greatly facilitate the interpretation of the various types of information [4].

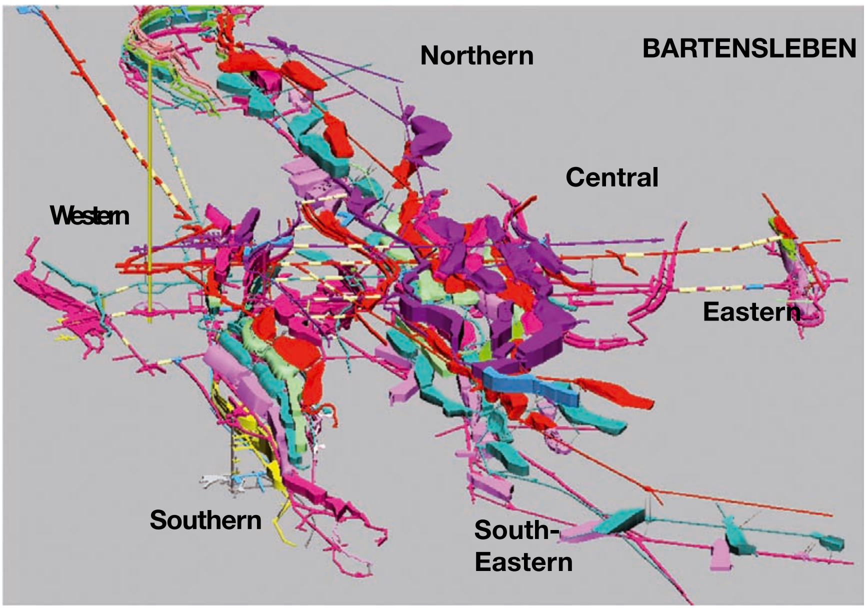

Fig. 1: 3D illustration of the halls of the ERAM

2 ERAM-SIS: Design of a spatial information system

The interpretation of geotechnical data requires a good understanding of the spatial context. A detailed 3D representation is most helpful – together with an interactive visualization tool. The backfilling activities in Morsleben rely on a comprehensive 3D model of the ERAM repository, created during the planning phase. The highly complex shapes were modeled in AutoCAD by digitizing hundreds of paper maps and sections [5] and were later refined by laser-scanning the most relevant halls [6]. In addition, a table describing the attributes (ID, name, volume, zone, altitude, level etc.) of all the cavities was set up as a relational database.

It became evident, that – to tap their full potential – the two sources of information had to be integrated into a single application. The Toporobot approach [7], originally developed for modeling and visualizing natural caves, offered a solution. It had been used before to model and document prehistoric mines (Bergbaumuseum Bochum) [8], to create an inventory of underground features in the cave of Milandre, and to build a web based spatial information system of the Mont Terri Rock Laboratory [9].

Such an information system combines 3D visualization with a database (Fig. 1 and Fig. 2). The ERAM-SIS displays the complex model of all mine passages and halls in real time and offers a fluid navigation in the 3D scene. In addition, all attributes are shown tabulated in spreadsheets, grouped by type of objects. As the views are bi-directionally linked, the user can either select from the spreadsheet and have the corresponding objects highlighted in 3D or pick objects in 3D to see the associated attributes in the spreadsheet. The database table allows to sort and search, to hide/show and color the objects as well as to structure the scene.

The information system offers specific functionality to assist with the analysis of geotechnical data. It calculates the volume of passages or halls, displays the distance between two selected points, and finds the smallest distance between two halls. It displays time series of measured data or observations and it animates the model to show the partially filled volumes. Maps and profiles schematically summarizing the geological situation can be shown or hidden interactively.

In addition to the integrated attributes and geo-referenced data, the program can handle links to further information such as reports, photos, maps, and profiles. Conversely, the information system can be used as an add-on application for a web browser. If this function is enabled, links contained in external documents (e.g. HTML, PDF, SVG) can launch the information system, point to a specific selection and/or have the camera fly to a particular view.

Initially, ERAM-SIS was designed to handle mostly static geometrical objects. But it became soon apparent that it would be particularly useful to display dynamical content such as the backfilling activities either in the planning stage or during its process. Thus, the information system was enhanced to display spatial phenomena in time. This allowed for instance to observe geophysical events in space such as clouds of microacoustical epicenters (Fig. 3). Visual feedback helped then to discover geophysical changes triggered by the backfilling activities.

During the planning stage, the system was used to control whether all predefined constraints (e.g. cavity A needs to be filled before cavity B) were fulfilled and to assist in the optimization of the operations.

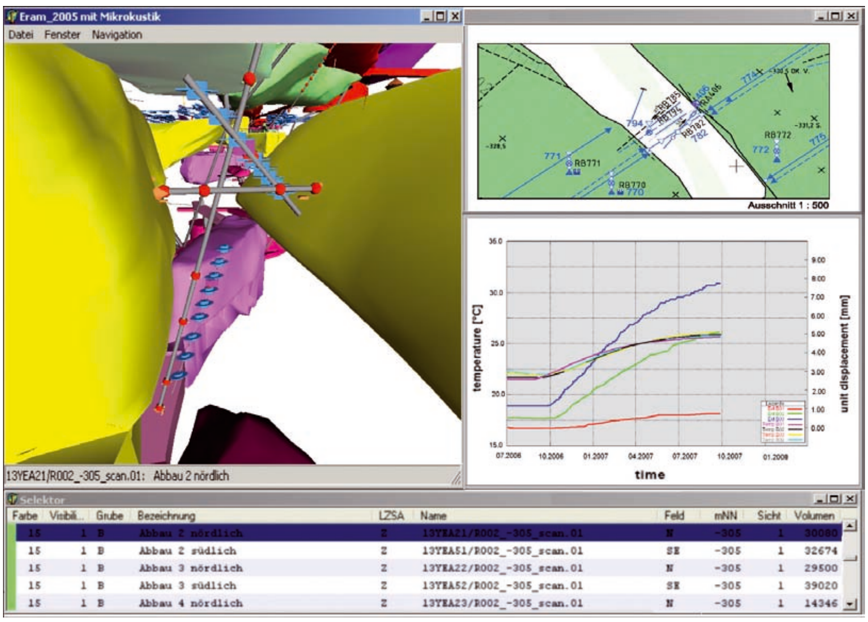

Fig. 2: ERAM-SIS 3D display of filtered and linked information

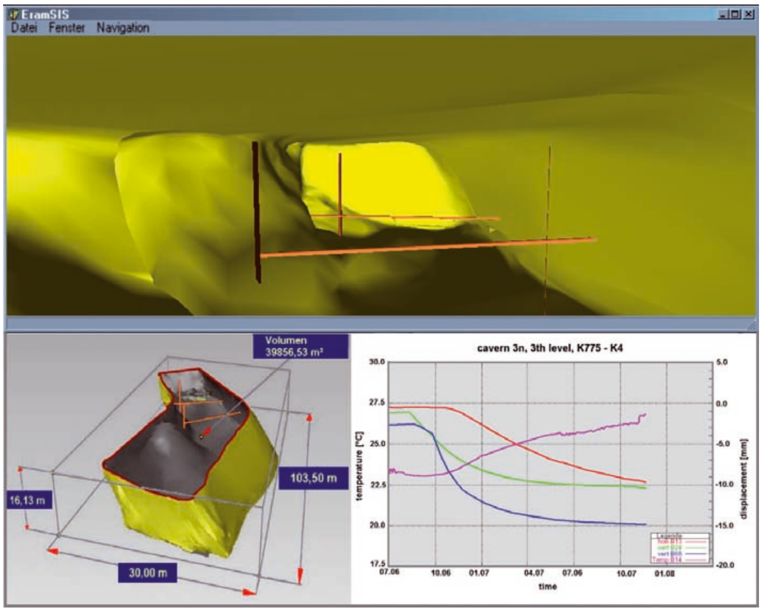

Fig. 3: 3D illustration of the halls and results from measuring systems

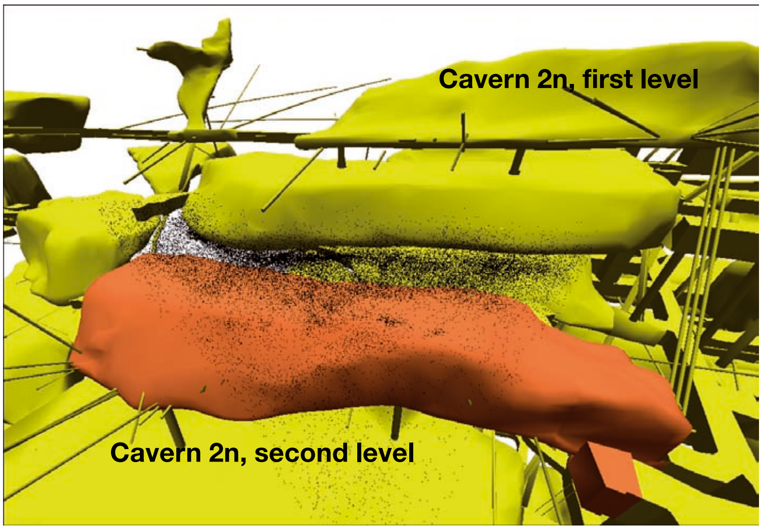

Fig. 4: Micro-acoustic signals above the material store

3 Experiences Gained in Using ERAM-SIS for the Observation Method

The geotechnical observation method has been developed to control rock behavior before, during and after backfilling activities in the areas of geomechanically exposed supporting elements [3].

There is a large quantity of automatically produced measurement results which have to be handled reliably and in a short time. To attain this ERAM-SIS is a suitable aid. It is possible to manage all data in a database which is connected with spatial information. By this it is much easier to interpret measurement results which are located in the rock.

The reliability of this system has already been proven. While backfilling a cavern on the 3rd level an unexpected acceleration of deformations was detected by the extensometers above the material store cavern on the 2nd level.

If the automatic evaluation determines unusual measurements it is important to identify the location and analyze the reliability of the result. In this case the ERAM-SIS is used. With this instrument it can easily be checked if other measuring systems nearby can confirm the first result. Especially for this purpose it was possible to include the micro-acoustic measurements in the interpretation. This system detected clear and strong signals in the determined area as well.

As an immediate measure the material store cavern has been evacuated and closed. The ongoing measures showed that the deformation progress did not stop. So it was decided to supplementary backfill this cavern, because it was unpredictable when or if this deformation progress would stop.

This is one example of how the program ERAM-SIS assists in making decisions which support the safe closure of the repository.

Additionally the application of the observation method to the ERAM requires some adaptations. E.g. new information gained during the process of backfilling can change the utilization of some caverns. This leads to modifications in the backfilling process and the measurement conception, too. For these aspects the ERAM-SIS is also a useful tool. By defining attributes that have to be kept while backfilling caverns it is possible to make a check-up during the whole backfilling progress which determines the locations where the attributes are not complied.

4 Conclusions

The spatial information system ERAM-SIS was designed to visualize the complex shapes of the cavities of ERAM and to integrate different types of information sources, e.g. measurements, geological information, photographs, inspection reports etc. The tool is meant to be used by a wide spectrum of users and offers sophisticated features in an intuitive interface.

Visual inspection and examination has proved to greatly assist with the interpretation of geotechnical data. In addition, it facilitates interaction between interdisciplinary co-workers.

Using this integrated visual approach, considerable insight has been gained already, while at the same time saving time and effort.

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Geomechanical integrity of waste disposal areas in the Morsleben repository

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Abstract

The Morsleben repository has been established in the old Bartensleben salt mine for the disposal of radioactive waste. Several parts of the mine, e.g. the southern, western, and eastern parts, were used requiring the analysis of the integrity of the salt barrier. To this end, numerous geomechanical finite-element calculations have been performed taking into account the specific geological situation and mining geometry as well as location-specific material parameters. The barrier integrity at each location was evaluated considering two criteria. The first criterion is related to dilatancy of rock salt; the integrity is guaranteed if rock stresses do not exceed the dilatancy boundary. The second criterion is related to fluid pressure; the integrity is guaranteed if the hydrostatic pressure of an assumed fluid column extending to the surface does not exceed the minimum principal stress at a certain location in the salt rock. The calculations show that dilatancy of the salt rock is restricted to the near vicinity of the rooms and hypothetical fluid pressure does not exceed the minimum principal rock stress at the outer contour of the salt barrier. Thus, the barrier integrity is given for each of the considered parts of the mine.

1 Introduction

For about two decades the Morsleben repository was used for the disposal of low and medium level radioactive wastes. The repository was established in the old Bartensleben mine, a former salt and potash mine consisting of several mining parts. Especially, the southern, the western, and the eastern parts which are located at the periphery of the mine were used for waste disposal. To assess the geomechanical stability of these structures as well as the integrity of the salt barrier geotechnical safety analyses are necessary. These analyses are based on geological and engineering-geological studies of the site, on laboratory tests and in-situ measurements, and on geomechanical model calculations. Model calculations are the most important part of the geotechnical safety assessment and comprise the geomechanical modelling of the host rock to simulate as closely as possible the conditions of the site and the behaviour of the rock, e.g. geology, repository or mine geometry, initial rock stress, as well as constitutive models and parameters (Langer, 1999).

While actual investigations are focused on the central part which is not used for disposal, but is the most critical area of the Bartensleben mine with the most considerable degree and unfavourable configuration of excavation (Büttner & Heusermann, 2004, Heusermann & Fahland, 2005, Fahland et al., 2007), this paper deals with the safety analysis of the real disposal areas of the Morsleben repository. To this aim, numerous two-dimensional finite-element calculations on characteristic cross sections of the several disposal areas (southern, western, and eastern parts) have been carried out. The results of numerical modelling have been used to analyse the integrity of the salt barrier in those parts mine from a geomechanical point of view.

2 Geological and Mining Situation

The Morsleben repository is located in the fault structure „Allertalzone“ (Fig. 1). The top of the salt structure is at approximately 140 m below mean sea level, respectively about 270 m below ground surface. The thickness of the salt structure varies between 380 m and 500 m. The exploration and modelling of the geological structure of the salt rock and the overburden in several characteristic cross sections of the different parts of the mine is based on the geological mapping of drifts, rooms, and numerous drillcores of the site as well as on ground-penetrating radar measurements (Behlau & Mingerzahn, 2001).

The salt rock is characterised by a distinct folding of the salt layers and a high amount of main anhydrite layers (z3HA) of the Leine-sequence. The structure of the salt rock in the central part includes the main stratigraphic units of the Zechstein strata (salt layers z2HS, z2SF, z3LS, z3OS, z3BK/BD, z3AM/SS, and anhydrite layers z3HA). The structure of the overburden and country rock includes the caprock, Bunter, Muschelkalk, Keuper, Jurassic, Cretaceous, and Quaternary layers.

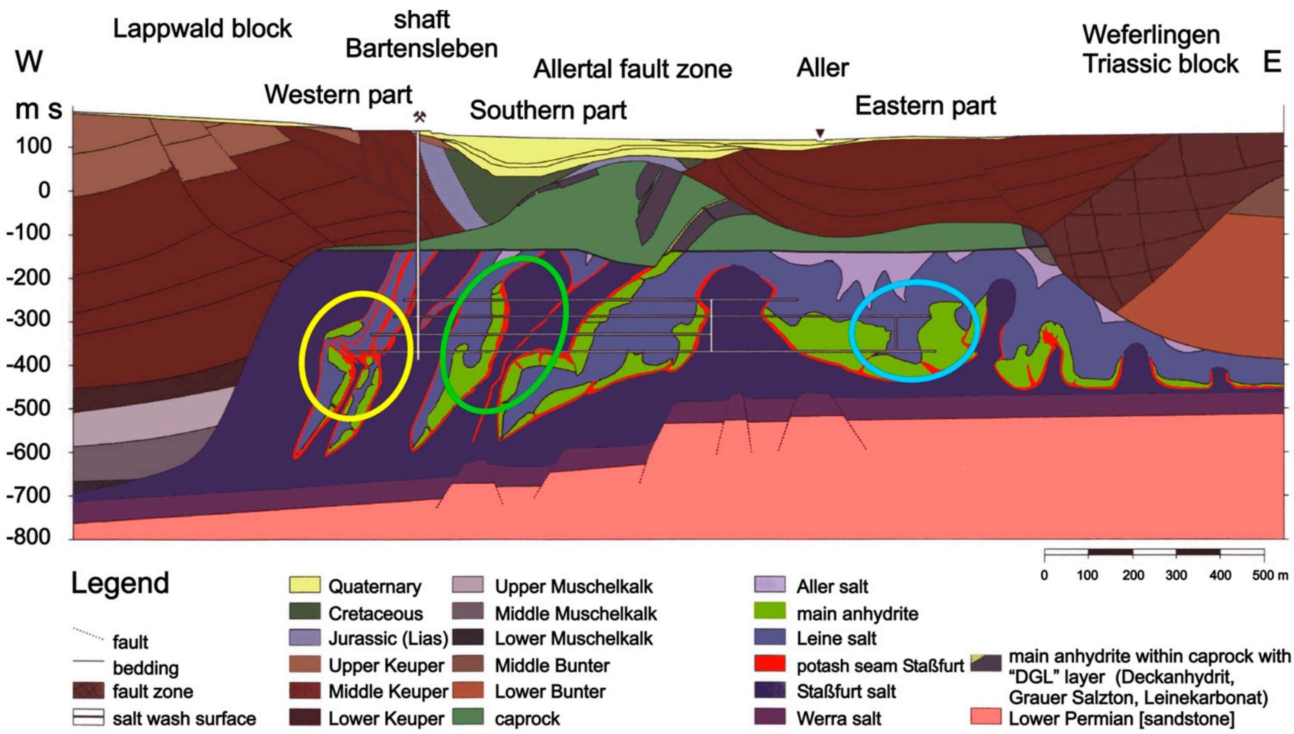

Fig. 1: Geological structure in the Morsleben repository (adopted from BfS).

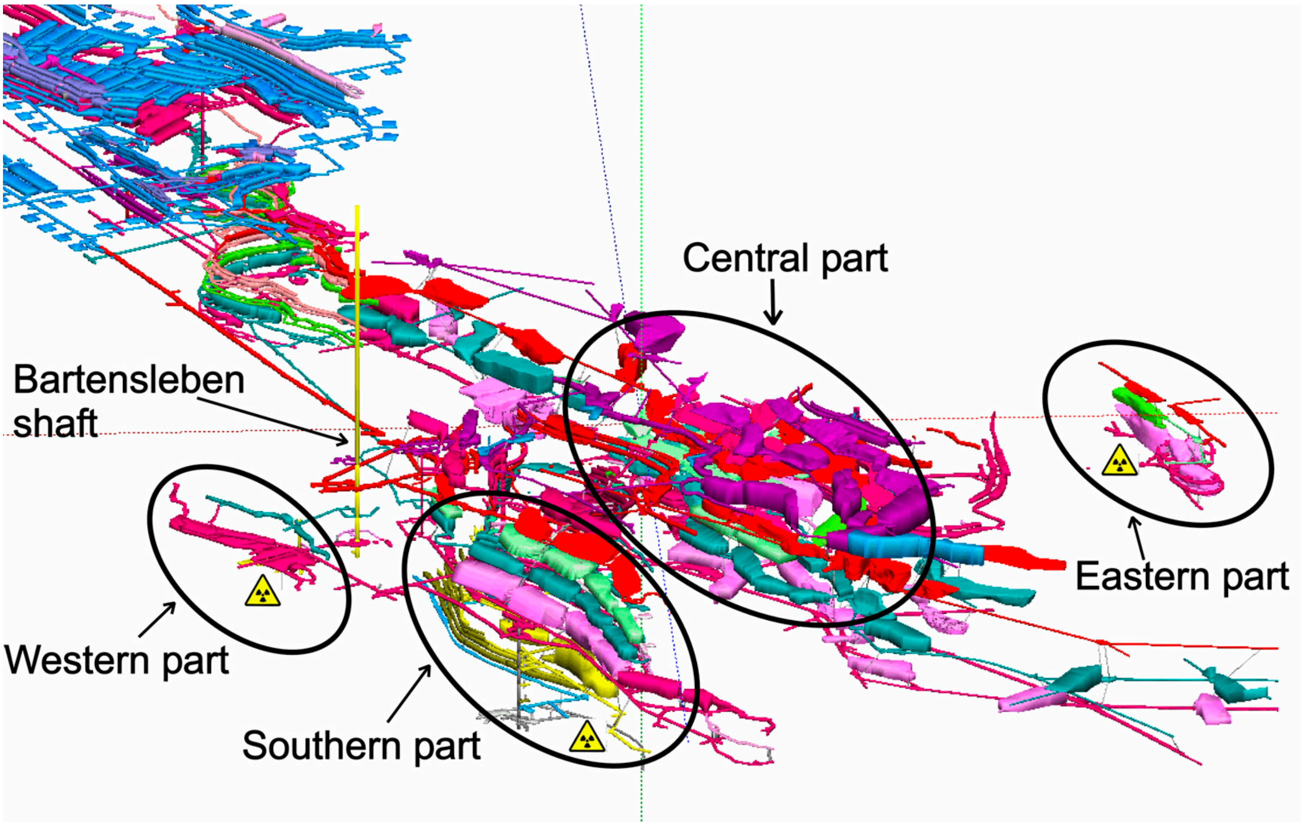

Fig. 2: Mining situation in the Morsleben repository: mapping and scanning data by DBE, model generated with ERAM-SIS (see Heller et al., 2004)

The Morsleben repository consists of the Marie mine and the Bartensleben mine, both connected by drifts on the first and third level. The Bartensleben mine includes several mining parts with four main levels: the northern, western, southern, southeastern, eastern, and central part. Most of the wastes are disposed of in the southern, western, and eastern parts. Fig. 2 shows an overview of the mining situation and old mining rooms excavated a couple of decades ago and located at different levels of the mine. From a geomechanical point of view, the central part which is not used for waste disposal shows the most unfavourable number and configuration of rooms with respect to size, shape, and arrangement in steep rows due to the strong inclination of salt layers. The southern part is characterized by a similar, to a smaller extent unfavourable configuration forming several pillars and roofs between the rooms.

3 Geomechanical and Numerical Modelling

To develop a geomechanical model as a basis of the numerical modelling, the geological structure of the salt rock and the overburden of the several disposal parts as well as the geometry of the rooms must be idealized and simplified (Plischke, 2007). Due to the two-dimensional modelling characteristic geological cross sections have been taken into account oriented more or less perpendicular to the axis of the geological structure and of the mining rooms. For the modelling it was assumed that the rooms were instantaneously excavated in the year 1940. Thus, up to now a time elapse of about 67 years must be regarded to analyse the recent stress and deformation state. The total time elapse considered in all models amounted 100 years up to the year 2040.

3.1 Eastern part

For the eastern part, the main units of the Zechstein strata (salt layers z2, z2SF, z3AM, z3SS, and anhydrite layers z3HA) and composites of the main units (z3LS-OS-BK/BD) are considered. The structure of the overburden was idealized taking into account the main layers caprock cr and Middle Keuper km (Fig. 3, left).

Finite-element modelling of the eastern part was performed on the basis of the idealized structure of the geological layers and the geometry of the old mining rooms. Fig. 3 (right) depicts a plot of a part of the 2-D model, 485 m in height and 650 m wide. The upper boundary of the model has a distance of about 130 m to the ground surface.

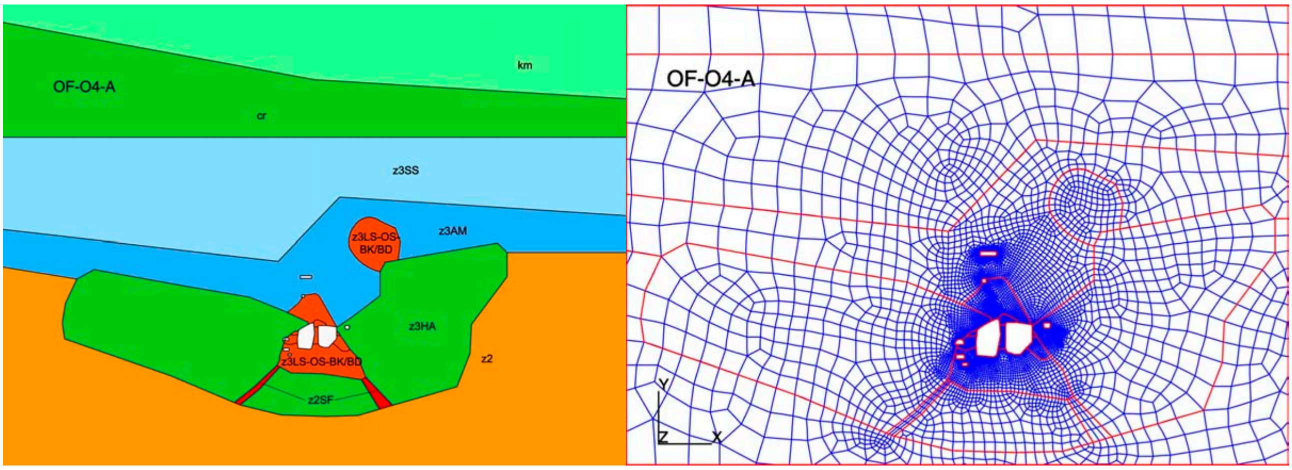

Fig. 3: Eastern part: Idealized geomechanical model (left) and part of FE mesh (right)

3.2 Western part

For the western part, the main units of the Zechstein strata (salt layers z2HS, z2SF, z2SF-UE, and anhydrite layers z3HA) and composites of the main units (z3-z4) are considered. The structure of the overburden was idealized taking into account the main layers caprock cr and Middle Keuper km (Fig. 4, left).

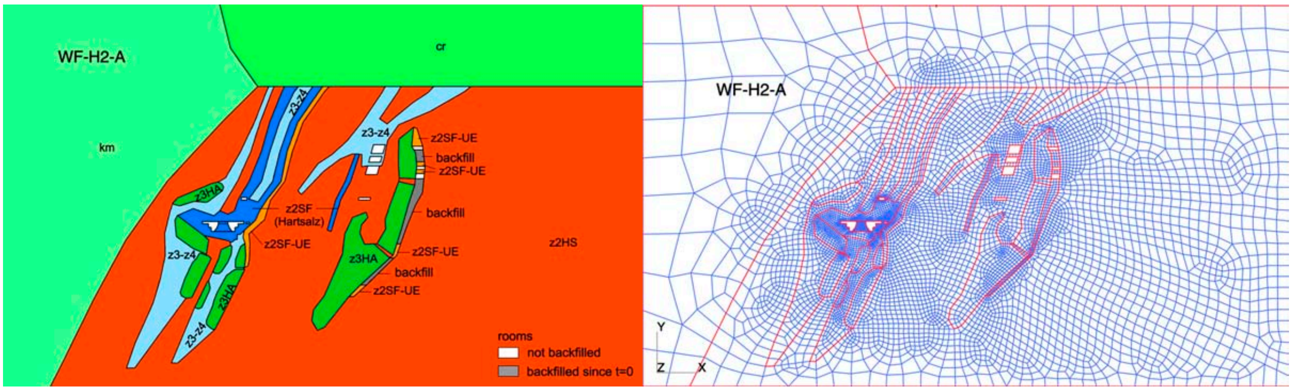

Fig. 4: Western part: Idealized geomechanical model (left) and entire FE mesh (right)

Finite-element modelling of the western part included the idealized structure of the geological layers and the geometry of the old mining rooms considering the two disposal rooms (left part of the structure) and the rooms of Lager B (right part of the structure). Fig. 4 (right) shows a plot of the entire 2-D model, 650 m in height and 1100 m wide. The upper boundary of the model has a distance of about 130 m to the ground surface.

3.3 Southern part

For the southern part, the main units of the Zechstein strata (salt layers z2HSO, z2HSB, z2HSW, z2W, z2SF, z3O, z3LS, z3AM, and anhydrite layers z1WA, z3HA) and composites of the main units (z3OS-BK/BD, z3-z4) are considered. The structure of the overburden was idealized regarding the main layers caprock cr, Keuper-Jurassic k-j, and Quaternary q (Fig. 4, left).

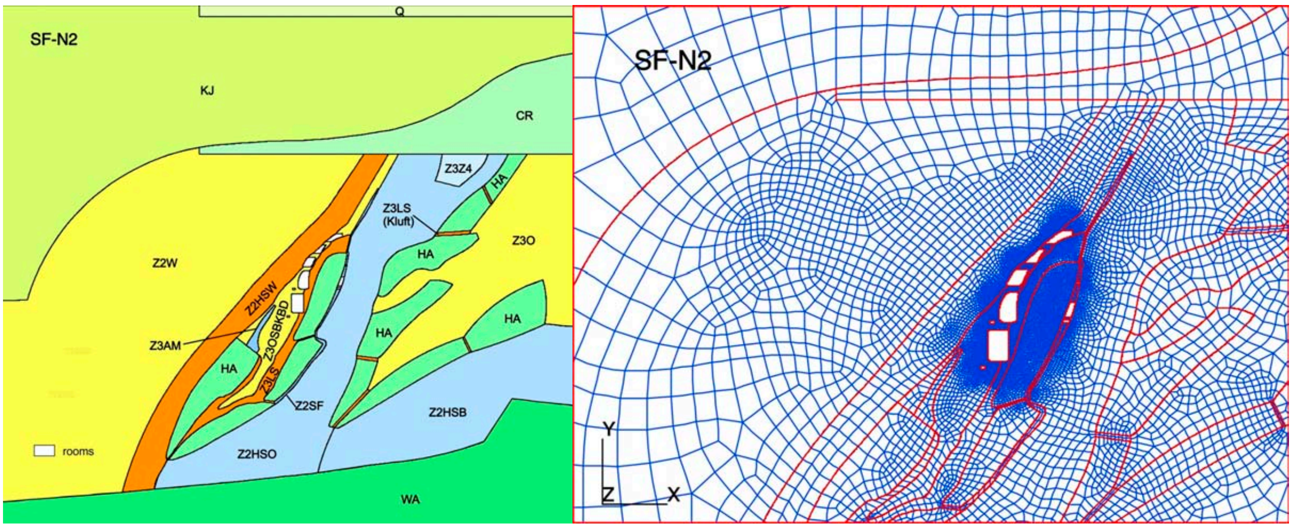

Fig. 5: Southern part: Idealized geomechanical model (left), part of FE mesh (right)

Fig. 5 (right) depicts a part of the finite-element model of the southern part. The idealized structure of the geological layers and the geometry of the old mining rooms were considered. The entire 2-D model is 800 m in height and 1000 m wide. The upper boundary of the model has a distance of about 120 m to the ground surface.

4 Material Behaviour

4.1 Creep

The idealized geological layers were classified with regard to their material behaviour. Anhydrite layers and the overburden were assumed to be elastic, the ductile rock salt layers were classified with respect to creep behaviour in terms of the steady-state creep rate. For the anhydrite layers, the modulus of elasticity was determined in laboratory tests from the post-failure stage to take the reduced stiffness of the jointed anhydrite blocks into account. Thus, a modulus of 30 GPa was used. The deformation behaviour of the ductile rock salt layers was described by a constitutive equation including both elastic and steady-state creep deformation. Additionally, the dilatant behaviour of rock salt was considered by a viscoplastic constitutive model. According to Hunsche & Schulze (1994), the effective steady-state creep rate can be calculated using:

$$\dot{\varepsilon}_{eff}^{cr} = A_{cr} \cdot e^{-\frac{Q}{RT}} \cdot \left(\frac{\sigma_{eff}}{\sigma^*}\right)^n \qquad (1)$$

with R = universal gas constant (8,3143 · 10^{-3} kJ/mol · K), T = temperature (K), σ_{eff} = effective stress (MPa), σ^* = reference stress (1,0 MPa) and the material parameters A_{cr} = structural factor (1/d), n = stress exponent (-), Q = activation energy (54,0 kJ/mol). The several types of rock salt layers mainly differ with respect to the structural creep factor A_{cr}. Thus, the creep capability of the layers can be taken into account using a factor A* related to the reference value A_0 = 0,18 1/d:

$$A_{cr} = A^* \cdot A_0 \qquad (2)$$

4.2 Dilatancy

A viscoplastic constitutive model was used to calculate dilatant deformations of the rock salt layers. Viscoplastic flow must be considered if the stress state exceeds the dilatancy boundary. Here, the dilatancy boundary is assumed to coincide with the yield function F which is described by a modified Drucker-Prager yield criterion:

$$F = \alpha \cdot J_1 + \sqrt{J_2^D} - k \qquad (3)$$

with J_1 = 1. invariant of stress tensor (MPa), J_2^D = 2. invariant of stress deviator (MPa^2), α = fictive angle of friction (-), k = fictive cohesion (MPa). The parameters α and k were determined to α = 0,2887 and k = 0. To determine the viscoplastic strain an associated flow rule is used:

$$\dot{\varepsilon}_{ij}^{vp} = \frac{1}{\eta} \cdot \langle F \rangle \cdot \frac{\partial \tilde{Q}}{\partial \sigma_{ij}}; \quad (\tilde{Q}=F) \qquad (4)$$

with η = viscosity [MPa·d], $\tilde{Q}$ = stress potential [MPa], and

$$\langle F \rangle = \begin{cases} 0, & \text{if } F < 0 \\ F, & \text{if } F \geq 0 \end{cases} \quad \text{[MPa]}$$

5 Model Calculations

The finite-element models of the several disposal parts of the Morsleben repository were established on the basis of the idealized structure of the geological layers (Fig. 1) and the geometry of the old mining rooms (Fig. 2). For reasons of simplification, two-dimensional plane-strain models were used in spite of the distinct three-dimensional development of stresses and strains around the rooms. This is valid since the rooms, especially in the southern and eastern parts, have a considerable length in the normal direction and the stress and deformation state calculated in a 2-D model is more unfavourable and „conservative" with respect to the assessment of the stability of the structure and the integrity of the salt barrier. Generally, reference models taking into account best-estimate parameters for the salt rock were used for all disposal parts. Additionally, a couple of model variations were calculated to consider different material behaviour or structural configurations.

Calculations were performed using the well proven and released finite-element code ANSALT developed by BGR (NIPP, 1991). Pre- and postprocessing of the data was carried out with the INCA/PATRAN tool. To analyse the stability and integrity of the structure the required data were calculated, e.g. deviatoric stresses, tensile stresses, effective strains, and displacements.

5.1 Eastern part

Fig. 6 (left) shows the distribution of deviatoric stresses around the rooms of the eastern part at time t = 100 years after excavation. Medium to low deviatoric stress values occur in the salt rock around and above the rooms. For the far field of the salt layers, low to negligible stress values occur. The highest stress values are obtained in wide parts of the surrounding anhydrite layers. This is caused by the creep of the salt structure, stress redistribution and stress relaxation around the rooms as well as subsequent accumulation of stress in the anhydrite.

Considering the calculated effective strain (Fig. 6, right), wide parts of the salt layers are subjected to very low to negligible strain values up to 0.25%. As expected, highest strain values up to about 3% were obtained for the near field around the rooms, especially for the pillar between the two disposal rooms.

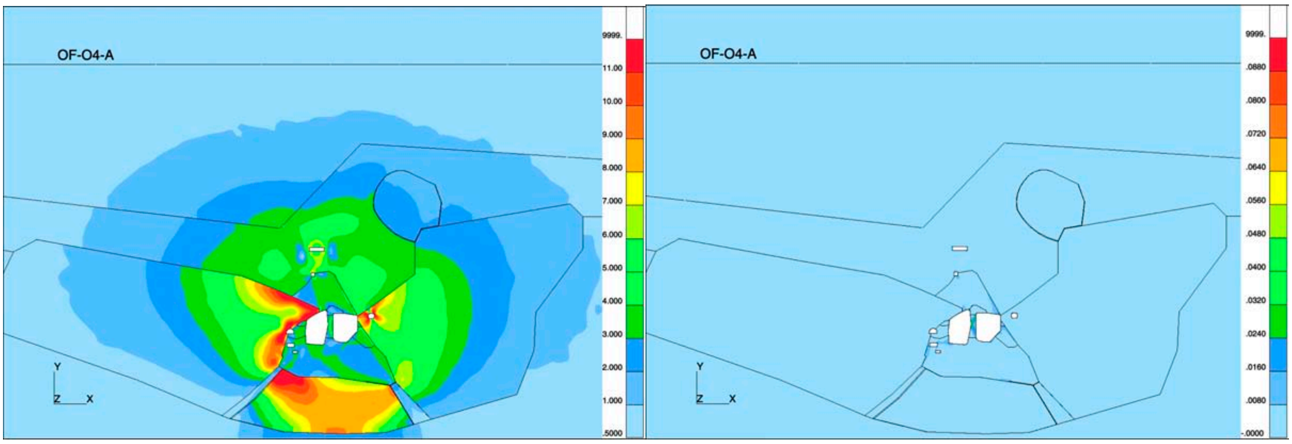

Fig. 6: Eastern part: Calculated deviatoric stress (left) and effective strain (right)

5.2 Western part

Fig. 7 (left) depicts the distribution of calculated deviatoric stresses around the rooms of the western part at time t = 100 years after excavation. Stress redistribution around the rooms takes place in wide areas of the salt rock caused by different creep properties of the several salt layers and by several stiff anhydrite layers distributed around the rooms. Stress redistribution reaches up to the overburden and country rock, but show only low deviatoric stress values of up to 5 MPa. Similar to the results obtained for the eastern part, the highest stress values are obtained in wide parts of the anhydrite layers. This is caused again by the creep of the salt structure, stress redistribution and stress relaxation around the rooms as well as subsequent accumulation of stress in the anhydrite. The increase of stress values in the anhydrite layers is uncritical with respect to the high degree of strength of the anhydrite.

Considering the calculated effective strain (Fig. 7, right), wide parts of the salt layers are sub-jected to very low to negligible strain values. Highest strain values of about 3% were calculated for the near field around the two disposal rooms and the rooms of Lager B.

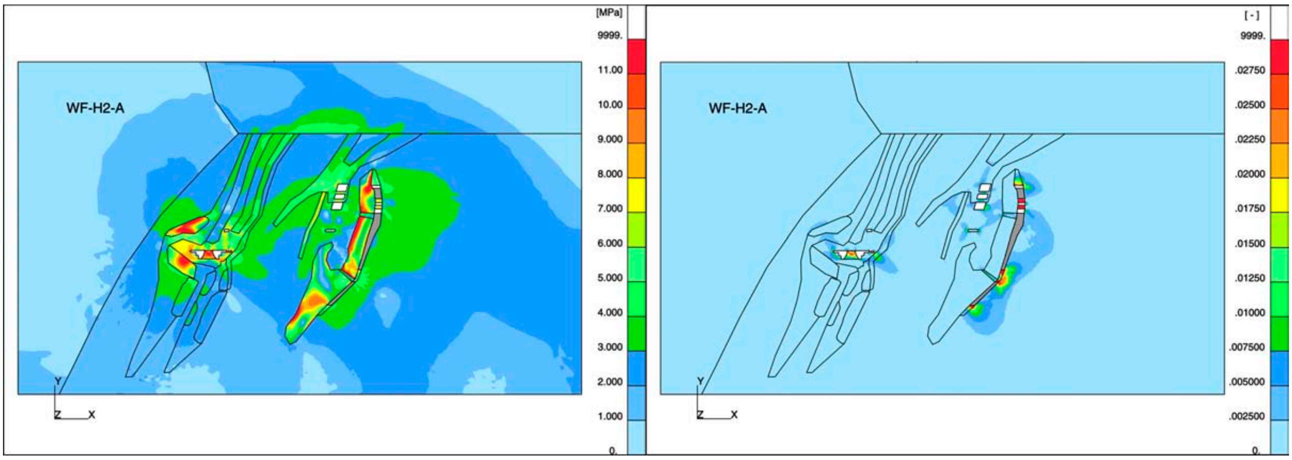

Fig. 7: Western part: Calculated deviatoric stress (left) and effective strain (right)

5.3 Southern part

The mining situation of the southern part is marked by several rooms located in at different levels of the mine. According to the pronounced inclination of the salt layers the excavation rows are arranged in a steep configuration forming several roofs between the rooms. Since these roofs are subjected to large deformations with subsequent fracturing, modelling of the mining situation covered two different cases. In a first step, a reference model was established assuming intact roofs. A second model took a total loss of bearing capacity of the roofs into account.

5.3.1 Reference model

In Fig. 8 (left), the distribution of calculated deviatoric stresses around the rooms of the southern part at time t = 100 years after excavation is plotted. Stress redistribution around the rooms occurs in wide areas of the salt rock caused by the geometrical configuration of the rooms and by several stiff anhydrite layers distributed around the rooms. This redistribution reaches up to the overburden (caprock) showing deviatoric stress values of up to 7 MPa. The highest stress values are obtained in wide parts of the anhydrite layers (up to about 16 MPa) and in the roofs between the rooms (up to 10 MPa). This is caused again by the creep of the salt structure and stress redistribution around the rooms as well as subsequent accumulation of stress in the anhydrite.

Considering the calculated effective strain (Fig. 8, right), wide parts of the salt layers are subjected to very low to negligible strain values of 0.8%. Highest strain values were calculated for the roofs between the rooms reaching a considerable amount of about 8%.

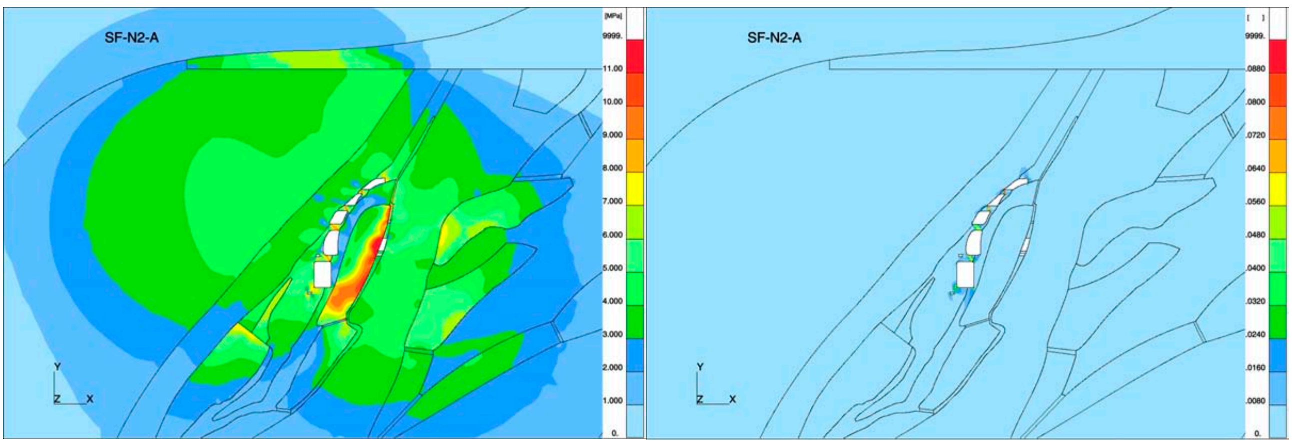

Fig. 8: Southern part: Calculated deviatoric stress (left) and effective strain (right)

5.3.2 Neglect of roof bearing capacity

Neglecting the bearing capacity of the roofs between the rooms, a reasonable increase of calculated stress and strain is obtained. Fig. 9 depicts the distribution of deviatoric stresses (left) and effective strains (right) at time t = 100 years after excavation. As expected, the stress redistribution in the salt structure is very similar to the results obtained with the reference model.

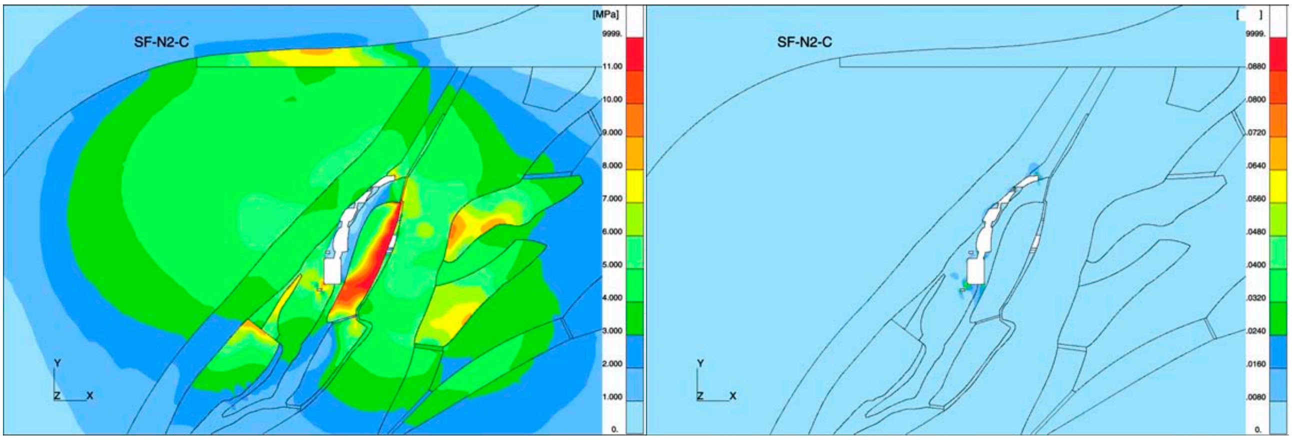

Fig. 9: Southern part without roof bearing capacity: Calculated deviatoric stress (left) and effective strain (right)

6 Analysis of Salt Barrier Integrity

6.1 Criteria

To analyse the integrity of the salt rock barrier from a geomechanical point of view, the following criteria were used (LANGER & HEUSERMANN, 2001):

- Dilatancy criterion (Fig. 10): The geomechanical integrity of the barrier is guaranteed if rock stresses do not exceed the dilatancy boundary; if this boundary is exceeded, micro-cracks will form and will cause progressive damage and increasing permeability of the salt rock.
- Hydraulic criterion (Fig. 11): The geomechanical integrity of the barrier is guaranteed if the hydrostatic pressure of an assumed column of brine extending to the ground surface does not exceed the minimum principal rock stress at the considered location of the salt body contour (e.g. top of the salt structure, contact area between salt and anhydrite blocks connected hydraulically to the overburden).

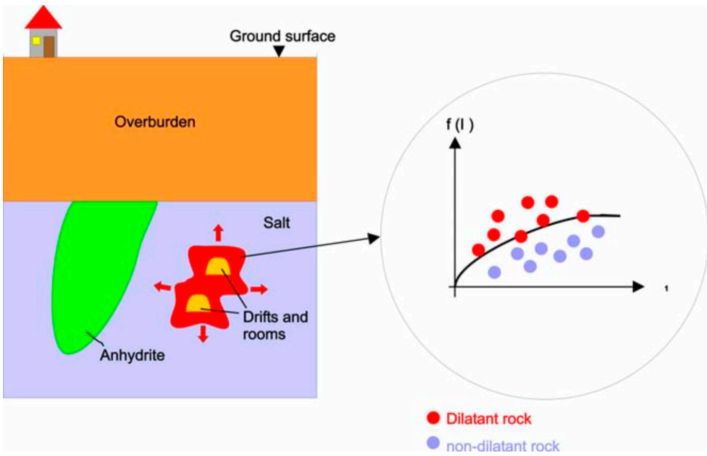

Fig. 10: Illustration of the dilatancy criterion

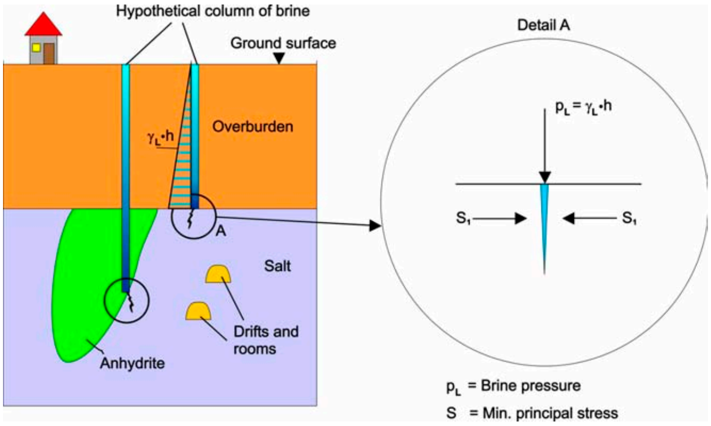

Fig. 11: Illustration of the hydraulic criterion

6.2 Eastern part

The dilatant rock zones of the eastern part calculated for a time elapse of 100 years after excavation are plotted in Fig. 12 (left). Excavation effects and creep of the salt rock cause the development of dilatancy around all rooms, as expected. Wide parts of the salt barrier between the disposal rooms and the top of the salt dome show no dilatancy. The development of dilatant rock zones with time is very slow.

Regarding the hydraulic criterion, the calculations yielded zones around the rooms of the eastern part with a distinct reduction of the minimum principal stress. From a very hypothetical point of view, the stress conditions appear to be unfavourable with respect to a fictive brine pressure which exceeds the minimum principal stress in the salt rock between the rooms and the anhydrite blocks (Fig. 12, right, yellow to red coloured areas). Since no hydraulic connection between the anhydrite and the overburden exists and wide parts of the salt barrier show neither dilatancy nor hypothetical exposure to brine induced fracturing, the integrity of the salt barrier in the eastern part is guaranteed.

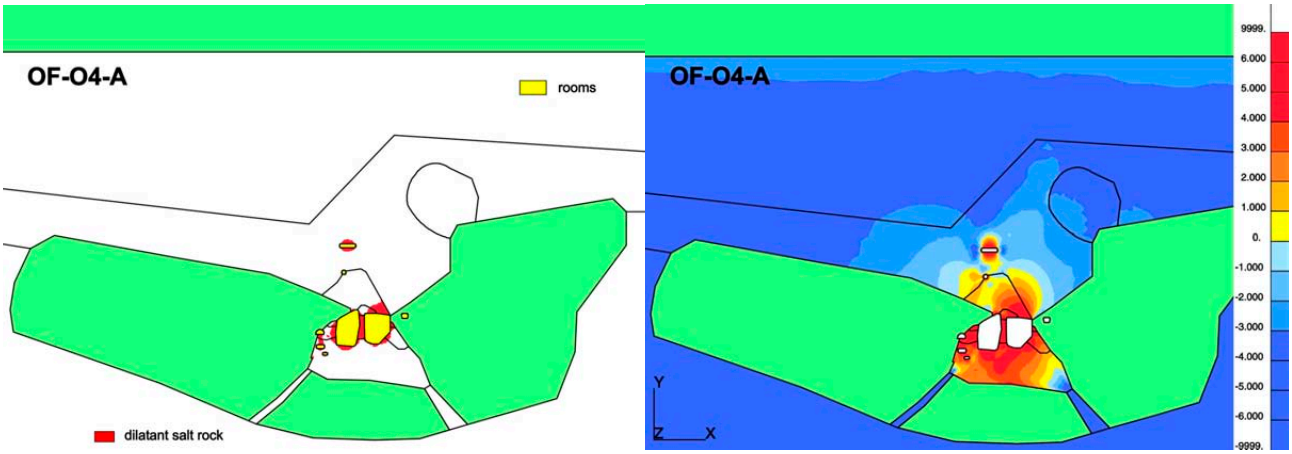

Fig. 12: Eastern part: Dilatant rock zones (left) and hypothetical rock zones affected hydraulically (right)

6.3 Western part

Fig. 13 (left) shows the dilatant rock zones of the western part calculated for a time elapse of 100 years after excavation. Excavation effects and creep of the salt rock cause the development of dilatancy around all rooms, as expected. On the opposite, wide parts of the salt barrier between the rooms and the top of the salt dome as well as the country rock show no dilatancy.

Fig. 13 (right) depicts the zones around the rooms of the western part with a distinct reduction of the minimum principal stress indicating a hypothetical exposure to brine induced fracturing (yellow to red coloured areas). Since no hydraulic connection between the anhydrite and the overburden exists and wide parts of the salt barrier show neither dilatancy nor hypothetical exposure to brine induced fracturing, the integrity of the salt barrier in the western part is guaranteed.

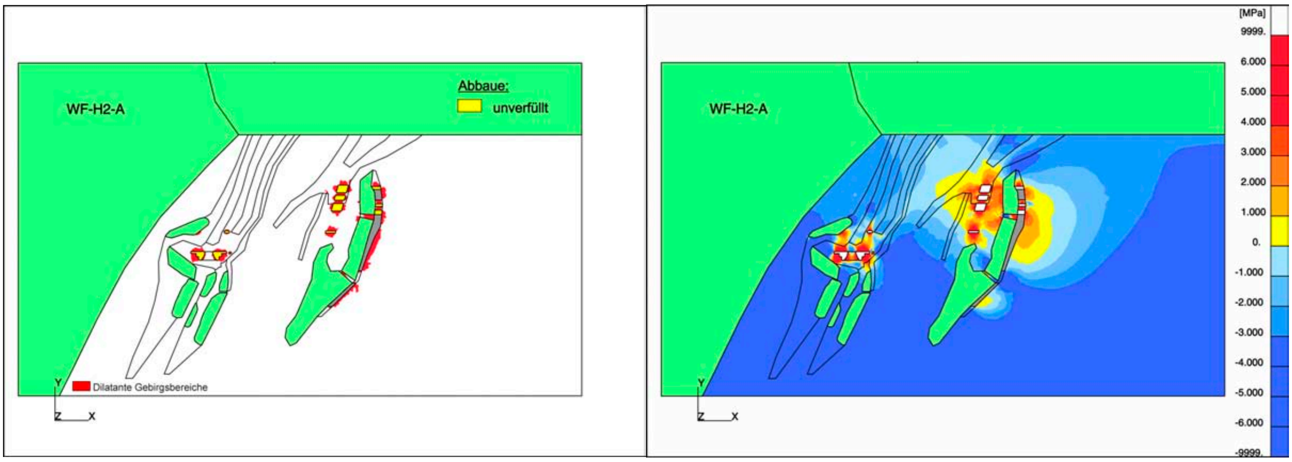

Fig. 13: Western part: Dilatant rock zones (left) and hypothetical rock zones affected hydraulically (right)

6.4 Southern part

The dilatant rock zones of the southern part calculated for a time elapse of 100 years after excavation are plotted in Fig. 14 (left). Excavation effects and creep of the salt rock cause the development of dilatancy around all rooms, as expected. Wide parts of the salt barrier between the disposal rooms and the top of the salt dome show no dilatancy.

Regarding the hydraulic criterion, zones around the rooms of the southern part with a distinct reduction of the minimum principal stress are obtained (yellow to red coloured areas). From a very hypothetical point of view, the stress conditions appear to be unfavourable with respect to a fictive brine pressure which exceeds the minimum principal stress in the salt rock around the rooms as well as between the anhydrite layers and the rooms. Since no hydraulic connection between the anhydrite and the overburden exists and wide parts of the salt barrier show neither dilatancy nor hypothetical exposure to brine induced fracturing, the integrity of the salt barrier in the southern part is guaranteed.

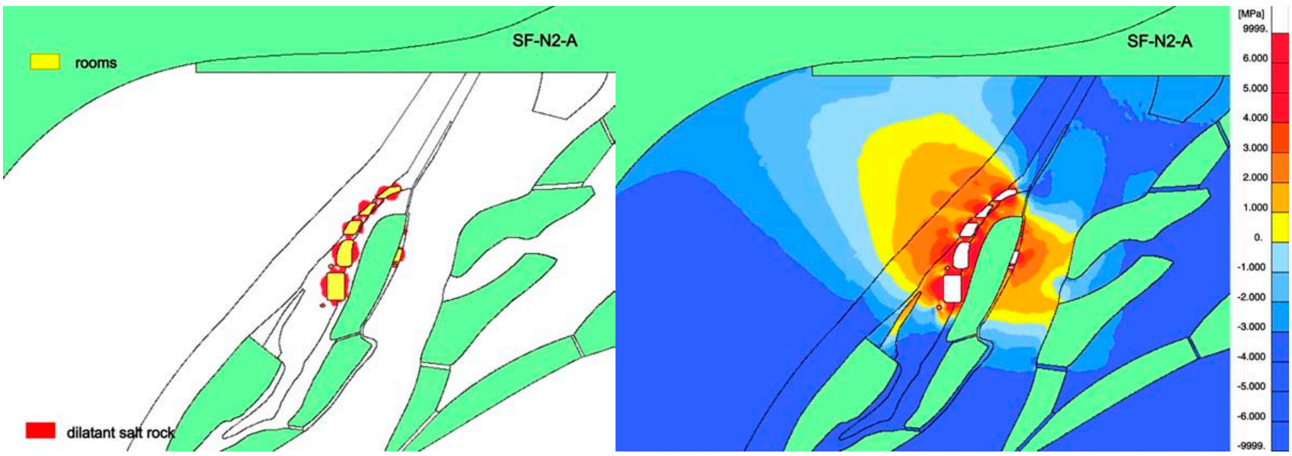

Fig. 14: Southern part: Dilatant rock zones (left) and hypothetical rock zones affected hydraulically (right)

If the bearing capacity of all roofs between the disposal rooms is neglected, wider parts of the salt rock show dilatancy and hypothetical exposure to brine induced fracturing (Fig. 15). Nevertheless, a sufficient area of the salt barrier, especially at the top of the salt dome, shows again no dilatancy and no critical stresses with regard to the hydraulic criterion. Thus, the integrity of the salt barrier in the southern part is guaranteed for this conservative case too.

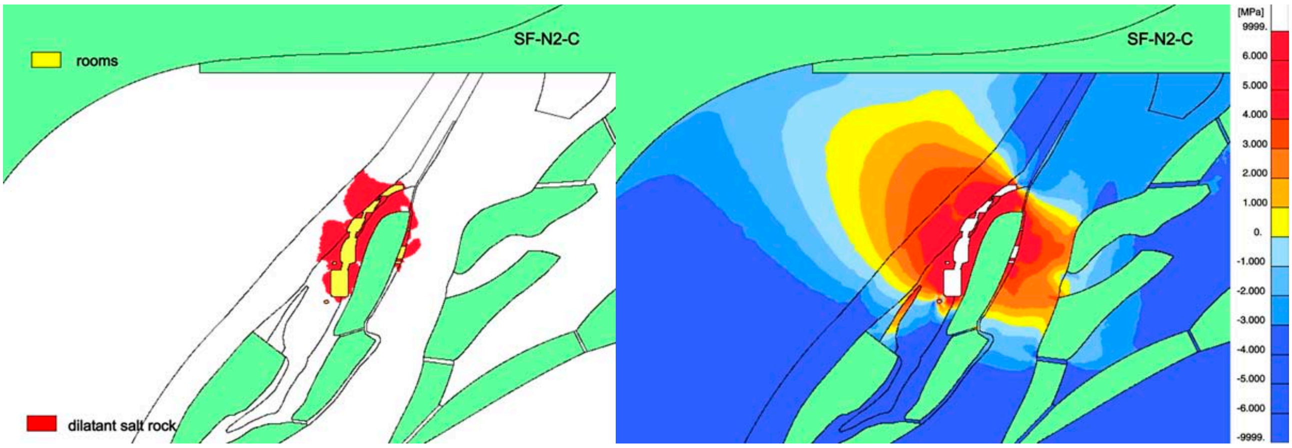

Fig. 15: Southern part without roof bearing capacity: Dilatant rock zones (left) and hypo-thetical rock zones affected hydraulically (right)

7 Conclusions

To assess the integrity of the salt barrier in the disposal areas of the Morsleben repository (i.e. eastern, western, and southern part of the Bartensleben mine), geomechanical model calculations have been carried out. Two-dimensional models of characteristic cross section of these mine parts were established based on idealized models of the geological structure. The calculations comprised the analysis of the recent stress and strain state of the salt barrier and the evaluation of the barrier integrity taking a dilatancy criterion and a hydraulic criterion into account.

The following results could be obtained by means of finite-element modelling:

- As expected, dilatant rock zones occur in the near field around the rooms of all disposal areas. Additionally, the salt rock between the rooms and several anhydrite layers nearby is subjected to dilatancy. Sufficient areas of the salt barrier show no dilatancy.
- Regarding the hydraulic criterion, zones around the rooms of all disposal areas with a distinct reduction of the minimum principal stress were obtained indicating a hypothetical exposure to fracturing induced by brine from the overburden.

Since wide parts of the salt barrier, especially at the top of the barrier, in the eastern, western, and southern part of the repository show no dilatancy and no hypothetical exposure to brine induced fracturing and no hydraulic connection between the anhydrite layers and the overburden exists, the geomechanical integrity of all disposal areas is guaranteed.

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Three Dimensional Analysis of Combined Gas, Heat and Nuclide Transport in a Repository in Rock Salt Considering Coupled Thermo-Hydro-Geomechanical Processes

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Abstract
To study the coupled thermo-hydrologic-geomechanical processes and their influence on gas and nuclide transport in a two phase flow configuration in a porous medium, a coupling of the thermo-hydrodynamic code TOUGH2 and the thermo-mechanic code FLAC3D is described and applied to analyze three dimensional gas, heat and nuclide transport in a repository for heat generating nuclear waste in rock salt. According to stress dependent hydrological properties, like, porosity, permeability and capillary pressure, the influence of coupled processes on a two phase flow can be relevant. The present procedure can be applied to quantify safety margin related to hydro-fracturing and dilatancy due to fluid pressure build-up and to define bounding analyses, if the hydrological properties, such as porosity, permeability, and capillary pressure, depending on mean effective normal stress are used.

1 Introduction

To assess the long term safety of a repository for heat generating nuclear waste in a deep geological rock formation, often groundwater flow into the repository is postulated. The water can react with the radioactive waste or with its containers and can gradually disassemble them. The radioactive substances after being dissolved in liquid phase can be transported out of the repository and subsequently can be released into the geosphere. The heat and fluid transport can be enhanced significantly by gas generation, mainly hydrogen due to corrosion of metallic materials in the repository. The two phase flow, pressure build-up due to gas generation and heat transport can influence substantially the mechanical behavior of the filling and sealing materials and host rock. The transient stress situation, especially, if the fluid pressure approaches the lithostatic pressure, can reactivate an unfavorably oriented fault or can lead to new fracturing. This can in turn affect the hydrological properties, like porosity, permeability and capillary pressure.

To study the coupled thermo-hydrologic-mechanical (THM) processes and their influence on gas and nuclide transport in a two phase flow configuration, the thermo-hydrological code TOUGH2 and the thermo-mechanical code FLAC3D are coupled. In GRS, TOUGH2 has been modified to investigate gas, heat and nuclide transport under various conditions. For instance, in [1], three dimensional gas, heat and nuclide transport in a repository in rock salt is analyzed with TOUGH2 considering porosity and permeability of crushed salt depending on pressure, temperature and rock convergence without considering mechanical effects. FLAC3D is developed for rock and soil mechanics and can treat some THM features for a single phase flow but not for two phase flow situation [2]. In [3], TOUGH2 and FLAC3D are coupled linearly to study gas and nuclide transport in a three dimensional isothermal system. In this paper, which extends the analyses of [3], the two codes are coupled sequentially to analyze a three dimensional non-isothermal system as proposed in [4].

2 Thermo-Hydrological Analysis with TOUGH2

To perform analysis with TOUGH2, which is a recognized tool to study three dimensional coupled fluid and heat transport of two-phase multi-component fluid mixtures in porous media, one of the several equation-of-state (EOS) modules must be linked with the main code of TOUGH2. The number and the properties of the fluid components are determined by the EOS module selected. The mass balance equation of each fluid component with contributions from gas and liquid phase includes advection and dispersion in both phases. In the heat balance equation, convection and conduction are considered. The velocities of gas and liquid phase are determined by multiphase extension of Darcy's law considering relative permeability of each phase depending upon liquid saturation [5]. Usually, the first fluid component is groundwater. In the equation of state module EOS7, which is used here, the second fluid component can be solute or denser salt water [6]. The third fluid component is a soluble gas (air or hydrogen). The complete steam table is used to calculate the liquid properties as a function of pressure, temperature and mass fraction of solute or salt water. The gas phase is treated as a mixture of ideal gases such as air or hydrogen and vapor.

For numerical simulation the region to be modeled is discretized into volume elements. The conservation equations are solved simultaneously with the integral finite difference method using Newton-Raphson iteration scheme. Time is discretized fully implicitly as a first order backward finite difference. This together with full upstream weighting of fluxes at the element interfaces can avoid time step limitations, mainly when a phase appears or disappears. The scalar quantities like pressure and temperature are determined at the center of the elements and the vector quantities such as velocities, mass and heat fluxes at the element interfaces.

3 Mechanical Analysis with FLAC3D

FLAC3D is an established three dimensional explicit finite difference code for engineering mechanics and can describe the behavior of structures built of soil, rock or materials that undergo plastic flow when their yield limits are reached. Materials are represented by polyhedral elements within a three dimensional grid that is adjusted by the user to fit the shape of the object to be modeled. Each element behaves according to a prescribed linear or non linear stress/strain law in response to applied forces or boundary restrains. To analyze coupled processes, fluid pressure and temperature can be prescribed by invoking the fluid and thermal configuration and the mean effective normal stress can be calculated as [4; 8]:

$\sigma_{mean} = (1/3)(\sigma_1 + \sigma_2 + \sigma_3)$, $\sigma_{mean,eff} = \sigma_{mean} + (p - \psi p_{cap})$.

The Bishop factor ψ, which depends on soil properties and should be derived from site specific data, approaches unity for saturated soil and zero for dry soil. Here, to simplify the analysis, ø is replaced by liquid saturation ($\psi = S_{Liquid}$). To study THM processes, it is to be noted that, total stresses, pressure and temperature are computed at the grid points (corner nodes of a zone) and principal stresses and average pressure at the center of a zone.

4 Sequential Coupling of TOUGH2 and FLAC3D

The original version of TOUGH2 with EOS7 is modified to couple FLAC3D sequentially. An identical numerical grid with equal numbers of FLAC3D zones and TOUGH2 elements is required for the coupling (Fig. 1). In the sequential procedure, the code TOUGH2 is executed as a ‘main program’ to perform thermo-hydrological analysis and the FLAC3D as a ‘subroutine’ to conduct a quasi static mechanical analysis (Fig. 2). It can be expedient to run the codes in an environment of the same operating system to ease the data transfer between the codes. In the beginning, the initial fluid pressure and temperature distributions are computed by TOUGH2 and transferred to FLAC3D. Since the TOUGH2 mesh uses one centre point within an element to determine pressure and temperature in element and since the FLAC3D grid points for temperature and pressure are located at the corners of the elements, data have to be interpolated from the mid-element of TOUGH2 to the corner grid points of FLAC3D. The volume averaged pressure and temperature at all grid points of FLAC3D can be determined as:

$p_n = [\Sigma p_k V_k] / \Sigma V_k$, $T_n = [\Sigma T_k V_k] / \Sigma V_k$.

This is a one possible scheme to compute the pressure and temperature at the grid points. Depending on problem, other schemes should also be considered. Subsequently, the initial stress distribution is calculated and a restart file is created. During the transient analysis, at the end of each time step, TOUGH2 calls FLAC3D and the current fluid and temperature distributions are transferred to FLAC3D. In FLAC3D, the new strain and stress distribution are computed by employing the previous restart file and the current fluid and temperature distributions of TOUGH2. At the end of the FLAC3D run, a new restart file is created and the current stress distributions are transferred back to TOUGH2. Since the stresses are determined at the centre of a zone, they can directly be allocated to the corresponding elements of TOUGH2. To continue the two phase flow and heat and nuclide transport analysis with TOUGH2, the hydrological properties, such as porosity, permeability and capillary pressure for all elements are determined basically as a function of mean effective stress at each time step. Usually, these coupling functions can be non-linear and should be determined by using site specific data. For instance, following functions are adopted:

$\Delta\sigma_{mean,eff}(t) = \sigma_{mean,eff}(t) - \sigma_{mean,eff}(t = 0)$, $\varphi_0 = \varphi(t = 0)$, $\varphi_{aux} = \varphi_0 \exp(a\Delta\sigma_{mean,eff})$.

Since porosity is described as a function of stress, which is in turn a function of pressure, the compressibility (TOUGH2 parameter) of the porous medium can be determined:

$\beta = (1/\varphi_{aux})[\Delta\varphi_{aux}(t)/\Delta p(t)]$.

This varying compressibility is then inserted in the conservation equations of the hydrological analysis to compute the porosity as a function of pressure and subsequently other properties:

$\Delta\varphi(t) = \beta\varphi(t)\Delta p(t)$, $k_0 = k(t = 0)$, $k = c\varphi^b$, $c \sim k_0$,

$p_{cap} = p_{Gas} - p_{Liquid} = p_{cap}(S_{Liquid})[(k_0\varphi)/(k\varphi_0)]^{1/2}$,   a, b, c: constant parameters.

Thus, the main state variables, pressure, temperature and stress are exchanged between the codes at each time step. The permeability and the compressibility of the porous media are kept constant during a time step. Depending on problem, other coupling functions can also be considered. This sequential method, in which the two codes are coupled at the end of each time step, can be improved and verified by an implicit iterative sequential coupling, in which the two codes are coupled at every Newtonian (physical) iteration within a time step. But, the iterative sequential coupling can increase the computational effort significantly.

5 Thermo-Hydro-Mechanical (THM) Analysis

A three dimensional simplified model of a non-isothermal repository system is considered to demonstrate the sequential coupling of TOUGH2 and FLAC3D (Fig. 3 and Table 1). The model with reasonable parameters consists of four material domains: upper 400 m as a barrier rock, lower 200 m rock salt as a host rock, a 10 m high repository within the host rock at the bottom and an excavation damaged zone (EDZ) around the repository. Initially, the complete system, except the repository, is flooded with groundwater. A transient temperature boundary condition at the bottom surface of the repository is introduced to simulate decay heat generation. To simplify the analysis, the radioactive substances in the repository are simulated by a single non decaying solute. The uniform gas generation in the repository is represented by hydrogen formation rate in three time segments [3]:

$0 \leqslant t \leqslant 1000$ years: linear increase from 0 to Q_{Gas},

$1000 \leqslant t \leqslant 5000$ years: Q_{Gas} = 45 kg/year,

$5000 \leqslant t \leqslant 6000$ years: linear decrease from Q_{Gas} to 0.

The fluid consists of three components: ground water, solute in liquid phase and hydrogen. The solubility of hydrogen in liquid phase is given by:

$X_{Gas\ in\ Liquid} = m_{Gas}/m_{Liquid} = (p/C_{Henry})(M_{Gas}/M_{Liquid})$.

To determine the relative permeabilities and the capillary pressure, the Van Genuchten functions are applied [5]:

$S_{Liq,Eff} = (S_{Liquid} - S_{Liq,Res}) / (1 - S_{Liq,Res})$, $S_{Gas,Res} = 0$, $k_{Liq} = kk_{Liq,Rel}$, $k_{Gas} = kk_{Gas,Rel}$,

$k_{Liq,Rel} = (S_{Liq,Eff})^{1/2}\{1 - [1 - (S_{Liq,Eff})^{1/\lambda}]^{\lambda}\}^2$, $k_{Gas,Rel} = (1 - S_{Liq,Eff})^{1/3}[1 - (S_{Liq,Eff})^{1/\lambda}]^{2\lambda}$,

$p_{cap} = p_b\,[(S_{Liq,Eff})^{-1/\lambda} - 1]^{(1-\lambda)}$, $\lambda = 0.77$, $p_b = 0.56\,(k_0)^{-0.346}$, k in m^2, p_b in Pa.

To determine the spatial stress conditions with FLAC3D, an isotropic elastic material for the cap rock (barrier rock) and repository is assumed. The rock salt (host rock) and EDZ are considered to be elastic visco-plastic materials and their behaviour is described by neglecting primary but including secondary creep rate:

$(d\varepsilon/dt)_{secondary} = D \exp[-A/\theta(t)]\,(\sigma^{dev} / \sigma_{ref})^5$, $\sigma^{dev} = [(3/2)\,\sigma_{ij,dev}\,\sigma_{ij,dev}]^{1/2}$, σ_{ref} = 1 MPa,

$\sigma_{ij,dev}$: deviatoric part of total stress σ_{ij}, D = 0.18 1/day, A = 6495 K.

Case THS31: In this reference case for the hydrological analysis with TOUGH2, the hydrological properties do not depend on stress ($\beta = 0$, φ = constant, k = constant).

Case THMS31: Sequential coupling: Using basically the same input data of case THS31, TOUGH2 and FLAC3D are executed sequentially at each time step considering porosity and permeability depending upon stress (for repository: $\beta = 0$, φ = constant, k = constant):

$\varphi_{Barrier,aux} = 0.05 \exp[5 \cdot 10^{-8} \text{ (1/Pa) } \Delta\sigma_{mean,eff}]$, $k_{Barrier} = (1.6 \cdot 10^{-14} \text{ m}^2) (\varphi_{Barrier})^4$,

$\varphi_{Host,aux} = 0.01 \exp[5 \cdot 10^{-8} \text{ (1/Pa) } \Delta\sigma_{mean,eff}]$, $k_{Host} = (1.0 \cdot 10^{-11} \text{ m}^2) (\varphi_{Host})^4$.

$\varphi_{EDZ,aux} = 0.02 \exp[5 \cdot 10^{-8} \text{ (1/Pa) } \Delta\sigma_{mean,eff}]$, $k_{EDZ} = (6.25 \cdot 10^{-11} \text{ m}^2) (\varphi_{EDZ})^4$.

To avoid numerical problems due to unrealistic values of the rock compressibility, especially, when $\Delta p(t)$ is too small between the consecutive time steps, the rock compressibility is not allowed to exceed a certain reasonable value, for instance, $\beta \leqslant 1\cdot10^{-7}$ 1/Pa.

Case THS33: This bounding hydrological case is same as THS31, but now a constant rock compressibility $\beta = 2\cdot10^{-8}$ 1/Pa for the entire model, except repository, is applied, which is derived as a maximum value from the case THMS31. This implies $\varphi = \varphi(p)$ and $k \sim \varphi^4$.

These three cases are computed with the modified version of TOUGH2/EOS7 up to the problem time of 10^4 years with a maximum time step of $8\cdot10^8$ s. The temperature at the centre of the repository for the case THS31 with constant hydrological properties in Fig. 4 indicates that the maximum temperature of 148 C is reached at t = 240 years. Fig. 5 shows the pressure distribution of case THS31 in the boundary plane (y = 5 m) at t = 1000 years, around which the maximum value occurs. The gas generation and the decay heat influence the pressure distribution significantly leading to the maximum value of 14.7 MPa in the repository, about 2.7 MPa above the lithostatic level. The temperature distributions in two boundary planes (y = 5 m and y = 45 m) at t = 1000 years are not very different, as the width in the y-direction is not very large (Fig. 6 and 7). The distributions of pressure, temperature, gas saturation and solute mass fraction in liquid phase at t = 10^4 years are depicted in Fig. 8 to 11. At t = 10^4 years the pressure in the repository is still around 7.8 MPa, about 1.8 MPa higher than the hydrostatic level (Fig. 8). Due to large capillary pressure difference between the rock salt, the excavation damaged zone and the repository and due to increasing gas solubility with increasing pressure, more gas is released at the ends than at the centre of the repository. Due to relatively low permeability of the host rock, the gas saturation remains below 1 % beyond 250 m from the repository in first 10^4 years (Fig. 9). The decreasing heat generation and the increasing heat transport, mainly via heat conduction, away from the repository reduce the temperature level in the host rock significantly causing a maximum temperature of 34 C in the repository (Fig. 10). The solute (nuclide) with initial mass fraction of 0.3125 in liquid phase in the repository does not migrate vertically beyond 200 m from the repository within 10^4 years (Fig. 11).

Assuming that the hydro-fracturing due to pressure build-up in the rock can occur, if the fluid pressure reaches the minimum compressive principal stress, the factor of safety related to hydro-fracturing can be defined as a ratio:

F_{Frac} = |min. compressive principal stress| / p; $F_{Frac} = 0$, if any principal stress > 0 or $\sigma_{mean} > 0$.

In Fig. 12, the factor of safety related to hydro-fracturing in the boundary plane y = 5 m at t = 100 years is depicted, around which maximum pressure occurs in case THMS31. In the upper area of the model, the safety factor is little smaller than in the lower area, as the difference between the fluid pressure and the lithostatic pressure increases with depth. Postulating that the risk of hydro-fracturing is given for a safety factor below 1.3, mainly the region in the rock salt right above the repository is affected; in major part of the model, the hydro-fracturing is not expected. Additionally, to characterize the mechanical stability of the rock salt, the factor of safety related to dilatancy (increase of volume due to opening or widening of cracks) can be defined as a ratio of the dilatancy boundary, which is derived from the experimental observations in [7], to octahedral stress:

$F_{dil} = \tau_{dil} / \tau, \quad \tau = (1/3) [(\sigma_1 - \sigma_2)^2 + (\sigma_1 - \sigma_3)^2 + (\sigma_2 - \sigma_3)^2]^{1/2}$,

$\tau_{dil} = 0.8996|\sigma_{mean}| - 0.01697|\sigma_{mean}|^2$, ($\tau_{dil}$ and σ_{mean} in MPa),

$F_{dil} = 0$, if any principal stress > 0 or $\sigma_{mean} > 0$; $F_{dil} < 1$: Mechanical stability is affected.

As the factor of safety related to dilatancy lies far beyond 1, the mechanical stability of the entire rock salt can be expected (Fig. 13). The criterion for the hydro-fracturing is clearly stricter than the dilatancy criterion, since for the dilatancy criterion all principal stresses should only be negative but for the hydro-fracturing criterion all principal stresses should be sufficiently negative. For an integral comparison, the pressure in the repository and the nuclide migration from the repository are presented in Fig. 14 and 15. The pressure in the case THMS31 with coupled mechanical effects is substantially lower than in the case THS31 without mechanical effects. In all three cases, the nuclide migration from the repository is nearly same, around 66 % of the initial mass of 10^7 kg in 10^4 years, as the net effect of the driving pressure difference and the effective permeability is not very different. Yet, the higher permeability in the cases THMS31 and THS33 leads to a little higher release than in the case THS31. The net effect of the driving pressure difference and the effective permeability on the nuclide migration is difficult to estimate without detailed numerical analyses. According to the postulated stress-porosity-permeability relationship, the impact of the coupled processes is noticeable, which can be enveloped reasonably well by the limiting hydrological cases THS31 and THS33. In other situations with a more sensitive stress-porosity-permeability relationship, the impact of the coupled processes can be substantial.

6 Conclusions

To study the coupled thermo-hydro-mechanical (THM) processes and their influence on gas and nuclide transport in a two phase flow configuration in porous media, a sequential coupling of the thermo-hydrodynamic code TOUGH2 and the thermo-mechanic code FLAC3D is described and applied to study three dimensional gas, heat and nuclide transport in a repository for heat generating radioactive waste in rock salt. The scoping coupled THM analyses show that the transport behaviour of the contaminated two phase fluid is noticeably influenced by the transient stress conditions. The present method can be applied to quantify safety margin related to hydro-fracturing and dilatancy due to fluid pressure build-up and can help to define bounding analyses, if the hydrological properties, like porosity, permeability, and capillary pressure, depending on mean effective normal stress are employed.

Symbols

A: normalized activation energy [K]; k: permeability [m^2]; M: molecular weight [g/mol]; m: mass [kg]; p: pressure [Pa]; p_b: bubble entry pressure [Pa]; Q: mass flow [kg/s]; S: phase saturation; t: time [s]; T: temperature [C]; V: volume [m^3]; X: mass fraction; β: rock compressibility [1/Pa]; ε: strain; φ: porosity; θ: temperature [K]; σ: stress (tensile: > 0, compressive: < 0) [Pa]; σ_j: principal stress (FLAC3D: $\sigma_1 \leqslant \sigma_2 \leqslant \sigma_3$) [Pa]; τ: octahedral shear stress [Pa]; ψ: Bishop factor.

Subscripts: aux: auxiliary; i: index of a TOUGH2 element; k: index of a FLAC3D connected zone; n: index of a FLAC3D grid point; s: time step.

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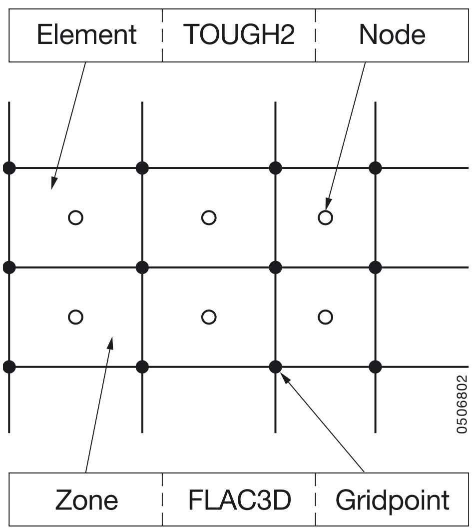

Fig. 1: Identical mesh for TOUGH2 and FLAC3D.

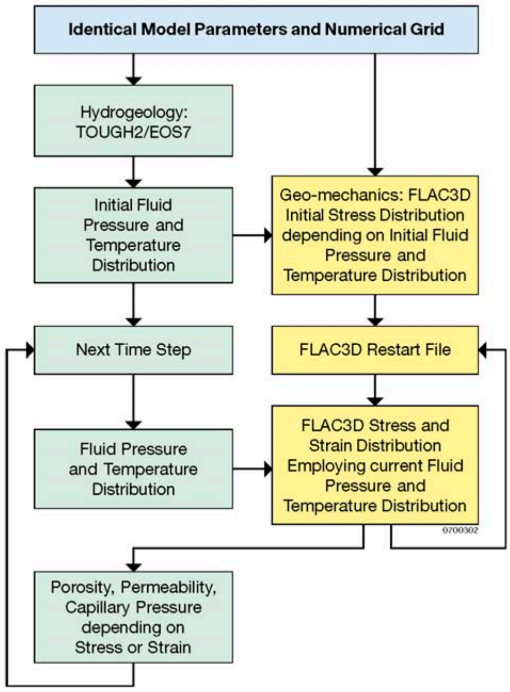

Fig. 2: Sequential coupling of TOUGH2 and FLAC3D.

Volume of repository	1E5 m^3
Density of liquid phase	$\rho_{Water}(p,T)$
Dynamic viscosity of liquid phase	$\mu_{Water}(p,T)$
Dynamic viscosity of hydrogen	8.95 E-6 Pas
Gas constant of hydrogen	4124 J/(kgK)
Molecular weight of liquid phase	18 g/mol
Molecular weight of hydrogen	2 g/mol
Henry constant for hydrogen, C_{Henry}	7.31E9 Pa
Mol. diffusion coefficient in liquid	5E-11 m^2/s
Porosity of barrier rock (cap rock)	0.05
Porosity of rock salt (host rock)	0.01
Permeability of barrier and host rock	1E-19 m^2
Porosity of repository	0.4
Permeability of repository	1E-12 m^2
Residual liquid saturation, $S_{Liq,Res}$	0.2
Residual gas saturation, $S_{Gas,Res}$	0.0
Compressibility of rock, $\beta = (1/\varphi)(\partial\varphi/\partial p)$	0
Thermal conductivity of total model	2 W/(mK)
Specific heat of total model	1000 J/(kgK)
Thermal expansion coeff. of total model	4E-5 1/K
Dry rock density of total model	2000 kg/m^3
Elastic bulk modulus of barrier rock	2E9 Pa
Elastic shear modulus of barrier rock	1.2E9 Pa
Elastic bulk modulus of repository	30E6 Pa
Elastic shear modulus of repository	2E6 Pa
Elastic bulk modulus of rock salt	18.12E9 Pa
Elastic shear modulus of rock salt	9.843E9 Pa

Table 1: Reference model parameters.

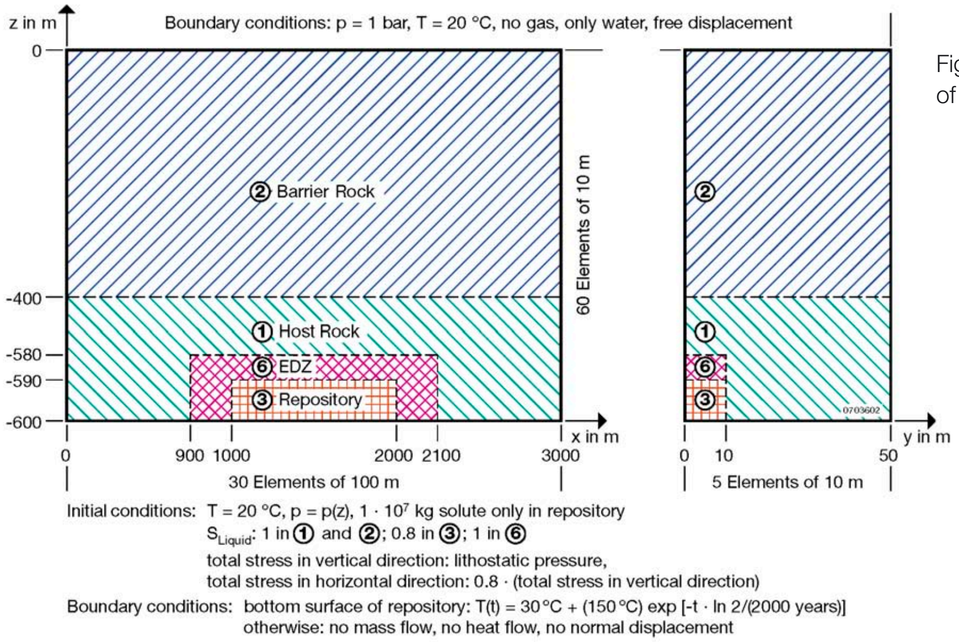

Fig. 3: Three dimensional model of a repository in rock salt.

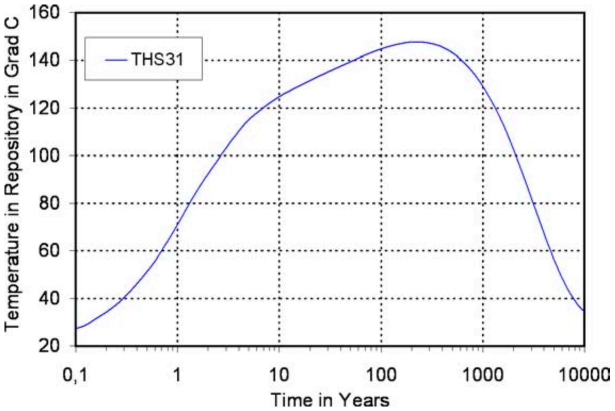

Fig. 4: Temperature in the centre of the repository in case THS31.

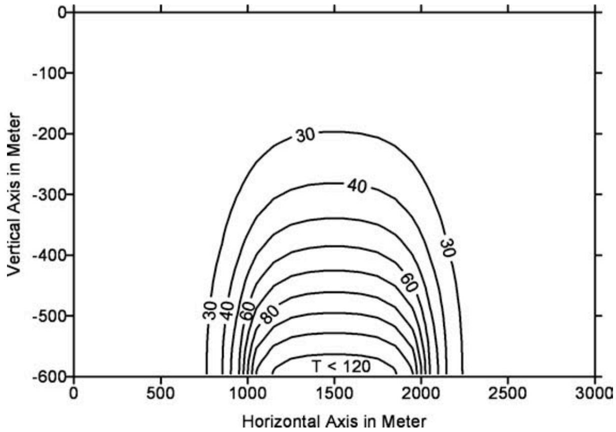

Fig. 7: Temperature (grad C) at y = 45 m and t = 1000 years in case THS31.

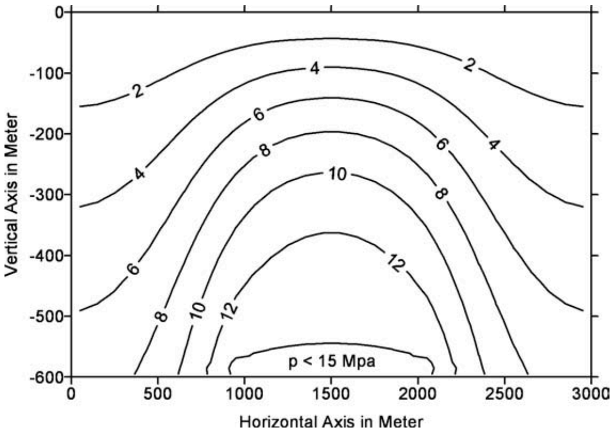

Fig. 5: Pressure (MPa) at y = 5 m and t = 1000 years in case THS31.

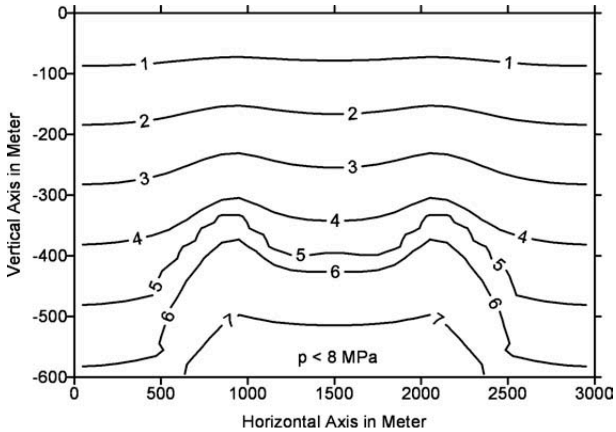

Fig. 8: Pressure (MPa) at y = 5 m and t = 10000 years in case THS31.

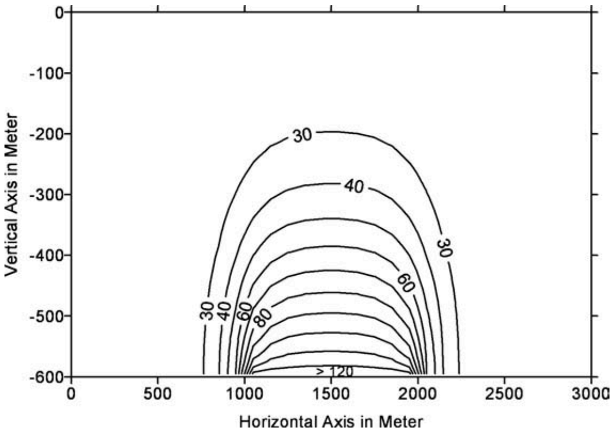

Fig. 6: Temperature (grad C) at y = 5 m and t = 1000 years in case THS31.

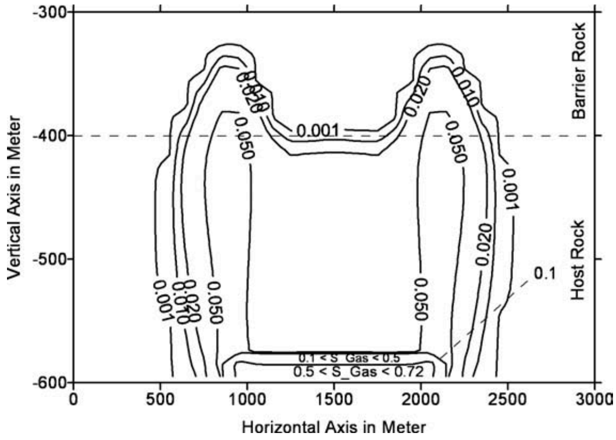

Fig. 9: Gas saturation at y = 5 m and t = 10000 years in case THS31.

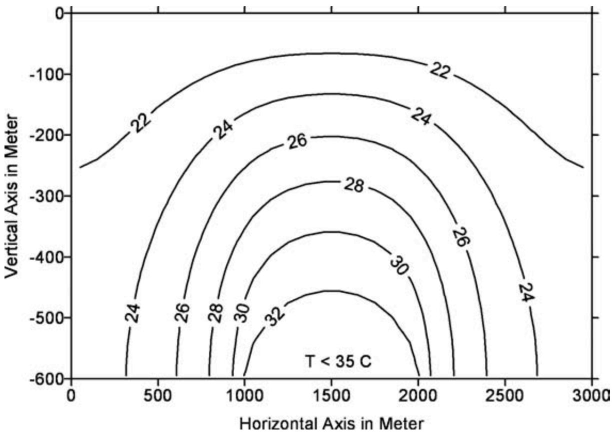

Fig. 10: Temperature (grad C) at y = 5 m and t = 10000 years in case THS31.

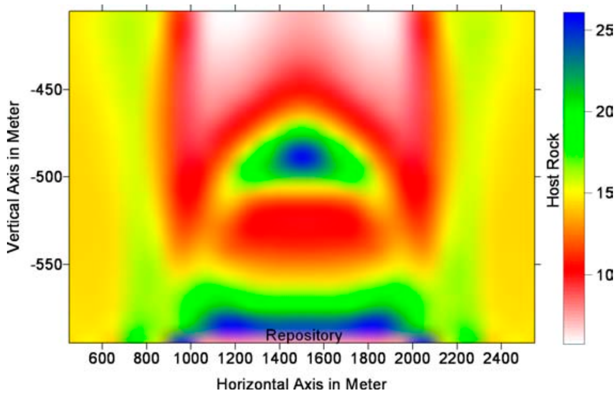

Fig. 13: Factor of safety regarding dilatancy at y = 5 m and t = 100 years case THMS31.

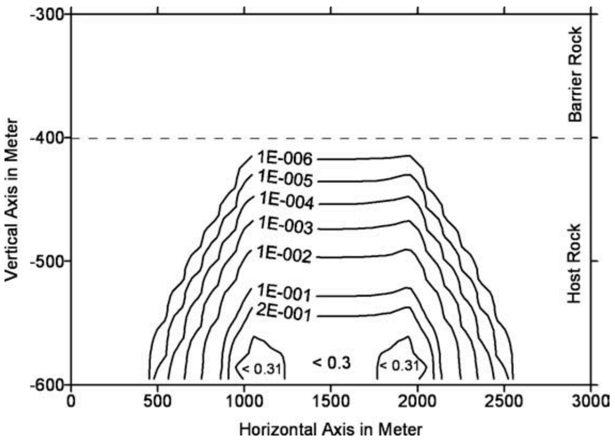

Fig. 11: Nuclide mass fraction at y = 5 m and t = 10000 years in case THS31.

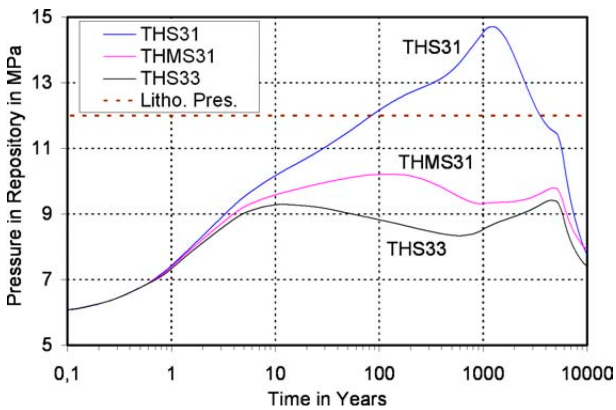

Fig. 14: Pressure at the centre of repository.

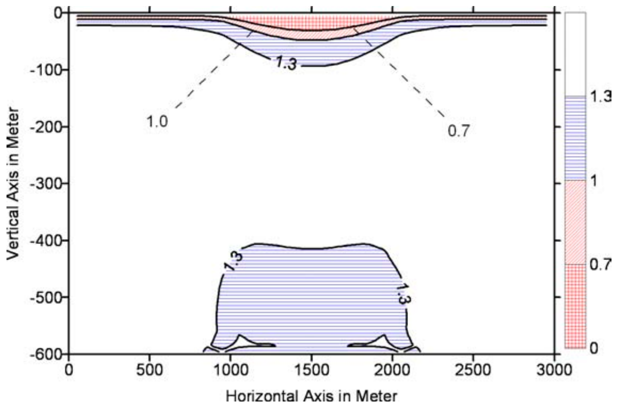

Fig.12: Factor of safety regarding Hydrofracturing at y = 5 m and t = 100 years in case THMS31.

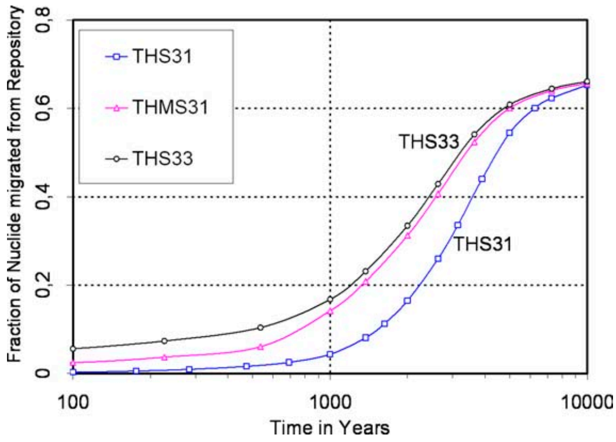

Fig. 15: Fraction of nuclide migrated from repository.

Application of the dilatancy concept to ascertain the damage and healing behaviour of rock salt

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Abstract

Besides other host rocks, rock salt formations are considered for the long-term storage of radioactive waste to exclude a threat to the biosphere. This means that the host rock's integrity has to be guaranteed during construction, operation and in the post-closure phase of a repository. Consequently, the contribution of the geological barrier to the safety of a repository has to be assessed by studying its natural characteristics and the main processes which make the transport of radionuclides possible or which can prevent the transport into the biosphere. In this context, the impacts of disturbance induced by the excavation of the underground facilities and long-term effects during re-compaction of the EDZ are the most important items. Therefore, understanding of the reciprocal action of damage respectively healing in rock salt is of vital importance for performing long-term safety analysis.

The integration of the relevant processes into the constitutive models requires consistent experimental data sets. The purpose of this paper is to illustrate the progress of experimental work performed in the last decade focusing on this issue. Firstly, the actual knowledge state of the dilatancy concept will be introduced, which provides the criterion to decide whether creep deformation without volume increase or dilatant deformation with propagating damage will occur. Then, the experimentally well documented deformation aspects, damage and the corresponding evolution of permeability, will be broadly discussed on the basis of field and laboratory tests. Also the actual state of numerical modelling describing damage will be briefly summarized. However, the description of the mechanical behaviour of salt is not complete if its favourable capacity for self-sealing of damage is not considered. Here some progress in the experimental work will be reported. The creation and evolution of the EDZ as well as of the hydraulic properties could be modelled quite well, however, it has to be stressed that self-sealing and healing are still open issues which require further investigation.

1 Introduction

Understanding the transport properties of rock salt and their relationship to its mechanical behaviour is of vital importance for the design and safety analysis of underground cavities, in particular with respect to the long-term storage of heat-generating radioactive waste and the storage of oil or hydrocarbon gas in salt caverns. The integrity of the geological barrier requires a sufficient tightness against fluids and gases which has to be guaranteed during construction, operation and in the post-closure phase of a repository which are schematically depicted in Fig. 1-1. Therefore, the understanding of the competing processes of damage respectively compaction and healing is of vital importance for the predictability of damage-related near-field processes and long-term effects, i.e. recovery of hydraulic integrity and subsequent gas generation:

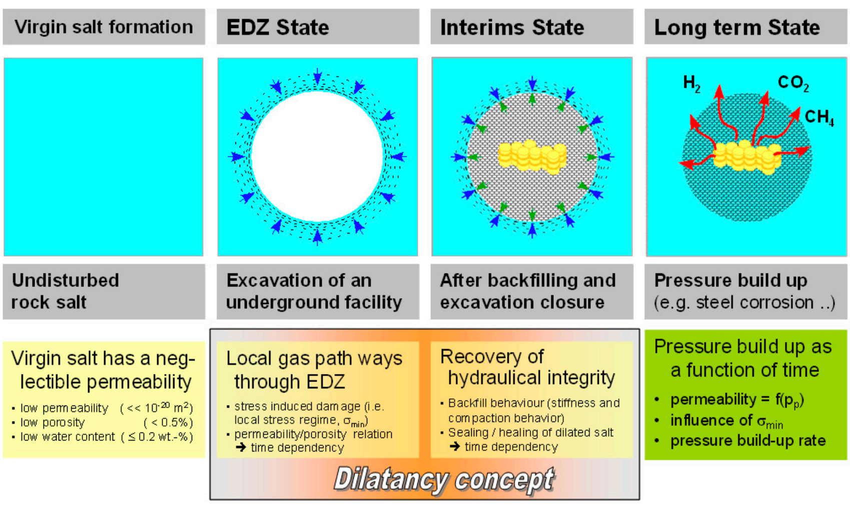

Fig. 1-1: Gas transport issues in a salt repository related to the dilatancy concept.

(1) Rock salt in undisturbed state is attributed to be impermeable for gases and fluids due to its low porosity and low permeability. During construction of underground openings in a rock salt formation, the change of stress state in the vicinity of these openings will affect the mechanical and hydraulic integrity of the surrounding rock salt by initiating local damage.

(2) The creation of the EDZ, and thus, the development of potential hydraulic pathways are closely related to stress dependent property changes, i.e. onset of dilatancy, as it was demonstrated through permeability measurements in field tests and under laboratory conditions. Depending on the order of permeability increase the EDZ has a high potential for gas transport in a repository, where gas will be produced by corrosion of metals or degradation of organic matter.

The EDZ in rock salt has been under systematic investigation for the last fifteen years as summarized during an European Commission Cluster conference and workshop held in Luxembourg in November 2003 [1]. In addition, the international conference „Saltmech6" in May 2007 in Hannover (D) gave a comprehensive overview about the actual knowledge regarding the mechanical behaviour of salt rocks and the coupled gas transport properties [2].

(3) In the post-closure phase (i.e. Interims state) of the repository, when the loading conditions will change to non-dilatant, subsequent healing (probably due to fluid assisted compaction creep) will take place restoring at least the initial gas tightness of the salt.

(4) In the long term state it is usually assumed that due to the subsequent gas production a time dependent pressure build-up with consequences of concern may occur. This aspect is separately discussed in the paper [3].

The deformation behaviour of rock salt depends on different micro-mechanical processes. For the modelling of these processes by constitutive equations and for the prediction of the long-term behaviour it is very important to distinguish between processes without dilatancy and those which are coupled with the evolution of dilatancy and damage. The principal behaviour of salt damage referred to the stress state is usually described on the basis of the so-called „dilatancy concept" which has been evaluated as a reliable basis for a prognosis of EDZ [4].

Based on this fundamental concept and referring to the general requirements of performance assessment (PA) studies the existing knowledge regarding the gas transport properties in rock salt will be briefly summarized related to the various phases of EDZ-evolution and its subsequent healing restoring salt integrity. Special account is given to the coupling between dilatancy and permeability as a base for a prognosis of the gas transport properties using numerical modelling of the mechanical behaviour of rock salt. Finally, conclusions are given about the state of knowledge obtained so far and remaining deficiencies.

2 The dilatancy concept

In contrast to crystalline rocks like granite, rock salt deformation is strongly influenced by pronounced visco-plastic behaviour. It reacts to the change in stress state by creep, which reduces the stress differences caused by the excavation. Creep takes place at constant volume and without damage of the salt. Thus, the risk of macro-fracturing is substantially mitigated. On the other hand, a micro-fractured EDZ with typical extents of several decimetres up to one or two metres develops around openings. Representing the relevant stress states, the dilatancy concept facilitates to distinguish between processes without dilatancy and those which are coupled with the evolution of dilatancy and damage [4].

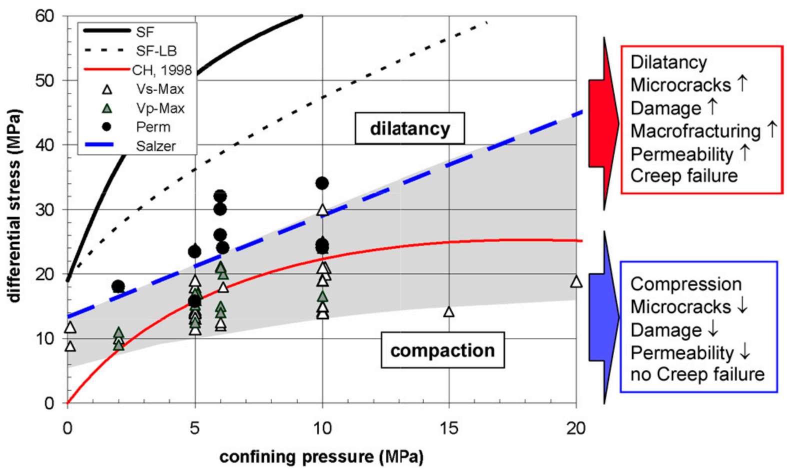

Fig. 2-1: The „dilatancy concept" - current understanding of EDZ nature and properties referring to the mechanical behaviour of salt in the stress space [4]. Experimental results from deformation tests are indicated where various micro-cracking sensitive physical parameters (Vp, Vs and permeability) were measured [6]. Short-term failure strength (compression) for rock salt (origin: BGR): SF respectively SF-LB (lower bound). Dilatancy boundaries: CH, 1998 – [4]; Salzer – [7].

In Fig. 2-1 the axes of the diagram represent the normal (σ) and the deviatoric (τ) stress, respectively. The stress space below the failure boundary is separated by the dilatancy boundary in the two domains, compaction and dilatancy. As long as the state of stresses remains in the non-dilatant compaction domain, the ductile rock salt deforms without any crack formation and without dilatant crack propagation. The transition in the state of stresses, where the deformation behaviour changes from ductile (no volume increase, $\Delta V \leqslant 0$) to dilatant deformation ($\Delta V > 0$) associated with permeability increase, corresponds to the dilatancy boundary.

Important property changes associated with dilatancy are additionally indicated. The dilatant domain is characterized by micro-cracking, causing accumulation of damage. Permeability and probability of creep failure are accordingly increasing. Air-humidity in the mine or fluids from inherent brine inclusions can permeate through the dilating salt, causing a humidity-assisted increase in ductility [8]. In the non-dilatant domain, rock salt is „compressible": micro-cracks are compacted, closed or even healed, and further micro-cracking is suppressed. Accordingly, permeability decreases and no failure will occur even during long-term deformation.

Although the existence of a dilatancy boundary is an experimental fact, it has to be mentioned that this boundary is more a transitional field than a distinct line, because the detection of onset of micro-cracking depends obviously on the sensitivity of the measured parameter. High-resolution ultrasonic velocity measurements (e.g. [9]) give clear hints of local onset of micro-cracking at lower stresses, whereby in axial compression tests Vs decreases sooner than V_p (the reverse is true under extensional conditions). Since a good agreement with the onset of humidity-induced creep acceleration was observed [8] this stress level has to be understood as the „lower damage boundary" which corresponds roughly to the older formula given by [4].

Measurements of the volume change during deformation proved opening of micro-cracks (respectively onset of dilatancy), primary at significant higher stress levels resulting in a dilatancy boundary as described for instance by [7]. Importantly, only at onset of dilatancy a simultaneous increase of permeability is observed (see Fig. 2-1).

Referring to the in situ case, by excavation of an opening in rock salt, the stress state in the vicinity is strongly disturbed. High deviatoric stresses occur, and the stress state is shifted into the dilatancy field. Damage evolution is controlled by the prevailing stress field, the creep properties of the salt, and the local lithologic heterogeneity. The closer the stress state is to the failure boundary, the faster damage evolves. The stress field is influenced by the geometry of the excavation and by technical means (e.g., lining or backfill in the excavation). Salt creep rate is a function of effective stress and temperature, but is also influenced by factors such as humidity and dilatancy [10]. Dilatancy leads to a reduction of strength of the salt. The deformation by creep and dilatancy results in a redistribution of stresses and a zone of low stress in the vicinity of the excavation.

When a supporting backfill or sealing structure is emplaced, creep leads the stress state to return below the dilatancy boundary, where only compression can occur, thus promoting healing and decreasing porosity and permeability.

3 EDZ formation – Damage and permeability evolution

3.1 Micro-mechanical model of salt deformation

The deformation behaviour of rock salt generally depends on the basic micro-mechanical processes which are active in correspondence with the dilatancy concept. This is illustrated in Fig. 3-1. Therefore, this concept is also the base for the so-called Composite Dilatancy Model (CDM), for details see [11].

In rock salt, the non-dilatant creep deformation is controlled by the trans-crystalline movement of dislocations in distinct glide systems [12]. For creep without volume change the Orowan equation is used to relate the macroscopic deformation rate to the dominant micro-mechanical processes

$$\dot{\varepsilon}_{cr} \sim b \cdot \rho \cdot v \qquad (3\text{-}1)$$

where $\dot{\varepsilon}_{cr}$ is the macroscopic strain rate, b the Burgers vector of a gliding dislocation, $\rho = 1/r^2$ the density of gliding dislocations with mean distance r, and v their mean velocity. The velocity v itself depends on stress, temperature and the parameters of the micro-substructure, Q is the activation energy for dislocations, M the Taylor-factor for cubic crystal symmetry (fcc), Δa the activation area for moving dislocations and σ^* the effective stress acting on these dislocations:

$$\dot{\varepsilon}_{cr} = \frac{b}{M}\frac{1}{r^2}\nu_0 \exp\left(-\frac{Q}{R\,T}\right)\sinh\left(\frac{b\,\Delta a\,\sigma^*}{M\,k_B\,T}\right) \qquad (3\text{-}2)$$

Dislocations may propagate on {110}-gliding planes in [110]-directions, where the unavoid-able interaction of the dislocations will cause dislocation pile-ups and the build-up of internal stress concentrations. It is generally accepted, that the evolution of damage is strongly coupled with the stress concentration at dislocation pile-ups during creep deformation (e.g. [13]).

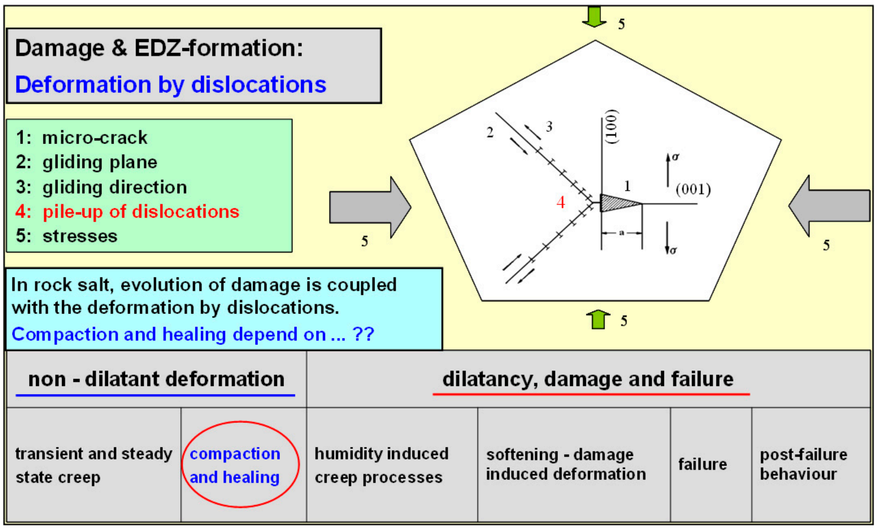

Fig. 3-1: Micro-mechanical model of non-dilatant deformation in rock salt on basis of rate controlling dislocation mechanisms and for their coupling with dilatancy related processes.

After the transient creep phase, where the micro-structure evolves as represented by Equation (3-1), the steady-state creep behaviour is achieved as soon as the generation of dislocations and the deformation hardening by dislocation pile-ups is compensated by thermally activated recovery processes - however, the steady-state creep behaviour will only hold as long as the stress generates no trans-crystalline micro-cracks at a pile-up of dislocations. This will happen in the dilatant stress domain.

Microstructural investigations demonstrate that at sufficiently low effective mean stress, dilatancy can occur, which involves different crack mechanisms. Fig. 3-4 shows both, oriented intragranular crack opening or tensile cracks (Mode 1) and, in addition, diffuse dilatancy, i.e. opened grain boundaries, producing significant permeability (see chapter 3.3)

Since rock salt exhibits no brittle failure behaviour, but even a ductile post-failure behaviour, also this process is described by CDM as a function of the creep rate. Incorporating humidity induced creep, which takes place only in dilated rock salt, the total strain rate $\dot{\varepsilon}_{tot}$ is expressed by the creep rate $\dot{\varepsilon}_{cr}$ where the impact of the humidity induced creep on ductility is denoted by F_h, that of the damage (i.e. damage induced weakening/softening) by δ_{dam}, and that of the post-failure behaviour by P_F

$$\dot{\varepsilon}_{tot} \sim P_F \cdot \delta_{dam} \cdot F_h \cdot \dot{\varepsilon}_{cr} \qquad (3\text{-}3)$$

In case of the deformation in the non-dilatant stress domain, these additional impact factors have the value of unity.

The damage function δ_{dam} depends on the irreversible volume change energy d_{dam} (briefly: damage energy) which evolves during the deformation under the conditions of the dilatant stress domain

$$\delta_{dam}\,(d_{dam}, \sigma_{min}) = \exp\,[\delta_1 \cdot (d_{dam}/\sigma_u)^{\delta_2 \cdot \sigma_{min}}]. \qquad (3\text{-}4)$$

The determination of the damage and the affected microstructural parameters, as described in [11] for the CDM - for example, is not a simple task, but it has to be pointed out that the various stages of damage can be convincingly simulated until failure. Alternative concepts are summarized in chapter 3.4.

3.2 In situ findings

The occurrence of the EDZ, and thus, the development of potential hydraulic pathways, is closely related to stress dependent property changes. This was demonstrated through permeability measurements in field tests at various sites since the beginning of the eighties.

It has been generally confirmed that the extent of the EDZ relies on the stress state and the geometry around the opening. Typically, it ranges between a few decimetres up to 1 to 2 m. Permeability measurements at the 800 m level of the Asse salt mine [14] showed, that the usual cross section of a drift with a flat floor and a domed roof leads to a larger extent of the EDZ below the floor compared to the walls and roof - corresponding to the state of stresses around a drift of that shape. Another well investigated example is the dam building project in the salt mine Sondershausen. Using conventional packer tests IBEWA determined the depth distribution of permeability in various temporal stages, as depicted in Fig. 3-2.

After 30 years lifetime of the original circular drift (Ø = 3 m) the extent of the EDZ is in the order of around 1 m, as indicated by decreasing permeability ranging from a maximum permeability of around 10^{-16} m^2 close to the drift opening to around 10^{-21} m^2 corresponding to the undisturbed rock. As a prerequisite for installing a bentonite-based sealing element, the drift was altered to a rectangular cross section. The low permeability measured one month after alteration demonstrates that the cut-off of dilated contour parts is a useful method to improve self-sealing effectiveness when constructing technical barriers. Repetition of the measurements after two years showed that the initial permeability profile is nearly restored.

In addition, it has to be mentioned that anisotropy effects corresponding to the acting stress field have always to be considered. Recently, a field test performed by GRS in the Asse salt mine, as measured by a near-drift system, nicely documents that the corresponding permeability behaves strongly anisotropic in the rock contour (e.g. [15]). In general - due to the highly anisotropic stress field in the drift contour - gas flow occurs preferentially in a plane parallel to the excavation surface whereas perpendicular gas mobilisation is inhibited.

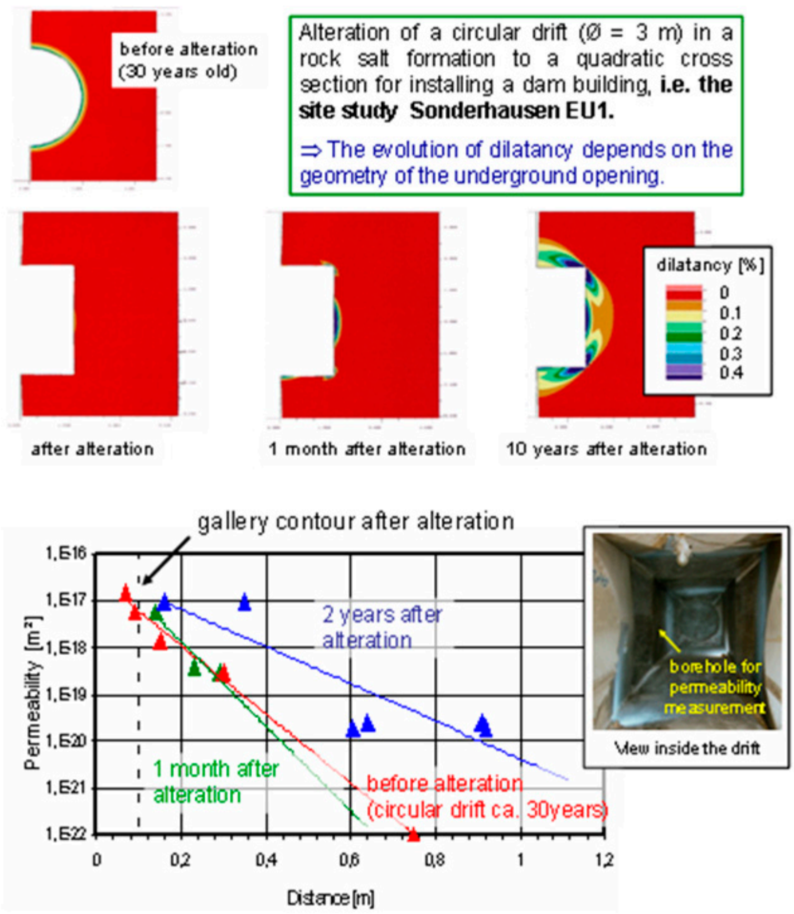

Fig. 3-2: EDZ-evolution in the rock contour of a 30 years old circular drift before and after alteration to a quadratic cross section. (top) Comparison of numerical calculations simulating the various drift states with time. (bottom) Results of repeated permeability measurements in the rock contour of the drift related to various phases of alteration (before and after – 1 month respectively two years, after [16]).

3.3 Outcome of laboratory investigations

During the last two decades, substantial progress has been made in the understanding of the effect of deformational conditions and parameters on the salt permeability on the base of laboratory tests on core samples at well defined conditions. Syn-deformational monitoring of various crack sensitive parameters (i.e. volumetric strain, permeability and ultrasonic wave velocities) facilitates to discriminate the actual state of damage during progressive deformation as discussed in chapter 2. Complete data sets are depicted in the Figs. 3-3 and 3-4.

Permeability under deviatoric stress conditions („dilatancy domain") evolves in different stages corresponding to stress, strain and time. In deformation experiments on natural rock salt (e.g. [17; 6]) and synthetic fine-grained halite [18] the onset of dilatancy is found to be accompanied by a drastic permeability enhancement of up to 5 orders of magnitude after the pore space had dilated by a small amount (< 1%), followed by a period of plus/minus constant permeability during strain hardening up to 10% axial strain or even more (compare Fig. 3-3).

Remarkably, as can be seen from Fig. 3-6, the order of permeability increase depends significantly on the acting minimal stress (for details see [5, 6]). This suggests that the evolution of permeability is not only a function of dilatancy but also of microcrack linkage as proposed by [18].

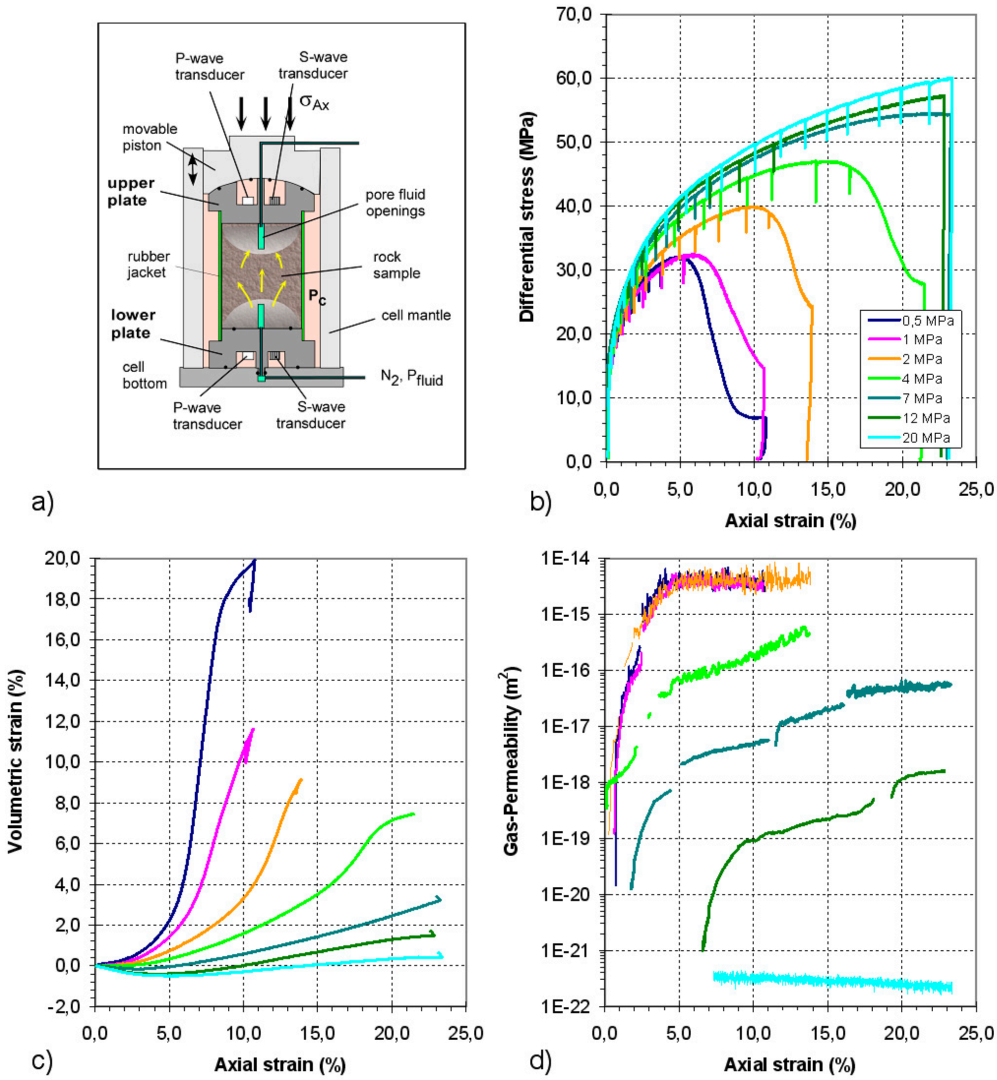

Fig. 3-3: Monitoring of damage during triaxial deformation – Experimental series of short term strength tests on Leine rock salt from the salt mine Teutschenthal/ Angersdorf. a) Experimental set up. Experimental results plotted vs. axial strain: b) Strength; c) volumetric strain and d) gas-permeability.

Also, the influence of the stress field affecting the permeability has been demonstrated [6]. As schematically depicted in Fig. 3-4, depending on the load geometry in the triaxial test, the micro-fractures are oriented preferably perpendicular to the minimum principal stress, i.e. relating to in situ conditions parallel to the excavation surface. Remarkably, also the relationship of the variation of the seismic velocities V_p and V_s depends on the stress state which again demonstrates the influence of crack geometry. However, it has to be mentioned, that a direct correlation of measured elastic wave velocities with transport parameters, i.e. permeability or porosity is difficult (compare Fig. 3-5).

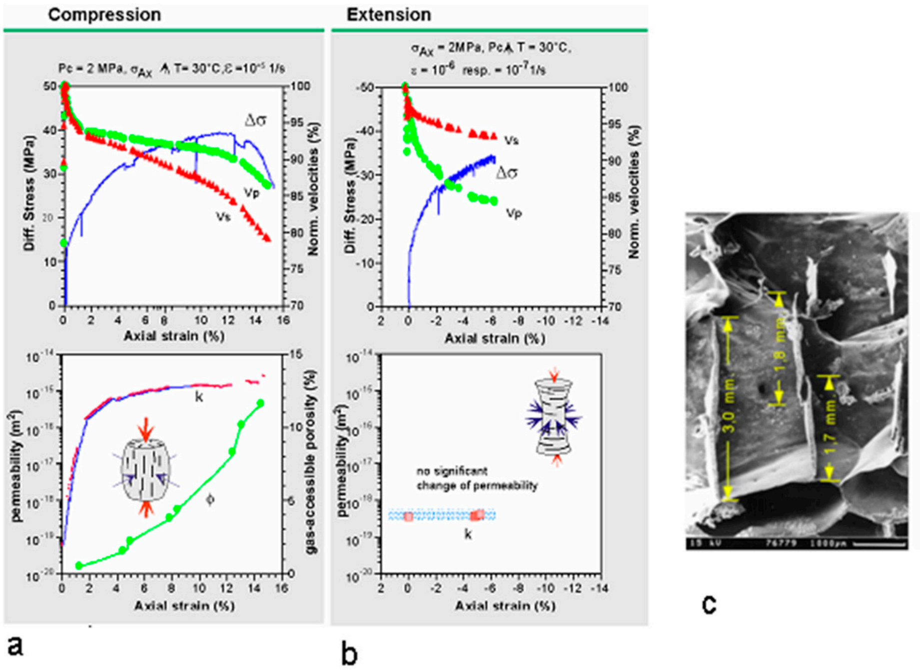

Fig. 3-4: Investigation of dilatancy, permeability, and seismic velocities under triaxial deformation (after [6]). a) compression: $\sigma_1 > \sigma_3$; b) extension: $\sigma_1 < \sigma_3$; c) REM-microstructure of dilated rock salt (pore space skeleton after impregnating the sample with epoxy and dissolving the soluble salt).

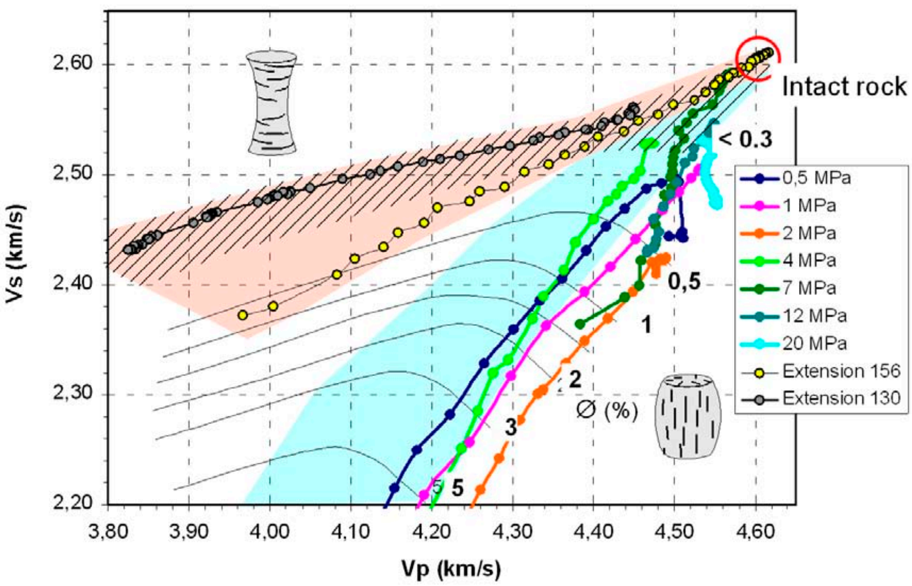

Fig. 3-5: Variation field of Vs vs. Vp together with isolines of porosity for rock salt. Data sets include the triaxial test series, depicted in Fig. 3-3 and older results of [6]. Note that the maximum Vp- and Vs-values correspond to the intact rock state without dilatancy.

3.4 Permeability-porosity relationships

The numerical modelling of fluid transport properties of dilated rock salt can be performed on the base of two approaches, either by a phenomenological description of experimentally derived relationships between the relevant parameters or simulating the fracture geometry and network properties based on textural investigations of damaged samples. Although the capability of the latter physical funded procedure seems to be obvious, the available micro-structural data base representing in situ conditions of the low-porous rock salt undergoing deformation might not be sufficient for the definition of the equation parameters (e.g. [19]). Therefore, the summary is constrained on the first approach.

In Kozeny-Carman's and other classic models permeability is described as proportional to simple integer powers of the relevant pore geometry parameters, i.e., porosity, hydraulic radius, tortuosity and/or specific surface area. For a given process, these parameters are usually assumed to be related to each other through power-law relationships, therefore leading to a power-law dependence of permeability on dilatancy respectively porosity (ϕ), possibly with a non-integer exponent:

$$k = k_0\, \phi^n \qquad (3\text{-}5)$$

Extensive laboratory testing aiming on deformation induced permeability changes were performed covering a wide range of loading conditions (see Fig. 3-6). Although there is a significant data scatter, it is obvious that a single power-law exponent does not always hold as porosity changes. One possible approach is to keep the power-law representation but with a variable exponent representing at least two parts of permeability evolution ([6]): (1) an initial steep increase due to progressive development of micro-cracks, and (2) beyond a certain threshold boundary a saturation state with moderate increase due to widening of created pathways. In addition, it is important to note that the threshold until reaching the saturation level in region (2) is obviously a function of σ_{min}.

Based on this concept, an improved description of the dilatancy induced permeability increase in rock salt under consideration of the minimal stress was recently presented in [20] (note the various model curves in Fig. 3-6):

$$k = \frac{k_{tp}}{\left(\left(\frac{\phi}{\phi_{tp}}\right)^{-n_1} + \left(\frac{\phi}{\phi_{tp}}\right)^{-n_2}\right)} \qquad (3\text{-}6)$$

n_1 and n_2 are constant inclination values according to the two relevant porosity/permeability slopes in the double logarithmic diagram. The other parameters are depending on the minimal principal stress σ_{min} (for parameters of the constitutive equations see Table 3-1):

$$k_{tp} = a_k \cdot \exp(-b_k \cdot \sigma_{min}) \qquad (3\text{-}7)$$

$$\phi_{tp} = a_\phi \cdot \exp(-b_\phi \cdot \sigma_{min}) \qquad (3\text{-}8)$$

Table 3-1: Parameters for describing permeability as function of porosity depending on the minimal stress σ_{min} (after [20]).

Parameter	Value	Parameter	Value
a_k	4.27E-14 m^2	a_ϕ	0.0263
b_k	1.26 MPa^{-1}	b_ϕ	0.3093 MPa^{-1}
n_1	4	n_2	1.07

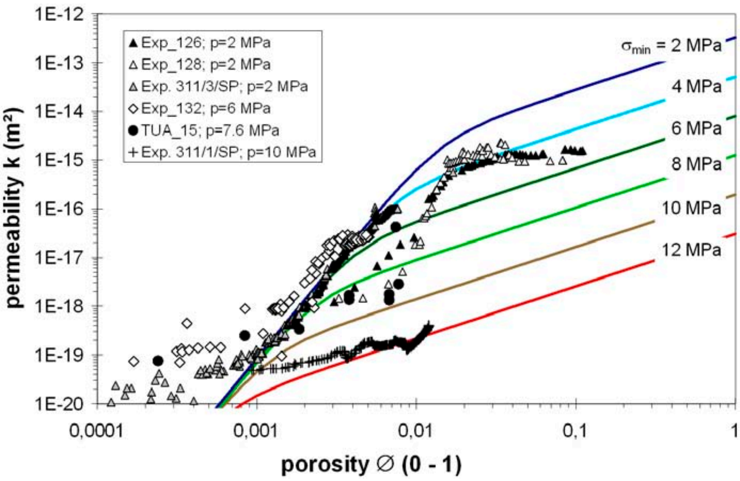

Fig. 3-6: Measured and calculated permeability/porosity relationships depending on the minimal stress (modified after [5]). The various modelling curves result from the relationship of [20].

Although the modelled permeability data seem to be slightly overestimated in region 2 the new concept of stress and porosity dependent permeability evolution has been successfully proved by [20] in the BAMBUS II-project.

3.5 Rock mechanical modelling of EDZ and associated property-changes

For the prediction of the mechanical behaviour of rock salt respectively the evolution of the EDZ and its coupled hydraulic behaviour, all those processes which contribute substantially to the time-dependent and spatial evolution of stress and strain in the material have to be taken into account (e.g. [21]). They comprise non-dilatant creep as well dilatancy and damage affected deformation processes. In general, each constitutive model describes a phenomenon (i.e. single deformation process) by one or a set of appropriate equations. Then, the combination of these modules reflects the overall coupled deformation processes, which allows beyond others the quantification of damage resp. porosity which can be connected to the hydraulic properties via a porosity/permeability relation of the form described in the foregoing chapter. Implementation of the model into various numerical (commercial or scientific) codes facilitates its universal application. Actual constitutive models and the used numerical codes are summarized in Table 3-2.

Exemplarily, Fig. 3-2 demonstrates the capability of numerical modelling by recalculation of the spatial and time dependent evolution of dilatancy around a drift in a salt mine where the cross section was changed from circular to quadratic. Note the induced stress concentration in the edges whereby most of dilatancy evolving with time is concentrated on the centre of the walls respectively the floor and roof. For comparison, permeability measurements were done at various depths in a borehole in the centre of the left wall. Repetition of the measurement nicely demonstrates the time and geometry dependent evolution of permeability.

Table 3-2: Actual constitutive models and numerical codes [21]. For the various abbreviations and details of the codes see the mentioned paper. The different codes are distinguished between finite element method (FEM), the finite different method (FDM) and the distinct element method (DE).

Institution	Constitutive model	Code
Hampel	Composite dilatancy Model (CDM)	FLAC (FDM)
BGR	CDM	JIFE (FEM)
IfG	Minkley	FLAC (FDM), UDEC (DE)
IfG	Günther/Salzer	FLAC (FDM), UDEC (DE)
INE-Pudewills	FZK-model	ADINA, MAUS (FEM)
TU Clausthal	Hou/Lux	MISES3 (FEM), FLAC (FDM)
IUB Hannover	Multimechanism Deformations Coupled Fracture (MDCF-IUB)	UT2D (FEM)
Uni Barcelona	COupled DEformation, BRIne, Gas and Heat Transport	CODE_BRIGHT (FEM)

However, a general problem has to be mentioned. The permeability will be an isotropic feature as long as the anisotropic crack formation is not explicitly taken into account. To the authors' knowledge, only models, which are based on damage mechanics as proposed by [22], incorporate anisotropy into a stress dependent porosity-permeability relation.

4 Post-closure phase – Recovery of hydraulical integrity

In the post-closure phase, after the end of excavation and backfilling the shear stress in the rock salt is continuously decreasing by creep until the isostatic state of stress is reached. Therefore, the state of stresses in the EDZ will consequently move from the dilatant into the non-dilatant domain. At last, this causes the re-compaction and the related decrease of the permeability in the rock salt of the EDZ.

Reestablishment of hydraulical integrity of the EDZ at in situ conditions, at least partial, has been recently confirmed by in situ investigations at a unique test site existing on the 700-m level of the Asse salt mine [14]. There, a cast steel liner of about 20 m length had been installed in a drift and backfilled with concrete as early as in 1914. Permeability measurements demonstrate that under the floor of the open drift, a typical EDZ with 1.5 m extent and permeability up to above 10^{-16} m^2 had evolved. Around the lined drift, however, the permeability had diminished to values between 10^{-20} m^2 and 10^{-19} m^2.

Although a considerable reduction was reached, the permeability of the undisturbed salt is attributed to significantly lower values, which may indicate that no real healing is obtained but self sealing due to compressive crack closure. It has to be stated that in this work the term „compaction" is preferred to describe such processes of decrease whether porosity or permeability. This term does not distinguish mechanically induced crack closure (due to hydrostatic or shear-enhanced compaction) from true healing (due to mass transfer by chemical processes like solution and reprecipitation, recrystallization etc., e.g. [18]). The latter will accomplish a recovery of the cohesion between crack planes. With respect to this definition the evolution of the temporal compaction and the permanent healing of rock salt have to be investigated with long-term tests using pre-damaged samples, i.e. after triaxial strength testing as depicted in Fig. 3-3.

The compaction is performed at T = 30 °C by the application of an isostatic pressure which is increased stepwise up to p_{iso} = 12 MPa and then decreased in steps again (whole testing time: ~ 70 d; for details of the performed experimental work see [23]). Isostatic loading of the dilated specimen results in a spontaneous but rather small decrease of permeability respectively porosity, as can be seen in Fig. 4-1 and Fig. 4-2. During constant loading sections, the porosity and the permeability are further decreasing, where a short transient behaviour with a more rapid decrease in the first stage of a section is detected until a more or less stationary decrease is obtained. The time dependence is plotted in Fig. 4-2.

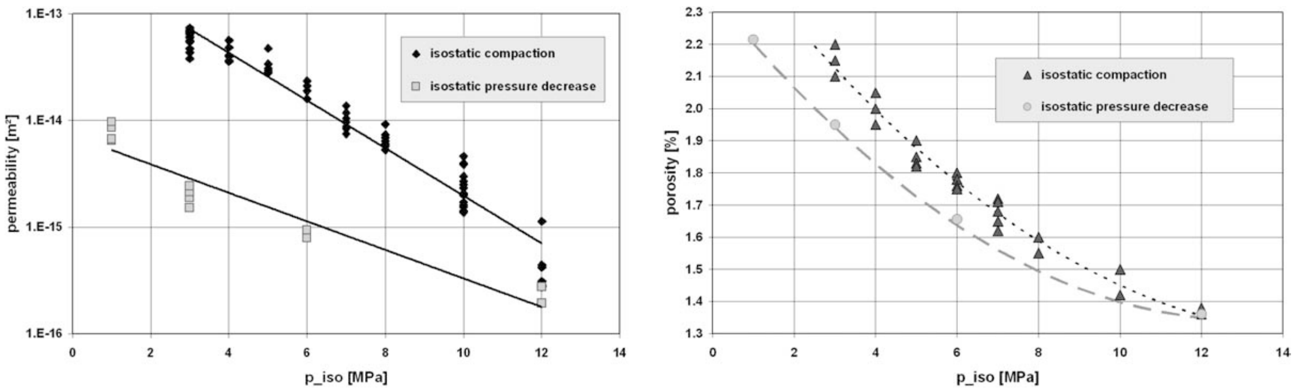

Fig. 4-1: Evolution of the general trends of the porosity (right part) and the permeability (left part) in dependence on the stepped isostatic pressure, where the time dependence is dominant (Fig. 4-2). Permanent decrease of porosity and of permeability (i.e. healing) is obvious.

However, stepwise unloading of the sample results in a spontaneous increase of both, permeability and porosity, which partially restores the initial state of dilatancy. Therefore, both figures clearly demonstrate that the permanent volume decrease (i.e. real healing) remains small. The permanent decrease in permeability achieves one order of magnitude, but decreases in between about three orders of magnitude during the stepwise increased isostatic pressure.

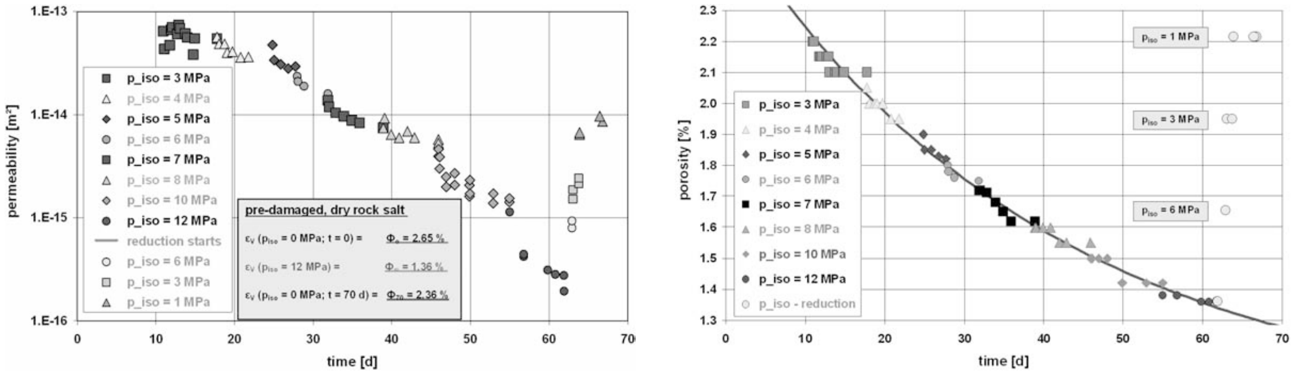

Fig. 4-2: Evolution of porosity (right part) and permeability (left part) in dependence on time during stepped isostatic loading with a time dependent transient compaction and decrease of the permeability, but a spontaneous de-compaction and permeability increase after 61 days of compaction.

The variation of the consecutive measured porosity and permeability during the compaction respectively decompaction cycle is compiled in Fig. 4-3. Remarkably in the double-logarithmic plot the relations between permeability k and porosity ϕ of both, compaction and decompaction cycle, follow straight lines but both slopes are much steeper than the relationships, as observed in the dilatant stage according to the formalism of [22].

However, it becomes obvious that the acting mechanisms for rock salt undergoing dilatant deformation with porosity production and the reverse, i.e. compaction of predilated or granular salt with porosity-destroying processes are different as usual (e.g. [24]). Although it seems clear that humidity effects may promote healing in dilated salt experimental data for quantifying the effect are not available. Nevertheless, in a first attempt, the hysteresis between the k-ϕ-trend lines is attributed to permanent compaction (i.e. healing).

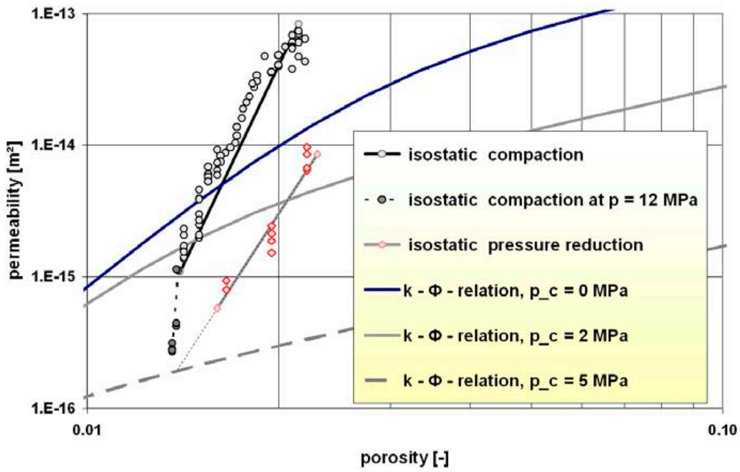

Fig. 4-3: Evolution of permeability in dependence on porosity during isostatic compaction. Dark solid line represents measurements during the stepwise increase of the isostatic pressure $p_{iso} \leqslant 10$ MPa, dark dashed line those during the section with $p_{iso} = 12$ MPa, and light solid line those during the reverse cycle, i.e. the stepwise reduction of p_{iso}. For comparison, the evolution of permeability and porosity during loading in the dilatant stress domain is plotted after [20].

Summarizing the results of long-term compaction experiments with durations of several months performed by BGR and IfG demonstrate that permeability of dry salt at a given hydrostatic pressure decreases exponentially with time:

$k = k_0 \cdot e^{(-a \cdot t)}$ with a = compaction coefficient and t = time (d) (4-1)

As shown in Fig. 4-4 the compaction coefficient a varies by a factor of 10 depending on pressure, whereas temperature seems to be of minor importance.

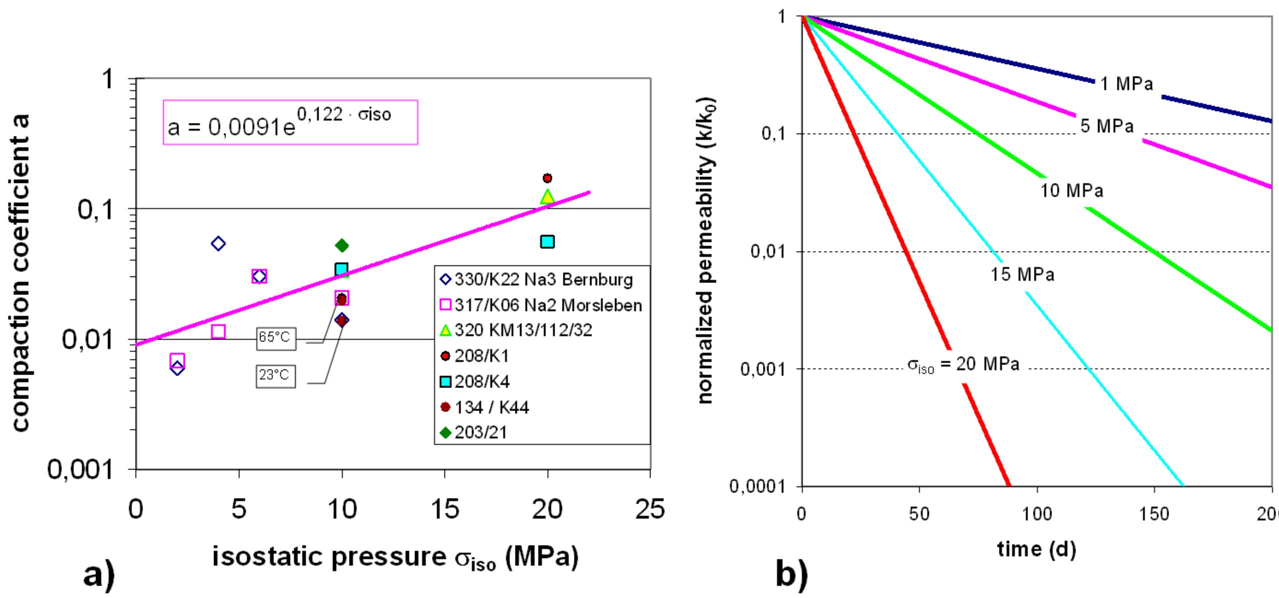

Fig. 4-4: Pressure induced permeability decrease with time. a) Evaluation of isostatic long-term compaction tests with continuous permeability monitoring (not shown here) using a simple exponential approach. b) Development of the normalized permeability with time at various pressure stages according to the respective compaction coefficients a as determined before.

5 Concluding summary and recommendations

Summarizing the previous sections, the following statements regarding the actual knowledge about damage and healing in rock salt can be made. They are based on the dilatancy concept and incorporate the relation with the gas transport properties of rock salt:

- ***EDZ-formation:***

Gas flow through rock salt is mainly restricted to the excavation damaged zone (EDZ). Well proven techniques for quantifying the extent and spatial geometry of the EDZ in salt formations are available (e.g. permeability, electrical, seismic....). During the excavation of an underground opening, dilatancy and EDZ evolution start to take place without delay:

- induced by deformation induced stress re-distribution due to the excavation
- EDZ evolves with time, depending on geometry and size of excavation, rock properties, excavation technique, mechanical support (e.g. backfilling, lining).

The extent of the EDZ is limited, i.e. no interactions between various zones. From a geomechanical point of view the EDZ is well understood:

- Based on sophisticated laboratory investigations with syn-deformational monitoring of various physical crack-sensitive parameters (e.g. permeability, volumetric strain, ultrasonic wave velocities) a comprehensive experimental data base is available.
- Verified rock mechanical models for predicting its initiation and evolution exist.
- Permeability-porosity relationships depending on σ_{min} are available.

However, the knowledge about hydro-mechanical coupling is not sufficient. Healing capacity of salt was proven but its time dependence and the relevant influence parameters (i.e. humidity) are not well understood.

• ***Recovery of hydraulic integrity - Sealing resp. healing***

Due to experimental difficulties it has to be considered that the experimental database is not sufficient, i.e. long-term compaction experiments with simultaneous permeability measurements are missing. The remaining challenges are:

- Understanding of physical processes which control the efficiency of healing in dilated rock salt with respect to humidity effects respectively pore pressures.
- Transition from dilated rock to natural rock conditions.
- Development of generally agreed constitutive models for the compaction of dilated rock salt.

Acknowledgements

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Prioritizing R&D for seven radioactive waste disposal options — an independent, interactive approach

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Abstract

A panel of university geoscience students has created a ranked list of research and development priorities concerning the disposal of radioactive waste. This prioritization is the result of a two month process during which the students were introduced to the disposal problem, submitted independent literature reviews, and developed selection criteria. Seven aspects regarding disposal were considered; geologic disposal in granite, in tuff/basalt, in salt, and in clay; sub-seafloor disposal; on-site disposal of mine tailings; and mineral encapsulation of radionuclides. The diversity of options provides the potential for dramatically different balances of geological, technological, political, and economic criteria in creating priorities. An emphasis on research and development forces the panellists to consider the role of near-term choices in changing the nature of long-term options. The prioritization process resulted in a method and ranking that has various similarities with the strategies and conclusions of other panels considering radioactive waste disposal options. Since the student panel was independent and not affiliated with stakeholders in the field, the similarity provides evidence for the robustness and objectivity of existing decision making processes. Furthermore, the unusual composition and focus of the student panel resulted in a noteworthy emphasis on the development of prompt and international solutions.

1 Introduction

Nuclear energy is a proven technology for future electricity generation. In 2002 there were 441 operational nuclear power reactors in the world, and another 32 under construction [17]. The main problem with nuclear energy is the safe management and disposal of the resulting radioactive waste [1]. On the other hand, despite public concerns about nuclear waste disposal, the technology still remains the largest proven carbon free generation option [26, 37], accounting for about one-fifth of the electricity produced annually in the US and about one-quarter of that produced in the UK [37]. Thus the future of nuclear power depends on how the safety issues regarding nuclear waste disposal are solved [26].

Global radioactive waste production statistics are not publicly available, but the USA (104 reactors) alone has some 160,000 spent fuel assemblies (45,000 metric tons) in storage, and it produces another 7,800 (2,000 metric tons) every year [38]. Mine tailings by volume far exceed the SF and HLW production. Of all the waste produced, some 5 per cent needs to be disposed of in a geological repository [4]. Fifty years ago, the US National Academy of Sciences already recognised the potential of disposing radioactive waste in deep geological formations [34], and it is now generally seen as the preferred means of disposal for high level and long-lived radioactive wastes [20]. Furthermore, the geological disposal of nuclear waste has been sanctioned by the International Atomic Energy Agency (IAEA) as the safest form of storage of high level waste away from human exposure.

Although there has been prolonged agreement as to the appropriateness of underground disposal, there is yet to be an implementation of proposed programs [20]. This is due to the fact that the discussion regarding the best overall substrate for geological disposal and whether to solve the waste problem nationally or internationally is still on-going. At the same time, discussions focusing on the immobilization and isolation of radionuclides in the underground repositories by adequate packaging in specific materials are directly linked to the choice for a particular repository substrate.

The following report is only supplementary to these debates, yet it offers a collective opinion from a distinctive perspective. It is based on an earth science related-project carried out by a panel of seven students enrolled in a Liberal Arts and Sciences college with no prior knowledge of radioactive waste disposal methods. This unconventional approach could be an interesting comparison against proverbial conclusions repeatedly drawn in the past, by panels consisting of highly-experienced, well-informed specialists. As Ch. McCombie (Arius, Switzerland) stated; the panels involved in drawing up site selection criteria often do this for the simple reason of their expertise. The student panel approach allows for a much more free-thinking discussion about the various disposal options and priorities for R&D.

The report discusses the criteria-based ranking of seven given disposal methods as priorities for research and development.

This includes the added options of the safe disposal of mine tailings and the possible improvement of containers for packaging of waste via incorporation in ceramics. Above-surface storage and transmutation research were not included due to the limited relation to the earth sciences; new mining techniques (ISL) were not included to keep the project focussed on the already existing waste.

It is hoped that this report will contribute to gain a broader perception; one which incorporates opinions reflective of a subsequent generation, on policies governing waste disposal and the use of nuclear energy in general. More importantly, it is expected that the approach may offer a small, if any, contribution to the scientific goal of the prolonged deliberation process.

2. Methods

2.1 Prioritization procedure

The ten week procedure for arriving at the final ranking for research and development priorities regarding seven cases concerning the disposal of radioactive waste included 8 stages. The procedure is summarized in Fig. 1.

1. Seven research and development options for radioactive waste disposal were selected. Only a single short list of options was derived, because of the small panel size (7 students) and the limited time allowed for the project. Four options (disposal in clay, salt, granite, and tuff) were provided in order to allow a wide range of siting issues to be considered. Sub-seabed disposal was included to gauge the panel's response to the environmental issues inherent to that option. Disposal of radioactive mine tailings allows the consideration of an understudied and more immediate class of environmental, health and socio-economic threats, a waste form with a different characteristic and volume, yet directly linked to the production of the other waste types through the resource cycle. The final option, encapsulation of radionuclides in minerals introduces the consideration of new technologies that can change the nature of the disposal problem by enhancing engineered barriers.

1. Selection of 7 options for R&D prioritization related to the disposal of radioactive waste
2. Basic theory regarding related geological, physical, chemical, hydrological, logistical, and technical aspects is presented and discussed with the student panel
3. Each student panel member writes a document with arguments for and against the assigned option by doing literature research and using an extended list of criteria (table1)
4. During a panel meeting the option evaluation documents are reviewed and discussed
5. Using a two-step ranking procedure, a ranking was agreed upon. The two steps were based on the overall balance of the individual option rankings and the subsequent voting on a priority ranking from high to low after discussing the results for the overall balance
6. A reduced list of criteria was produced (table 2) by analyzing the arguments for or against the various options as based on the extended list of criteria (table 1)
7. Interactive poster presentation of the prioritization rankings and criteria to experts in the field of radioactive waste processing, storage and disposal
8. Final ranking based on traditional decision matrix which also takes into account the weight of the comments of the experts on a certain criteria and on specific R&D criteria

Fig. 1: Prioritization procedure for 7 R&D research options concerning radioactive waste disposal.

2. The panellists were given an introduction to the nuclear fuel cycle and radioactive waste disposal problems. This included an overview of the properties of nuclear fission, radioactive decay, radiation properties, a description of state of the art above-ground storage, and lectures on relevant topics in geology, hydrology, geochemistry, seismology, and tectonics. This stage

provided the panellists with the background knowledge needed to interpret relevant literature.

3. Each panellist was assigned one of the seven options to explore and evaluate using the information provided in stage 2 and by consulting literature. A month time was given to write a literature review and describe the strengths and weaknesses of the assigned option as a research priority. Panellists were also asked to present the strongest possible arguments for supporting their particular option. This encouraged them to take on some of the characteristics of a typical panel having a diversity of areas of expertise and of personal biases. An important element in this process was the use of an extended list (table 1) of possible criteria to consider when arguing for the pursuit of the assigned disposal option. This list is fairly typical of those considered by nuclear waste policy panels, although less exhaustive than most.

4. The information from the seven literature reviews was shared and debated for two hours during a panel meeting. Written review reports prepared by the panellists were distributed to all students and studied. The merits of the various options were discussed as a panel, allowing equal time for each option with several rounds of questions and answers. .

5. A consensus was reached on the ranking of the options following a two-step ranking procedure in a subsequent panel meeting that lasted 2 hours. Each panellist was asked to submit a simple ordered ranking of the seven options, and these rankings were then summed to create an initial draft of the panel's priority list. Subsequently, each panellist stated the strengths and weaknesses for each option based on the most weighed criteria in their individual ranking process. After a discussion of the arguments the panellists voted again on the ranking. In this case, single votes (cast blind) from each member for a favourite option were gathered first. After the most popular 1st choice was identified, a single vote for 2nd choice from the remaining options was used to identify the panel's 2nd choice before the remaining options were considered in the same way to create the rest of the ranking. This two-stage ranking procedure explored the effects of differences in panel voting methods and of the influence of sharing information during ranking discussions.

6. In the same panel meeting a reduced list of prioritization criteria was developed Each of the strengths and weaknesses cited during the ranking process was classified according to the original list of criteria (table 1) presented to the panellists. Those criteria not used were discarded from the original list. The remaining criteria were combined into broader categories in such a way as to create a short list of criteria felt to have roughly equal weight in the ranking process. The original lists of strengths and weaknesses were then classified using the new criteria (table 2). This post-hoc characterization of selection criteria and weighting resembles a simple and practical realization of logistic regression analysis applied to an interactive multiple criteria decision process [32].

7. The criteria for each option and the R&D rankings and assessments were summarized and submitted for review by experts in the field by means of an interactive conference poster (presented at Reposafe, November 6-9, 2007). The poster allowed reviewers to submit their own rankings and argue for or against a certain R&D option. This provided the panellists with access to expert testimony and information to which they had not previously been exposed.

8. The expert comments were evaluated and discussed by all panellists after the conference in a 2 hour panel meeting session and used to create a final ranking for R&D options. First a general eligibility ranking for the various options for radioactive waste disposal was agreed upon. This involved the making of a more traditional decision matrix using the reduced list (table 2) of criteria. Next, the implications for this ranking using criteria explicitly related to research and development were used to determine the final ranking.

Table 1: Extended list of criteria

Initial criteria list for the evaluation of the 7 R&D options for RW disposal
• (human) exposure risk - (IM, IA)
• terrorism - (IA)
• transportation - (TF, AV)
• retrievability - (IA, TF)
• geological abundance of substrate - (AV)
• availability requirements - (AV, PF)
• environmental impact - (IM, AV)
• catastrophe risks - (IM, IA, AV, PF, TF, FF)
• monitoring and maintenance options - (TF, FF)
• radionuclide mobility - (IM)
• realization, operation, maintenance/closure cost - (TF, FF)
• political acceptability - (PF)
• geological hazards - (IM, IA, AV, PF, TF, FF)
• rock stability for repository creation - (AV, TF)
• groundwater table and flow - (IM, AV, TF, FF)
• stability of governments - (PF)
• communication to present and future generations - (PF,TF)
• required engineered barriers - (IM, IA, TF, FF)
• governmental stability - (PF)

Table 2: Reduced list of criteria

Reduced criteria list for the evaluation of the 7 R&D options for RW disposal	
• Immobility	(IM)
• Inaccessibility	(IA)
• Availability	(AV)
• Political feasibility	(PF)
• Technical feasibility	(TF)
• Financial feasibility	(FF)

2.2 Selection of criteria

The six main criteria categories to evaluate the characteristics of the various RW disposal options encompass one or more criteria in the extended list (table 1 and 2). Since the criteria of the long list fall in several of the six main criteria categories (as indicated between parentheses next to each criterion in the long list), the approach can be considered a fuzzy analysis of the various RW disposal research options. The analysis does not include the weighing of the suitability for R&D efforts, but rather, serves as input for R&D considerations. The six categories of criteria can be described as follows:

The immobility criteria consider the general containment of radionuclides in geological nuclear waste disposal, taking into account all possible pathways to exposure to the biosphere, humans in particular. These pathways include fluid transport mechanisms such as transport in water, oil or other media that could bring the waste in contact with people or increase background radiation to a level that exceeds the allowed dose. The geochemistry of the rock composition of a repository is a second factor that is discussed in its contribution to the mobility of waste. Additionally important is the proneness of a particular stratum or site to seismic hazards or alternatively, those features of the host material that will promise minimal transport of the waste in a seismic event.

The inaccessibility criteria include the protection against unwanted invaders of the repository and against the formation of unorthodox entry ways to the RW resulting from (geological) hazards. At the same time, inaccesability is also a factor hindering monitoring, maintenance, and retrieval and re-use. So depending on the choice of whether or not to allow accessibility, this criterion will have a high or a low score. The student panel is convinced that an unnoticeable closure and sealing of a deep repository is favorable for guaranteeing proper safety in the near and far future.

The availability criteria in the context of nuclear waste disposal refer to the global and local abundance of current disposal options. Also considered is the availability of a particular disposal option allowed by governments and the (longer-term) availability of other resources required for implementation.

The category of political feasibility criteria encompasses the likelihood that all stakeholders (e.g. governments, peoples, environmental movements and businesses) involved are in agreement and adhere to international or national regulations. The chance of regulations being modified or created to allow certain RW disposal solutions is generally unpredictable.

The issue of technical feasibility is often given a high weight in decision making processes. It focuses on physical realization as well as on available techniques. Also considered is the required technology to prevent malfunction or accident to such a degree that it is guaranteed that no major environmental disasters can occur. Technical feasibility is highly interconnected with the issue of financial feasibility, since most of the technical challenges are surmountable, as long as money is not an issue.

The financial feasibility criteria incorporates issues such as development, maintenance and closure costs, marginal cost, relative environmental cost, and the economics of scale. Usually financial feasibility is of high importance for the choice of a repository option, but not always in the right way. Often the tendency is to choose the cheaper option over the safest one. However, long-term safety is certainly worth the money.

3 Results and Discussion

3.1 Individual evaluation of the seven RW disposal options

After the panel was introduced to the theory and science of the various aspects involved in the disposal of RW, each member evaluated and characterized the suitability of their assigned disposal option in an individual report. The results were then summarized and can be found in the next paragraphs.

3.1.1 Mineral barriers

Containment in minerals, which is defined as the vitrification of nuclear waste in durable accessory minerals, can potentially achieve the goal of immobilisation of long-lived radionuclides that are present in nuclear waste. Accessory minerals are mainly ceramics such as zircon ($ZrSiO_4$), apatite ($Ca_5(PO_4)_3(F,Cl,OH)$) and monazite ($LnPO_4$), which are present in host rock in trace amounts. Accessory minerals will serve as a natural alternative to Synthetic rock (Synroc) or the already commercially used borosilicate glass, in packaging of Spent Nuclear Fuel prior to deep geological disposal. It is conceived that the combination of radioisotopes with minerals will create a more stable form of containment for the waste.

Most accessory minerals of interest are found in nature, commonly associated with a considerable amount of radioisotopes. Accessory minerals have a comparatively higher thermodynamic stability than borosilicate glass (a material currently used) and are durable under high amounts of irradiation, minimising radionuclide mobility [19, 28]. Additionally, minerals prove quite cost-effective in the long-term as a containment option as they are without a shelf-life and may provide a solution to the accumulation of mine tailings, which may serve as a ready source of accessory minerals. Moreover, the possibility of using extremophilic bacteria to incorporate nuclear waste in products of their metabolism that are analogous to mineral candidates, would prove an effective alternative to the mining of these minerals if the use of mine tailings as ores is unfavorable. It is thought that certain bacteria may be able to indirectly make use of the radioactivity of the nuclides as an energy source for their metabolism or alternatively, reduce isotopes to insoluble states [1, 3, 23].

There are a number of factors that pose a challenge to implementation. There still is a sustained need for deep geological disposal of radioactive vitrified waste and thus there will also be the need for repositories that provide suitable host rocks for the minerals; these will likely be rock types with which the minerals are naturally associated, potentially demanding an international site if suitable locations are not available nationally. An added challenge concerns the potentiality that several minerals or mineral aggregates will have to be employed as they differ in radionuclides with which they are naturally associated. Finally, most transuranic elements rarely occur naturally, so it is only possible to establish empirically their behaviour in accessory minerals, a process which is time consuming and expensive in the short-term.

3.1.2 Salt

Deep disposal in rock-salt is one of the more feasible and reasonable options. Isolation, costs, and availability are the main advantages. Thermal properties and economical exploitation of the salt are the main obstacles to realization.

In more detail, the advantages of using rock salt as a host rock for a geological repository are to following. Firstly, due to its high plasticity [7], isolation of the nuclear waste is relatively high. The speed of deformation is low enough to allow filling the site before it seals itself. Monitoring and maintenance of the site will therefore only be necessary for the first couple of centuries. Secondly, salt has a high thermal conductivity, reducing the risk of container overheating. Thirdly, the potential size of a repository in salt is to be large enough to offer an international solution, making it interesting for the national economy of the host country as well.

There are, however some challenges to overcome. Firstly, research has shown that in some cases the salt composition can be altered under the influence of radiation and heat. The altered composition can become explosive, thus posing a big threat on the safety of the site. This deserves high research priority, because it can make or break the case for geologic disposal in salt. Secondly, salt deposits have historically been exploited, meaning they have multiple access shafts/ramps. This poses a threat to both inaccessibility and environmental isolation. Thirdly, if water enters the salt dome, corrosion rates of containers, as well as radionuclide mobility, are high.

3.1.3 Volcanic Tuff

Extensive research has been conducted to investigate possible storage of radioactive waste in volcanic tuff. Most of this research was carried out in the USA for their proposed storage site at Yucca Mountain. However, many scientists still doubt whether the Yucca Mountain site is suitable as a RW repository and also the efficiency of tuff in general as a host rock for such a repository.

The arguments in favour of the Yucca Mountain repository mainly concern the possible aridity of the site. Because it is a mountain, the repository can be build above the ground water table. For a geological repository it is important that radionuclides do not reach the ground water table and vice versa; therefore this is a definite plus. Furthermore, the region also experiences very little precipitation, which adds to the argument that repository can be a very arid repository. Another argument is the possibility of a 'hot repository' in volcanic tuff. This means that the repository will not be cooled and will thus stay hot enough to vaporize water for several thousands of years. Finally Yucca Mountain is isolated from major population centres and for that reason there is no strong NIMBY response [8].

Critics of volcanic tuff as a host rock and of the Yucca Mountain site claim that because the number of years during which a repository has to function is so large, there is no way of guaranteeing the region will stay as arid as it is now. A possible increase in precipitation (which is not unlikely according to future climate models) could be fatal, for water can travel through the faults and cracks present in tuff [10, 33]. Therefore, the engineered barriers for such a repository have to be able to withstand thousands of years of corrosion. There is no way to test this in a laboratory. Last but not least, there is the threat of a volcanic eruption. While the chances of such an eruption appear to be small [8], the potential damage it could cause is tremendous.

3.1.4 Sub-seabed

Sub seabed disposal appears to be a good option for nuclear waste disposal, since the distance from mankind is large. When buried within oceanic sediments, the material would not be easily accessible for the so-called „dangerous" parties. The waste would be laid to rest beneath 5000 meters of waters and several hundreds of meters of clay-rich mud. The sediments of the chosen sites have characteristics that are favorable for the burying of radioactive materials. The thick muds generally have very fine particles and therefore have a low permeability. In case a canister breaks open, the chances are small that the radionuclides will be brought up to the ocean floor. The adsorption capacity of the sediments for radionuclides is high [14], lowering even further the chance of their release. The ocean functions as a potent dilutor.

In theory, the technology that is needed for sub seabed disposal of nuclear waste is already available. Deep-sea drilling techniques could be used to place the containers hundreds of meters underneath the sea floor [38]. Radioactive waste that is disposed under the seabed must be packaged in corrosion-resistant containers or glass. Sub-seabed disposal utilizes sites in the ocean that are most stable and predictable, which usually means on the abyssal plains. The selected locations are far from any plate boundaries and would minimize chances of disruption by turbidites, volcanoes, earthquakes and any other seismic activity.

Future research would have to focus on in situ testing, for it is important to get a detailed overview of the properties of the site as well as the behaviour of the radioactive waste. More studies need to be done on the method of placing (and retrieving) the containers. It is very important that the canisters are not damaged during their placement (and possible retrieval) and that the material for the canisters can withstand the pressure difference between the surface and the ocean floor.

The idea of sub-seabed disposal might be easier to sell to society than land-based disposal, due to the „not-in-my-backyard“ syndrome. For seabed disposal, technical matters are not the main source of concern. Several technical solutions can be found for the problems that come with sea bed disposal. International laws that impose general obligations upon states to protect the marine environment form a bigger challenge. The most important restrictions derive from the London Dumping Convention (LDC) which states that „any deliberate disposal at sea of wastes or other matter from vessels, aircraft, platforms or other man-made structures at sea“ [23] is prohibited. The challenge is to come to a general agreement with all nations to allow sub-seabed disposal.

3.1.5 Granite

Next to sub-seabed, granite is the most abundant option for a geological repository. Since it is not abundant in every country [32], international cooperation would be required if granite is to be the only repository option in use. One of the reasons why granite is a promising host rock for disposal of RW is the low mobility of the radionuclides due to the low porosity and permeability. Weathering usually concerns the top 50 meters, while a repository is located much deeper. Faults in the rock can be monitored and thus a site with as few faults as possible can be selected. All major cracks are formed during the forming of the granite and those sites are the most dangerous when it comes to further cracking. Sites with only small cracks have a much lower risk of cracking further [5].

The accessibility of granite is low, since it is a hard rock in which drilling is difficult and expensive. This makes it expensive to build a repository in granite, yet the high safety of the repository can outweigh the high costs. In view of terrorism, the low accessibility is a positive feature.

3.1.6 Mine Tailings

The residues from mining and milling operations are called mine tailings [25]. Uranium mine tailings contain hazardous elements as radium, radon, thorium, arsenic, selenium and molybdenum [1], which will leak into the environment if the mine tailings are not properly disposed of. In high concentrations elements like Se, As, Mo are toxic.

Mine tailings are disposed of on-site, in the vicinity of the mine, for a number of reasons. Firstly, they have a large volume, rendering transportation infeasible. For example, the Mary Kathleen uranium mine (Australia) produced a total of 31 Mt of material from open pit mining [2]. The tailings slurry from acid leaching of uranium from the ore was pumped into a retention area of about 30 ha, 1.5 km from the mill. Secondly, there are regulations, governing the rehabilitation of mining sites. For example, the multiple layer concepts [20] are applied when tailings are to be disposed of in tailings impoundments.

On the other hand, on-site disposal of mine tailings poses a few problems. Firstly, in areas where the mine is located near a river, a slurry pipeline has to be built to relocate the tailings to another site deemed suitable. For example, the Moab (Utah) uranium mill tailings pile located adjacent to the Colorado River, released contaminants by seepage into the river and the feasibility of relocating the tailings to a more qualified disposal site using a slurry pipeline was explored [16]. Secondly, bringing back tailings to the dug out mine (for storage) necessitates their treatment, otherwise radon gas emission will continue and the risk of groundwater contamination will be enhanced [39].

3.1.7 Clay

Many natural analogues suggest that clay layers are a suitable substrate for building a repository for RW. Examples are Alligator River (Australia) [15], the Cigar Lake Uranium ore deposits (Canada) and the Oklo natural reactors (Gabon) [2, 9], and Pocos de Caldas (Brazil) [2], with regard to long-term U migration; Loch Lomond lakebed (Scotland) [30], with regard to middle-term ^{129}I migration; clays at Orciatico (Italy) [11], with regard to thermal alteration; preserved wood at Dunarroba (Italy) [35], with regard to clay's isolation capacity and biodegradation processes; and the Littleham Cove native copper deposits (England) [31], with regard to the potential stability of copper canisters.

There are several reasons why clays deserve research preference for disposal of RW. First of all, clay has the ability to self-heal, which gives it great advantages over non-plastic rocks: in case of an earthquake it can re-seal the repository; stress from geologic deformation is buffered; it can heal the Excavation Disturbed Zone. Compared to all plastic rocks and sediments, it can do so the quickest due to high creep rates [18]. Secondly, modelling of processes in clay like flow, radionuclide diffusion and retardation, and mechanical stress distribution, is more straightforward than in other media and thus requires less research. This means that less exploration effort may need to be done than for some of the other options, and that possible failure of the Engineered Barrier System under the strongly buffering conditions of a clay layer is very accurately predictable. Therefore, the spread of radionuclides after Engineered Barrier System failure in a water-saturated plastic medium is more homogeneous than in fractured systems or non-saturated plastic media. Thirdly, the retardation in clays due to sorption is more efficient than in rocks. The anoxic conditions in clay are perfect for preserving canisters and for immobilizing most important nuclides.

There are also some challenges which still need to be met. Some anionic radionuclide species (^{129}I, ^{36}Cl) do not interact with the clay as a Natural Barrier [18]. Therefore, the Engineered Barrier System needs to be designed to retain these radionuclides, and possibly also some other cationic radionuclides, because their sorbed mobility may be quite high (as already known of Ce and Sr) [4]. Also, more study on the behaviour of different types of clay under repository pressure and temperature is needed. Furthermore, knowledge in areas like hydraulic conductivity, creep rate, gas transport pathways and mechanical strength, is far from complete. The bigger challenge while this research is on-going, is to develop a uniform international reference library of Safety and Performance Analyses for clay disposal sites.

3.2 Initial Ranking and Review

The composition of the panel members was quite diverse; the seven students follow –besides geosciences - different programmes within the science curriculum of the Liberal Arts and Sciences College and take various courses in social sciences (e.g. law, argumentation and analysis) and arts and humanities (e.g. music) as well. The panel can therefore be characterized as broad minded and free-thinking.

Since the wide range of issues and theory to be addressed had to be introduced in a consecutive manner, it took about 6 of the total 10 weeks prioritization procedure before the panel was up to speed with analysing and evaluating their cases. The decision making part for the ranking of the options was relatively quick, especially after the criteria had been summarized into six main categories. The long list of criteria (table 1) was not workable for extensive comparison, especially since the process was not automated.

During the first ranking in which a two-step procedure was followed the effects of differences in panel voting methods were explored, one being the summation of individual rankings of the seven options and one being the democratic vote on the ranking of the options after a debate of the summation results. As it turned out there was no change in the rankings. The initial ranking was as follows: 1. Sub-seabed, 2. Salt, 3. Clay, 4. Mine tailings, 5. Granite, 6. Mineral barrier, 7. Tuff.

This ranking was submitted for external review in order to give the inexpert panelists an opportunity to react to expert commentary in reconsidering their choices. The review process took the form of an interactive poster where experts could give their ranking based on the six criteria (Reposafe, November 6-9, 2007). The poster presented the condensed versions of the option summaries given above together with an invitation to deliver any comments to the panelists who monitored the poster. A fair number (>13) of experts participated in this process. The most notable result was that the R&D prioritization for sub-seabed and mine tailing varied significantly among reviewers. Sub-seabed disposal was either supported as a top priority (in agreement with the panel's first ranking) or recommended for demotion to lowest priority, a strong indication of the difficulty in achieving large scale consensus on that option. Reviewers dismissed panel concerns about possible explosive effects of strong heat sources on salt. Disposal in granite received strong support because of the advanced state of the development of this option. More surprising support was received for a high prioritization of mine tailings disposal because of the possible near-term instability of many storage sites.

There were also important comments about the general nature of the prioritization process. Several reviewers remarked upon the consistency of the criteria and ranking presented with those that had resulted from much more extensive processes (e.g. CoRWM). The panel's support for an international solution was considered noteworthy. Finally, it became clear that a more explicit distinction should be drawn between the criteria used to rank disposal options and those used to prioritize options for research and development.

3. 3 Revisions and final ranking

The panel reconvened a week after the external review session in order to discuss the expert comments and determine a revised and final ranking. At this meeting a simple decision matrix was developed (Fig. 2), which can be considered an additional step to the two-step method in stage 5 of the prioritization process. In constructing this matrix, the panel agreed to apply the criteria to the ranking of the options as international disposal alternatives. Subsequently, the panel then defined R&D criteria that could be used to perform a post-hoc adjustment of the disposal option ranking to produce a ranking of research and development options. These criteria for R&D prioritization were agreed upon by all panel members.

Although R&D criteria could be added as seventh criteria category and be weighed equally together with the other six main criteria, the panel chose instead to evaluate the balance of the six main criteria and the weakest and strongest criterion for each option therein as one of the considerations for R&D. Other criteria for prioritization of the options for R&D are the time-frame in which a solution needs to be found. With existing plans to build another 32 nuclear plants, solving the waste problem associated with the nuclear resource cycle is an urgent matter. Therefore, if a relatively small research effort is needed towards realization or improvement of an option, that option could have R&D priority over an option with a higher ranking based on the six main criteria, but requiring extensive research. The present situation of the waste is also a consideration. Does the waste pose an immediate threat, or can we buy time with intermediate solutions while constructing permanent solutions? Obviously, if political consensus (international or national) is conceived to be a time-consuming process for a certain option, than despite its qualities, the option may be ranked lower than based on its suitability alone. Another consideration is that a more generic solution is preferred, so, options that are considered good but are scarcely available or not uniform in character, should be considered later than options that are more abundant. Finally, resources (energy and materials) for R&D and implementation should not outweigh the energy produced by a nuclear power plant. The amount of money required for R&D is given a low weight in the prioritization of R&D options, as the panel states that safety is far more important.

	Clay	Granite	Minerals	Salt	Seabed	Tailings	Tuff
Immobility	+++	++	+++	+++	+++	+	-
Inaccessibility	+	+	0	-	++	- -	0
Availability	+	+++	++	+	+++	na	- -
Political Feas	0	+	+	+	- - -	++	-
Tech. Feas	+	+	-	++	+	+++	++
Finan. Feas	0	-	-	++	- -	++	++
R&D ranking	**3**	**1**	**4**	**2**	**5**	**1**	**6**

Fig. 2: Criteria matrix for the 7 RW disposal options; +++ = very high, 0= neutral, - - - = very low, na = not applicable. Subsequent R&D ranking is based on considerations explained in the text

Together with the incorporation of the external comments about individual options, the clear distinction drawn between the criteria used to rank disposal options and those used to prioritize options for R&D resulted in a different ranking from the initial ranking (Fig 2). Most notably is that the sub-seabed disposal dropped in ranking significantly, since the panel became much more pessimistic about reaching political consensus within a short time-frame about the choice of ocean sites, the disposal methods and operational procedures including those issues surrounding RW transport over longer distances. If retrievability becomes a requirement in the future should transmutation be a possible and desired option, the high inaccessibility score may be a negative instead of a positive feature. The geologic disposal in granite and salt rose in ranking, because these options appear to be already well and successfully developed (granite in Sweden and Finland, salt in Germany). Granite received preference over salt due to its wider abundance and much more limited accessibility. In addition, this option could become a strong international solution with further development. It was decided that mine-tailing disposal would have to be considered separately from the ranking of the other options. Mine tailings pose an immediate problem to the environment and are not contained. Since the problems of disposal at both ends of the fuel cycle would be significantly increased by a switch from fossil fuels to nuclear power, taxing the limits of the temporary storage mechanisms now in place, rapid development of solutions to both problems seems a necessary condition for the consideration of this particular carbon alternative. This problem needs to be solved simultaneously with the search for repositories and barriers and is therefore also ranked on top. Due to the relative limited abundance of clay and its lack of uniform composition, and due to the vast amount of research that would need to be carried out and doubts about its technical feasibility to use minerals as an engineered barrier, these options received intermediate rankings. Tuff was again ranked last of all, because of the limited availability and because of climate change concerns.

4. Conclusions

The general recommendations of the student panel concerning R&D for RW disposal are perhaps best summarized by two short imperatives: act now; act internationally.

The first imperative refers both to the importance given to addressing immediate concerns (mine tailings) and to the ranking of the options. The panel noted that climate change adds to their sense of urgency. The panelists sense of urgency was further enhanced by the overall consistency of their rankings with those expressed by poster viewers and in the findings of much larger committees such as CoVRM. This was seen as evidence that consensus support for practical options is achievable in the short term.

The second imperative refers to an emphasis on the development of options that provide solutions for most or all countries dealing with RW disposal. Mine tailings and granite disposal again serve as examples. The former are present in a number of presently less developed nations (e.g. Gabon). The latter is available in many parts of the world so that development of the technology could benefit many nations (and could allow for a rapid expansion of capacity if nuclear energy use grows). At the same time, granite is an example of a solution that would require international cooperation for nations such as The Netherlands, whose only internal geologic disposal options were viewed as less safe by several panel members.

Additional observations concern the role of technical information in the process of building consensus among an inexpert panel. Good access to the latest research proved to be a deciding factor in prioritizing options. The updating of the knowledge base of the panel via interaction with expert reviewers proved to have a significant impact and seems to have made it easier to achieve consensus. This was reassuring, in that there had been some concern that the injection of new information would confuse the panelists. It was also clear that too much focus on studies to determine the absolutely optimal solution was not considered helpful. Once a class of options was shown to be similar with respect to issues of safety, the focus of the panel shifted away from technical details in the literature and toward progress reports that helped identify the option that was best developed. (parallel to geologic over borehole listing in CoRWM chapter 11)

In conclusion, given the political simplicity of the principle of urgent and international responsibilities; and given the successful use of technical literature by an inexpert panel in forming a ranking consistent with that of larger efforts such as CoRWM; it seems likely that development priorities like those given here can be implemented with consensus support in the near future, as long as efforts are made to provide broad access to the relevant information.

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Environmental Impact Assessment of a Final Dis-po-sal (High Level Active Waste)

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Abstract

An Environmental Impact Assessment (EIA) of a final disposal site of High Level Active Waste is in Germany a legal obligation. The results of the environmental impact assessment are important for the acceptance of a final disposal site by the public. For transparency in a public discussion, a common understanding about the investigation area and assessment procedure of the environmental impact assessment is required.
During the construction phase of a final disposal site relevant environmental impacts are expected, caused by land use as well as the noise and air pollution caused by transport vehicles. Environmental impacts during the operational phase of the final disposal site are relatively low, especially of non-radiological effects, and can be compared to big facilities for interim storage of radioactive material. Thus an EIA for a final disposal site is feasible.
To minimize environmental impacts, especially non-radiological impacts during the construction phase, it is important that comparable alternative sites, alternative routes for transportation and alternative technical methods exist.
This article shows only an extract of the Öko-Institute's investigations on EIA of final disposals, a topic in which the Öko-Institut gained a lot of experience by reviewing many EIA-reports of nuclear facilities for different German Authorities.

Definition of the area around a final disposal for the investigation of the environmental impact

The area, in which the environmental impact of a final disposal site has to be investigated, is the area in which the emissions of different activities of the final disposal can be measured as an immission. The area of investigation of the environmental impact is a spaciotemporal function.

The spatial dimension of investigation for environmental impact is an area in which immission exceeds a de-minimis-level. This de-minimis-level has to be defined for every kind of immission. In an ideal case it is a spherical circle around the source of emission (e.g. irradiation of a single source without shielding). Immissions in the underground do not necessarily have an impact on subjects of protection such as man, the health of man, animals, plants, biodiversity, soils, water, air, climate or landscape. A harmful effect on subjects of protection only occurs when the source of emissions is not buried deep enough in the underground (e.g. noise) or when contaminated matter is able to reach the biosphere e.g. via groundwater.

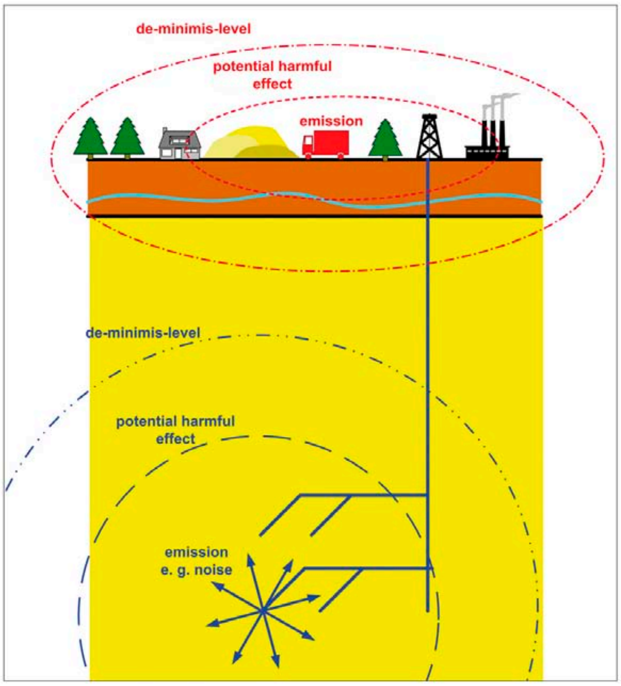

Fig. 1: Examples for different spatial dimensions of different sources of emission, different de-minimis-level and different potential harmful effect

The temporal dimension which has to be considered for the investigation of the environmental impact is the time span during which immissions, exceeding the de-minimis-level, impact subjects of protection.

Significant harmful effects on subjects of protection depend on the intensity and the lifespan of the source, on the existence of the subjects of protection in the area of immission and on the sensitivity of the subjects of protection.

Short-term impacts with low intensity are not as significantly harmful as long-term impacts with nearly the same intensity (e.g. drawdown of groundwater level for short or longer periods). Impacts with extra high intensity within short periods can have more harmful effects than impacts of the same kind but with lower intensity because of dissemination of intensity over a longer period (e.g. high intensity of impact because of much more transport vehicles over a short period versus the same amount of transport figures disseminated over a longer period).

The environmental impact of a final disposal site depends on the different activities as well as the intensity and the duration of activities during different phases of a final disposal site. A final disposal site can be subdivided into six phases:

Phases of a final disposal	Duration
Investigation	ca. 8 years
Construction	ca. 8 years
Operation	ca.50 years
Sealing	ca. 8 years
Post closure	up to 1 Million years

Examples of potential environmental impacts which differ from phase to phase of a final disposal

Land use

The potential negative effect of land use depends on the value of land for animals and plants, especially protected species. Land use during the aboveground investigation depends on the amount of drilling investigation and the necessity of new infrastructure. To drill more than 100 m deep in the underground a land use of 2.000 m^2 is necessary per deep borehole /Thiele 2005/. The amount of land use during aboveground investigation can be estimated to be around 10.000 m^2.

During the underground investigation phase land use is higher because buildings for a mine are needed aboveground and the infrastructure has to be extended. The land use for buildings and infrastructure is about 40.000 to 70.000 m^2. Additional land use for the stockpiling of excavated bedrock is the same amount during the underground investigation phase. In summary during die phase of underground investigation die land use is about 110.000 m^2.

The construction phase of final disposal site causes the highest land use because buildings and infrastructure will be extended for handling of radioactive waste and the area for stockpiling of excavated bedrock will increase. The additional land use is estimated at 200.000 m^2.

To minimize the land use of valuable country side during the construction phase, investigations of alternative sites for the final disposal are necessary to compare the impact of land use with respect to the value of the local nature.

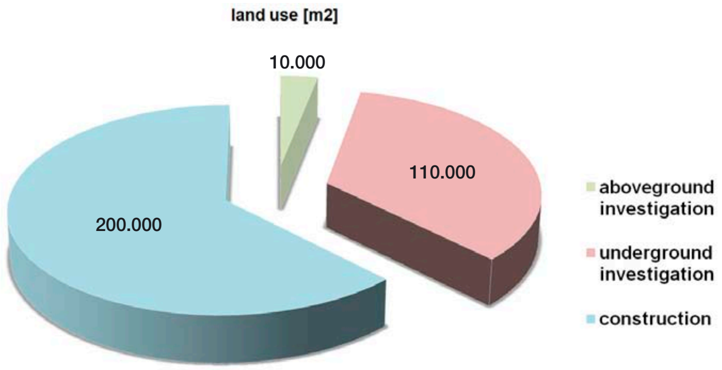

Fig. 2: Estimated land use during different phases of a final disposal site

Air pollutants, dust and noise caused by vehicles and construction site equipment

Vehicles, especially lorries and construction site equipment, cause air pollutants like nitrogen oxides, sulfuric oxides and carbon particulate emission in the vicinity of the final disposal site and along the transport routes. These can result in impacts on the surroundings and nature. The significance of the effects basically depends on the intensity of the impact and the distance between the source of emission and the subjects of protection.

Air pollution, dust and noise are very relevant during the phase of underground investigation and the construction of the final disposal site, because during this relative short time period the greatest amount of bedrock will be excavated and transported from the mine to a stockpile. During the operation phase the amount of transports per week is not as high as during the construction phase because the backfilling of bedrock depends on the time for the encapsulation of radioactive waste into the bedrock. For the encapsulation of canisters into the bedrock additional time is needed for radiation protection measures for workers, for the building of technical barriers around the canisters and for the quality control.

Assuming that 60.000 to 110.000 m^3 bedrock have to be transported within two years necessary for the underground investigation, 20 to 26 lorries per week will be needed for the transport. During the construction of the final disposal site for a time period over 8 years, 800.000 m^3 bedrock und 150.000 m^3 concrete must be transported /Nagra 2002/. Under these conditions transport figures of 37 to 58 lorries per day are expected.

During the operational phase of a final disposal site the necessary amount of transports is the same as during the construction phase but the period of transportation lasts some decades without peaks of traffic.

Environmental impacts by immissions caused by transport vehicles can be minimized by comparing alternative sites for a final disposal and stockpiling, alternative transport routes and alternative transport facilities. This approach was taken in Finland where four different final disposal sites and alternative transport routes were taken into consideration. For the environmental impact assessment of final disposal in Finland four sites for final disposal and additional different transportation routes for every site had been considered /Posiva 1999/.

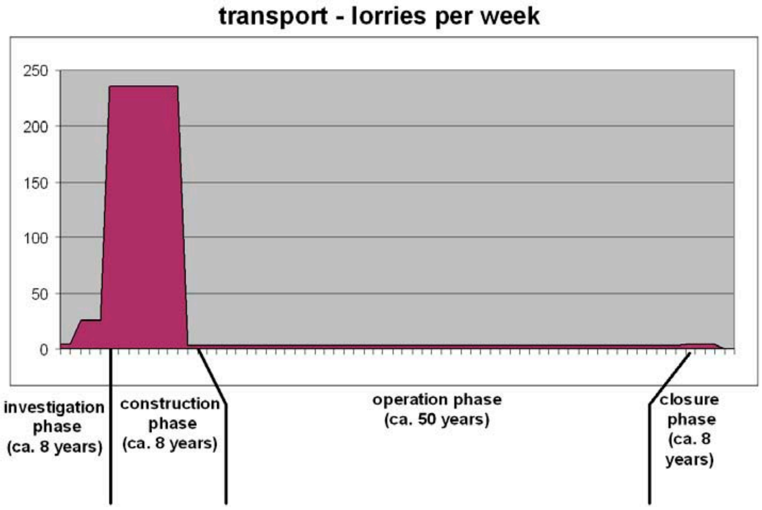

Fig. 3: Transport figures, lorries per week during different phases of a final disposal

Conclusion

The intensity of environmental impact differs during the several phases of a final disposal. Especially during the construction phase non-radiological environmental impacts as transport emissions and land use are significant. The result of an environmental impact assessment depends on the intensity and duration of emissions as well as on the existence and sensitivity of subjects of protection in the area of investigation. An Environmental Impact Assessment of a final disposal site is feasible but a minimization of impacts is requested and is possible by comparing alternative sites for the final disposal, alternative routes for transportation (especially of bedrock) and alternative facilities/vehicles for buildings and transportation.

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