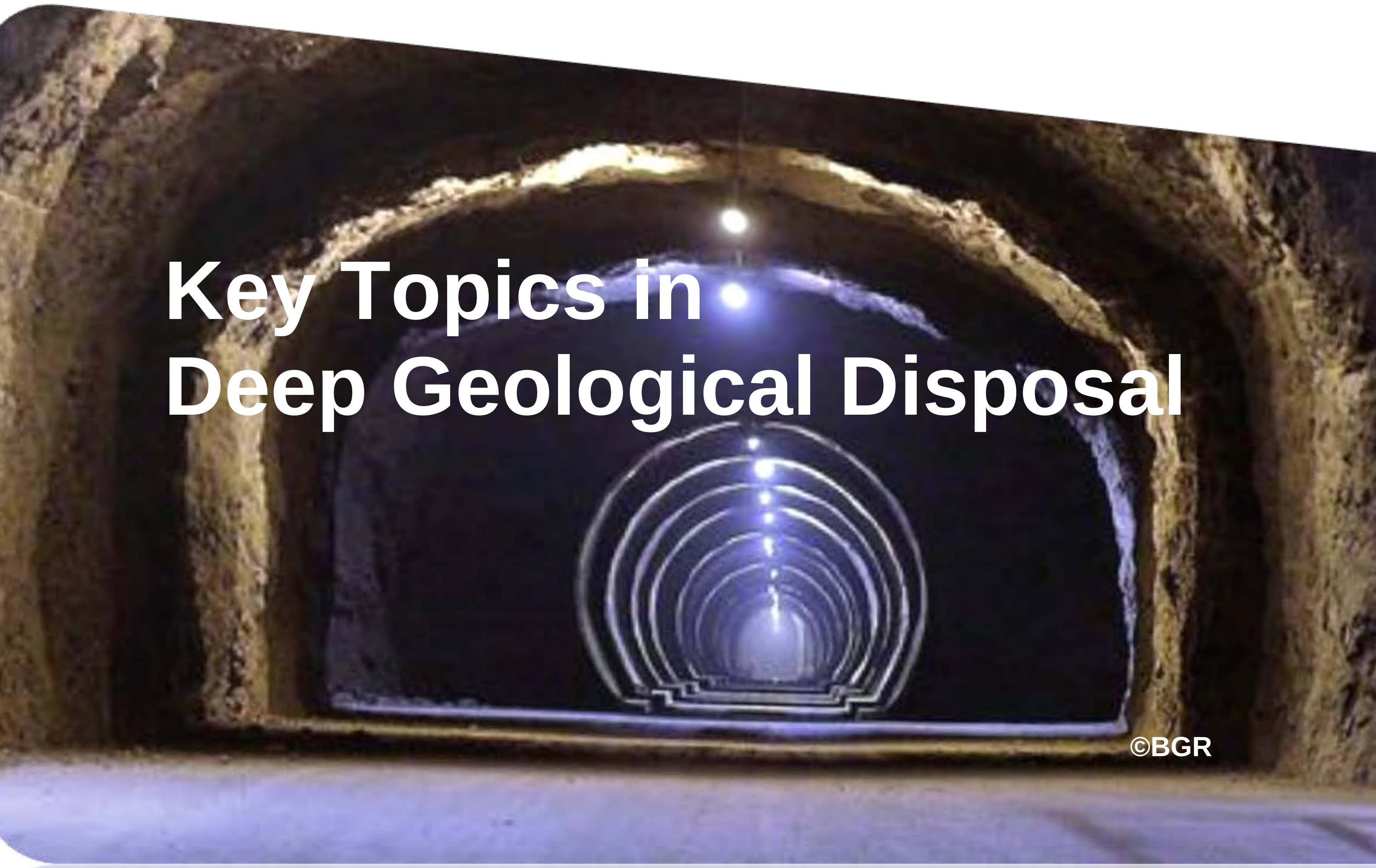
# Key Topics in Deep Geological Disposal




**Cologne, 24 – 26 September**

**BOOK OF ABSTRACTS**

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

S1-01

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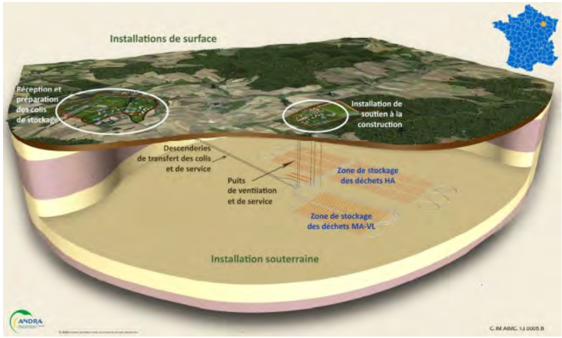
The French classification of waste is defined by the National management plan of radioactive materials and waste updated every three years. High level waste (HLW) and intermediate level long-lived waste (ILW) are planned to be disposed of in a deep geological repository subject to authorization (Cigéo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

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

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

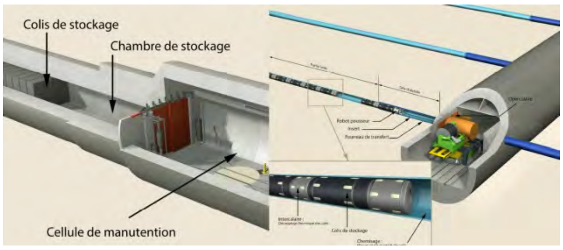
If licensed the disposal facility will be implemented at a depth of 500m, in the Callovo-Oxfordian clay layer (argillites) which has been investigated in the Meuse/Haute Marne Underground research laboratory (URL). The underground facility will be located in a 30 km$^2$ defined area close to the URL.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




Overall view of the disposal facility

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

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




Disposal cells for ILW(left) and HLW (right)

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

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

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

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

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

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

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# Preliminary Safety Analysis of the Gorleben Site

S1-02

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

The objective of the project "Preliminary Safety Analysis Gorleben (Vorläufige Sicherheitsanalyse Gorleben -VSG) was to assess whether repository concepts for the Gorleben site or other sites with a comparable geology could comply with the safety requirements governing the disposal of heat-generating radioactive waste in Germany, published by the Federal Ministry of Environment, Natural Conservation and Nuclear Safety (BMU) in 2010, based on currently available knowledge.

For this purpose, the results obtained so far from research on disposal in salt domes were compiled and assessed. The repository concepts for rock salt site were to be further developed, a long-term safety analysis was to be conducted and the results to be assessed based on the safety requirements of the BMU /BMU 10/ to identify the need for further research. Further, it was investigated whether the methodology used in the VSG is suitable for a future site comparison.

About 80 scientists from nine institutions participated in the VSG. The project was divided into four levels. The basic level comprised the geoscientific site characterisation, quantification of the volumes of waste arising and the development of a safety concept. This served as a basis for developing the technical concepts for the Gorleben site, which could then be subjected to a system analysis. In the synthesis, the results were assessed and future research needs shown.

The site description has already been documented to a large extent before the start of the project. However, results from recent exploration work were also considered in the VSG. During the project, the repository concepts had to be adapted to changes in the volumes of waste expected to arise. This was the first time that a BMU safety requirement had been implemented in a safety and verification concept for a specific site.

Concepts for the emplacement of casks in drifts or in boreholes were examined. The additional optional emplacement of waste with negligible heat generation was taken into consideration. The requirement that emplaced waste should be retrievable resulted in significant conceptual changes compared to initial considerations, in particular for the concept of borehole emplacement.

For the scenario analysis of these concepts, 115 FEPs and assumptions were derived that led to a probable reference scenario and 17 alternative scenarios. For different load cases, the numerical system analysis addressed the questions whether the integrity of the salt barrier will be maintained, whether solutions may enter waste emplacement areas, and which radiological consequences of the release of radionuclides are to be expected.

The results showed for the different load cases that there is no entry of solutions into waste emplacement areas and that compliance with the radiological acceptance criteria for the simplified verification procedure can be verified for the solution path. According to the state of the art in science and technology, the simplified verification procedure is possible for the gas path if emplacement of casks can be postulated that will be gas-tight for 500 years. If the reference points for determining the RII (Radiological Insignificance Index) are positioned in the lower shaft seal, the RII values are well below 1 (usually < 10–20) for all alternative scenarios. This shows that under the assumptions that have to be postulated in the VSG project, radiological safety can be verified according to the BMU safety requirements.

The synthesis report /FIS 13/ contains a detailed presentation of the results on the containment capacity, on the robustness, on uncertainties and load capacity as well as proposals on optimisation and identification of further R&D needs. The VSG methodology is, in principle, also suitable for other potential repository sites in salt domes.

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# Disposal of Radioactive Waste in Swedish Crystalline Rocks

S1-03

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

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

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

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

S1-04

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

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The paper summarizes the results and findings of the study "Impacts of Safety Requirement Retrievability on existing Repository Designs and Requirements for new Repository Concepts" conducted by DBE TECHNOLOGY GmbH (Bollingerfehr et al., 2014). The paper starts with a review of the "Safety Requirements Governing the Final Disposal of Heat-Generating Radioactive Waste" (BMU, 2010) and presents a conceptual idea of how retrievability could be realized in Germany. The study establishes a basis for further development and new designs for repository concepts considering German safety requirements and presents conceptual adaptations for application of retrievability in existing repository concepts.

Since the 1980's generic emplacement concepts have been developed. Those concepts were developed consistent with the former safety requirements from 1983. Full scale transportation and emplacement techniques have been tested at surface demonstration facilities consistent with these requirements.

In 2010 new safety requirements were issued that focused on retrievability; making it a strict licensing requirement. Retrievability (Rückholbarkeit) is the planned technical option to remove emplaced radioactive waste containers from the subsurface repository throughout the operational period. The new safety requirements generate several additional technical requirements that have to be met.

In order to meet these requirements measures have to be implemented that encompass the entire operational life-cycle of a repository. These include for example optimization of waste package and drift spacing, use of adequate drift lining systems as appropriate, adaption of ventilation and cooling systems, and monitoring and radiological protection measures during retrieval.

There are inherent contradictions that must be addressed to maintain a retrievability option while still achieving the desired safety function of a repository, which is the permanent safe disposal of the waste. Retrievability must not impact the passive safety barriers and their long-term safety function. Refining the repository concepts using the presented strategies may help to resolve this inherent conflict. This is demonstrated using the existing emplacement and retrievability concepts developed in the Preliminary Safety Analysis Gorleben (VSG) (GRS, 2011) and (GRS, 2012). In the VSG both horizontal drift emplacement and deep vertical borehole emplacement are considered with respect to retrievability.

The goal of retrievability is achieved by implementing a "re-mining" strategy to meet the long-term safety requirements. This strategy is characterized by maintaining the previously developed emplacement concept with minor adaptions where necessary. The void space around the waste packages is still backfilled to create the passive barrier safety function. Should a future waste retrieval decision be made, the galleries will require re-excavation.

Retrieval options with respect to the existing repository concepts for HLW disposal in clay formations have to be developed. The implementation of the "re-mining" strategy seems to be a reasonable approach; however, additional technical challenges must be answered. For example the design of a disposal drift lining system will have an impact on operational safety as well as on retrievability and the long-term performance of a potential repository.

Based on actual state of the art DBE TECHNOLOGY GmbH plans to advance the implementation of retrievability option in German repository concepts for heat generating high level radioactive waste in both salt and clay formations. This includes answering open questions of operational safety, mining and retrieval equipment, ventilation, radiological protection and also demonstrating the technical feasibility of the retrieval technologies.

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S2-01

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

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

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

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

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

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

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

S2-02

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

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

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

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

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

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# Current State of Knowledge on Long Term Behavior of Highly Active Waste Forms

S2-03

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Though physicochemical processes related to the behavior of spent nuclear fuel (SNF) and highly active waste (HAW) glass in a deep geological repository are of complex nature, knowledge on underlying mechanisms has considerably advanced over the last years. Several key issues for the long-term safety assessment of SNF and HAW glass disposal are related to the interaction of water with the waste forms. Even if geological and geotechnical barrier systems may prevent to some extent formation water from contacting the waste forms, intrusion of aqueous solution into disposal rooms has to be taken into account within the long-term.

Upon contact with water, the alteration of SNF and the consecutive release of radionuclides involve the combination of many different processes, which can be grouped into two stages: A fast / instant release of a fraction of radionuclides and a much slower and long-lasting radionuclide release that results from the alteration and dissolution of the UO2 fuel matrix. The so-called Instant Release Fraction (IRF) comprises fission products which are segregated during irradiation to the pellet / cladding gap, fractures in the pellet and grain boundaries as well as activation products within the cladding. According to a review of Kienzler et al. (2013) the IRF depends mainly on linear heat rating, burn-up and other in-reactor fuel operating parameters. As a major result of recent studies, it turns out that alteration of the UO2 fuel matrix is significantly inhibited under the strongly reducing conditions induced by container corrosion and consecutive hydrogen production. In contrast to earlier results, radiolysis driven oxidative fuel matrix dissolution appears to be less relevant for most repository concepts (e.g. Eriksen et al., 2012). Still, open questions remain related to the exact mechanism of the protective hydrogen effect on SNF matrix corrosion and the impact of potentially interfering naturally occurring groundwater trace components.

Various models have been developed to quantify the release of radionuclides and to describe the long term behaviour of HAW glass. Radionuclide release is associated to either alteration of the glass matrix or ion exchange. For these processes penetration of water into the glass network is a critical step. Water inward-diffusion is expected to be coupled by an inter-diffusion process to the outward-diffusion of alkali ions and boron (Kienzler et al., 2012). Besides the release mechanism, the fate of radionuclides is controlled by their retention in the glass gel layer and incorporation into glass alteration products, such as hydrous alumosilicates. These retention mechanisms depend strongly on the chemical speciation of the radionuclides. Applying sophisticated speciation techniques (like X-ray absorption spectroscopy), present studies focus on the determination of the concentration of rare fission products and their speciation in HAW glass.

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

S2-04

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

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

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

In addition, a second site in Europe has been identified where natural low alkali cements exist in combination with both saline groundwaters and brines, allowing examination of saline groundwater/low alkali cement interaction. Once again, the preliminary results of the study will be reported here and a comparison between the two sites will be made, highlighting the different impact of the differing groundwaters on the cement.

Although precise dates for the systems examined are not yet forthcoming, this is a focus for future efforts at both sites, allowing more precise definition of the probable longevity of low alkali cements in repository environments.

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

S2-05

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

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

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

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

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

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

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# Consensus Shaping and Safe Space Public Participation Processes

S3-02

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Stakeholder involvement and public participation is recognized as a necessary part of nuclear waste management. It is possible to see three driving forces behind this development: 1) a deliberative movement with its roots in the 1960s leftish movement (a philosophical argument), 2) the need for public acceptance of proposed solutions (a practical argument), and 3) the need for awareness and clarity before crucial decisions are made (a practical and democratic argument). Depending on which driving force dominates an initiative for public participation, the initiator is likely to prefer certain public participation processes before others. The initiator has to select among a large number of processes already having been implemented or design a new one tailored for the specific situation. The process choice is crucial for which stakeholders are able to join and which goals can be reached.

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

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

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

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

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

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# Installation of a Radioactive Waste Disposal Facility: The Necessity of Building Up Durable Links Between the General Public and Radioactive Waste – Feedback from Experience in France

S3-03

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

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

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

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

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

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

# The Swiss Approach to Finding Compromises in Nuclear Waste Governance

S3-04

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In Switzerland, as well as in many other countries, which are planning to install underground repositories for their high-level nuclear wastes, social conflicts have arisen in the process of searching and exploring a suitable site. In Switzerland, a stale-mate in decision-making occurred when an application for underground exploration for a repository site for low- and intermediate level waste was rejected in a cantonal referendum. As a result, the Swiss parliament decided to abolish the cantonal veto right and instead install an official step-wise site selection procedure with an optional national referendum at the very end of the selection process. To this end, they choose and adapted a well established planning instrument ("sectoral plan").

In this, a strong focus was put on issues of transparency and public participation. Ethical considerations are significant factors in this as the civility of decision-making is an important criterion. Public participation in the sense of "having a say" is though limited to aboveground facilities; in all other questions participation is limited to consultation. Consultation takes place in several forums designed for this purpose, especially in the so-called regional conferences, in which representatives from the interested public and stakeholders are members. The tasks those regional conferences should fulfil are fixed in the sectoral plan.

In setting up this approach, the Swiss authorities took up ideas developed by the German Committee for the Development of a Site Selection Procedure (AkEnd 2002) and international organisations, such as the Nuclear Energy Agency`s Forum on Stakeholder Confidence (e.g. NEA 2003).

Involving the public in decision-making requires changes in the "design" of decision-making processes. Institutions need to learn to coordinate such an involvement of stakeholders, which also means being open to their questions and demands. This is not an easy task as different stakeholders have very different ideas and expectations regarding their involvement. Compromises from all sides are necessary in order to make such arrangements work.

One central compromise in the Swiss case was to abandon the cantonal referendum but at the same time involve the public already in the conception of the sectoral plan through focus group sessions and public consultations.

In those consultations, the Swiss government profited from the strong Swiss consensus democratic tradition. They went beyond the usual consultation procedures by talking not only to established collective actors, but also to the interested public. It showed that, even though time consuming, this approach was insofar a success as all stakeholders agreed to take part in the new site selection procedure. A working compromise had been found.

During implementation it showed that this compromise was not fixed, but that corrections were necessary in order to keep all stakeholders "on board". Tensions still occurred and cannot all be solved.

In this lecture, the Swiss approach to repository siting will be analysed based on governance theories and discussed regarding lessons to be learned for the German case.

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# Same, Same but Different: A Comparative Perspective on Participation and Acceptance in Siting Procedures in France, Sweden and Finland

S3-05

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Deep Geologic Disposal (DGD) has been indicated by a large majority of scientific and technical experts — and endorsed by national governments — as the most adequate way of disposing highly radioactive waste (HLW). It has been claimed that this is the only available option to ensure long-term security and not to burden future generations. However, the paradigm of DGD as the sole solution started getting eroded and some concessions were made also in supporter circles. So the ENEF Working Group "Risks" (2009:3) claimed that "[..] it is nevertheless recognized that there are diverging views in some groups and that there are remaining concerns in the public about geological repositories". Although the advocates of permanent closure of wastes in DGD continue to be numerous, some countries are following other options. Reversibility and Retrievability (R&R) in geological disposal are now seen as a "prudent approach" (OECD/NEA 2012:3). Nowadays, the R&R option is included in several national waste disposal concepts as a possible way to adjust to progress in science and technology and to respond to societal pressures.

People, regardless of their views of nuclear power, realise that radioactive waste needs to

e.g. on siting criteria, is no longer confined within the scientific and techno-political sphere and has involved the relevant stakeholders, including the civil society. However, nuclear waste governance varies from country to country.

The presentation analyses the governance of spent fuel and radioactive waste governance and compares national approaches in Finland, Sweden (both in crystalline rocks) and France (in clay). These three Euratom Member States are forerunners in the siting for DGD. The procedures in these countries differ as they were based on voluntary schemes (Sweden), decisions with strong local community support (Finland) and top-down decision making (France). Sweden and Finland show extraordinary positive attitudes towards nuclear energy (Eurobarometer 2010). In France – where less than 50 % of the population supports nuclear energy – the final decision on site selection is subject to the outcome of a public debate, whereby the opponents have decided to boycott it.

The presentation focuses on participation, whether different modes of governance with diverse requirements concerning transparency and communication have been at work. Moving from the fact that variance exists with regard to the relevant national institutional, legal, industrial and energy frameworks we take evidence from national case studies (Mez et al. 2014) and look for different and common patterns.

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# Social Dimensions of Nuclear Waste Disposal

S4-01

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Nuclear waste disposal is a two-faceted challenge: a scientific and technological endeavour, on the one hand, and confronted with social dimensions, on the other. In this paper I will sketch the respective social dimensions and will give a plea for interdisciplinary research approaches. Relevant social dimensions of nuclear waste disposal are:

(1) *Safety standards*: Frequently, determining safety standards is regarded an issue of natural and engineering sciences. However, in spite of the fact that these disciplines are needed in this field they cannot cover all questions involved. For example, the question "how safe is safe enough" and determining criteria for answering it necessarily include ethical and social issues and are in need of a political legitimisation.

(2) *Disposal 'philosophy'*: Decision-making between different 'philosophies' and approaches how to dispose nuclear waste (e.g. in a deep geological formation, close to the Earth's surface, or in long-term but still interim technical facilities) is not possible on the basis of technical criteria alone but rather needs ethical and social criteria. This is because the decision on the disposal 'philosophy' also will touch justice issues, the distribution of risks among people living today and future generations, and also the type of risks involved.

(3) *Siting*: The determination of the site where to establish the waste disposal is probably that field with the highest consensus that the social dimension is not only involved but even dominant. In many countries the etsbalished procedures of determining national sites for nuclear waste disposal have failed so far. Protest and rejection based on different reasons are not exceptional but seem to be the rule (except few cases). Defining and implementing a process of determining an appropriate site with respect to safety and other criteria in a civilian way remains a challenge in many countries. Participatory and inclusive approaches observing ethical issues of fairness, equity, and transparency are looked for. However the criteria and approaches from natural and engineering sciences must not be forgotten in these processes.

(4) *Operation*: The operation phase of a nuclear waste facility needs technical knowledge and competence but also appropriate capabilities to deal with its social environment in a civilian and open way. Trust is crucial not only for determining the site but also in operating the respective facility. This is even more ambitious in the presence of monitoring activities.

Taken altogether we conclude that the issues of justice (just distribution of burdens and risks), of trust (e.g. by transparency) and of fairness (concerning the opportunities to intervene into decision-making procedures) are of highest importance. These are not merely additional to natural and engineering sciences' inputs but, rather, technical and ethical criteria and issues often are unseparably intertwined (e.g. in determining safety standards). This situation makes close interdisciplinary cooperation necessary.

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# Inclusive Expertise in a Site-Selection Process – Experience, Some Lessons and Reflections Beyond Boundaries

S4-02

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The nuclear community recognises that the long-term safety of repositories "is not … a rigorous proof of safety ... but rather a convincing set of arguments" (NEA 1999). It has, however, been difficult to "live" its socio-technical nature (NEA 2013). Albeit the waste problem is driven by technology and, indeed, a technological constraint, in the end, it has to be solved by society.

Building upon the defence-in-depth principle, the concept of integral, technical and societal, robustness was developed (Flüeler 2002/06). A system is "socially robust" if most arguments, evidence, social alignments, interests, and cultural values lead to a consistent option (Rip 1987). Based on international experience (Flüeler 2002/05/06/14), we propose a 3-step approach for a site-selection procedure:

**Step 1: Discuss – comprehensive societal discourse.** First, discuss the issue from all conceivable angles. "Involvement of stakeholders" cannot mean to call for as many *individuals* but for as many *perspectives* as possible to systematically incorporate all relevant facets in the multi-dimensional discourse. The aim is to lay all pertinent aspects (values, norms, context, evidence …) open, to successfully "close" issues, and proceed to the following phase.

This process has to be overseen, *e.g.,* by a widely credible and trustworthy body. In 2002, Flüeler suggested a "National Council for the Safe [and Secure] Governance of Radioactive Waste" as the guardian of the process, the Swiss expert committee EKRA foresaw a "Disposal Council" – an idea in the meanwhile somewhat embraced by the Federal authorities (BFE 2008). It should be pluralistically composed, independent of the industry yet knowledgeable and not driven by daily politics.

**Step 2: Decide – "common ground" in goals and stepwise strategy.** The goals have to be prioritised so as to adopt the stakeholders' respective responsibilities. Conflicting goals exist. The concept of "sustainability", a complex goal indeed, encompasses protection of, and leeway for, future generations. Transferred to nuclear waste, both passive safety and "active" control or surveillance need due care and attention in parallel (see Swiss concept to integrate monitoring into final disposal, *ib.*).

No consensus will be reached "at heart", in the stakeholders' core beliefs. Society must agree, though, on three levels: 1. Problem recognition (waste exists, problem to be tackled, eventually "solved", at least set on track to be solved); 2. Main goal consensus (degree of protection and intervention); 3. Procedural strategy ("rules of the game").

**Step 3: Implement – start programme and prepare long term knowledge basis.** In view of a successful transfer of knowledge, it is vital to explore contextual issues and tacit/implicit knowledge – they determine the degree of societal understanding of the eventual disposition system. Unless the rationale of conceptual reasoning is appropriately handed over to next generations, the entire undertaking is bound to failure.

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

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

S4-03

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ENTRIA ("Disposal Options for Radioactive Residues: Interdisciplinary Analyses and Development of Evaluation Principles") is a joint research project funded by the German Federal Ministry of Education and Research BMBF, grants 02S9082A-E) and carried out by 12 departments from German universities and major research institutions and one partner from Switzerland. The scientists participating in ENTRIA represent natural sciences, civil engineering, philosophy, law, and social sciences. Recognising that all these disciplines need to interact when radioactive waste management is concerned, the project is aiming to investigate, and to develop evaluation principles for, three waste management options: Deep disposal without retrievability measures, emplacement in deep formations with monitoring and retrievability measures, and prolonged surface storage.

The scientists involved are well aware of the fact that this is not the first time that such options are evaluated, and many of them have developed their preferences over the years and decades of their professional life. Nevertheless, screening literature, documents and media showed that the conceptual arguments or their communication to various audiences and stakeholders were not as robust in arguing that stakeholders and media are convinced. Therefore, there is apparently a need to systematically compile, structure, evaluate, and communicate basic features, advantages and drawbacks of these options. This will not spare society the decision(s) to be made, but will hopefully make such decisions better informed. Apparently, the relevant articles of the recently adopted German Site Selection Act (StandAG) are based on similar considerations.

In order to facilitate interdisciplinary communication and co-operation, the project is organised in three so-called vertical projects, each addressing one of the management options and all mainly treated by natural scientists and civil engineers. In addition, overarching aspects such as "Synthesis, co-ordination and communication", "Technology Assessment and Governance", "Ethical and moral substantiation, legal prerequisites, and implications", and "Interdisciplinary Risk Research" are addressed by interdisciplinary so-called transversal projects.

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

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

Furthermore, the project is increasingly working on joint interdisciplinary products such as journal articles, PhD theses, and events for the interested public held by interdisciplinary teams. As a very first step towards an interdisciplinary synthesis, the senior scientists from all teams published a memorandum naming the specific challenges and target conflicts of importance when deciding about management options. All these products required dedicated working approaches which will be addressed in the presentation.

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

S4-04

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

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

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

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

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

**Acknowledgements**

The research leading to these results has received funding from the European Atomic Energy Community's 7th Framework Programme (FP7/2007-2011) under grant agreement n° 269909.

# Nuclear waste and hazardous waste in the public perception

S4-05

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

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

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

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

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

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

P1-01

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In summer 2010 the German Federal Government launched a preliminary safety analysis to assess whether the salt dome at Gorleben is suitable to host all heat-generating radioactive waste generated by German NPPs. On basis of the “Safety Requirements Governing the Final Disposal of Heat-Generating Radioactive Waste as at 30 September 2010” (BMU 2010) a repository concept was developed which was optimized to a certain extent during project evolution. The repository design was tailored to the specific geologic environment at the Gorleben site and served as the main technical basis for the site-specific preliminary safety analysis (VSG).

For comparison, two different emplacement strategies were considered. One strategy considered that all waste packages will be disposed of in horizontal drifts of a repository mine, while the other strategy considered that all waste packages will be disposed of in deep vertical boreholes drilled into the floor of the repsotory mine.

The fundamentals, the design approach, and the resulting repository designs for two different emplacement strategies (GRS 2011) will be presented in this paper. The designs took into account an updated set of fundamental data regarding the amounts and types of expected heat-generating waste and the documented results of the exploration of the Gorleben salt dome. The designs aimed to transfer the new “Safety Concept and Safety Assessment Concept”, i.e. the methodology how to demonstrate operational and long-term safety, into technical solutions for repository components, systems, and processes.

The respective final technical design results (GRS 2012) for both options (emplacement in drifts/emplacement in vertical boreholes), will be presented. These include first the selection and description of waste-specific (SF/ reprocessing waste) waste packages for both strategies. Based on the expected waste inventory, 3D-thermal calculations were carried out to determine canister, drift, and borehole distances and to provide suitable design parameters for the layout of emplacement drifts and fields and the entire repository. The design includes the layout of infrastructure areas and drifts for the transportation of waste packages and excavated rock salt material as well as the measures for backfilling. Transport and emplacement processes and techniques for both strategies were developed and described. Several of them have already been demonstrated in full-scale at surface test facilities.

According to the safety requirements designs have to meet retrievability requirements for a German HLW repository. The consequences of this requirement and suitable methods for the design, licensing, construction, and operation of the repository were analysed and described. With regard to long-term safety, assessments of how to exclude criticality of disposed SF packages during the lifetime of the repository were performed as well.

With regard to possibilities for design optimizations, the ventilation system for repository operation was investigated in detail. Preliminary calculation results showed that it is possible to provide sufficient amounts of fresh air in all mine areas and to transfer all exhaust air to the surface. In addition to this, the structures of drifts and emplacement fields were reconsidered in order to find possibilities to keep excavation to a minimum.

The repository technical design work showed that the Gorleben salt dome provides sufficient space on a single level to host a repository for all the heat-generating radioactve waste and spent fuel arising in Germany.

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# A Thermodynamic Reference Database for Nuclear Waste Disposal

P1-02

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

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

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

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

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

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

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

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

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

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# Waste Classification and Choice of Geological Repository Concept: Ukrainian Case

P1-03

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Currently, 15 WWER reactors are being operated at four Ukrainian NPPs. Four RBMK reactors were operated at Chernobyl NPP, one of which was destroyed in the 1986 disaster.

It is expected that more than 15,000 tHM of SNF and almost 200,000 $m^3$ of waste will be generated by the existing NPPs. The accident at the Chernobyl NPP contributed by app. 3.5 million $m^3$. This 'accidental' waste contains high concentration of alpha-emitting nuclides.

It has been realized that the current Ukrainian waste classification is not suited for the disposal of all existing waste. It allows the use of only two types of repositories: near-surface for short-lived waste and deep geological repositories for long-lived waste.

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

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

According to this scheme, all Ukrainian waste can be divided into seven classes (DBE, 2012):

- Exempt waste;
- NORM-waste (disposal not governed by nuclear legislation);
- VLLW: Very low level waste (landfill type repository);
- LLW: Low-level waste (near-surface repository);
- ILW: Intermediate-level waste (geological repository at intermediate depth);
- HLW: High-level waste (deep geological repository);
- DSS: Disused sealed sources (disposal as VLLW, LLW or ILW).

It was suggested not to create a special class of 'accidental' waste. This waste should be disposed of only inside the Chernobyl exclusion zone (CEZ), using special, less restrictive waste acceptance criteria.

The implementation of the new classification scheme leads to a reduction of disposal costs.

A decision to dispose of 'accidental' waste as VLLW or LLW in the CEZ and about a co-disposal of ILW and HLW in a deep geological repository will allow a 10-fold decrease in total costs. If ILW and HLW will be disposed of separately in different repositories, the costs will decrease by 40 times (DBE, 2012).

This economical effect makes the use of both concepts of geological repositories appealing in Ukraine:

- repository at the intermediate depth for ILW disposal, like SFR (Sweden);
- deep geological repository of borehole type for the disposal of SNF and vitrified HLW, like VDH-concept (Sweden).

Other non-economic benefits of the deep borehole disposal concept for Ukraine were discussed in (Shestopalov, 2006).

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# Geological and Geotechnical Limitations of Radioactive Waste Retrievability in Geologic Disposals

P1-04

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

In the present research the use of five host rocks types, flatly and steeply-bedded salt rocks, clay, claystone and crystalline rocks, for deep emplacement of HAW is addressed and their performance due the retrievability measures are compared based on their characteristic hydrologic and mechanical behavior. The favorable features for each host rock and their response to keeping the access drifts and emplacement open are compared and evaluated.

The mechanical behavior of both types of rock salt is quite similar. Rock salt in natural storage conditions has a very low permeability and the construction-induced damage is reduced by self-healing. Rock salt is capable to creep, therefore the stress deviator has to be reduced. If not, the EDZ will increase and the geological barrier will be degraded. The cavities in salt rock do not need a pit supporting system.

Clay has a very low self-support therefore a heavy support system is required particularly when the disposal vaults and access tunnels are keeping open for a long period of time. Due to stagnating pore-water the sediment is sensitive against desiccation.

Claystone, due its lithified structure, has a low self-support too. However their required pit support system is lesser extent as in clay. As a rock it has a latent joint network which can result in flow paths for ground water.

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

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

P2-01

Achim Brunnengräber & Daniel Häfner

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

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

P2-02

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

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

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

# Independent monitoring of a release from the Waste isolation Pilot Plant in New Mexico, USA: Results and Purpose

P3-01

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

**Acknowledgements**

This research is funded by U.S. Department of Energy (US-DOE) through a grant DE-FG29-91AL7416 from the Carlsbad Field Office (CBFO).

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

P4-01

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In nature, $SiO_2$-minerals, in particular quartz, are the most important weathering residues. Moreover, silicate phases are widespread secondary formations. They form as a result of a temperature or pressure decrease as well as an increase in salt content and/or a decrease in pH-value. The latter influences dominate in saline environments. These facts and the low solubility of silicates suggest the use of $SiO_2$-containing materials for sealing measures, e.g. for the construction of barriers in repositories that are located in salt formations. The grouting of fractures in rocks and building constructions or the filling of pore spaces in salt backfill require additional favorable physical properties, e.g. a negligible small particle size and a low viscosity. All these chemical and physical requirements are met by sodium silicate solutions.

Sodium silicate solutions, generally called water glass, are alkaline solutions of sodium oxide and silicon dioxide. Major products of the reaction with pure halite or sodium chloride solution are amorphous silicate and halite. Salt formations contain a variety of Mg-containing minerals and solutions. Experiments and observations made during in-situ grouting, illustrate that these $Mg^{2+}$ ions, together with water, are fixed in the structure of magnesium oxichlorides and Mg-silicates (Engelhardt & von Borstel 2014). All reaction products have a low solubility. For example, laboratory experiments show that highly concentrated $MgCl_2$-solutions (cf. Q-/IP21- or R-/IP19-solution) have Si concentrations of < 1 µg/g which is lower than in low-saline waters (< 5 µg/g), which are controlled by the solubility of macro-crystalline quartz. The results document the long-term stability of the fracture or pore fillings. Like $Mg^{2+}$ or $Ca^{2+}$, experiments demonstrated that silicate solutions are able to fix $Co^{2+}$, $Ni^{2+}$, $Sr^{2+}$, $Ba^{2+}$, $[UO_2]^{2+}$, $Cs^+$, dissolved in NaCl- and NaCl-$MgCl_2$-solutions, in insoluble minerals.

One important aspect, which is fundamental to rate the grouting process quantitatively, is that sodium silicate solutions can be considered as Newtonian fluids. Their reactions with salts and salt solutions induce a rapid gelation, which results in an exponential increase in viscosity and the development of a yield strength. Most notable is the dependence of the reaction rate on the dissolution rate of the salt mineral, the temperature, and the composition of the water glass. Long-term changes in the mineral formations comprise a structural ordering of the amorphous silicates. This process is coupled with a release of $Na^+$ cations, which induces halite crystallization.

With regard to the grouting of fractures and pores, the size of the colloidal particles in sodium silicate solutions can be neglected. This fact opens the possibility to simultaneously fill and seal fractures and to measure the permeability of excavation damaged zones. In order to determine the permeability, a stepwise increase of the pressure and the performance of pressure drop measurements are necessary. The measured pressures and flow rates were evaluated in the same way as the procedures that are used in instationary permeability measurements. As a consequence of the high grouting pressures, which were eventually adjusted, the procedure also fulfills the task of a local test loading.

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P4-02

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

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For repositories rock salt based materials are often used for sealing systems. In this case the most favourable material is crushed salt. For shaft sealing elements crushed salt may be used in backfill columns. For instance within the future shaft closing concept of WIPP-Site in Carlsbad NM [1, 4] highly compacted crushed salt columns are envisaged. But till now shaft sealing elements with very low permeability after compaction were still not yet realized, because in situ compaction of crushed salt is an extensive operation.

To limit the settlement and the brine permeability of the backfill column crushed salt must be in situ compacted to a maximum of packing density. The packing density of technical compacted crushed salt depends on the following parameters:

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

According to GLAUBACH el al. [3] and other authors grain size distribution of the crushed salt has an important effect on maximum packing density. As a result the well-known FULLER distribution with a FULLER-exponent of n=0.5 is used to optimize grain size distributions of crushed salt as a practical approach in engineering. As a reference, the well-known REPOPERM [5] material was used to evaluate the effort of the optimization of the grain size distribution. The calculated total porosities of the optimized mixtures are significant lower than the achieved porosities of the REPOPERM material.

The approach of using swellable clays for doping crushed salt mixtures according to BUTCHER [2] and STÜRENBERG [6, 7] was successfully adopted.

Based of the former work of GLAUBACH et al. [3] an empirical optimization process is described to optimize the water and clay content. As a result the water and clay content could be reduced to a range, which is nearly the half of the suggested content according to BUTCHER [2] and STÜRENBERG [6, 7]. Furthermore, a relation between the spec. compaction energy and the optimum water / clay content exists and can be used to adopt the mixture design to the in situ available compaction procedure.

The lab-scale compaction results are not transferable without considering further demands to large scale practice.

Typically underground compaction procedures are generally comparable to the common earthwork compaction. Therefore, several field tests using earthwork machinery for compacting optimized salt/clay-mixtures were realized. Results and limits will be discussed and further demands revealed within the lecture and paper.

The described results are part of the ELSA-Project. This project is kindly supported to 100 % by the Federal Ministry of Economics and Technology (BMWi) under the project number 02E11193A. The Project management isunder assistance of the Project Management Agency Karlsruhe (PTKA).

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

P4-03

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For repositories in rock salt, containment of the radioactive waste is provided by the geologic barrier salt in combination with geotechnical barriers. Therefore, demonstrating the integrity of drift and shaft seals is an indispensable part of the long-term safety case.

In this study, probabilistic methods are applied to assess two crucial parameters: The fictitious abutment length for a shaft seal and the effective permeability of a drift seal (dam), i.e. the integral entity for the whole structure including contact zone and damaged salt zone. For the seal permeability, the question arises how to derive it based on permeability measurements with a limited number of samples due to cost restrictions. Furthermore, it is of interest which conclusions can be derived regarding the minimum length of drift seals if the failure probability should be smaller than e.g. $10^{-4}$.

It is possible to consider the small-scale data as a sample for which its probability distribution is representative for the large-scale behaviour. In other words, the whole structure is considered as a homogeneous entity in which the permeability complies with the distribution derived from the small scale measurements. According to the distributions of the material parameters quantiles can be estimated as characteristic values and inserted into the limit state function. However, using just the $10^{-4}$ fractile of the measurements leads to demands concerning the length of a drift seal which are neither plausible nor viable.

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

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

The work presented was funded by the German Ministry for Economic Affairs and Energy (BMWi, grant 02 E 10729) and carried out in the frame of a sub-contract awarded by DBE Technology.

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# Thermal Loading of Bentonite: Impact on Hydromechanics and Permeability

P4-04

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

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

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

# Radionuclide Solubility Control by Solid Solutions

P5-01

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

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

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

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

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P5-02

# Barrier Function of a Corroding Iron Based Container

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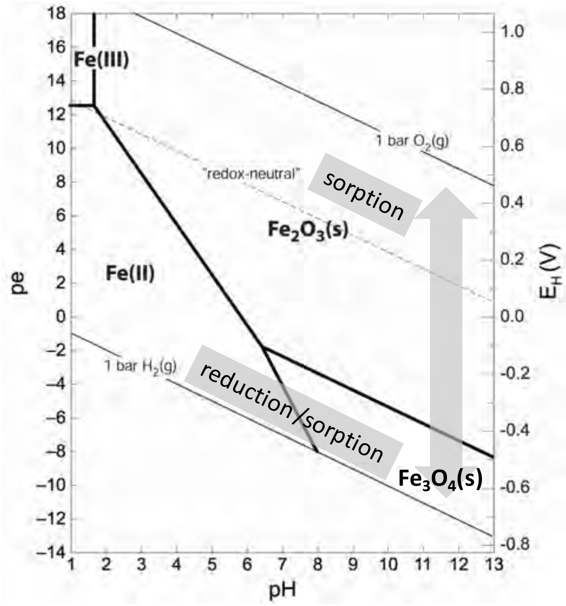
Thick-walled cast-iron containers encapsulate highly radioactive waste in a deep geological repository and provide mechanical stabilization against lithostatic pressure and shear forces. Intrusion of water and corrosion, however, limits container integrity to a certain time period of several hundreds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

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

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




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

- surface induced reduction of actinides (e.g. U, Np, Pu) and fission products (e.g. Tc, Se) by which they transform to reduced and poorly mobile species

- surface complexation of redox inactive radionuclides (e.g. Am) at reactive surfaces by which migration is retarded

- formation of secondary solid phases (e.g. $FeSe_x$)

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

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

P5-03

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

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

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

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

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

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

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

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

P5-04

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

For the requirements of waste conditioning, special cement mixtures have been developed. These include special mixtures for the solidification of evaporator concentrates, borate binding additives, and for spilling solid wastes. In recent years, low-pH cements were strongly discussed especially for repository applications, e.g. (Céline CAU DIT COUMES 2008; García-Siñeriz, et al. 2008). Examples of relevant systems are Calcium Silicate Cements (ordinary Portland cement (OPC) based) or Calcium Aluminates Cements (CAC). Low-pH pore solutions are achieved by reduction of the portlandite content by partial substitution of OPC by mineral admixtures with high silica content. The blends follow the pozzolanic reaction[1] consuming $Ca(OH)_2$. Potential admixtures are silica fume (SF) and fly ashes (FA). In these mixtures, super plasticizers are required, consisting of polycarboxilate or naphthalene formaldehyde as well as various accelerating admixtures (García-Siñeriz, et al. 2008).

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

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

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

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

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

# Iron Corrosion in Concentrated Saline Solutions at T and P Conditions in High-Level Radioactive Waste Rock Repositories: A Thermodynamic Study

P5-05

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Metallic corrosion of steel-based spent fuel canisters is one of the processes that will inevitably occur in a repository of radioactive waste when aqueous solutions come into contact with the waste. The corrosion process can only be understood taking thermodynamic relations, reaction kinetics and transport processes into account. Thus, thermodynamic aspects of Fe(II) and Fe(III) in concentrated solutions of the hexary oceanic salt system constitute an essential frame for modelling and predicting corrosion processes in repositories in salt rock formations and argillaceous formations. Water from different sources and traces of oxygen trapped in the backfill salt initiate an aerobic corrosion phase which further evolves under reducing conditions where corrosion proceeds with hydrogen evolution (Wang et al 2001). The stability of corrosion products generated upon contact of steel containers with saline solutions may have a large impact on the transport and migration of radioactive contaminants that have been mobilized from the waste matrix. For instance, zero-valent iron was tested for the removal of uranium by reductive precipitation of uranyl to less soluble U(IV)-compounds and the adsorption of uranyl cations on iron corrosion products (Farrel et al 1999). The removing rate by reduction of U(VI) to slightly soluble U(IV) species depends greatly on the activity of solution species and the nature of the corrosion products .

The corrosion rate can be controlled by the nature of precipitation products, which may be constituted by oxy-chlorides, sulfates, sulfides and carbonates, depending on the composition, temperature and pressure of the salinary surrounding media and the availability of water (by its transport rate into the direct proximity of the waste canister). This work aims the construction of a self-consistent data base for prediction of the thermodynamic stability of the canister. The first step consisted in the determination of the activity of solution species at high temperatures and high ionic strengths, which was modeled by applying the Pitzer formalism (Pitzer 1991). A set of temperature dependent binary and ternary Pitzer parameters for the system [Fe(II),Fe(III)]-K-Na-Cl-SO4-H2O was generated from water activity and solubility data acquired from an extensive literature review and water activity data obtained from own isopiestic experiments. Interaction Pitzer parameters and solubility constants of minerals belonging to the hexary oceanic salt system were taken from the THEREDA reference data base (*http://thereda.de*).

The solubility of iron oxides, sulfides and carbonates at high temperatures were obtained from an extended revision of literature data and using the Helgeson-Kirkham-Flowers (HKF) model for predicting thermodynamic properties of ionic species. Using the complete set of data, stability Pourbaix-type diagrams at temperatures from 0°C to 90°C were constructed by using the program The Geochemist´s Workbench with a self-generated parameter file in a wide range of composition and temperature of the corroding media.

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# Speciation of Neptunium after Diffusion in Opalinus Clay

P5-06

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

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

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

This work was financed by the Federal Ministry of Economics and Energy (BMWi, contract no. 02E10166, 02E10981) and TALISMAN (contract no. TALI-C01-15).

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# Evolution of the Bentonite Barrier Under Glacial Meltwater Intrusion Conditions

P5-07

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

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

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

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

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

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

P5-08

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

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

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

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

**Acknowledgements**

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

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# Study of the Influence of Hydrogeological Conditions in the Upper Aquifer System on Radionuclide Migration from a Deep Geological Repository Using a 2D Groundwater Flow and Transport Model

P5-09

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In course of implementation of an IAEA-coordinated Research Project (IAEA, 2013) and of a National Academy of Science of Ukraine Project #1050 (Shestopalov, 2011), a case groundwater flow and transport modeling was performed to characterize the influence of hydrogeological conditions in upper aquifers of a storey water exchange system on the forecast of possible radionuclide migration from a deep geological repository (DGR) as applied to geological formations of Chernobyl Exclusion Zone in Ukraine.

A 2D (section) groundwater flow and transport model of a prospective site within the study area is developed using the PMWin Groundwater modelling system (Ciang, & Kinzelbach, 2001). The transport model accounts for advection, dispersion, and equilibrium sorption processes.

The DGR site is represented by a watershed with definite surface infiltration recharge and groundwater discharge into a river at distance 5 km and a shallow well at distance 3 km from the surface point above the DGR location. The model domain corresponds to 2D vertical section (depth 1.5 km, length 5 km) around the DGR location.

In the vertical direction the model section includes the upper active water exchange zone in the sedimentary water-bearing deposits to depth 300-400 m and the underlying low groundwater exchange zone in the granitoid formations with the DGR at depth 800-1000 m. An inclined fracture with relatively higher conductivity is considered ascending from the DGR upward to the well location.

The initial data included the topology (depths) of main geological strata, assessed hydraulic conductivity and porosity of deposits, groundwater heads observed in aquifers of active water exchange zone (sedimentary cover and upper zone of fractured granites), and assessed transport (sorption) parameters.

The main simulation task was study of the influence of model parameter variations on the contaminant travel time from the DGR and relative contaminant concentrations reached at the discharge locations (well and river) in course of implementation of the different simulation scenarios.

An important conclusion is made that the higher drainage activity in the well and river and higher hydraulic conductivity in the upper aquifers may provide a “safer” situation in the deep DGR location. The possible reason is that in such conditions the most part of the downward groundwater flow concentrates in the upper subsurface aquifers of the watershed. Hence, the hydrogeological conditions in the top water-bearing layers have a key influence on the water exchange intensity in the deep zone of DGR location.

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

P5-10

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

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

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

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

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

P5-11

# The Correlation Between Selenium Adsorption and the Mineral And Chemical Composition of Taiwan Local Granite Samples

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Selenium-79 (Se-79) is a radioactive isotope of selenium, which is a highly mobile nuclides since Se-79 exists as an anion species in the groundwater. For an anionic species, the transport of Se-79 could be controlled by metal oxides in minerals such as goethite and hematite (Jan et al., 2008). It was verified that the transport of selenium in shallow surface environments depends on the amount (concentrations?) of these metal oxides in the soils and sediments.

However, when dealing with deep geological repository, the transport of Se-79 becomes less predictable because of the lower content of metal oxide in the host rock such as granite. In order to conduct a reliable performance assessment of repository, it is very important to establish the correlation between selenium adsorption and the properties of potential host rock, in this study, the mineral and chemical compositions of Taiwan local granite. From this point of view, selenium adsorption experiments (Se(IV)) were conducted with 54 different Taiwan local granite samples collected from depths ranging from 100 ~ 400 meters below the surface. The synthesized groundwater adopted in this study is based on the chemical composition of in situ groundwater (Tsai et al., 2008).These granite samples represent a variety of deep geological environments, including intact rock, groundwater intruded zones, and some weathered samples. Based upon our preliminary results no redox reaction was observed and several conclusions could be drawn from the point of view of linear regression.

First, the correlation coefficients between the Kd values and the mineral and chemical compositions are very low (R-square values are often < 0.2). This points out the complexity of these geological samples and strongly suggests more efforts should be invested to acquire more relevant information.

Second, the correlation between the selenium Kd values and the content of iron oxide (R-square 0.110) is much higher than that between the CEC of these granite samples (R-square 0.001). This clearly indicates that the minerals that are able to underlie cation exchange reaction would not be the minerals regulating the transport of selenium nuclides.

Third, a relative high correlation coefficient between the variation of pH before and after selenium adsorption and iron oxide content was observed. The release and uptake of proton reflected as the variation in pH implies that surface complexation reaction is much likely accounting for selenium adsorption.

Interestingly, the linear correlation between Se(IV) adsorption Kd values and geochemical properties are greatly improved when we further extracted those data collected from the fracture zones out of others. Importantly, data belonging in this group are often the outliers presented in the figures. This means these minor minerals presenting in the fracture zones are the dominant ones controlling the transport of nuclides.

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# Self-Sealing of Excavation Induced Fractures in Clay Host Rock

P5-12

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

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

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

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

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

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

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

# Critical Evaluation of German Regulatory Specifications for Calculating Radiological Exposure

P5-13

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

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

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

# An Introduction to the Transport Properties Research Laboratory at The British Geological Survey and its 50+ Years Experience in Geological Disposal Research

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

Three key areas explored are:

(i) baseline characterisation of hydromechanical properties,

(ii) influence of stress path and stress history on transport properties and

(iii) transmissivity of fractures, faults and discontinuities (e.g., wellbore interfaces).

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

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

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

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

P5-15

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

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

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

P5-16

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

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# Microorganisms in potential host rocks for geological disposal of nuclear waste and their interactions with radionuclides

P5-17

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

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

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

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

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

**Acknowledgements**

The authors thank the BMWi for financial support (contract no.: 02E10618 and 02E10971), the BGR for providing the clay samples and Velina Bachvarova, Sonja Selenska-Pobell and Gert Bernhard.

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

P5-18

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

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

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

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

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

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

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

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

**Acknowledgements**

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

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

S6-01

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

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

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

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

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

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

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S6-02

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

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

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

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

The proof of function is performed iteratively consisting of two main parts – the so-called preliminary dimensioning (GRS 2012a) relying on a preliminary design of the closure system and the detailed proof of function (GRS 2012b), which includes minor design modifications. The preliminary design and dimensioning serves as a basis for FEP selection and scenario analysis as well as for a preliminary long-term safety assessment to rate whether the closure system will probably will be able to guarantee safe containment of the waste. In a second step brine and gas pressure build up rates resulting from long-term safety assessment in turn are used to prove proper function of the closure system according to technical regulations in force.

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

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

S6-03

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

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

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

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

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

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

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

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

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# Versi: A Method for the Quantitative Comparison of Repository Systems S7-01

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

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

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

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

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

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

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

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The presented work has been initiated and financed by Bundesamt für Strahlenschutz BfS, Deutschland and is published in:

Resele, G. *et al.*, (2010) WS2044 – VerSi – Methode für den quantitativen Vergleich von Endlagersystemen, BfS, Deutschland.

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

S7-02

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Within the framework of the R&D project "AnSichT", DBE TECHNOLOGY, BGR and GRS are developing a methodology to demonstrate the safety of a HLW repository in claystone in Germany. The methodological approach is based on a holistic concept which links the legal and geologic boundary conditions, the disposal and closure concept, the demonstration of barrier integrity, and the long-term analysis of the repository evolution.

For the geological boundary conditions the criteria of the safety requirements (BMU 2010) and the results of an evaluation of German argillaceous rock formations (Hoth et al. 2007) are applied (Reinhold & Sönnke 2012). To demonstrate the safety the description of the geological situation and appropriate models are required. Considering the diverse geologic conditions in Germany two geological models were developed. The bases for the geological models are borehole data, data from seismics or in-situ data from underground research laboratories (e.g. URL Mont Terri). The developed geological model for Northern Germany represents a typical situation of the eastern part of the Lower Saxony Basin (Reinhold et al. 2013). The Barremium & Hauterivium formation in the Lower Cretaceous are considered as host rock formations. The generic model for Southern Germany is situated in the north alpine Molasse Basin with the Opalinus Clay in the Middle Jurassic as the host rock formation.

Representative parameters for the hydraulic and petrophysical rock properties are assigned to the geological units of the models (Jahn & Sönnke 2013). The specifications include average values and bandwiths. Out of the bandwidth certain parameters were selected for numerical simulations, which are used for the demonstration of the barrier integrity (Nowak & Maßmann 2013).

Further geologic boundary conditions for the safety demonstration are geoscientific long-term predictions of the evolution of the geosphere during the demonstration period of 1 million years. According to Hoth et al. (2007) the considered areas comprise parts of the North German Basin and a smaller part of Molasse Basin in Southern Germany. Due to the different geological developments of the northern and the southern areas, different implications have to be considered.

The description of the geological situation and the development of suitable models, the selection of representative parameters and the geoscientific long-term predictions provide a fundament for the system analysis, which is a main part of the safety assessment.

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

S7-03

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In Finland, Posiva Oy has submitted a construction licence application for a KBS-3 type spent nuclear fuel repository to be constructed at the Olkiluoto site in South-Western Finland. The crystalline bedrock of Olkiluoto consists of supracrustal high-grade metamorphic rocks, mainly various types of gneisses, and secondarily of igneous rocks such as pegmatitic granites and diabase dykes.

The Finnish regulations specify several favourable properties that the repository host rock must have in order to fulfil its safety functions as a natural barrier. There are also regulations on classification of the rock structures and properties that may have an impact on the long-term safety and that should be taken into account in the repository construction. To address these requirements Posiva has been developing rock classification systems applicable to the construction of the repository for more than fifteen years now.

During the years, along with the site selection and more detailed studies including also underground studies at the selected site, the rock classification systems have evolved. At the beginning, the identification and classification of large-scale deformation zones and the definition of respect distances between the deposition tunnels and these potentially unstable or transmissive zones guiding the repository layout were key issues. The properties in the rock mass between the deformation zones were at first addressed by considering only the engineering properties of the rock. Later the focus has been on selecting the host rock suitable for the deposition holes and tunnels emphasising the long-term safety aspects.

The Rock Suitability Classification (RSC) system (McEwen et al. 2012) that is currently being used incorporates criteria for hydraulic properties, mechanical stability of the host rock and the chemical properties of groundwater, and has been reinforced by a well-defined classification process. The criteria are derived from target properties, which set the general long-term safety related requirements for the host rock. Measurable or observable host rock parameters are incorporated into the rock suitability criteria with the aim of fulfilling the target properties. The RSC method is carried out at different scales: repository, panel, tunnel and deposition hole scales. These coincide with different stages of repository design and construction, proceeding from the layout design of the whole repository to the implementation design and construction of panels consisting of central tunnels and several deposition tunnels and, finally, deposition holes.

The Rock Suitability Classification system is currently being tested in the rock characterisation facility ONKALO at Olkiluoto, at the actual repository level and conditions representative of the future repository to be constructed as an enlargement of the ONKALO facility. The main outcome of the demonstration activities performed to date indicates that the criteria developed are applicable in practice and that the stepwise research, construction and decision-making protocol can be applied successfully. However, needs for further development of some of the criteria, further streamlining the flow of investigations, design and construction, as well as the associated decision-making and documentation process were also identified.

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# Application of fundamental aquatic chemistry to the Safety Case and the role of thermodynamic reference data bases

S8-01

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

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

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

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

# Assessment of the Long-Term Safety for SFR

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

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

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

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

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

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

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

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

# *In-Situ* Experiments to Investigate Rock Matrix Retention Properties in ONKALO, Olkiluoto, Finland

S8-03

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Spent nuclear fuel from nuclear power plants, owned by TVO (Teollisuuden Voima Oy) and Fortum, is planned to be disposed to a repository at a depth of more than 400 meters in the bedrock of Olkiluoto (Eurajoki, Finland). The repository system of multiple release barriers consists of both manmade and natural barriers. The surrounding rock acts as the last barrier if other barriers fail during passage of the millennia. Therefore, safe disposal of spent nuclear fuel requires information on the radionuclide transport and retention properties within the porous and water-containing rock matrix along the water conducting flow paths.

To this end, various experiments are being performed and planned within ONKALO, the underground rock characterization facility in Olkiluoto, as a part of a project “Rock matrix rEtention PROperties” (REPRO). The research site locates at depth of 420 meters close to the repository site. The aim is to study diffusion of elements and sorption properties of the rock matrix in real in-situ conditions. It is known that due to sorption and matrix diffusion migration of radionuclides is retarded. However, relevance of the retention in repository conditions has remained unclear and impact of retention varies from nuclide to another. In addition, parallel laboratory experiments conducted offer information about difference between in-situ and laboratory conditions to diffusive and transmittive properties of bedrock as well as parameters for analyzing in-situ experiments.

At the moment three types of experiments are being performed or prepared in the REPRO project. In first type of experiment short concentrated pulse of selected radionuclides is injected to a water flow through artificial fracture and their breakthrough is measured from the other end of the fracture (water phase diffusion experiment, WPDE1). The 1 mm thick fracture is formed by 2 meters long packered section of drill hole and a flow guide placed on center of the drill hole. In second type of experiment concentrated mixture of selected radionuclides is injected into a meter long packered section of drill hole and a concentration of the radionuclides are followed in two observation bore holes 10 cm away from the injection bore hole. Third type of experiments are similar to the first one, except these will be performed in gas phase. These experiments are about 10000 times faster to perform than experiments in the water phase.

The WPDE1 experiment using radionuclides was performed, during the spring and summer of 2012. In this experiment a flow rate of 20 µl/min and a mixture of HTO, $^{22}$Na, $^{36}$Cl and $^{125}$I was injected to the flow. Breakthrough of radionuclides were determined as a function of time with an online gamma detector and by taking laboratory samples from the outflow. As a result pore diffusion coefficient of $1\times10^{-12}$ $m^2/s$ was determined for $^{36}$Cl and $7\times10^{-12}$ $m^2/s$ for HTO, $^{22}$Na and $^{125}$I, respectively. Furthermore for $^{22}$Na and $^{125}$I distribution coefficient of $6\times10^{-5}$ $m^3/kg$ was determined. As assumed, sorption behavior was not observed for HTO and $^{36}$Cl. Currently similar experiment using lower flow rate (10 µl/min) and HTO, $^{22}$Na, $^{36}$Cl, $^{133}$Ba and $^{85}$Sr as tracers is running. For experiments under preparation detailed predictive analysis will be presented.

# Studies on Spent Nuclear Fuel Evolution During Storage

S8-04

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

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

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

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

# Corrosion of Spent Fuels from Research and Prototype Reactors Under Conditions Relevant to Geological Disposal

S8-05

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The reference inventory of high-level nuclear wastes designated for geological disposal in Germany as used within the preliminary safety assessment for a geological repository in the Gorleben salt dome ("vorläufige Sicherheitsanalyse Gorleben", vSG) includes various types of spent nuclear fuels from research and prototype reactors, besides LWR spent fuels and vitrified high-level wastes (Peiffer et al., 2011). This paper will discuss the results of and conclusions from corrosion experiments on spent fuels from prototype high-temperature reactors (HTR) and research reactors that were performed under repository relevant conditions and provided the basis for the derivation of respective source terms in the vSG (cf. Kienzler et al., 2013).

Migration of radionuclides from a repository into the biosphere occurs mainly via the water pathway after the waste comes into contact with groundwater. Static leaching experiments were performed with salt brines as well as clay pore waters on irradiated aluminium-clad uranium-silicide ($U_3Si_2$-Al) dispersion fuels from research reactors. Measured corrosion rates are up to 4 orders of magnitude higher than matrix corrosion rates of $UO_2$-based LWR fuels under similar conditions. The crystalline (e.g. layered double hydroxides) and amorphous corrosion products effectively retain many radionuclides such as Am and Eu, whereas more "mobile" radionuclides such as Cs and Sr are predominantly released into solution. Ongoing research on the mechanistic understanding of the radionuclide retention processes (sorption and/or structural uptake) on a molecular level indicate also efficient retention of anionic species (e.g. of I, C, and Se) and provide additional knowledge to support the safety case for nuclear waste disposal.

The $UO_2$-based fuel elements for prototype HTR consist of up to 10,000 small fuel kernels with diameters of about 500 µm, embedded in a moulded graphite sphere with a diameter of about 60 mm. The fuel kernels are coated with pyrocarbon (PyC) layers (BISO) or PyC and SiC layers (TRISO). Leaching experiments with complete spent HTR fuel elements indicate extremely low radionuclide release rates under repository conditions, originating from U impurities in the graphite.

Investigations of isolated $UO_2$-TRISO coated fuel particles show a fast release of some fission and activation products into the gas phase (e.g. $^{14}CO_2$, $^{85}Kr$) and the aqueous phase (e.g. $^{137}Cs$) after cracking of the TRISO coating. In contrast, the corrosion of the fuel matrix was insignificant throughout the duration of these experiments (about 300 days). The results demonstrate that the HTR fuel elements represent a very stable waste matrix as long as the integrity of the SiC coatings of the kernels is not impaired.

An extensive knowledge base on the behaviour of these special fuel types has been build up throughout the last two decades. Open questions regarding the radionuclide release mechanisms under repository conditions will be discussed.

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

S9-01

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

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

S9-02

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

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

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

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

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

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

S9-03

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

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

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

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

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

# Lessons learned in Demonstration Projects Regarding Operational Safety During Final Disposal of Vitrified Waste and Spent Fuel

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

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

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

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

The steps to the full-scale test were:

- designing a shaft hoisting system based on the payload (transport system plus waste package) and designing the emplacement level
- selecting the Koepe system with cage and counter weight
- designing the loading and unloading devices
- performing probabilistic safety analyses and reliability study
- full-scale tests

All the devices were designed, fabricated, and tested in accordance with the applicable specific requirements, taking into account the payload of 85 t. All components were successfully tested.

A second test series in 1994/1995 (Filbert 1995) was aimed at demonstrating the feasibility of rail-bound handling system, horizontal transportation, and drift emplacement of self-shielded containers loaded with spent fuel. Here, emphasis was placed on the development and construction of components, such as an emplacement device, a transport cart, and a mining locomotive. Their capabilities of working under normal operating conditions and under conditions of operational disturbances were demonstrated in a full-scale surface test facility in order to ensure the safe handling of waste packages.

A third test series in 2008/2009 (Filbert 2009) was aimed at demonstrating the feasibility of a rail-bound handling system, horizontal transportation, and vertical borehole emplacement of BSK 3. All full-scale demonstration tests were carried out in a surface facility using canister dummies with the same dimensions and masses as real BSK 3 canisters.

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

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

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

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

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

S9-05

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

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

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

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

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

The project is of particular interest to regulatory authorities, technical safety organizations and waste management organizations responsible for the development and operation of geological disposal facilities.

The project is planned to be completed in 2015. The outcome of the project will be compiled as a technical report, which provides views and expectations regarding how the safety during operational phase of radioactive waste geological disposal facility can potentially impact on post-closure safety.

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