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Book of Abstracts

THEME

Underground Disposal Unit Design & Emplacement Processes

For a Deep Geological Repository

"Operational & Safety Considerations"

ANDRA

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List of Paper Abstracts

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PAPERS

Paper #1

RADIOACTIVE WASTE MANAGEMENT IN THE CZECH REPUBLIC

Vitezslav Duda / RAWRA

Radioactive waste and spent nuclear fuel are generated in the Czech Republic as a consequence of the peaceful use of nuclear energy and ionising radiation in many industries, particularly in the generation of nuclear energy, health care (therapy, diagnostics), research, and agriculture. The current extent of utilisation of nuclear energy and ionising radiation in the Czech Republic is comparable with that of other developed countries. The Concept of Radioactive Waste and Spent Nuclear Fuel Management is a fundamental document formulating government and state authority strategy for the period up to approximately 2025 (affecting policy up to the end of the 21st century), concerning the organizations which generate radioactive waste and spent nuclear fuel. The Concept puts forward solutions to provide for the disposal of waste in compliance with requirements for the protection of human health and the environment without excessively transferring any of the current impacts of nuclear energy and ionising radiation utilisation to future generations. The Concept was approved by the government of the Czech Republic in 2002. According to the Concept high level waste and spent nuclear fuel generated at the Dukovany and Temelín nuclear power plants will eventually be disposed of in a deep geological repository. Such a repository should commence operation in 2065. Work aimed at selecting potentially suitable sites began in 1992, but the final site has not yet been determined. In compliance with the aforementioned Concept, the Radioactive Waste Repository Authority (RAWRA) is responsible for finding two suitable sites till 2015. The current stage of evaluation covers the whole territory of the Czech Republic and involves complex criteria and requirements. On the basis of current findings RAWRA suggested six potential sites for further investigation at the beginning of the year 2003.

Paper #2

DEVELOPMENT AND DEMONSTRATION OF PROTOTYPE TRANSPORTATION EQUIPMENT FOR EMPLACING HL VITRIFIED WASTE CANISTERS INTO SMALL DIAMETER BORED HORIZONTAL DISPOSAL CELLS

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The work described in this paper was carried out by Andra (Agence Nationale pour la Gestion des Déchets Radioactifs) within the framework of the ESDRED Project. ESDRED is co-funded by the European Commission as part of the sixth Euratom Research and Training Framework Programme (FP6) on nuclear energy (2002/2006). Over a period of 4+ years Andra, working with a variety of Contractors mostly specialising in nuclear orientated mechanical applications, successfully designed, fabricated and demonstrated 2 prototype high level waste transport systems. One system was subsequently modified for transporting heavy prefabricated engineered barrier components. Both systems were designed to conform to the French national disposal concept as it was defined at the start of the ESDRED Project in 2004. In particular this included deep geological disposal in clay, disposal of both vitrified waste and spent fuel canisters, the ability to retrieve these canisters using the emplacement equipment and the placement of heavy sand/bentonite buffer rings which would encapsulate the canisters. Nothing in the design of the Andra emplacement equipment would preclude its utilisation in horizontal openings in other types of geological settings.

The first system developed is designed for the emplacement of CU1 type spent fuel canisters (1.25m OD, 5.39m long, weighing 43t). These are emplaced in 45m long horizontal bore holes (disposal cells) with a diameter of 3.3 m, which have been excavated in clay host rock and outfitted with a carbon steel lining, buffer rings and an inner steel sleeve inside the buffer rings. It is this inner steel sleeve, with an ID of only 1.35m which is the actual emplacement location. The technological innovation is the use of air cushion pallets as the transport mechanism in a small diameter (1.35m) circular opening, whereas air cushions are normally only used on flat surfaces. Once designed and proven this system was modified for the transport of other heavy loads notably packages of sand/bentonite rings weighing 17 tons and having a diameter of 2.25 metres.

The second system developed is designed for the emplacement of C type vitrified waste canisters (0.590m OD, 1.60m long, weighing 2t). These too are emplaced in 40m long horizontal bore holes (disposal cells) which have been excavated in clay host rock and outfitted with a 0.620 ID carbon steel lining. The technological innovations include the fabrication of a pushing robot (using pneumatic toric jacks for propulsion) which is less than 1m long and weighs less than 1 ton and the fabrication of dummy canisters which incorporate ceramic inserts acting as sliding runners.

For both systems simplified versions were built early in the project to quickly, but convincingly, prove the fundamental concepts. This too is described in the paper. Finally full scale versions of both systems were built and each includes a remotely operated emplacement machine, capable of emplacing a dummy canister that has a mass and geometry identical to the real canisters.

The paper starts by defining the rationale behind the design of the disposal cells and the functional requirements of the systems. It goes on to describe the various components that make up the 2 emplacement systems and the general set-up used for the test work and the demonstrations.

The main results of the 2 test programmes are detailed and the main test cases carried out are listed. The difficulties encountered from commissioning through final demonstration, together with the weak points of each system and the remedial solutions considered, are also explained. Results are discussed. The main achievements are highlighted. The paper concludes with a discussion regarding future enhancements and the likely prospects for an industrial application.

The successful completion of the 2 main test campaigns confirms the feasibility of emplacing canisters containing long lived HLW in 40m to 45m long horizontal bore holes (disposal cells) with only minimal clearance between the canister and the disposal cell wall. The developed technology is considered to be mature enough for a potential industrial application.

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Paper #3

THE SWISS CONCEPT FOR DISPOSAL OF SPENT FUEL AND VITRIFIED HIGH-LEVEL WASTE

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Management of spent fuel (SF), vitrified high-level waste (HLW) and long-lived intermediate-level waste (ILW) is based on the concept of deep geological disposal, namely long-term, effective isolation of the waste in suitable deep rock formations. The first project studies carried out by Nagra in this respect already lie more than 20 years in the past (Nagra 1980), when disposal in the crystalline basement and in clay was considered. The strategy developed by Nagra over the years agrees well with the concept of "monitored long-term geological disposal" as formulated by (EKRA 2000) and contained in the new Nuclear Energy Act of 2003 (KEG 2003).

This paper provides an overview of the concept for facilities and operation of a deep geological repository for SF/HLW/ILW (Nagra 2002a), as prepared for the 'Entsorgungs nachweis' project, together with a geological synthesis report for the Zürcher Weinland (Nagra 2002b) and a report on long-term safety (Nagra 2002c).

The facilities and operation concept look at the feasibility of constructing a repository in the Opalinus Clay of the Zürcher Weinland. It also provides project-specific input for analysing and demonstrating the long-term safety of such a repository. The individual structural elements and facility components for which the feasibility study was conducted are brought together as a modular system to form a stand-alone reference project. They can be adapted later to meet local features and requirements.

The main focus of the paper shall be on selected system elements concerning design, layout and operational aspects including operational safety.

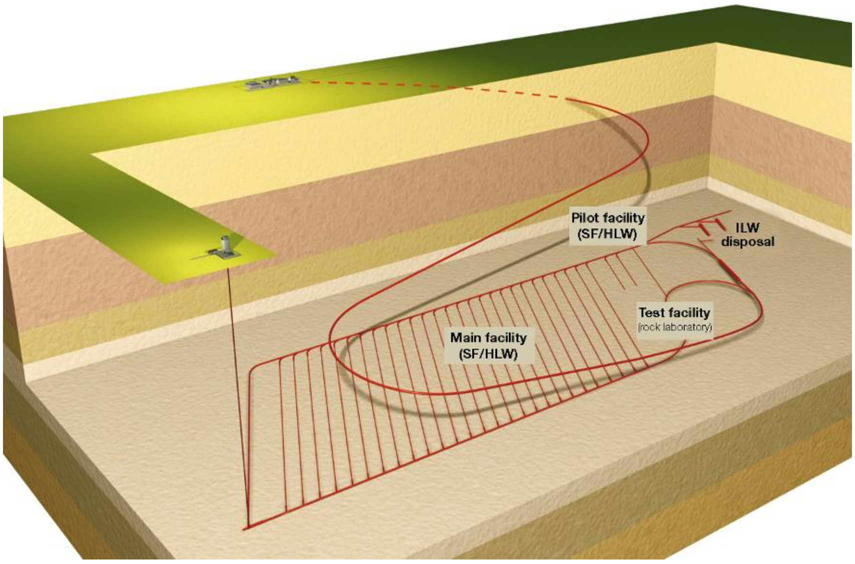

Fig. 1 3D view of the deep geological SF/HLW/ILW repository in Opalinus Clay

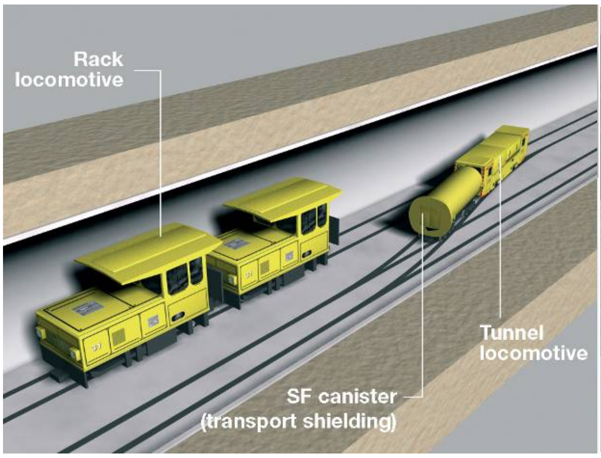

Fig. 7 SF canister in the central area:
the tunnel locomotive has taken over the canister (in transport shielding) from the rack locomotives

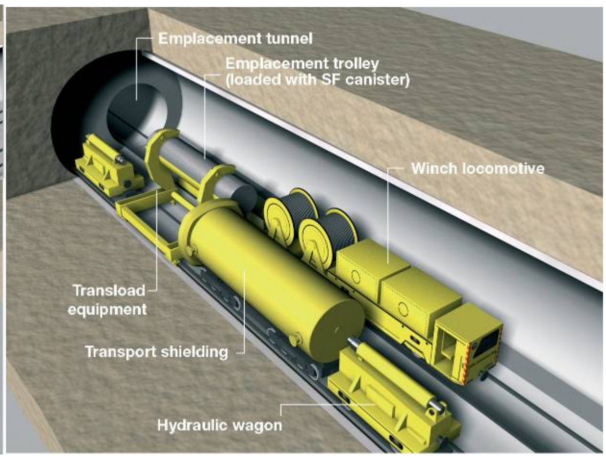

Fig. 8 SF canister in the transition area:
the canister has already been removed from the transport shielding and transferred from the transload equipment to the emplacement trolley.

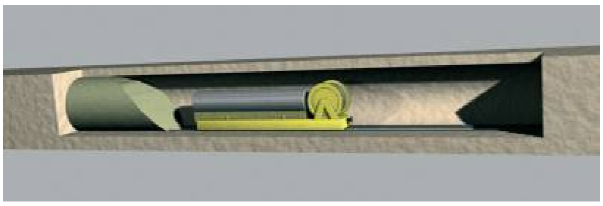

Fig. 9 Emplacement trolley with spent fuel canister at emplacement position

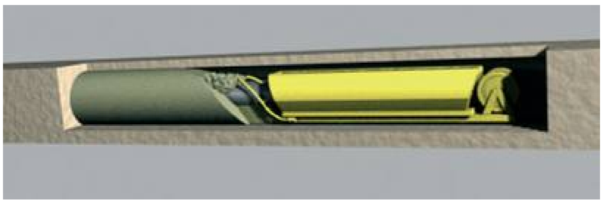

Fig. 10 Wagon emplacing granular bentonite material as tunnel backfill

Paper #4

EXPERIMENTAL PROGRAMME TO DEMONSTRATE THE VIABILITY OF THE BELGIAN SUPERCONTAINER CONCEPT FOR HLW

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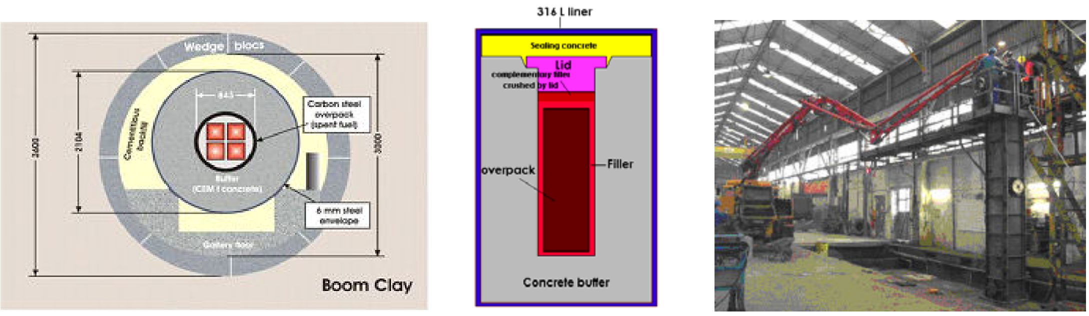
The EIG EURIDICE (a joint venture between the Belgian Organisation for Radioactive Waste Management – ONDRAF/NIRAS – and the Belgian Nuclear Research Centre – SCK•CEN) is responsible for performing large-scale tests, technical demonstrations and experiments to assess the feasibility of a final disposal of vitrified radioactive waste in deep clay layers. This is part of the Belgian Research and Development programme managed by ONDRAF/NIRAS.

The current Belgian reference design for vitrified HLW and spent fuel assemblies is the so-called Supercontainer design. The vitrified waste canisters or spent fuel assemblies are enclosed in a carbon steel overpack which has to prevent contact between water from the host formation and the waste during the thermal phase. In order to maintain favourable chemical conditions to avoid corrosion during this period (several hundred or even thousand of years), the overpack is surrounded by a high alkaline concrete buffer of about 70 cm thick. The buffer also provides permanent radiological shielding for the workers, simplifying handling and other operations. All the components of the Supercontainer are constructed in above ground installations, thus creating favourable QA/QC conditions. After the emplacement of the Supercontainers in the disposal galleries (Figures 1 LEFT and MIDDLE), the remaining space will be backfilled.



Figures 1: Design of the supercontainer. LEFT: cross-section of a disposal gallery with a supercontainer for spent fuel. MIDDLE: different components of a supercontainer. RIGHT: test in column to verify the construction of the buffer

Tests to demonstrate the viability and the construction feasibility of the supercontainer design have been initiated. The viability programme includes:

- Tests to verify the feasibility to construct and emplace the components of the supercontainers.
- Tests to verify the feasibility to backfill the disposal galleries once the supercontainers are placed.

Supercontainer construction

Tests in column to verify the construction feasibility (risk of cracking) of the buffer with two different types of concrete (a self-compacting concrete – SCC - and a rheoplastic concrete RPC) were performed in collaboration with the Belgian concrete factory Socea (figure 1 RIGHT). A characterization programme on cores from these columns is occurring.

In an actual case, the prefab shell (concrete buffer) will be constructed in normal workshop conditions because the waste is not yet present. The next step, being the filling of the annular gap between the overpack containing the heat-emitting waste and the concrete buffer, is technically more challenging.

Indeed, the inserted waste will generate a significant amount of heat - up to 100°C can be expected at the interface overpack-filler - influencing the installation of the filler.

Two material types are currently considered as filler: a cementitious grout or a powder (e.g. portlandite). To evaluate candidate materials and installation techniques, small scale tests will be conducted under relevant temperature and geometric conditions. For these tests, hot-cell conditions will not be mimicked. The key issues to be studied here are:

- Finding an appropriate combination of material and installation technique;
- Optimisation of the dimensions of the annular gap for this combination;
- Study the interaction between fresh, non-hardened concrete and the heat source (for the grout type only).

The emplacement technique of the lid and sealing concrete has not yet been fixed. Various mechanisms will be examined, mainly the choice between using end pieces that are pre-cast or cast immediately in the Supercontainer. This aspect will also be the object of an experimental programme.

Backfilling disposal galleries

In the framework of the EC ESDRED project EIG EURIDICE and ONDRAF/NIRAS, investigate different technologies to apply the backfill. A wide variety of materials was tested and two options to apply the backfill were investigated: gunite the gap with a granular material and backfill the gap with a grout.
As result of the guniting tests, a list of appropriate materials was established: pure sand (siliceous), pure bentonite (MX-80), a 75/25 mixture of MX-80 with sand, a mixture of MX-80 with cement and sand only slightly enriched with cement. Another important achievement was the robustness of the custom designed and built projection machine which operated duly under harsh conditions.
The tests on the grout injection method were also successful. A custom designed grout was injected in a constant flow by a grout pump. To approach the expected conditions at the time of injection in the real-life situation, the mock-up temperature was kept at 40-50°C. After hardening of the grout, several core samples were taken to analyse the strength and thermal conductivity of the backfill, these analyses are currently ongoing. A slice was cut from the mock-up using a diamond cable and the perfect filling of the gap was illustrated (see Figure 2 LEFT).

At the beginning of 2008 a large scale backfill test in mock-up with grout will be performed. The mock-up is being constructed (Figure 2 RIGHT).

Figures 2: Mock-ups for backfill tests with grout. LEFT: small-scale mock-up after slicing. RIGHT: construction of 30 m long large-scale mock-up

Paper #5

OPERATIONAL SAFETY FOR A KBS-3H REPOSITORY

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SKB

In Sweden both vertical (KBS-3V) and horizontal (KBS-3H) disposal of the copper canister with spent nuclear fuel are studied. The purpose of this paper is to present a preliminary analysis on Operational Safety for a KBS-3H repository highlighting the main areas where the two concepts differ. The main differences between the two KBS-3 concepts are the activities in the reloading station and in the disposal drift:

1. In the KBS-3H concept, the canister and buffer are placed inside a perforated steel cylinder and disposed of as a package, normally called the Super Container (SC).
2. In the KBS-3H concept, the SC container is emplaced with a special deposition machine designed for handling heavy loads in a long drift with small diameter. The KBS-3H drift has a diameter of only 1.85m and the SC has a diameter of 1.765m.
3. The KBS-3H concept does not require backfilling of the deposition drifts but installation of distance blocks are needed between the SC for not exceeding allowed maximum temperatures in the buffer.

The artist impression of the KBS-3 repository is shown on Figure 1 and the location of the reloading station is shown on the figure as well as the arrangement of the horizontal drifts.

The main differences between the KBS-3V and KBS-3H concepts are mainly in the reloading station and activities associated with emplacement of the SC and distance blocks.

This paper is therefore focused on presenting the risk analysis for these two areas.

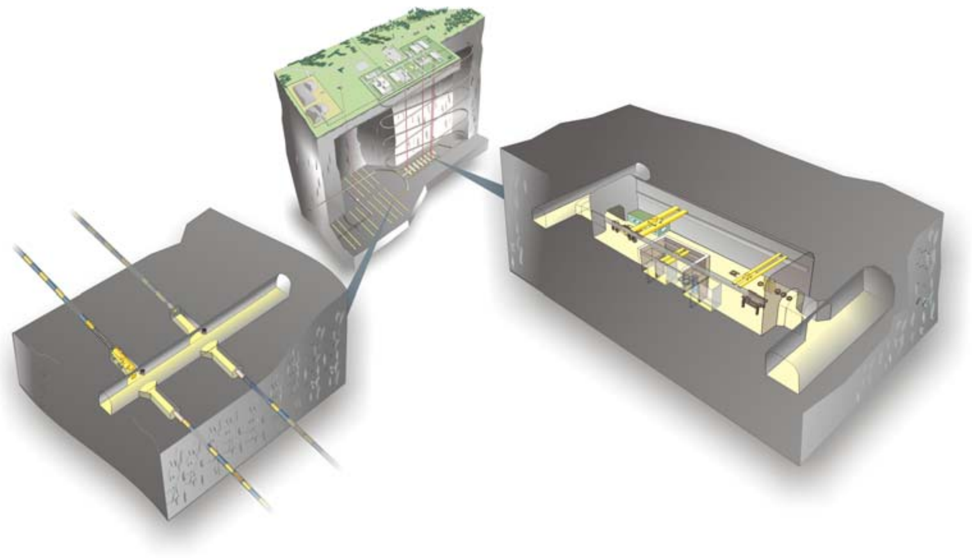

Figure 1. Artist impression of the KBS-3H repository with an enlargement of the reloading station and some disposal drifts.

The reloading station will have a different function in concept KBS-3H compared to KBS-3V. In KBS-3H the main activities in the reloading station are:

- The preassembly of the SC is done in the reloading station but outside the shielded handling cell. The function of the shielded cell is to provide radiation protection during the transfer of the copper canister from the transport cask into the SC. This preassembly will be done inside the Transport Tube standing vertically on a small wagon in the Transport Tube pit and with the top gamma gate removed. First, the SC perforated shell is placed inside the Transport Tube and then all buffer material, the top lid of the shell and the Transport Tube top gamma gate. The complete SC is now inside the Transport Tube except the canister. Further, the top gamma gate is not bolted to the shell of the Transport Tube at this stage. The Transport Tube will now be transferred with the wagon into the shielded cell for receiving the copper canister and welding of the top lid to the SC shell.

- In the handling cell, the top gamma gate is removed as well as the top plate and the top bentonite plug.
- The transport cask is unloaded from the ramp vehicle and the outer lid is removed as preparation to enable lifting the canister out of the cask. The cask is then placed on a small wagon in the transport cask pit for transport inside the handling cell. Inside the handling cell the inner lid of the transport cask will be removed and the cask is open.
- The canister can now be lifted out of the cask and placed inside the buffer material in the SC. With the canister in place the top bentonite block is placed on the canister inside the shell. The top plate of the shell is then installed and welded to the SC shell. All these activities are done inside the Transport Tube.
- After completion and control of the welding, the top gamma gate can be bolted to the Transport Tube. The Transport Tube is now ready to be removed from the handling cell.
- The empty transport cask for the canister can now also be prepared for removal including reinstalling the inner and outer lids. These operations do not differ compared with KBS-3V except the removal of the canister which is done inside the handling cell thereby providing radiation protection during the transfer of the canister from the cask to the prepared SC in the Transport Tube.
- The last operation in the reloading station is the transfer of the filled Transport Tube to the trailer for transport to the niche in the repository area for emplacement.

Obviously there are a number of additional risks compared to KBS-3V during the assembly of the SC in the reloading station. But the work in the shielded handling cell must also be compared to the actions where the copper canister has to be lowered into the deposition hole filled with buffer rings in the repository area in the KBS-3V concept.

Additional risks, compared to KBS-3V are identified in the following handling steps:

- Lowering of the copper canister into the SC filled with bentonite and lifting of the Transport Tube with the SC.
- The canister is moved from the transport cask into the SC inside the Transport Tube in the Shielded handling cell.
- The shielded handling cell contains equipment that may cause additional risks, e.g. increase of the fire ignition probability.

The risk in the transport of the transport cask with the canister in the ramp is the same as for concept KBS-3V.

The risk in the transport of the loaded Transport Tube to the niche in the deposition area is similar to the transport of the canister inside the shielded tube in the deposition machine for the KBS-3V concept.

When the Transport Tube with the SC has been transported to the repository area, the following activities are performed:

- Placing the Transport Tube in the correct position outside the drift.
- Docking of the Transport Tube to the deposition drift
- Docking of the Deposition machine to the Transport Tube
- Initialization of the deposition process for the SC.
- Transportation of the SC inside the deposition drift
- Emplacement of the SC in the deposition drift
- Transport and emplacement of distance blocks in the drift
- Construction of the sealing plug (similar activities as for the KBS-3V concept)

Conclusion about failure during emplacement of the SC and distance blocks

Compared to the KBS-3V concept the following advantage is identified:

- The lifting of the SC is done with water cushions and the lifting height is very low. No risk for dropping the waste package.
- No backfilling of large deposition drifts, about 6 000 m^3 each, is required.

The following disadvantages from a safety/availability point of view are identified:

- The consequences of malfunction of the deposition equipment including fire need to be investigated further and the small diameter of the drift makes it difficult to carry out correcting actions. However, it would be possible to take back the deposition machine for repair if anything happens.
- If the ground water flow into the drift is too high piping in the buffer may occur. Installations of compartment steel plugs in a drift may solve this problem but it is still a risk that must be investigated further. The requirements on ground water management are higher for KBS-3H then for KBS-3V.

Paper #6

OPERATIONAL SAFETY AND RADIATION PROTECTION CONSIDERATIONS IN DESIGNING A HLW REPOSITORY IN GERMANY

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In Germany the reference concept for disposal of heat generating radioactive waste considers emplacing canisters with vitrified waste in deep vertical boreholes drilled from the drifts of a repository mine in salt at a depth of 870 m. Spent fuel is to be disposed of in self-shielding POLLUX casks in horizontal drifts. An optimized disposal concept anticipates emplacing unshielded canisters with vitrified HLW and canisters containing the fuel rods of 3 PWR or 9 BWR fuel assemblies in boreholes with a diameter of 60 cm and a depth of up to 300 m.. In all cases the void space between POLLUX cask and drifts and canisters and borehole wall will be backfilled with crushed salt.

1. Operational Safety

Based on a detailed description of all underground disposal operation steps, the possible impacts on the disposal operations were analysed and the need for further studies determined. The disposal operation steps comprise e.g. rail bound transport from the shaft to the emplacement drift and emplacement process itself. As possible impacts the following occurrences were considered: ventilation failure, power supply failure, rock mechanics impact including cross-section convergence, irregular floor uplift and rock fall, brine and natural gas intrusion, derailing of transport carts and finally internal fire.

2. Radiation Protection

According to the German Atomic Energy Act (AtG), the design, construction and operation of a nuclear site like a final repository has to be licensed by the responsible authority. The Radiological Protection Ordinance and further guidelines i.e. concerning the emission and immission of released radioactive nuclides or the risk analysis of possible failure, build the basis for the licensing procedures.

To ensure adequate protection against undue radiation exposure the repository is divided into different radiological protection areas. Generally, the handling of shielded waste packages above und under ground (including all the pathway of transport and emplacement area) takes place in a controlled area with restricted access. Due to the high radiation dose rate level, handling of unshielded canisters is carried out in exclusion areas (transfer cells). Other areas of the repository, in which conventional mining activities are carried out, e.g., disposal drift excavation, as well as the areas giving access to the controlled area are monitored areas. With this, the areas where the mine air can potentially be contaminated are separated from conventional areas each one having a separate ventilation regime.

It should be pointed out that only conditioned waste packaged in qualified containers is acceptable for disposal, and that independent package quality control procedures prior to delivery are in place. Nevertheless, a radiation protection entrance check is performed to confirm compliance with the disposal package acceptance rules.

Radiological monitoring at the repository includes dose rate surveillance at certain points of interest as well as the surveillance of exhaust air discharged from the surface buildings and from the repository mine. The instruments measure volatile aerosols and radioactive gases. Further radiological monitoring of the air in the site and its vicinity as well as of waters and soil in the surroundings is carried out. Meteorological data are also registered.

The personnel is kept under radiological surveillance by measuring the air at work places, the individual dose and if necessary, the incorporation. Design dose targets for exposed workers were considered for the (in principle) avoidable incorporation of radionuclides and the unavoidable external radiation exposure. To achieve this, a specific ventilation regime has to be put in place, as well as a sound balance between radiation source shielding and shielding of the working places. Where necessary, remotely controlled waste handling will be applied. In general terms, exposure to external radiation is much more important than the effective dose by incorporation.

3. Incident analysis

Incident analysis concerning internal events with possible release (impact on packages by mechanical interactions, fire, explosion, criticality, heat etc.) or external events (seismic, lightning, flood, airplane crash etc.) governs the layout of the repository and the disposal conditions.

Long-term and operational safety assessments by the use of deterministic and probabilistic safety analysis are discussed.

Paper #7

OPERATIONAL SAFETY AND RADIOPROTECTION CONSIDERATIONS WHEN DESIGNING THE ILW-LL DISPOSAL ZONE

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As for any other nuclear industrial facility, in a radioactive waste repository the various waste disposal operational activities from construction to closure can present a risk to human(workers and public) and the environment. In accordance with the December 30, 1991 French Waste Act, Andra has conducted feasibility studies regarding the disposal of HLW & ILW-LL waste in a clay host formation. The "Dossier 2005 - Clay" includes a description of the operational safety analysis that was conducted for ILW-LL waste disposal in underground horizontal drifts. The objective of this paper is to present that safety analysis and its impact on the design at the feasibility stage.

The safety analysis covered the operations from the reception of the waste transport casks to the disposal of the waste disposal package in its final emplacement location inside the disposal cell. Since the surface facilities' operations are similar to those of other nuclear ones, this paper focuses on the specificity of the deep repository, i.e. the operational safety and radioprotection aspects applied to the deep disposal drift.

Andra has selected an ILW-LL design based on large horizontal drifts (diameters of 10 to 12m, and lengths of 250m). The primary waste packages are put inside a specific concrete overpack before their disposal. These overpacks are remotely stacked inside the horizontal drifts.

The operational safety analysis aims to ensure that risks are kept under control through provisions in the design of the repository and by operating the facility in compliance with operational requirements and the safety functions. The requirements and the safety functions, developped at this stage of the feasability studies, will be explained. The operational safety analysis is structured around physical components and real activities (construction, operation, closure) through a dedicated risk analysis.

Due to the large variety of different ILW-LL waste, in order to identify the potential measures employed to counter each risk, there was a need to conduct the safety analysis in relation to four major hazards. These will be developed in the paper. The first one is the basic nuclear risk inherent in normal operations due to external exposure by irradiation (and also internal exposure by inhalation). Irradiation of ILW-LL waste packages are generally well above the 2mSv/h level. The three other ones are related to the following particular accidental situations:

- the ILW-LL waste package drop during its emplacement in a disposal cell,
- the explosion associated with the emission of gas (due to hydrogen potential accumulation) ,
- the fire breaking out in underground installations during construction or while in operation,

The present paper will detail the above-mentioned items, the analysis methodology, the preliminary results of the risk analysis, and in particular the mitigating design measures taken. It will conclude with some suggestions regarding the ongoing development and design evolution that are needed in view of meeting the 2015 goal of a licensing permit application in accordance of the June 28, 2006 French Waste Act.

Paper #8

PRACTICAL AND SAFE IMPLEMENTATION OF DISPOSAL WITH PREFABRICATED EBS MODULES

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The use of prefabricated EBS modules ("PEMs") to minimise the problems involved with handling compacted bentonite and ensuring that it is emplaced to established quality levels has been investigated in various national programmes for disposal of both HLW and SF. To date, however, this has tended to be decoupled from studies of related operational aspects such as assessing / minimising the consequences of use of concrete for support structures, ensuring ease of tele-operated reversal of waste packages during emplacement (e.g. in the event of operational disturbances) / retrieval at a later time, logistical optimisation (especially for programmes with large waste inventories) and cost minimisation.

It is clear that specific aspects of operational safety and practicality can be considerably enhanced if designs are modified with a focus on them. It is trickier to provide optimised solutions, which simultaneously address all these critical points. Nevertheless, with a bit of lateral thinking, it appears possible to devise options that may not only ease the operational phase, but may also actually improve post-closure safety case robustness – although improved, more realistic performance assessment codes and databases will be needed to demonstrate this rigorously.

To illustrate this approach, an example will be presented based of disposal of vitrified HLW in a fractured hard rock; the general principles involved are, however, also applicable to other higher activity wastes and other host rocks. Key aspects of the design are:

- Optimisation of PEM design for both short-term and long-term performance
- Development of a rail emplacement system which eases remote handled emplacement / recovery
- Large diameter, lined emplacement tunnels to ensure operational robustness
- Use of multi-package overpacks (e.g. 6 HLW containers in each PEM) and short tunnels to ease emplacement logistics
- Backfilling with a non-swelling sacrificial pH buffer (eases handling and improves performance)

The resulting design builds on information gained during design and equipment development studies carried out in Japan and, in particular, the experience gained in large-scale demonstration projects carried out in URLs worldwide. Although some novel aspects of this approach will need to be demonstrated, this is balanced by eased implementation and, potentially, significant cost reductions.

Paper #9

INTEGRATED SAFETY CASE DEVELOPMENT FOR DEEP GEOLOGICAL REPOSITORIES

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2 - McKinley Consulting, Baden/Dättwil, Switzerland

The term "safety case" has been extensively discussed in recent years (e.g. NEA, 2004), emphasising that the convincing demonstration of safety needed to license a facility involves more than a quantitative performance assessment. There has, however, tended to be a focus on post-closure safety; as projects move closer to implementation, however, construction and operational safety become increasingly important. In Japan, this is particularly topical, as a recent amendment of the law governing geological disposal has established a requirement for stepwise production of a comprehensive safety case that will specify operations up to final repository closure.

Safety concerns start at the point of planning site characterisation, involving both assessing direct hazards associated with characterisation technology and the potential of such work to degrade the post-closure performance of the site (as evaluated by baseline monitoring). Evaluation of operational logistics has highlighted how critical the EBS design is for the reference case of emplacement of 5 packages of HLW per day. The emplacement process will be carried out by remote handling, hence needs to be robust and to fail safe in the event of any kind of operational perturbation. Any recovery and remediation work resulting from perturbations should, wherever possible, be possible with tele-operated equipment.

The post-closure safety case is based on the performance of a number of barriers, which are established during construction, operation and closure. Such barriers must be confirmed using quality assured methods, supported as required by monitoring. The requirement for an integrated assessment means that even the final process to end institutional control and transfer any liabilities from the implementer needs to be considered at present, even though this will undoubtedly be refined and tailored to the site characteristics over the many decades that will pass before this occurs.

The paper will illustrate how the repository design in general – and the operational procedures in particular – can be optimised in order to improve practicality. This not only reduces risks to operators, but also, due to improved quality assurance, contributes towards building a more convincing case for post-closure safety.

NEA, (2004). Post-closure Safety Case for Geological Repositories - Nature and Purpose, OECD/NEA, Paris, France.

Paper #10

ESDRED PROJECT MODULE 1
FROM BENTONITE POWDER TO EB UNITS:
AN INDUSTRIAL PROCESS

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Introduction

This paper is a digest on the development of Module 1 of the ESDRED Project of the European Community. The ambition of the Project itself is to define and manufacture an engineered barrier designed to line horizontal cells in order to demonstrate the technical feasibility of such concept for radioactive waste disposal. Module 1 focuses on the definition and manufacturing of scale-1 bentonite-based rings and disks, as well as all associated gripping, handling, assembling (in four-unit sets), transportation and conservation means.

The French National Radioactive Waste Management Agency (ANDRA) hired a pool of other organisations, including the French Atomic Energy Commission (CEA) and two compagnies, *Minéraux et Produits Chimiques* (MPC) and SEGULA *Ingénierie* with a view to manufacturing and testing the rings. That temporary pool of companies, named GME *(groupement momentané d'entreprises*), had the role to develop all the industrial processes necessary to the fabrication of buffer rings and discs on scale 1, as well as the related means and procedures.

The project started in July 2004 by the creation of a working group at the instigation of the CEA. The working group developed a strategy in response to a call for tenders by ANDRA. As the foundations of the GME took shape, the Legal Services of the CEA, SEGULA TECHNOLOGIES and MPC drafted a convention by which the three companies agreed to work in close relationship throughout the project.
ANDRA accepted the technical arguments of the GME and enacted the contract on 4 April 2005.

Buffer definition

The main specifications of the buffer are the hydraulic conductivity (10^{-12} m/s), the thermal conductivity (1.2 W/(m.K)) and the gas permeability. The physical properties of the buffer as specific gravity, swelling pressure and composition are derived from these specifications.
A complete laboratory scale THM study carried out during the year 2005 proposed a buffer formulation based on a compacted 70 % of MX80 bentonite and 30% of sand mixture, with a water content of the powder in the order of 12% and a compaction pressure of 80MPa. The resulting compacted material has a bulk density of 2,200 kg/m3, a dry density of 1,960 kg/m3, a dry density of clay in the mixture of 1,770 kg/m3, a thermal conductivity close to 1.5 W/(m.K) and a residual swelling pressure equal to 2 MPa after a 20% expansion of its volume. The weak hydraulic conductivity of the MX80 is slightly affected by the presence of sand and remains lower than 10^{-13} m/s.

Various tests conducted on the compacted material at both the laboratory and industrial scale provided useful information for the design and fabrication of ring compaction mould at scale 1 such as the quantification of the post swelling or a PTFE spray for stripping purposes.

Fabrication of the buffer rings and discs

The initial challenge was to find a suitable compacting device capable of applying a pressing force of 40,000 t. Investigations concluded on the possibility of conducting compaction operations by using the press at AUBERT & DUVAL (Figure 1), in the INTERFORGE plant, at ISSOIRE (France), where activities are mostly metallurgical in nature and where the press has a very heavy load plan.

Figure 1: 65000 tons press of INTERFORGE, Issoire, France

The GME launched investigations in preparation for the preliminary project brief on the mould and initiated studies on the clay material. In parallel, the GME carried out a technological watch for cost-optimisation purposes, after the price of raw materials (especially steel) escalated during the first half of 2005. Those events led the GME to orient the design of heavy-tonnage items towards cast iron instead, in order to reduce both purchasing and machining costs.

As the process took shape, so did the architecture of the mould. The GME met with AUBERT & DUVAL in order to present the project and to convince them to dedicate a few hours of production time on their press to the fabrication of rings and disks.

Manufacturing the mould is divided in several steps. FERRY CAPITAIN is in charge of making the essential parts of the mould. In the light of its experience and available means, LE CREUSOT MÉCANIQUE was selected by the GME to assemble the mould and to ensure its final set-up. ANDRA, together with experts of CLAY TECHNOLOGY, attended the factory-acceptance meeting at Le Creusot, at the end of May 2006 (Figure 2). In parallel, the GME managed the design, realisation and set-up of associated means: vacuum spreader, stripping kit, turn-around station, containers, pallets, etc. At the end of the test and qualification phases of the clay material, the GME mixed it and conditioned it into big bags that were brought bring to the production site in order for rings and disks to be made.

All relevant means were transferred to Issoire for the first pressing set-up on 12 June 2006. That experiment generated a single ring, helped to identify the necessary evolutions of the process and constituted the first experiment for all GME and INTERFORGE partners.

The problems encountered during the second pressing test on 3 July 2006 led the GME to make important decisions regarding the evolution of the mould behaviour, associated means, press use and teams. The mould and the stripping kit were sent back to LE CREUSOT MÉCANIQUE in order to be modified in accordance with the GME's specifications. In parallel, the GME and INTERFORGE worked together with a view to optimising press means and the execution of the operations.

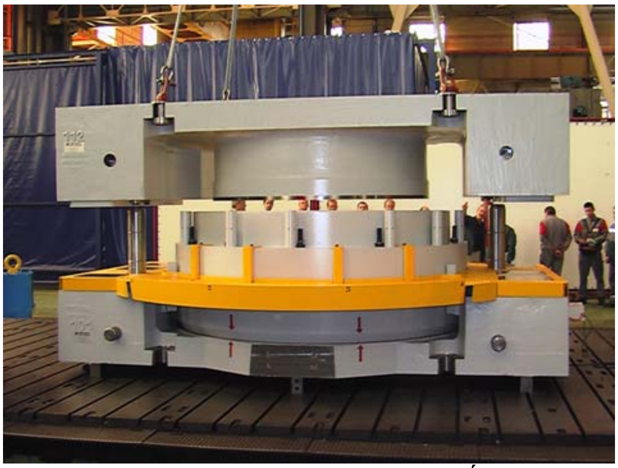
Figure 2: Test of the mould at LE CREUSOT MÉCANIQUE facilities

The third pressing test took place from 5 a.m. to 9 p.m. on 18 December 2006. During that time, nine rings and two disks were produced (Figure 3). Production was continuous throughout the day with a reproducibility rate that guaranteed a sound homogeneity in the production. The GME restored the press hall to its initial operating conditions the next morning and shipped its entire output and means to the MPC hall, at Limay (Yvelines Department, France).


Figure 3: A ring just after stripping

A dimensional control confirmed that rings and disks were consistent with specifications, which means that assembly and containering operations could start. The Project ended, in terms of production, with the successful assembly operations that were performed at Limay, during the first half of 2007.

The three conditioned assemblies were deposited into their specific containers and are currently stored at Limay, together with all equipment that was developed for the project.

Paper #11

FULL-SCALE DEMONSTRATION OF EBS CONSTRUCTION TECHNOLOGY, (I) BLOCK, PELLET AND IN-SITU COMPACTION METHOD

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The long-term safety of geological repository for high-level radioactive waste in Japan is secured by a multibarrier system including an engineering barrier system (EBS) which consists of bentonite-based buffer material and overpack sealed vitrified wastes. Remote handling and remote emplacement technologies of overpack and buffer material are significant on the performance of EBS and operation safety of geological repository. It is extremely important to show applicability of remote handling and remote emplacement technologies widely at various site conditions by technological development of the geological repository, because specific candidate site has not selected yet in Japan.

In this study, test results of full scale demonstration were provided for basic EBS concepts, BLOCK, PELLET FILLING and IN-SITU COMPACTION, shown in Fig.1 and Fig.2. An alternative concept, PEM method, will be presented by the paper (II).

- **Bentonite Block**

Applicability of manufacturing technology of buffer material was verified by manufacturing of full scale bentonite ring which consists of one-eight (1/8) dividing block (Outside Diameter :OD 2,220mm H300mm). Density characteristic, dimension and scale effect, which were considered the tunnel environment under transportation, were evaluated. Vacuum suction technology was selected as handling technology for the ring. Hoisting characteristic of vacuum suction technology was presented through evaluation of the mechanical property of buffer material, the friction between blocks, etc. by using a full-scale bentonite ring (OD2,200mm, H300mm). And design of bentonite block and emplacement equipment were presented in consideration of manufacturbility of the block, stability of handling and improvement of emplacement efficiency.

- **Bentonite Pellet Filling**

Basic characteristics such as water penetration, swelling and thermal conductivity of various kinds of bentonite pellet were collected by laboratory scale tests. Applicability of pellet filling technology was evaluated by horizontal filling test using a simulated full-scale drift tunnel (OD2,200mm, L6m) . Filling density, grain size distribution, etc. were also measured.

- **In-Situ Compaction of Bentonite**

Dynamic compaction method (heavy weight fall method) was selected as in-situ compaction technology. Compacting examination which used a full scale disposal pit (OD2,360 mm) was carried out. Basic specification of compacting equipment and applicability of in-situ compaction technology were presented. Density, density distribution of buffer material and energy acted on the wall of the pit, were also measured.

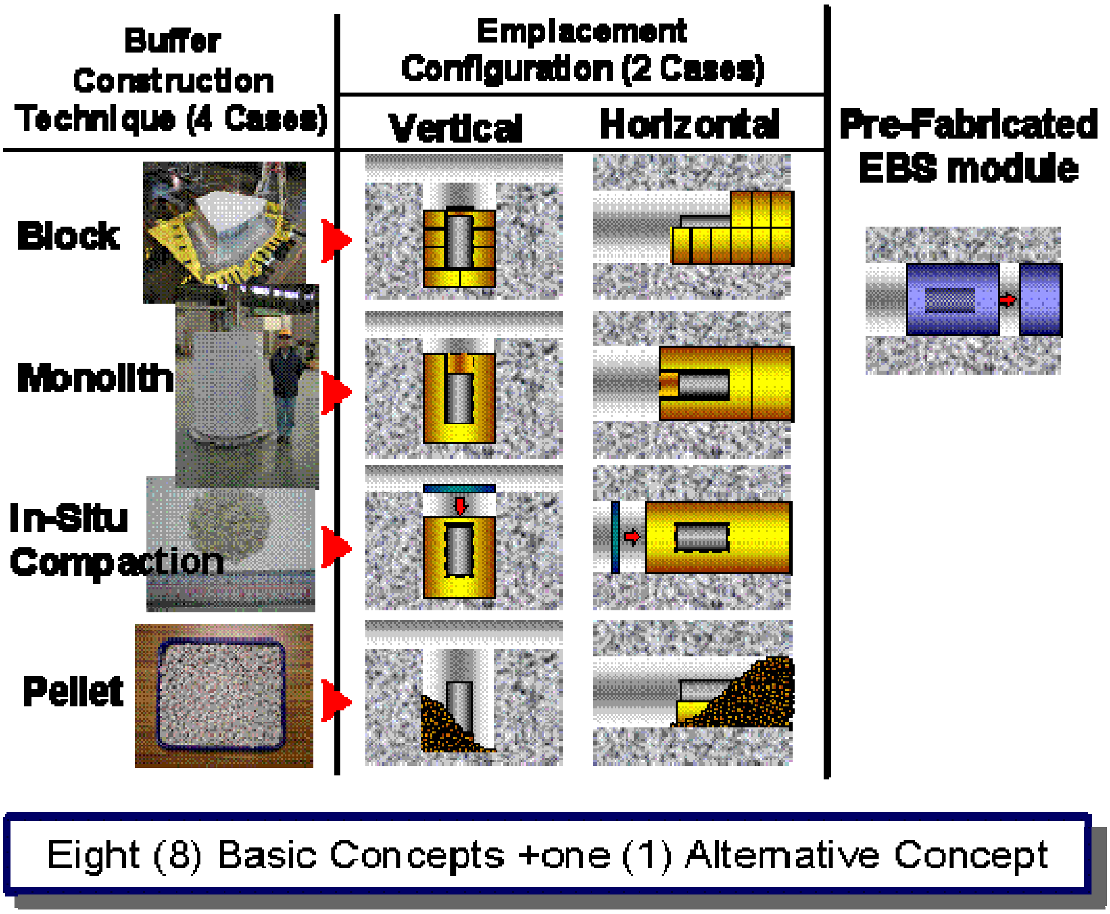

Fig.1 Various Emplacement Concepts

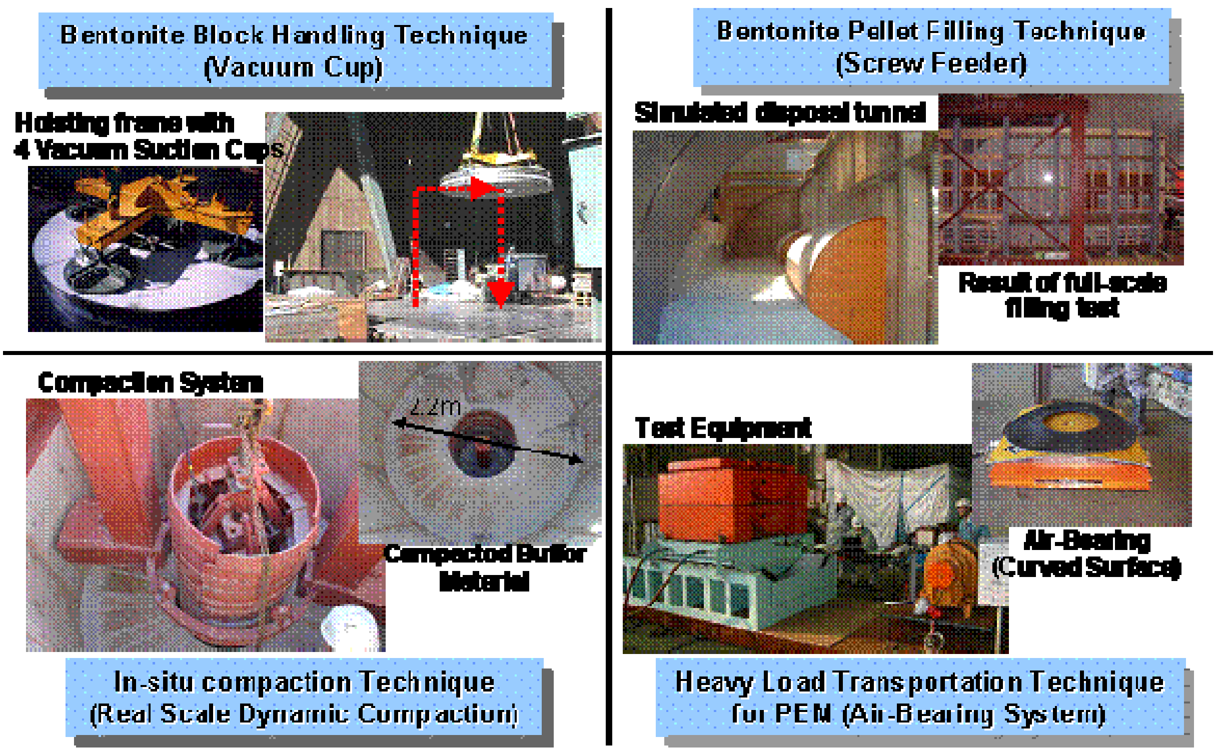

Fig.2 Full Scale Demonstration Tests

Paper #12

FULL-SCALE DEMONSTRATION OF EBS CONSTRUCTION TECHNOLOGY, (II)DESIGN, MANUFACTURING AND TRANSPORTATION OF PRE-FABRICATED EBS MODULE (PEM)

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Geological disposal concept of Japan is to seal a vitrified waste(HLW) inside the overpack which is placed in a stable geological formation surrounded by a bentonite-based buffer material. Transportation and emplacement technologies are significantly affected on the quality and the performance of EBS(engineering barrier system). Pre-fabricated EBS module(PEM), which contains one unit of buffer material and overpack, is transported and emplaced in the drift tunnel after it is assembled at aboveground facility. This method has technical advantages due to its simplified handling procedure, while there are several works should be done about construction and heavy load transportation techniques.

In this study, PEM was investigated as a full-scale demonstration for the design, manufacturing and construction by using simulated buffer material and overpack in consideration of horizontal emplacement. Also near full-scale tests were conducted to examine the applicability of air-bearing system which can be used to transport a heavy load at the drift tunnel as for PEM.

With regard to PEM casing, design requirements were selected from the viewpoints of EBS performance and operation safety issues. The construction procedure was examined in consideration of the shapes of buffer material, which are previously positioned inside the casing. And design procedure of the casing was also examined and presented. A full-scale PEM casing as a longitudinally two-part divided cylinder type with connection flanges was manufactured by using carbon steel plate.

The wall thickness of this non-leak tight type PEM casing was evaluated its mechanical integrity by 2-dimensional stress analysis in consideration of the emplacement condition on the drift tunnel basement. Mechanical integrity of a percolated type casing was also examined its mechanical integrity.

Air-bearing unit, which originally apply to a flat/smooth surface, was modified to fit a curved surface of the drift tunnel. Two(2) units were aligned with two parallel lines, which estimate to be able to lift 12 tons, about two-fifth(2/5) of the total weight of full scale PEM. On the conducted transportation tests of the air-bearing units, considering the surface roughness of the drift tunnel, especially for its unevenness, capability and availability of the run-over such gaps were investigated. And effect of covering sheets which can improve the gapped surface into relatively smooth was also examined by using several candidate materials. Through these tests, combination of the covering sheets and the maximum available height difference were evaluated and identified. Also the maximum traction force to toe the loading was measured to design the air-bearing system.

Paper #13

DESIGN AND EMPLACEMENT OF BENTONITE BARRIERS

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Bentonite is a well known sealing and buffer material for waste disposals in hard rock conditions. Bentonite was also tested as a sealing material for rock salt formations. Under salt conditions the hydraulic conductivity is higher and the swelling pressure is lower then under freshwater conditions using the same bentonite dry density. Because rock salt conditions are more ambitious for bentonite, the new research results of material design and emplacement technology are very useful for applications under freshwater conditions.

Commercial available bentonites were tested by measurements of hydraulic conductivity and swelling pressure at various dry densities. The result is the required emplacement dry density for each bentonite. Following an industrial production process for high compacted bentonite materials with low initial water content was developed. Bentonite blocks for drift sealings and compacts / granules for bulk mixtures for shaft sealings have been successful tested in large scale in situ tests in salt mines.

Since 1998 about 500 t of bentonite blocks have been produced by the company Preiss-Daimler Industries GmbH – Feuerfestwerke Wetro. The blocks have a standard size of (250 x 125 x 62,5) mm. A proper fit of the blocks to the rock contour can be formed by sawing. The emplacement as dry brickwork is simple and reliable for the rough conditions in underground mines.

The recent shaft sealing systems consist of a binary mixture of air dry compacted bentonite-compacts and -granules (moisture content 7 – 10 %). This material design and the production technology were developed in cooperation with the K+S Group. Both components of the bulk mixture (compacts and granules) are now produced at the plant "Bergmannssegen-Hugo" (K+S Group). This material defines the actual best state of the art for bentonite sealing materials for long term stable shaft sealing systems, especially under difficult conditions like in salt mines. Since 2004 about 3000 t of bentonite compacts and high quality granules have been produced. For this reason 3 shaft sealing systems (Salzdetfurth mine) were realised. Furthermore 8 shaft sealings are intended to be build up to 2015 (closed mine "Merkers").

With compacted granulate of various grain size a wide range of bentonite-sand-mixtures with defined hydraulic conductivity and swelling pressure may be designed. The wide experience with this bentonite materials (blocks, various bulk mixtures) is very useful for applications as buffer and sealing materials for radioactive waste disposals.

Paper #14

THE SURFACE MOCK-UP KENTEX: ON THE THERMAL-HYDRO-MECHANICAL BEHAVIORS IN THE BUFFER OF A KOREAN HLW REPOSITORY

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The concept for a disposal of high-level wastes (HLW) in Korea is based upon a multi barrier system composed of engineered barriers and its surrounding plutonic rock (Kang et. al., 2002). A repository is constructed in a bedrock of several hundred meters in depth below the ground surface. The engineered barrier system (EBS), which is similar to the configuration considered by many other countries, consists of the HLW-encapsulting disposal container, the buffer between the container and the wall of a borehole, and the backfill in the inside space of the emplacement room, to isolate the HLW from the surrounding rock masses. Figure 1 shows the schematic picture of the EBS of the Korean reference disposal system (KRS) and the dimensions of the emplacement room and borehole were recommended from a preliminary design calculation (Kwon et al., 2001).

The engineering performance of a HLW repository is dependent, to a large extent, upon the thermal-hydro-mechanical (THM) behaviors in the buffer which are complicated by the processes such as the decay heat generated from the HLW, the ground water flowing in from the surrounding host rock, and the swelling pressure exerted by compacted bentonite. For this reason, the Korea Atomic Energy Research Institute (KAERI), to investigate the THM behaviors in the buffer of the KRS, planned large-scale tests to be conducted in two stages: a surface mock-up and then a full-scale "in situ" test. This paper deals with the surface mock-up called as “KENTEX” and presents the THM behaviors in the buffer which have been investigated from the KENTEX test.

The KENTEX is a third scale of the KRS (Figure 2). It consists of five major components: a heating system, a confining cylinder, a hydration tank, bentonite blocks, and sensors and instruments. The heating system measures 0.41 m in diameter and 0.68 m in length, which includes three heating elements in its inside, capable of supplying a thermal power of 1 kW each. The confining cylinder, which plays a role of the wall of a borehole excavated in the host rock, is a steel body with a length of 1.36 m and an inner diameter of 0.75 m, the inside wall of which is lined with layers of geotextile and the outside wall of which is mounted with 24 nozzles with two metal filters inserted into the inside of each, to uniformly apply the groundwater to the outer surface of the bentonite blocks (i.e., hydration surface). The bentonite blocks are fabricated of “Kyungju” bentonite [Lee, 2004] which is being considered as a candidate buffer of the KRS. Total of 176 blocks are emplaced in 16 sections of the confining cylinder. The bentonite blocks have an average value of 13 % of a water content and the average dry density of the bentonite blocks in the confining cylinder is 1500 kg/m^3. Total of sixty eight sensors are installed to measure the temperature, humidity (eventually, water content), and total pressure. And the heater control and data acquisition are operated automatically by means of a computer program.

The KENTEX test which started on May 30, 2005 is now under a successful operation. The T-H-M behaviour in the bentonite blocks may allow us to draw preliminary and quantitative conclutions. The temperature reached a steady state in a short time after the test start. The temperature was higher as it became closer to the heater, while it became lower as it was farther away from the heater. The water content had a higher value in a part close to the hydration surface than that in a heater part. The relative humidity data suggested that a hydration of the bentonite blocks might occur by different drying-wetting processes depending on their position. The total pressure was continuously increased by the evolution of the saturation front in the bentonite blocks and thereby the swelling pressure. There was also a contribution of a thermal expansion of the bentonite blocks near the heater and the capillary force in the dry bentonite blocks which the water did not reach from the hydration surface.

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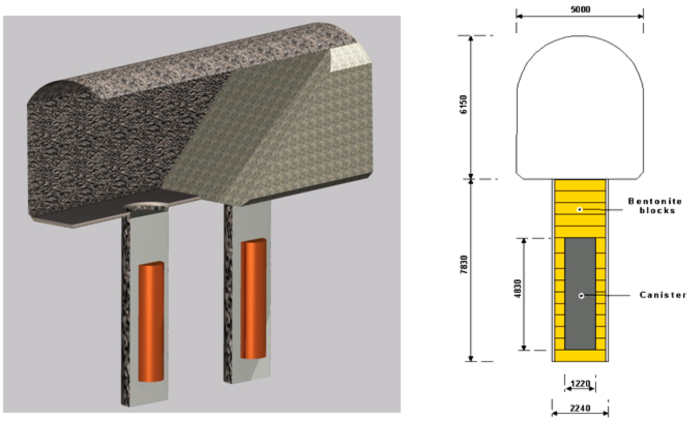

Figure 1. Schematic picture of the engineered barrier system of the KRS.

Figure 2 Picture of the surface mock-up KENTEX

Paper #15

THE EC/NEA ENGINEERED BARRIER SYSTEMS (EBS) PROJECT: LESSONS LEARNED ON DISPOSAL SYSTEM DESIGN AND DEMONSTRATION

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Deep underground disposal is the internationally favoured option for the long-term management of radioactive wastes containing significant amounts of long-lived radionuclides (e.g. spent fuel and high-level waste). Disposal systems for such wastes comprise a system of barriers that acts to isolate and contain the wastes. The barriers include the natural geological barrier and the Engineered Barrier System (EBS). The EBS may comprise several components, including the waste form, container, buffer, backfill, and tunnel seals and plugs. The presence of several barriers with complementary safety functions is designed to reinforce effective performance of the disposal system and to enhance confidence in its safety.

The OECD Nuclear Energy Agency and the European Commission have, over the last five years, sponsored a state-of-the-art report and a series of workshops to develop a greater understanding of how best to design, construct, test, model and assess the performance of engineered barrier systems, and how to integrate these aspects within the safety case for disposal.

Disposal system design and optimisation is necessarily an iterative process that involves a range of studies to:

- Define the requirements of the disposal system and the EBS, and take account of waste- and site-specific constraints that influence the design.
- Understand the materials of the EBS components and the processes that may affect them as the disposal system evolves.
- Model the behaviour, and assess the performance, of the EBS components and the whole disposal system under the range of conditions that may occur.
- Test and demonstrate that the EBS can be manufactured, constructed and installed satisfactorily.
- Provide reasonable assurance that the disposal system will provide an acceptable level of safety.

This paper presents examples from various disposal programmes and discusses lessons that may be drawn relating to disposal system design and the use of underground tests.

Many useful large-scale experiments have been conducted in underground laboratories that have allowed an assessment of the feasibility of methods for tunnel construction, waste package emplacement, buffer and backfill emplacement, tunnel seal construction etc. In general, these tests have been successful and have shown that the necessary techniques for manufacturing and installing EBS components are feasible and available. In some cases, tests have shown that designs or techniques need to be adjusted, or have enabled identification of the factors to be taken into account in future optimisation studies. Further trials of some methods are still required, particularly at the repository or industrial scale. Further experiments are also likely to be required to increase understanding of the long-term behaviour of the EBS after installation.

Paper #16

EXPERIMENTAL EVALUATION OF HYDRAULIC RESISTANCE OF COMPACTED BENTONITE/ BOOM CLAY INTERFACE

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Introduction

In the frame of in-situ PRACLAY Heater experiment to be performed in HADES URF in Mol, in order to study the feasibility hydraulically to cut-off the Excavation Damaged Zone (EDZ) and the RC of the disposal galleries with a seal in a horizontal drift (horizontal seal), an annular seal composed of compacted bentonite will be installed between the heated zone and the access gallery (PRACLAY seal test). According to numerical scoping calculations, heating until 80 °C will induce a pore pressure order of 3.0-3.5 MPa. As this horizontal seal should play a role to avoid the hydraulic shortcut to the access gallery, it is important to check the performance of the seal under these hydraulic and thermal conditions. The aim of the present work is to evaluate the hydraulic resistance of the interface between the compacted bentonite and the host rock Boom clay.

Material and experimental procedures

Compacted MX80 bentonite (LL = 520%, I_p = 474) was used. The provided bentonite has an initial water content of 10%; it was sieved at 2 mm and compacted to a dry density of 1.75 Mg/m^3. The two devices used are presented in Figure 2. The first one (Figure 2*a*) is a percolation cell (50-mm inner diameter). The soil specimen (5 mm high) is confined between two porous stones. The diameter of the soil specimen is smaller than the inner diameter of the cell, defining a gap between the soil and the cell wall. The water inlet of the cell is connected with a volume/pressure controller. The second device (Figure 2*b*) concerns a constant-volume oedometer (50-mm inner diameter) equipped with vertical and horizontal stress sensors. The diameter of the soil specimen is equally smaller than the inner diameter of the cell and a volume/pressure controller is connected to the lower base of the cell. The gap between the soil specimen and the cell wall represent the unavoidable technical gap between the bentonite ring and the gallery wall (Boom clay).

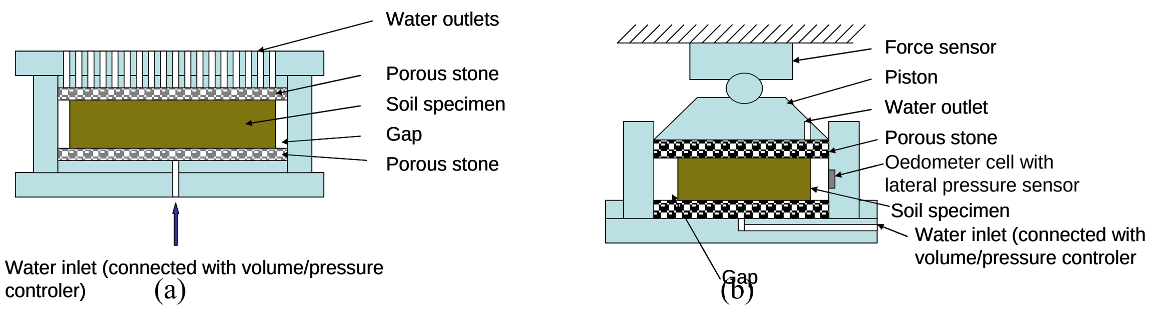

Figure 2. (a) Percolation cell; (b) Oedometer cell.

Experimental results

Figure 3a presents the results obtained from a test performed with the percolation cell (Figure 2a). The initial diameter of the specimen is 46.4 mm, which corresponds to a gap of 1.8 mm. During the first 2 h, water was injected with a constant rate of 1 mm^3/s and the pressure (*p*) was recorded. It can be observed that *p* increased to 150 kPa and decreased suddenly to 70 kPa at 0.5 h. After that, *p* increased continually until 25 MPa at 2.2 h. Figure 3b presents the results obtained from a test performed with the oedometer cell (Figure 2b). The initial diameter of the specimen is 46.0 mm (corresponding to a gap of 2.0 mm). When water was injected at a constant rate of 1 mm^3/s, *p* started to increase after 1 h while the vertical stress (σ_v) and the horizontal stress (σ_h) started to increase immediately. Figure 3b shows that *p* and σ_h increased and fell down suddenly several times and the breakthrough values increased with time. The water pressure was kept constant at 50 kPa from 7 h to 20 h and the water injection was then restarted. It can be observed that *p* increased rapidly to 10 MPa without any fall.

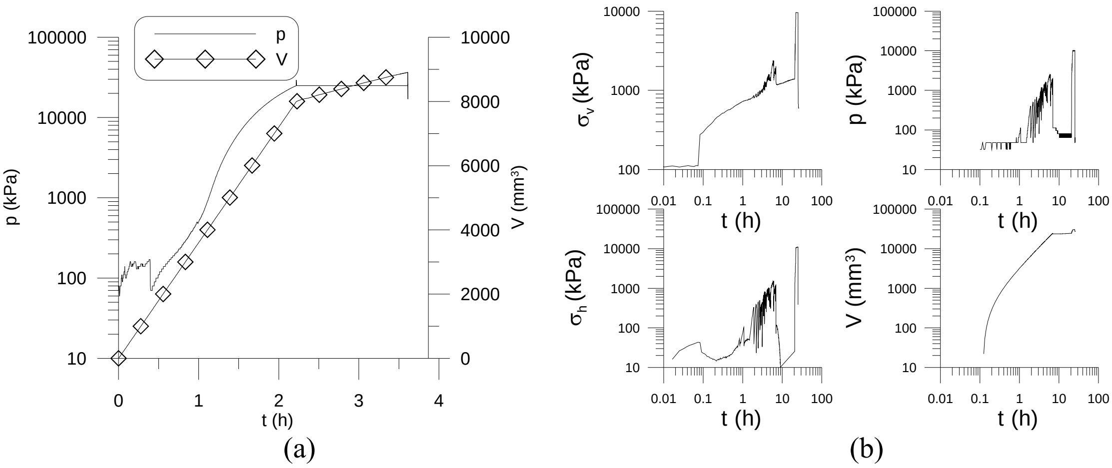

Figure 3. (a) Changes in volume and pressure in a test with the percolation cell; (b) Changes in volume, pressure, vertical and horizontal stresses in a test with the oedometer cell.

When starting the tests, water is injected in the cell and flows freely through the gap between the soil specimen and the cell wall. That explains why water pressure measured at the beginning was equal to zero. When the soil is in contact with water, it swells rapidly and reduces the gap, which increases the pressure of water. The pressure falls observed during the first hours took place when the water pressure reached the hydraulic resistance of the soil/wall interface. Marcial et al. (2006) observed equally that this breakthrough pressure increased with time. The increase of radial stress evidenced the rapid swell of the soil specimen. Teachavorasinskun and Visethrattana (2006) equally observed that the hydraulic resistance of compacted sand-bentonite mixture increases with the increases of overburden pressure.

It’s believed that the performance of seal depends highly on this interface characteristic. In the present work, several tests have been carried out using the percolation cell with three gap values (2.0, 1.8, and 1.6 mm) and at two temperatures (20 °C and 80 °C). In addition, several tests were performed using the oedometer cell with three values of gap (2.0, 1.8, and 1.6 mm) at 20 °C. All the tests show a hydraulic resistance of the soil/wall interface higher than 5 MPa at completion of soil swelling. This pressure is higher than the pore pressure estimated by numerical calculation in the PRACLAY Heater experiment (3.5 MPa). That confirms the performance of the compacted bentonite seal under the foreseen hydraulic and thermal conditions.

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Paper #17

DEVELOPMENT OF NON-INTRUSIVE MONITORING TECHNIQUES – ESDRED & TEM PROJECTS AT MONT TERRI AND THE GRIMSEL TEST SITE

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The EC Integrated Project, IP ESDRED (Engineering Studies and Demonstration of Repository Designs) was commissioned to establish a sound technical basis for demonstrating the safety of disposing of spent fuel and long-lived radioactive wastes in deep geological formations and to underpin the development of a common European view on the main issues related to the management and disposal of radioactive waste. The in situ development of non-intrusive monitoring techniques is included within the programme as an important component of the overall ESDRED programme.
Monitoring will play an important role in providing information to the repository operator and to society in general and to support decision-making about if and when, to move from one phase to the next. The challenges, when constructing engineered barrier systems (EBS) to isolate the waste, is the ability to monitor the waste and the barriers, once isolated. Conventional monitoring systems depend upon wires or cables to transfer information from the monitoring sensors outside the barriers. Monitoring sensors also have a limited lifetime and new sensors cannot be emplaced without disturbing the barrier. The development of non-intrusive monitoring systems which do not rely upon “hard-wired” connection, thus providing the opportunity for continued monitoring after isolation.

The ESDRED partners developed a programme utilising cross-hole seismic tomography to monitor an experimental demonstration by Nagra at Mont Terri Underground Research Laboratory (URL). The work programme includes PhD studies to conduct full seismic waveform analysis and to develop algorithms to address natural anisotropy in the Opalinus clay at Mont Terri.

Following on from the ESDRED developments, some of ESDRED partner organisations identified a further opportunity for developing in situ monitoring techniques utilising the construction and testing programme of a low pH shotcrete plug being constructed in granite at the Grimsel Test Site, this being part of another study, funded within the ESDRED programme. The TEM (Testing and Evaluation of Monitoring Techniques) Project at Grimsel was initiated to provide a unique opportunity for simultaneous comparison of 3 monitoring methods - wired signal transmission from the EBS, wireless data transmission using magneto-inductive techniques, and observation through non-intrusive geophysical techniques. The novelty of TEM is that the application of each of these techniques is under a realistic condition, where a bentonite barrier is emplaced in a 3.5m diameter tunnel and isolated with an approximately 4m long low-pH shotcrete plug.

The report describes the work programmes and some of the results from the ESDRED and TEM programmes at both sites.

Paper #18

IMPLICATIONS OF THE USE OF LOW-PH CEMENTITIOUS MATERIALS IN HIGH ACTIVITY RADIOACTIVE WASTE REPOSITORIES

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One of the most accepted engineering construction concepts for high radioactive nuclear waste of underground repositories considers the use of low pH cementitious materials, in order to avoid the formation of an alkaline plume fluid which perturbs one of the engineered barriers of the repository, the bentonite. The accepted solution to maintain the bentonite stability, which is function of the pH, is to develop cementitious materials that generate pore waters with pH $\leq$ 11, because the corrosion velocity of the clay is significantly reduced below this value. The IETcc-CSIC has focused the research activity on low-pH cementitious materials using two cements: Ordinary Portland Cements (OPC) and Calcium Aluminates Cements (CAC). In both cases, the achievement of a low-pH environment implies the use of high content of mineral admixtures to prepare the binder. Obviously, the inclusion of high contents of mineral admixtures in the cement formulation modifies most of the concrete “standard” properties and the microstructure of the obtained cement products.

When designing a concrete based on low-pH binders, not only the functional requirements have to be reached but also the modifications of the basic properties of the concrete must be taken into account. Besides, due to the location and the long service life of this type of products, their durability properties must be also guaranteed.

This paper deals with the procedure followed in the design of a specific application of low pH cements; for instance, the shotcrete plug fabrication. The challenge of this type of use (shotcreting) is more complex taking into account that requires the employment of additives that must be compatible with the concrete mixture. Furthermore, their effectiveness must be assured without increase the pH above the admissible levels. Therefore, their compatibility with admixtures is tested in the present work.

The compliance of the requirements for a shotcrete plug was evaluated at laboratory scale and real scale, making different trials with the shotcrete. Parameters such us the workability of the basic concrete, the quality of fresh and hardened basic concretes, the pumpability of the mix and the properties of the hardened shotcrete were tested. Furthermore, some parameters related to the durability of the cementitious materials are being analyzed, such us the resistance of cementitious materials to long term water aggresion and the influence of the high contents of mineral admixtures in the shrinkage of these cementitious materials.

The results show that it is possible to obtain a low-pH shotcrete appropriate to be used in the underground facilities of a nuclear waste repository.

Paper #19

FULL SCALE DEMONSTRATION OF SHOTCRETE SEALING PLUG UNDER REALISTIC WORKING CONDITIONS

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Module #4 of the IP ESDRED aims at the demonstration of the technical feasibility, at an industrial scale, for the closure of deep geological repositories for the disposal of high activity wastes in compliance with requirements on operational safety, retrievability and monitoring. Both the construction and closure of a deep geological repository will require the use of big amounts (up to thousands of tons) of cementitious materials for the construction of auxiliary structures needed for the operation of the repositories, in particular temporary or permanent plugs.

One main concern for the use of concrete in radioactive waste repositories comes from the potential chemical interaction with the disposal components, which can undergo physicochemical transformations and changes in their radionuclide confinement properties. The reduction of the pH of the concrete is a long-term safety issue to avoid this interaction.

Another key issue addressed in relation to the feasibility of the construction of concrete sealing plugs in a real repository is the introduction of the shotcreting technique. This technique provides a very good contact between concrete and rock, filling all voids and holes, even at the roof part. In addition, a good quality shotcrete has a lower porosity and permeability than standard concrete, and can be easily reinforced using fibres if needed. Another practical advantage is that forms are not needed, and therefore the plug can be constructed very quickly, which is a critical factor in a real repository, in cases when a fast temporary or permanent closure of a gallery or drift is required. In terms of safety, shotcrete arms and robots make possible to perform this operation in a semi-automated mode, with the operator situated at some distance from the working face.

Although the utilization and performance of standard shotcrete in conventional construction works is well known, there is no experience in either the workability or the performance of shotcrete formulated to obtain a final low-pH product and, therefore, testing of this specific material under realistic conditions is needed.

The research activities carried out in this sense within the IP ESDRED have provided a low-pH concrete formulation suitable of being shotcreted. In a series of field tests, this concrete fulfilled the established functional requirements in terms of low pH, long distance pumpability and sprayability. Thereafter, a short low-pH shotcrete plug was successfully constructed and tested (load test to determine its bearing capacity) at the Äspö URL. The feasibility of the construction in accordance to the established requirements was demonstrated, and the plug behaved as expected, showing a good enduring capacity under mechanical load. The results from the test provided valuable information on the mechanical behaviour of confined granite-shotcrete interfaces, which has been used for improving the plug design calculations.

As a final step, a full-scale low-pH shotcrete plug has been constructed in the Grimsel URL to check the feasibility and performance of this type of plug construction under realistic conditions – swelling pressure exerted by the saturated bentonite and the local hydraulic gradient. The construction was successfully carried out in winter time, with no access by road to the Laboratory, and producing the concrete ‘in situ', within a restricted space, what demonstrated its feasibility in the toughest conditions.

The proposed paper is mainly focused on the construction of the full-scale tests and the results obtained.

Paper #20

FINAL REPOSITORY FOR SPENT NUCLEAR FUEL IN GRANITE – THE KBS-3V CONCEPT IN SWEDEN AND FINLAND

Stig Pettersson, Bengt Lönnerberg
SKB

Both Sweden and Finland has advanced plans for design, construction and operation of the final repositories for direct disposal of spent nuclear fuel. Both countries have the same type of host rock – granite. They are also investigating alternative concept for disposal, vertical or horizontal disposal of the canisters with encapsulated spent nuclear fuel, normally called KBS-3V or the KBS-3H disposal concept.

The development of the KBS-3V concept started around 1980 and is the reference method for both SKB in Sweden and Posiva in Finland. However, extensive development work is ongoing since 2001 with KBS-3H in order to bring that concept to the same maturity as KBS-3V. This presentation deals with the design and operation of the KBS-3V based on the work done within Sweden and SKB but the development is Finland is identical and it is a close cooperation between SKB in Sweden and Posiva in Finland.

In Sweden, the site investigation for location of the repository has been concentrated on two sites, in the Oskarshamn area, about 350 km south of Stockholm, and the Forsmark area, about 180 km north of Stockholm. For information it can be mentioned that Finland plans to locate their repository in the vicinity of the Olkiluoto nuclear power plant site, about 300 km north of Helsinki.

The site investigation is completed and the selection of site is scheduled to mid 2009 and sending in the application for location and construction of the repository is scheduled to end 2009. After receiving all necessary permits, construction time and commissioning will take about 7 to 8 years and operation is expected to start about 2020.

The KBS-3 system is based on a multi barrier concept and the work with compiling the design requirements for the underground part of the deep repository has been ongoing for some time within the SKB organisation. Today the design requirements for the underground part are documented in a big number of reports that has been produced by specialists and working groups over rather a long time period.
For each barrier the following will be determined during the development work:

- Specification
- Design determining parameters
- Dependency on other parts of the repository barriers
- Design determining situations during construction and operation
- Design determining processes after closure of the repository

For the design and optimisation of the different parts of the disposal concept the following has to be considered:

- Long term safety after closure
- Safety during operation
- Safety during construction
- Environment issues
- Technology and feasibility
- Costs
- Possibility to retrieve the canisters with spent fuel

Over the years, a number of generic studies of the layout of the operational area(s) above ground and underground facilities of the repository have been performed. Different access routes from the ground level to the repository level at 500 m below ground have also been investigated. The access routes studied are mainly by shafts only or a ramp access for the heavy and bulky transports in combination with different service shafts. Further, a ramp alternative could be arranged as a spiral or as a straight ramp in combination with service shafts.

The selected reference alternative with a combination of shafts and a spiral ramp is illustrated in Figure 1. In the evaluation process of the access routes it is important to have a good knowledge of the cost and time schedule for shaft sinking down to the repository level in order to compare this with a ramp access. The time needed for the excavation of the shaft down to the repository level will be about 18 months shorter comparing with ramp. Other factors that may influence the selection between access routes are constructability, operational safety and long term safety.

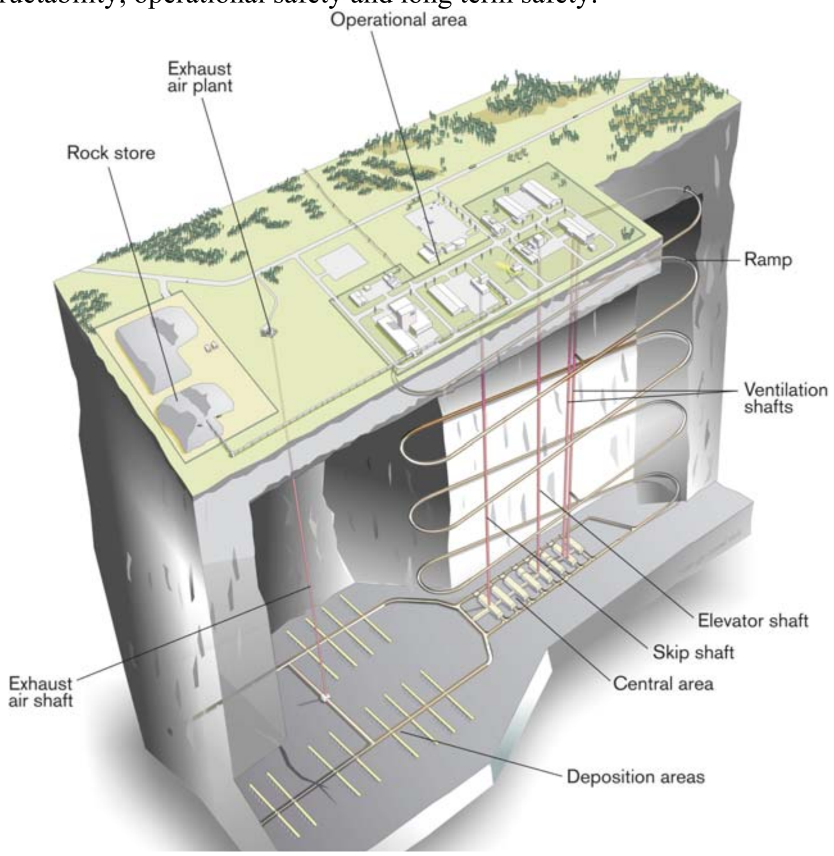


Figure 1. Illustration of the deep repository and the main functions.

In the reference case, the deposition tunnels are separated by 40 metres and the spacing between the deposition holes is six metres. The latter distance is determined by the need to limit the temperature on the canister surface.

The operational area comprises a terminal building for receiving transport casks containing encapsulated fuel, a production building for preparation of buffer and backfill material, a supply building for electric power supply, buildings for offices and personnel and a restaurant building.

When all canisters in one deposition drift have been emplaced the backfilling and final sealing of the drift can start. It has been discussed to carry out stepwise backfilling but the reference is that we do the backfilling in one step. The backfilling will be done with pre-compacted blocks of swelling clay and with some additional pellets for filling the void between the blocks and the rock wall and the roof of the drift. The principle for emplacement of the blocks is still not decided but different methods and equipment will be tested.

The backfilling of the about 300 m long disposal drifts will be a challenge. The speed for backfilling must be high as we must avoid piping and problem with water. We plan to take down about 350 – 400 tons of backfilling material per 24 hours. The backfilling of one drift is estimated to take 10 – 12 full weeks working all days in the week and around the clock.

The transport logistic for the backfill material from the production building in the operational area on ground down to the repository level and out into the drift and feeding to the emplacement equipment as well as filling the void between the blocks and the walls and the roof will not be an easy task.

When the backfilling is completed it is time for construction of the sealing plug. SKB is investigating different designs of this plug. The plug will be a cast low-pH concrete plug but it is still open if it will be a short reinforced plug or a longer taped plug or if we need a bentonite plug between the backfill and the concrete construction.

The operational safety of the repository will be presented in the "Preliminary Safety Assessment Report" and the long term safety (Safety Case) will be presented in a report called Safety Report Site (SR-Site). A preliminary version of the SR-Site has already been presented and has been subject to an international review. /SR-Can/

Paper #21

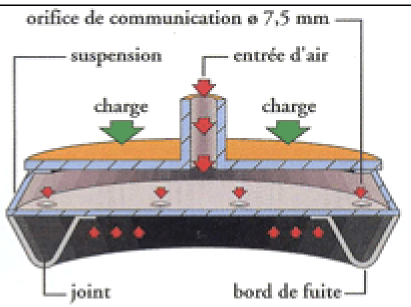
APPLICATION OF THE AIR/WATER CUSHION TECHNOLOGY FOR HANDLING OF HEAVY WASTE PACKAGES IN SWEDEN AND FRANCE

Stig Pettersson, Erik Thurner
SKB

Disposal of radioactive waste in deep repositories requires handling and emplacement of heavy waste packages. The weight of these packages can be in the order of 25 – 50 ton and they have to be handled in limited spaces as it is expensive to excavate and backfill openings in the repository. This will require special technology that can meet the requirements for safe operation and still not require excavation of large openings. Air/water cushion systems are used world wide in the industry for moving of heavy components. However, this technology for handling of heavy waste packages in narrow drifts has not been used earlier.

This paper will describe the work done by SKB and Posiva within the framework of the ESDRED Project. ESDRED is co-funded by the European Commission as part of the sixth Euratom Research and Training Framework Programme (FP6) on nuclear energy (2002/2006). SKB in Sweden and Posiva in Finland have since 2001 performed a joint project for developing the KBS-3H disposal concept and in 2004 it became part of the ESDRED Project.

The basic principal of the air cushion technology is that components as shown in Figure 1 are arranged as a pallet with normally four air cushions. Depending on the load the number of cushions will be adapted. With normal air pressure of 0.2 MPa the lifting capacity per cushion is in the range of 1 to 20 tons. It also possible to use cushions that can work with an air pressure of up to 0.4 MPa and the lifting capacity will then increase to 2 to 40 tons.



This illustrates the principal of the individual air cushion. The rubber cushion is fixed to a support plate and air/water will fill the rubber cushion and lift the load. The cushion with the load will then float on a film of air.
Inlet for air or water/media
Parking support plate

Figure 1. Illustration of the design of an air cushion.

The waste package in the Swedish/Finish KBS-3H concept consist of a perforated steel container that contains the copper canister with spent nuclear fuel and the buffer material. The weight of the container, normally called Super Container (SC), is in the order of 46 tonnes with a diameter of 1.765m and a length of 5.57m. Between each SC we need a spacer block in order to get a predetermined distance between each SC for reducing the heat load and temperature on the canister surface and buffer material. The super containers are placed in 300 m long drifts with a spacing of 25 to 40 m between each drift. The distance between the drifts and each SC is determined by the thermal conductivity of the host rock and the decay heat of the fuel.

The deposition equipment for KBS-3H became part of the ESDRED project and is partly financed by EC within the 6th Frame Programme from February 2004. However, before that SKB had carried out prototype tests using both air and water cushion in order to prove the feasibility to use this technology also for cylindrical objects and surfaces.

Figure 2. Set-up of equipment at the test site at Äspö HRL, level -220m. The Super Container is inside the transport tube with the radiation shielding gates open. The control room is on the left side on this photo.

The Factory Acceptance Tests (FAT) was done at the CNIM factory in France in February 2006 before delivery of the equipment to Äspö. However, during FAT all tests could not be performed with real conditions and at the initial start of Site Acceptance Test (SAT) it was discovered that we could not control the balance of the SC and it even derailed during the testing and preventive actions had to be taken.

To ensure a proper function of the guides on the slide plate the lifting height must be limited. It was therefore decided to change the original water cushions to new water cushions with less lifting height and also with less sensitivity to load variations.

The tests performed so far have shown that the emplacement equipment is operating effectively for transport and deposition of Super Containers with a weight of 45 tons in horizontal drifts.. Further tests are however required to verify the availability and the reliability of the equipment for a longer period of time.

It has also been concluded that the water cushion technique is sensitive to load variations and the system is also sensitive to the alignment in the set-up between the transport tube for the Super Container, the deposition drift and the start tube for the deposition machine.

The full paper will outline the experience and actions taken to improve the operation of the equipment.

Paper #22

DEVELOPMENT OF EXCAVATION TECHNOLOGIES AT THE CANADIAN UNDERGROUND RESEARCH LABORATORY

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Many technological advances have been made at Canada's Underground Research Laboratory (URL) in the fields of excavation methodology, excavation design, the modelling of rock response and the characterization of the severity or rock damage around excavations. Several countries, Canada being among them, are developing concepts for disposal of used fuel from power generating nuclear reactors. As in underground mining operations, the disposal facilities will require excavation of many kilometres of shafts and tunnels through the host rock mass. The need to maintain stability of excavations and safety to workers will be of paramount importance. Also, radioactive waste repository excavations will ultimately need to be backfilled and sealed to maintain stability and minimize any potential for migration of radionuclides, should they escape their disposal containers. The method used to excavate the tunnels and shafts, and the rock damage that occurs due to excavation, will greatly affect the performance characteristics of repository sealing systems. The underground rock mechanics and geotechnical engineering work performed at the URL has led to the development of excavation technologies that reduce rock damage in subsurface excavations.

At the URL, damage around the shaft and underground tunnels constructed in hard granitic rock was minimized through controlled blasting techniques. These techniques reduce the amount of overbreak and fracturing outside the design tunnel perimeter by reducing the charge-energy in the blastholes close to the final walls. The techniques essentially eliminated overbreak and reduced fracturing around the shaft and tunnels located in low-to-intermediate stressed, moderately fractured rock on the 240-m-deep level and more hightly stressed, sparsely fractured rock on the 420-m-deep level. Controlled blasting results in better quality excavation with a smaller blast-induced Excavation-Damaged Zone (EDZ). Optimizing excavation shapes to reduce the compressive stress in the rock was also a successful technique for reducing the EDZ.

This paper discusses the excavation methods used to construct the Canadian Underground Research Laboratory and their application in planning for the construction of similar underground laboratories and repositories for radioactive wastes.

Paper #23

EMPLACEMENT TECHNOLOGY FOR THE DIRECT DISPOSAL OF SPENT FUEL INTO DEEP VERTICAL BOREHOLES

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In the early sixties it was decided to investigate salt formations on its suitability to host heat generating radioactive waste in Germany. In the reference repository concept consequently the emplacement of vitrified waste canisters in deep vertical boreholes inside a salt mine was considered whereas spent fuel should be disposed of in self shielding casks (type POLLUX) in horizontal drifts. The POLLUX casks, 65 t heavy carbon steel casks, will be laid down on the floor of a horizontal drift in one of the disposal zones to be constructed in the salt dome at the 870 m level. The space between casks and drift walls will be backfilled with crushed salt. The transport, the handling und the emplacement of POLLUX casks were subject of successfully performed demonstration and in situ tests in the nineties and resulted in an adjustment of the atomic law.

The borehole disposal concept comprises the emplacement of unshielded canisters with vitrified HLW in boreholes with a diameter of 60cm and a depth of up to 300 m. In order to facilitate the fast encapsulation of the waste canister by the host rock (rock salt), no lining of the boreholes is planned. With regard to harmonize and optimize the emplacement technology for both categories of packages (vitrified waste and spent fuel) alternatives were developed. In this context the borehole emplacement technique for consolidated spent fuel as already foreseen for high-level reprocessing waste was reconsidered. This review resulted in the design of a new disposal package, a fuel rod canister (type “BSK 3”), and an appropriate modified transport and emplacement technology. This concept (called BSK 3-concept) provides the following optimization possibilities:

- A new steel canister of the same diameter (43 cm) as the standardized HLW canisters applied for high-level waste and compacted technological waste from reprocessing abroad can be filled with fuel rods of 3 PWR or 9 BWR fuel assemblies.
- The standardized canister diameter provides the possibility to apply a common transfer and handling technique for both categories.
- The BSK3-canister is tightly closed by welding and designed to withstand the petrostatic pressure at the emplacement level.
- The residual heat generation of a canister loaded with fuel rods burned up to 50 GWd/tHM will enable it’s emplacement in a salt repository already after about 3 to 7 years following reactor unloading of the fuel assemblies. This has been evidenced by thermal calculations.

The BSK 3-concept, therefore, may provide a considerable reduction of necessary effort in terms of time and costs. Consequently a research program was launched to develop and test the necessary technical components and to transfer this emplacement technology into a new state of the art. The R&D work is going to be performed as a part of the 6th Framework program of the European Union. The BSK 3 concept there is an essential element of an Integrated Project, which is dealing with the development and demonstration of repository relevant transport and emplacement techniques. The IP ESDRED will be performed by a consortium of 13 partners from 9 European countries. Financial support is provided by waste management agencies and by the European Commission. The activities performed by DBE TECHNOLOGY GmbH are co financed by the German Project Management Agency Karlsruhe and the manufacturing of the components by the German nuclear industry represented by GNS. To develop the BSK 3 emplacement technology to technical maturity it was decided to carry out a comprehensive series of demonstrations tests in a 1:1 scale. The target of these tests are to prove the feasibility and safety of the technical system and it’s different components. In addition, the results of these tests should provide all information required for licensing this new back end technology, satisfying the legal requirements for a German final repository.

The main components are:

- a BSK 3 canister, containing rods of 3 PWR or 9 BWR fuel elements,
- a transfer cask for internal transport of BSK 3 canisters,
- an emplacement device,
- a borehole lock and
- a transport unit consisting of a transport cart and a battery driven mine-locomotive for rail bound transport in the repository.

The main components of the emplacement system are displayed in figure 1 below.

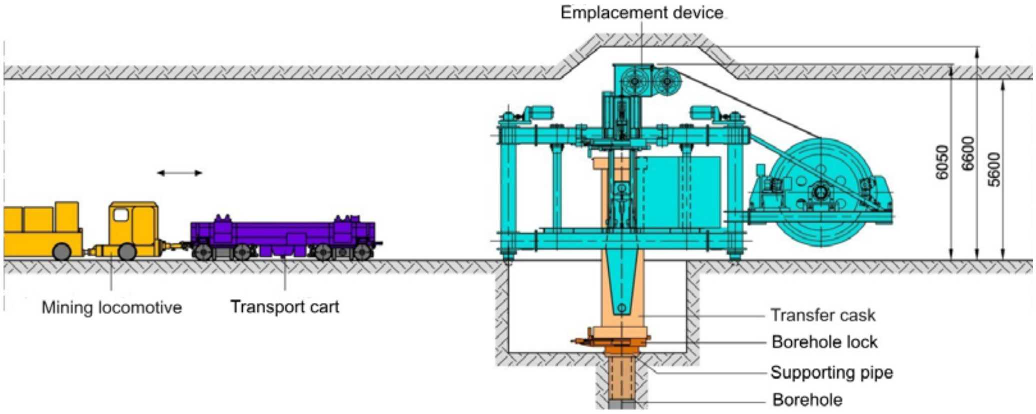

Figure 1: System for the emplacement of spent fuel canisters into deep vertical boreholes

All these components were designed in detail in the last few years and are going to be manufactured from summer 2007 to spring 2008. The test bench will be a former turbine hall of a power station in Lower-Saxony. The reconstruction of this hall to a test bench will be done in parallel to the component manufacturing. Thus, the planned series of demonstration and reliability tests with the emplacement system will be performed during the year 2008.

The results of the design and manufacturing of the components of the BSK 3 emplacement system will be shown as well as first results and photographs from the demonstration tests.

Paper #24

9+ YEARS DISPOSAL EXPERIENCE AT THE WASTE ISOLATION PILOT PLANT

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With almost a decade of operating experience, the Waste Isolation Pilot Plant (WIPP) has established an enviable record by clearly demonstrating that a deep geologic repository for unconditioned radioactive waste in rock salt can be operated safely and in compliance with very complex regulations. WIPP has disposed of contact-handled transuranic (TRU) waste since 1999 and remote-handled TRU waste since 2007. Emplacement methods range from directly stacking unshielded 0.21- 4.5 m3 containers inside disposal rooms to remotely inserting highly radioactive 0.89 m3 canisters into horizontally drilled holes (shield plugs placed in front of canisters protect workers inside active disposal rooms). More than 100 000 waste containers have been emplaced, and one-third of WIPP's authorized repository capacity of 175 000m3 has already been consumed.

Principal surface operations are conducted in the waste handling building, which is divided into CH and RH waste handling areas. Four vertical shafts extend from the surface to the disposal horizon, 655m below the surface in a 1000m thick sequence of Permian bedded salt. The waste disposal area of about 0.5km2 is divided into ten panels, each consisting of seven rooms. Vertical closure (creep) rates in disposal rooms range up to 10cm per year. While one panel is being filled with waste, the next one is being mined. Mined salt is raised to the surface in the salt shaft, and waste is lowered down the waste shaft. Both of these shafts also serve as principal access for personnel and materials. Underground ventilation is divided into separate flow paths, allowing simultaneous mining and disposal. A filter building near the exhaust shaft provides the capability to filter the exhaust air (in reduced ventilation mode) through HEPA filters before release to the atmosphere.

WIPP operations have not exposed employees or the public to radiation doses beyond natural background variability. They consistently meet or exceed regulatory standards and expectations. Process improvements continuously reduce cycle times and costs. During the past few years, regulators have approved configuration changes that eliminated some unnecessary tests and activities. Many more could be targeted to further reduce vulnerability, while maintaining and even enhancing safety.

While WIPP is licensed to dispose of only defense-related TRU waste, past experiments and performance assessments have shown that heat-generating high-activity waste could also be safely isolated in salt (and without prior vitrification). Thus, beyond its current restrictions, WIPP helps pave the way toward permanent isolation of all categories of radioactive waste.

Paper #25

EMPLACEMENT FEASIBILITY FOR A MULTI-TIER, EXPANDED CAPACITY REPOSITORY AT YUCCA MOUNTAIN, NEVADA

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A geological repository at Yucca Mountain has been proposed for the disposal of spent fuel from the US commercial reactors and other radioactive waste. A legislative capacity of 70,000 MTHM has been set by the Nuclear Waste Policy Act of 1982, including 63,000 MTHM of commercial spent nuclear fuel (CSNF), the projected amount of CSNF that will be produced by about 2014. Policy issues remain as to how to handle waste that is generated beyond 2014 from a growing nuclear industry in the US. The Electric Power Research Institute (EPRI) is independently evaluating the technical, rather than legislative, limit of CSNF that could be safely disposed at Yucca Mountain. Geological, thermal management, safety and cost factors have been recently evaluated by EPRI (2006; 2007) for grouped emplacement drifts and/or a multi-tier repository. EPRI's evaluation of emplacement feasibility for a multi-tier concept is described here.

Expanded capacity concepts as envisioned for Yucca Mountain (EPRI, 2006; 2007) assume excavation of one or two additional levels of drifts parallel to (right-hand side of Figure 1) or above and/or below the original drift excavations (left-hand side of Figure 1). For the latter multi-tier concept each 'tier' or 'level' would essentially replicate the original layer with a 30-m separation between tiers. This arrangement essentially doubles or triples the capacity of the repository for a two- or three-tier design, respectively.

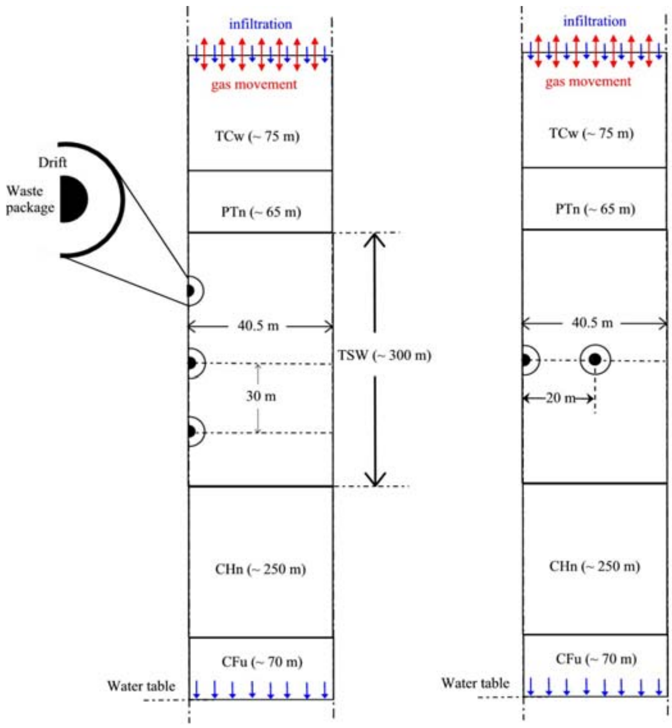

Figure 1. Illustration of the conceptual expanded capacity repositories (EPRI, 2006)

The main issues that affect the feasibility of expanded capacity design are; (i) ventilation requirements; (ii) radiation hazards; (iii) thermal and thermo-mechanical constraints.

***Ventilation*:** The repository design involves waste packages mounted in close proximity to each other in 600-m long drifts that remain open and actively ventilated for at least 50~100 years. Analyses, conservatively assuming that all three repository levels operate simultaneously, indicate no technological obstacles in meeting ventilation requirements for sustained simultaneous operation based on current industrial/ mining practices.

Radiation Hazards: The presence of the waste-filled drifts 30m above or below other drifts might cause elevated radiation exposures to workers during drift excavation and subsequent operations. It is well established that a 1-m or so cylinder of concrete, as used in dry cask surface storage at nuclear facilities, is sufficient to reduce radiation emanating from the concrete to safe levels. Radiation intensity declines exponentially with thickness of the 'container'. Thus, radiation levels in the adjacent tunnels will be negligible.

Thermal effects: The natural rock temperature at Yucca Mountain is 24ºC. Analysis indicates that heat generated by radioactive decay of the waste-filled drifts will cause a maximum temperature increase of <60ºC above ambient (i.e., 84º C) at a radial distance of 30m from the filled drift. Calculations indicate that the readily achievable removal of 75 watts /m of drift length is sufficient to bring the temperature in the drift to ambient 24ºC. Removal of heat generated during the rock excavation process by tunnel boring machine (TBM), a more severe challenge, has been proven in deep mines in South Africa. The ventilation requirements to drive supplementary drifts at 30-m from a waste-filled drift and maintain a 24°C drift environment will be no different from the requirements when driving waste emplacement drifts already excavated.

Thermo-mechanical effects: Increase in the rock temperature due to heat generated in the waste - filled drifts will produce thermal stresses in the rock that will be superimposed on the pre-existing rock stresses. The increased stresses will decline in proportion to the square of the distance, expressed as a ratio of the drift radius (2.75m), from the heated drift. For example, at 25 m (i.e. 9 radii) spacing, the increase in stress at the supplementary tunnels will be $(1/9)^2$ or approximately 1%.

Based on existing engineering practice and experience, it is concluded that is technically feasible to double or triple the CSNF disposal capacity of a Yucca Mountain using multi-tier or grouped emplacement drift design concepts.

References

EPRI, 2006, *Room at the Mountain: Analysis of the Maximum Disposal Capacity for Commercial Spent Fuel in a Yucca Mountain Repository.TR-1013523,* Electric Power Research Institute, Palo Alto, CA.

EPRI, 2007, *Room at the Mountain: Analysis of the Maximum Disposal Capacity for Commercial Spent Fuel in a Yucca Mountain Repository.TR-1015046,* Electric Power Research Institute, Palo Alto, CA.

Paper #26

PROPOSED DEEP GEOLOGIC REPOSITORY FOR LOW AND INTERMEDIATE LEVEL RADIOACTIVE WASTE AT BRUCE NUCLEAR SITE, TIVERTON, ONTARIO

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A Deep Geologic Repository (DGR) for the long-term management of operational Low and Intermediate Level Radioactive Waste (L&ILW) is being proposed by Ontario Power Generation at the 900-ha Bruce Nuclear site. The Bruce Nuclear site is located approximately 225 kilometres northwest of Toronto near Tiverton, Ontario. The project is currently undergoing environmental assessment. This paper describes the current concept for Ontario Power Generation's proposed DGR. The underground repository concept is comprised of horizontally-excavated emplacement rooms arranged in panels with access provided via two vertical concrete-lined shafts. The emplacement rooms would be constructed at a depth of about 680 m within limestone. This limestone formation is laterally extensive and is directly overlain by 200 m of low permeability shale. The low-permeability diffusion-controlled geosphere immediately surrounding the repository will assure long-term isolation of the L&ILW.

L&ILW will be retrieved from nearby above-ground storage structures and then transferred to the DGR. L&ILW will also be shipped directly from the nuclear generating stations to the DGR. In this concept, most waste packages are retrieved and transferred "as is" with shielding added, as necessary, to protect workers. The waste packages are lowered by hoist to the repository horizon and then transferred by forklift or, in the case of large and heavy packages, by rail car to emplacement rooms. Waste packages are stacked within emplacement rooms by forklift or gantry crane and, when full, the rooms are isolated by closure walls. It is expected that the repository will be open for at least 50 years to receive L&ILW from the operation of Ontario's nuclear reactors. When filled with waste and after receipt of all necessary regulatory approvals, the repository will be sealed by placing low permeability concrete and clay-based plugs in each shaft. A preliminary safety assessment indicates that predicted peak radiological impacts of the sealed repository will be many orders of magnitude below regulatory criteria.

POSTERS

Poster #1

INTRODUCTION TO ESDRED

Jean-Michel Bosgiraud, Wolf K Seidler
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This presentation consists of 4 pre-existing posters (each 60W X 80L) which have been used since the beginning of the Project to provide a brief overview of the Project for the benefit of those not previously familiar with it. Given the theme of the conference, which is largely based on ESDRED work, it seems essential that this information be prominently displayed within the front of the poster area. The first poster provides some of the key data such as who the participants are, the time frame, the budget and of course it highlights the objectives. The next 3 posters provide an overview of the 4 Technical Modules, providing in each case the conceptual designs on which the ensuing work was planned to be based.

The titles of the 4 posters are as follows:

POSTER 1	**ESDRED – Engineering Studies and Demonstration of Repository Designs**
POSTER 2	**Module 1 – Buffer Construction Technology for Horizontal Disposal Concepts**
POSTER 3	**Module 2 – Waste Canister Transfer and Emplacement Technology for Horizontal and Vertical Disposal Concepts**
POSTER 4a	**Module 3 – Heavy Load (15 to 50 metric tons) Emplacement Technology for Horizontal Disposal Concepts**
POSTER 4b	**Module 4 – Low pH Cement for Shotcrete Sealing Plug Construction Technology**

These posters do not belong directly to any one of the Conference themes in particular but rather they belong to the Conference in general in the sense that they relate to all the themes.

Jean- Michel Bosgiraud (Andra), the Project Coordinator's technical representative will be present at the poster display during the designated hours.

Poster #2

SELF SEALING BARRIERS OF CLAY/MINERAL MIXTURES THE SB PROJECT AT THE MONT TERRI ROCK LABORATORY

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Since about two decades, geological clay formations are investigated on their suitability to host a repository for high-level radioactive waste. Gas generated by anaerobic corrosion of waste containers or by radiolysis of water in the host formation may lead to the development of high gas pressures in the repository near field which in turn can lead to fracturing of the host rock if the gas pressure exceeds the least principal stress or tensile strength of the rock.

To avoid such a scenario two technical possibilities can be considered. One is to provide an adequate gas storage volume in the backfill of the disposal rooms and another one would be to seal the disposal rooms with an optimized sealing material which allows the gases to migrate continuously out of the disposal room thereby keeping the gas pressure at a safe low level.

In this regard it has been found that moderately compacted clay/sand mixtures may represent a reasonable alternative to highly compacted bentonite so far envisaged as buffer in various multi-barrier repository concepts. In contrast to highly compacted buffers clay/sand mixtures exhibit a comparably low gas entry/break through pressure in the saturated state while providing an adequate sealing potential due to swelling of the clay minerals in consequence of water uptake from the host rock. The evolution of high gas pressure in the repository near-field due to corrosion of the waste containers will thus be avoided while possible migration of radionuclides from the waste matrix through the buffer will be diffusion controlled.

In 2003, GRS started the SB-project which comprises laboratory investigations and mockup testing to investigate the principle suitability of clay/sand mixtures as sealing material as well as full-scale in-situ experiments at the Mont Terri Underground Rock Laboratory to test and demonstrate the advantageous sealing properties of moderately compacted clay/sand-mixtures under representative in-situ conditions.

The poster will summarize the results obtained from the laboratory investigations as well as such obtained so far in the in-situ experiments which were taken into operation in late 2005 and which are expected to run for some further period of time.

Poster #3

EMPLACEMENT TESTS WITH GRANULAR BENTONITE

H.P. Weber[1], M. Plötze[2]

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2. ETH Zurich, Institute for Geotechnical Engineering, CH-8093 Zurich, Switzerland (michael.ploetze@igt.baug.ethz.ch)

Introduction

The objectives of NAGRA's granular bentonite emplacement testing within the ESDRED Project are as follows:

- Testing and demonstrating of suitable granular buffer installation techniques on a full scale in surface facilities;
- Verification if the requirements are fulfilled and optimization of different parameters.

The general objectives and the know-how from previous experiments lead to an ambitious project specific target value for the emplacement dry density of 1500 kg/m^3.

The technological project ESDRED: "Engineering Studies and Demonstration of repository Designs" is co-funded by the European Commission (EC) as part of the sixth Euratom research and training Framework Programme (FP6) on nuclear energy (2002-2006).

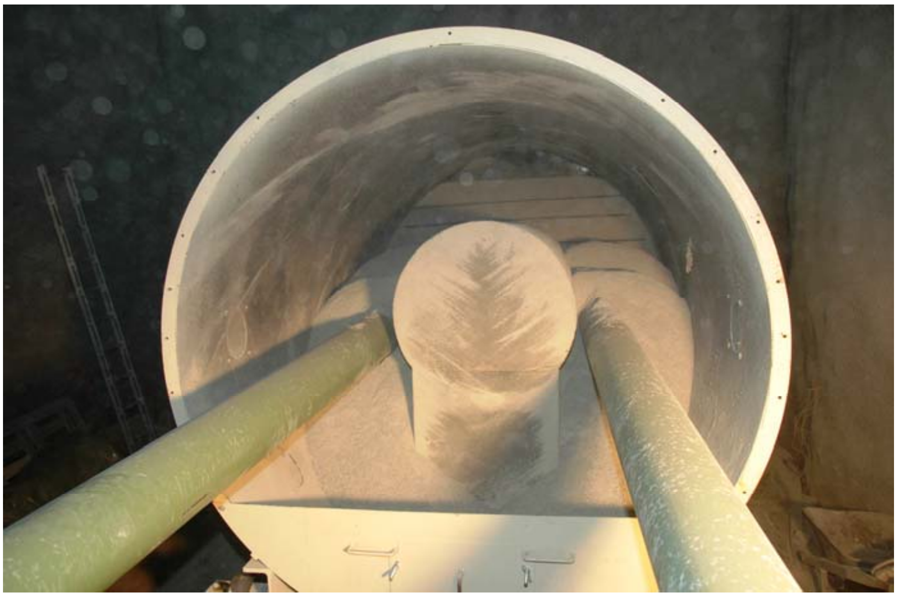
Experimental concept

In previous experiments Nagra executed various tests with different bentonite types of buffer material. For this project we decided to use Wyoming bentonite. The sodium-bentonite MX-80 was delivered in a conditioned, slightly granulated state to improve the pourability and pelletizing. The production of the granulated material was done in the Rettenmaier facilities in Holzmühle, Germany. During pelletization of the bentonite, an increase of the bulk grain dry density from 1.17 g/cm^3 to 2.10 g/cm^3 with simultaneous halving of porosity was achieved.

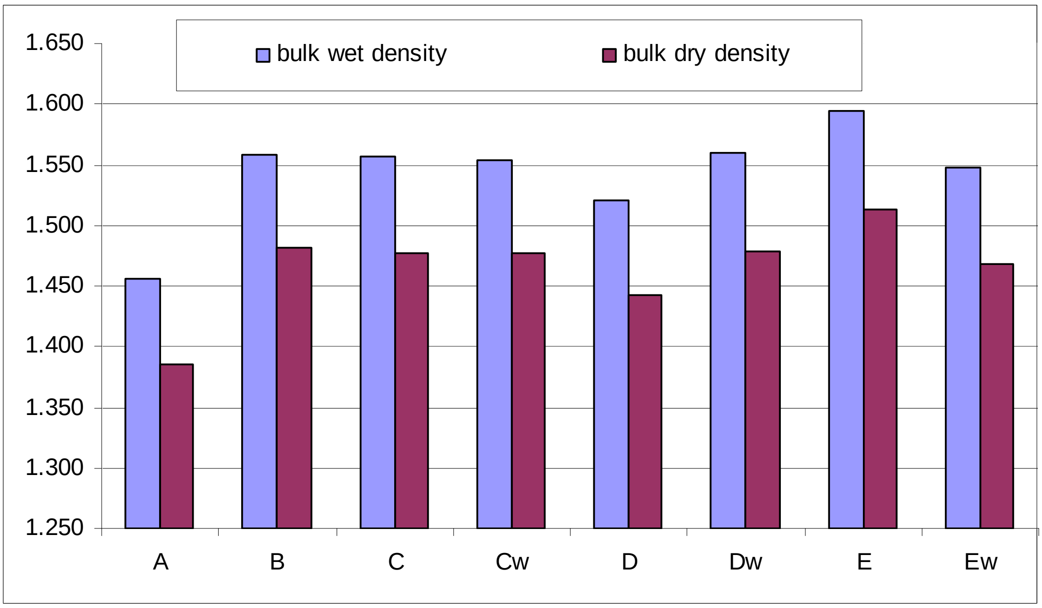
The built twin auger system to emplace the buffer material has a total length of about 9 m and a weight of 1350 kg. The length of the two auger casings is 7.0 m, the diameter of the tubes are 0.2 m. The feed rate can be controlled by the auger turning speed. The rotating screwing motion of the auger moves the materials to the end of the outer casing tube where the material either falls off the end of the auger freely or can push the material out into the existing bentonite mass. The maximum feed rate is actually 7 m^3 of granular bentonite material per hour. The maximum filling volume of the steel cylinder was about 7 m^3, resp. about 10 tons of emplaced granular bentonite material.

As part of the testing, a comprehensive laboratory testing program was executed to investigate the performance of the system. Bentonite samples were taken sequentially from the "big bags" before filling for grain size distribution, water content and bulk density. After every backfill operation of the model, the following parameters were investigated:

- global bulk wet density
- particle size distribution measurements of the granular bentonite material before emplacement and after emplacement sampled at selected points at the outer surface of the steel model
- water content measurements of the granular bentonite material before and after emplacement
- in addition, other properties of the granular material as mineralogy, swelling pressure, thermal conductivity, etc. were determined in the laboratory of ETH Zurich.



Results
After each emplacement test, the bulk wet density of the whole silo emplacement was measured. The bulk density is the net weight of buffer material over total volume of the test section. The bulk densities of the granular bentonite material show only small changes for different admixtures of fine granular bentonite and coarse granular bentonite material. The water content increased only slightly during the test runs from 5.0 % to 5.8 %. The results are reflected in the following graph:



A	100 % coarse rounded granular material, embedded in two layers
B	92 % coarse, 8 % fine, two layers
C	85 % coarse, 15 % fine, two layers
Cw	85 % coarse, 15 % fine, two layers
D	70 % coarse, 30 % fine, two layers
Dw	70 % coarse, 30 % fine, repeat run, two layers
E	64 % coarse, 28 % fine, 8 % briquettes, two layers
Ew	64 % coarse, 28 % fine, 8 % briquettes, repeat run, only one layer

Poster #5

LPH SHOTCRETE FOR ROCK SUPPORT ON OPALINUS CLAY: SPRAYING TESTS IN THE HAGERBACH TEST GALLERY

Thomas Fries[1], H.P. Weber[1], V. Wetzig[2]

[1] National Cooperative for the Disposal of Radioactive Waste (Nagra) CH-5430 Wettingen / Switzerland

[2] Test Gallery Hagerbach Ltd., CH-7320 Sargans / Switzerland

Within the ESDRED[1] project Module 4, Work package 3.2: Low pH Shotcrete for Rock Support, NAGRA has been continuing recipe development and demonstration tests previously carried out in Sweden. NAGRA selected the VSH Hagerbach Test Gallery[2] (www.hagerbach.ch) for carrying out these tests.

Low pH concrete has a different binder composition compared to normal concrete, which has a pH of around 13. The low pH is achieved by a binder composition where 40 wt% of the Portland cement is replaced with microsilica. This composition means that the hydrates have a different structure and will result in a groundwater pH of around 11 on the long term.

The objectives of the tests at Hagerbach were to further develop a suitable sprayed concrete mixture with low pH and good pumpability that meets the functional requirements for rock support shotcrete. In terms of aggregates and Portland cement, the wet mix also had to consist of components commonly used for the purpose of rock support in Switzerland. The tests had to demonstrate that this low pH shotcrete is suitable for rock support in Opalinus Clay sedimentary rock.

The tests were carried out in three steps in autumn 2006:

- In the first step (phase 1), the Swedish base recipe, corresponding to a 10-2 mix from the field tests at Äspö (Lagerblad, 2006), was adjusted without changing the cement/microsilica ratio and type of set accelerator in order to achieve a pumpable wet mix. This wet mix was sprayed onto test panels (700 mm x 700 mm) using different amounts of alkali-free set accelerator. Various tests with fresh and hardened concrete were carried out.
- In the second step (phase 2), low pH shotcrete was applied to, and tested on, test panels prepared with approximately 4.7 m2 Opalinus Clay samples.
- In the third step (phase 3), a large-scale field test was performed by spraying 6 m3 wet mix onto approximately 20 m2 of unsupported horseshoe-shaped underground excavation at the Hagerbach underground facilities.

The poster will summarise the results and discusses some of the main findings in the context of using shotcrete as rock support in underground radioactive waste repositories. Recommendations for further steps are also given.

In order to assess the performance of low pH shotcrete as compared with that of ordinary shotcrete, the results are compared with a typical wet mix shotcrete (reference mix) for underground excavation support purposes.

However, the tests at Hagerbach demonstrated that low pH shotcrete can be produced which, in terms of concrete (fresh and hardened state) and application (pumping, spraying) properties, is comparable with ordinary wet mix sprayed concrete for rock support and therefore is a feasible alternative for rock support in radioactive waste repositories.

---

[1] ESDRED **E**ngineering **S**tudies and **D**emonstration of **RE**pository **D**esigns

[2] VSH Hagerbach Test Gallery Ltd. Rheinstrasse 4, P.O. Box 64, CH-7320 Sargans, Switzerland

Fig. 1: Fresh concrete test (flow table test) of LPH shotcrete (wet mix).

Fig. 2: Spraying tests on Opalinus Clay panels.

Fig. 3: Horseshoe tunnel profile for the large-scale field test.

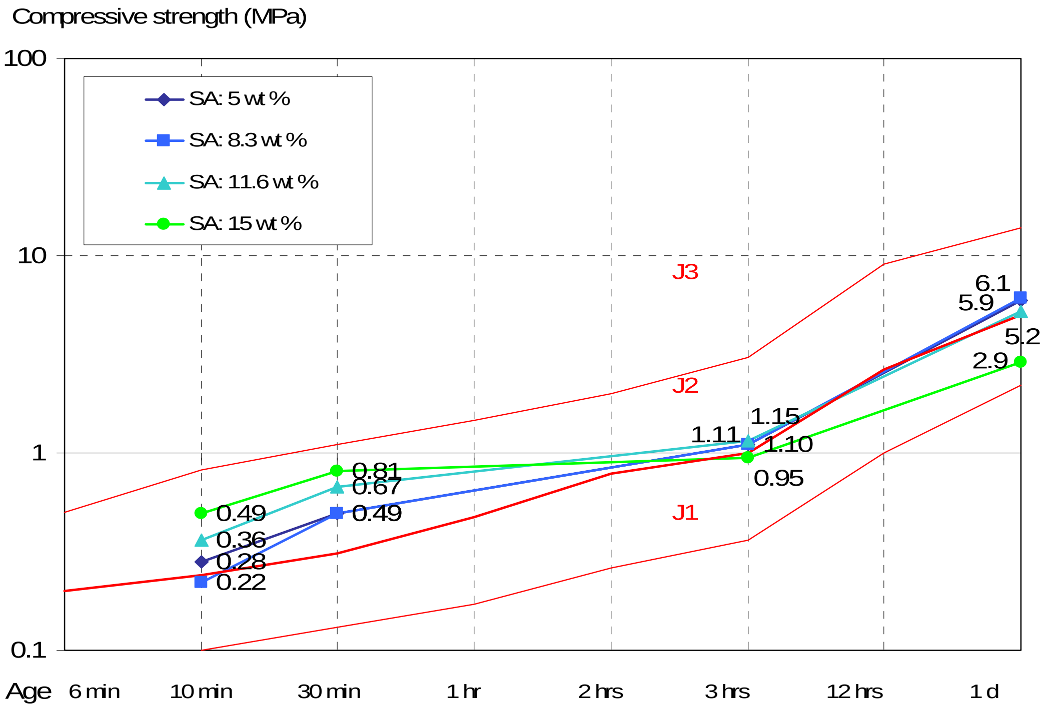


Fig. 4: Measured early compressive strength at different ages for varying amounts of set accelerator (SA).

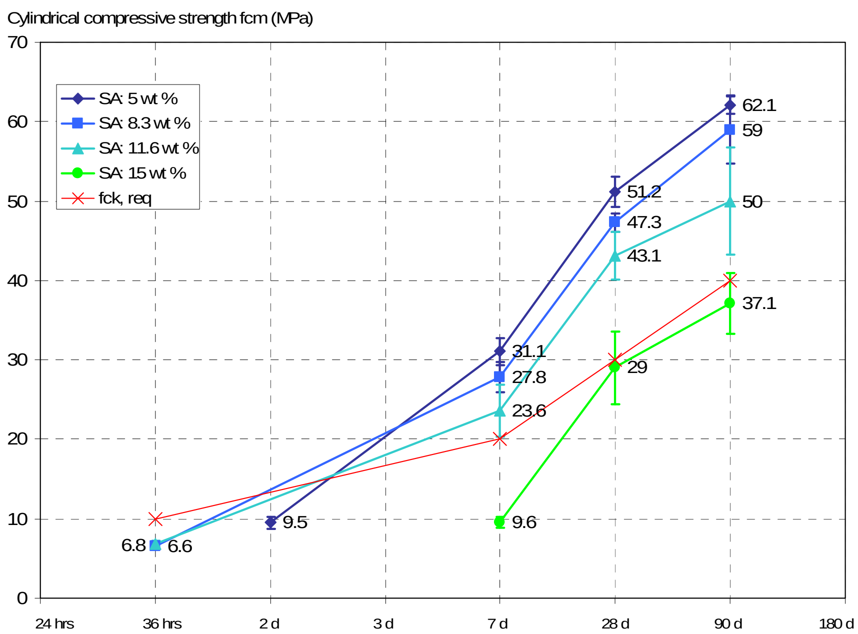


Fig. 5: Measured cylindrical compressive strength at different ages and for different dosages of set accelerator.

Poster #6

CAVERN DISPOSAL CONCEPTS FOR HLW/SF: ASSURING OPERATIONAL PRACTICALITY AND SAFETY WITH MAXIMUM PROGRAMME FLEXIBILITY

Ian G. McKinley[1], Mick Apted[2], Hiroyuki Umeki[3] and Hideki Kawamura[4]

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Most conventional engineered barrier system (EBS) designs for HLW/SF repositories are based on concepts developed in the 1970s and 1980s that assured feasibility with high margins of safety, in order to convince national decision makers to proceed with geological disposal despite technological uncertainties. In the interval since the advent of such “feasibility designs”, significant progress has been made in reducing technological uncertainties, which has lead to a growing awareness of other, equally important uncertainties in operational implementation and challenges regarding social acceptance in many new, emerging national repository programs. As indicated by the NUMO repository concept catalogue study (NUMO, 2004), there are advantages in reassessing how previous designs can be modified and optimised in the light of improved system understanding, allowing a robust EBS to be flexibly implemented to meet nation-specific and site-specific conditions.

Full-scale emplacement demonstrations, particularly those carried out underground, have highlighted many of the practical issues to be addressed; e.g., handling of compacted bentonite in humid conditions, use of concrete for support infrastructure, remote handling of heavy radioactive packages in confined conditions, quality inspection, monitoring / ease of retrieval of emplaced packages and institutional control.

The CAvern REtrievable (CARE) concept reduces or avoids such issues by emplacement of HLW or SF within multi-purpose transportation / storage / disposal casks in large ventilated caverns at a depth of several hundred metres. The facility allows the caverns to serve as inspectable stores for an extended period of time (up to a few hundred years) until a decision is made to close them. At this point the caverns are backfilled and sealed as a final repository, effectively with the same safety case components as conventional “feasibility designs”. In terms of operational practicality and safety, the CARE concept has the advantages of large dimensions of the emplacement caverns, utilising well-established technology for waste package handling and ensuring ease of monitoring (even inspection) and retrieval of the waste inventory for an extended period of time. The long open period decreases the heat loading at closure, allowing significantly higher emplacement densities with reduced costs. Although caverns will include cement-based ground support, carbonation during the extended open phase will reduce subsequent post-closure issues of hyperalkaline leachates, which can be further reduced by using multi-layer buffer/backfills. Post-closure safety assessment indicates that performance will be at least as good as conventional designs – but improved models are needed for more rigorous analysis.

NUMO, 2004. Development of repository concepts for volunteer siting environments, NUMO-TR-04-03.

Poster #7

SELECTION AND THM CHARACTERISATION OF THE BUFFER MATERIAL

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Context

One of the main objectives of Module 1 of the European ESDRED Project is to manufacture large rings and to use them as engineered barriers in order to demonstrate the technical feasibility of the current concept for radioactive-waste disposal [1]. As the entity responsible for Module 1, the French National Radioactive Waste Management Agency (Andra) hired a pool of other organisations, including the French Atomic Energy Commission (CEA) and two companies, *Minéraux et produits chimiques* (MPC) and *SEGULA Ingénierie* with a view to manufacturing and testing the rings. In that context, the CEA Study Laboratory on the Behaviour of Concretes and Clays (*Laboratoire d'étude du comportement des bétons et argiles* – LECBA) was in charge of defining and characterising a suitable swelling-clay material (bentonite) capable of fulfilling very precise specifications.

Experimental work

This poster aims at describing methodology and technology used to define the formulation of an adapted material and compiles the results of the characterisation studies. The chosen material consists in a mixture of 70% bentonite (MX80-type Wyoming sodic montmorillonite) and 30% sand. The parametric study concerning the compaction capability of that mixture helped to determine a density and water-content domain meeting the prescribed specifications.
Thermo-hydro-mechanical measurements were taken on the compacted material at both the laboratory (figure 1) and industrial scales. A formulation was proposed with a water content of the powder in the order of 12% and a compaction pressure of 80 MPa. The resulting compacted material has a bulk density of 2,200 kg/m3, a dry density of 1,960 kg/m3, a dry density of clay in the mixture of 1,770 kg/m3, a thermal conductivity close to 1.5 W/(m·K) and a residual swelling pressure equal to 2 MPa after a 20% expansion of its volume [2].

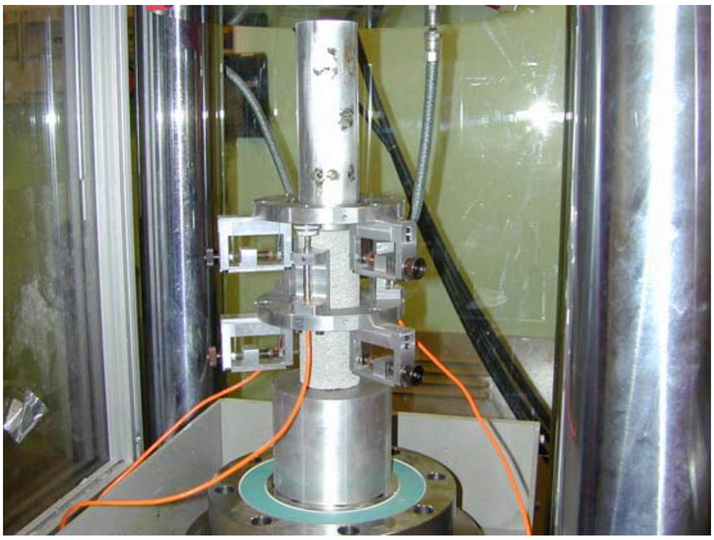

Figure 1 Device used for Young modulus measurement on compacted cylinder

The formulation proposed is certainly not the only option. Naturally, the decision to select that formulation does not take only into account specifications in terms of the THM performances of the material, but also its capability to be produced industrially, including, the resistance of the mould, homogeneity, post-swelling and the conservation of the compacted rings.

The various tests conducted to the laboratory thus largely contributed to the definition and the design of the various components necessary to the realisation of the rings on scale 1. This industrial stage is the subject of a second poster proposed on the ESDRED project module 1.

References

[1] J.-M. BOSGIRAUD., "ESDRED Project-Module 1. Scope of Work. Design, Fabrication, Assembly, Handling and Packaging of Buffer Rings". ANDRA/DP/TE, C CC ASTE 04-0520/B, April 2004.

[2] C. GATABIN, G. TOUZÉ, Pierre BILLAUD, Christophe IMBERT, William GUILLOT. ESDRED Project-Module 1. Selection and THM Characterisation of the Buffer Material. Technical Report. E.NT.0GME.05.0005/B, 2006 November.

Poster #8

ESDRED PROJECT, MODULE 1
DESIGN, FABRICATION, ASSEMBLY, HANDLING AND PACKAGING OF BUFFER RINGS

C. Gatabin[1], **J-L. Guyot**[2], **S. Resnikow**[3],

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3 MPC, Zone portuaire, 62, route du Hazay, 78520 Limay, France (serge.resnikow@mpcfr.com)

Context

One of the main objectives of Module 1 of the European ESDRED Project is to manufacture large buffer rings and to use them as engineered barriers in order to demonstrate the technical feasibility of the current concept for radioactive-waste disposal [1]. As the entity responsible for Module 1, the French National Radioactive Waste Management Agency (Andra) hired a pool of other organisations, including the French Atomic Energy Commission (CEA) and two companies, *Minéraux et produits chimiques* (MPC) and *SEGULA Ingénierie* with a view to manufacturing and testing the rings. That pool of companies, named GME, had the role to develop all the industrial process necessary to the realisation of buffer rings and discs on scale 1, as well as related means such as transport, the handling of the rings, the assembly by sets of 4, the packaging of buffer ring assemblies, all the procedures, etc...

Course of the Project

That work proceeded over 2 years from April 2005 at April 2007. This poster has the aim of describing the advance and the various stages of the realisation by focusing the positive points and the encountered difficulties.

The first stage was the design and the fabrication of the buffer rings and discs, starting with a selected buffer formulation [2]. This item includes the following points:

- Choice of a press capable of large pressing area and 40.000 tonnes force (figure 1),
- Design and realisation of a mould adapted to the press and associated procedures,
- Design and realisation of a handling system to remove rings out of the mould and associated procedures,

The second stage was the fabrication of rings and discs. This work was carried out in factory INTERFORGE (Issoire, France) in 3 shifts of approximately 30 hours on the whole. In this operation 13 rings and 2 discs were manufactured. A ring weighs 4 tons with a 0.5 m thickness and with a 2.3 m outer diameter. The central hole has a diameter of 0.69 m. The dry density of rings and discs reaches 1.98 Mg/m^3. Those dimensions are in respect with the expected specifications (figure 2).

Then the buffer rings and discs were transported to Limay, close to Paris, where they were assembled by set of 4 and packaged for long term conservation.

Figure 1: Filling the mould with ESDRED mixture powder

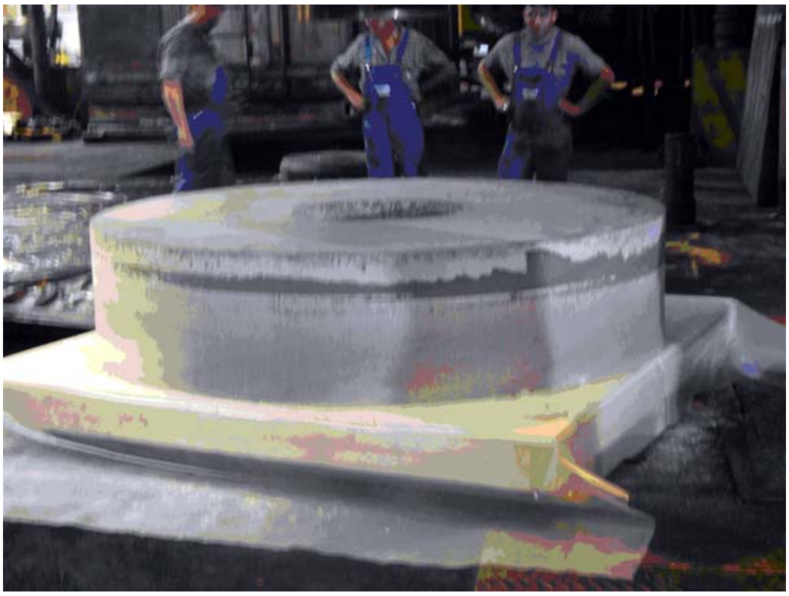

Figure 2: A buffer ring after removing of the mould

References

[1] J.-M. BOSGIRAUD., "ESDRED Project-Module 1. Scope of Work. Design, Fabrication, Assembly, Handling and Packaging of Buffer Rings". ANDRA/DP/TE, C CC ASTE 04-0520/B, April 2004.

[2] C. GATABIN, G. TOUZÉ, Pierre BILLAUD, Christophe IMBERT, William GUILLOT. ESDRED Project-Module 1. Selection and THM Characterisation of the Buffer Material. Technical Report. E.NT.0GME.05.0005/B, 2006 November.

Poster #10

THE KBS-3H CONCEPT – HORIZONTAL DISPOSAL

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In 2001 SKB prepared an R&D programme for the KBS-3H concept (horizontal deposition) as an alternative disposal method to the base case KBS-3V (vertical deposition) concept. Both methods are based on the so-called KBS-3 concept with multi barrier systems but the orientation of the emplaced canister differs.

In the autumn of 2001, the boards of SKB and Posiva decided to start the common R&D programme for the alternative KBS-3H disposal method. The purpose of the programme, carried out over the period 2002-2007, is to find out if KBS-3H can be regarded as a suitable alternative to the KBS-3V concept. In order to evaluate the potential of the concept and to determine if the R&D work related to the KBS-3V concept should continue the following main activities were planned:

- Design and fabrication of the deposition equipment.
- Development of the KBS-3H design and of all components to be emplaced in the drift
- Development of the related repository layout by taking into account the Olkiluoto geological site model.
- Excavation of two horizontal drifts at the Äspö Hard Rock Laboratory (HRL) to be used for demonstration of the deposition equipment and related components.
- Compiling of a preliminary Safety Case for the KBS-3H concept with Olkiluoto as a reference site.
-

The KBS-3H deposition equipment is since February 2004 a part of the technological project ESDRED: "Engineering Studies and Demonstration of repository Designs" and is co-funded by the European Commission (EC) as part of the sixth Euratom Research and Training Framework Programme (FP6) for nuclear energy (2002-2006)".

The final repository for spent fuel is planned to be constructed at a depth of about 400-500 metres in crystalline bedrock, in both Sweden and in Finland. In the KBS-3H repository concept, multiple canisters containing spent fuel are emplaced in approximately 300 m long deposition drifts. These are excavated with a small positive inclination starting at the mouth i.e. at the transport tunnel. Each canister, with its surrounding bentonite buffer and a perforated steel shell, called Super Container, is assembled in a handling cell, located in a cavern in the central area of the repository level, prior to emplacement in the drift. Each Super Container is placed between two compacted bentonite distance blocks (spacers). In addition to providing the appropriate spacing to meet thermal loading requirements, the other main purpose of the distance blocks in the KBS-3H system is to separate the Super Containers from each other hydraulically, thus preventing the possibility of pathways for flow and erosion of bentonite along the drift.

The transportation of the Super Container in the drift is based on lifting the container with a water cushion pallet and then advancing in 1.5 metre increments by alternating the advance of the container and of the slide plate located underneath the pallet. The same machine will also be used for the transportation of the distance blocks.

The spent fuel canisters and the buffer material are identical in both the KBS-3V and the KBS-3H concepts but the introduction of the Super Container into the disposal drift and the process of emplacing multiple Super Containers in a single drift are new features compared to the KBS-3V concept. Therefore the function and behaviour of the Engineered Barrier System (EBS), after emplacement, is different and thus the long-term performance (safety case) for the two concepts will also be different.

The main activities for the project listed above have now been completed and the evaluation and reporting of the work done up to 2007 is ongoing. The general experiences are very promising. However, more testing and development work needs to be done and both SKB and Posiva have decided that they will jointly continue the development work with KBS-3H during 2008 and 2009.

Poster #11

PERFORMANCE ASSESSMENT STUDIES EVALUATE THE LONG-TERM SAFETY OF THE DISPOSAL SYSTEM OF RADIOACTIVE WASTE FROM CERNAVODA NPP

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Especially during the last ten years, a part of Romanian research program "Management of Radioactive Waste and Spent Fuel" was focused mainly on applicative research for the design of near-surface disposal facility, which intends to accommodate the low and intermediate radioactive waste generated from Cernavoda NPP. In this frame, our contribution was at the acquisition of technical data for the characterization of the future disposal facility.

In the present, the project of the disposal facility located on the Saligny site near Cernavoda NPP, must be licensed.

As regards to the safe disposal, the location of final disposal, the Saligny site, has been characterized through five geological formations which contain potential routes for transport of radionuclide released from disposal facility, in the receiving zones (potential receiving zones), into liquid and gaseous phases.

The technical characteristics of the disposal facility were adapted at the Romanian disposal concept using the reference data from IAEA technical report [1]. Input parameters which characterized from physical and chemical point of view the disposal system, were partially taken from literature.

The performance assessment studies, which follows the preliminary design development phases and the selection, describes how the source term is affected by the infiltration of water through the disposal facility, degradation process of engineering barriers (reflected in the distribution coefficient values) and solubility limit.

The studies regard the evaluation of the source term, sensitivity and uncertainty analysis provide the information on "how" and "why" were evaluated, following:

- Radiological safety assessment of near-surface disposal facility on Saligny site;
- Complexity standard assessment of the Engineering Barriers Systems (EBS);
- Identification of the elements which must be elaborated for the increase of the disposal safety and the necessity for new technical data for the characterization of the disposal facility.

In the frame of the performance assessment, sensitivity and uncertainty analyses for the Saligny disposal facility have been conducted, consulting associate activity for the three phases corresponding to the disposal:

- Operational period;
- Institutionalized control;
- Post-operative control.

In source term evaluating study and of the sensitivity and uncertainty analysis, we took into consideration radionuclide Co-60, Cs-137, H-3 and C-14, these representing the most relevant radionuclide generated by the operation of the nuclear power plant.

The sensitivity and uncertainty analysis applies to Saligny disposal facility, correlated with special parameters that are influencing the release of radionuclide from repository, was conducted by using a computer code which yielded results allowing the characterization of the disposal facility at the end of the operational period and to eliminate the uncertainties[2].

REFERENCES

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Poster #12

CRITICALITY AND DOSE RATE MODELLING FOR COPPER DISPOSAL CANISTER WITH SPENT RBMK-1500 NUCLEAR FUEL

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There is only one nuclear power plant in Lithuania - the Ignalina NPP. After final shutdown of INPP Unit 1 in 2004 and Unit 2 in 2009 total amount of spent nuclear fuel will be approximately 22 thousands of fuel assemblies. All these assemblies should be stored about 50 years and after that disposed of. International consensus exists that spent nuclear fuel (SNF) and long-lived high level radioactive wastes are best disposed of in geological repositories using a system of engineered and natural barriers. During 2002-2005 investigations on possibilities to dispose of SNF in Lithuania were performed with support of Swedish experts.

A generic repository concept for RBMK-1500 SNF disposal was developed. For crystalline rock in Lithuania the KBS-3 concept developed by SKB for the disposal of PWR and BWR spent nuclear fuel in Sweden is adopted because of the very comprehensive knowledge of the performance of such a repository that has emerged from SKB's extensive studies, including methods for handling and application of waste, buffer and backfill. The preliminary characteristics of the copper canister for RBMK-1500 SNF are: diameter 1050 mm, length 4070 mm, and it can be loaded with 32 RBMK-1500 SNF half-assemblies.

The estimation of the criticality and dose rates of the disposal canister with spent RBMK-1500 nuclear fuel is one of the most important issues for the safe SNF handling and temporal storage at disposal site, and of cause disposal.

Criticality and dose rate calculation models for copper disposal canister with RBMK-1500 SNF was developed using SCALE computer codes system. For the modeling it was assumed that canister is fully loaded and contains 32 RBMK-1500 fuel half-assemblies. Assumption that the fuel half-assemblies contain only fresh, undepleted fuel (no credit for burnup) with 2.8% ^{235}U enrichment was made for criticality calculations. For dose rate calculations it was assumed that 2.8% ^{235}U enriched fuel has burn-up of 30 MWd/kgU, irradiation time 3 years and cooling time 50 years.

In the paper the results of criticality variation with density of the water in the canister will be presented. Also modeling results of dose rate on of canister surface dose rate and the dose rate at different distance from the canister will be presented.

Poster #13

ESTIMATING THE DUAL PERMEABILITY PARAMETERS OF THE FRACTURED ENVIRONMENT OF THE RICHARD NUCLEAR WASTE REPOSITORY (RAWRA, CZECH REPUBLIC)

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When modeling the flow and contaminant transport in the surroundings of underground nuclear storage facilities, we are in most cases dealing with a fractured rock environment. Dangerous nuclear wastes stored for long-term periods are usually deposited either in a deep rock, fractured environment or in a heavy clay environment. According to IAEA requirements, such wastes must be permanently stored, from the time that the material is considered as a waste product, in a so-called deep underground storage facility, which has been subjected to a detailed study. The conditions for such a structure are precisely laid down in the IAEA documentation. However, although these requirements were already introduced in 1993, and according to the EU rules full implementation is planned from 2018, none of the recent producers have been able to fulfill the conditions. According to an edict of the Czech government from 2001, implementation of this solution is planned from 2065, which means that nuclear wastes produced recently in this country are stored under different conditions, in so-called sub-surface storage facilities. One such facility is the Richard Nuclear Waste Repository, at Litoměřice. For security reasons, no further handling of the stored wastes (which have already been located in their position for decades) is under consideration. Investigations into the security of this facility have recently been carried out. Previous complex studies of the facility had not taken into account the preferential flow through the fracture network.

The mainstream approach used in groundwater flow and contaminant transport modeling in a fractured rock environment is based on so-called equivalent porous media. The whole complex of fractures together with the matrix permeability is substituted by a porous medium with the equivalent Darcy's K value. The variability of the velocity field due to flow through a system of fractures is neglected. However, the calculation of the 3D finite element is fast and stable.

An approach generally referred to as dual permeability can give more accurate results. According to this model, the water flow with contaminant transport is considered to flow partly through the rock matrix with its known K values and partly through a so-called fast domain with K values related to some fast media, e.g. sand. The main problem is to make a correct estimation of the coefficients of the dual media. The K values reflect the rate of the fast and slow media and other coefficients describing the transit between the fast and slow domain. To evaluate these characteristics, a model fracture network on a sample (10x10m) was determined with exact geometrical properties (see Structural Analysis, Geotip report, 2001). The flow behavior is therefore easily analyzable when using the computer codes dealing with a geometrically-described fracture network. When inverse modeling methods are used, a dual permeability parameter estimation can be easily made. The final output of this study provides a first step toward a more detailed analysis of the Richard Nuclear Storage Facility.

Poster #14

DEVELOPMENT OF NON-INTRUSIVE MONITORING USING CROSS-HOLE SEISMIC TOMOGRAPHY TECHNIQUES

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The EC Integrated Project, IP ESDRED (Engineering Studies and Demonstration of Repository Designs) was commissioned to establish a sound technical basis for demonstrating the safety of disposing of spent fuel and long-lived radioactive wastes in deep geological formations and to underpin the development of a common European view on the main issues related to the management and disposal of radioactive waste. Development of non-intrusive monitoring techniques is included within the programme as an important component of the overall ESDRED programme.

A key challenge of monitoring is the ability to monitor the waste and the barriers non-intrusively so as not to compromise their performance. An opportunity was identified for non-intrusive monitoring, using cross-hole seismic tomography, to monitor the changes at different phases in an existing Nagra experimental demonstration, the HG-A experiment, at Mont Terri. The HG-A experiment consists of a 1000mm diameter micro-tunnel which is back-filled with a sand/bentonite mixture and closed with a hydraulic mega-packer. In different phases the backfill is saturated with water and de-saturated using nitrogen; to allow measurement of the degree of saturation, gas storage and pressure build-up. Monitoring was conducted for each phase from 2 boreholes drilled perpendicular to the micro-tunnel, using non-intrusive seismic tomography techniques. Development of novel full waveform modelling and inversion schemes are currently being undertaken to analyse the monitoring data.

The partners recognised a further opportunity for developing *in situ* monitoring techniques utilising the construction and testing programme of a low pH shotcrete plug being constructed in granite at the Grimsel Test Site under ESDRED Module 4.Project TEM (Testing and Evaluation of Monitoring Techniques) at Grimsel was initiated to provide a unique opportunity for simultaneous comparison of 3 monitoring methods - wired signal transmission from the EBS, wireless data transmission using magneto-inductive techniques, and observation through non-intrusive geophysical techniques. The novelty of TEM is that the application of each of these techniques is tested under realistic conditions.