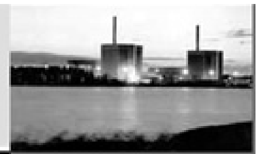
TOPSEAL 2006
Olkiluoto 17.9-20.9 2006



TOPSEAL 2006

Transactions

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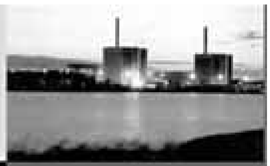

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SESSION I : International Perspectives on Radioactive Waste Management

Current Status of the French Radioactive Waste Disposal Programme

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Abstract

The 15 years of research prescribed by the Law of 1991 are now over. Their results led to the promulgation of a new planning act on 28 June 2006 detailing the applicable conditions and process for the pursuit of further programmes. It sets 2015 as the deadline to submit the statutory application in order to commission a deep geological repository for high-level and long-lived radioactive waste by 2025. The new law also sets the prescribed framework for the management programmes of the different waste categories.

As the years went by, experience kept accumulating and helped us to advance. Today, we are able to draw some lessons concerning the success factors of the most difficult projects in complex environments. Significant progress was recorded not only on the scientific and technical scales, but also and mainly with regard to governance and decision-making.

The *Planning Act of 28 June 2006* concerning the sustainable management of radioactive materials and waste marks a new step in the French legislation. It represents the natural outcome of the 15 years of research instigated by the Law of 30 December 1991. It is known also as the “Bataille Law” from the name of Christian Bataille, MP, who drafted it and monitored its enforcement as a member of the Parliamentary Office for Scientific and Technological Assessment (*Office parlementaire d’évaluation des choix scientifiques et techniques* – OPECST). Many advances were made concerning not only scientific and technological knowledge, but also governance. The new law opens brand new prospects in relation with those different aspects.

Lessons from experience

Andra’s history began in 1969 with the implementation of the *Centre de stockage de la Manche* (CSM), the first disposal facility for low-level and intermediate-level short-lived radioactive waste. It continued with the commissioning of the *Centre de stockage de l’Aube* (CSA), the second disposal facility for the same waste categories at Soulaines, Aube. A large number of lessons were drawn from the first experiment in the field of site selection, implementation, design and operation. In this paper, a few examples will serve to illustrate how initial weaknesses needed to be corrected for the CSA’s sake. The first point deals with the selection of a suitable site. All initial disposal operations at the CSM were essentially justified by the proximity of waste-generating facilities.

The first waste packages were deposited in trenches on a site with a complex geology and hydrogeology that made the safety demonstration all the more difficult to validate. First and foremost, the CSA was selected for its intrinsic characteristics: a very simple geometry consisting of a draining-sand layer above a clay layer. The sand layer allows potential seepage waters to be transferred to a single outlet that is easy to characterise and monitor. The clay layer prevents any contamination risk of the deeper aquifers and, consequently, any radionuclide transfer over several kilometres.

However, selecting the site on the sole basis of natural, geological and hydrogeological criteria would not have been sufficient to implement a disposal facility without taking into account also the human and social environments. The quality of the CSM’s relationships with its environment derives from the

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sound integration of its activity and of its staff within the local fabric. Many crisis situations occurred since 1969, but each of them helped to progress towards improved information and transparency. One might add, though, that the climate of confidence was not only due to the quality of communications. A responsible management, including the retrofitting of certain structures when necessary, as well as safety studies and demonstrations concerning the environment, human beings and their health, were most instrumental in building confidence. The improvements achieved at the CSM, sometimes due to crisis situations, were taken into account for the implementation of the CSA. The sound integration of Andra's staff within the local fabric, together with a communication programme that is increasingly available to all and a financial-incentive policy, have all contributed to the creation of the CSA.

Concurrently, the first safety assessments showed the advantages of multi-barrier mechanisms and the CSM was restructured according to that principle. Integrating the multi-barrier approach as early as the design stage of the CSA facilitated significantly the implementation procedure. Although that approach is now considered classical, it is still quite recent and consists in prescribing performance criteria for the packages, the structures and the site. It helped, for example, to specify the strict characteristics for waste packages, which are used today as waste-acceptance criteria (WAC). The classification of radioactive waste is based on the possibilities to dispose of them. Those possibilities rely on the WACs through combined criteria relating to the activity and the radioactive half-life of the radionuclides contained in the waste.

Disposal structures also benefit from new design and construction approaches. From the very beginning, the monitoring and the closing of the site were both taken into consideration. The seepage-water-collection networks run under the CSA's overall system of existing and projected structures. Their purpose is to monitor precisely the presence of any seepage, to trace back its origin, as well as to collect and analyse samples of it. Those networks, together with the piezometers spread through the environment are monitored on a permanent basis and the corresponding results are published every six months. Fortunately, no anomaly has been detected so far at the CSA and in its environment since commissioning in 1992. An experimental cover was installed right from the start in order to provide hindsight and experience concerning the selection of sound design and construction options when time comes to shut down the site definitely and to move on to the monitoring phase.

With regard to operation, controlling compliance with WACs on the waste-production sites and upon delivery of the packages at the CSA proves to be one of the major achievements. The computerised monitoring of the content of waste packages also ensures a finer management and a better control of their position within the disposal cells. Hence, suitable methods and tools have been developed in order to ensure a state-of-the-art traceability and to track down the origin of any anomaly, if need be.

The know-how transmission between the CSM and the CSA proved most fruitful. It also worked most successfully for the implementation and the commissioning of the disposal facility for very-low-level radioactive waste (*Centre de stockage pour les déchets de très faible activité* – CSTFA) at Morvilliers, located close to the CSA.

Figure 1: Aerial view of the CSA.

Figure 2: Aerial view of the CSTFA.

Research context in 1989

On the other hand, when new studies were launched in 1989, they were met with an opposition movement that was sometimes violent. However, the conditions were quite different. A broad research programme had been initiated without any prior information in order to study disposal options for high-level and long-lived radioactive waste. Doubts were raised and opposition developed due to the lack of information, thus leaving the field open to all sorts of interpretations and fears. Apart from the lack of information, the situation worsened due to the ignorance surrounding the issue. Until then, the only issue to be addressed had been low-level and intermediate-level waste, whose half-lives lie within the order of 30 years, a historical timescale that everybody is able to grasp. All of a sudden, though, citizens were requested to change their reference timescale and to jump to one million years, a timescale with which geologists may be familiar, but totally out of the boundaries of a layman's imagination. In addition, since people had just lived the Chernobyl catastrophe, it is easy to understand that they would associate it readily with the disposal of radioactive waste generated by nuclear power plants, that was now at stake, and increase their own anxiety.

Within such a context, the Prime Minister declared a moratorium that lead to the adoption of the Law of 30 December 1991 in which the following innovating approaches were prescribed:

- a stepwise decision-making process;
- the study of alternative solutions;
- independent assessment;
- information;
- the independence of the agency responsible for radioactive-waste management in relation to waste producers.

Main steps and achievements since 1991

Soon after the adoption of the Law and the corresponding decree, the government entrusted upon Mr Christian Bataille, MP, to organise a public consultation with a view to seeking potential implementation sites for underground research laboratories (URL). In the meantime, however, the situation had changed drastically, since the information mission was now given to a member of Parliament, detailed monitoring and control conditions were prescribed and financial-incentive modalities were clearly defined. In 1993, four candidate sites were selected among approximately 30 voluntary applications: a granite site under a sedimentary cover located in the Vienne; a deep marl site located in the Gard close to the Rhône River, as well as two sites located in Callovo-Oxfordian argillites, one in the Meuse and the other in the Haute-Marne. Those two last sites were rapidly combined to form the Meuse/Haute-Marne Site.

As early as 1994, and for a period of two years, detailed investigations were launched. At this stage they were limited to the form of surface boreholes and in geophysical-measurement campaigns. Concurrently, local information and oversight committees (*commission locale d'information et de suivi* – CLIS) and incentive funds were set in place. Based on the conclusions of its studies, Andra submitted three applications to authorise the implementation of URLs on the Vienne, Gard and Meuse/Haute-Marne Sites. In 1997, the applications were the subject of public inquiries. Relevant territorial communities were called upon to express their views and confirmed their willingness and, consequently, their agreement to host such URLs.

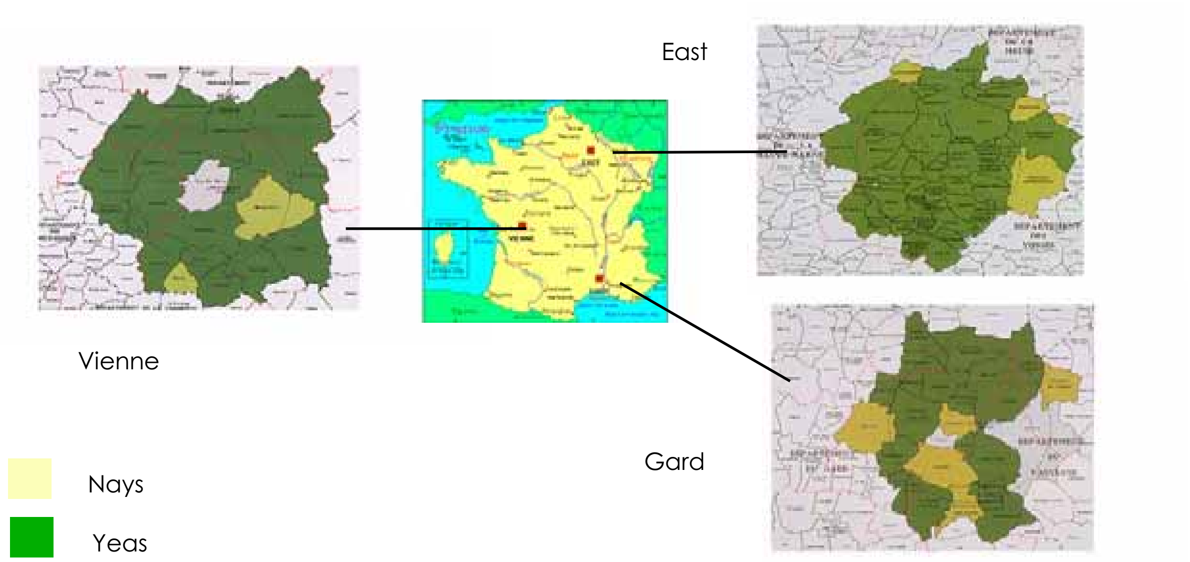

Figure 3: Vote results of the territorial communities in 1997 concerning three sites:
Vienne, East (Meuse/Haute-Marne) and Gard Sites.

The government's decision following the review of the applications by the competent services confirmed the continuation of operations in the Meuse/Haute-Marne with the creation of a URL in Bure. On the other hand, both the Gard and Vienne Sites were abandoned. In parallel, the government set in place a research mission in order to find a new granite site, but the project was not met with any local support and was finally terminated in 1999 following numerous opposition movements. In the meantime, the preliminary work started for the construction of the Meuse/Haute-Marne URL. The construction phase and the first experiments took place from 2001. The accumulated information served as the basis for the *Dossier 2005 Argile* that was submitted to the government in mid-2005.

The scientific and regulatory assessments of the *Dossier* were entrusted to the National Review Board (*Commission nationale d'évaluation* – CNE) and the Nuclear Safety Authority, respectively. The French government also requested that an international peer review be carried out under the aegis of the OECD Nuclear Energy Agency (NEA). The resulting opinions were quite remarkable and very encouraging for Andra and for its scientific and technical partners, thus emphasising the quality of the work undertaken according to the best international standards and concluding to the feasibility of a deep geological repository.

The government also wished that a national debate be organised concerning the long-term management of radioactive waste. Consequently, it called upon an independent entity, the National Commission on Public Debate (*Commission nationale du débat public* – CNDP). After six months of preparation, the debate included 13 meetings that were held in different cities from September 2005 to January 2006. Scientific and technical themes, management strategies and governance were discussed at length. In its final report, the Commission stressed the existence of a general demand for:

- all waste categories to be taken into account by the legislation;
- the need to improve governance regarding radioactive-waste management;
- the advantages of a stepwise decision-making process;
- and the need for a true economic-incentive programme for the territories on which any deep geological repository would be implemented.

Lastly, the report of the OPECST, published in March 2005 by Messrs Birraux and Bataille, MPs, analysed the results of investigations from the standpoint of management strategies. It concluded to the complementarity of the three research areas prescribed by the Law of 30 December 1991: partitioning and transmutation, deep geological disposal and long-term storage.

Presentation of the Law of 28 June 2006

The new act falls in line with the approach adopted by the Law of 1991 by prescribing specific deadlines for the different management solutions to be enforced. For partitioning and transmutation, industrial prospects relating to the investigations for the fourth generation of reactors shall be established by 2012. With respect to a reversible repository within a deep geological formation, all relevant elements shall be gathered in order for the corresponding application for the implementation of a deep geological repository to be submitted and reviewed by 2015 and for the repository to be commissioned in 2025. This date is compatible with the production schedule of high-level and long-lived waste by the French nuclear-fuel-cycle industry.

The new act also provides two essential elements in areas that were not addressed by the Law of 1991. It meets one of the recommendations formulated during the public debate and advocates a true national management policy, not only for radioactive waste, but also for radioactive materials, whether recoverable or not, by instituting the National Radioactive Waste Management Plan. Besides setting specific deadlines for high-level and long-lived waste, the act also prescribes that a decision be made by 2013 for graphite and radium-bearing waste, a category of low-level but long-lived waste. Hence, within the next few years, all categories of radioactive waste will have been attributed a relevant management solution.

	Short-lived (half-life < 30y years)	**Long-lived (half-life > 30y years)**
Very low level (VLL)	VLL Waste Disposal Facility (Aube)	
Low level (LL)	LL/IL Waste Disposal Facility(Aube)	Investigations on repository projects Commissioning in 2013
Intermediate level (IL)		
High level (HL)	Investigations conducted in accordance with the Law of 30 December 1991, and now with the Planning Act of 2006	

Figure 4: Classification of radioactive waste and status on disposal solutions.

Moreover, the new law establishes the legislative framework for the dismantling of nuclear facilities and, particularly, for the secured financial provisions to be constituted by waste producers in order to ensure that an amount of 68 billion euros, which is currently deemed adequate, be available. Parliament will participate in the control of those financial provisions and in their appropriation as dedicated assets in the companies' accounts.

Lastly, the law strengthens the socio-economic incentive programme applicable to the territories concerned by the implementation of a potential waste repository. It reinforces the status of the existing public-interest groups (*groupement d'intérêt public* – GIP) devoted to the local development in the Meuse and the Haute-Marne with a view to involving the nuclear industry in local industrial projects, while improving the status of the local consultation and information structure for elected officials and citizens.

Beyond its industrial mission regarding radioactive-waste management, its research mission notably with respect to high-level and long-lived waste, and its information mission to disseminate relevant knowledge, the major evolutions of Andra's mandate involve:

- leading investigations on waste storage;
- designing, implementing and managing waste-storage and disposal facilities;
- taking over orphan waste and sites (public-service mission).

Success factors

The accumulated experience certainly constitutes the most important capital since it has allowed us over the years to improve operational and safety procedures and should prove beneficial for the implementation of any new facility. However, it is not sufficient to undertake field investigations without preparing adequately the various publics and without launching a broad information campaign beforehand. What happened in local communities in 1989 highly proved that point, since over and above the triggered opposition to the implementation project and the associated work, the events instilled a climate of doubt at National level. Consequently, it was necessary to gain back public confidence, a challenge that proved all the more delicate since the progress to be achieved after a crisis period is always broader in scope. The purpose is by no means to move from indifference to confidence anymore – a situation that represents a larger gap on a value scale – and requires a much more significant effort altogether.

The 1989 crisis will have served to provide a structured process with regular controls and prescribed deadlines as clearly indicated in the Law of 1991, thus marking a decisive step with regard to governance for radioactive-waste management.

Based on the certainties formulated by the technicians, the Law of 1991 was able to introduce alternative solutions. In order to undertake the disposal of radioactive waste within a deep geological formation, it is necessary to verify that there are no other suitable means to eliminate or to process that waste. That is the purpose of the investigations on partitioning and transmutation, the first area over the last 15 years of prescribed research. The second solution involves deep geological disposal itself. By prescribing that the reversibility of such an option be ascertained, the Law of 1991 introduced a brand new concept, at least in the mind of many people, since the approach moved from a rationale based on burying the waste and, to some extent, on forgetting about it, to the approach of a responsible manager who may be called upon not only to recover the waste during a certain timescale, but also to ensure the monitoring of the facilities and of their environment, sustaining thus any further decision either to retrieve waste packages or to close disposal drift or access tunnels. The most important criteria upon which a decision would be taken is the comparison between the monitoring results with those predicted from all the studies and models. Lastly, the third area involved stabilising, conditioning and storing the waste over the long term, which represents another form of long-term alternative before considering a final solution.

Policy-makers have also placed research programmes and results under the strict control of the CNE, whose responsibility is to submit an annual report to Parliament and the government. The assessment of management strategies is entrusted upon the OPECST. A deadline was set in 2006 in order to take stock of the investigations and to recommend whether to move on to further steps.

Besides programme control, two accompanying structures were set on the sites themselves in order to involve and to give responsibilities to local representatives, elected officials, administrations and associations. The first structure consists of the CLIS, whose mandate is to monitor the evolution of investigations and to inform the public on the related programmes, their advances and their results. The second structure, consisting of various public interest groups (GIP), manages the incentive funds dedicated to local development projects.

Through the political framework and the local incentive programmes for each site, a general system is set for both the national and local supports of new implementations, as in the case of the Meuse/Haute-Marne URL. Inversely, once the government decided not to pursue the granite project in the Vienne, the mission prescribed by the government to seek a new site was not met with the essential local support for its implementation.

Among the success factors, the evolution of Andra's views has also played an important role. In the past, Andra was but a simple agency within the French Atomic Energy Commission (*Commissariat à l'énergie atomique* – CEA), one of the major actors in the nuclear field. Thanks to the Law of 1991, Andra's status evolved into a public establishment, independent from waste producers and reporting to three supervisory ministries: Industry, the Environment and Research. That evolution reflected also the direct implication of the State in safe management practices for the protection of human beings and their environment. Once again, it also involved a switch-over since the purpose was to make clear that not only was the State responsible for managing and conducting research on radioactive waste, but it was also exerting a full control on policies that might have been imposed otherwise by waste producers.

Lastly, the confidence deficit also resulted from the general lack of transparency. The regular publication of an updated national inventory of all radioactive waste present on French soil has largely contributed to re-establish part of that confidence and all the more since the successive editions have been enriched progressively with new information, notably with regard to military activities, which are normally secret in nature.

The Law of 1991 provided a very rich framework with a view to reviving essential values, such as responsibility, not only in the sense of the collective responsibility towards radioactive-waste management, but also the responsibility of each actor, each political decision-maker at the national scale, each local supporter of new facility projects, each scientist, each association and, of course, each waste producer through the requirement to fund the relevant programmes.

At the scale of important decisions such as the future of radioactive waste, the statutory 15-year timescale also allowed a stepwise approach within the decision-making process, including a major threshold to be crossed when the Law of 28 June 2006 was promulgated.

A sustained information programme is necessary for the sound conduct of such projects. Its purpose is to ensure that everybody is able to follow their evolution, to assess their achievements and to organise a debate in order to improve and to share the general reflection. Provided that projects and investigations are launched on a clear, transparent and stable basis that has been determined from the very beginning of the process and that continues to be shared among the different actors, Andra is readily in a position to relay messages in accordance with its communication and information mission. That basis must therefore help to reinforce confidence and, in order to achieve that goal, must abide by the two following principles: the possibility to verify information and the requirement never to leave a pending issue without a legitimate answer. A wide communication approach was launched to reach the various publics with a special emphasis on the clarity of the messages to be disseminated, particularly with regard to complex scientific issues. The primary objective is to explain and never to run the risk of introducing doubt by creating a mental block of the general public when faced with a seemingly unexplainable complexity.

Communication documents are made available free of charge, in either paper or electronic version, or are downloadable from Andra's Web site (www.andra.fr); they represent the Agency's main information vectors. Guided tours of the different surface disposal facilities and of the Meuse/Haute-Marne URL also help in improving public confidence by providing the public an opportunity to observe *in situ* the quality of the Agency's achievements. In 2005, for example, no less than 13,000 visitors were greeted on Andra sites.

Lastly, one of the requirements to control the external processes prescribed by the Law relates to the quality of internal activities. It gives rise to the different assessments mentioned above and it is also confirmed in its application by the delivery of ISO-9001 and ISO-140001 certificates, which implies the strict observation of the commitments taken by the Agency, especially with regard to deadlines.

Examples of specific recent achievements

A major milestone was reached in 2006 with the promulgation of the Law of 28 June, which opens up new prospects for managing the entire set of radioactive waste categories, including high-level and long-lived residues. In the meantime, surface disposal facilities continue to operate and to undergo specific developments, such as the experimental work carried out in the Meuse/Haute-Marne URL.

Surface disposal facilities

Close to 25,000 m^3 of radioactive waste were delivered at the CSTFA. At the CSA, 15,514 m^3 of radioactive waste were accommodated in 2005 and the filling rate is progressively reaching the 20% mark.

In both facilities, specific solicitations from producers have led to consider storing bulky items.

The waste intended for disposal in Andra facilities exists in the form of packages whose shapes and dimensions are mentioned in detail in the handling specifications within those facilities and in the deposit specifications within disposal structures. However, some waste categories may have a complex and bulky shape that requires major cutting operations in order to be conditioned into acceptable packages within those facilities.

In certain cases, the direct transfer of such large-size waste may provide additional radiation-protection benefits by reducing the intervention time on dismantling worksites.

Andra has already taken over the responsibility for such waste at both the CSA and the CSTFA. In the first case, which deals with low-level and intermediate-level waste, specific structures have been designed to accommodate reactor-vessel covers originating from *Électricité de France*'s (EdF) nuclear power plants. Six covers were received in 2005, bringing to nine the total number of disposed covers.

Figure 5: Transfer of a reactor-vessel cover towards the disposal-cell area (*left*) and placement in a disposal cell (*right*).

In the second case, the VLLW Disposal Facility received the exchange bottles from the Saint-Laurent A NPP as well as used transport containers from Cogéma Logistics. Andra is currently studying the possibility to take over the steam generators of the Chooz A NPP and the light neutron shields from the Creys-Malville NPP.

A parallel reflection is under way in consultation with waste producers concerning the definition of a waste inventory suitable for direct disposal in order to anticipate the facility's future operational and equipment requirements with sufficient lead-time.

Andra's capability to take into account waste with atypical characteristics after carrying out feasibility and safety studies and after obtaining any relevant agreements also helps to reinforce the confidence of not only waste producers who are provided with a suitable reference structure that is able to grasp a difficult issue and, above all, to provide them with an adequate solution, but also of the public in general who appreciates the fact that a safe solution exists in each case.

Meuse/Haute-Marne Underground Research Laboratory

The construction of the Meuse/Haute-Marne URL was completed with the junction of the connecting drift between the two shafts and the opening of the drifts and structures prescribed in the initial programme. Final operations are under way in the shafts with the installation of the permanent lift machinery.

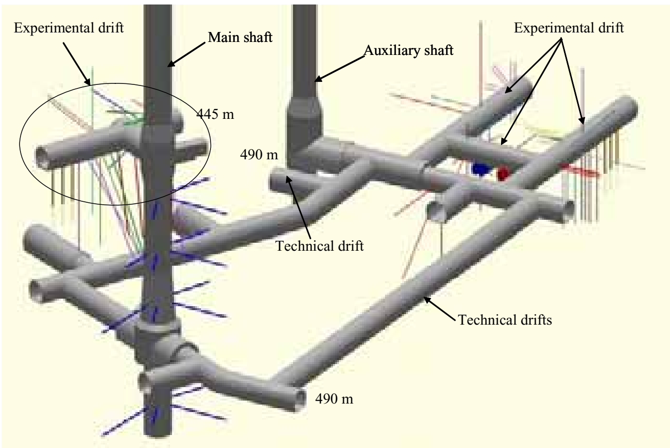

Figure 6: Layout of the Meuse/Haute-Marne Underground Research Laboratory.

The experimental programme started as early as shaft-sinking operations at the end of 1999 and culminated with the opening of the drift located at a depth of 445 m. During excavation operations, a fine geological and hydrogeological survey was undertaken to study the overall sedimentary profile down to the roof of the Callovo-Oxfordian formation at a depth of 420 m. Work in the argillites not only provided detailed geological surveys, but also allowed for observations and measurements to be made concerning the mechanical behaviour and the resistance of the formation to excavation. A large number of preliminary experiments were conducted in the niche before the actual experimental drifts located at a depth of 490 m became operational, thus allowing for the first data to be collected and for the initial set-up to be made in preparation for the further experiments to be performed at the bottom.

Figure 7: Drilling operations for the installation of geomechanical-measuring devices during the sinking of the main shaft.

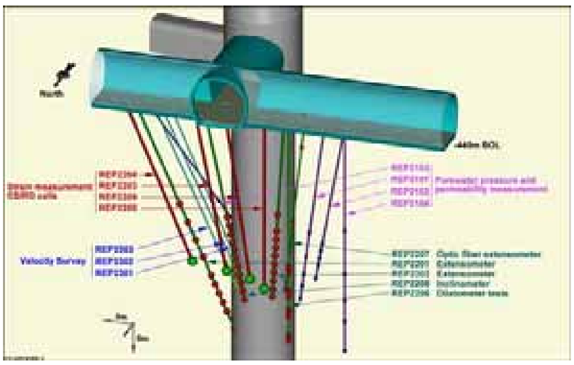

Figure 8: Drilling of boreholes from the experimental drift (-445 m) in order to monitor the hydrodynamic characteristics around the shaft during shaft-sinking operations.

One of the initial motivations to create a niche was to excavate observation and measurement wells in the immediate vicinity of the main shaft before its sinking and, thus, to record all signals likely to describe the mechanical behaviour of the argillite during and after excavation. The impact of excavation on hydrodynamic behaviour was also analysed thanks to the instruments set in place. In addition, the first diffusion experiments were conducted, and confirmed the characteristic values of transfer properties that had been obtained from laboratory specimens.

At a depth of 490 m, the number of observations, measurements and experiments intensified in order to confirm primarily the validity of existing data. One of the challenges was to collect and to characterise water samples within an environment known for its low water content in the order of a few percent and mostly for its very low mobility (horizontal permeability in the order 10^{-13} m/s).

A last mention should also be made that mechanical measurements were used to study various means to seal disposal structures. In order to prevent any risk of hydraulic short-circuit through the excavation-disturbed zone (EDZ), various cuts were made on the drift walls and filled with bentonite briquettes.

Since the launching of operations on the Meuse/Haute-Marne URL Site, 27 deep boreholes were drilled from the surface and 130 scientific boreholes were drilled from the 40-m-long niche and from the different drifts extending over a total length of 484 m at a depth of 490 m.

Current research programme until 2015

Today, the first phase of the experimental programme is fully operational. The programme for the second phase was prepared and will be launched progressively with a view to reducing further the margins of uncertainty, confirming acquired data and submitting the formation to various solicitations (e.g., heat or the presence of exogenous materials, such as hydraulic binding agents or steels).

Technological tests are also scheduled in order to verify the feasibility of certain methods and processes to be implemented before a future disposal facility may be implemented and operated.

Since the Act of 2006 prescribes that no disposal site may be proposed if its host geological formation has not been submitted to various studies within a URL, all investigations will need to be concentrated in the Bure area. Proposing a site will be the main challenge before submitting by 2015 at the latest an authorisation application to implement a waste repository. A 200-km^2 transposition zone has been therefore delineated around Bure: its geology is well described and its characteristics are deemed sufficiently similar to those of the Bure Site to be directly transposable to the zone. Both a 2-D seismic campaign, followed by a 3-D seismic campaign at a lower scale, are scheduled to take place before 2009 in order to verify the absence of any major tectonic accident.

Seeking a suitable site for the authorisation application to implement a waste repository also requires a sustained information, consultation and co-operation effort with local communities who will be expected to participate in a territorial project integrating the repository project. Besides the significant incentive programme prescribed by the Law, industrialists have committed themselves to supporting local-development endeavours. Many industrial implementations are already planned in the region. The political, social and economic dimensions have been incorporated as early as possible and are very closely monitored by the government.

Investigations continue at the Meuse/Haute-Marne URL and the information generated by the related experiments keeps accumulating. The major research themes include the furthering of the knowledge and the representation of the phenomena involved in the operation of the waste repository, the study of their couplings, upscaling over the entire expansion of the repository and over timescales beyond the field of experience, as well as the description of mechanical, hydraulic, thermal and chemical transients, especially to provide a finer knowledge in preparation for the reversibility study and the associated monitoring means.

Among the important issues that gave rise to recommendations by the various assessment authorities, a special mention should be made of the need to further knowledge about the EDZ, radionuclide migration within the Callovo-Oxfordian argillite and the behaviour of gases. Another recommendation addressed the need to create demonstrator models not only for building and closing disposal structures, but also for transferring and emplacing waste in those structures. Those suggestions were incorporated in the research programme to be conducted until 2015.

All those elements, whether scientific, technical, socio-economic or political, will nurture the public debate prescribed by the Law of 2006 in preparation for the authorisation application to implement a disposal facility for high-level and long-lived waste. A sustained programme has therefore been developed and includes the preparation of a first report by 2012 with a view to organising the debate and to finalising the report by 2015.

Conclusion

The last 15 years of hard work that recently resulted in the promulgation of a new act were marked with several significant developments with regard not only to scientific and technical expertise, but also to human behaviour and governance, especially in the context of complex social challenges.

Today, new issues are at stake, but are in direct line with the ones experienced so far. The Law of 2006 opened the way to the implementation of a waste repository for high-level and long-lived radioactive waste. It also refined the national radioactive-waste management mechanism by instituting the National Management Plan for Radioactive Materials and Waste, thus prescribing suitable management methods, whether under study or already implemented, for every waste category. The Plan will take into account the different programmes, including those for the disposal of high-level and long-lived waste, as well as of graphite and radium-bearing waste, for which the Law prescribes that a suitable facility be commissioned by 2013.

All those projects will need to be accompanied by a sustained effort in favour of information, communication, training and diffusion of know-how. Such an endeavour will require a strong implication of all stakeholders with a constant concern not only to share their knowledge and the new data, but also to promote them within the framework of a project of national interest.

Before closing, I would like to reiterate how strongly I feel about international co-operation. Without it, Andra would not have been able to develop its experiments as efficiently as it did in the underground laboratory and would have desperately fallen short of a most fruitful debate and of the resulting advances. As in the case of the first phase, Andra is inviting all foreign teams to participate in its ongoing experimental programmes at the Meuse/Haute-Marne URL.

HLW DISPOSAL IN GERMANY – R&D ACHIEVEMENTS AND OUTLOOK

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ABSTRACT

The paper gives a brief overview of the status of R&D on HLW disposal. Shortly addressed is the current nuclear policy. After describing the responsibilities regarding R&D for disposing of heat-generating high-level (HLW) waste (vitrified waste and spent fuel), selected projects are mentioned to illustrate the state of knowledge in disposing of waste in rock salt. Participation in international projects and programs is described to illustrate the value for the German concepts and ideas for HLW disposal in different rock types. Finally, a condensed outlook on future activities is given.

1. Introduction

In Germany in 2006, 17 reactors at 12 nuclear power plants (NPP) are in operation. In the first six months the total output power was about 86 TWh, about 5 % more than during the same time period in 2005. Still about 30 % of the total electricity production is contributed by the German NPP. To date 19 reactors (power reactors and prototype facilities) are in the decommissioning phase or have been decommissioned.

Nuclear matters were discussed during the coalition negotiation talks end of last year. It was decided to maintain the status-quo because of opposing views with respect to the use of nuclear energy and to keep up decisions made in the consensus agreement of 2001, which was the basis for the nuclear phase-out. It was agreed upon, however, to continue and expand safety research on NPP. The coalition parties also confirmed that it is acknowledged that the safe disposal of radioactive waste has to be ensured and the Government will *"tackle this issue in a speedy and result oriented manner. We intend to solve this question by the end of the current electoral term*." (1) In April 2006, the Government's new energy concept lasting to 2010 was discussed by high-ranking politicians and experts from electric utilities at the first national energy summit. Working groups were established to prepare documents for the next meeting in fall 2006. (2)

2. Current nuclear waste policy

Reprocessing of spent fuel (SF) elements in France and in the UK has been terminated. The vitrified waste, still in France and in the UK, will be taken back in due time according to existing international agreements. Shipment of the reprocessed waste from France and the UK to the central interim storage facilities is allowed. The shipments to the reprocessing plants in France and the UK were discontinued. Shipment of HLW and spent fuel (SF) from the interim storage facilities or the on-site storage facilities to the repository site will not be allowed before a deep geological repository is in operation.

Two central interim storage facilities are operational. At Ahaus (North Rhine-Westphalia) the BZA (Brennelement-Zwischenlager Ahaus) facility is used to store both SF elements and thorium high-temperature reactor (THTR) fuel elements. At Gorleben (Lower Saxony) the BLG (Brennelement-Lager Gorleben) facility is used to store SF elements and the reprocessed vitrified waste. Two de-centralized interim storage facilities for spent fuel elements at the sites of two decommissioned reactors are in

operation, too. At each NPP an on-site storage facility will be operated to store the SF for a period of 40 years. All facilities are licensed by the Federal Office for Radiation Protection (BfS). (3)
The lifetime of existing nuclear reactors is determined by the limited electrical output (Atomic Energy Act (AEA), 2002) Preliminary calculations, based upon the AEA, show that energy production will stop in the year 2022. Especially this issue still is a matter controversial national discussion against the background of international developments and initiatives.
There is consensus to dispose of all types of wastes in deep underground repositories in Germany, i.e., neither export nor import of radioactive waste is allowed. A decision about the rock type that will finally host the repository for heat generating waste is still pending. There are some discussions of having a sort of site selection process using some of the ideas of the AkEnd (4). However, this might postpone the target of the Government to start the operation of a repository in 2030.
In Germany, rock salt was the favourite host rock for a deep underground repository for vitrified waste and SF. R&D that was performed for years has lead to a sound base of knowledge. However, based upon a governmental decree in connection with the phase-out decision, the investigation of other favourable host rocks became also subject to R&D activities.
Moreover, the idea to have only one single repository for all kinds of radioactive waste still is under discussion.
The year 2000-moratorium that halts the exploration of the Gorleben salt dome still is effective. Only on-site maintenance measures are permitted. A crucial point connected with the moratorium was to clarify questions related to conceptual and safety-related issues for all suitable host rock types. The Federal Office for Radiation Protection (BfS) awarded contracts to national and international groups to work at these questions. The reports were reviewed, presented and discussed during an internal workshop one year ago (3). At present the synthesis report, its conclusions and the consequences for the Gorleben moratorium are being reviewed by GRS, the main German TSO. Presently, there is no indication that the moratorium would be finished prematurely.
For Schacht Konrad, planned as a repository for non-heat generating intermediate and low-level waste, the licensing procedure is completed after more than two decades. However, law suits delayed the immediate implementation. On March 8, 2006, the complaints were rejected by a high administrative court because they were irrelevant. There was no appeal possible. Yet, the complainants will go to the higher administrative court to appeal against this decision. Hopefully, after the decreed license and the Federal Government's decision that Konrad is to be operated as a repository, the start of operation might be around the year 2013.
In Morsleben the stop of short-lived long- and intermediate-level waste emplacement was decreed. At present, among others things, the main activities comprise activities necessary for licensing and closure of the mine and the repository areas.

3. Responsibilities for R&D

The 5th Energy Research Program of the Federal Government "Innovation and New Technology" is the framework for R&D activities. The Ministry of Economics and Technology (BMWi) is the ministry in charge. The Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU) and the Ministry of Education and Research (BMBF) are involved, as well. Besides the main topics "Renewable energies" and "Energy efficiency" nuclear safety research and waste disposal are parts of the program. (2)

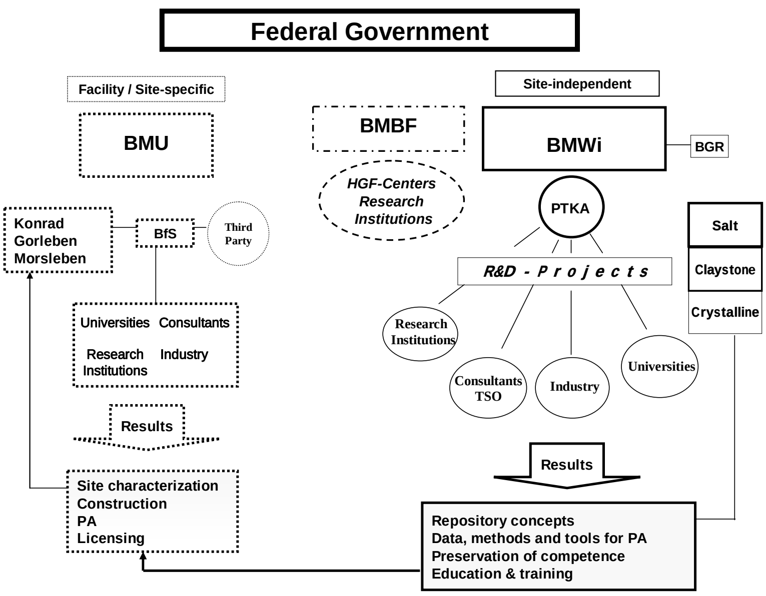

Fig.1. Responsibilities for R&D

Activities related to HLW disposal are basically the responsibility of BMU, the BMWi, and BMBF.
Leading principles are safety and responsibility for present and future generations, thus being in line with international conventions. Subsumed under this is the Government's legal obligation to provide repositories for radioactive waste. Basis for funding site-independent research by projects is the Research Concept of BMWi. (5) In the beginning it focused on R&D activities related to disposal in rock salt. After the changes in politics and the new nuclear policy streamlining became necessary in some areas to prioritize R&D and to put emphasize on other host rock types than rock salt. The R&D projects funded by BMWi (Fig. 1, right) can be assigned to applied basic research, defined as basic research being conducted with the prospect to create a sound basic knowledge in a long perspective helping to solve foreseen, consisting or future problems. The Project Management Agency Forschungszentrum Karlsruhe (PTKA) acting on behalf of BMWi is the managing and supervising unit. It helps to transfer programmatic contents into projects, provides support with regard to conceptual work, controlling, evaluation, and general management tasks. This respective research is carried out by industrial companies, consultants, technical support organizations (TSO), universities, and research institutions. The results can principally be used by legal bodies, authorities, reviewers, operators, private industry, and other stakeholders.
BMBF is funding primarily basic research conducted by the national research scientific-technical and biological-medical centres that constitute the Helmholtz-Association. Research on waste disposal is carried out in the national research centres Karlsruhe and Jülich. BMBF is also responsible for funding R&D on underground disposal of chemotoxic waste. The respective projects are also coordinated and supervised by PTKA.
BMU, the German regulatory body, is responsible for the disposal projects and the related facility- / site-specific R&D. On behalf of BMU, BfS initiates and coordinates this R&D. BfS is in charge of activities regarding construction and operation of facilities for disposing of radioactive waste using the expertise of third-party organizations. Project-specific activities are performed mainly by research centres, consultants, universities, and industrial companies. The results are directly used by BfS for site characterization, performance assessment, and license application.
By law, the costs for facility-specific R&D are paid according to the “polluter-pays-principle“, e.g., regarding Gorleben, the electric utility industry.

4. Achievements

During the last decades a lot of R&D on HLW disposal in rock salt was performed. One of the most important achievements was the development of the Direct Disposal concept. Starting in 1985, after having performed some preliminary studies, this ambitious R&D program was successfully finished officially in 1995. The goals of the four subprograms, a) spent fuel conditioning and cask development (POLLUX casks and canisters), b) demonstration tests (emplacement and handling technologies for heavy payloads, THM behaviour of crushed salt backfill), c) conceptual design of the disposal systems (System analyses), and d) laboratory tests were achieved. All large-scale tests were successfully executed and could be concluded according to schedule. (6) For scientific reasons, the last test within this program, the TSDE (Thermal Simulation of Drift Emplacement) or EC-co funded BAMBUS experiment was prolonged and was finished in 2004 (7). It was the last it-situ test performed in the Asse salt mine. As a consequence of this program direct disposal of spent fuel became a disposal option legally equivalent to reprocessing. Now it is the only legally accepted way to dispose of spent fuel and vitrified waste.
Lessons learnt from all these experiments were that validation of function and reliability of large technical equipment is feasible, that material behaviour can be described adequately by experiments and modelling and that legal requirements can be fulfilled. Moreover, it is commonly acknowledged that the use of large-scale or full-scale in-situ demonstration experiments, also performed in underground research laboratories, is indispensable. This is not only essential for scientific reasons but also because of its importance to get public acceptance for safe and secure handling of technology.
Besides the engineering work and large-scale experiments, laboratory experiments and research focused on contributions for performance assessment has been performed to improve the knowledge. Especially the tools and instruments to be used in modelling and performance assessment were further developed substantially and tested in several national and international projects.

An important experiment, coordinated by PTKA and managed by the Kali und Salz (K+S) company, was the BMBF-funded large-scale shaft sealing experiment carried out with regard to the underground disposal of chemotoxic waste. (8) This joint project aimed at planning, constructing, and testing of a long-term stable sealing system. The investigations were accompanied by laboratory experiments and numerical modelling. It could be shown that the experiment can be considered to be representative for sealing elements to be used in real shafts. The project results are used by K+S in the closing activities of three shafts of a salt mine. Although the results were not intended to be used for radioactive waste disposal a lot of expertise was gained that can be used in a synergistic way.
The knowledge gained about disposing of heat-generating waste in rock salt during the past decades has reached a certain status of maturity. It was shown that technological problems can principally be tackled. Techniques for the emplacement of spent fuel and vitrified waste are at hand. A lot of knowledge has been accumulated about the behaviour of rock salt and crushed salt backfill. Databases and models were permanently improved. Instruments to be applied in safety assessments exercises are available. Yet, there are some open questions left that could be answered in the years to come. Moreover, up to now there are no indications that rock salt is not suitable to accommodate a repository for heat generating waste.
The Government decided that other host rock types should be investigated. To reach an adequate level of knowledge and expertise as compared to rock salt in due time is a challenge. Therefore, R&D activities were intensified. The respective projects are focused mainly on argillaceous rock and comprise feasibility studies (9), laboratory experiments and modelling. The results achieved to date have contributed to a certain level of knowledge and expertise. (e.g., 10, 11).

5. International cooperation

A lot of these activities are integrated in international programs and activities in underground research laboratories. There are various reasons for that approach. Experience showed that the fruitful and successful cooperative work has both contributed a lot to the national knowledge and fostered the exchange between researchers in scientific and technological areas. Problems connected with large R&D projects can be tackled much more efficiently by an international "job-sharing" effort, both to reduce risks and costs. An additional important aspect surely is the positive effects on education and training on human capital involved in all the activities. International cooperation is mainly based on agreements between Governments, scientific institutes, universities or national research centres, and on agreements with the European Commission. Very positive outcomes arise by the overlapping of projects in these areas resulting in building up multidisciplinary and multinational networks. It is acknowledged that it becomes more and more valuable to cooperate in joint international projects, e.g. in URLs, both to share the financial burden but also to use the combined and sound expertise of national and international experts to solve common research tasks. Against the background of these positive aspects and because of its responsibility for R&D as well as a sign for the importance attached to international cooperation, international activities are directly funded or co-funded by the German government, respectively BMWi. Moreover, the benefits of international cooperation can be communicated to the public showing that there is a common understanding in the world-wide scientific community to solve the task of waste disposal together in a multinational effort, safety-oriented and responsible in due time.
When rock salt was the favoured material to host a repository in Germany, and the Asse mine was used as sort of URL, a series of projects were carried out in cooperation with institutions from countries then also interested in rock salt (e.g., France, the Netherlands, Spain, US). Most of these experiments were co-funded by the European Commission in the respective Framework Programs. (e.g. 12)
Yet, German research institutions have participated in foreign projects and programs related to other host rock formations, because it was important to understand possible other candidate host rock formations and to get information and knowledge for evaluating the pros and cons.

A lot of projects were and still are performed in underground laboratories. During the last years the participation focused on topics like development of techniques and technologies for site characterization, for repository construction and operation, and performance assessment. In the meantime the knowledge in some areas has become quite advanced. Therefore, future work will increasingly focus on specific problems.

Concerning crystalline rock, the ongoing R&D activities focus on experiments performed in the URLs in the Swiss Grimsel Test Site (GTS) and in the Swedish HRL Äspö. On a minor scale, there is collaboration with Russian institutions.
The activities in the GTS started about two decades ago. Close collaboration with international partners took place in a series of projects during the six investigation phases of the GTS. (13) The investigations comprised host rock characterization, considering the hydrogeological, petrophysical, and mechanical properties, projects, the study of the EBS behaviour (i.e. FEBEX, GMT),and the study of colloid and radionuclide migration. A lot of basic knowledge was gained, a detailed system understanding was created, sophisticated computer codes, measuring devices and methods, and state-of-the-art analytical tools, were developed.
In 1995 cooperation in the HRL Äspö started because it was deemed necessary to extend the knowledge on types of crystalline rock other than those of the Grimsel granite. Cooperation started in projects focusing on developing and testing of instrumentation and methods for underground rock characterization, studying the behaviour of the EBS, development of numerical flow and transport models as well as studying radionuclide migration (in particular actinides), and the impacts of colloids and microbes. Tools, measuring devices, sophisticated analytical instruments and method used in GTS were transferred and very successively used in the HRL Äspö. (14) At present German scientists participate in six projects in the HRL. (15)
Since 2001, there has been a cooperation agreement between BMWi and ROSATOM (the former MINATOM). The very first project within the collaboration aims at developing a proposal for a site investigation and selection program for a generic repository in crystalline rock. German and Russian experts in geology, technology and safety assessment are working on a concept for borehole disposal of vitrified waste in granite in the area of Krasnoyarsk. (11, 16)
R&D in argillaceous rocks started late compared to the activities in crystalline rock. Initially it was mainly focused on plastic clay, and on minor scale, on indurated clay. Now emphasise is put on indurated clay. Because there is no URL in Germany it is appropriate to participate in the Swiss Mont Terri URL and the French URL in Bure. In the Mont Terri URL projects related to all clay-relevant issues are performed. German scientists are involved in several projects. (17) At the URL at Bure activities focus on issues that are comparable to and complementing the Mont Terri activities. Topics are the study of THM-properties of the clay, the characterization of this material, and participation in selected in-situ tests. The entire work in the URLs is accompanied by laboratory experiments and modelling.

5.1 Framework Programs of the European Commission

The participation in the Framework Program of the European Commission still is an essential part of the national research policy and is considered as crucial and necessary for reasons like to gain expertise, to transfer experience, to exchange knowledge, to increase excellence, support mobility and scientific exchange. In the 6th Framework Program German research institutions participate in nearly all projects/initiatives either as coordinator or as member of the scientific body. Participation takes place in the projects NF-PRO (Understanding and Physical and Numerical Modelling of the Key Processes in the Near-Field and their Coupling for Different Host Rocks and Repository Strategies), ESDRED (Engineering Studies and Demonstrations of Repository Designs), FUNMIG (Fundamental Processes of Radionuclide Migration), PAMINA (Performance Assessment Methodologies in Application to Guide the Development of the Safety Case, under negotiation), the Network for Actinides Sciences ACTINET, and the Project “Co-ordination of research, development and demonstration (RD&D) priorities and strategies for geological disposal”(CARD). (18)
Because of the benefits resulting from past Programs, the participation is justified also in the 7th EC-Framework (EURATOM) Program currently being prepared.

6. Outlook

With regard to HLW disposal in rock salt the status of knowledge is well advanced. Some key projects being carried out will have an important impact on the salt concept. Their results will be a milestone for future decision making. The first one, being part of the EC-ESDRED project aims at completing and optimizing the Direct Disposal concept by full-scale demonstrating the emplacement of SF in vertical boreholes. The second project deals with the development of an advanced safety concept for a

HLW waste repository. Performance Assessment exercises were performed within the framework of German R&D projects at the end of the 1980s and in the first half of the 1990s. Since then remarkable developments improved the basis for developing an advanced safety case. A joint R&D project is being performed to identify major needs for further R&D. Moreover, using the approach of proving the safe enclosure without release for the expected evolution of the repository is considered to be more appropriate for a rock salt repository and takes advantage of its specific properties.
During the last few years in Germany the knowledge concerning indurated clay increased. The project results achieved so far, including the Clay study (19) of BGR, the German Geological Survey, allow a better and more qualified estimation and evaluation of the pros and cons of HLW disposal in clay. However, to reach a state-of-the art like in rock salt, there are R&D efforts by projects pursued addressing e.g., conceptual and safety related questions or all problems connected to host rock characterization.
The Research Concept of BMWi is presently subjected to a sort of revision, evaluation, and discussion, inter alia, by experts from several German research institutions. This activity is of special importance against the background of streamlining and focusing the research activities both concerning budget constraints and adaptation to future demands, priorities and perspectives.

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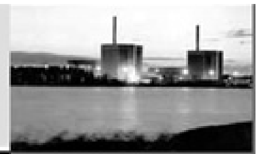

SESSION II : Research, Development and Demonstration for Radwaste Storage and Disposal

PERSPECTIVES FOR DEEP GEOLOGICAL FORMATION DISPOSAL RESEARCH IN FRANCE BEYOND 2006

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ABSTRACT

One finalised aim underlies research conducted on the feasibility of geological disposal: the possibility of having a reversible disposal system available. A model has been drawn up to provide a framework for the analysis and propose possible research content between 2006 and 2015.

This period will correspond to the move from the feasibility to a development, optimisation and detailed study phase. It aims at answering any questions raised by reviewers and develops forward the study of a repository. It will also correspond to the consolidation of scientific data, greater understanding of certain mechanisms and an approach of scientific and technical integration. Another goal of the period 2006-2015 would be to collect elements for a decision on the sitting issue through an extended survey.

This phase of development includes confirmation of the data acquired during the previous phase and over relatively long periods, optimisation of repository concepts and detailed study of their main components, the production of full-scale mock-ups or simulations to validate the main technological design points and refining of data extrapolation methods.

1. Background

The December 1991 Act on radioactive waste management research defines three avenues of research for studying long term management solutions concerning high level long lived radioactive waste. It provides for a milestone in 2006 in order to assess the research conducted over the past fifteen years and define the way forward at the end of this research programme. To this end, Andra submitted the initial and final versions, in June 2005 and December 2005 respectively of its deep geological disposal feasibility report. This report gives a general scheme that could be applied, setting out the underlying assumptions and possible uncertainties. It is based on scientific and technical considerations which, when the time comes, may need to be confronted with the social, political or regulatory framework under which any studies could be conducted.

2. The results of fifteen years research (1991-2005)

At the end of fifteen years of research, the report submitted in 2005 by Andra emphasizes that the basic feasibility of a clay medium repository is achieved. A number of elements support this conclusion.

4.1 The Meuse Haute-Marne site offers favourable geological conditions

The Callovo-Oxfordian layer combines some very useful properties, matching expectations for the design of a repository in a clay medium. Firstly, the layer is very thick (130 metres right below the Meuse/Haute-Marne URL site), with large volumes which are not affected by faults. Its geological history is well-known. Since its deposition this history has been quite undisturbed, which is a major argument for confirming its homogeneity and its extreme stability. Seismicity is very low. The layer contains very little water, whose movement is extremely slow, due to its very low permeability. Physical and chemical characterizations further show that it has a strong ability to retain and trap most of the chemical elements and radionuclides present in the waste. It is suited to excavation mining techniques and underground construction work only causes moderate disturbances, which do not mean, *a priori*, creating preferential flow paths. There is a wide zone of more than 200 km^2 within which these properties exist in principle (so-called transposition zone). The geological medium therefore intrinsically offers favourable characteristics making it suitable for hosting a repository.

4.2 Architectures have been prepared to take advantage of the favourable geological conditions

Engineering studies have defined simple and robust disposal concepts suited to the characteristics of the argillaceous layer, taking the utmost advantage of its qualities. These concepts include cautious choices leading to design margins. The work has not been pursued to the optimization stage, but has established that the proposed architectures were realistic, capable of being constructed and used for waste disposal without any special difficulty. These architectures contain many arrangements which foster overall safety, such as subdividing the repository in various separate zones. Furthermore, operational safety and safety studies based on lessons learnt from other mining or nuclear facilities show the possibility of safe operation without notable impact on the environment.

4.3 Reversibility at the heart of the investigatory approach and expressed in concrete practical terms

The architectures drawn up for the repository have been chosen with in mind a possible reversibility process under the best conditions possible. Andra has developed an approach to reversible disposal which can be defined as the possibility of gradual, flexible management of the repository in stages. The objective is to allow future generations freedom of decision in waste management. Consequently, Andra has opted not to set a preconceived duration for reversibility. This involves offering as great a flexibility as possible in the management of each stage, allowing for the possibility of maintaining the status quo before deciding on the next stage or going backward. The design of the repository (modular architecture, simplified operation, design and choice of durable materials, etc.) aims at allowing the widest possible choices. A period of reversibility of several centuries could be envisaged through monitoring and conventional maintenance work.

4.4 The safety analysis shows the absence of significant impact on the environment

Would closing the repository be decided, a detailed assessment has been made of its behaviour over time and its possible impact on man and environment. On the basis of the acquired scientific data and the proposed repository architectures, a post-closure analysis of the evolution of the repository has been performed. The evolution of the repository under normal conditions has been represented and modelled using computational tools integrating recent advances in digital simulation (ALLIANCES platform). The objective was to examine the efficiency of the repository safety functions. Using various indicators, the analysis shows that the main safety functions ("preventing water circulation", "limiting the release of radionuclides and immobilising them" and "delaying and attenuating migration") were fulfilled by the proposed system. The cautious (conservative), or even pessimistic choices made provide significant safety margins. Thus, all the assessments display a high degree of robustness. The analysis has shown that these conclusions were valid not only in normal situations, representative of the most probable evolutions, but also in altered, therefore considerably more pessimistic, configurations.

At the end of the calculations carried out as part of the safety model under normal evolution, repository performance meets, the dose limits recommended by The Basic Safety Rule III.2.f, with large margins. In conclusion, the safety approach supports the repository feasibility study.

3. The basis of the 2006-2015 program on the disposal in the Callovo-Oxfordian formation

Following its production in June 2005, the "Dossier 2005 Argile" has been extensively reviewed by the French National Review Board (CNE), the French Safety Authority, an international review team established by the NEA as requested by the French Government as well as by the scientific committee of Andra. A national debate was also organized mainly during the last trimester of 2005 the conclusions of which were synthesised in a report.

These different reports were transmitted to the Parliamentary Office of Assessment of Scientific and Technological Choices (OPECST) and the concerned French authorities in order to prepare the 2006 Act on radioactive waste management which was voted on June 28, 2006. This law provides Andra with new milestones and more specifically indicates that the Agency will have to provide a request for authorization of construction of an underground disposal, that could be evaluated in 2015.
Andra also carefully analysed the evaluation reports in order to take into account the various recommendations in the revised versions of its scientific, technical and experimental programs. The final and detailed organisation of the work which will be conducted by Andra during the next 9 years will be completed by the end of 2006.

4. The 2006-2015 program

The proposed program aims at defining the main detailed design elements of a possible repository and carrying out pre-industrial tests. This includes confirmation of the data acquired during the previous phase and over relatively long periods, optimisation of repository concepts and detailed study of their main components, the production of full-scale mock-ups or simulations to validate the main technological design points and refining of data extrapolation methods. It covers the detailed definition, testing and optimisation of disposal design options.

From a more scientific viewpoint, research essentially deals with two major problems:
- changes in scale. This means examining the validity, at larger scale, of data acquired over limited intervals of time and space, as well as specifying the location that could be envisaged,
- validation of the understanding of phenomena and their interactions (full-scale and *in situ*) and their modelling (detailed analysis of couplings).

From a technological viewpoint, the important issues are related to the construction of full-scale repository infrastructures and as well to handling or monitoring operations. In this respect, the Meuse/Haute-Marne laboratory continues to acquire data in order to confirm the previously obtained ones and conduct technological tests directly within the medium under consideration.

Main efforts will thus be focussed on:
- The qualification of the Meuse/Haute-Marne site, with data validation through long-term experiments, data extrapolation and extended geological survey (borehole drilling and geophysical campaign) in order to delimit as accurately as possible the geological zones of interest to site the possible repository facility.
- The consolidation and initial optimisation of engineering studies, with a view to drawing up a full report on the planned installations and processes, as well as the detailed study of the facility main components,
- The technological testing, and production of industrial scale demonstrators or prototypes,
- The scientific programmes underlying optimisation of concepts and experimental testing,
- The definition of appropriate measuring instruments and operational simulation tools,
- The consolidation of safety assessments and more accurate quantification of safety margins.

4.1 Consolidation and optimisation of engineering studies

The engineering studies conducted in the previous phase have enabled the specific components of a disposal facility (installations and processes) to be basically designed. This new phase will allow an overall and detailed view of the installations necessary for the repository operation:
– the underground structures will be defined more precisely, in terms of their dimensioning, construction methods and equipment, so that they can be reproduced experimentally in the underground laboratory (demonstrators). The aim will be to achieve a technical-economic

optimisation of underground installations. Finally, a detailed definition (plans, dimensioning) and design justification report will be produced;

– the surface nuclear facilities previously studied in terms of their construction and operating principles will be studied in more detail and all the surface arrangements will be briefly examined. This will include nuclear buildings (reception and surface storage of packages, disposal package conditioning facilities, etc.) and technical buildings (current, fluids, etc.);
– the transfer, emplacement and possible retrieval of disposal packages will be studied in more depth, in particular for the design of the necessary systems, which will be tested (demonstrators);
– operating safety studies will be continued linked to the previous ones.

The engineering studies will also have to examine in detail how the different activities fit in with each other, particularly in terms of the timing of construction and operation activities. Special emphasis will be put on the reversibility analysis.

4.2 Accompanying scientific programmes

The design of the repository, particularly its dimensioning, is highly dependent on the way the geological medium responds to the perturbations or disturbances it causes. In this respect, additional studies need to be conducted on several thematic areas. The following should be specifically mentioned:

- detailed understanding of near-field thermo-hydro-mechanical interactions, in particular for C waste (vitrified waste) disposal cells with their large number of interfaces. These studies will be closely associated with technological demonstration activity. They will more specifically deal with the first hundreds and thousands years of evolution of the repository with a special focus on the reversibility period (few centuries)
- the gas management with a better appraisal of their production by packages and of their behaviour within the cells, taking into account coupled effects;
- thermal dimensioning criteria, referring in particular to geochemistry and thermodynamic studies;
- the mechanistic approach to radionuclide transport processes in order to make a link between the different scales of investigation and facilitate data transposition;
- specific formulation of materials (structure concrete, engineered barrier buffer clay and backfill materials) related to their expected performances and their implementation in defined concepts; these studies will also be closely associated with technological demonstration activity;
- the detailed modelling of the behaviour of waste matrix and packages in repository conditions.

4.3 Technological tests, demonstrators and industrial prototypes

Technological testing in the underground laboratory has two aims: on one hand, *in situ* and full-scale testing of structure construction processes and the corresponding techniques and tools; on the second hand, validation under *in situ* and full-scale conditions of the scientific knowledge previously acquired from samples or at intermediate scales. These full-scale tests are part of the progressive change of scale approach linked to design iterations.

To this end, structures could be excavated starting out from the underground laboratory infrastructures. Furthermore, additional exploratory drifts will be excavated. Beyond geological survey, they would provide feed-back for industrial-scale excavation method design (technologies becoming progressively more industrial). Most of these demonstration operations will last beyond 2015.

Several tests are scheduled:

- the construction of C waste cells, for excavation and sleeve emplacement. Waste package emplacement and possible retrieval will also be tested. With the installation of heating devices, thermal behaviour measurements would be recorded, in particular to observe thermo-hydro-mechanical phenomena around disposal cells;
- the construction of a B waste cell, through excavation techniques with industrial scale potential (implementation, performance, safety), for geomechanical characterisation and verification of compatibility with package handling processes;

- the construction of a drift with a normal section (access drifts), in which it will be possible to test the excavation process;
- a full-scale sealing test (for example in the previous drift), including the installation of the anchoring key, seal core emplacement and performance tests.

Beyond the demonstration of construction or closure operations, it is also important to test certain processes or technical devices specific to the operating phase, in order to control their actual performance. For this, prototypes can be made and tested in a surface facility (in the workshop, for example), then possibly later underground beyond 2015.

4.4 Measuring devices and simulation tools

The underground installations of a geological repository will have to be regularly monitored and checked. This involves the observations and measurements necessary for the purposes of maintenance and maintaining safety, on the one hand, and gaining the knowledge of repository behaviour and evolution necessary for a step-wise reversible disposal management, on the other hand. The instrumentation technologies and measuring devices required for this purpose will have to be developed (when they do not exist) or tested *in situ*.

The simulation programme undertaken in cooperation with the CEA during the previous period and aiming at installing a simulation platform, will be continued. The objective is the availability of an effective tool suited to the detailed design and monitoring of a disposal facility. Developments should provide an operational tool for safety calculations, which could integrate the growing complexity of coupled thermo-hydro-mechanical and chemical phenomena as they become understood.

4.5 Qualification of the site: validation of data over the long term, extrapolation of data and geological survey at repository scale

The feasibility phase enables the behaviour of the repository, its components and the surrounding rock to be understood in the medium or long term by extrapolations or mechanistic modelling based essentially on data acquired for the short term. In order to achieve optimisation and reliability of the approach (engineering, safety), the acquisition of experimental data over longer periods of time is essential. To do this, measurements will still be recorded on experiments started before 2006. This will allow long-term records in order to validate the main parameters (geomechanical, thermal, gas transfer, diffusion, etc.). Moreover, new data will be obtained from experiments started before 2006 which had not provided yet complete results; this concerns, for example, long-term diffusion experiments, gas migration tests or observation of the behaviour of materials *in situ*.

It should be possible to transpose data obtained in the underground laboratory to the geographical area within which a repository could be sited. The transposition carried out during the feasibility phase led to the definition of a zone representing a 200 Km^2 equivalent geographical area, in which the data obtained inside the underground laboratory zone can be transposed. Within it, it is necessary to specify a zone with the most favourable geological, hydrological, geochemical and geomechanical characteristics in terms of the confinement provided by the layer and of the hydrogeological pattern. Based on a combination of direct and indirect surveying methods (borehole drilling and 2D high definition seismic measurements), the program will aim at completing the characterisation of this zone, in order to propose a more restricted one which could correspond to a possible repository footprint and on which more detailed geological exploration would be carried out (seismic 3D).

5 Summary

The studies conducted over the 2006-2015 aim to prefigure an industrial scale project. The technical report produced at the end of this phase will give an overview of the planned installation. It will provide its detailed definition, supported by scientific knowledge and technological tests for the most significant elements. Under these conditions, a draft preliminary safety report could be issued. The concerned geological medium will have been surveyed and all the elements necessary for sitting a possible facility will be available.

DISPOSAL OF SPENT FUEL FROM GERMAN NUCLEAR POWER PLANTS – PAPER WORK OR TECHNOLOGY?

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ABSTRACT

The reference concept “direct disposal of spent fuel” was developed as an alternative to spent fuel reprocessing and vitrified HLW disposal. The technical facilities necessary for the implementation of this reference concept – the so called POLLUX-concept, e.g. interim storages for casks containing spent fuel, a pilot conditioning facility, and a special cask “POLLUX” for final disposal have been built. With view to a geological salt formation all handling procedures for the repository were tested aboveground in a test facility at a 1:1 scale. To optimise the concept all operational steps are reviewed for possible improvement. Most promising are a concept using canisters (BSK 3) instead of POLLUX casks, and the direct disposal of transport and storage casks (DIREGT-concept) which is the most recent one and has been designed for the direct disposal of large transport and storage casks. The final exploration of the pre-selected repository site is still pending, from the industries point of view due to political reasons only. The present paper describes the main concepts and their status as of today.

1. Introduction

Since the late seventies the German industry has been developing an alternative to spent fuel reprocessing and vitrified HLW disposal. It is called "direct disposal of spent fuel". The feasibility was examined and safety aspects were evaluated. The Federal Government concluded that the technology of direct disposal had to be developed. A reference concept based on the triple purpose cask POLLUX for transport, storage and final disposal as well as a conditioning technique that separates fuel rods from the structural parts of the fuel assemblies were developed.

In 1994 the Atomic Energy Act was amended allowing the direct disposal of spent fuel without prior reprocessing. The assumed cost advantages of direct disposal of spent nuclear fuel compared to reprocessing gave reason to follow the path more and more. Moreover, the concept of phasing out nuclear energy was pursued by a red-green government and on June 30th, 2005, the delivery of spent fuel to the reprocessing facilities was abandoned. So, as of today, direct disposal is the only route of spent fuel out of nuclear power plants.

Competition among various techniques of electricity generation forced utilities and the nuclear industry to strictly control their costs and to finish unsettled issues as far as possible. All cost saving potentials at all steps along the route towards direct disposal have to be explored. At the same time the highest levels of safety standards as well as flexibility for future improvements, decisions and regulatory needs have to be maintained. So, the reference concept is still under investigation and potentials for optimisation have been identified.

The main and today most promising option is called BSK 3-concept. The most recent option under investigation is called DIREGT-concept, and aims at the direct disposal of large transport and storage casks. The main steps of these three options are shown in figure 1.

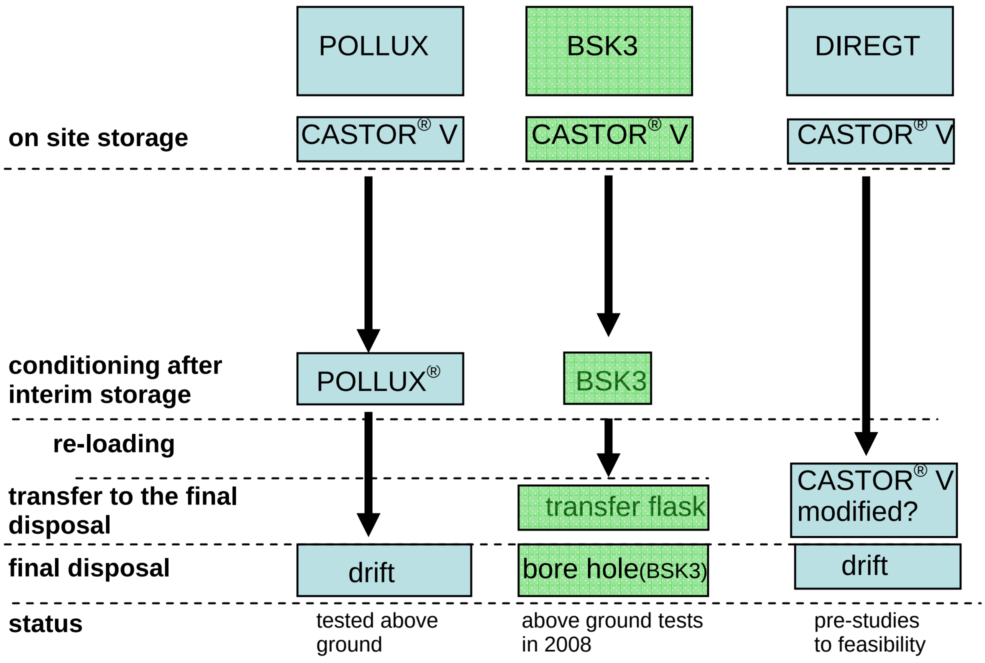

Fig. 1 Direct disposal of spent fuel: Reference concept and two alternatives under investigation

2. Disposal scheme for high-active and/or heat-generating wastes

All options for direct disposal must be seen in context with disposal of reprocessing residues (s. figure 2). About 6,000 tHM will be reprocessed and about 10,000 tHM will go to direct disposal. These quantities are calculated based on the regulations laid down in the Atomic Energy Act that allows the operation of an existing nuclear power plant for about 32 years. No new nuclear power plants will be licensed.

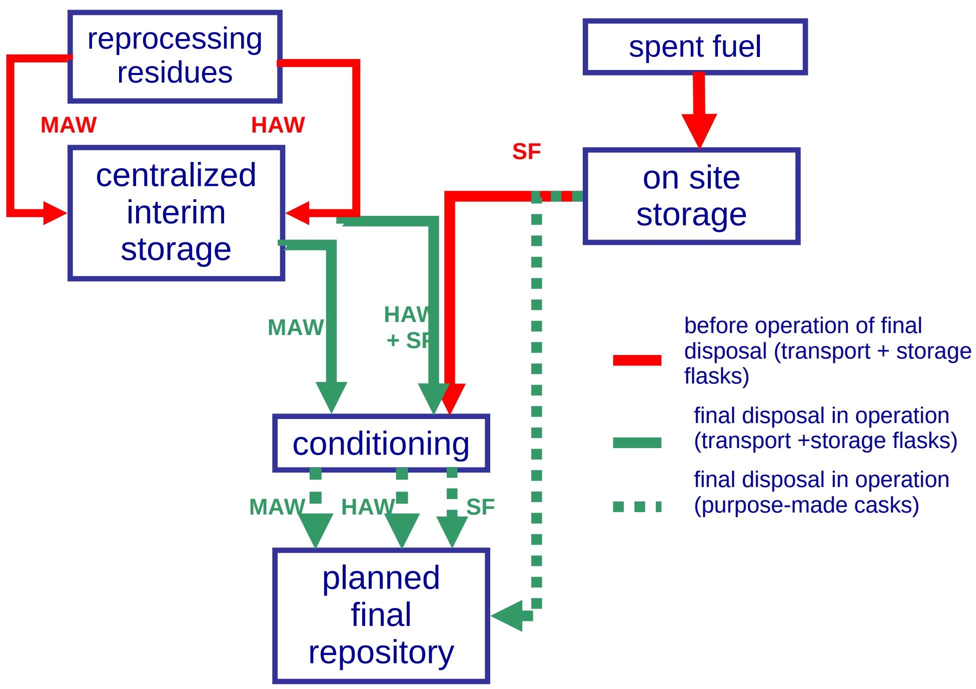

Fig. 2 Routes of waste concerning SF

In total, reprocessing will lead to about 4,000 HAW-canisters containing vitrified fission products and about 6,000 MAW-canisters containing compacted structural parts of fuel assemblies. Prior to disposal they will be stored in large transport and storage casks. A total of about 300 casks containing HAW- and MAW-residues is expected, 63 casks with HAW are already in storage at the Gorleben site. The disposal concept of today envisages to place all canisters in boreholes. So, canisters have to be transferred from the transport and storage casks into a transfer cask that serves as "shuttle" between the de/reloading station and the repository. The transfer cask is transported into the repository just onto the top of the borehole, the canisters are placed into the borehole and the transfer cask is ready for the next transfer. De/reloading of transport and storage casks was planned to be carried out in the same conditioning facility that is intended to be used for the direct disposal route.

The reference concept POLLUX for fuel assemblies also relies on large transport and storage casks and on the same type of interim storage, but differs concerning the final disposal technique.

3. Spent fuel disposal using disposal casks in the drifts of a repository: The reference concept POLLUX

Main steps are (s. figure 1):

- loading of spent fuel into transport und storage flasks after spent fuel has sufficiently cooled down in the pools of the nuclear power plant
- transfer of the flasks into an interim storage at the site of the nuclear power plant
- interim storage of flasks until conditioning or final disposal is possible
- transport of flasks to the conditioning facility or the final disposal site
- conditioning by separating fuel rods from structural parts of the fuel element
- final disposal of fuel rods and structural parts

3.1 Transport and Storage of Fuel Assemblies

To minimise costs of transport and interim storage, large casks, mainly of the CASTOR® V-type, are used. These contain up to 19 fuel elements from PWR and 52 fuel elements from BWR. The cask body is made of ductile cast iron, neutron shielding is achieved by polyethylene bars assembled in uniformly distributed drillings in the cask wall. So far, more than 100 casks have been loaded with fuel assemblies in Germany, they are stored on site, except for those 6 casks that are being stored at the Ahaus site and those 5 casks that are being stored at the Gorleben site.

3.2 Pilot Conditioning plant

The pilot conditioning plant at the Gorleben site was completed in 1999. License for operation was granted in 2000. According to the license the plant is operated in a stand-by modus to accept and repair casks when necessary. As of today, the throughput is limited to 35 tHM per year. --- Fig. 3 shows the hot cell where fuel assemblies are separated into fuel rods and structural parts. This, and their subsequent packing into cans is necessary to ensure sub-criticality in the repository.

Fig. 3 Hot cell of the pilot conditioning plant

3.3 Final Disposal Cask

The cans containing the fuel rods are inserted into the POLLUX cask. The POLLUX cask was developed as a triple-purpose cask for transport, storage and final disposal and is to be disposed of in drifts of a salt dome repository. The safety analysis report and the licensing documents according to the regulations of the Atomic Energy Act were submitted to the licensing authority and its independent expert for obtaining the flask approval certificate according to the transport regulations (type B(U)F) and the storage license according to the acceptance criteria of the Gorleben interim store. A drop test programme was carried out in 1994. The application was withdrawn due to the fact that precise and reliable design requirements cannot be identified until the exploration of the Gorleben salt dome is completed. Figure 4 shows the basic design of the POLLUX final disposal cask.

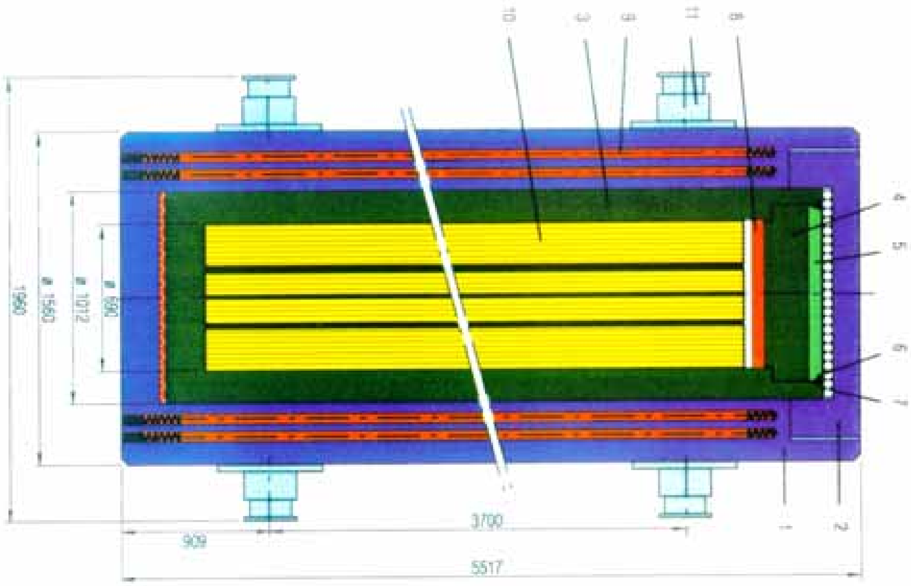

Fig. 4 POLLUX-cask for final disposal of fuel rods from spent fuel

It consists of the shielding cask with an inscrewed lid and an inner cask with bolted primary and welded secondary lid. The fuel rods are inserted into the POLLUX-cask in cans. The cylindrical wall and the bottom of the inner cask are made of fine-grained steel and extruded in one piece. The body of the shielding cask also consists of one piece and is made of ductile graphite iron. Two rows of boreholes in the shielding casks wall contain neutron moderator material. A prototype cask has undergone tests in the Pilot Conditioning plant in Gorleben. The cask will contain fuel rods from up to 10 PWR fuel assemblies or fuel rods from up to 30 BWR-fuel assemblies.

3.4 Tests for drift disposal with the POLLUX-cask in an aboveground test facility

These tests performed in 1994/1995 were aimed at demonstrating that rail-bound handling, horizontal transportation, and drift disposal of shielding-disposal casks with a weight of 65 t (POLLUX) loaded with spent fuel assemblies are technically feasible. Here, emphasis was put on the development and construction of components, such as an emplacement device, a transport cart and a mining locomotive. Their capabilities of working under normal operating conditions and under conditions of operational disturbances were demonstrated at a full scale aboveground test facility in order to guarantee the safe handling of waste packages. Figure 5 shows the mining locomotive, transport cart, emplacement device and the dummy cask.

Fig. 5 Components of the disposal system

The transport and battery-operated disposal locomotive is state-of-the-art. It is constructed in so-called open order with heading and trailing cabin and is the result of efforts to improve mining locomotives from ergonomic, operational and safety-technological points of view. The transport cart was built with a special carriage to shift the center of gravity of the POLLUX cask to a lower position. The emplacement device for drift disposal (ELVIS) was made of components that are used under comparable conditions in other technical fields and was designed for a load of 65 t. The dummy cask which was used for the tests has all the features of the POLLUX-cask.

4. Spent fuel disposal using borehole-technology: BSK3-concept

To simplify the operation of the repository and to avoid large and expensive disposal casks the BSK3-concept was developed. It is also based on the conditioning of fuel assemblies, but instead of using the POLLUX cask fuel rods are inserted into a canister that can be placed in boreholes. Thus, the same handling procedures and techniques that are used for the disposal of reprocessing residues can be used for the final disposal of spent fuel. The concept of the transfer cask has only recently been developed and is still being refined. Supported by the EU and the German Federal Ministry of Economics and Technology, DBE TECHNOLOGY GmbH is preparing an aboveground test facility to simulate all relevant handling procedures. The German utilities are involved via GNS and will provide all necessary hardware components. Figure 6 shows a sketch of the BSK 3. The BSK 3 is designed to contain fuel rods from up to 3 PWR-fuel elements or from up to 9 BWR fuel elements.

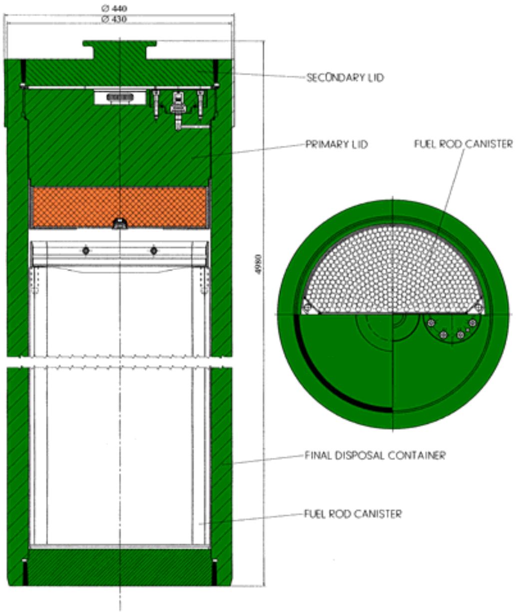

Fig. 6 Sketch of a BSK 3

Preliminary studies show that the costs for the repository operation can be reduced significantly and the total costs for casks for spent fuel can be reduced by about 50%.

5. Direct disposal of large transport and storage casks: DIREGT-concept

Based on the estimated quantity of HM a total of about 900 – 1000 transport and storage casks for spent fuel and about 300 transport and storage casks for reprocessing residues (mainly CASTOR®-casks) will be stored in interim storages and scrapped later if the POLLUX-concept or concepts based on borehole-disposal technology (BSK 3- concept or borehole-technology for reprocessing residues) are used. The feasibility of the direct disposal of transport and storage casks is being investigated. The implementation of this concept - called DIREGT-concept in figure 1 – would offer the potential to avoid the separation of fuel rods from structural parts and would avoid custom made final disposal casks. Moreover, it has been estimated that the conditioning and the de/reloading process will be most probably the bottleneck in terms of time, and the conditioning time may define the operation time of the repository. It is investigated whether safe sub-criticality can be ensured with final disposal of transport and storage casks (TSC) in the drifts of the repository. Results achieved so far give rise to an optimistic answer. In addition, it is investigated whether the heat transfer from the casks meets the requirement that the temperature in the salt formation stays safely below 200° which is the critical temperature for the salt formation in Gorleben. The crucial point will be the aboveground interim storage time necessary to let heat production rates of TSC fall below acceptable rates and whether those time spans are acceptable. All investigations and studies performed so far, do not exclude the feasibility of direct disposal using large transport and storage casks. Technical equipment for the shaft transport and the transport in the drifts of the repository has been judged as feasible. Work will continue, final results of the relevant research programme financed by GNS on behalf of the German utilities are expected in 2008. This concept has a cost saving potential which is even greater than that one of the BSK 3 concept.

6. Summary and Conclusion

The technical features for the direct disposal of spent fuel from German nuclear power plants have been thoroughly investigated and a reference concept has been established. Facilities are in operation or could be operated from the technical point of view. Concepts for optimisation show a clear cost saving potential; they may simplify processes and lower operational risk without any compromise in longterm radiological safety aspects. Test programmes will be performed within the next years. So, the answer to the title's question "paperwork or technology?" must be "technology". The missing link "exploration of a final disposal repository" and "operation of a selected final disposal site" does not lie within the responsibility of the utilities, but - as laid down in the Atomic Energy Act - lies within the government's responsibility. The German utilities are ready to solve the task "disposal of spent fuel" within the next 20 years. So, they call for a restart of the exploration of the Gorleben salt dome as soon as possible. Figure 7 shows the technical facilities at the Gorleben site: In the background the Pilot Conditioning Plant, the Interim Store, and in the front the site of the exploration mine Gorleben.

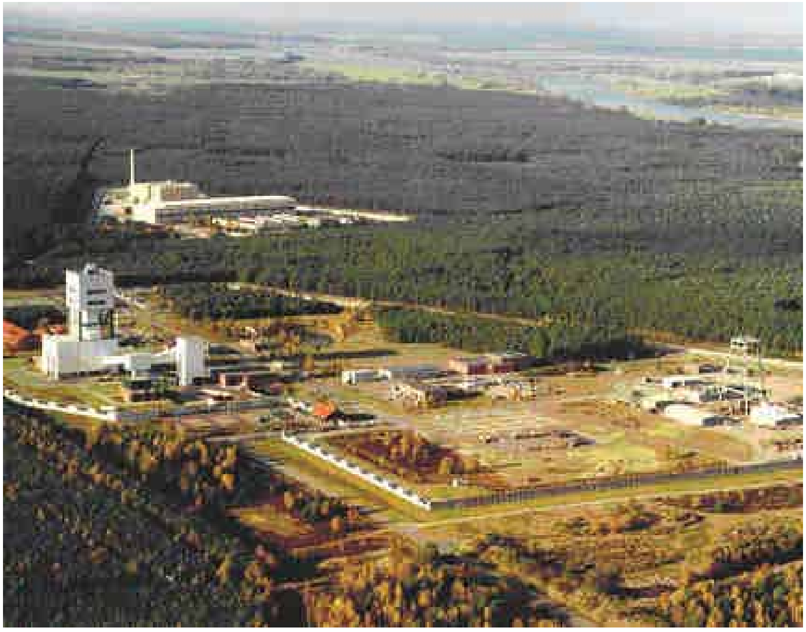

Fig. 7 Facilities for the direct disposal of spent fuel at the Gorleben site

DISPOSAL OF SPENT FUEL IN VERTICAL BOREHOLES IN SALT - AN INDUSTRIAL DEMONSTRATION PROGRAM -

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ABSTRACT

In order to synchronize and optimize the emplacement technologies used for both categories of waste (vitrified waste and spent fuel) the borehole emplacement technique for consolidated spent fuel was reconsidered in Germany. The appropriate design resulted in a fuel rod canister of the type “BSK 3”. This BSK 3 steel canister is of the same diameter as the standard HLW-canister which allows the use of a common transfer and handling technique. The canister is tightly sealed by welding and designed to withstand petrostatic pressure. Thermal calculations showed that the emplacement of a BSK 3-canister into a vertical borehole in a salt repository is possible after only about 3 to 7 years after reactor unloading of the fuel assemblies. Thus, the emplacement of BSK 3-canisters allows a complete revision of the schedule of spent fuel disposal concepts and may lead to a considerable reduction of time and costs. Accordingly, a research program was launched in 2004 within the framework of the 6th European Research Program to develop the transport and emplacement components, the functionality and reliability of which are to be tested in a one-year demonstration phase which will commence at the beginning of 2008.

1. Introduction

In the context of the repository development program for the Gorleben site the vertical borehole emplacement concept was developed as one option for the disposal of heat generating waste in rock salt. In this disposal concept unshielded canisters containing vitrified HLW or spent fuel are to be emplaced in vertical boreholes with a depth of 300 m which are situated in disposal zones at a depth of 870 m. The disposal zones consist of several disposal fields, each comprising several disposal drifts connected by transport drifts. Each disposal drift consists of a certain number of equally spaced boreholes. In order to facilitate the fast encapsulation of the waste by rock salt, no lining of the boreholes is planned. Figure 1 shows the disposal concept for the emplacement of waste canisters into vertical boreholes in a repository in rock salt.

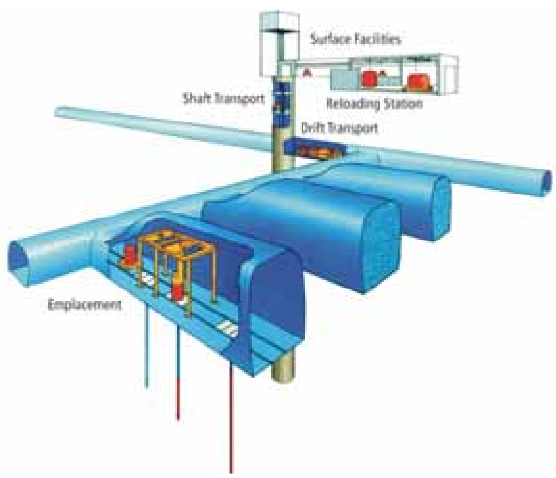

Fig. 1 Disposal concept for the emplacement of waste canisters into vertical boreholes in rock salt

To synchronize and optimize the emplacement technologies for both categories of waste (vitrified waste and spent fuel) the borehole emplacement technique for the direct disposal of spent fuel rods was reconsidered. The corresponding design resulted in a canister of the type “BSK 3”. This steel canister is of the same diameter as the standard HLW-canister which allows the use of a common transfer and handling technique. The canister is tightly sealed by welding and designed to withstand petrostatic pressure. The emplacement of BSK 3-canisters into vertical boreholes of a salt repository is possible after only about 3 to 7 years after reactor unloading of the fuel assemblies.

Accordingly, it was decided to set up an RTD-program to develop the necessary components for demonstrating and testing the functionality and reliability of a suitable emplacement technology. The research project is funded by the German Ministry of Economics and the German Nuclear Industry. In the context of an EC call for proposals for Integrated Projects in waste management in 2003, the BSK 3 emplacement concept was recommended to a consortium of mainly European Waste Management Organisations and accepted eventually as part of the five-year Integrated Project ESDRED. 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).

2. Types of waste canisters

The high level waste to be taken into account in the repository layout in Germany consists of three different types (see figure 2): Heat generating vitrified HLW which is stored in standard Cogema canisters, low heat-producing highly-active technical waste, i.e. mainly hulls and claddings, which is stored in CSD-C canisters, and spent nuclear fuel rods which will be packed in BSK 3-canisters. BSK 3-canisters are designed to hold the fuel rods of three PWR or 9 BWR fuel assemblies. The BSK 3 consists of a cylindrical steel canister with a height of 4.99 m and a diameter of 0.43 m. The wall thickness is 50 mm and the total mass is 5,200 kg. The quantities of waste canisters considered for repository design amount to:

HLW-canisters:	4,778
CSD-C-canisters:	8,764
BSK 3-canisters:	5,525

3. Emplacement system

The emplacement system developed for the handling and disposal of BSK 3-canisters comprises a transfer cask which provides appropriate shielding during the transport and emplacement process, a transport unit consisting of a mining locomotive and a transport cart, and an emplacement device. During the conceptual phase, different options were taken into consideration in order to find a technically feasible and safe emplacement system. Figure 2 shows the components of the system selected for the transport and emplacement of BSK 3-canisters in an underground emplacement drift. The emplacement device swivels the transfer cask from the transport cart, tilts the cask into an upright position and lowers it down onto the top of the borehole lock.

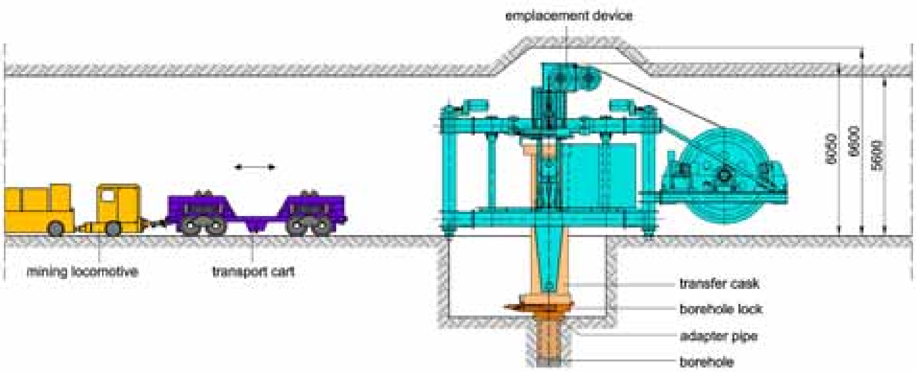

Fig. 2 Transport and emplacement system for BSK 3 canister disposal

3.1 Transfer cask

Aboveground, the BSK 3-canister is inserted into the transfer cask (Figure 3) which is transported by the transport cart through the shaft to the emplacement drift underground. The mining locomotive drives the transport cart with the transfer cask to the emplacement device. The body of the transport cask consists of a thick-walled hollow cylinder made of cast iron with nodular graphite (GJS) or cast steel. The wall thickness and wall structure of the cask body are designed in accordance with the requirements concerning mechanical strength and gamma and neutron shielding.

There are two drill hole lines in the cask wall on separate circles which are filled with polyethylene rods which serve as neutron moderators. The neutron shielding at the base and lid is effected by flattened disc-like neutron moderators. Direct radiation transitions are avoided by means of constructional measures. The locks are made of stainless steel and are screwed to the cask body. The flat slide latches integrated into the locks work like drawers and run in slide bars. When in locked position the flat slide latch is kept in place by two locking bolts set into the side walls.

The transfer cask does not have any inherent mechanism to open and operate the flat slide latches. Cask opening and closing is effected at the base by means of the borehole lock mechanism and at the lid by means of the emplacement device (shielding cover). During the lowering processes opening bolts in the borehole lock and shielding cover mechanism slide into recesses in the latches and open them. At the cask base the opening bolts of the borehole lock are part of the borehole latch. The opening of the borehole causes the simultaneous opening of the transfer cask.

There are two diametrically positioned trunnions at the ends of the cask body to facilitate the handling of the transfer cask. The trunnions have an additional offset collar next to the collar which during transport serves to fasten a lifting sling when in the surface transfer area and to allow it to be picked up by the emplacement facility.

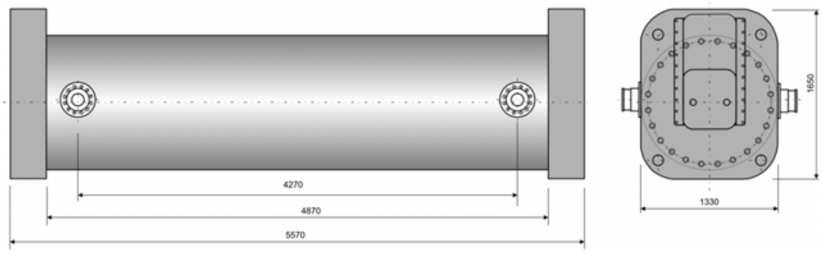

Fig 3 Transfer cask

3.2 Transport cart and mining locomotive

The transport cart (Figure 4) is used for on-site transportation of the transfer cask from the surface to the emplacement location. It has four axles and a pick up attachment under the middle of the chassis for transportation by stationary lifting gear and a coupling at each end for traction units.

The inner collar of the transfer cask trunnions serves as a bearing when the transfer cask is transported by the transport cart. The design of the frame parts of these bearings takes the required operating space for the loading (swivel girder) of the emplacement facility into consideration. To lift the transfer cask free from the transport cart half-covers, a lifting distance of 200 mm is necessary. This distance makes allowance for a 30 mm protrusion of the half cover hinges above the middle of the carrying-pegs and a max. spring relaxation of the transport cart of 20 mm.

For the experiments the existing battery-powered locomotive will be used. In the final repository the locomotive gage will have to be widened

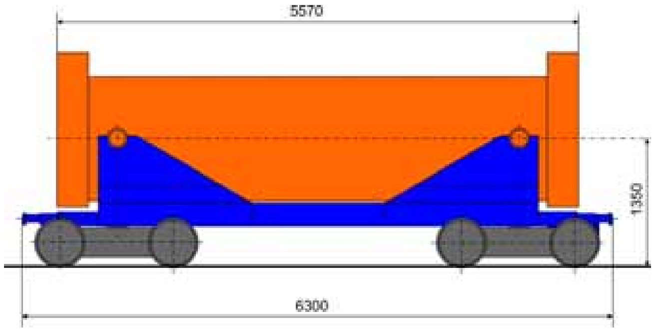

Fig. 4 Transport cart

3.3 Emplacement device

The emplacement device, Figure 5, is the central device of the entire emplacement process. Accordingly, it is equipped with all means for the safe handling of the transfer cask and BSK 3-canister. In a final repository an emplacement cycle usually includes the following procedures: The BSK 3-canister is delivered in the transfer cask on the transport cart. In the emplacement drift the transfer cask is lifted off the transport cart by the emplacement device and swivelled into an upright position after the transport cart has been removed. After lowering the transfer cask onto the borehole lock and opening the transfer cask and borehole lock, the BSK 3-canister is lowered to the planned position in the borehole with the canister grapple. The canister grapple is removed and the transfer cask and the borehole lock are closed. After swivelling the transfer cask into a horizontal position, the transport cart is again driven under the emplacement device and the transfer cask is placed on the wagon. Finally, the transport cart and transfer cask are driven out of the emplacement drift for reloading. Accordingly, the emplacement device basically consists of the following assembly units:

- Lifting gantry
- Flap-frame with controls
- Swivel gear
- Canister lifting gear including hoist cable and lifting tackle
- Shielding cover

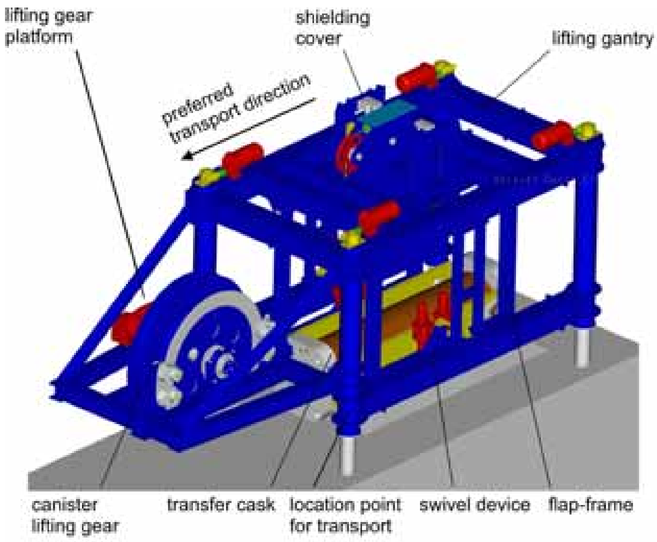

Fig. 5 Emplacement device

3.4 Borehole lock

The borehole lock (Figure 6) provides the sealing of the borehole and consists of a body and a flat slide latch as well as the equipment for the slide latch guidance and the slide motor. The upper part of the body is collar-shaped to take the transfer cask. Four guide pins at the base of the collar help line up the transfer cask. The flat slide latch is a massive block-shaped steel body which is moved by a spindle and a geared engine unit. The flat slide latch has two opening bolts to unlock the borehole lock and to mechanically connect it to the transfer cask's locking latch.

The lower part of the body is flange-shaped to allow its connection to the borehole support pipe. In the upper, inner part of this offset flange ducts arranged in segments carry used air from the borehole via a ring channel to the connection with the ventilation plant.

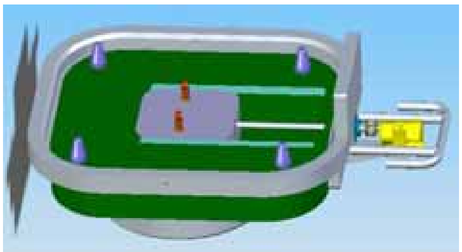

Fig. 6 Borehole lock

4. Demonstration tests

The demonstration tests will be performed in the former turbine hall of a power station in the village of Landesbergen in the vicinity of Hanover, Lower-Saxony. The testing procedure is to simulate the feasibility of emplacing BSK 3-canisters in boreholes of up to 300 m depth in salt rock.

In the demonstration test facility, emplacement will be simulated for two depths. The borehole will be simulated by a sheet steel metal casing. Only one BSK 3-canister will be used for the tests. In the first test, the BSK 3-canister is lowered down to the lower position in the borehole. The canister grapple of the emplacement device is released and removed. Subsequently, the BSK 3-canister is removed from the borehole and placed into the transfer cask on the transport cart.

In the second test, the BSK 3-canister is lowered down to the upper emplacement position in the bore-hole which is simulated by installing a seal into the sheet steel metal casing. As in the first test, the BSK 3-canister is lowered into the borehole and then removed again.

5. Outlook

The components for the demonstration test will be manufactured in 2007 and will be available for test purposes at the beginning of 2008. Thus, the test series will start with the site acceptance test followed by functionality and demonstration tests. During the test series which will last for 10 - 12 months, the functionality and reliability of the components as well as of the entire system will be tested. Eventu-ally, a qualified emplacement system should be available for industrial application.

KBS-3H – DEVELOPMENT OF THE HORIZONTAL DISPOSAL CONCEPT

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ABSTRACT

SKB and Posiva are performing an R&D programme over the period of 2002-2007 with the overall aim to find out whether the KBS-3H concept can be regarded as an alternative to the KBS-3V concept for disposal of spent nuclear fuel. A feasibility study of the KBS-3H concept was carried out in 2002, followed by the setting up of basic design in 2003, and since 2004 the demonstration phase is ongoing, ending with the evaluation of the potential of the concept in 2007. In order to find out whether the concept can be regarded as a viable alternative to the KBS-3V concept demonstration and design work involve development of excavation technology of the drift, detailed studies on the function of the buffer bentonite, deposition equipment and methods for construction of low-pH shotcrete plugs. The investigations related to long-term safety are based on difference analyses between KBS-3V and KBS-3H and focus on KBS-3H specific processes. By the end of 2007 the KBS-3H concept will be reported including a preliminary safety case of the concept with Olkiluoto in Finland as a reference site.

1. Introduction

In 2001 SKB prepared an R&D programme /1/ for the KBS-3H (horizontal deposition) concept as an alternative disposal method to KBS-3V (vertical deposition). Both methods are based on the so-called KBS-3 concept with multi barrier systems but the orientation of the canister differs, Fig. 1. The repository for spent fuel is planned to be constructed at a depth of about 400-500 metres in crystalline bedrock.

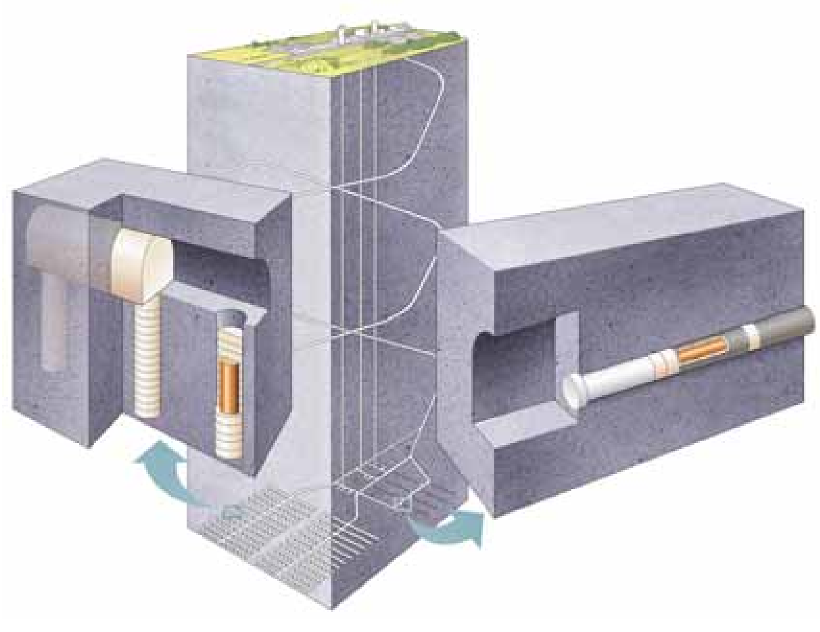

Fig. 1. *Principal for the multi barrier KBS-3 concept showing both the vertical deposition (KBS-3V) and the horizontal deposition (KBS-3H).*

In the fall of 2001, the boards of SKB and Posiva decided to start the common R&D programme on the alternative disposal method KBS-3H. The purpose of the programme carried out over the period 2002-2007 is to find out if KBS-3H can be regarded as an alternative to the KBS-3V concept. In order to fulfil the needs to evaluate the potential of the concept and if the R&D work should continue with the aim to become a viable alternative to the KBS-3V concept the following main activities are being carried out:

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

It should be noted that the deposition equipment since February 2004 is 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) on nuclear energy (2002-2006)".

The development of the KBS-3H design and compilation of the safety case for the concept are the main topics of this paper.

2. KBS-3H concept

In the KBS-3H repository concept, multiple canisters containing spent fuel are emplaced in approximately 300 m long deposition drifts, slightly inclined towards the transport tunnel (Fig. 2). Each canister, with its surrounding bentonite buffer and a perforated steel shell, called supercontainer, is assembled in a handling cell in a cavern in the central area at repository level prior to emplacement in the drift. Each supercontainer is placed between two compacted bentonite distance blocks. 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 supercontainers from each other hydraulically, thus preventing the possibility of pathways for flow and advective transport along the drift.

The transport of the supercontainer in the drift is based on lifting of the container with a pallet with water cushions and alternating relative movement of the container and slide plate. The same machine will also be used for the transport of the distance blocks.

The spent fuel canisters are identical in both KBS-3V and KBS-3H and also the buffer material but the introduction of the steel supercontainer and also the serial emplacement of supercontainers in a 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 differ in some aspects.

3. Development of KBS-3H design

At present there are two different variations (called candidate designs) of KBS-3H design: a) Basic Design (BD) and b) design based on Drainage, Artificial Watering and air Evacuation (DAWE). BD alternative is based on assumption that the distance blocks will seal the supercontainer units in wet sections stepwise in sequence independently of each other. In DAWE design the drift can be artificially filled with water after plugging one compartment by a steel plug and sealing the compartment. The distance blocks will then also swell and isolate the supercontainer units simultaneously. These two KBS-3H candidate designs are at the moment being developed to proper level of details based on Olkiluoto bedrock data in order to evaluate the feasibility of the KBS-3H concept in 2007.

Both designs are based on the principle of dividing the deposition drift into compartments (Fig.2). The number of compartments depends on the water inflows and site-specific bedrock structure. The compartmentalisation is implemented by using a novel steel plug capable of taking the full hydrostatic force at 400-500 m level.

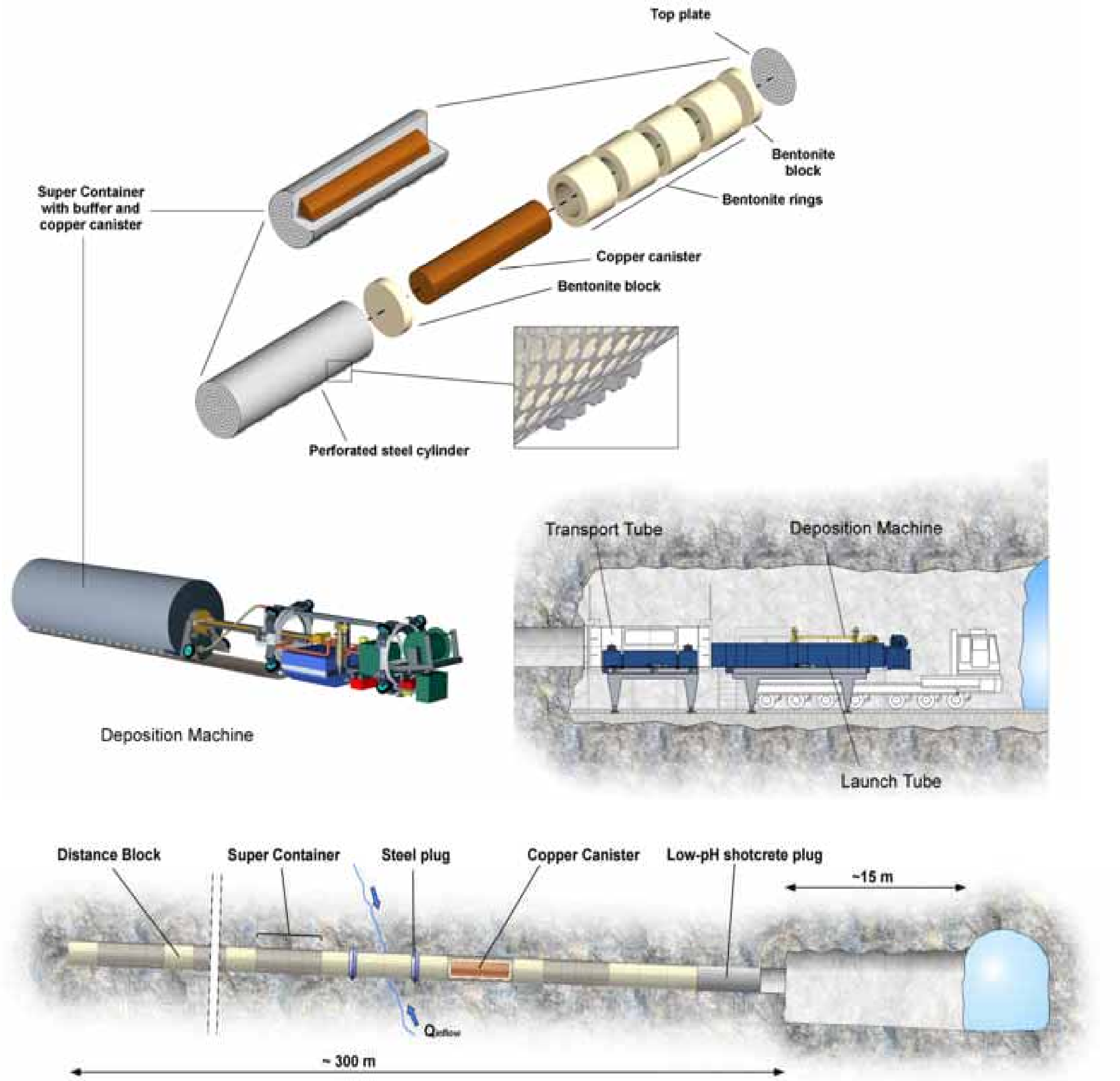

Fig.2 *Supercontainer, deposition equipment and the general arrangement of the drift.*

One important design factor of the concept is the groundwater control, which will affect the utilisation degree significantly. Therefore new approaches and techniques, such as a Mega Packer type injection device for post grouting by using low pH grouting materials, are investigated in order to improve the management of water inflows

Since the design includes new and not proven components, the functional analysis and testing of the design are significant part of the work. The work include testing of the most important design components at proper scales from laboratory up to full scale testing at the Äspö HRL in the two horizontal demonstration holes at -220 m level. The objective of these tests is to demonstrate that the individual fundamental components of the KBS-3H design function properly and that the design in total will fulfil the specified requirements and is sufficiently robust.

In order to evaluate the feasibility of the KBS-3H concept, an Olkiluoto specific layout adaptation is carried out to assess the site utilisation degree of the concept and to provide basis for safety assessment.

4. KBS-3H Safety Case

A preliminary Safety Case for the KBS-3H concept will be compiled at the end of 2007 with Olkiluoto as a reference site.

Only the significant differences with respect to KBS-3V are evaluated in detail, because the KBS-3H concept rests on the broad technical and scientific foundation of the KBS-3V experience with canister development, spent fuel studies, buffer studies, geosphere and biosphere issues and safety analysis. Thus, the requirements for providing a comprehensive synthesis of information are considerably reduced relative to that of a typical safety case. This 'difference analysis' approach is the foundation of the work for the KBS-3H safety case.

The processes that require special attention in both design and long-term safety studies are the early evolution of the system, related to the heterogeneous re-saturation of the buffer and distance block along the drift. The hydraulic conductivity, density and swelling pressure of the bentonite surrounding the canisters at the end of the period of transient THMCBG (Thermal, Hydraulic, Mechanical, Chemical, Biological and Geological) processes are key properties ensuring that the buffer carries out its safety role in both the KBS-3V and KBS-3H concepts. Thus, any processes during this transient phase leading to a loss or re-distribution of buffer mass are of particular concern. In the case of KBS-3H, piping and erosion have been identified as possible mechanisms leading to such a loss or re-distribution of mass /2/. Piping and erosion become less likely as the buffer saturates over time and increasing swelling pressure is exerted on the drift wall. Thus the magnitude and rate of increase in the hydraulic pressure differences that might create pipes and drive pipe flow are both relevant quantities. For the geosphere processes are, due to localised water flow effects, complex gas related processes in early phase, and possible reactivation of fractures in near-field rock (gas pressure) identified.

A key difference between KBS-3H and KBS-3V is the presence of additional steel components (e.g. supercontainer, steel plug) in KBS-3H that will corrode and introduce a complex set of gas issues in safety assessment, such as eventual gas bubble transport and transport of volatile radionuclides, accumulation of gas along the top of the drift, and its effect on groundwater transport. The eventual chemical alteration of bentonite, effect on rheological properties, swelling and hydraulic conductivity of bentonite due to the steel components are also of concern and these are being addressed /3, 4/.

Features, events and processes specific for KBS-3H will be described, analysed and discussed in the Process report. The report primarily aims at presenting the scientific knowledge and understanding of the internal processes and at providing a long-lasting basis for future assessments. As regards radionuclide transport analyses specific issues related to the KBS-3H are e.g., the fate of volatile radionuclides in gas phase (e.g. C-14). Fate of volatile radionuclides released from a defective canister during the re-saturation phase is somewhat different in KBS-3H vs. KBS-3V due to gas generation by the supercontainer and other structural materials.

The description and analyses of the evolution of the site and repository from emplacement of the first canisters in the repository over various transient phases into the far future forms together with the Process report the scientific basis for the preliminary Safety Case for a KBS-3H type spent fuel repository at Olkiluoto (Fig.3).

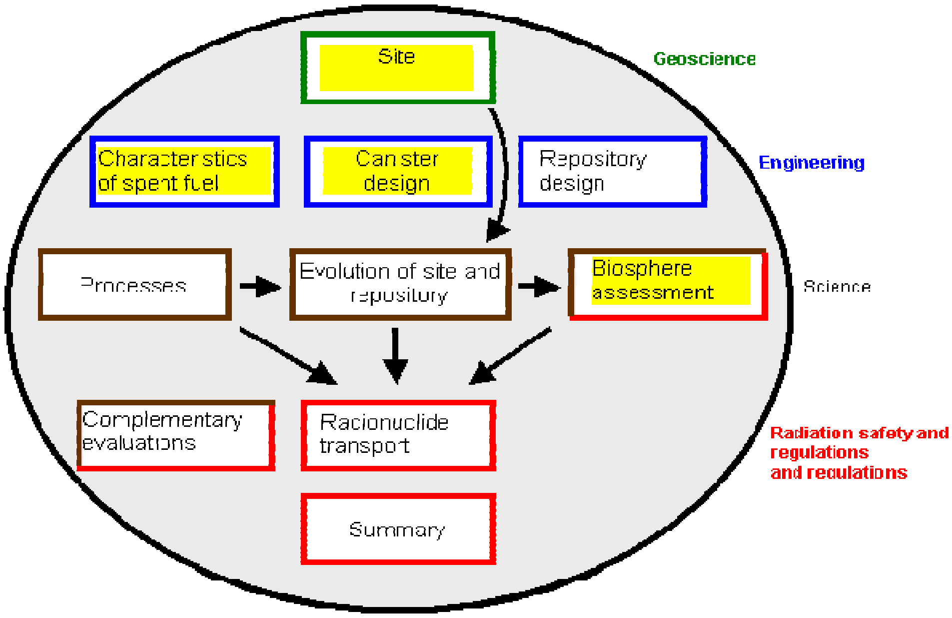

Fig. 3:*The reporting structure for the KBS-3H safety case 2007, The same reporting structure applies to the safety case for a KBS-3V type spent fuel repository at Olkiluoto /5/. The colours of the boxes indicate the areas covered by the different reports and arrows show the most important transfers of knowledge and data. Filling indicates reports common to the KBS-3V and -3H safety cases.*

5. Summary

The work within the KBS-3H concept study is reaching the milestone 2007 for the evaluation of the feasibility of the KBS-3H as an alternative for the KBS-3V concept.

The results of the evaluation of full-scale testing of the design components and the technical equipment needed in the concept will together with the safety case form the basis for the decision by SKB and Posiva how to proceed with the project, and for planning of further steps in development and testing work.

The KBS-3H-specific safety case focuses on major differences between the two KBS-3 concepts, which are largely related to the emplacement of the supercontainer and distance blocks and their subsequent evolution. Provided these aspects lead to limited differences in long-term evolution and performance of the system, there is expected to be little difference in the long-term safety of KBS-3V and KBS-3H.

6. References

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THE BELGIAN DEMONSTRATION PROGRAMME FOR THE DISPOSAL OF HIGH-LEVEL AND LONG-LIVED RADIOACTIVE WASTE

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ABSTRACT

The EIG EURIDICE is responsible for performing large-scale tests, technical demonstrations and experiments so as to assess the feasibility of a final disposal of vitrified radioactive waste in deep clay layers. This programme is part of the Belgian Research and Development programme managed by ONDRAF/NIRAS. The research infrastructure includes the Underground Research Facilities HADES (URF HADES) in the Boom Clay geological formation and surface facilities. The achievements of the demonstration programme are the demonstration of the construction of shafts and galleries at industrial scale, the characterisation of the hydro-mechanical response of the host rock, and the "OPHELIE mock-up" a large scale hydration test under thermal load of pre-fabricated bentonite blocks. The future works will consist mainly in the realisation of the “PRACLAY experiments” including a large scale heater test. The results of this test will constitute an important input for the Safety and Feasibility Cases 1 (SFC-1, 2013) and 2 (SFC-2, 2020).

1. Introduction

The Boom Clay layer, a tertiary plastic clay, was chosen as a study case for the geological disposal of high-level and long-lived radioactive waste. In Belgium, the R&D programme on this topic was initiated at the Belgian nuclear research centre (SCK•CEN) in 1974. The URF HADES was constructed at a depth of 223m for R&D purposes. The first construction phase started in 1980 and since the URF HADES has been expanded several times. Figure 1 shows the construction history. The primary purpose is conducting various in-situ experiments to study the feasibility of HLW disposal in the Boom Clay layer. HADES is currently managed by the Economic Interest Grouping EURIDICE, a joint venture between SCK•CEN and NIRAS/ONDRAF.

Since previous research yielded promising results, the R&D programme is more and more tending towards large scale and demonstration tests. The realisation of the demonstration programme “the PRACLAY project” is the main mission of EURIDICE. The PRACLAY project includes:

- The extension of **URF HADES** consisting in the construction of a second shaft, a connecting gallery between the second shaft and the existing laboratory, an experimental gallery perpendicular to the connecting gallery and the ventilation building leading to the demonstration of the industrial process for constructing the underground disposal infrastructure;
- The ***in-situ PRACLAY experiments*** aiming to demonstrate that Boom Clay is suitable, in terms of performance of the disposal system, to undergo the thermal load induced by the vitrified waste;
- The ***surface PRACLAY experiments*** to demonstrate the technical construction and placement of the engineered barriers, as well as researching the interaction of these with the host rock.

2. The extension of the URF HADES

The demonstration that we can construct a repository infrastructure, using an industrial technique, while controlling the disturbances at an acceptable level in terms of performance of the disposal system, is now well advanced with the experience gained with the construction of the HADES extension. The extension consists in the realisation of a second shaft (1997-1999) and the construction of a connecting gallery (2001-2002).

The hydro-mechanical behaviour of Boom Clay around the excavation and its evolution with time is now well characterised and the high sealing capacities of Boom Clay have been proven [1].

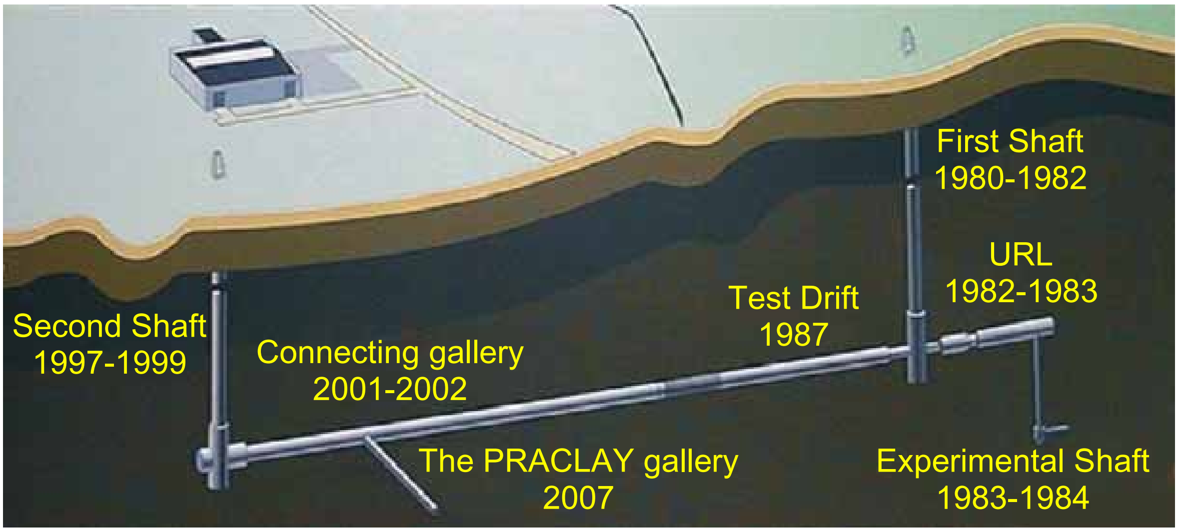

Fig 1. The construction history of HADES

2.1 The second shaft

The second shaft has an effective diameter of 3 m and widens out to an effective diameter of 5m at low level. The ground freezing technique was used to sink the shaft through the water-bearing sands. In this section the definitive lining consists of prefabricated concrete rings with an 8-mm thick outer steel casing. The gap between the preliminary lining and the definitive lining was filled with asphalt. The part in Boom Clay was realised without freezing [2]. In this section the definitive lining consists of poured concrete in direct contact with the Boom Clay.

The feasibility of digging in unfrozen Boom Clay from the top (-190 m) to the middle (250 m) of the Boom Clay layer has been demonstrated. During the excavation of the second shaft, the mechanical behaviour of the rock was quite homogeneous, irrespective of the depth. During the construction of the starting chambers, significant slip planes were observed. Their symmetry around the shaft axis indicated that the fractures were induced by the excavation work. Active support installed immediately after the excavation considerably reduced the opening of the fractures and the risk of detachment of blocks.

2.2 The connecting gallery

The construction of the connecting gallery by an industrial technique has been an important milestone in the demonstration programme for the disposal of high-level radioactive waste. Four main requirements were set to limit the extent of the zone disturbed by excavation: maximising the construction rate (minimum 2 meters a day), minimising the overexcavation, minimising the length of the unsupported zone, and choosing a stiff lining.

The construction of the connecting gallery required the prior construction of a mounting chamber, namely of a chamber that would be large enough to enable the tunnelling machine and the associated equipment to be assembled. The tunnelling machine (see figure 2) was composed of three main parts: a road header to excavate the rock, a 2.3 metre long shield to protect the workers and control the convergence, and a bird-wing erector to install the lining segments. The excavated diameter by the road-header was slightly smaller than that of the shield. It was therefore the shield that imposed the final, smooth excavation profile. The lining technique was the wedge block technique, an expandable-lining technique thought to be suitable for lining galleries in the Boom Clay at about 223 m depth while minimising convergence [3].

Fig 2**.** The tunnelling machine used for the construction of the connecting gallery

2.3 The EC CLIPEX project

The excavation of the connecting gallery from the second shaft towards the existing Test Drift provided a unique and original opportunity to monitor the hydro-mechanical parameters of Boom Clay ahead of an excavation front. The EC CLIPEX instrumentation programme (Clay Instrumentation Programme for the Extension of an Underground Research Laboratory) enabled the instantaneous hydro-mechanical response of the clay during excavation of the connecting gallery to be characterised with high reliability [4]. The host rock has been instrumented both in the zone to be excavated and around it.

The numerical simulations using Mohr-Coulomb and Modified Cam-Clay models gave reliable blind predictions in terms of displacement and pressure on the lining thus allowing an optimum design of the tunnel machine. One important finding of the project is the unpredicted observation of hydraulic perturbation deep inside the formation. Current model developments are made to explain the variation of pore water pressure in the far-field (about 60m from the excavated front) considering the delayed effects through the viscosity of the Clay skeleton.

2.4 The SELFRAC project

The creation of an excavation disturbed or damaged zone is expected for all geologic formations. Macro- and micro-fracturing, and in general a rearrangement of rock structures, will occur in this zone resulting in significant increases of permeability to flow. Implications of the higher permeability of the damaged zone and its time evolution under various repository scenarios need to be evaluated as part of waste repository safety assessment. Within the EC SELFRAC project, various issues, such as processes creating fractures in the excavation damaged zone, the degree of permeability increases, potential for sealing or healing (with permeability reduction) in the zone were investigated [5].

Laboratory tests were conducted to characterise the sealing process by monitoring the evolution of the flow properties along a fracture. Results of these tests show that for Boom Clay sealing occurs very quickly after the flooding of the fracture. During the sealing process the permeability decreases up to value close to the permeability of intact Boom Clay (about 10^{-12} m/s).

The in situ experiments in Boom Clay have allowed to follow the evolution with time of the hydro-mechanical behaviour of Boom Clay around a gallery excavated by industrial technique and to quantify the effect of the sealing processes on the hydraulic conductivity evolution. The radial extent of the fracture zone around the gallery was about 1 m. However a slight increase of the hydraulic conductivity was measured up to 6-8 m into the host-rock. It was shown that two years after the excavation the interconnected fractures zone was reduced from 1m to less than 60 cm around the gallery. The hydraulic conductivity in the sealed zone and beyond in the host-rock remains lower than 2.5 10^{-11}m/s. We can therefore conclude that the repository system would not be adversely affected by the excavation process of the repository infrastructure.

3. The in situ PRACLAY experiments

The main objective of the PRACLAY experiments is to verify that Boom Clay is suitable, in terms of performance of the disposal system, to undergo the thermal load induced by the vitrified waste. The test will focus on the study of the combined effect of the EDZ (Excavation Damaged Zone) and the thermal impact at repository scale. The influence of the temperature increase on the EDZ evolution as well as the possible additional damage created by the thermal load will be studied. The impact of the THMC (Thermo-Hydro-Mechanical and Chemical) response on the transport properties of the Boom Clay will also be assessed. A long term (more than 10 years) large scale heater test would be representative of the most penalizing conditions that could be encountered in the real disposal. The results of the test will constitute an important input for the SFC (Safety and Feasibility Case) -1, 2013 and -2, 2020.

The design of the PRACLAY experiments is now fixed based on numerical simulations. The design phase included the definition of the geometry, the boundary conditions and the instrumentation programme. We developed the PRACLAY in-situ experiments to be design-independent to overcome possible future changes in the reference disposal design. The PRACLAY experiments will be performed within "The PRACLAY Gallery", which will be 45 m long with an internal diameter of 1.9 m, lined with concrete segments and perpendicular to the connecting gallery. The heated length will be about 30 m. The PRACLAY in-situ experiments regroups a set of three tests (see figure 3):

- The gallery and crossing test
- The heater test
- The plug test

The excavation will be performed under the protection of a shield and using the wedge block system for the lining. The method has to allow an excavation rate (excavation + installation of the lining) of minimum 2m/day. At the end of the gallery a stop test will be carried out in order to assess the difficulty to restart the tunnelling machine after a stopping period of one week. The construction of the PRACLAY gallery requires a steel reinforcement ring at the crossing with the connecting gallery. According the design of the connecting gallery the maximum possible diameter for the opening in the lining of the connecting gallery is 2.55 m. Consequently, the nominal diameter of the extrados of the PRACLAY gallery has been fixed to 2.5 m taking into account the convergence. A diameter about 2.5 m corresponds to the range of diameters considered for the repository designs.

The results of the gallery and crossing test will give additional information for the optimisation of the tunnel excavation and will demonstrate the feasibility to construct a crossing between an access gallery and a disposal gallery.

The large scale heater test is considered as a generic issue for all repository design actually considered by ONDRAF/NIRAS since at a distance larger than a few metres from the waste, the influence of the specific design on the temperature profile is limited. It has to demonstrate that the damaged zone due to thermal load of the host rock remains acceptable in terms of long term performance of the repository. It will be important to verify that fracturing remains acceptable and that the decrease of effective stress due to the increase of pore water pressure will not lead to the liquefaction of Boom Clay. The impact of the chemical processes on the transport properties of the Boom Clay will also be investigated. It has been chosen for the PRACLAY In Situ Experiment, to use a heater system imposing a as constant as possible temperature of about 80°C at the extrados gallery wall. However a second heater working at constant flux will also be installed as a back up of the first heater in case of failure. This back-up heater can be retrieved during the PRACLAY Heater Test.

Plugs (within disposal galleries, between disposal and main galleries, between main galleries and shafts) are considered, at least as a conservative measure, in the overall repository design in order to e.g., limit interactions between various repository zones (compartimentalisation through cutting the hydraulic connection along the gallery lining and EDZ), increase the resilience of the repository to intrusion, and avoid gas migration. The “PRACLAY Plug Test” aims at demonstrating that it is possible to cut-off hydraulically the EDZ and the engineered barriers of the disposal galleries with a horizontal plug.

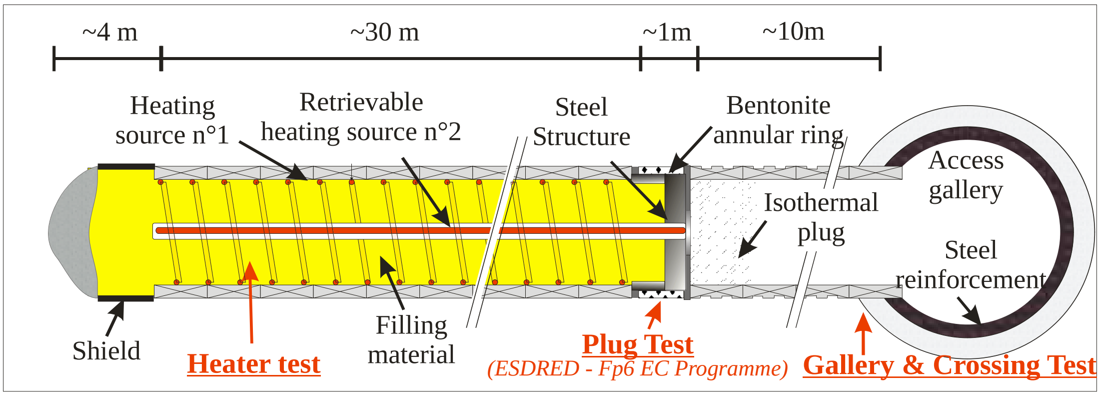

Fig 3. The PRACLAY experiments

The PRACLAY gallery will be constructed in 2007 so that the experiments can be installed in 2008. The start of the heating phase is planned in the first semester of 2009. The EC project TIMODAZ (Thermal Impact on the Damaged Zone Around a Radioactive Waste Disposal in Clay Host Rocks) has been introduced in the frame of the 6th Framework Programme and is now under contract negotiations. The project includes lab and field experiments, modelling and model validation through benchmarking. After such an independent validation, the codes will be applied to perform predictive simulations of the PRACLAY experiments. In total 8 countries are represented (BE, FR, CH, DE, NL, ES, CZ, UK).

3. The Surface PRACLAY Experiments

The PRACLAY Surface Experiments include the feasibility study of the construction and the handling of the engineered barriers. For these tests, it is currently understood that in situ conditions are not required, i.e., that the influence of, and the interactions with, the host formation do not condition the short term performances of the engineered barriers. Hence, it is not primordial to test these elements in situ. It was therefore decided to test them on surface which should enable better control of the experimental conditions and be more cost-effective.

3.1 The OPHELIE mock-up

The OPHELIE mock-up (see Figure 4) deals with the reference design valid in the middle of the 90s. It simulated a section of a waste disposal gallery, in order to prepare the in situ PRACLAY Experiment and to review several technical aspects of its design. The mock-up focused on the engineered barriers of the disposal system: the buffer material, a mixture of FoCa Clay (60 wt %), sand (35 wt %) and graphite (5 wt %), the metallic disposal tube and the hydration system. The mock-up also allowed a large-scale investigation of the THM behaviour of the buffer material as well as of its interactions with the other barriers.

Globally, regarding its thermo-hydro-mechanical properties, the buffer material fulfilled its role: after four years of hydration and heating, it kept a low permeability and a high thermal conductivity. Although it swelled and filled all the technological voids, the swelling pressure remained low and the swelling process was not homogeneous.

The OPHELIE mock-up highlighted the complexity to determine and to understand the main processes (THM, chemical, corrosion …) controlling the behaviour of the engineered barriers in a saturated environment at temperatures exceeding 100 °C [6]. Associated with the presence of chlorides observed in the buffer, these observations were at the origin of the decision taken by ONDRAF/NIRAS to re-examine in-depth the Belgian reference design for the disposal of vitrified HLW and to develop the concept of a "Supercontainer" [7].

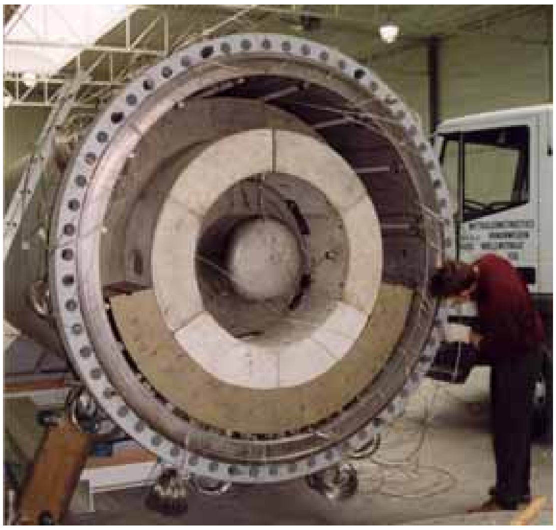

Fig 4. The OPHELIE mock-up

3.2 The demonstration of the “Supercontainer design”

ONDRAF/NIRAS selected “the Supercontainer design” as the new reference design. A more homogeneous cement-based material offers, amongst others, the advantage to create long-term chemical conditions much more favourable for the lifetime of the metallic barriers. Moreover, cement-based materials are well known because of the large number of applications and existing studies. The new reference design also considers a lower linear thermal power resulting in temperatures below 100 °C in the engineered barriers. Although some aspects related to feasibility and technological issues were already, implicitly, considered during this selection process, the technological feasibility to construct the engineered barriers system (EBS) and its different components remains, to a large extent, still to be demonstrated. This demonstration programme will start in 2007.

4 Conclusions

At this time, the study of the feasibility for Boom Clay is well advanced. The effect of a large scale thermal load on the behaviour of Boom Clay is an important key issue remaining to be studied. Indeed the impact of the thermal load generated by the waste is particularly important since it will significantly affect the temperature and the stress profiles on the whole thickness of Boom Clay in the short term after the disposal. Therefore the early transient THM perturbation might be the most severe impact that the repository system will undergo on a large spatial scale and in a relatively short period of time. In order to demonstrate that Boom Clay will behave as predicted under a thermal load a large scale heater test within "The PRACLAY experiments" is planned. The performance of a horizontal plug will be tested in the same experimental drift. The results of these large scale tests will be a milestone in the choice by the Belgian government of a disposal strategy for radioactive waste. Parallel to the in-situ PRACLAY experiments, the feasibility of the new reference design, the "Supercontainer" will be studied.

Acknowledgement

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GENERIC REPOSITORY CONCEPT FOR RBMK-1500 SPENT NUCLEAR FUEL DISPOSAL IN CRYSTALLINE ROCKS IN LITHUANIA

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ABSTRACT

During 2002-2005 investigations on possibilities to dispose of spent nuclear fuel (SNF) in Lithuania were performed with support of Swedish experts. Disposal concept for RBMK-1500 SNF in crystalline rocks in Lithuania is based on Swedish KBS-3 concept with SNF emplacement into the copper canister with cast iron insert. The bentonite and its mixture with crushed rock are also foreseen as buffer and backfill material.
In this paper modelling results on thermal, criticality and other important disposal characteristics for RBMK-1500 SNF fuel emplaced in copper canisters are presented. Based on thermal calculations, the distances between the canisters and between the tunnels were justified. Criticality calculations for the canister with fresh fuel with 2.8 % ^{235}U enrichment demonstrated that effective neutron multiplication factor k_{eff} values are less than allowable value of 0.95. Dose calculations have shown that total equivalent dose rate from the canister with 50 years stored RBMK-1500 SNF is rather high and is defined mainly by the γ radiation.

1. Introduction

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 (SNF) will be approximately 22 thousands of fuel assemblies. All these assemblies should be stored about 50 years and after that disposed of. The capacity of existing SNF dry storage facility at the Ignalina NPP is for 80 casks. New interim dry storage facility that will accommodate remaining SNF is under implementation.

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 the assessment of possibilities for disposal of SNF in Lithuania was performed with the support of Swedish experts. Extended studies on selecting of suitable geological formation had led to the conclusion that crystalline rock and argillaceous rocks are the primary candidates for disposal of SNF and long-lived intermediate level waste (ILW) in Lithuania [1]. In Lithuania, a crystalline basement occurs at depth of 200-2300 meters below the land surface. The prospective area of crystalline basement was confirmed as occurring in the southern Lithuania with the depths ranging from 210 m to 700 m, while in most of the Lithuania territory, the depth of the basement exceeds 700 m, reaching 2300 m in the west [1].

A generic repository concept for RBMK-1500 SNF disposal in crystalline rocks in Lithuania developed during mentioned studies is presented in this paper. Modelling results on thermal, criticality and other important disposal characteristics for RBMK-1500 SNF fuel emplaced in copper canisters performed using computer code FLUENT for thermal calculations, and code system SCALE 4.3 for criticality and doses calculations, also are presented in this paper.

2. Repository concept

The repository concept for SNF disposal of in the crystalline rocks in Lithuania is based on the repository concept developed in Sweden for SNF disposal of in the crystalline rocks (KBS-3). According to this concept SNF is emplaced in the copper canister with cast iron insert. Disposal canister and bentonite buffer of 0.35 m thick surrounding it are vertically emplaced in the crystalline rocks at the depth of 500 m. This disposal method is known as KBS-3V. At the present vertical and horizontal SNF canister disposal are under investigations by SKB (Sweden) and POSIVA (Finland) [2]. The advantage of KBS-3H compared to the reference design (KBS-3V) is that the deposition tunnels are not needed in that design. The absence of deposition tunnels reduces the excavated rock volume by about 50 %. This results to less environmental impact during construction, cost savings and reduced need for ventilation and drainage during construction and later on operation.

The main engineered barriers of the geological repository are waste form, waste overpack (canister), buffer and backfill materials for backfilling of the void space between the canisters and host rock. The container is one of the most important component of the multibarrier system. Two conceptual approaches are possible: corrosion allowance and corrosion resistance. The first involves the use of readily corrodible metals (e.g. mild steel and cast iron) with sufficient thickness to delay container failure for some thousands of years, i. e. until the short lived fission products in the wastes have decayed. The second involves the uses of corrosion resistant materials (e.g. copper or titanium alloys) that are intended to prevent water access for much longer periods (up to 100 000 years), possibly even until all the most mobile radionuclides have decayed and the waste hazard has declined to levels similar to those of natural uranium ore [3]. For SNF disposal of in the crystalline rocks the preference is often given to the copper canisters. Under the reducing water condition as it usually prevails in deep crystalline rocks the copper has very high corrosion resistance. Based on KBS-3 concept SNF disposal canister will be composed of two components: an outer corrosion protection of copper and a cast iron insert with channels for the fuel half-assemblies in order to improve the mechanical strength. The wall thickness of the copper canister is 50 mm and minimum wall thickness of the cast iron is 50 mm. Taking into account the results of the criticality, dose rate assessment and thermal calculations presented in the next chapters as well as taking into account the existing experience in the canisters shifting and emplacement technology it was proposed to load 32 half-assemblies of RBMK-1500 SNF in one disposal canister. Based on preliminary assessment the reference canister would be of 1050 mm diameter and 4070 mm length. For Lithuanian SNF disposal purposes about 1400 canisters should be employed.

Detailed repository design is clearly highly specific to waste type and to geological environment, but there are some general principles in it's design. Due to already mentioned advantages KBS-3H could be proposed as a reference as a reference design for Lithuania. As KBS-3H design is under the development in Sweden and Finland yet, thus KBS-3V is left as an alternative one, if KBS-3H is shown as not feasible and safe. There is no decision yet if the long-lived ILW will be disposed of in the same repository as the SNF or separately. In case of the first alternative possible layout of the repository is presented in Fig. 1. The main elements of repository are:

- an access shaft, transport tunnels, central waste receiving facilities and a shaft;
- an array of SF emplacement tunnels (deposition drifts);
- emplacement tunnels for long-lived ILW.

The waste emplacement tunnels, main tunnels and the transport tunnels would be excavated at a depth of 300-500 m in the crystalline basement. The SNF canisters would be disposed of at 1.2 m distance from each other in horizontally bored emplacement tunnels using so called supercontainer concept [3]. According to this concept each container with its surrounding bentonite buffer and a perforated steel shell (supercontainer) is placed between two compacted bentonite distance blocks in the emplacement tunnel [3]. The length of 250 m of SNF emplacement tunnels is accepted at this stage of investigations. The diameter of the SNF emplacement tunnel is proposed to be 1.85 m like in Sweden. The centre to centre distance between the emplacement tunnels is about 40 m. The length of shafts is dependent on the repository layout and selected site. Taking into account the distance between the SNF canisters and emplacement tunnels as well as the number of canisters to be disposed of, the deposition area for SNF will cover approximately 0.4 km^2 area.

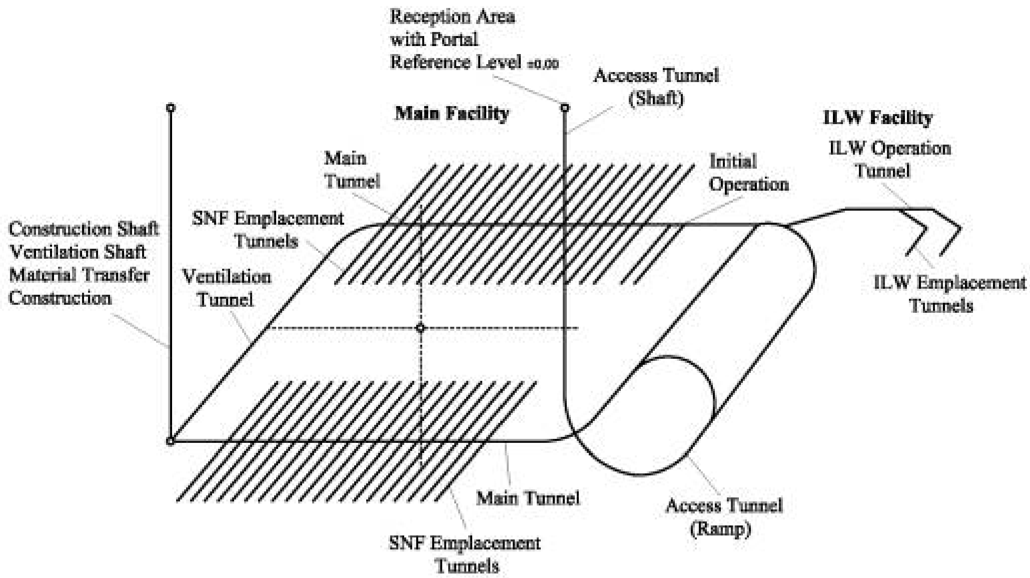

Fig. 1. Plan view of the repository for RBMK-1500 SNF disposal of in crystalline rocks

3. Performance assessment

3.1 Dose rate

Dose rate values will be important when spent nuclear fuel after interim dry storage will be emplaced in canisters and transferred to the repository. Calculated dose rate level indicates what measures should be introduced (for example, remote handling, additional shielding) to meet radiation safety requirements. The γ dose rate outside the canister is of importance for radiolytic disintegration of water also. The canister design criteria require that the γ dose rate not exceed 1 Gy/h (1 Sv/h if only β, γ radiation) in order to minimize the importance of the process [4].

Sequences SAS2H and SAS4 from SCALE 4.3 computer code were used for dose rate assessment of the canister with RBMK-1500 SNF. The main assumptions for the modelling of fuel assembly irradiation were following:

- RBMK-1500 fuel assembly that consists of 18 fuel rods was homogenized and in the reactor's fuel channel was described as an element of 5 concentric cylinders;
- Fuel enrichment 2.8% ^{235}U, burn-up 30 MWd/kgU, irradiation time 3 years, cooling time 50 years;
- For dose rate calculations axial burn-up distribution of fuel assembly was not taken into account.

Neutron and gamma radiation forms the total equivalent dose rate and from the canister with 50 years stored RBMK-1500 SNF the total equivalent dose rate is app. 500 mSv/h (Fig. 2).

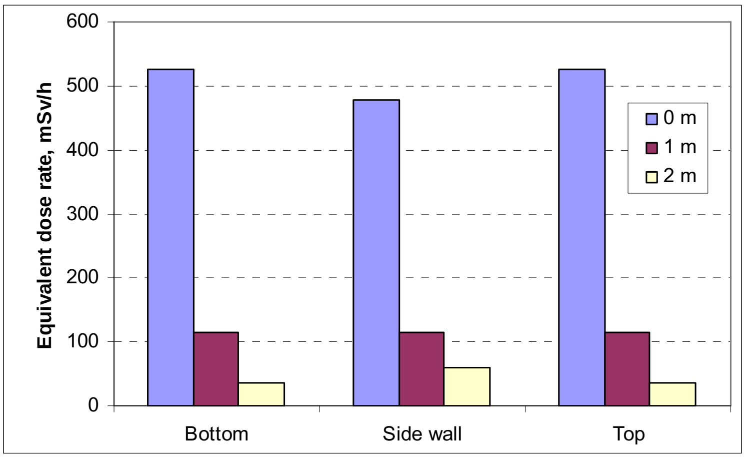

Fig. 2. Total equivalent dose rate values on the copper canister surface and at some distances

The results of dose rate calculations show that total equivalent dose rate is formed mainly by the γ radiation (more than 99.9%); neutrons forms only insignificant part of total dose rate. Dose rate calculations have shown that dose rate values on the surface of the copper disposal canister with RBMK-1500 SNF are rather high in comparison to SNF storage casks, but do not exceed the limit of 1 Sv/h which is maximum allowable dose rate value according to Swedish KBS-3 concept.

3.2 Criticality

Neutrons with suitable energy can cause nuclear fission particularly in ^{235}U, ^{239}Pu and ^{241}Pu in the SNF. Intruding water into the damaged canister can slow down the neutrons to suitable energies. A criticality conditions are assessed with the effective multiplication constant k_{eff}. Criticality analysis for copper disposal canister loaded with (2.8% ^{235}U enrichment) fresh 32 RBMK-1500 fuel half-assemblies was performed using SCALE 4.3 computer codes system. The following main conditions and assumptions were accepted for the criticality calculations:

- Maximum loading of the canister, i.e. insert of the canister contains 32 cylindrical holes each with fuel half-assembly inside;
- Discrete representation of the fuel rods is used in the geometry description. This means that each half-assembly consists of 18 fuel rods;
- The fuel half-assemblies contain only fresh, undepleted fuel (no credit for burnup) with 2.8% ^{235}U enrichment;

The variation of effective neutron multiplication factor k_{eff} (including 3 standard deviations) with water density for copper disposal canister shows that k_{eff} values continuously increasing when water density is increasing and maximal k_{eff} value of approximately 0.61 is reached when water density is 1.0 g/cm^3. The main requirement of the criticality safety is that effective neutron multiplication of the system containing fissile material must be less than 0.95. For copper disposal canister when long-term processes (corrosion, degradation, etc.) are not taken into account, k_{eff} values are less than allowable value of 0.95.

3.3 Thermal calculations

Temperature evolution calculations were based on the decay heat of the SNF, thermal and geometrical data of all parts of the repository. Temperature evolution in SNF tunnels was calculated using FLUENT 6.1 code.

Two cases were analyzed in total assuming that the buffer is partly saturated (low thermal conductivity) and fully saturated (high thermal conductivity) by water uptake from the surrounding rock. The results of time-dependant temperature evolution in the tunnel with horizontally emplaced SNF canisters show that, for the partly saturated (low thermal conductivity) bentonite, the peak temperature of $\approx$ 92 °C (Fig. 3, curve 1) on copper canister surface is reached within few years. In case of fully saturated (high thermal conductivity) bentonite the temperatures are much lower than in the case before. The highest temperature of ≈ 72 °C (Fig. 3, curve 2) on the canister surface is reached within 30 years.

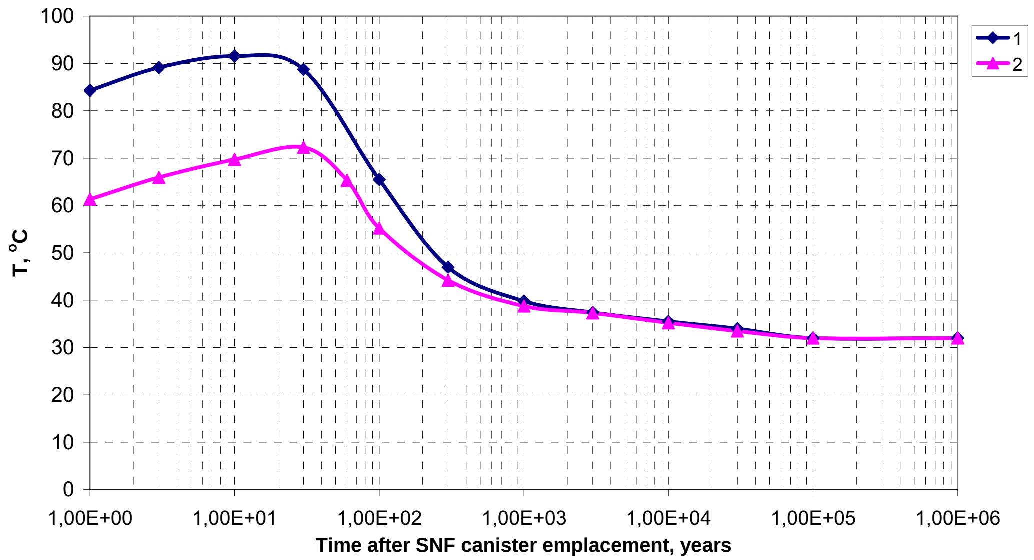

Fig. 3. Time-dependant temperature evolution on the surface of the SNF disposal canister in case of partly (1) and fully saturated (2) bentonite

The requirement that the surface temperature of the canister may not be exceed 100˚C [5] can always be met by choosing a suitable spacing between the canisters or by adjusting the fuel content in the canisters. The results of temperature assessment around the canisters loaded with 32 RBMK-1500 SNF half-assemblies show that a maximum heat output of 784 W per canister at the time of waste emplacement will satisfy the temperature constrain. Calculation results justify the chosen distance between canisters 1.2 m.

4. Summary

Generic repository concept for RBMK-1500 SNF disposal in the crystalline rocks in Lithuania was proposed based on Swedish KBS-3 taking into account the preliminary results on criticality, dose rate and thermal calculations of RBMK-1500 SNF.

5. Acknowledgement

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EXPERIMENTAL AND MODELLING STUDY ON THE LONG-TERM PERFORMANCE OF THE ENGINEERING BARRIER SYSTEM OF TRU WASTE REPOSITORY

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ABSTRACT

The long-term properties of the barrier system of the TRU waste repository will be assessed using the model that analyzes the geochemical reaction and the mass transport in that system. But there are data and models whose validity isn't adequately clear yet. RWMC had started the project of laboratory scale tests, natural analogous studies and numerical model analysis to improve the reliability of that model. The result of the four years of the study made it possible to change some conservative assumption to more realistic ones, and to show longer time stabilities of the repository system.

1. Introduction

In Japan, TRU wastes are planned to be disposed in the tunnel in hundreds meters depth with the engineered barrier system (EBS) of low hydraulic conductive bentonite/sand mixture and low diffusive cementitious materials. The cementitious materials would slowly dissolve in the groundwater to form high pH and high Ca concentration zone around it, and both cementitious and bentonite materials would alter in that environment, and change the performance of EBS.

Conventional performance assessments conservatively treated the effect of these phenomena by assuming degraded properties of EBS for whole period of concern. RWMC of Japan had been trying to model the long-term alteration phenomena of EBS to understand the time dependent performance of it. With this model, initial high performance properties of EBS could reasonably and reliably considered in the assessment. The effects of the dissolution and mineralogical changes of minerals could also be considered in, which weren't sufficiently considered in the conventional assessment.

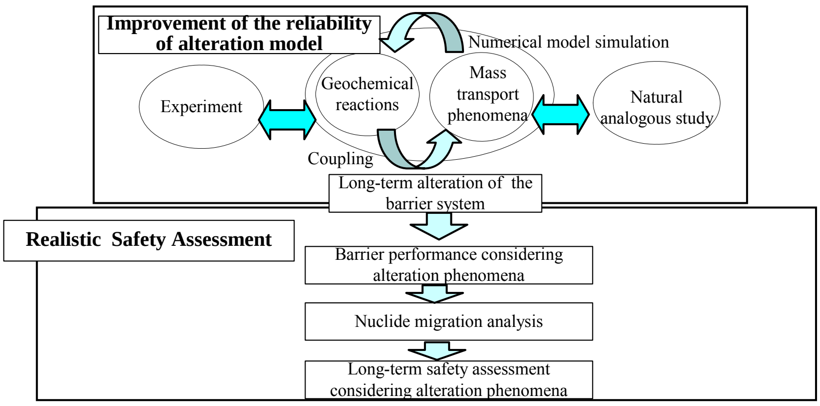

Fig.1 Experiment, natural analogous study and numerical model simulation are combined to improve long-term safety assessment

RWMC had constructed the first model in 2002 with the knowledge at that time, and started the project of laboratory scale alteration experiment, natural analogous studies and numerical model analysis to reduce the uncertainties in the model, parameters and assumptions in it. The purpose of the experiments is to precisely examine the alteration phenomena of months to several years.

Natural analogous data were used to observe the result of very long-term alteration. And these experimental and natural analogous data were compared with the numerical model.

2. Key issues and experiments

RWMC had discussed about the phenomena that can affect the performance of EBS, and selected important ones to make study plans to solve those problem. Most essential and important issue to consider the alteration of EBS is the geochemical model (i.e. thermodynamic data, dissolution rate, selection of minerals identified in EBS, and the position where secondary minerals are formed). In this project, experiments are mainly concentrated to identify the reactions in the system.

Crack is a specific property of cementitious materials, and mass transport related with chemical reaction might be greatly affected by this feature. Here, experiments were made to see the reactions would clog or broaden the crack in various water conditions. If crack is clogged by secondary minerals, cementitious material can be treated as highly resistible region for mass transport.

Though OPC has been treated as typical cement, fly ash cement (FAC), which has finer pore structure, might be preferable cementitious material. To use FAC, its chemical and physical properties should be studied sufficiently. Reaction with saline water is another important issue, because high concentration NaCl and relatively high Sulfate content can quite differently react with materials of EBS. Experiments and main results of them are summarized in Table 1.

Table 1. Experiments and results to solve key technical issues

Key technical issues	Experiment	Results of the experiment
Geochemical model of FAC	Batch dissolution test of FAC	Dissolved chemical species cold be predicted using geochemical model for OPC with modification of initial mineral contents.
Dissolution of OPC in saline water	Batch dissolution test of OPC in saline water.	Dissolved major species and pH could be predicted using geochemical model for OPC. Concentration of S, Mg and formation of friedel-salt, ettringite had differences.
Reversible reaction of analcime to montmorillonite	Montmorillonite reprecipitation test in simulated pore water (120～150℃)	Trace of the reprecipitation of montmorillonite couldn't be detected.
Alteration of montmorillonite to analcime under saline water condition	Batch alteration test of montmorillonite in high Na concentration solution. (80-120℃)	Alteration to analcime was identified in Na concentration > 0.1mol/L, pH range of 11-13.
Clogging or dissolution at the crack in cementitious material	Diffusion and flow column test with cracked mortar specimen.	Most of cracks were clogged in the experiments.
Secondary minerals and its location at the interface of bentonite/cement.	Long-term alteration test of bentonite contacting with mortar (3 years)	Experiments are now going on.
Identification of precipitated CSH	Development of separation method of CSH from bentonite.	Using heavy solution method, added CSH could be separated from bentonite.

3. Numerical analysis

Reflecting the results of those experiments, numerical model analysis was performed to predict the long-term evolution of the EBS. In this analysis, geochemical reactions in each materials and diffusive/ advective solute transport were calculated simultaneously. Furthermore, the change of the physical properties (i.e. hydraulic conductivity (Kw), diffusivity (De) and porosity (ε)) was estimated in accordance with the mineralogical alteration of the EBS.

Analysis code is PhreeqC-Trans, which is the improved code of USGS's PhreeqC[1)] to treat the change of mass transport conditions according to the geochemical reactions. Thermodynamic database for minerals is Spron.JNC[2)] and A.Atkinson's data[3)]. Relationship of Kw and effective gel density of bentonite is from Mihara's data[4)]. Relationship of De and ε of cementitious material is from Yasda's data[5)].

Analyzed cases are listed in Table 2. In conservative case, De of cementitious material is estimated as the average of crack and intact matrix region. Contrary, in the reference case, De of matrix is selected. To consider the effect of mass transport properties in cementitious material, two cases are considered. The first one is crack model, in which crack and matrix is modelled in 2-dimentional geometry. The other case is monotonous increase of De of concrete. Numerical simulation indicates that, according to the precipitation of CSH, De of cementitious materials will decrease, but the location where CSH precipitates isn't clear yet.

For the selection of reacting minerals, three cases were considered. The first one treats reprecipitation of montmorillonite, which is predicted by simulation when pH decreases, but that isn't identified in the experiment. Alteration of montmorillonite to laumontite is also predicted but isn't identified. Dissolution rate of montmorillonite is known to be small, moreover when pore water approaches to equilibrium, it approaches to zero. This effect is more evident if there is non-liner relationship between dissolution rate and saturation index. This type of relationship is selected in "Second TRU progress report" [6].

FAC and saline water case were also analyzed, because the result of the experiment showed that geochemical model of cement minerals can be adopted for FAC or saline water condition.

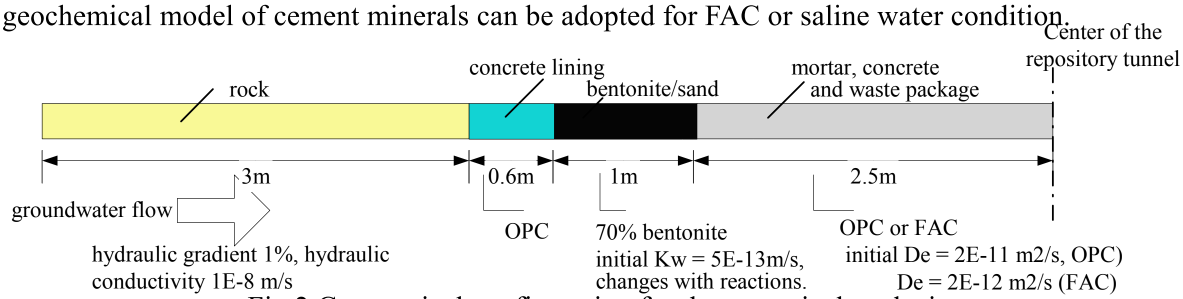

Fig.2 Geometrical configuration for the numerical analysis

Table2 Numerical model analysis cases

Conservative case	Conservative high De in mortar/concrete	De in mortar/concrete is estimated as the average of crack and matrix region.
Reference	Reference case (OPC)	De of concrete is De of matrix. Instantaneous equilibrium is assumed.
Mass transport property of cementitious material	Crack model case	Crack and matrix of concrete is model in 2-dimensional geometry.
	Monotonous increase of De of concrete	Decrease of De with the decrease of ε by the formation of CSH is conservatively neglected.
Selection of the minerals	No reprecipitation of montmorillonite	Reprecipitation reaction of montmorillonite from analcime is inhibited.
	No formation of laumontite	Laumontite is excluded from secondary minerals list.
	Kinetic dissolution of montmorillonite.	Dissolution rate of montmorillonite is calculated by Sato-Cama's equation[6].
Cement type	FAC case	Fly ash cement is used instead of OPC
Groundwater type	Saline water case	Ground water around the repository has high salinity.

4. Results and discussion

Distribution of the minerals after 10,000 years is described in Fig.3. Remarkable change of minerals could be seen at the interface of bentonite and cementitious material. In the bentonite layer, montmorillonite and chalcedony dissolve to form zeolite such as analcime or laumontite. In the cementitious material, portlandite and CASH dissolve to form CSH of low Ca/Si ratio. At conservative high De case, most of montmorillonite are predicted to dissolve, whereas the crack model case shows similar result to that of the reference case. These results indicate that, supply of Ca and other ions aren't dominated by the diffusion through the crack but by the dissolution of the intact cement matrix. Moreover, the clog of the crack in mortar is shown by the test. These results justify expecting low De in the cementitious materials. In low diffusive FAC case, altered zone is small and most of montmorillonite remains even after 10,000 years.

These alteration phenomena strongly affected by selection of the secondary minerals. If laumontite is excluded, most of montmorillonite remains even OPC is used.

When kinetic dissolution of montmorillonite is considered by Sato-Cama's equation, dissolution of montmorillonite in the repository is very small.

Most remarkable alteration of montmorillonite is seen in saline water case. In this case, all of montmorillonite is dissolved to form analcime within 1,000 years.

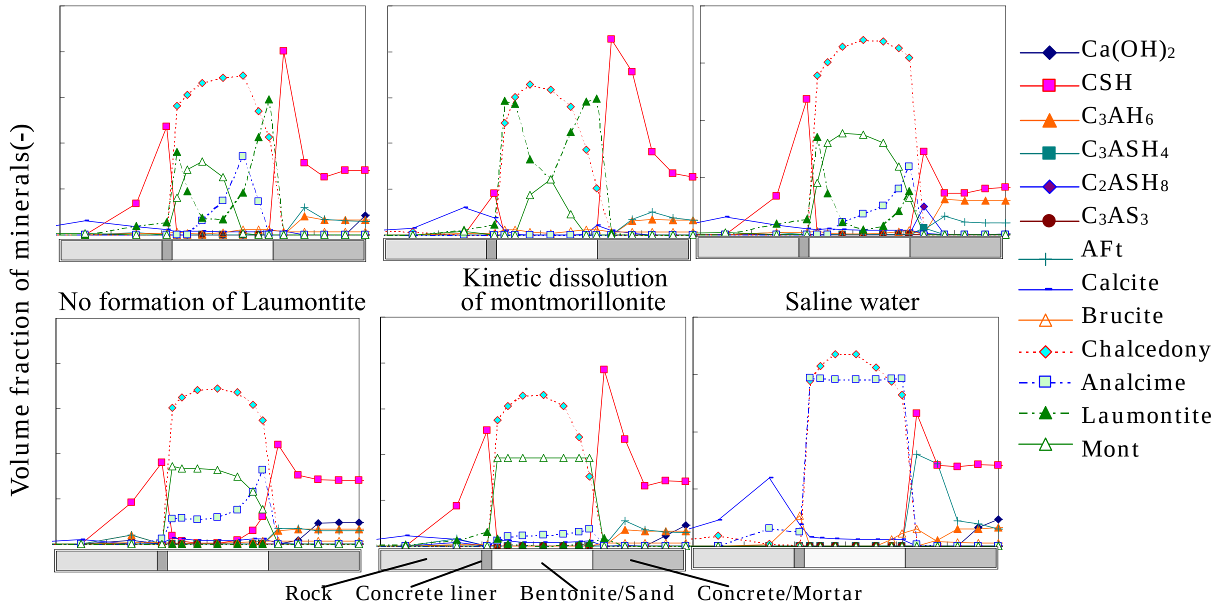

Fig.3 Distribution of minerals in EBS after 10,000 years. The extent of alteration depends on De, selection of minerals, dissolution rate and groundwater contents.

Time dependent Kw in the bentonite layer is shown in Fig.4. Kw increases with the dissolution of montmorillonite, and other associated minerals. Kw of the reference case is smaller than that of conservative high De case. It maintains about 1E-11 m/s until 100,000 years. If FAC is used, smaller Kw for long period is expected. In addition, if kinetic dissolution of montmorillonite is considered, Kw will remain about 2E-12 even after 100,000 years.

As the result of these calculations, if the reliability of realistic parameter and model is validated, stability of the EBS can be shown for longer-term.

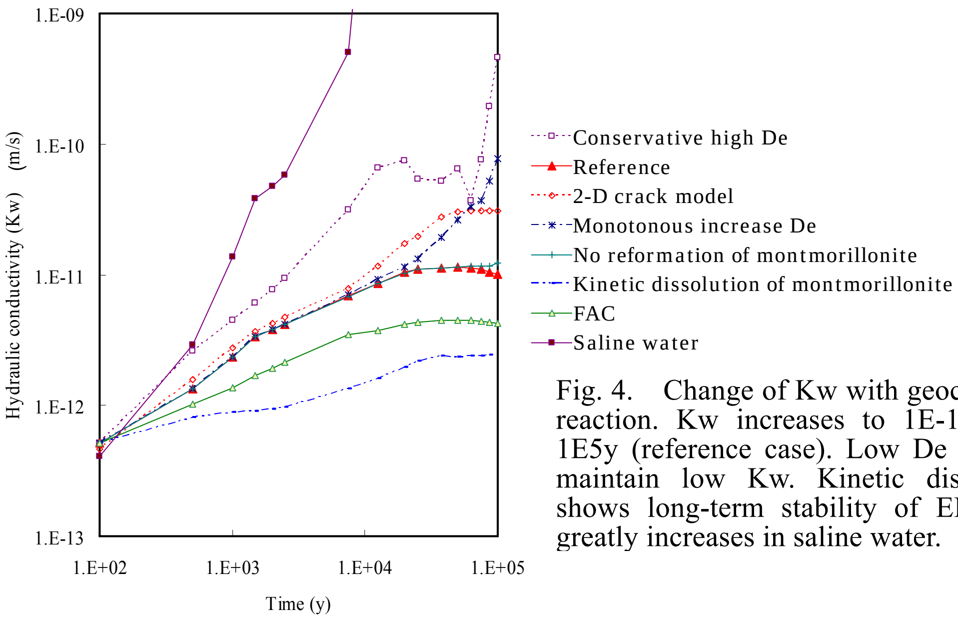

Fig. 4. Change of Kw with geochemical reaction. Kw increases to 1E-11m/s in 1E5y (reference case). Low De leads to maintain low Kw. Kinetic dissolution shows long-term stability of EBS. Kw greatly increases in saline water.

Kw and De of the EBS are plotted on Fig.5 at every 20,000 years. In this figure, contour of normalized nuclide flux at near field is also plotted. Initial properties of each condition are located at the left side of the graph. According to the time evolves, hydraulic conductivities increase and De decreases in most cases. Painted circle is the value used in first TRU report. At that time, low De of cementitious materials was conservatively neglected. On the other hand, in some cases, Kw after long time exceeds the "conservative" value. That is because, many mineralogical reactions are considered, instead ion-exchange of Na type montmorillonite to Ca type only was considered at first TRU report. However, with the precise knowledge about those geochemical reactions, it can be possible to show the performance of the EBS with confidence.

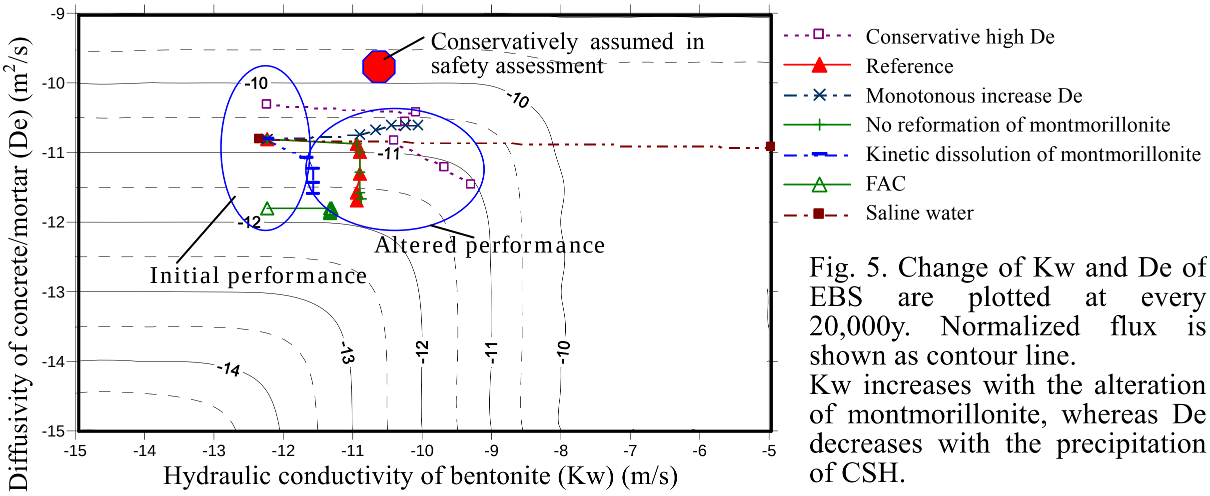

Fig. 5. Change of Kw and De of EBS are plotted at every 20,000y. Normalized flux is shown as contour line.
Kw increases with the alteration of montmorillonite, whereas De decreases with the precipitation of CSH.

5. Conclusion

Reflecting the results of the experiment, long-term model analysis was performed. With the modification of conservative assumptions to the realistic ones made it possible to expect higher performance of the EBS for longer time. Whereas, there remain technical issues to be solved, i.e.

- The location where CSH precipitates at isn't clear from the experiment. That can affect the change of ε and mass transport properties at the interface region.
- In most cases, cracks of mortar were clogged, but they aren't fully re-produced by model.
- Though chemical reaction in batch experiment could be understood by model calculation, there are differences in column experiment where mass transport also takes important role.

More detailed information of the geochemical reactions and mass transport will be shown by the results of the long-term alteration test of bentonite contacting with mortar.

6. Acknowledgment

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MODELLING OF NUCLIDE MIGRATION FOR SUPPORT OF THE SITE SELECTION FOR NEAR SURFACE REPOSITORY IN LITHUANIA

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ABSTRACT

Construction of the near surface repository (NSR) for disposal of short-lived low-and intermediate-level waste (LILW) is planned in Lithuania. Reference design of the repository was prepared. Site selection process is going on. Environmental Impact Assessment (EIA) Program and Report were prepared and are under review by regulators.
Releases of radionuclides to water pathway and potential human exposure after closure of the NSR have been assessed for support of the site selection for NSR installation.
Two candidate sites were taken under consideration. The assessments have been performed following ISAM methodology recommended by IAEA for safety assessments of near surface disposal facilities. The conceptual design of NSR as well as peculiarities of geological and hydro-geological environment relevant to each candidate site is taken into account. The results of the analysis as part of EIA Report are presented in the paper. It is demonstrated that estimated impact of potential radionuclide migration for both candidate sites is below dose constrain established by regulations of Lithuania.

1. Introduction

There is only one nuclear power plant in Lithuania – Ignalina NPP. Two similar units with installed capacity of 1500 MW (each) were commissioned in 12/1983 and 08/1987 respectively. But the first Unit of Ignalina NPP was shutdown December 31, 2004, and second Unit will be shutdown before 2010 taking into consideration substantial long-term financial assistance from the EU, G7 and other states as well as international institutions.

In relation with Unit 1 decommissioning implementation of new technologies for treatment and conditioning of radioactive waste are under way. Construction of the near surface repository for disposal of short-lived LILW is also planned. Reference design of the repository was prepared in 2002 [1]. Operation of the repository is planned until 2030 while the Ignalina NPP will be dismantled and the conditioning of radioactive waste will be performed.

Site selection process is going on. Environmental Impact Assessment Program and Report [2] were prepared and are under review by regulators. After geological engineering investigations North-eastern Lithuania and vicinity of Ignalina NPP in particular are identified among the best suitable regions for a near surface repository. Short distance from the Ignalina NPP, relatively favourable social-economic conditions (low population density, low land economic potential) and good level of geological characterization are the main positive features of Ignalina NPP region.

In this paper releases of radionuclides to water pathway and potential human exposure after closure of disposal facility have been assessed for support of the site selection for NSR intended to construct in Lithuania. Two candidate sites, Galilauke and Apvardai, are taken into consideration.

2. Methodology

The assessments have been performed following ISAM methodology [3] recommended by IAEA for safety assessments of near surface disposal facilities. It contains key components as follows:

1. Assessment context;

2. Description of the disposal system;
3. Radionuclide migration scenario development and justification;
4. Model formulation and implementation;
5. Calculations;
6. Analysis of the results;
7. Confidence building.

Safety assessment of candidate sites built upon framework of the ISAM is provided in the next sections.

3. Assessment context

Dose constraint of 0.2 mSv per year is required for nuclear installations in Lithuania and stands for radiological criteria in the present assessment;

The evolution of the repository within institutional control period for 300 years (100 years for active control and 200 years for passive control) and during subsequent period is considered in order to assess potential releases of long lived radionuclides.

4. Description of the disposal system

According to the concept [1], the repository would consist of 50 vaults with total disposal volume of 100 000 m^3. It is estimated that repository will occupy an area of about 40 ha including waste disposal zone of 3 ha area.

Only finally conditioned solid or solidified short-lived low- and intermediate-level waste that meet waste acceptance for disposal criteria will be disposed off in the repository. The cemented radioactive waste (cement matrixes) containing ion-exchange resins, perlite and sediments are considered in the present safety assessment. Radionuclide inventory containing *short-lived* H-3, Cs-137, Sr-90 as weak sorbing and Pu-241 as strong sorbing as well as *long-lived* C-14, I-129, Ni-59, Ni-63 as weak sorbing and Nb-94, Tc-99, Am-241, U-234, U-235, U-238, Np-237, Pu-238, Pu-239, Pu-240 as strong sorbing with total initial activity estimated to ~1.1×10^9 MBq is under investigation.

After disposal of radioactive waste the repository will be closed by constructing additional engineered barriers. The engineered barriers of closed repository consist of concrete vaults surrounded by low-permeable clay-based material and the whole system is being covered by long-lasting and erosion-resisting cap, Fig. 1. A fence at distance of 150 meters from the vaults will surround a territory of the repository. Sanitary protected zone (SPZ) of 300 m radius is planned for the facility.

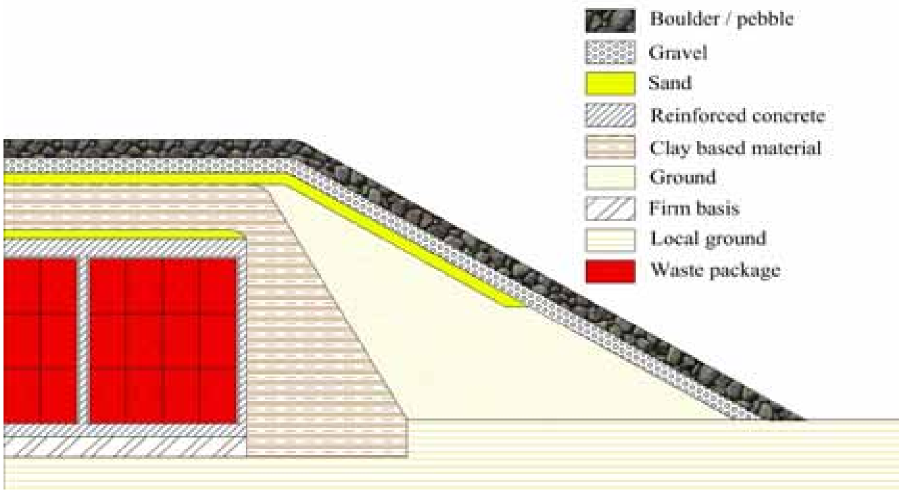

Fig. 1. Cross-section of the vault after closure of repository

A rather complex structure of the vadose zone has been assumed (generalized) as clay loam of 30 m thickness for Galilauke site, 3.5 m thickness for Apvardai site. The aquifer mainly consists of sand layer of 10 m and 2.2 m thickness at Galilauke and Apvardai site respectively.

The typical exposure pathway, exactly the well installed 150 m from the edge of the repository (next to the fence surrounding the repository), is determined for Galilauke site, but two exposure pathways, namely the well distant 150 m and the lake distant 1300 m from the edge of the repository have been determined in case of Apvardai.

5. Scenario development

Considering defined states of the engineered barriers (intact/normally degraded/completely degraded), two scenarios have been developed: *normal evolution scenario* and *barrier degradation scenario*.

The natural degradation of the barriers is assumed in case of *normal evolution scenario*. Hydraulic conductivity of clay barrier determines minimum infiltration rate (~0.02 m/yr) through the repository within analyzed period in this case.

In case of *barrier degradation scenario* the minimum infiltration rate through repository is assumed for period of institutional control (300 years) while maximum value of infiltration rate (~0.2 m/yr) is assumed after completion of institutional control due to sudden collapse of engineered barriers.

It is assumed that characteristics of geology and hydrogeology as well as biosphere remain stable within analysed period of time.

6. Model formulation

The conceptual model is developed on the basis of processes prevailing in the components of disposal system: due to water infiltration governed by state (evolution) of the engineered barriers the radionuclides leached from the waste packages (cement matrixes) through the bottom of the repository by diffusion/advection prevailing in the medium are transported toward vadose zone and further to the aquifer. Finally contaminants are discharging into well/lake water from which is used by humans for their daily needs. Radioactive decay is also taken into account.

7. Calculations

- Radionuclide migration in the repository and vadose zone has been assessed using 1-D equation of advective-diffusive transfer with respect to processes of dispersion and radioactive decay. Model is implemented in DUST computer program [4];
- Radionuclide migration through the aquifer has been assessed solving 1-D equation of advective transfer with respect to processes of dispersion and radioactive decay. Model is implemented in GWSCREEN computer program [5];
- Radionuclide transport and potential exposure to human in case of lake exposure pathway is implemented using AMMBER software [6]. 1-D differential equation modelling the radionuclide exchange between the components of the Lake system has been solved with respect to radioactive decay.

8. Analysis of the results

After assessment of migration of 18 radionuclides through the disposal system it is find out those only 12 radionuclides will reach biosphere zone. Cs-137, Sr-90, Pu-238, Pu-241, Ni-63, Am-241 will decay on the way in both cases of investigated scenarios, i.e. normal evolution as well as barrier degradation.

Normal evolution scenario

The estimated dose mainly should be resulted by C-14. After period of ~9-10 thousand years it should be of two orders of magnitude less than dose constrain of 0.2 mSv/year, i.e. negligible in case of Galilauke site. Very similar situation is revealed for Apvardai site in both cases of exposure pathways, well and lake.

Barrier degradation scenario

The estimated dose mostly should be resulted by C-14. After period of ~3-4 thousand years the maximum dose should be 0.036 mSv/year, i.e. factor of 5 less than dose constrain of 0.2 mSv/year in case of Galilauke site (Fig. 2). For Apvardai site in case of well (more important exposure pathway in comparison to lake), Fig. 3, it should be 0.16 mSv/year, i.e. ~20% below the dose constrain.

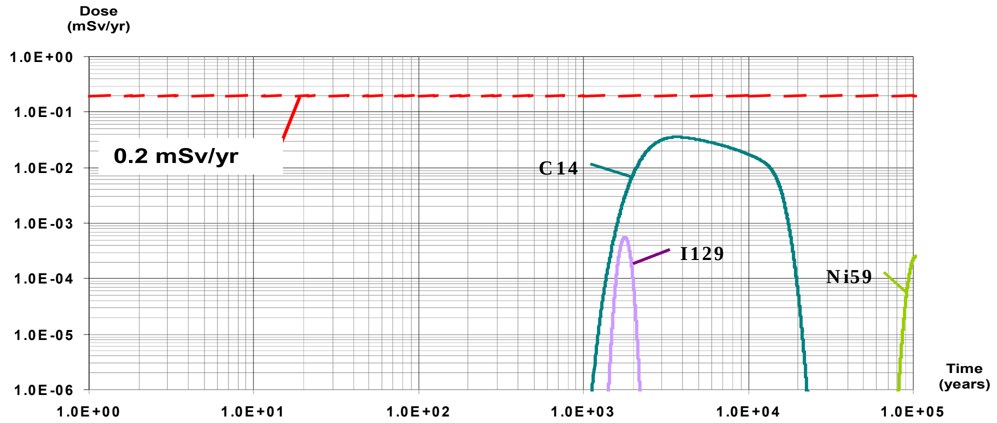

Fig. 2. Dose rate for well exposure pathway at Galilauke site in case of barrier degradation scenario

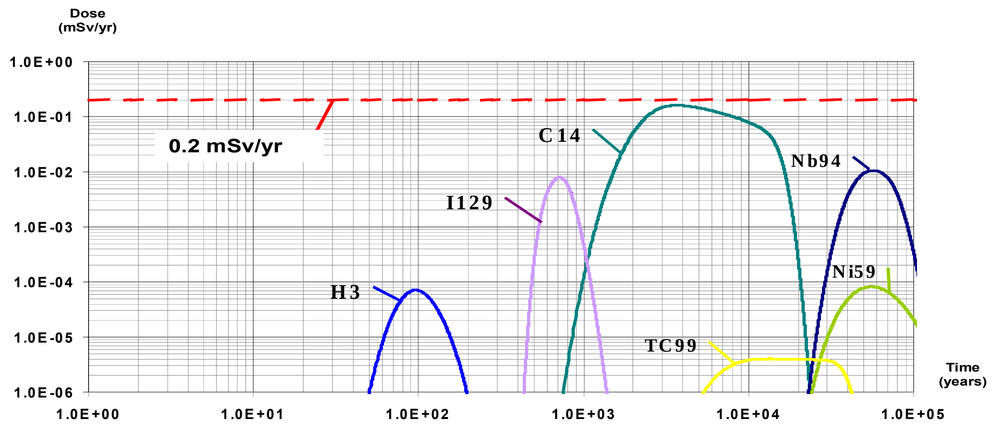

Fig. 3. Dose rate for well exposure pathway at Apvardai site in case of barrier degradation scenario

In general, the preliminary safety analysis has demonstrated that the dose constrain should not be exceeded at both candidate sites, moreover expected doses should be less at Galilauke site in comparison to Apvardai.

9. Confidence building

1. The results of the assessment should be accepted as conservative due to assumptions as follows:

- The percolation/saturation time through the repository is not taken into account;
- The radionuclide leaching from waste matrix is not modelled (instant release is assumed);
- Solubility limits of radionuclides are not taken into account;

- Initial activities of radionuclides are estimated for the beginning of operation period (not closure);
- 1-D dispersion is considered in transport analysis in aquifer zone for most radionuclides;
- Higher than average values for foodstuff consumption rates of local resident is used for dose assessment;
- Possible restrictions on activities within sanitary protection zone are not taken into consideration (i.e. well installed at the boundary of the repository).

2. Three types of uncertainties are analyzed:

(i) *Scenario uncertainties.* The alternative horizontal flow direction in vadose zone towards well has been investigated in case of Apvardai site. It appears that that dose is by 2-3 orders of magnitude less in comparison with case or vertical flow in vadose zone towards aquifer.

(ii) *Parameter uncertainties.* The variations of biosphere parameters (transfer factors, consumption rates, yield values, etc.) have been investigated for C-14 as most critical radionuclide using probabilistic simulation (Monte Carlo). The variation interval of doses is obtained and the upper bound of the interval is found within the limits of dose constrain.

(iii) *Model uncertainties.* The assessment results carried out for C-14 as most critical radionuclide using DUST for the near field and vadose zone, GWSCREEN for aquifer and, AMBER for biosphere zone are compared to the case when all components of the disposal system are modelled using only AMBER code. The difference only of 10 % in case of normal evolution scenario and 20 % in case of barrier degradation scenario is obtained.

10. Conclusions

After preliminary assessment of the potential releases of radionuclides from near surface repository to the groundwater pathway with respect to the estimated radionuclide inventory, conceptual design of the disposal facility intended to construct in Lithuania as well as peculiarities of the two candidate sites called Galilauke and Apvardai it is possible to conclude that:

1. Expected doses estimated for both candidate sites should be below the dose constraint of 0.2 mSv per year established by regulations of Republic of Lithuania.

2. Expected doses should be less in case of Galilauke site in comparison to Apvardai site.

3. Due to rather conservative assumptions taken into account for the analysis of radionuclide migration the estimated doses should be considered as overestimated.

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PRE-CONCEPTUAL STUDY ON THE REVIEW FRAMEWORK FOR THE RADIATION SHIELDING SAFETY OF THE PWR SPENT FUEL CASK INTERIM STORAGE IN KOREA

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ABSTRACT

In Korea, 20 nuclear power plants are in operation and lots of spent fuels are on the onsite storage. The onsite storage capacity in Korea is supposed to be full around at the year of 2016 and interim storage facilities could be considered to be constructed before 2016. A review framework to evaluate the radiation shielding safety of the interim storage facilities is developed in this study. It includes acceptance criteria, review procedures and activities of independent analyses. A case study is performed to apply the review framework. Modeling the review reference storage, evaluating the source terms and calculating the photon fluxes are performed. It is shown that the application of the review framework could satisfy the regulatory demand that would arise in the near future in the review area of the radiation shielding safety of the interim storage in Korea.

1. Plan for Interim Storage for Spent Fuel

Spent fuels generated from nuclear power plants in Korea are on the onsite storage in each unit. The onsite storage capacity in Korea is supposed to be full around at the year of 2016. Some plants had already encountered the excess of their initial storage capacity. For those plants of PWR type, the storage capacity has been expanded by adopting and implementing high-density wet storage racks. For PHWR reactors, an on-site dry storage facility has been constructed so as to expand their storage capacity.

The latest national policy for spent fuel management in Korea was made in 2004. It is that spent fuels should be stored within each nuclear power plant until 2016 by the additional expansion of the onsite storage capacity and the spent fuel management strategy including the construction of the interim storage facility shall be decided in a timely manner through national consensus by public consultation among stakeholders[1].

The detailed schedule for the interim storage for spent fuel is not yet decided. But, it can be seen that the interim storage facility might be in preparation by 2016 and the related researches should be ongoing. Also, it is supposed that a regulatory demand on the safety evaluation would arise before the year of 2016. In this study, a review frame for the radiation shielding safety of interim storage for PWR spent fuels is pre-conceptually developed to prepare for the regulatory demand. A case study is performed to apply the review framework.

2. Review Framework for the Radiation Shielding Safety

The main purpose of the study is to set up a review framework to evaluate the radiation shielding safety of the interim storage facilities. The review framework developed in this study is pre-conceptual, for the details of the interim storage facilities are not determined and the framework can be changed.

The review framework includes acceptance criteria, review procedures and activities of independent analyses. The acceptance criteria are input to review procedures and the compliance with the criteria is output from the review procedures. Review procedures demand independent analyses such as source terms evaluation and shielding calculation. The simple diagram of the review framework is in Fig 1.

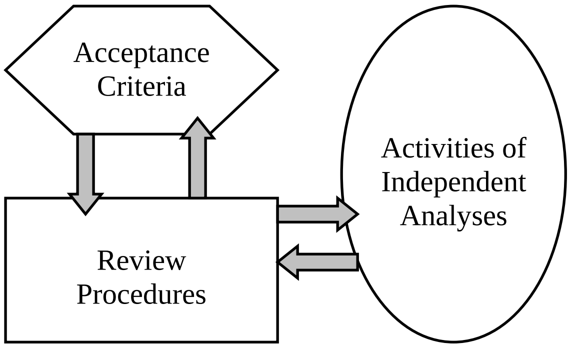


Fig 1. A simple diagram of the review framework

2.1 Acceptance Criteria

The acceptance criterion is assumed pre-conceptually to be the dose limit of 0.25mSv/hr at the boundary of the controlled area under the normal condition. It is from the U.S. Federal law of 10CFR72.106. It might be changed at the final review framework when the setup of the criteria of the interim storage in Korea is completed.

2.2 Review Procedures

Review procedures are consisted of several successive steps. A series of reviews for various areas is performed. This is the main review stage and the review conclusion will be produced.

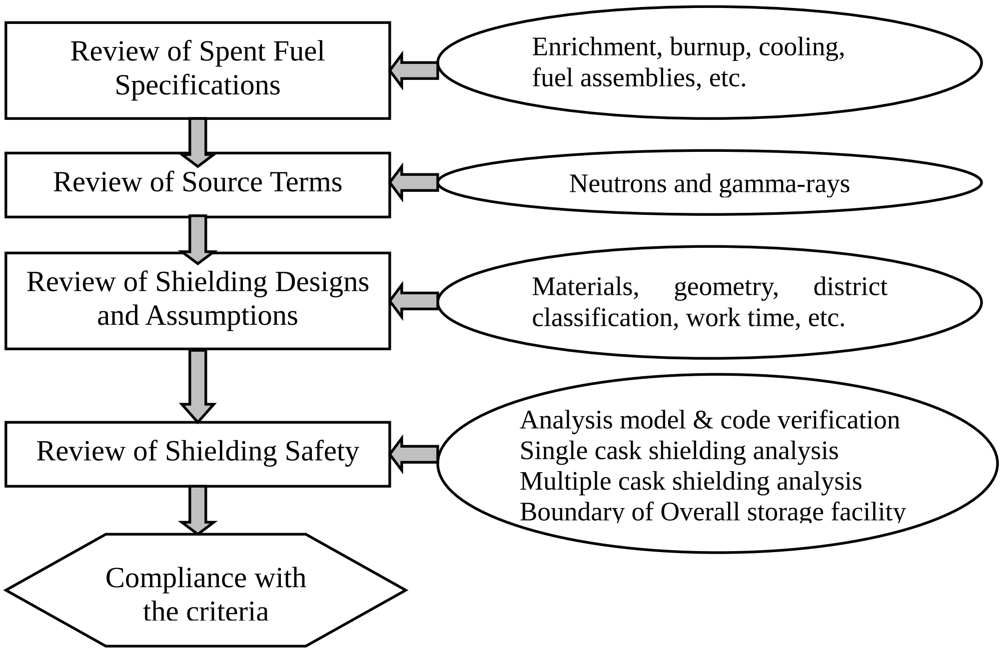


Fig 2. A simple diagram of review procedures

2.3 Activities of Independent Analyses

In the review procedures, independent analyses for source terms and shielding are required. To perform these analyses, proper tools could be chosen such as ORIGEN2.1, ORIGEN-ARP, MCNP, QAD, DORT, SCALE, etc. It is recommended that the different approach to modeling and codes from those of the utilities be applied in the independent analyses.

3. Case Study

For case study, some parts of the review framework are pre-conceptually applied to the PWR spent fuel cask interim dry storage. A review reference storage model is established and preliminary analyses for source terms and shielding performance are performed.

3.1 Review Reference Storage Model

A review reference is modeled for interim dry storage of the PWR spent fuels whose characteristics are 5 w/o U-235 enrichment, 50,000MWD/MTU burnup, 7 years of cooling and a storage capacity of 24 fuel assemblies.

3.2 Source Terms

The energy spectrums and the intensities for neutrons and gamma-rays are calculated and applied as the radiation source terms for the shielding analyses. ORIGEN-ARP (SCALE 5.0)[2] code and the embedded libraries are utilized to evaluate the source terms of the spent fuels.

3.3 Shielding Model and Analysis

To perform the shielding analysis, SAS4 (SCALE 5.0)[2] and QAD-CGGP-A[3] codes are used. Homogenization of the spent fuels, three-dimensional geometry modeling and the dose conversion factor of ANSI/ANS-6.1.1-1977[4] are applied. For a case of the single cask storage, dose rates are calculated by SAS4 and QAD-CGGP-A. For multiple casks such an array of 2x5, QAD-CGGP-A code is utilized due to its array function. Simplified geometry modeling of the cask storage is in Fig 3.

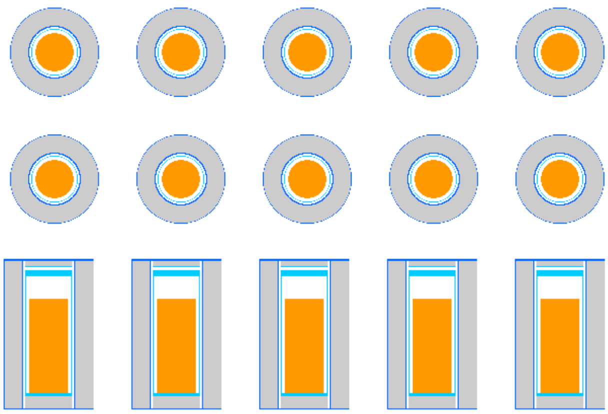


Fig 3. Simplified Cask Geometry Modeling of 2 x 5 array

3.4 Results

Dose rate are calculated to the 400 meters away from the cask storages. When the direct radiation is only considered and the effects of radioactive effluents not included, then, the distance in compliance with the dose limit of acceptance criteria is shown to be around 200 meters away for the one cask storage. The curves of dose rates for one cask storage are in Fig 4.

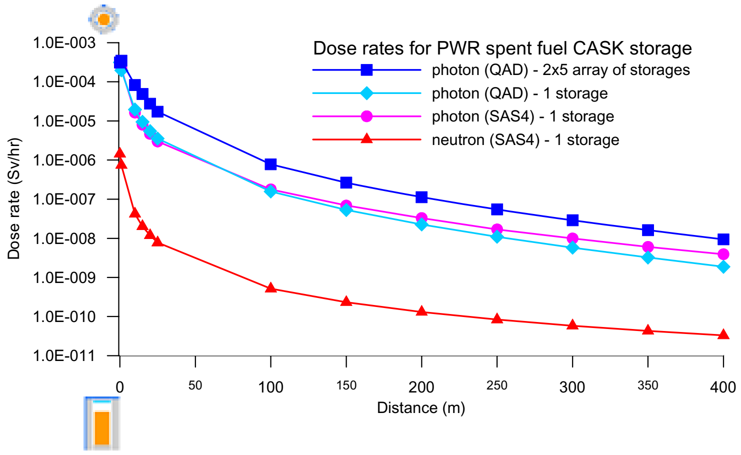
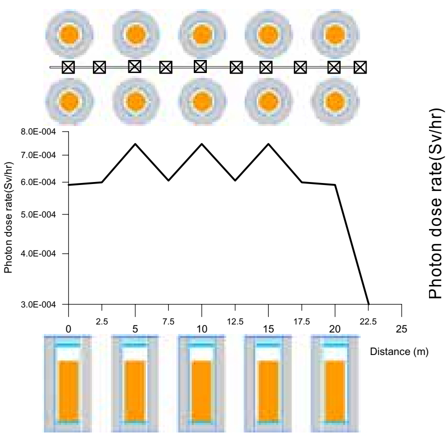


Fig 4. Dose rate for the one cask storage

For the array of 2x5 cask storages, the dose limit is satisfied as the distance of 300 meters. The photon dose rates within the cask storage and with the distances are shown in Fig 5 and Fig 6, respectively. At the final review stage, all the contributions to the dose including radioactive effluents would be summed up to evaluate the boundary.



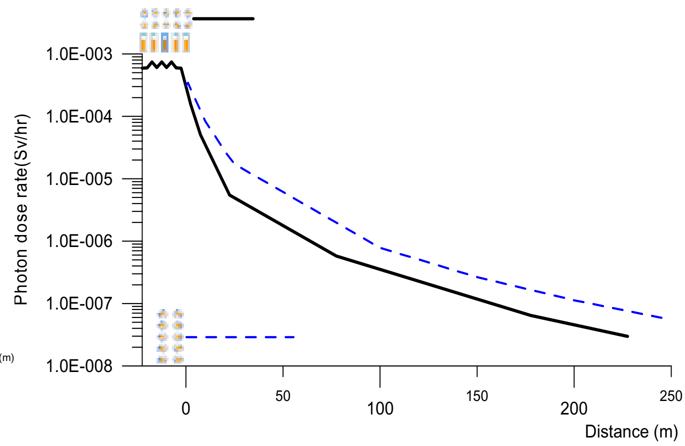


Fig 5. Photon dose rates within the array

Fig 6. Photon dose rates for the array

4. Conclusion

In this study, a pre-conceptual structure of the review framework for the shielding safety evaluation is developed and preliminary activities of independent analyses including modeling the review reference storage, evaluating the source terms and calculating the photon fluxes are performed. Now the review framework is being developed and not completed yet. Currently activities of analyzing the effects of neutron streaming, Co-60 gammas and radioactive effluents are missed. The further study will be continued to set up the review framework completely to cover the missed issues. After the review framework is completed later, it is expected that the application of this review framework will satisfy the regulatory demand that will arise in the near future in the review area of the radiation shielding safety of the interim storage in Korea.

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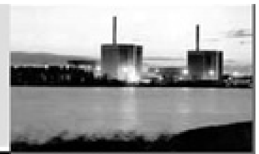

SESSION III : Experience with Existing LLW/ILW Storage and Disposal Facilities

PRESENT ISSUES FOR CENTRE DE LA MANCHE DISPOSAL FACILITY

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ABSTRACT

Centre de la Manche disposal facility officially entered its institutional control period in January 2003. Andra performs monitoring of the environment and of the capping system in order to prepare further phases that should become more and more passive. A detailed "long term memory" has been established in order to provide future generations with the relevant information about the facility.

1. Introduction

Centre de la Manche disposal facility is the first French radioactive waste surface disposal facility. It was operated between 1969 and 1994. 527,000 m^3 of packages of low and intermediate level short lived waste have been disposed in the 12 ha of the facility. To prepare its closure capping works were performed between 1991 and 1996, water tightness being provided by a bituminous membrane. The facility officially entered its institutional control period in January 2003.

Figure 1: Centre de la Manche disposal facility

The way the facility will be monitored will evolve in a step by step approach. An update of the safety report is required in January 2009 in order to take into account collected data from the environmental monitoring. The report should also include proposals for modifications of the facility, in particular the capping system, in order to make the monitoring more and more passive.

2. 25 years of operation and improvement of Centre de la Manche

During 25 years of operation of Centre de la Manche, continuous efforts were made to improve operating conditions of the facility . The design of Soulaines-Dhuys (Aube Prefecture) disposal facility that superseded to Centre de la Manche in 1992 and the way this facility is operated are the result of these permanent efforts to improve, in particular the design of the disposal structures and waste package specifications.

2.1 The design of the disposal structures

When the facility started up the only requirements by the 1969 Order of Creation of the facility were the followings:

- Direct disposal in the ground is authorised for waste that is conditioned in such a way that leaching by water does not induce any hazard (waste that is grouted in concrete or metallic drums);
- Low level waste in drums may be disposed in the ground provided that it is covered by a plastic and bitumen protection and that a drainage system is implemented at the bottom of the disposal cell;
- Intermediate level waste and bulk waste must be grouted in concrete disposal cells.

Figure 2: concrete disposal cells

Figure 3: disposal vaults with a steel reinforced concrete bottom slab

From this initial simple design improvements were made and resulted in two major changes:

- The fact that water going inside the disposal vault has to be considered as an effluent and to be clearly separated from rain water. Therefore a specific collecting system was implemented between 1979 and 1982 in an underground monitoring gallery;
- The inclusion of a steel reinforced concrete bottom slab that provides resistance to the burden of the disposed waste packages and that provides water-tightness in order to prevent infiltration of effluents in the water table. Therefore direct disposal in the ground was quickly abandoned.

These principles have been used since 1984 for the design of vaults. Effluents are transferred to AREVA's reprocessing plant close to Centre de la Manche.

2.2 Waste packages specifications

The Order of 1969 creating the facility was referring to the maximum concentration in drinking water to define the disposal modes, water being the main contamination vector. When operations started at the facility, there was practically no activity limit for the waste, except to define the packaging and disposal modes.

In 1984 Fundamental Safety Rule RFS I.2 formalised the incorporation of the long-term safety objectives into acceptance criteria. It took into account that, in the post-institutional control phase after no more than 300 years, the intrinsic safety of the disposal facility relied partly on the initial activity limitation in long-lived emitters of the disposed waste.

As a complement, Fundamental safety Rule RFS III.2e imposed systematic waste packaging and established minimal characteristics, particularly with regard to containment, with which packages must comply depending on the nature and activity of the waste. A minimum mechanical strength was required in order to integrate packages in the disposal architecture. Andra reflected all these requirements in its technical specifications in 1985.

2.3 The capping system of Centre de la Manche

Due to the changes in the design of vaults or in waste specifications, it was decided to implement a capping system that would be able to sustain strains that might be caused by settlement of waste packages in the eldest parts of the facility. Therefore a bituminous membrane was selected. Such a membrane, 5.6 mm thick, has good water tightness properties (significantly less than the target of a few litres per square meter and per year). Its maximum strain before failure under a tensile stress is 45 to 50%.

The “roof” of the capping system is made of successive tilted sectors of 25 meters with slopes between 6 and 14% above the repository. Because of the limited area of the facility, the slopes on the side are steeper (about 25°).

Different layers are implemented. In particular they include permeable levels under and above the membrane and coarse material (1.2 m) to protect the membrane against intrusion by animals or roots. Infiltrated water above and under the membrane is collected by drains and can be monitored.

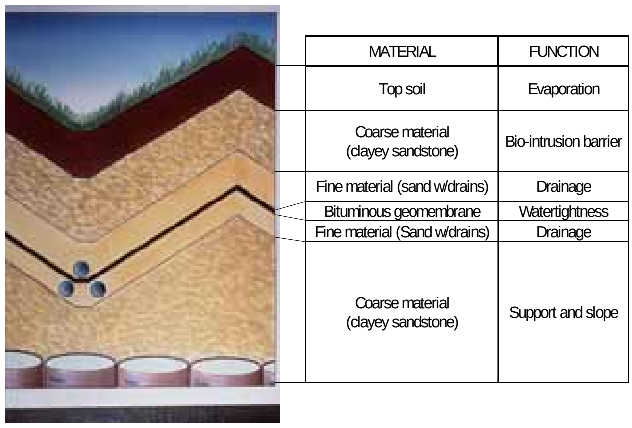

Figure 4: Typical cross section of the capping system

3. The institutional control period

3.1 Monitoring the capping system

Monitoring of the capping system is performed through 3 main undertakings:

- The surveillance of surface movements by periodic topographic measurements. These measurements actually show settlements on the “roof” in some places. A maximum displacement of 50 cm since 1996 is detected in one place.
  However other displacements are measured on the slopes of the capping system that correspond to the consolidation or slide of the covering material along the membrane. These movements are continuous with a velocity of 1 or 1.5 cm per year (3.5 cm per year in one sector).

- The monitoring of collected water in the different networks of the facility.
  Measurements since the beginning of capping works show that the collected volume of water from the vaults has been divided by 75 and its activity by 130 to 270. This volume should be reduced as some infiltrations that do not percolate through the vaults are detected in connecting pipes at the boundaries of the repository.
  The target value of infiltrated value is achieved: 0,08 l/m^2/year in drains under the membrane, 2 l/m^2/year from the vaults.

Effluents collected from vaults	**1991**	**2005**
Volume	21,000 m^3	280 m^3
Beta activity (but ^3H)	1.3 GBq	0.008 GBq
^3H activity	1,700 GBq	6.2 GBq
Alpha activity	0.4 GBq	0.003 GBq

Table 1: effluents collected from vaults

- Diagnostics of the membrane by periodical samplings every five to ten years. Two samples (2.5 m X 3 m) were taken in 1997 and two in 2005. Tests are performed on the membrane (thickness, mechanical strength, water tightness), on the bitumen (softening temperature, asphaltene content) and on welded areas (mechanical strength). No indication of ageing can be detected at present.

Figure 5: membrane sampling

Figure 6: replacement of the sample

3.2 Environmental monitoring

Chemical and radiological measurements are performed to verify that the facility complies with its applicable regulation, in particular in terms of discharge. These measurements enable to assess its impact. Measurements are made in the water networks, in rivers, in the ground water, in vegetation and in the air.

As a tritium contamination was detected in the river in 1976, groundwater is monitored by 73 boreholes drilled around the facility and its vicinity. Tritiated waste was retrieved in 1977 - 1978, however tritium diffused in the ground. The average tritium activity at the boundaries of the facility is now decaying (by a factor 2.35 between 1996 and 2005 – 6240 Bq/l in 2005), even if in some boreholes the level of activity remains significant (maximum: 190,000 Bq/l, the other values are between 100 and 24,000 Bq/l). The activity in the river is 100 Bq/l and 700 Bq/l in a contributory stream.

The present impact on a critical group living along the river is less than 0.8 µSv/year.

3.3 Keeping memory of the facility

The duty of memory constitutes a statutory requirement for Andra and the Agency pursues four major objectives:

- to inform future generations about the existence and the contents of the site, especially with regard to the risk of human intrusions, in case the facility was forgotten;
- to facilitate the understanding of the observed phenomena;
- to ensure that any relevant corrective actions be carried out under safe conditions, if necessary;
- and to allow for future generations to make any decision concerning the future of the site, especially in response to technical and societal developments.

In order to be sure that future generations will always have the proper means to keep abreast of all developments concerning information systems, Andra copied the documents on “permanent” paper. “Permanent” paper complies mainly with an international standard (ISO 97.06). Provided that it is handled with care and that it is kept in suitable premises, it is designed to remain stable over several centuries and therefore constitutes a sound solution in response to technical evolutions.

Figure 7:
copy of documents on permanent paper

Documents for a detailed long term memory are selected when relevant concerning potential risks. For Centre de la Manche, thirteen scenarios of potential incidents were developed in accordance with the safety report. Those scenarios are divided into three levels. For each of those scenarios, questions are raised, and the required knowledge is explained for each answer in order to provide feedback elements allowing future generations to make their own decisions. The purpose is to provide information to the operator of the facility (or by any person who, in the far future, would be in charge of managing or transforming it).

The “detailed memory” of Centre de la Manche was compiled at the beginning of the monitoring phase of the facility in 2003, and includes 10,732 documents of a total of 442,938 pages, and spread over 60 linear metres. A copy was transferred in 2004 to French National Archives, another copy is kept by Andra.

A “synthesis memory” of about 400 pages will also be established for local and national decision makers. It will contain the most important information (historical and descriptive briefing notes, summaries of inventories and regulatory applications, etc…).

4. Conclusion

The update of the Centre de la Manche safety report in 2009 will include proposals for the transition from a very active monitoring to an active monitoring period. Proposals will concern possible improvements and modifications of the capping system in order to deal with the problem of steep slopes, of settlement of ancient waste. It will provide the results of research concerning the service life of the membrane and about other substituting materials. Some technical elements of the facility will be investigated with a goal of simplification and to have passive operating systems. A review of the environmental monitoring program will be performed. The active monitoring period will be followed by a passive monitoring system in a step by step licensing approach.

Centre de Soulaines-Dhuys disposal facility (in the Aube prefecture) that superseded to Centre de la Manche takes benefit of the experience gained since 1969. At the end of 2005 about 183,000 m^3 of waste packages have been disposed in Centre de Soulaines-Dhuys, which has a capacity of 1,000,000 m^3. In particular Andra has a very cautious approach with the acceptance of waste containing tritium.

Figure 8: experimental capping system

As 110 disposal vaults have been constructed, a continuous area in the eastern part of the facility will be occupied by closed vaults in a few years. It will be then possible to begin partial capping work. The reference design uses clay as an impermeable material. An experimental capping system is being tested since 1995. Its results will also be used for Centre de la Manche studies in the same way that Centre de la Manche feed back will be used to prepare a report on the capping system of Centre de Soulaines-Dhuys to be submitted to the regulatory body in 2012.

The experience feedback from the implementation of the detailed memory of the Centre de la Manche was instrumental in highlighting the need to establish that memory as documents were produced. Hence, every document produced by Andra or received from its suppliers is input in the Agency’s information system (document software for content management) and “ticked off” or not to indicate whether it will be part of the detailed memory or not. Those documents are printed on a regular basis on permanent paper (two copies). In 2005, the first shipment to the French National Archives was performed for documents concerning the first 10 years of operations at the Centre de Soulaines-Dhuys disposal facility.

Operational experience from SFR – Final repository for low- and intermediate level waste in Sweden

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ABSTRACT

SFR, the Swedish Final Repository for Radioactive Waste, has been in operation since April 1988. It was designed for short lived LLW/ILW from the operation and maintenance of all Swedish Nuclear Power Plants.

The first stage was constructed for 63 000 m^3 which was assumed to give a margin and flexibility for the preliminary operational period. Today this volume represents the whole prediction of operational waste. Until the end of 2005 SFR has received 30 930 m^3 waste.

In average it has been 2-3 derivations per year at the repository. The most derivations happened in the years 1993-1995, and that was also the years when the repository received the most volume of waste. The most of the derivations those years was related to the waste packages.

The doserate to the personal has always been very low in the latest years the collective dose has been under 0,1 mmanSv/year.

1. Introduction

SKB (Swedish Nuclear Fuel and Waste Management Co) is owned by the Swedish Nuclear Power utilities and has been appointed as responsible for the management of Sweden's radioactive waste. The final repository for radioactive operational waste, SFR, has been in operation since 1988. All the short-lived waste; low level waste (LLW)and intermediate-level waste (ILW) from the operation and maintenance of the nuclear power plants is disposed in SFR, along with radioactive waste from medical use, industry and research.

SFR is owned by SKB and the operation and maintenance of the repository is performed by Forsmarks Powergroup who is contractors to SKB.

SFR has five different rock chambers for disposal of different kind of waste. The most active waste is disposed of in a concrete silo surrounded by a clay buffer. The other four chambers consist of a cavern for LLW (BLA), two caverns for concrete tanks with dewatered ion exchange resins (BTF1 and BTF2), and a cavern for ILW (BMA). BMA and the silo are for ILW and the three other caverns are for LLW.
The repository is located close to the nuclear power plant at Forsmark, in crystalline bedrock, 60 m under the bottom of the Baltic Sea. The entrance is at the Forsmark harbour and two tunnels leads to the disposal area, 1 km from the shore. SFR consists of an aboveground section and an underground section. The above ground section consists of office, workshop, terminal building for transport containers and the ventilation building.

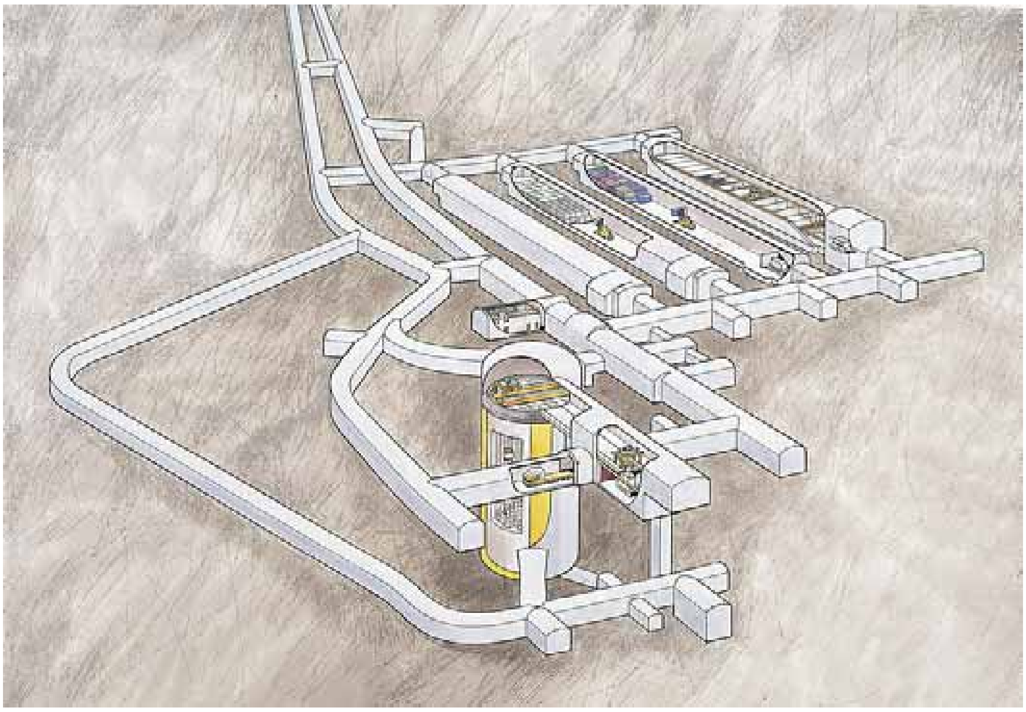

Fig 1. SFR

The five rock chambers have different barrier systems. The most active waste is deposited in a 50 m high concrete silo surrounded by a clay buffer. Besides the silo the repository consists of four more simple rock caverns. These four caverns have a length of 160 m each.

In the three caverns for LLW the waste packages are handled with a fork lift truck. In the cavern for ILW in the silo the waste packages are handled with remote controlled overhead cranes.

2. Operation experience compared to design phase assumptions

SFR has been in operation since 1988. When SFR was constructed in 1980 the repository was dimensioned for prediction of operational waste, 90 000 m^3. The first stage was constructed for 63 000 m^3 which was assumed to give a margin and flexibility for the preliminary operational period. Today this volume represents the whole prediction of operational waste.

SFR was designed for disposal of 30 m^3 waste per day but today only about 1000 m^3 waste is disposed a year. This because of that the waste producers have become better on minimising the amount of waste and the very LLW are disposed of at the power plants in surface disposals.

In the beginning of operation the transport vehicle that transports the waste down to the repository was remote controlled and electrically operated without any driver in the vehicle. We soon discovered that it was better to have a driver in the vehicle but it was still electrically operated. This gave us a lot electrical problems because of among other thing the humidity in the repository. So today the vehicle is diesel powered.

It’s important that the slope down to the repository is kept free from snow and ice in the winter season. Otherwise it can be problems for the fire brigade to drive down to the repository in case of fire. For that reason a electrical cable was laid in the roadway to melt the snow, this has risen the electrical costs. To minimise the costs for heating of the roadway we nowadays use waterborne heating using the heat from the ventilation from the rock.

A large problem in the repository is the high humidity; in late summer the humidity are 100 %. Because of the high humidity in the repository we have problems with corrosion of metal parts. For example we have had high costs for exchange of ventilation tubes in the repository in the end of the 90ies and we now have a new exchange campaign of metal parts ahead of us. We also have some corrosion problems on some waste packages. (See 4.3) We are now investigating how we can try to reduce the humidity in the repository and we try to warm the ingoing air as much as possible. To do that we use two heat pumps.

3. Radiation protections and occupational dose rate experience

The only part of the handling in SFR that is done manual of unshielded waste packages are deposition of waste in three of the disposal caverns, BTF1, BTF2 and BLA. The handling in those caverns is done whit forklift trucks.
The maximum surface dose rate of the waste in BTF1 and BTF2 is 10 mSv/h and in BLA 2 mSv/h.
In the original safety report the calculated doses were 7,5 mSv/year for disposal work. At that time the assumption was that 250 transport containers should be disposed each year. The representative quantity of transport containers is 30-60/year.
Also the transportation of the containers down to the repository caverns was supposed to give a dose of 7,5 mSv/year. In that case an assumption is made of 250 transport containers, a driving time of 3 hours/container and a doserate of 0,01 mSv/h. This is also a very conservative value compared with the operational experiences.

Transportation and deposition of the containers have given rise to very low personal doses.
From the operation start in April 1988 to the year of 2005 the collective and personal doses has been low. In the first 12 years when the deposition activity was higher the collective doses has been under 1 mmanSv/year. After the year of 2000 when the volume of disposed waste has been lower, the collective doses has been under 0,1 mmanSv/year.

The tasks that have given rise to the highest doses in the repository are different closure methods. For example, in one of the caverns, drums with ashes are disposed. These drums are grouted with concrete. The first time this was done, it gave rise to a dose of 6 mmanSv. The method has after that been modified to minimise the doses and the fourth and latest time the dose was 2 mmanSv.

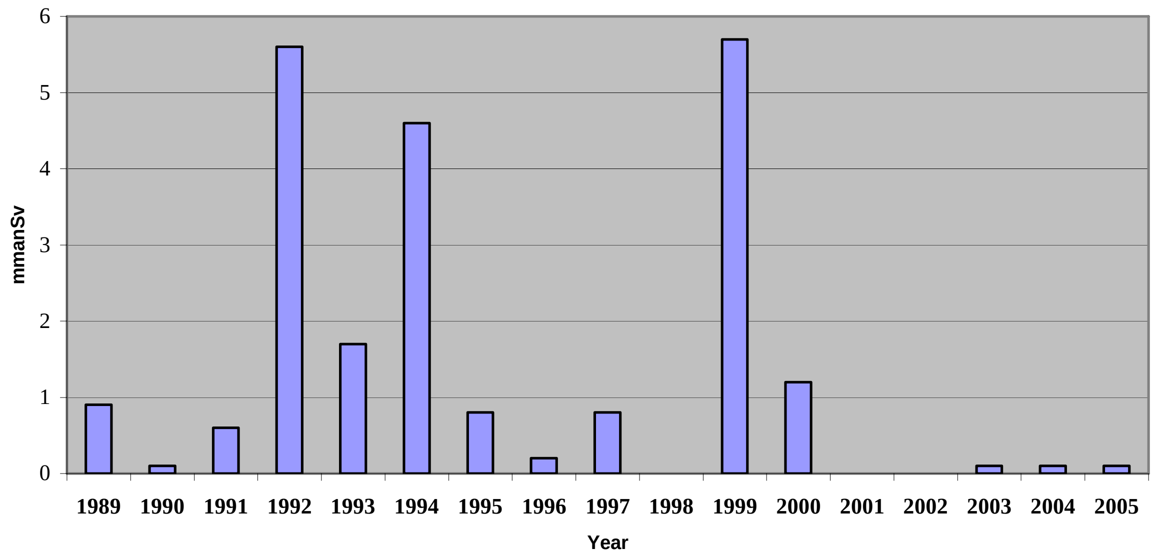

Fig 2. Collective doses at SFR

4. Type of incidents reported and corrective actions

In average it has been 2-3 incidents per year at the repository. Most incidents happened in the years 1993-1995, and that was also the years when the repository received the most volume of waste. The most of the incidents those years was related to the waste packages.

4.1 contaminated transport containers

Some transport containers has been contaminated from the waste packages. The contamination consists in several cases from contaminated water from corroded steel drums. The corrosion was caused by a fault in the drier equipment at the waste producer. After that the routine for the control of the container was made more stringent and all steel drums that could be corroded were put in a steel box before transportation to the repository.

4.2 Containers with broken lid

ISO containers with low level waste are being disposed in the repository and there fore the waste producer use used containers. Sometimes the conditions of these containers were relatively bad and containers with hole in the sides and on the lid arrived to the repository. To solve this problem the requirements on the waste producer was made stricter and the waste producer now use special lids for the containers sent to SFR.

4.3 Detected contamination in drainage water

When SFR was constructed it was assumed that the drainage water that passed throw the disposal compartments in the cavern for intermediate level waste, BMA, could be slightly contaminated. Therefore a separate collecting system for the drainage water was constructed. The drainage water collected in this system are analysed for activity.

In the summer of 2005 the activity level was increasing but the value was still under the limiting value. The probable cause for the increasing of the activity is that one or more of the steel drums in the compartment has been corroded. It is the same waste type that gave the problem with contaminated transport containers see 2.3.1. Too stop the increasing of the activity in the drainage water a mobile roof has been constructed over the compartment with this waste to avoid the water to drip over the waste drums. Today almost no water comes from the compartment. The compartment will be sealed as soon as possible.

This incident show us that the drainage system work and that we in a early stage find out when we have activity in the drainage water and we can take care of the problem immediately.

ASPECTS ON THE ACCEPTANCE OF WASTE FOR DISPOSAL IN SFR

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ABSTRACT

When licensing a final repository for radioactive waste certain assumptions have to be made concerning the waste. These assumptions cover radionuclide inventory and non-radiological materials and its physical and chemical impact on the waste, the repository and on the environment.

Development of new waste treatment systems and waste packages at the waste producer site aim at finding solutions and products that can be stored, transported and disposed of safely and are economically sound.

This paper discusses some aspects concerning development of new or modified waste products. It highlights the importance of analysing the whole sequence in treatment, handling and disposing the waste. The process should be to find an optimal solution for the whole system, considering the fact that what is best in one step it not necessary best for the whole system, including the post closure issues.

1. Introduction

When a nuclear facility plans for a new waste processing system or modification of an existing it is of outmost importance to analyse the whole chain of operations from the main operation of the facility to the final storage of the rest products produced as a consequence of the main operation. This analysis should contain aspects like cost of operation and for final storage of waste, safety of personnel, long-term storage considerations, etc.

This paper discusses some of these considerations from a repository operator's point of view, specifically for the SFR-repository (Final repository for operational radioactive waste) located near the Forsmark nuclear power plant.

2. Waste acceptance criteria on a waste package for final disposal

2.1 General set of Waste Acceptance Criteria for waste disposed off in SFR

A set of general and qualitative criteria for acceptance of waste for disposal in SFR has been developed. These criteria are given to the waste generators and SKB by the authorities, the Swedish Nuclear Power Inspectorate (SKI) and the Swedish Radiation Protection Agency (SSI). For each type of waste the criteria have to be quantified as far as possible. As a result there will be a unique set of acceptance criteria for each waste type.

Below is given a qualitative set of criteria that should be considered and when appropriate specified in the Waste Type Description (WTD) that must be delivered with each type of waste. The criteria are valid for all steps in the handling of waste. Limiting, quantitative values on each parameter are used as acceptance criteria.

<u>Design, geometry and dimensions</u>
The design, geometry and dimensions of a waste package shall be in compliance with the systems for handling and transportation and with the appropriate disposal part of the repository.

Weight
The weight of a waste package must not exceed the limits set by the handling and transport systems. The distribution of mass within a waste package must not jeopardise the stability during handling operations such as lifting and stacking.

Marking
Each waste package shall bear a unique identification marking. The marking shall be such that it pertains until backfilling around the package takes place in the emplacement vault. The marking shall be documented in the waste producer's register and in the SFR register. It shall also enable the package to be localised in the repository.

Radionuclide inventory
The contents of gamma emitting radionuclides shall be known in terms of species and quantities for each waste package.

An account of the contents of alpha and beta emitting radionuclides shall be given. The accuracy of this inventory shall be sufficient to assure compliance with given limits for different kinds of packages and different emplacement cavities in the SFR.

Surface dose rate and dose rate at a certain distance
The maximum external dose rate, measured and reported as surface dose rate and/or dose rate at a certain distance (normally 1 m) from the outer surface of the package, shall be lower than the limits applicable for the facilities where the packages are stored and the equipment used for their handling. Limits for transportation shall be taken into account.

Surface contamination
Transferable contamination by radionuclides, i.e. contamination that might be released from the outer surface of the waste packages during normal handling or pouring of water for a short time, shall be kept within authorised limits.

Internal radiation
Internal dose rates as well as the internal integrated radiation dose, must not be as great that processes induced by radiation, e.g. radiolysis, affect the properties of the waste form, packaging and the barrier functions in the repository to an unacceptable extent.

Homogeneity
Solidified liquid and wet waste shall be homogeneously distributed to an extent that radionuclides never occur in such concentrations that the above mentioned radiological properties will be affected to an unacceptable extent.

During package and grouting of solid waste, components etc., the active material shall be emplaced in the packaging in such a way that the activity becomes distributed throughout the packaging as homogeneously as possible so that the mechanical and physical-chemical properties assigned to the waste form from the aspects of safety and radiation protection are not jeopardised and can be assessed with sufficient accuracy.

Composition and structure
The chemical composition and structure of the waste form and its packaging shall be known and defined to such an extent that it allows an assessment of the material properties of the waste package.

Liquids
Waste packages are not allowed to contain free liquid that, due to leakage, might lead to unacceptable radiological consequences.

Corrosion resistance
Waste packaging shall have a durability against external and internal corrosive attacks that is sufficient with regard to conditions before backfilling around the package or sealing of the repository cavity.

Gas formation
Gas formation rate and volume, caused by the composition and structure of the waste form or package, shall not jeopardize the safety before closure of the repository or, after that, give rise to unacceptable disturbance of the barrier functions of the repository. Different mechanisms and processes for gas formation shall be regarded, e.g. radiolysis, biological decomposition, metal corrosion and other possible transformation processes. The content of organic substances and other biodegradable materials and metals whose corrosion might give rise to gas evolution shall be specified with sufficient accuracy to ascertain assessment of the consequences with respect to radiation protection. Wastes in the form of compressed gases are not to be disposed of in the repository.

Combustibility and fire-resistance
Combustible waste and waste forms shall be packaged in such a manner and have such characteristics that the risk of self ignition is negligible. Combustible waste shall be sufficiently well characterized and specified in terms of quantities and composition to permit the necessary precautions to be taken. Any fire shall be prevented from spreading through appropriate measures. Explosive materials are not allowed in the waste.

Chemical reactivity
Waste packages shall not contain substances that, due to their nature and quantities, might jeopardize the stability of the waste packages or the barrier functions of the repository to an unacceptable extent. Complexing agents shall be avoided as far as possible.

Leaching
Leaching of radionuclides from waste packages must be within the limits given for transport of radioactive materials. The leaching properties must be in compliance with the assumptions made for calculation of the long-term safety of the repository.

Mechanical strength against external stresses
The mechanical strength of the waste packages against external stresses such as pressure, strain, bending and impact, shall be sufficient to preclude unacceptable releases of radionuclides during foreseeable incidents and accidents. The waste form shall have a structure and homogeneity that is in compliance with this requirement.

Mechanical stability
The structure and volume of the waste shall be such that they do not deteriorate to an extent, leading to unforeseen release. Examples of such processes are swelling of the waste form under pressure build-up and degradation of mechanical strength caused by changes in temperature.

2.2 Important concerns regarding the long-term safety of the repository

Traditionally the major focus on waste management has been on production and workers safety at the waste generators site, during transport to interim storage and handling in the repository. These are all very important aspects but are not sufficient. As the level of knowledge and modelling abilities on final disposal have been more and more sophisticated, the requirements on analyses and reporting have increased on the Safety Analyses Report (SAR) of the final repository for operational waste.

Thus, already before a waste processing operation at the waste generator can be allowed to start or the mode of operation changed for their main operation, the waste generator must analyse all consequences that the production of the waste can have on the chain of events from processing at the site to the possible risk of high radiation doses to the environment in the future.

3. Case study for start of a new operation at an NPP

At the Ringhals nuclear power plant work is ongoing to reduce the overall water release to the recipient. A general principle is to try to use as little water as possible, but this can only decrease the use of water to a certain degree, still the dominating method to minimise water release is reuse of the water.

In an ambitious effort Ringhals NPP has started a project looking at all aspects on the water "issue". The project examines all waste water streams from the site's 3 PWR's and one BWR. Important questions that need to be studied and answered are e.g.: should all water be treated in one plant; in one for the PWR's and one for the BWR; or should Ringhals 1 and 2 (BWR and PWR) and Ringhals 3 and 4 (2 PWR's) have separate treatment facilities. Further, the plant is looking at methods to clean the water for reuse, and methods to produce a waste package that is well suited for production, handling and disposal in the SFR-repository.

3.1 Evaluating some vital factors influencing the long-term safety of the repository

Contrary to the "technical approach" to solving a problem, i.e. find a most efficient technical solution to the water clean-up, Ringhals decided to look at what properties a final waste package should have, so that the authorities could accept it for long-term storage in the SFR-repository without too much further investigations. Regardless of which method Ringhals finally will choose for purifying the water for reuse, the waste produced will be a radioactive more or less liquid salt residue. As the liquid waste water treatment plant in Ringhals is a cement solidification unit it is of vital importance to choose a solidification process that can be treated by the Ringhals waste plant, and that the properties of the produced waste package are such that the long-term safety of the SFR-repository will not be jeopardized.

One problem with cement solidification of PWR-waste is that boron is a cement retarder, hence, with an improper amount of boric acid in the waste stream it will not be possible to produce a solid cement waste matrix. Further, a high salt content is likely to significantly react with the cement matrix, in many cases without causing significant harm to the cement structure, but especially the anions sulphate, chloride and carbonate will react with the cement and can cause serious fractures and damage of the waste matrix, and in the worst case directly or indirectly by a swelling waste package cause damage on the main concrete structure of the SFR.

With these concerns in mind, Ringhals decided to thoroughly investigate how to produce a robust cement waste matrix, and how this cement waste matrix may interact with the SFR repository structure, and as a consequence how the long-term safety of the SFR repository may be influenced by this new waste type. Thus, Ringhals started a major laboratory programme studying different cement waste solidification formulas ("solidification recipes") in their newly equipped cement laboratory (This program will not be further discussed in this paper.). But they also studied the long-term influence this new waste type may have on the SFR-repository, in order to produce an as complete as possible WTD for the application to SKB and the authorities.

Three types of salt waste streams were studied both in laboratory and theoretical modelling, one with a fairly low salt content (ca 10-15% dry substance), one with significantly higher salt content, and finally one waste type with only the salt and basically no free liquid present.

3.2 Theoretical modelling of cement – salt interaction

The long-term salt - cement interactions were modelled using a modified version of the PHREEQC-2 code [1]. To the original PHREEQC-2 code were added subroutines for a "dynamic" porosity and diffusivity, i.e. after each run a new porosity was calculated based on the mineralogical changes of the system, and the diffusivity was calculated using the updated porosity. The work is on-going and will be presented by Ringhals later this year, but an example of the cement-salt mineralogical composition as a function of time up to 100 000 years after repository closure is given in Figure 1, and the porosity change as a function of time for the same part of the studied system is given in Figure 2.

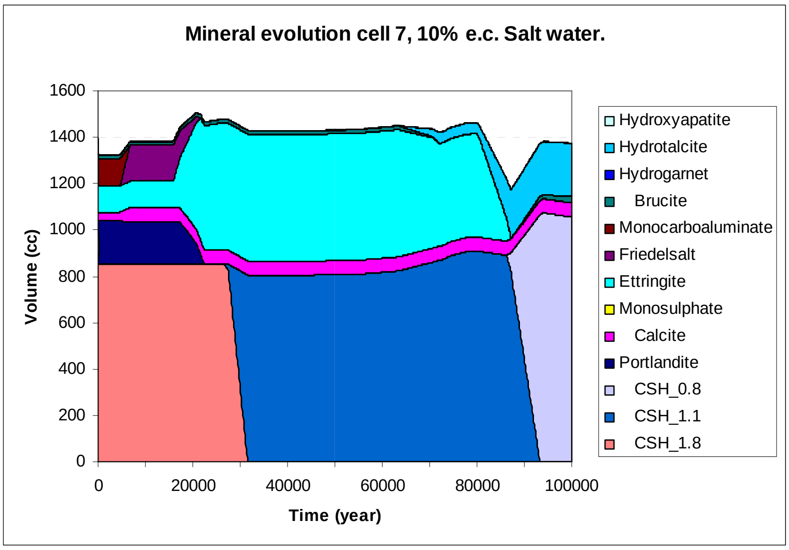

Figure 1. The mineral evolution in cell 7, the outer part of the cement-encapsulated evaporator concentrate, when exposed to salt water (10% e.c.). The order of the minerals is as given in the captions, i.e. from top to bottom.

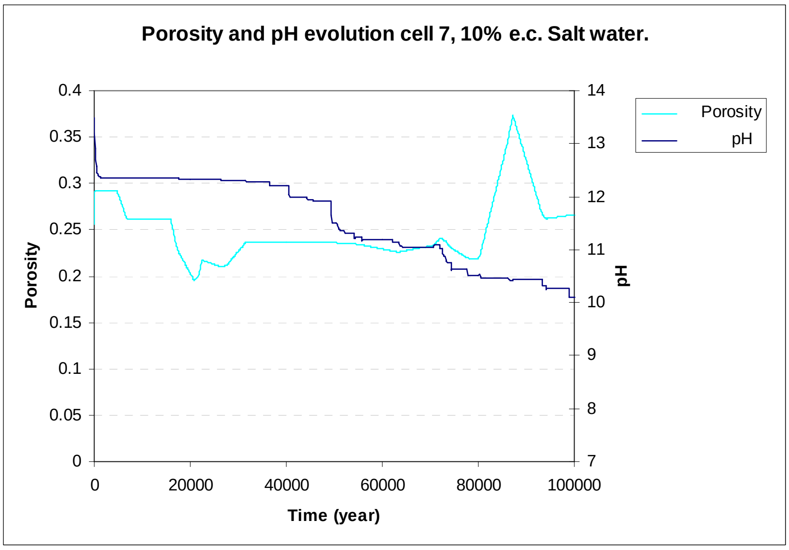

Figure 2. The porosity and pH evolution in cell 7, the outer part of the cement-encapsulated evaporator concentrate, when exposed to salt water (10% e.c.).

4 Conclusions

All aspects must be included in the evaluation of the best technique to use, when planning for a new or modifying an existing waste treatment facility which will produce a radioactive residue that need to be taken care off. To speed up the time from starting a project for treatment of radioactive waste, to when the system can go into "industrial operation" it is of outmost importance to look at the whole chain of operation including the final storage, and the influence the new waste type may have on the long-term safety of the final repository. Without this complete analysis it will be difficult or impossible for the competent authorities to issue a license for operation, and if the waste type is not well-suited for long-term storage, a proposed or in the worst case already commissioned treatment unit may have to be rebuilt or completely decommissioned.

5 References

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HYDRAULIC CAGE CONCEPT FOR WASTE CHAMBERS AND ITS TECHNICAL IMPLEMENTATION FOR THE UNDERGROUND RICHARD REPOSITORY, LITOMĚŘICE, CZECH REPUBLIC

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ABSTRACT

Richard Repository is a near surface underground repository for low and intermediate level radioactive waste of institutional origin. In the course of a joint Phare Project carried out together with the repository operator, the Czech Radioactive Waste Repository Authority (RAWRA), DBE TECHNOLOGY developed a new concept for the closure of individual waste chambers. Main technological element of this concept is the installation of a hydraulic cage around the waste chambers by attaching a gravel layer to all sides of the chamber. This hydraulic cage will prevent the development of advective flow through the waste/concrete body within the backfilled chambers by eliminating the pressure gradient as driving force for such a flow. Thus the transport of radionuclides will be restricted to diffusive fluxes, which results in a considerable decrease of potential radiological impact. In the course of the project the closure of a certain chamber system within the mine was planned up to a grade of detail, which allowed its direct realization as a pilot closure study, which started in the beginning of 2006.

1 Introduction

In the past, RAWRA as operator of the Richard Repository prepared a preliminary plan for closure of the facility including a safety assessment demonstrating the long-term safety of the disposal system (2002 SA, /1/), which is part of the repository license documentation.

In 2003 RAWRA launched an international call for tenders for the review and further development of the repository closure concept and for the detailed design of the closure of a chamber in the Richard Repository, which was awarded to DBE TECHNOLOGY GmbH. General objective of this project named “Solution for Closure of a Chamber in the Richard Repository” was to demonstrate the feasibility of safe closure of one or more disposal chambers and to improve the safety of the low-level waste packages disposed of in the Richard Repository. This Phare Project, which ended in the second half of 2005, was largely financed by the EC and co-financed by the Czech Ministry of Finance.

2 The Richard Repository

The Richard repository for radioactive waste from institutional waste producers is located in the outskirts of Litoměřice, on the shores of the river Labe, in Northern Bohemia, some 100 km Northeast of Prague. Richard is a former limestone mine, in which during WW2 an underground facility for military production was installed. To this aim a series of caverns were excavated, which at later times remained unused. The layer of limestone containing the tunnels and caverns is horizontal and approximately 4 m thick and the heights of the tunnels and caverns are up to this height. The site is located at about 265 m above sea level, and has a maximum thickness of overlaying strata of about 70 m.

With the start of nuclear research and isotopes use in medicine and industry in Czechoslovakia in the nineteen fifties the need for a facility to dispose of resulting waste arose, eventually leading in the middle of sixties to the installation of a repository in the central part of the former Richard mine (ca. 200 x 400 m). A tunnel through the repository allows access to the caverns. Packages of waste are stacked in the caverns on their sides and the entrances to the caverns were bricked up once the caverns were full of waste packages. The waste is mostly packed in standard 200-liter drums, which comply with radiation protection criteria for contact-handled waste. Up to now, some 25,000 waste packages and thereby a significant activity of about 10^{15} Bq have been disposed of with a significant proportion of long-lived radionuclides, mainly ^{241}Am, ^{239}Pu, and ^{238}Pu.

3 Reasons for the Development of a New Closure Concept

The former closure concept building the basis for the 2002 SA foresaw that the waste chambers containing stacks of waste drums would be backfilled with concrete. For this purpose the filled waste chamber would have been sealed by concrete walls, through which concrete would have been pumped into the chambers with the objective to backfill the chamber to 100%. The concrete for backfilling was supposed to have low hydraulic conductivity, below 10^{-10} m s^{-1}, and low shrinkage. The rest of the repository (main drift, entrance etc.) would have been sealed by concrete plugs while adjacent tunnels and caverns were planned to be backfilled by material with lower requirements.

In the 2002 SA the repository is modelled as a homogenous mixture of waste and concrete. A certain percentage of the precipitation infiltrates the marlstone in the overburden, percolates downwards and through the repository advectively transporting radionuclides out of the former mine. As release scenarios mainly two scenarios are identified. For both, the Town Well Scenario and the Farm Scenario, it is assumed that contaminated water from the mine will travel further downwards through the lower marlstone layer into the aquifer, from which at a certain horizontal distance contaminated water will be pumped up again to serve as drinking water in the first scenario and water supply for the operation of a small farm in the second scenario. As a third potential way of radionuclide release the direct release of contaminated mine water into the biosphere was considered. This possibility was excluded on the basis that high performance sealing of the access tunnels would prevent this to happen. Still as a reference case the annual dose rate was calculated assuming that mine water would be used as sole source for drinking water by an individual without any prior dilution (see Figure 1).

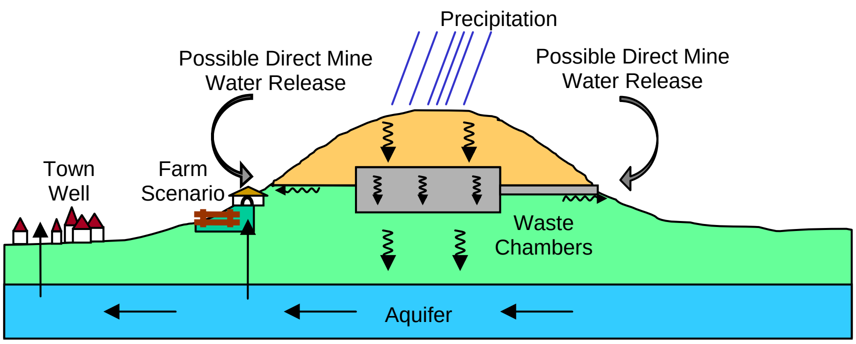

Figure 1: Schematic view of contamination pathways for the different scenarios.

While within the limits of the model used, the 2002 SA rendered exposure values for future generations below current regulatory limits for the Town Well and the Farm Scenario, the reference case yielded high annual dose rates well above regulatory limits. A comprehensive compilation of the closure concept and the site documentation including the results of the 2002 SA can be found in the Safety Report 2003 /2/.

When reviewing the safety assessment, we considered that the direct release of contaminated mine water could not be excluded just by sealing the entrance tunnel in such a way that mine water would not be able to bypass this barrier. Instead, we judged that for several reasons it would be very difficult to totally prevent direct release and even more difficult to prove that such release would be prevented in the long term. The main reason for this evaluation is that even if total long-term sealing of all access tunnels to the Richard Repository could be achieved, which would be difficult to prove, this would still not prevent the possible direct release of mine water because the hydraulic conductivity of the overlying and underlying marlstones is lower than the hydraulic conductivity of the 3-4 m thick limestone layer around the repository. Water inside the repository, therefore, will be released preferably through the limestone than through the underlying marlstone provided no vertical fractures with high permeabilities exist in that marlstone layer, which can easily be reached from all areas of the mine. Also the probability that preferential pathways might exist along the limestone base or fractures inside the limestone with even higher permeabilities than the limestone itself is considered rather high.

The assumption made for the reference case in the 2002 SA that contaminated water from the mine is released without any dilution into the biosphere and is used there by an individual as sole source for drinking water certainly is very unlikely. However, the possibility that mine water is not flowing into the aquifer but is released without much dilution at the slope of the hill into the biosphere does not seem to be very improbable. It was considered necessary, therefore, to develop a changed closure concept, which would prevent the radiological consequences from possible scenarios associated with the direct release of mine water. These considerations lead to the development of a closure concept involving the installation of a hydraulic cage around the waste chambers.

4 Hydraulic Cage Concept

The main idea of Hydraulic Cage Concept is to exclude the build-up of a pressure gradient across the disposal chamber by implementing a high permeable layer around the chamber as preferential pathway for possible groundwater inflow. Thus, the former radionuclide isolation system of the repository, which was based only on the principle of radionuclide containment by enclosing the waste with low-permeability barriers, is complemented by a redundant barrier based on an alternative, totally different working principle: avoiding water flow through the waste by eliminating the flow driving force.

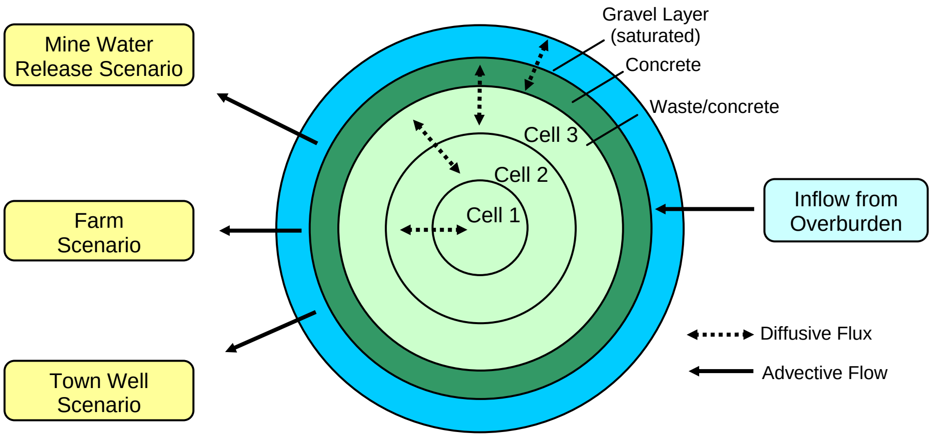

Figure 2: Source term model used for the safety assessment for the Hydraulic Cage Concept.

Due to capillary forces the concrete body might soak up water, but even if the concrete were 100% saturated, without the driving force of a pressure gradient no groundwater flux through the concrete body would result and accordingly no advective transport of radionuclides would take place. This would also be the case if the repository system as a whole was 100% saturated. The normal evolution scenario thus will be changed in such a way that no release of radionuclides will occur apart from diffusive fluxes between the waste/concrete body and the gravel layer (see Figure 2). Although, in the course of time e.g. carbonation might increase the initial diffusivity of the concrete, the difference in permeability between the low permeability zone of the gravel layer and the concrete is expected to remain at several orders of magnitude.

Flux through the concrete body can only occur if continuous fractures throughout the whole body exist. Such fractures might develop as the result of seismic incidents or other accident scenarios. Even in that case, groundwater possibly infiltrating from above in the vicinity of such a fracture will have a low tendency to pass through it, given the negligible hydraulic resistance of the hydraulic cage.

As demonstrated by the updated safety assessment taking into account the changed source term /3/, potential hazards for members of the critical group arising from the direct release of mine water are significantly reduced. For the first 2000 yrs after closure annual dose rates are reduced by more than 4 orders of magnitude. But also for the normal evolution scenarios the changed concept leads to significant improvements in regard to possible radiation exposure. For both scenarios - considered as representative for normal evolution - the peak annual dose rates derived from safety calculations were reduced by a factor 4 and the respective peak time was shifted several thousand years into the future.
In addition to reducing the potential hazards related to the different scenarios, the implementation of the Hydraulic Cage Concept also drastically reduces the probability that either of these scenarios will ever occur at all. Taking into account that, with the Hydraulic Cage System implemented, radionuclide release out of the waste/concrete body will take place only via diffusive transport, there will be hardly any release at all as long as no permanent flooding of the mine results from that water inflow. A rather likely situation, which for the former closure concept would still lead to the release of radionuclides.

5 Technical Implementation for individual Chambers

After reviewing the different technical implementation alternatives, a rather simple to apply and robust solution for the construction of the hydraulic cage, analogous to techniques used in tunnel building, was developed. In principle the stacked waste is backfilled with low-permeable concrete as foreseen in the preliminary closure plan. In addition, a layer of pure concrete, which again is enclosed in a gravel layer with high hydraulic conductivity, surrounds the waste/concrete body. A drainage and monitoring system allows monitoring the performance of the hydraulic cage before repository closure, thus allowing verifying the proper functioning for a period of up to several decades.

As the chamber-system selected for the pilot closure project was part of the non-operated area of Richard II, as first steps, the planning foresees clearance of the chambers from debris and loose rocks and securing roofs and walls with rock bolts. Subsequently the floor and the cage for the surrounding gravel layer are being prepared to reach the situation illustrated in Figure 3. As to be seen from this figure the floor consists of three layers. The underlying concrete layer together with the small concrete side walls will isolate the drainage layer in the floor from the surrounding rocks. This will allow infiltrating mine water to be conducted to the entrance of the mine for monitoring purposes. Another reason for shielding the lower marlstone, exposed at the bottom of the chamber and at the lower part of the walls, from the potentially saturated gravel layer is the fact that this marlstone has a certain swelling potential, which might have a negative influence on the permeability of the gravel layer. The gravel layer, at the chamber floor represents the lower part of the hydraulically conducting layer surrounding the waste chamber. During operational times, water entering this layer will be drained through normal water pipes leading to the tunnel entrance. Gravel filled gaps between the concrete roadway and the small concrete side walls provide the connection between the bottom gravel layer and the gravel layers to be installed at the walls. According to sensitivity analyses the thickness of the surrounding layer of pure concrete, of which the concrete roadway on top of the gravel layer represents the lower part, was set to 40 cm.

The highly permeable layer at the walls and the ceiling will be realized by attaching a steel mesh (10 cm x 10 cm x 6.3 mm) to the extruding rock bolts (about 1 rock bolt per m²). The steel mesh is to be properly fixed to the rock bolt extensions using appropriate extensions. At the wall side of the steel mesh, a geotextile mat (alternatively steel gauze could be used) has to be attached to keep the gravel inside the interstice between steel mesh and wall or roof. To complete the highly permeable layer, drainage material will be blown into the interstice between the chamber contours and the layer of steel mesh and geotextile. To prevent fresh concrete or cementitious suspension from flowing into the gravel layer during later backfilling of the chamber a watertight layer has to be attached to the steel mesh from inside the chamber. This function will be fulfilled by of a shotcrete layer of 5-10 cm thickness, which also will increase the working safety.

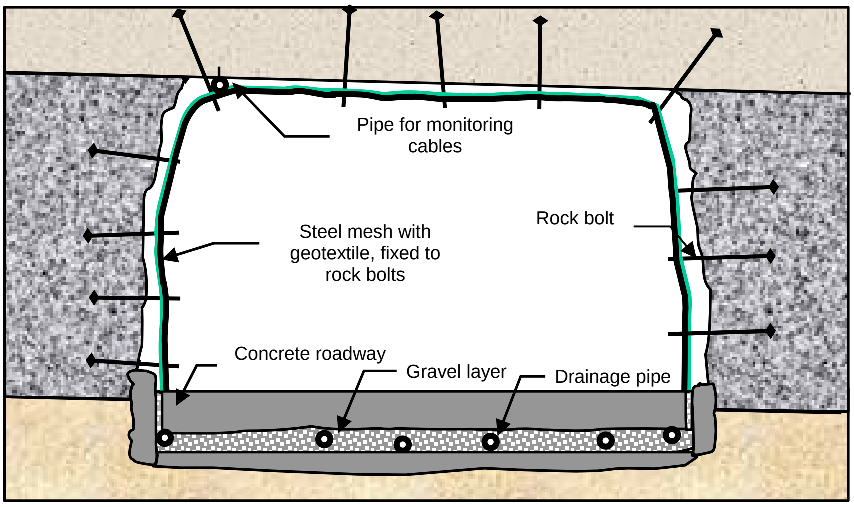

Figure 3: Cross section of disposal chamber after installation of cage for gravel layer. The different colours of the surrounding rock layers refer to: Upper Marlstone (top), Limestone (middle) and Lower Marlstone (bottom).

The construction of a 40 cm layer of pure concrete at the walls and the ceiling of the chamber can be carried out automatically during the later backfilling of the chamber if care is being taken during waste disposal that waste packages are kept in a respective distance to the walls. This requires, however, the stabilization of waste packages to prevent them from rolling or shifting towards the walls. Depending on the form and weight of the waste packages this might not be trivial. It was therefore planned to construct the 40 cm side walls prior to waste disposal, which allows an unproblematic stacking of drums or other waste packages. After this construction step the chamber is prepared for waste disposal. Completion of the enclosing layer of pure concrete around the waste/concrete body will be achieved by filling the top most part of the chamber during the later backfilling process

For the sake of simplification, in the description above the fact has been ignored that in order to comply with certain quality requirements the chamber system has to be subdivided into several segments with maximum lengths of about 20 m, which will be backfilled individually. The detail planning therefore foresees the construction of partition walls, which not only serve as separating walls for the backfilling process but also as supporting structure for the chamber-system prior to its backfilling.

Meanwhile the realization of preparation, waste disposal, and subsequent backfilling of the chamber-system has been started in the course of a further Phare Project as a pilot study for the feasibility of the technical implementation. There is a second paper to be presented at this conference, describing the experiences from the construction work and showing examples of the “real world” version of the hydraulic cage /4/.

6 References

/1/ Chambers, A.V., R. Cummings and B.T. Swift: Performance of the Richard Repository, Serco Assurance, Harwell, 2003.

/2/ RAWRA: Safety Report of the Radioactive Waste Repository Richard. Final Report. RAWRA, Prague, September 2003.

/3/ Haverkamp, B. and E. Biurrun: Safety Assessment and justification of the proposed solution for closure. DBE TECHNOLOGY Report DBE-RCH-TSK-07, July 2005

/4/ Kucerka, M.: Technical Realization of a Closure Concept for a Chamber-system in the Underground Richard Repository in the Czech Republic. Transactions TopSeal 2006, Olkiluoto, 2006.

Technical Realization of a Closure Concept for a Chamber-system in the Underground Richard Repository in the Czech Republic

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Introduction:

The Phare project CZ 632.02.04 "Realization of closure of a chamber in the Richard repository as input for establishing a safety case" is a follow up implementation phase of the Phare project, CZ 01.14.03 "Solution for closure of a chamber in the Richard repository".

Main objective of both projects is to propose and realize a disposal system in selected chambers of the Richard repository, which will eliminate burden from the past practices in waste management during the first phase of the Richard repository operation (1965 – 1980) and which will improve its overall long term safety.

This objective will be assured by realization of the concept of so called "hydraulic cage", which technical solution was developed by DBE TECHOLOGY within the Phare project CZ 01.14.03. The solution is described in the previous presentation "Hydraulic Cage Concept for Waste Chambers and its Technical Implementation for the Underground Richard Repository, Litoměřice, Czech Republic" (Bernt Haverkamp et all).

Realization of the hydraulic cage closure system is divided in 5 basic phases:

- Detailed realization design and technologic procedures development
- Preparatory work and clean up of the chambers
- Construction of the hydraulic cage and concrete structures for emplacement waste
- Removal, inspection, conditioning and relocation of "historical waste" into new chamber segments
- Backfilling of voids between waste packages – closure of the segments.

After the project completion RAWRA will launch a program on evaluation of a long term behavior of the backfill material and sealed waste packages, as an input for verification and validation of data necessary for the repository safety assessments.

Basic project information:

Project CZ 632.02.04:

Phare contractor:	EREBOS – podpovrchová výstavba spol. s r.o., M. Svatoňovice
Subcontractors:	TUBES spol. s r.o., Praha, (design)
	AGE, a.s. Praha (technical supervision)
	DBE TECHNOLOGY, GmbH, Germany (consultancy)
RAWRA contractor:	ALLDECO.CZ a.s. (waste treatment and relocation)

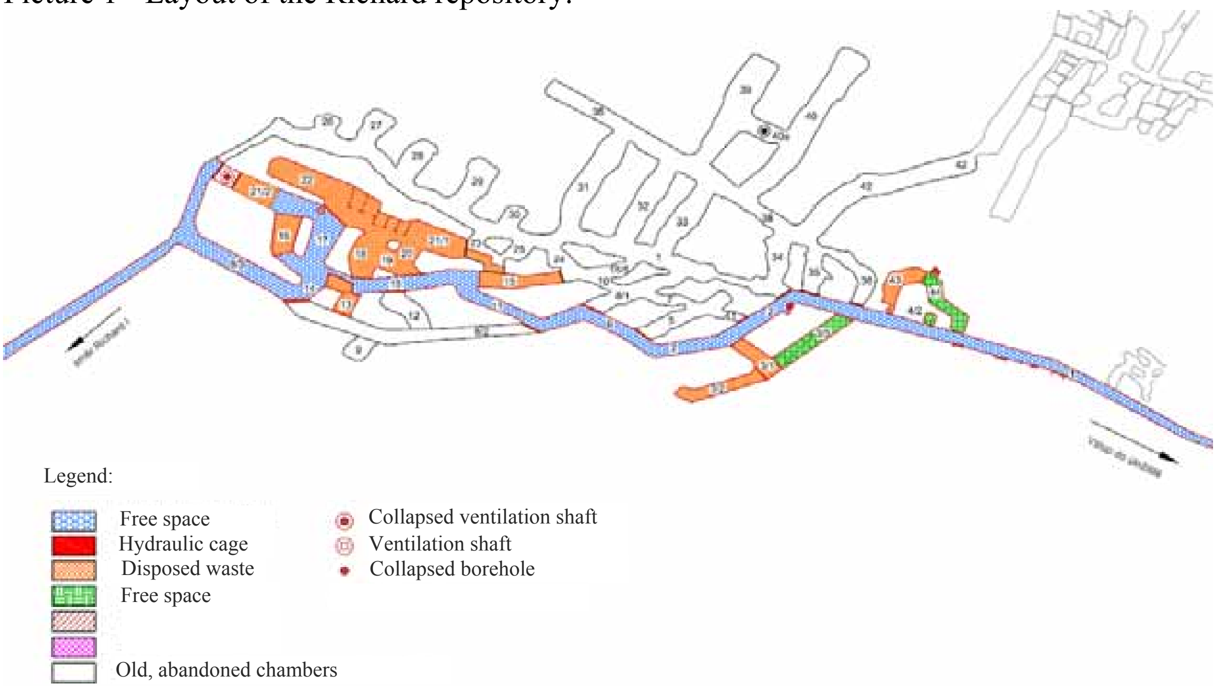
Time schedule:	Planed	Real
TOR approval	3Q. 2003	3Q. 2005
Contractor selection	1Q. 2004	July – Sept. 2005
Project start up	1Q. 2004	5. Dec. 2005
Project completion	3Q. 2005	June 2007
Disbursement period	Nov. 2005	Nov. 2007

Budget:

Phare contract	- construction and backfilling	1 000 000 EUR
RAWRA co-financing	- relocation, conditioning waste	264 000 EUR
	- reconstruction of Richard cabling	151 000 EUR
	- reconstruction of drainage system	38 000 EUR

The project includes realization of reconstruction and closure of the chambers 8/2, 9 and 12. Historical waste will be removed from the chambers 22, 18 and 19. Debris from clean up of the chamber 8/2, 9 and 12 were removed into the chambers 4, 5, 7, 8/1 and 10. Situation is shown in the Picture 1 below.

Picture 1 - Layout of the Richard repository:



Realization activities

The contract on the Richard chamber closure realization, with the selected company EREBOS – podpovrchová výstavba, was signed on 9.11.2005. The contract does not include activities connected with handling, conditioning and disposal of the waste packages, before the chambers backfilling. These are provided by specialized company ALLDECO.CZ selected by RAWRA and contracted in May 2006 from the co-financing resources.

1st Phase

The project kick of meeting was held on 28 November 2005 and the project activities started on 5th of December.

First activities were focused namely on development of a detailed realization design of chambers reconstruction and preparation of all necessary documentation for licensing of planned actions. What concerns the project there are two main licensing authorities – the Czech Mining Office and the State Office for Nuclear Safety.

In parallel, EREBOS prepared its infrastructure for the project realization and training of the EREBOS personnel on emergency preparedness, radiation protection and other relevant internal instructions of RAWRA was held.

Special attention was focused also on preparation of the QA and QC plan, to achieve and ensure planed and/or required parameters of controlled materials and processes.

During the design development some changes of solution were proposed and consequently discussed with DBE Technology. The main changes are described below in the table 1.

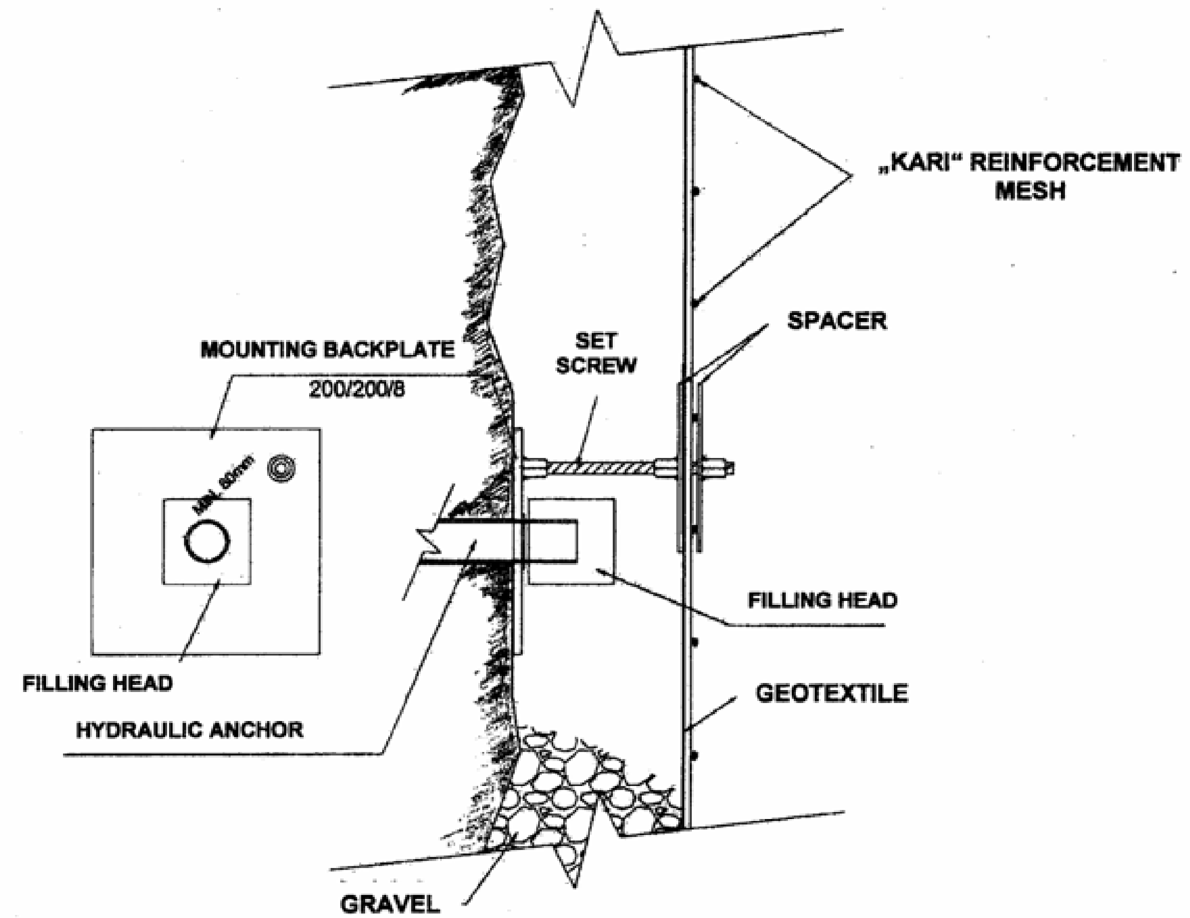
For construction of the hydraulic cage a unique system of the cage mounting on the chamber walls was developed. This system is very flexible and allows the cage construction without any limitation concerning the chamber walls shape – see Picture 2

To avoid random, uncontrolled formation of shrinking cracks during the concrete setting, a system of controlled shrinking joints is used. It includes inserting of a wooden conical lath into the floor or wall before its concreting in distance of some 6 - 8 m. After setting of the concrete the lath is removed and the gap is filled with a special expanding concrete. In the line of this dilatation joint will be created a gap between the waste packages minimally 10 cm wide, so the joint will be overlaid by pure concrete layer. The joint construction scheme is shown in the Picture 3.

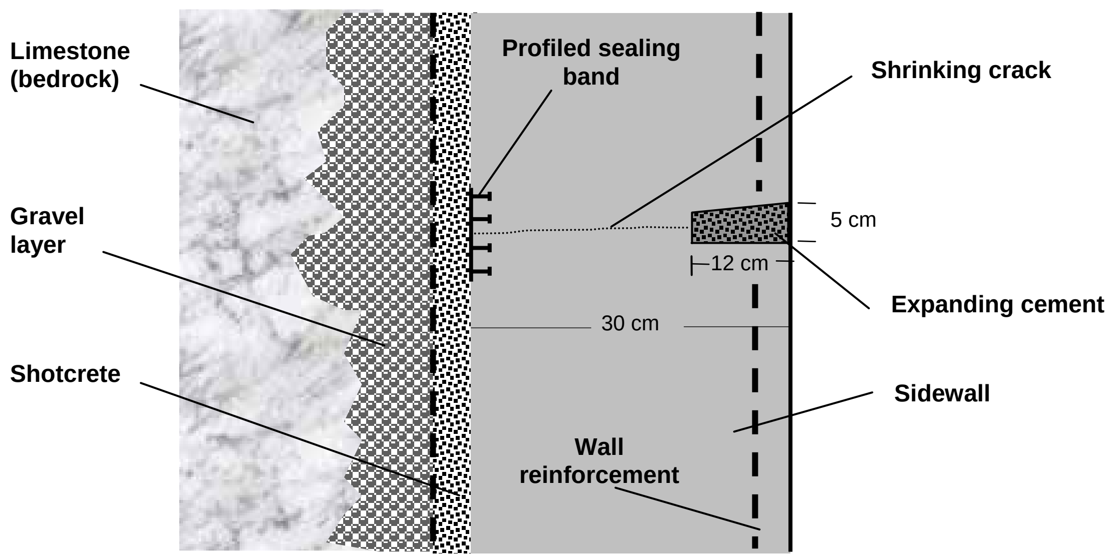
Table 1: Changes in the realization design in comparison with proposed technical solution.

	Technical solution according to 2005 closure plan (DBE Technology)	**Changed technical implementation according agreement of RAWRA, EREBOS and TUBES**
1	Fixing of the wire gauze to steel KARI – mesh to separate gravel layer from shotcrete	Wire gauze replaced by geotextile
2	Construction of concrete pillars for controlled drum stacking; Backfilling of 40 cm space between walls and RAW with backfilling concrete during closure of chamber segment	Construction of concrete side walls prior to drum stacking as advanced part of the later backfilling with shrinking joints (6-8 m apart) 50 mm cover to reinforcement of the walls (KARI-mesh 6,3/100/6,3/100)
3	Application of 10 cm layer of shotcrete (SB C 20/25) to separate gravel layer with wire gauze/geotextile from inner chamber.	Application of 5 cm layer of shotcrete (SB C 20/25 X0)
4	Roadway construction as 40 cm reinforced concrete layer (40 cm, C30/37)	Realization without reinforcement but in two subsequent concreting processes within 1-2 days.
5	Chronological separation between disposal of compacted 50-l drums waste and disposal of 200-l drums	Concurrent disposal of different kind of waste
6	Backfilling of voids between compacted 50-l drums within containers in advance to disposal or alternatively in compartments within chamber segment Backfilling of remaining voids during closure of chamber segment	Backfilling of voids between drums: (mainly 50-l drums) stepwise, backfilling of remaining voids between different waste packages and free space (40 cm) towards ceiling and upper walls for complete chamber segment after disposal during closure of chamber segment

Picture 2: Hydraulic cage mounting on the chamber wall



Picture 3: Controlled concrete wall shrinking joint



2nd Phase

The full-scale activities concerning the chambers reconstruction started on 2 January 2006. Firstly, the chambers 4, 5, 7, 8/1 and 10 were prepared and stabilized for accepting debris from the reconstructed chambers. During the clean up, more than 600 m^3 of debris rock was removed from the chambers 8/2, 9 and 12 and disposed in the above mentioned chambers. This phase was completed in first decade of March. Anchors used for fixing ceilings were the same as for mounting the hydraulic Cage.

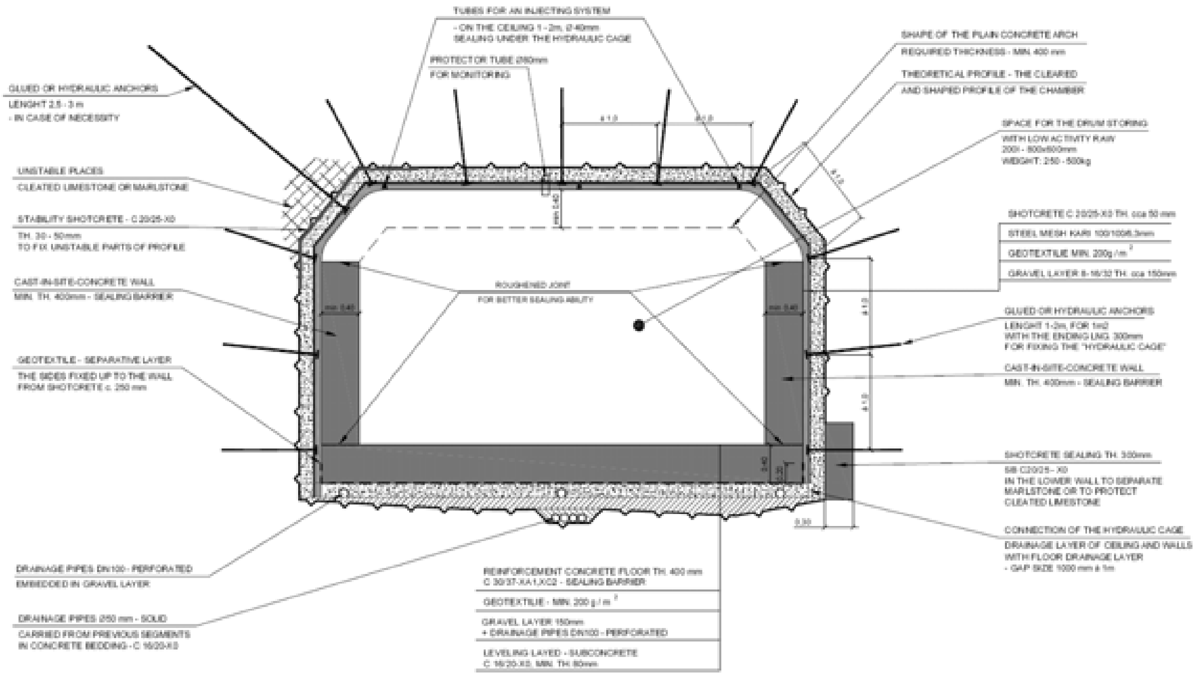
Picture 4: Chamber 8/2 before clean-up

Picture 5: Chamber 8/2 after clean-up

3rd Phase

This Phase includes construction of hydraulic cage, supporting frames, floor with drainage system and concrete walls as shown in the Picture 6. It started in second decade of March, hydraulic cage on walls and ceiling was completed in June, in July were completed and cleaned up 3 segments for accepting waste.

Picture 6: Cross-section of the disposal chamber



Quality control of used concrete during the construction work confirmed its compliance with the requirements from the technical specifications prepared by DBE TECHNOLOGY.

Table 2: Parameters of concrete used for the walls and floor construction

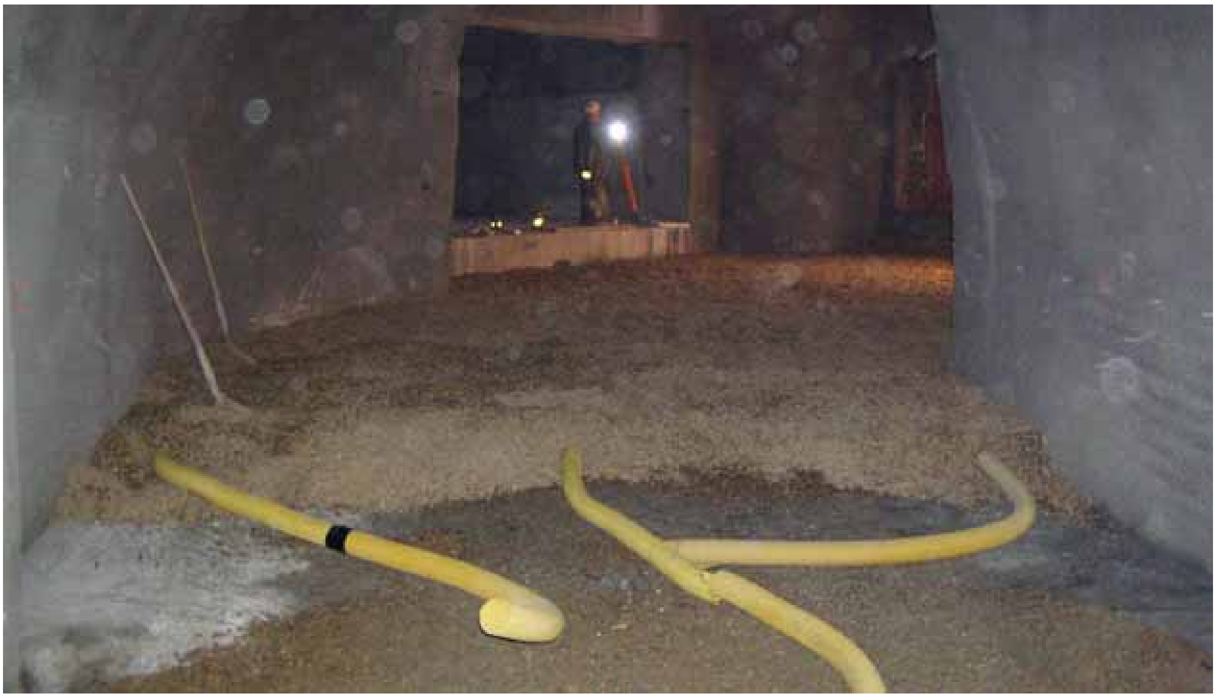
Parameter	Required	Achieved
Compressive strength	30 MPa	35 – 50 MPa
Content of cement	300 Kg/ m^3	400 Kg/ m^3
Water penetration depth	75 mm	16 mm
Temperature during concrete setting	Max. 60°C	Max. 32°C

Up today remains realize construction of floor and walls in the last chamber segment; this will be completed within next two weeks.

Following pictures illustrate the work realized:

Picture 7, 7a: Details of the hydraulic cage

Picture 8: Construction of bottom drainage system



Picture 9: Formwork for the wall concreting

Picture 10: Completed part of the chamber 8/2

4^{th} Phase

For inspection and compaction of waste packages ALLDECO.CZ erected a separate working place. It is situated in the chamber 17 next to the chamber 22, from which the historical waste will be removed. In this place are installed a hydraulic compactor for compacting the 50 liter barrels, electronic weight and radiation control equipment. The compactor has filtrated exhaust (with high efficiency HEPA filter) connected to the main ventilation system. Each waste package is measured on gamma exposure rate and neutron flux and each barrel content is visually controlled by opening a lid. Depending on results of waste content checking, the shift foreman takes a decision on compaction or other treatment of the waste package. Up to now approximately 300 of historical waste packages were inspected and treated. Some of the packages are in a very bad condition – heavily corroded. These barrels are inserted into plastic bags and consequently into new 100 l barrels. Void space will be backfilled by concrete "in situ".

Following pictures illustrate the historical waste handling.

Inspection of the waste packages content

Compacted 50 liter drum

Waste drums in the chamber 22

Corroded drums

Compacted waste in the transporting containers

5th Phase

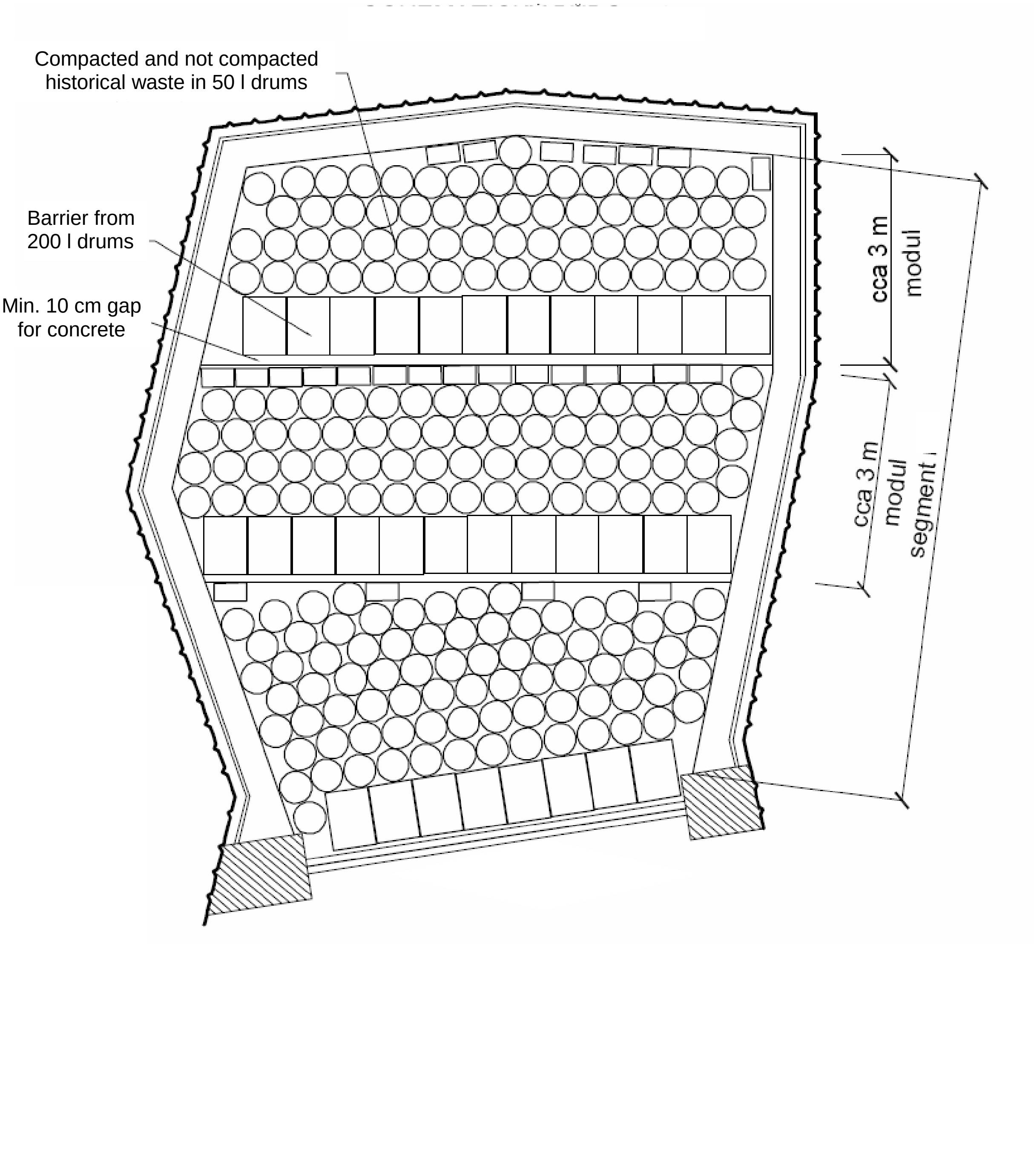
After evaluation of various proposals and concepts of emplacement and backfilling of the waste packages was agreed the following concept. This concept we consider as an optimized solution from radiation protection, as well as from the backfilling technology point of view.

Backfilling will be held in modules that will be wide approximately 3 meters. From 200 l waste drums will be stepwise created a barrier supporting the pile of compacted and not compacted 50 l waste packages, up to the height of the concrete wall. Then will be erected a formwork and the waste pile will be backfilled with concrete. On the top of the backfilled waste will be inserted waste packages in that manner that a void space between the hydraulic cage and waste packages will be minimally 30 cm, to ensure sufficient isolating concrete layer. After stepwise backfilling of the modules, whole segment with remaining not backfilled upper layer of waste packages will be backfilled together at once, so a compact concrete “head” will be created.

Backfilling of the waste packages does not started yet, but we expect that it will start in the end of September, after receiving the license from the Mining Office. During that period a system of monitoring of backfilling process will be realized. EREBOS will run tests of concreting technology (mixing, pumping) using variable concrete mixtures, to ensure optimized parameters of the concrete.

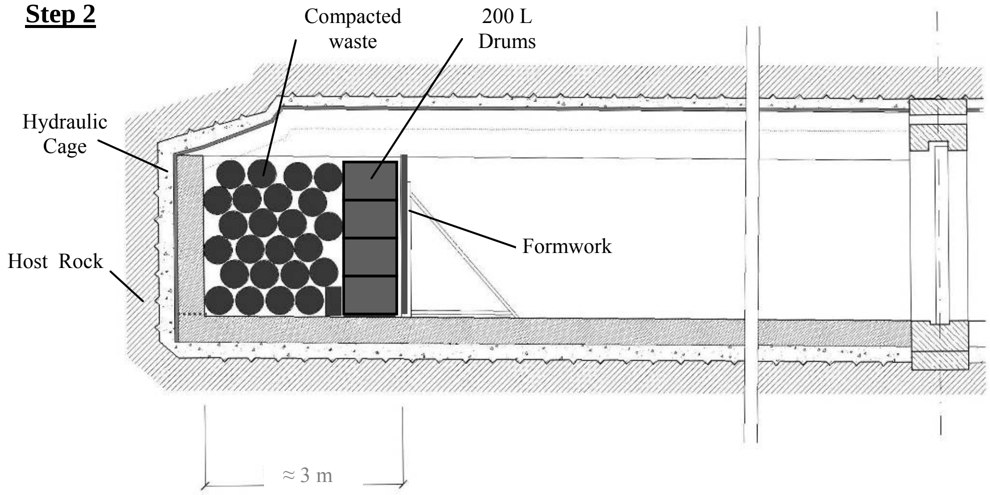
The proposed system of waste emplacement and backfilling is visible from the following pictures.

Layout of the chamber 9 filled with the waste packages

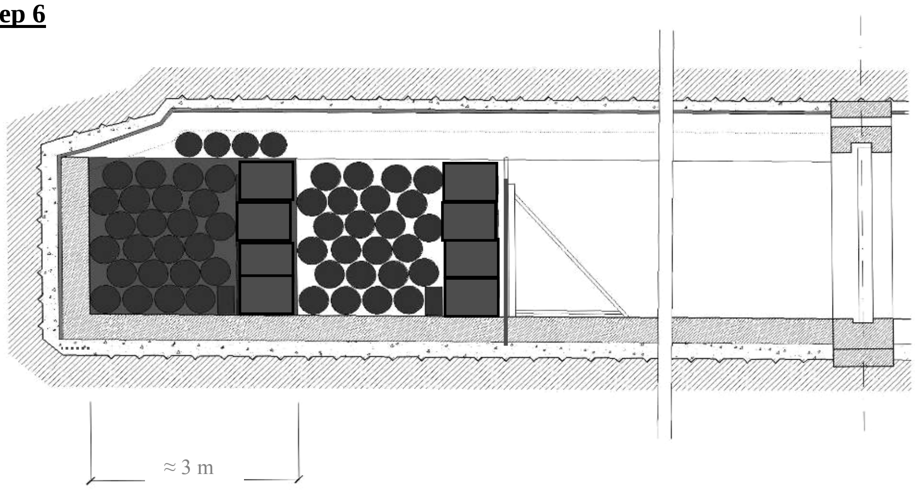


Sequencing of the waste emplacement and backfilling

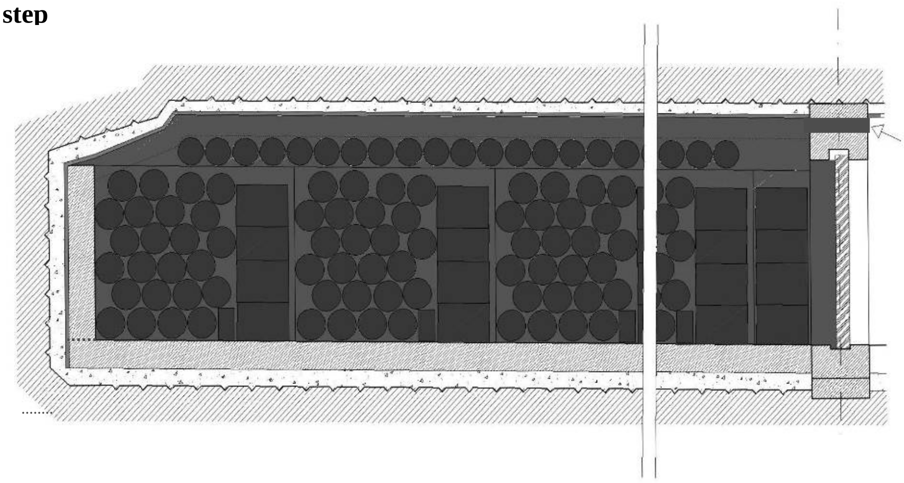
Step 2



Step 6



Final step



NEW DEVELOPMENTS IN LOW LEVEL RADIOACTIVE WASTE MANAGEMENT IN SPAIN

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ABSTRACT
El Cabril disposal facility was commissioned in 1992 and is a key element in LILW Management in Spain. It is a vault-type surface disposal facility with a total internal volume of 100,000 m^3. The installation also has facilities for waste treatment and conditioning, verification and characterisation, interim storage and other ancillary equipment. This paper includes a brief description of the facility, the operational experience, the design improvements and new developments in waste acceptance procedures, safety assessment and the related research programme. The paper also refers to the new disposal facility intended for very low activity waste, under construction at the same site. This facility, a part of El Cabril nuclear installation, will have a maximum capacity for 130,000 m^3 of very low activity waste. Its construction started in February 2006, after the evaluation of the nuclear safety authority and the environmental impact statement procedure.

1. Introduction

The Ministry of Industry, Tourism and Commerce (MITC) is responsible for the radioactive waste management policies; the Nuclear Safety Council (CSN), an independent body reporting to the Parliament, is responsible for nuclear safety and radiological protection regulation and enforcement; and the Ministry of the Environment (ME) is responsible for the Environmental Impact Statements. The cabinet has recently approved, in June 2006, the sixth General Radioactive Waste Plan (GRWP) [1], a document where the national policy in this field is reported.

ENRESA is responsible for the long term management of all categories of radioactive waste and for nuclear installations decommissioning operations as well. ENRESA owns and operates El Cabril Low and Intermediate Level Radioactive Waste (LILW) disposal facility, which was commissioned in October 1992. Since then, it has been a key element in waste management in Spain.

The total internal volume of the existing disposal vaults is 100,000 m^3, corresponding to 35,000-50,000 m^3 of primary waste packages as delivered by the producers, depending on the waste types. The total expected volume of LILW arising from the Spanish nuclear programme is estimated in 176,300 m^3 [1], as shown in figure 1.

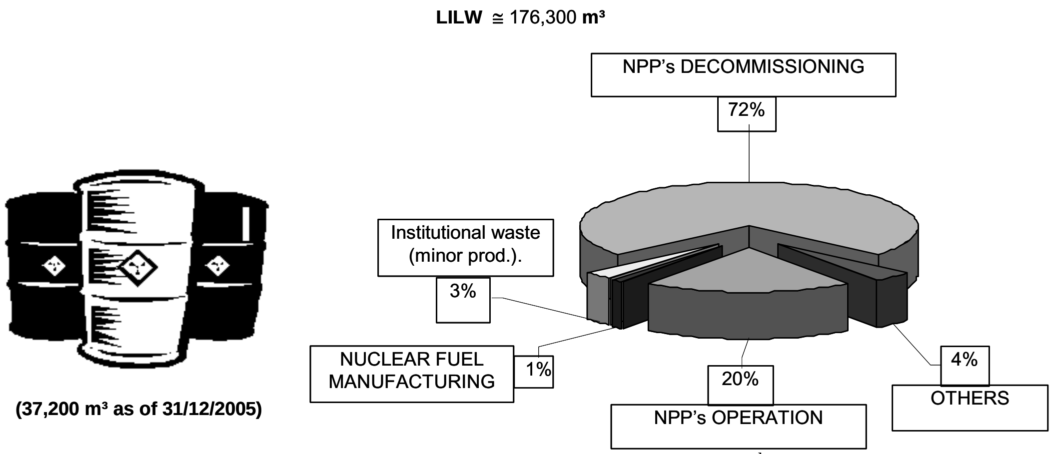

Fig. 1- Expected LILW volumes according to the 6th GRWP

An important part of this grand total would be very low activity waste, mainly arising from decommissioning operations. 53% of the existing capacity at El Cabril has already been filled-up by summer 2006, while still a good margin exists in terms of the maximum activity authorised. Together with operational safety and reliability, the main axis of development in LILW management are: A good administration of the remaining vaults' volume, through volume reduction in co-operation with the waste producers, and through optimisation of the occupancy rate supported by the revision of the waste acceptance criteria (WAC); The construction of a new disposal facility for very low level waste (VLLW); and the improvement of confidence building, through updating and enhancing the safety assessment, the surveillance of the facility and the research to improve the knowledge of the site characteristics, the waste forms and the isolation barriers behaviour.

2. General Description of the facility

El Cabril is located in Southern Spain. The site was an exhausted Uranium mining reserve and was used for radioactive waste storage since 1961. The main objective of the facility is the final disposal of all LILW from the country. Other additional aims of the facility include waste treatment and conditioning; waste characterisation and verification, interim storage, and auxiliary installations to support operation and maintenance.

Fig. 2 - El Cabril LILW Disposal Facility General View

The disposal concept is a multibarrier surface disposal system. Waste packages (mainly 0.22 m^3 drums and 1,3 m^3 metal boxes) delivered by the producers, respecting the WAC, usually are reconditioned in concrete containers to form an 11 m^3 final package or disposal unit, which constitutes the first barrier. The internal volume of the concrete container may be back-filled with mortar grout, or may be used to condition institutional liquid waste or contaminated ashes. These packages are place inside 24x20x10 m concrete vaults. Once the vault is completed with 320 11-m^3 concrete containers, it is backfilled with gravel and a closing slab is constructed and coated with an impervious painting. After completion of, at least, one of the two areas in which the disposal zone is divided, an engineered multi-layer cap will be built. Beneath each row of disposal vaults there is an inspection drift, where two drainage systems are installed, one for rain water collection from the vaults not yet in operation, one for the vaults containing waste packages. Each row of vaults is served by a metallic shelter on wheels, for weathering protection and supporting the 32 tonnes overhead lifting crane.

The installation is laid out in two main zones: The disposal zone, containing the 28 existing vaults, and the rain water collection pond and a factory to prefabricate the concrete containers as well; And the auxiliary buildings zone, including security, administration, workshops, storehouses, and the treatment and conditioning building (where the main systems are incineration, super-compaction, grout preparation and injection, and ashes treatment), and characterisation laboratories (with waste core extraction and specimen preparation cell, gamma spectrometry scanning room, leaching test room, and sample preparation and measurements radiochemical laboratories).

3. Operational experience

The number of primary packages actually disposed of at El Cabril by the end of June 2006 is 98,548. Most of them (95,800) are 220-litre drums (63% are solidified waste in cementitious matrix; 37% are compactable waste drums). There are 1920 480-litre drums and 828 1.3-m^3 metal boxes as well. In addition we have some 7000 packages at El Cabril in interim storage, most of them pre-classified as VLLW, waiting for the commissioning of the new VLLW disposal vaults under construction at the site.

No major operational incidents have occurred, although some minor ones can be reported as the drop of a drum because of the rupture of the lifting cable with local dissemination of material inside a building or the drop of a concrete container at the mortar injection post due to a wrong positioning of a trolley, which required special means for handling.

An interesting finding was the appearance of some liquid in the water collection system of the vaults. After a sound investigation of potential in-flow from the upper slab and construction joints, which included water tightness tests of flooding such roof and joints, capillary rise from the water table, together with evaporation and condensation due to thermal differences was analysed as the origin of such water collection. The air gap among containers and vaults walls produced a seasonal difference of temperature of a few degrees, which provokes water vapour diffusion from the walls to the concrete containers in summertime or from the concrete containers to the walls in winter. There are changes in the liquid saturation in the concrete and finally some condensation at the cold surface, with a maximum volume of a few litres per day in the vault with more water collected.

A surveillance programme including more than 1000 samples in 118 sampling points has shown no abnormal values and average collective dose since 1992 has been 12.2 mSvxperson with no internal contamination recorded in the annual internal dosimetry checkings.

4. Major design changes

Main design changes since the facility start-up were:

In relation to disposal some changes were:

- The slopes of the closing slab (from four directions to one direction to the external side) and some adaptations to the changes in structural concrete and cement regulations;
- The possibility to use the gap among drums, inside the concrete container to condition some type of cemented solid wastes (smelting ashes).
- The possibility to accept in some conditions 480-littre drums where 220-littre drums had been reconditioned

In relation to treatment and conditioning equipment it may be stressed:

- New grout injection system, because it was identified as the bottle neck in the facility operation.
- Grinding, Leaching and Electrolysis equipment for pre-treatment of contaminated ashes and cemented hazardous waste, to fulfil the requirements to be part of the injection grout.
- Decontamination room for treating some waste.
- Tests for incineration of NPP'S waste
- Improvements in the compactor system drain collection, due to presence of liquid in amounts larger than expected.

In relation to waste characterisation and verification, an extension of the radiochemical laboratory has been built and new gamma scanning equipment has been set-up in order to increase the number of verification tests to ensure the limitation of activity determination uncertainties. In relation to interim storage a new building was needed as well as the adaptation of three final disposal vaults to be used s interim storage while waiting for treatment previously to disposal.

5. Very Low Activity Radioactive Waste disposal

Significant amounts of radioactive waste with very low activity (VLLW) have already been produced, especially in some incidents in the steel industry, and larger amounts are foreseen from nuclear power plants decommissioning. The total volume to be managed is estimated in 80,000-120,000 m^3 depending on the clearance and enhanced decontamination policies. A Parliamentary Committee recommended to the Government the development of a facility specifically intended for this type of waste not to misuse the existing disposal capacity at El Cabril, considered strategic and designed to dispose of waste with higher specific activity. The application for this facility was presented in May 2003. MITC granted the execution licence for the disposal cells, after binding report from the CSN and environmental Impact Statement by ME on February 2006. Civil works are now under way; the project schedule foresees a construction delay of 18 months

Although the disposal principles for VLLW remain the same that for the rest of LILW, a different barriers' lay-out has been adopted (figure 3). The Spanish regulation applicable to hazardous waste disposal [4], based on the corresponding European Directive [5], was taken as a major reference for the design criteria. Each cell presents a one-meter artificial geological barrier of compacted clay, complemented with geo-bentonite to provide a five-meter equivalent clay barrier. The isolation barrier has also a High Density Polyethylene (HDPE) film above which there is a leachate collection system.

BARRIERS

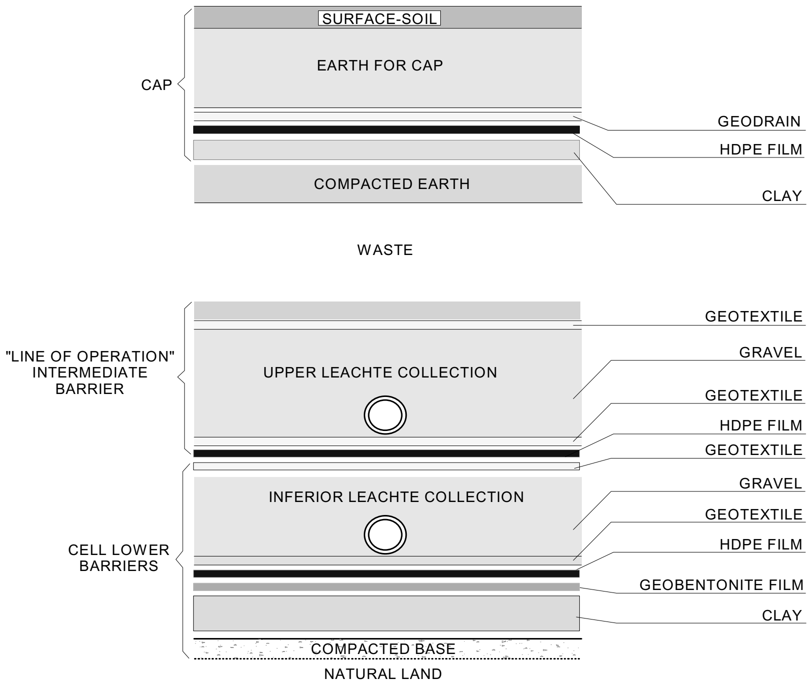

Fig. 3 - VLLW disposal cell: barriers scheme

The VLLW disposal zone is organized in four large disposal cells with 35,000- 45,000 m^3 capacity each. Each cell is divided into so-called "lines of operation", which are protected by a light roof structure (to minimise the volume of leachate to treat) (Figure 4). Each single "line of operation" has an additional HDPE film and leachate collection to help identifying where the collected liquids come from.

DISPOSAL PROCESS

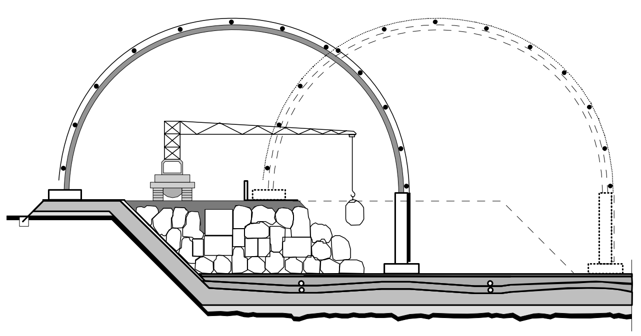

Fig. 4 - VLLW disposal cell: operation scheme

6. Current safety assessment and associated research

The developments on assessment on post-closure safety have been mainly centred on: The adaptation and the updating of the assessment methodology to the current international practices; The review and the enhancement of scenario development approach; The review of the conceptual model, the mathematical formulation and data use to represent each scenario; The extension and the improvement of uncertainties and sensitivity analyses; and providing a coherent link between waste acceptance criteria, engineered barriers design requirements and criteria, and site additional data with the assumptions and the parameters used in the safety assessment. Taking into account the most recent approaches to safety assessment, the efforts have been centred on reorganization of the information included on the safety report.

The performance of the disposal system under both present and future anticipated conditions, including events associated with the normal evolution of the facility and less probable events, was review. The approach adopted relay on a systematic identification and consideration of Features, Events and Processes (FEPs). The set of scenarios developed, although are basically the same as those defined previously, enhance their justification and a defensible presentation of the system.

A critical review has been carried out in order to assure that the models representing the scenarios are adequate and appropriate. The computer codes used are adapted to the specific characteristic of the system. A special care have been put on providing coherent link between design requirement and criteria regarding waste packages/disposal units, engineered barriers and site. The improvements in each of the aspect mentioned above have contributed to identify, document and quantify (when it is possible) the system uncertainties.

An important effort is carried out to link in a comprehensive way the design requirements and the design criteria of the component of the disposal system with the assumptions and the parameters used to describe the component behaviour. The effort has been focused on the following aspects: activity confinement, permeability and durability.

ENRESA research programme includes a limited number of research projects in the LILW field. The main objectives in this area are:

- Development of new treatment systems, focussed on volume reduction, and including a plasma torch furnace for LILW treatment.
- Development of characterisation techniques for different matrices
- Activity measurement methodology for radionuclides difficult to be measured.
- Graphite waste management.
- Clearance of materials with an extremely low activity.
- Behaviour of concrete barrier materials under disposal conditions
- Construction of a pilot earthen cap.
- Instrumentation of pilot containers, disposal vaults, etc
- Model development to integrate migration model in hydro-geo-chemical models

7. Conclusions

An overall LILW management system exists in Spain, which allows the waste generators to get rid of their waste in a safe an efficient manner. Nonetheless, to have a system running in an appropriate way do not permit any relaxation and we have to be prepared to answer the growing social and regulatory requirements. A careful administration of the existing disposal capacity and the new disposal facility for very low activity radioactive waste under construction are crucial elements for being able to manage the expected volumes of waste to be generated.

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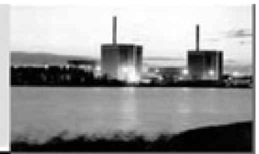
[3] Real Decreto 1481/2001, de 27 de diciembre, por el que se regula la eliminación de residuos mediante depósito en vertedero (BOE de 29 de enero de 2002

[4] Directiva del Consejo de la Unión Europea relativa al vertido de residuos. (1999/31 CE, de 26 de abril de 1999).



SESSION IV : Planned Activities for Geological and Near-Surface Repositories for all Waste Types

LAST DEVELOPMENTS IN THE BELGIAN DISPOSAL PROGRAMME FOR LOW AND INTERMEDIATE SHORT-LIVED WASTE

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ABSTRACT

After an historical reminder of the several phases of the Belgian program for the disposal of low and medium level short-lived waste since the creation of ONDRAF/NIRAS and the bad results obtained in the 90's by using a pure technical approach, the presentation will explain the main lines of the new methodology developed, as a consequence of the government decision of 16 January 1998 in ONDRAF/NIRAS to improve local acceptance for the disposal project.

The way local partnerships were created with four nuclear municipalities under the form of a non-profit organization with a clear mission, the functioning, on a voluntary base, of the different partnerships during four to six years and the concrete results obtained until now using this very innovative method will be addressed.

The last developments of the Belgian program for the disposal of low and medium level and short-lived waste will be presented, including the recent and very important decision of the Belgian government of 23 June 2006 to dispose of the low and medium active short-lived waste in a surface disposal installation on the territory of the municipality Dessel.

1. Introduction

ONDRAF/NIRAS, the national Agency for radioactive waste and enriched fissile materials, estimates the total volume of waste that will be produced until 2060, i.e. the end of the dismantling activities, at 70 500 m³ of short-lived low and medium active (category A) waste.

The council of ministers of 23 June 2006 took an important decision regarding the long term management of category A waste. He decided to dispose of the low and medium active short-lived waste in a surface disposal installation on the territory of the municipality Dessel. This decision makes it possible to develop the next phase of the program so that a disposal installation may be realized concretely.

2. Disposal of category A waste: the failure of the pure technical approach

ONDRAF/NIRAS started working on the long-term management of short-lived low and medium level waste shortly after its creation in 1980. Practiced on a regular basis in Belgium until the early eighties, sea disposal of conditioned low and medium level waste had indeed become very uncertain in 1984, when Belgium decided to adhere to the international moratorium of 1983 between the signatory countries of the London Convention on sea pollution. This decision prompted ONDRAF/NIRAS to launch studies to look for another solution, which would be safe and technically acceptable, for the final disposal of this type of waste on Belgian territory.

In 1994, ONDRAF/NIRAS published the NIROND 94–04 report. This report concluded the feasibility of disposing of at least 60% of the short-lived low and medium level waste produced in Belgium at surface level, while strictly following the recommendations of the various relevant international organizations. It also identified 98 zones on Belgian territory as potentially suitable, according to the bibliographical survey carried out, for hosting a surface repository for short-lived low and medium level waste. The multidisciplinary scientific advisory committee set up by ONDRAF/NIRAS' Board of Directors to examine the report issued a globally positive evaluation, but recommended extending the research to fields related to economics and human sciences.

Far from going unnoticed, the 1994 report was rejected unanimously by all the local councils on the list. To its surprise, ONDRAF/NIRAS had caused a general outcry. And yet, had it not been given the responsibility to develop and propose, through an objective and rational approach, a safe solution to the radioactive waste problem? Neither the political authorities nor ONDRAF/NIRAS had realized in due time what the implications were in the field of public consensus when it turned out to be necessary to look for a favorable geology outside the existing nuclear sites. As a result, the publication of the NIROND 94-04 report in April 1994 leaded to a public deadlock.

The working method applied in the past by ONDRAF/NIRAS aimed to select the future disposal site for short-lived low and medium level waste on the basis of a scientific approach that had been very carefully worked out by its experts. ONDRAF/NIRAS thought – maybe rather naively – that the actual setting up of a repository would cause no problems once it had been proven that the chosen site was one of the best possible choices from a technical point of view. ONDRAF/NIRAS looked at this time for a solution for the radioactive waste problem in an objective and rational manner. Gradually, the agency understood the necessity to take into account the socioeconomic aspects of setting up a final repository on the national territory.

3. Breaking the stalemate

In 1995, in an attempt to break the stalemate, the government commissioned a study by ONDRAF/NIRAS on the possible alternatives to surface disposal. The final report, the NIROND 97–04 report, published in 1997, compared surface disposal with deep disposal and prolonged interim storage. It recommended that the government should base its decision on ethical considerations. Indeed, ONDRAF/NIRAS supports the view that the current generations are responsible for ensuring that future generations will not have to actively take care of the management of the radioactive waste they will have inherited.

On the basis of this report the Belgian federal government opted, on 16 January 1998 for a final or potentially final solution for the long-term management of short-lived low and medium level waste. The government also wanted this solution to be implemented in a progressive, flexible and reversible manner. With this decision, the prolonged interim storage option was abandoned in favor of either surface disposal or deep geological disposal.

At the same time, the government entrusted new missions to ONDRAF/NIRAS, to allow the government to make the necessary technical and economic choice between surface disposal and deep geological disposal. ONDRAF/NIRAS was assigned to develop methods, including management and dialogue structures, necessary to integrate a repository project at local level. Furthermore, ONDRAF/NIRAS had to limit its investigations to the four existing nuclear zones in Belgium, namely Doel, Fleurus, Mol–Dessel, and Tihange, and to the municipalities interested in preliminary field studies.

4. A new concept: the local partnership

After the government's decision of 16 January 1998, ONDRAF/NIRAS set up a work program based on a new work methodology. The idea of local partnerships was developed to ensure that every party liable to be directly affected by a collective decision has an opportunity to express its opinions. The local partnership project is an attempt to address the low and medium level waste disposal-siting problem through both technical research and concept development, and interaction with the (local) stakeholders. The partnership concept was developed by researchers in Social and Political Sciences of the university of Antwerp (UIA) and the research group SEED (Socio-Economic Environment Development) of the university of Luxemburg (FUL), on the basis of intense dialogue with ONDRAF/NIRAS.

The idea behind the partnership concept stems from the presumption that collective decision-making in a democratic environment is always a process of negotiation. Different interests, opinions and values are thereby weighted one against the other. This weighting of interests is something that should be done by the stakeholders and not for them.
By creating partnerships, ONDRAF/NIRAS intended to bring the decision-making process closer to the public, and to lower the threshold for active participation. The purpose was to create a representative body of the different stakes involved in this decision making process. On the one hand this is necessary to obtain a complete picture of the viewpoints, interests, needs and values that are at stake in this particular community, regarding this particular issue.
The general interest of the community will be the outcome of a process of dialogue and discussion among these different stakes. As many stakeholders, with as many different backgrounds and opinions as possible, should therefore be invited to actively participate in the partnership. Local partners should represent different political, economic, social, cultural and environmental movements or organizations within the community.
On the other hand, this setup should provide the key to creating an inclusive, transparent, flexible and stepwise decision making process that can be considered to be sustainable and fair by all parties. Even if, in the end, not everybody is completely happy with the outcome of the process, the fact that is was seen as fair, representative and transparent, can still make the outcome an acceptable one for the entire community.
Discussing in depth the pro's and con's of a low and medium level nuclear waste repository in the surroundings, however, is not something that can practically be done through public hearings with several hundred people attending. Therefore, it was decided to work out an adapted, clear organisational structure that fits the goal. This is why the local partnerships were set up as non-profit organisations of volunteers willing to discuss whether and under which circumstances they can possibly accept a repository; and with the mandate to work out an integrated pre-proposal of a repository, integrated in a broader added value project designed to fit the specific environment supported by the local population.
The concept of local partnership was first discussed with different local stakeholders and, on their recommendation, adapted to meet local needs.

5. Partnership: a non profit organization with a clear objective

A local partnership should be considered as a representative democracy on a micro level. Overseeing the whole "operation", a *general assembly*, uniting representatives of all participating organizations, decides on the main course and sets out the beacons for the actual discussions. The general assembly appoints an *executive committee*, in charge of the day-to-day management of the organization. The committee is, amongst many other things, responsible for the co-ordination of working group activities, decision making on budget spending and the supervision of the project co-ordinators.
In several *working groups*, all different aspects of the implantation of a low level waste repository in the community are being discussed. Here all relevant existing research is taken into consideration, the need for additional studies is evaluated and independent experts whose opinion is considered as relevant are invited to participate in the debate. The working groups concentrate on technical aspects, such as siting and design, environment and health, safety assessment as well as on social aspects: local development. The working group Local Development analyses socio-economic issues and projects, formulates prioritisation criteria and founding modalities. The more technical working groups evolve from general information through specific information on siting and the disposal concept towards a final disposal concept. The working groups report regularly to the executive committee. They are composed of both representatives of the organizations that founded the partnership, as well as individual citizens who expressed an interest to participate actively in this discussion forum. Within those working groups, the ONDRAF/NIRAS representative enters into direct dialogue with the local community, interested in hosting

the project. Questions, reactions and suggestions from the public, required the organisation to rethink many aspects of the initial concept or project.
Since all these people participate on a voluntary basis, at least two full time *project co-ordinators* need to be employed by the partnership. These project co-ordinators take care of administrative and communication tasks and support the working groups both logistically and scientifically.
It was considered important that the partnership should have its seat at the heart of the community concerned. A partnership is not a field office from ONDRAF/NIRAS, but an independent local organization in which ONDRAF/NIRAS participates as the only non-local partner amongst a multitude of local stakeholders. This location “on site” gives the partnership a “face”. A clearly visible presence in the community creates awareness amongst the not participating citizens and the premises of the partnership can serve as an open platform where citizens can come with their questions, remarks interests, fears, values or concerns. In order to allow the partnership to work independently, each partnership receives an annual budget from ONDRAF/NIRAS of approximately 250.000 EUR.
Maybe the most important and probably the most innovative aspect of the partnership approach, is that the partnership does not only decide (or at least advises to the community council) on every details of the repository concept and where it should (or should not) be implanted. Through the partnership, the local community can decide on what they consider to be the necessary conditions (technically, environmentally, aesthetically, socially, etc.) for such a repository.
Furthermore, within the partnership, an accompanying local project that seeks to bring added value to the community will be developed to obtain an integrated project creating a win-win situation.
When finally, all, or at least a majority of the parties involved come to an agreement on what their integrated project should look like, this is presented to the municipal council which decides to accept or reject the proposal, adding or not some specific conditions. Since the final word in this matter lies with the municipal council, it is also essential that council members are fully aware of the implications of their decision. To avoid the risk of conflicting interests between local politicians and the other members of the community, an active involvement of the representatives of the political arena is hence encouraged.
It was the responsibility of the federal government at last to make a choice between surface disposal or deep disposal and to decide where the repository should be implemented.

6. Most important achievements of the programme

As a result of the new approach, three local partnerships have been created ; the first one with the municipality of Dessel (creation of STOLA-Dessel in 1999), the next one with the municipality of Mol (creation of MONA in 2000) and the third one with the municipalities of Farciennes and Fleurus (creation of PaLoFF in 2003).
On 5 November 2004, after five years intensive work, the STOLA-Dessel partnership submitted its report to the municipal council. The MONA local partnership presented its findings to Mol municipal council on the 27th of January 2005. Both local partnerships proposed two different technical disposal concepts, one on surface, the other in the Boom clay formation present in the underground of their municipality. Both local partnership considered the disposal of category A waste as acceptable, provided that all their conditions are met. These conditions relate to various areas. The concerns of the local communities about the possible effects of a repository on health, safety and the environment are reflected in a number of concrete and strict conditions regarding the disposal concept. Furthermore, the local inhabitants expect, as initially promised, that a disposal project will bring social, cultural and economic added value, which will benefit the future development of the municipality. Finally, they demand continuous participation in monitoring the future development of the project and explicit their appreciation of the contribution made by the municipality for solving this important social problem.
The municipal council of Dessel pronounced itself unanimously on this dossier on 27 January 2005. The municipal council of Mol pronounced itself on 25 April 2005. The municipalities of Dessel and Mol and

their respective partnerships declared themselves ready to consider the possibility to accept a disposal site on their own territory at condition of the respect of specific conditions associated to the implementation of the disposal infrastructures.

As before, local participation will still constitute a critical factor for success in future discussions. In the community of Dessel a new partnership STORA has been founded on 27 April 2005. This partnership will not only do the follow-up of the STOLA file but will also discuss on the management of all radioactive waste stored on the territory of Dessel. Meanwhile, partnership MONA modified its statutes and its name on 24 November 2005 but kept the original acronym unchanged.

On 21 December 2005, the partnership in Fleurus and Farciennes, who had developed a common half-buried disposal project, decided to submit the final report of PaLoFF to both local councils. But on 23 February 2006, the municipality councils of Fleurus and Farciennes decided to put an end to the integrated project and consequently to the participative process.

In May 2006, ONDRAF/NIRAS submitted his final report to the federal government. This report contained all the information necessary to take the political decision regarding the future of the program with full knowledge of the facts. This report also puts an end to the activities linked to the development of integrated pre-projects by the local partnerships.

On the base of this final report the council of ministers took on 23 June 2006 the decision to dispose of the low and medium active short-lived waste in a surface disposal installation based on the technical concept developed by the partnership STOLA, on the territory of the municipality Dessel.

This decision marks the transition to a new stage: the stage of the detailed studies (approximately 5 years) in which the licence application files that are necessary to start the construction of the repository will actually be prepared. As the municipalities have pronounced on the conditions that they lay down for a possible repository on their territory, the concrete implementation of the local conditions will be discussed with all stakeholders in the next stage of the decision-making process. At the end of the detailed studies period, all parties involved should have reached a final agreement fixing the rights and obligations of all the parties. Only when all parties will be in formal agreement with the municipality's conditions, does the conditional candidature become definitive. By order of the government, the continuity of the participation process will be ensured not only with the selected municipality of Dessel but with the municipality of Mol as well.

The construction phase will take another 4 to 5 years including the period for bringing the installation into operation. The repository could thus become operational in 2016 at the earliest.

The operational stage, i.e. filling the repository, will take about thirty years and will be followed by the final covering and closure of the repository, and by a monitoring phase of a few hundred years.

The total cost of the pre-project phase (1998 – 2006) is approximately 20 MEUR_{2006}, of whom 2,8 MEUR_{2006} for the working of the partnerships.

The cost estimation for the detailed studies varies between 65 and 85 $\text{MEUR}_{2006.}$

The cost estimation for the disposal from the beginning of the construction to the end of the period of institutional control varies between 360 and 510 MEUR_{2006}.

7. Lessons drawn so far

Co-decision making is a dynamic but time consuming process that requires a permanent dialogue on how to realise a project. Close interaction with local stakeholders is an absolute necessity. Maintaining the continuity of the approach is also vital. Mutual learning and understanding, respect, transparency, openness, ability to listen are key elements. The real question is not so much the acceptance but how integrating a repository project in the social and cultural context of a specific place.

THE SWISS HIGH-LEVEL WASTE PROGRAMME: STATUS AND FUTURE CHALLENGES

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ABSTRACT

After about 25 years of studies and investigations covering both the crystalline basement as well as the overlying sediments in northern Switzerland, Nagra submitted at the end of 2002 comprehensive project documentation on the feasibility of safe disposal of SF / HLW and long-lived ILW in Opalinus Clay (Project *Entsorgungsnachweis,* or “demonstration of disposal feasibility” in English). The recently published reviews by the Swiss safety authorities all came to a positive conclusion about the project. The review phase was followed by a broad public consultation phase in 2005. Based on the results of the review and the public consultation phase, the Swiss Government (the Federal Council), in its meeting on 28 June 2006, decided to accept the demonstration of disposal feasibility. Furthermore, Nagra recently published a report on the siting possibilities for a SF / HLW / ILW repository from the geological point of view. All these documents show that a level of maturity has been reached that allows moving towards siting of such a repository. As a first step in the siting process the Swiss Government is currently preparing a site selection procedure that also defines the corresponding criteria; a first draft has been published in June 2006. Thus, the Swiss HLW programme is currently moving towards the important and challenging phase of deciding on the site for repository implementation, involving all relevant stakeholders.

1. Introduction

In Switzerland, nuclear power production by the current five plants contributes approximately 40% of the electricity consumed; the other 60% come from hydro power. The first commercial nuclear power plant went into operation in 1969, and since the late 70’s investigations have been underway with respect to the safe disposal of high-level radioactive wastes in Switzerland (see the broad review of options reported in 1978 [1]). A first milestone was reached with submission of Project Gewähr in 1985, which looked at the feasibility of safe disposal of high-level waste in the crystalline basement of northern Switzerland ([2]). This study was based on a regional field programme (deep boreholes, 2D reflection seismics, investigation of regional geology, etc.), lab work and studies. In their review the Swiss authorities came to the conclusion that long-term safety and engineering feasibility were adequately demonstrated but that siting feasibility – the likelihood that a sufficiently large block of crystalline rock with the properties assumed in the study could be found and characterised with sufficient reliability – was not yet fully convincing due to the complex tectonic situation in northern Switzerland. The authorities also required that the investigations should be extended to sedimentary formations. Already in the field programme for the crystalline basement, Nagra had characterised the most promising sedimentary layers overlying the crystalline basement. From this data, together with data from other sources and taking into account the good general understanding of the geology of Switzerland, Nagra published a first interim report on the broad geological possibilities in sedimentary rocks in 1988 [3]. Based on this interim report and two other interim reports ([4], [5]), in 1994 an agreement with the relevant Swiss authorities was reached that field investigations for the HLW programme should focus on the Opalinus Clay (a Jurassic claystone formation) as a host rock and the Zürcher Weinland as a potential siting region for a comprehensive demonstration of disposal feasibility. After extensive investigations (including regional geological studies, experiments at the Mont Terri rock laboratory, and using information from other sources), Nagra submitted at the end of 2002 comprehensive project documentation on the feasibility of safe disposal of SF / HLW and long-lived ILW in Opalinus Clay in the potential siting region of the Zürcher Weinland (Project *Entsorgungsnachweis*), see [6], [7], [8]. The recently published reviews of the Swiss safety authorities and their experts, as well as the review by an international review team under the auspices of the OECD / NEA, which was published in April 2004, all came to a positive conclusion about the project ([9], [10], [11], [12]). The review phase was followed by a broad, three-month public consultation phase in the fourth quarter of 2005. Based on the results of the review and the public consultation phase [13], the Swiss Government (the Federal Council) concluded on 28 June 2006 that disposal feasibility of SF / HLW / ILW in Switzerland had been successfully demonstrated [14].

However, besides the Opalinus Clay and the potential siting region of the Zürcher Weinland, other potential host rocks and siting alternatives also exist. Because of the public interest in the focussing on the Opalinus Clay and the Zürcher Weinland in 1994 ([5]), in 2002 a German – Swiss commission asked the AkEnd (a German working group that had the responsibility to develop a site selection process for Germany) for a review of the process that led to focussing the work on Opalinus Clay and the Zürcher Weinland. In their review the group came to the conclusion that the selection process had been conducted, by international standards, in a state-of-the-art manner. Furthermore, the group considered that the selection – based on safety criteria – of the Zürcher Weinland as the preferred option for the siting of a HLW repository was justified ([15]). Due to the continuing interest in alternatives, and in response to a request by the responsible Federal Minister, Nagra published in 2005 a report on the siting options for a HLW repository from the geological point of view ([16]), the results of which are summarized below in Section 2.

The information available and the documents produced show that a level of maturity has been reached that allows moving towards siting of a HLW repository. As a first step in the siting process the Federal Office of Energy is currently preparing a document defining a site selection procedure along with the corresponding criteria ([17]). It is expected that this site selection procedure will be approved by the Swiss Government in 2007 after a period of broad consultation with the cantons, the neighbouring countries and different interest groups. The siting process will allow extensive public participation.

2. Evaluation of siting options for a repository for HLW

There is broad agreement that in the evaluation of siting options, safety has highest priority. Thus, in the first phase in the evaluation of siting, the main emphasis is placed upon those geological properties that are essential for safety and that ensure engineering feasibility; in the next phase, other issues like land use planning, environmental impact issues and socio-economic aspects also need to be considered.

In the recently published report on the siting possibilities for a HLW repository [16], Nagra applied a step-wise narrowing-down procedure to systematically evaluate the different geological options that Switzerland offers:

– In a first step, those parts of Switzerland that are less suitable with respect to long-term stability and / or are tectonically very complex were identified and were excluded from further considerations.

– In a second step, all the different rock types in the area not excluded in step one were assessed with respect to their potential for being a suitable host rock.

– In a third step, the spatial distribution of the potential host rocks in the geologically stable parts of Switzerland was derived (resulting in distribution maps). Taking into account larger scale (regional) geological features / elements, potential siting regions were identified.

– Finally, in a fourth step, the different resulting regions were assessed on a case-by-case basis.

The resulting siting options for HLW from the geological point of view are depicted in Figure 1 (from [16]) and can be summarised as follows: From the geologic stability point of view the Alps, the region south of the Rhinegraben as well as the Folded Jura are excluded from further considerations, the latter mainly due to its tectonic complexity. Within the remaining parts of Switzerland – the (eastern) Tabular Jura and the Plateau-Molasse – the analysis of stratigraphic profiles led to the conclusion that (from bottom to top) the crystalline basement, the Opalinus Clay (in some parts combined with other low permeability strata below and above the Opalinus Clay) and the Lower Freshwater Molasse are potentially suitable host rocks. However, they differ markedly in their safety-relevant characteristics: The Opalinus Clay - a sedimentary rock deposited in a shallow marine environment - offers excellent radionuclide retention properties (very low permeability, good sorption properties, self-sealing of repository-induced fractures (e.g. EDZ)), due to its significant clay content and compaction, fine pore structure and homogeneity. The Lower Freshwater Molasse – a rather heterogeneous fluvio-terrestrial formation consisting of fine grained sediment layers (marly mudstones, etc.), which are intersected by more permeable sandstone channels – has less favourable transport properties. Also, the crystalline basement in northern Switzerland is intensely fractured due to its tectonic history and therefore also has less favourable transport properties than the Opalinus Clay. Due to their heterogeneity, both the Lower Freshwater Molasse and the crystalline basement are difficult to characterise and even after extensive characterisation, significant residual uncertainties with respect to transport properties are expected (the so-called undetected fast channels). This implies that for these two host rocks, it may be necessary to adopt a different safety concept: In contrast to the Opalinus Clay – where the host rock is the key barrier providing safety – in these rocks there may be the need to use very long-lived canisters (e.g. canisters with a copper shell as in Finland / Sweden) to achieve a convincing safety case.

In the Opalinus Clay of northern Switzerland several potential siting regions have been identified, including (from east to west): Zürcher Weinland, Nördlich Lägeren, Bözberg and Jurasüdfuss. There are again distinct differences between these potential siting regions, the differences being the properties of the under- and overlying rocks (in some regions these provide an additional strong transport barrier), the tectonic regime (the influence from Alpine orogeny ranges from being negligible to being clearly established), the lateral extent of non-disturbed rock, the depth of the repository horizon in the host rock, the boundary conditions for surface exploration (Quaternary cover, topography, housing density, etc.) with clear advantages for the Zürcher Weinland from the point of view of geology and safety.

However, this evaluation purely from the safety / geology point of view is considered not to be sufficient for final site selection; other factors need also to be taken into account. The site selection procedure currently under preparation by the Federal Office of Energy (already mentioned above) will also address land use planning, environmental issues and socio-economic aspects. This procedure will be based on the existing Land Use Planning Legislation (Sectoral plan or, in German, *Sachplan,* [17]). Interested parties (cantons, neighbouring countries, etc.) and the public will be involved in defining the site selection process.

3. Developing the future RD+D programme

After the positive review of project *Entsorgungsnachweis* by the Swiss authorities, the RD+D programme is currently being revised by Nagra. Already in the documentation of Project *Entsorgungsnachweis* key elements of the future RD+D programme were included. Currently, the reviews by the authorities are being analysed for further input and also experience abroad is being taken into account in defining the RD+D goals for next few years. The key elements of the RD+D programme will be included in the "Waste Management Programme" (in German: *Entsorgungsprogramm*) that is required by law. Nagra is currently developing this programme on behalf of the Swiss waste producers. Issues to be addressed in the RD+D programme include:

– the engineered barrier system (waste matrix stability, canister materials and design, properties of bentonite pellets used as buffer, etc.),
– the host rock (transport properties (including sorption), gas migration, geomechanics, etc.),
– repository design (encapsulation facility, different repository components, etc.),
– process models, systems analysis and safety analysis.

The RD+D programme will continue to rely on the underground rock laboratory at Mont Terri and also on the results from other URLs (e.g. Bure, Äspö, Grimsel). In addition, the Waste Management Laboratory at the Paul Scherrer Institute with its extensive infrastructure (hot laboratory, synchrotron light source, etc.)

and other competence centres (e.g. at the University of Berne) will continue to play an important role. Furthermore, international co-operation will continue to be a key element in the future RD+D programme. The results from RD+D together with site-specific information and the results from other projects will be used as technical input for the site selection process which eventually will lead to a general licence application for a HLW repository.

4. Summary and conclusions

After about 25 years of studies and investigations (including regional and localised field investigations as well as extensive research, development and demonstration projects in the two underground rock laboratories at Grimsel and Mont Terri) covering both the crystalline basement and the overlying sediments, a significant level of maturity of the technical programme has been reached. Furthermore, the recently revised Nuclear Energy Law and the corresponding Ordinance provide a suitable basis for further developing the HLW programme. As a next step, a detailed site selection procedure is currently being developed by the Swiss authorities. Based on the good understanding of geology and the features that contribute to safety, it should be possible to move towards selecting a site for which a general licence will be submitted. This site selection process, however, will be very challenging and requires not only a high-level technical programme and the ability to plan and develop projects that consider the needs of the region (land use planning, environmental impact assessment, etc.), but also interaction skills for the dialogue with a broad spectrum of stakeholders involved in the process.

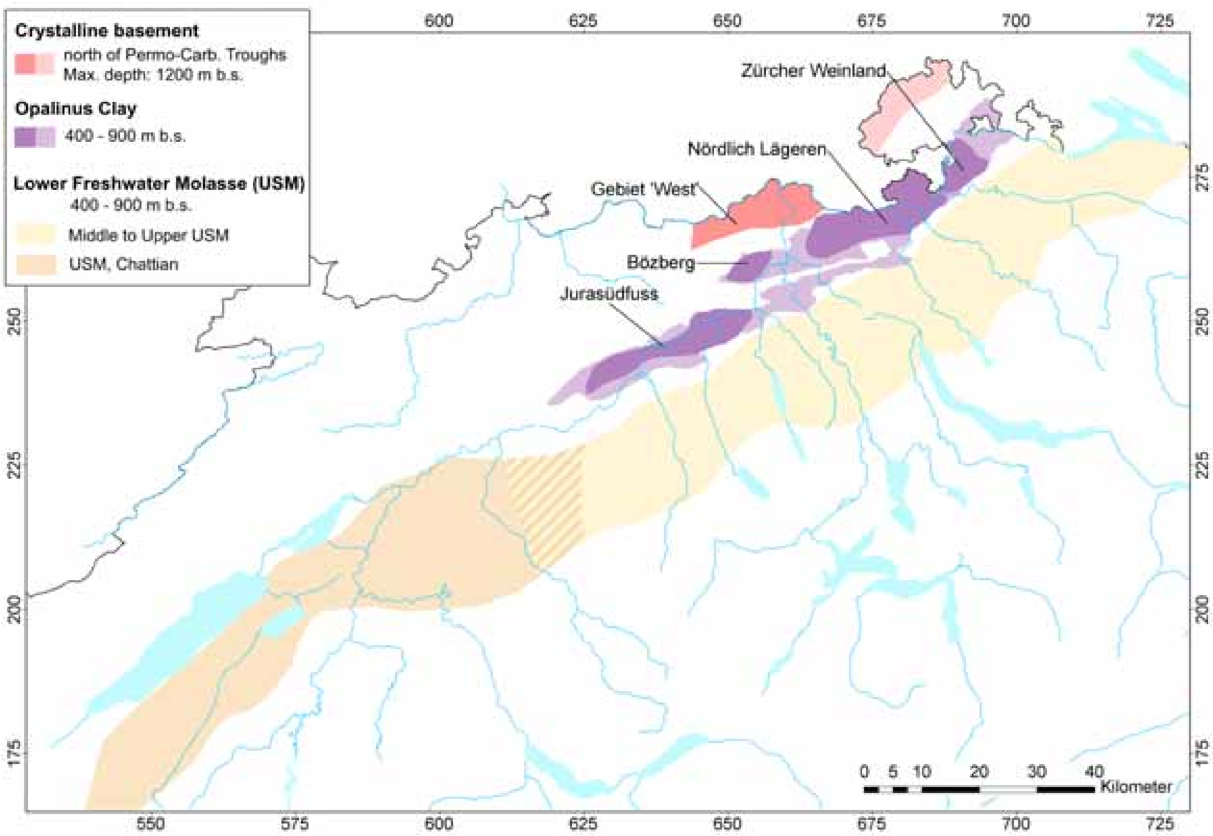

Fig. 1. Spatial distribution of potential host rocks at suitable depth and potential siting regions for a SF / HLW / ILW repository in Switzerland.

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FUTURE EXTENSION OF THE SWEDISH REPOSITORY FOR LOW AND INTERMEDIATE LEVEL WASTE (SFR)

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ABSTRACT

The existing Swedish repository for low and intermediate level waste (SFR) is licensed for disposal of short-lived waste originated from operation and maintenance of Swedish nuclear power plants.

The repository is foreseen to be extended to accommodate short-lived waste from the future decommissioning of the Nuclear Power Plants.

Long-lived waste from operation, maintenance and eventually decommissioning will be stored some years before disposal in a geological repository. This repository can be build either as a further extension of the SFR facility or as a separate repository.

This paper discusses the strategy of a step-wise extended repository where the extensions are performed during operation of the existing parts of the repository. It describes the process for licensing new parts of the repository (and re-license of the existing parts).

1. Introduction

Sweden has today (the year 2006) 10 commercial nuclear power reactors in operation (7 BWR:s and 3 PWR:s) at three sites along the seacoast. These nuclear power plants produce almost 50 percent of the electricity used in Sweden. The time schedule and conditions for shut down are still open. Based on technical, safety and economical considerations, the operation of a reactor may continue until 40 to 60 years of operation or even longer. The two reactors at the Barsebäck site (BWR) have been shut down in the year 1999 and 2005 respectively. The shut down was done before 30 years of operation for political reasons.

The 12 Swedish commercial NPP’s were commissioned during a short time span, from 1972 to 1985. Consequently the operation should be terminated between the year 2012 and 2025 if an operating lifetime of 40 years is assumed. Dismantling of a reactor close to a still operating unit will not commence until the other unit has ceased operation. With this strategy the first reactors in Sweden should start dismantling around the year 2020, after removal of the spent fuel. This is also the year scheduled for opening the extended repository for disposal of decommissioning waste.

2. The existing repository

The existing repository for short-lived radioactive waste (SFR 1) has a capacity to accommodate waste from operation of the remaining reactors in Sweden up to almost 60 years of lifetime (Figure 1). There might be a need for an extension for some of the intermediate level waste produced during operation and maintenance.

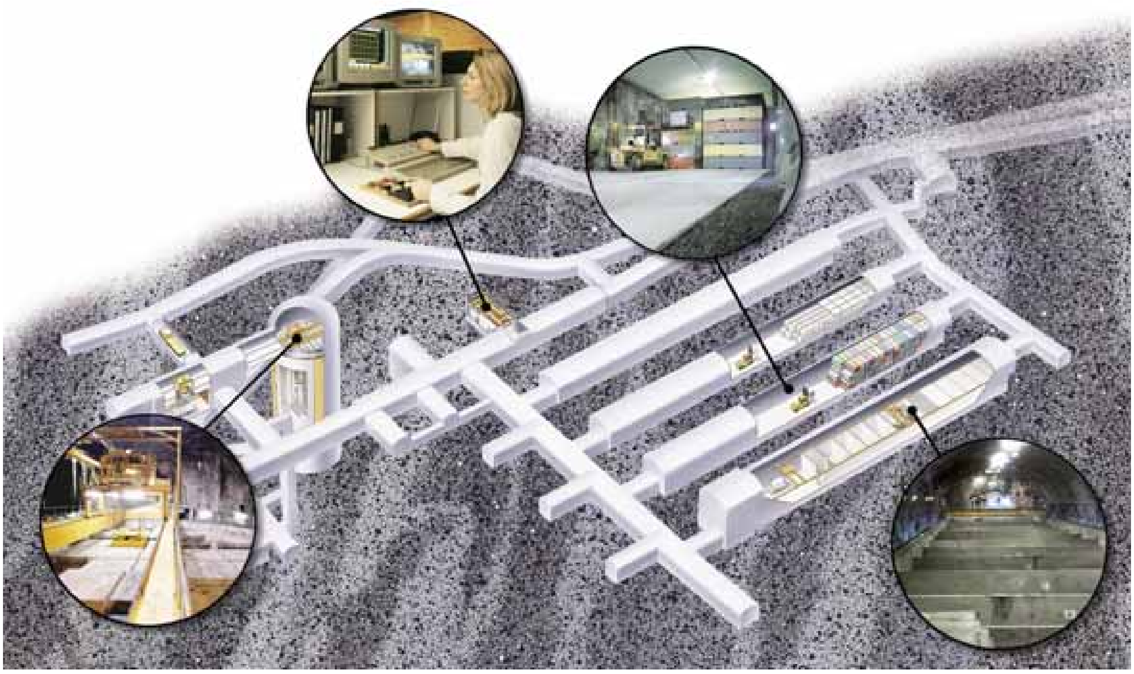

Figure 1. The existing repository, SFR, for operational low and intermediate level waste

3. The step-wise approach to extension

The situation today is that Sweden has reactors already shut down while other reactors are foreseen for a life extension to 60 years or more. The need for a repository for decommissioning waste is spread over a long time period and therefore a step-wise extension of the repository is planned. A first step would be to accommodate waste from early shut-down reactors e.g. the reactors at the Barsebäck site. This first phase of extension may also include capacity for some operational waste. The following step of extension of SFR is depending of the lifetime of the reactors and of the strategy for dismantling adopted by the different operators (Figure 2).

The total volume of short-lived low and intermediate level decommissioning waste has been estimated to approximately 150.000 m^3 (processed, packaged and ready for disposal). The two reactors in Barsebäck will together give rise to about 18.000 m^3 of radioactive waste for disposal in SFR. Another 200 m^3 of long-lived waste is anticipated for interim storage before the eventual disposal in a separate repository for long-lived waste. The exact quantity of decommissioning waste is depending on many factors like the strategy for processing the waste, on the requirements for free release measurement and on the time schedule for dismantling.

When the existing repository was planned and constructed a step-wise extension for operational waste was foreseen. The fist step (63.000 m^3 of waste) which is now in operation was supposed to last 10-15 years. After that an extension to a total of 90.000 m^3 was planned. To facilitate the extension the tunnel system was prepared for further excavation in the way that niches were prepared for new disposal chambers. The idea was to excavate the new chambers during operation of the first phase. As a preparation for future decommissioning of the nuclear power plants also niches to caverns for decommissioning waste were excavated already during the first construction phase.

The originally planned second step for operational waste had a disposal capacity of 30.000 m^3 of waste. This space will now be available as part of the first step for decommissioning waste. The exact volume available can not be determined until a detailed rock characterisation has been performed. After the site investigation a detailed lay-out and design study can be done. The total available volume is assumed to be in the range of 50.000 m^3 in this stage. The total volume of decommissioning waste for disposal in the SFR will be determined based on experience from dismantling end demolition of the first reactors. When the second stage is needed, today estimated to 100.000 m^3 depends on the lifetime of the reactors and of the strategy for dismantling taken by the individual utility owner.

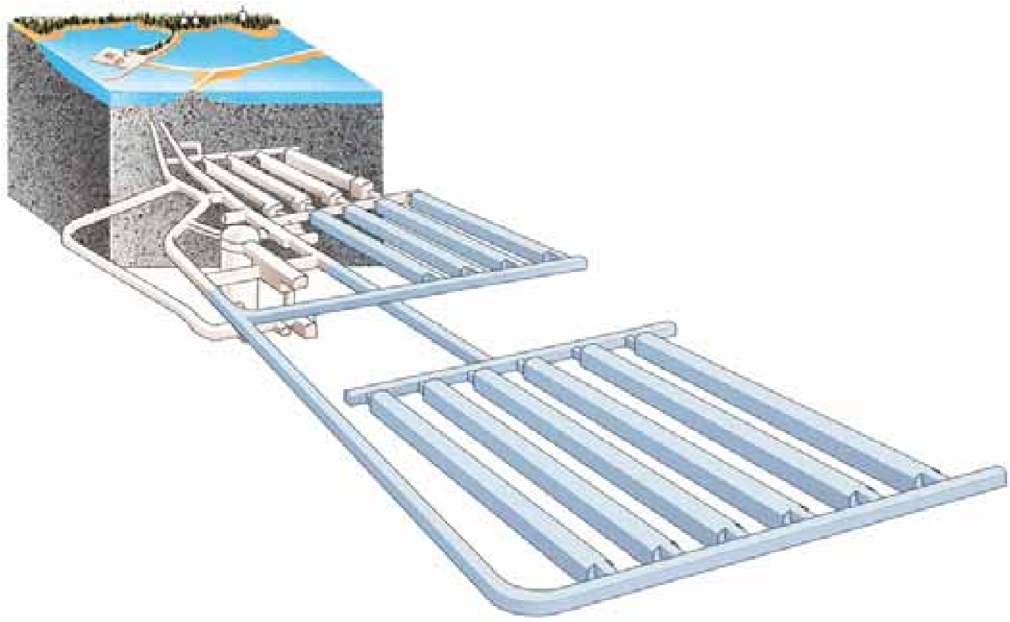

Figure 2. Extended repository for disposal of decommissioning waste. First phase close to the existing part and the second phase in the foreground.

4. Legal aspects

There is no national policy requiring a certain starting point or endpoint of decommissioning. When the dismantling will start is a decision to be taken by the owner of the power plant. Only if there is radiation protection or safety hazards the Swedish authorities may require an earlier decommission. Since no major decommissioning project has been performed so far the conditions to be achieved have been given by the Authorities on a case-by-case basis. It is the responsibility of the owners of the power plants to plan and to execute the decommissioning of their plants. The responsibility to take care of its own waste is stipulated in the Act on Nuclear Activities and is included in the license to operate a nuclear facility. Generic decommissioning studies are performed by the Swedish Nuclear Fuel and Waste Management Co, SKB.

Following the final shut down of a reactor and removal of spent fuel it is assumed that the dismantling will commence, preferably after a thorough decontamination of the process systems. In addition a period of 5 – 15 years might be used for decay of short-lived radionuclides. If and how long this period will be has to be decided by the utilities for each nuclear power unit from an ALARA standpoint.

5. Scheduled activities

Investigations at the SFR site are expected to start next year (2007) with the aim to license and carry out the extension so that the first decommissioning waste can be disposed of in the year 2020. The site investigation will cover the fully excavated site and the safety assessments for the long term period will include a rough estimate of the fully excavated repository and a detailed analysis of the first step.

During the extension temporary walls are put into the niches and the excavation (drill and blast technique) will be from the "back-side" of the area. The so called construction tunnel will be used for transport of personnel and blasted rock. After excavation and installation of necessary equipments the temporary walls are removed and disposal activities may commence in these new parts.

The excavation needs to be licensed both according to nuclear/radiological laws and regulations and according to the environmental law. A new environmental impact study is needed and the application will be for short-lived radioactive waste from operation, maintenance and decommissioning of nuclear power plants and similar waste from research, hospitals and industry. By re-licencing the existing repository to cover also waste from decommissioning an optimal use of the repository is possible.

6. Summary

The existing repository for short-lived radioactive waste is licensed for operational waste only. A re-licensing is needed to accept also waste from decommissioning. The waste volume from operation and decommissioning is based on annual prognosis for operational waste and on decommissioning studies performed by SKB and the utilities. As decommissioning probably will be carried out over a rather long time period excavation will be in steps. The first step is planned for operation from the year 2020. It will mainly be built to allow for disposal of waste from dismantling of the first reactors but also for some operational waste where the existing facility is not sufficient. The following extension(s) will be scheduled according to operational lifetime of the reactors and the strategy of the utility owners for decommissioning.

OPG'S DEEP GEOLOGIC REPOSITORY FOR LOW AND INTERMEDIATE LEVEL WASTE – RECENT PROGRESS

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ABSTRACT

This paper provides a status report on Canada's first project to build a permanent repository for the long-term management of radioactive waste. Ontario Power Generation has initiated a project to construct a deep geologic repository for low- and intermediate-level waste at the Bruce Nuclear Site, at a depth in the range of 600 to 800 m in an Ordovician-age argillaceous limestone formation. The project is currently undergoing an Environmental Assessment and consulting companies in the areas of environmental assessment, geoscientific site characterization, engineering and safety assessment have been hired and technical studies are underway. Seismic surveys and borehole drilling will be initiated in the fall of 2006. The next major milestone for the project is the submission of the Environmental Assessment report, currently scheduled for December 2008.

1. Introduction

Ontario Power Generation (OPG) is responsible for the safe management of the radioactive wastes arising from the operation of 20 CANDU reactors in the Province of Ontario, Canada. The purpose of this paper is to provide an update on a project to construct a Deep Geologic Repository (DGR) for the long-term management of the low- and intermediate-level wastes (L&ILW) arising from these reactors. The location of the proposed DGR is the Bruce Nuclear Site which is located about 225 km north-west of Toronto, between the towns of Kincardine and Port Elgin, on the east shore of Lake Huron.

In 2002, the Municipality of Kincardine, the host community for the Bruce Nuclear Site, signed a Memorandum of Understanding (MOU) with OPG to jointly study options for the long-term management, at the site, of all L&ILW arising from the operation, refurbishment and decommissioning of OPG-owned reactors in Ontario. All L&ILW generated by these reactors are now in interim storage at OPG's Western Waste Management Facility (WWMF) which is located on the Bruce Nuclear Site, along with the eight reactors currently operated by Bruce Power under a lease agreement.

In the joint study with Kincardine, consultants were hired to identify various options and conduct geotechnical, safety assessment, and environmental and social impact studies. Following completion of the studies Kincardine council indicated a preference for the deep repository option and a Hosting Agreement based on this option was negotiated in late 2004. A community poll was conducted in early 2005 and the Council's position was solidly endorsed.

Documents describing the above processes and studies in more detail can be found on the project website at www.opg.com/dgr.

Following the successful community poll, and a formal decision by OPG to proceed with the DGR project, activities have been progressing on a broad front. Detailed work plans have been developed in the areas of environmental assessment, geoscientific Site Characterization, engineering and safety assessment and supporting contracts have been put in place. As well, extensive stakeholder engagement activities are continuing in order to ensure that all stakeholders are kept aware of project developments and that any concerns are being addressed. This paper describes these current activities in more detail.

2. Geologic Setting

The Palaeozoic rocks underlying the Bruce Nuclear Site are comprised of a near-horizontally layered, undeformed sequence of carbonates, shales, evaporites and minor sandstones within the Michigan Basin. This sedimentary sequence is approximately 800 m thick resting upon the crystalline Precambrian basement. The repository is currently targeted for a argillaceous limestone formation at a depth of about 660 m below surface. This formation is overlayed by a 200 m layer of low permeability shale. These Ordovician-age shales and limestones are expected to have rock mass hydraulic conductivities between 10^{-13} to 10^{-12} m/s and under these conditions it is expected that solute transport in these media would be diffusion controlled.

3. Design Concept

Figure 1 provides an illustration of the current conceptual design of the facility. In the concept the underground repository would be comprised of a series of horizontal emplacement rooms, some dedicated to low-level waste and some to intermediate-level waste. Access to emplacements rooms would be by shafts and access tunnels. It is currently assumed that the repository would need to be designed for about 160,000 m^3 of waste to handle all operational and refurbishment wastes from the 20 OPG-owned reactors. The design assumes that low-level waste containers will be transferred as is from their current above-ground storage locations, while the intermediate-level wastes containers would be placed in sacrificial concrete shields, as required by occupational dose considerations, prior to being transferred underground.

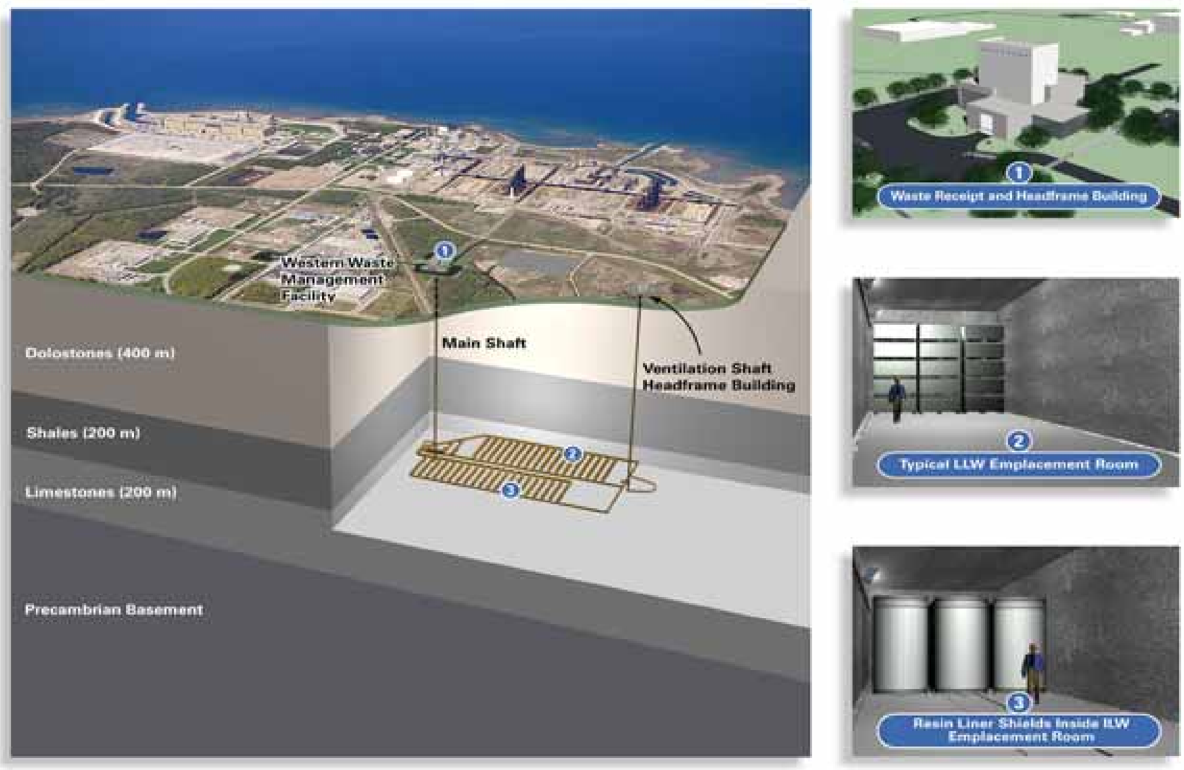

FIG. I. Proposed Deep Geologic Repository for L&ILW on Bruce Nuclear Site

4. Safety Concept

The essence of the safety case for the repository is being built around the following key arguments:

- The rock is very old and stable,
- The geology and hydrology are predictable at repository depth,
- The deep groundwaters at the site are very old and do not mix with surface waters,
- Post-closure dose estimates are very small because mass transport is diffusion controlled,
- Earthquakes, glaciations or other natural events will not disrupt the repository,
- Gases generated by wastes will remain within the rock,
- The repository is safe from inadvertent human intrusion, and
- The repository can be built and operated safely using proven technologies.

Detailed site characterization and safety assessment studies are expected to support the above arguments through multiple lines of reasoning.

5. Current Activities

5.1 Environmental Assessment

An Environmental Assessment (EA) for the proposed DGR, in accordance with the Canadian Environmental Assessment Act, is currently underway. The Canadian Nuclear Safety Commission (CNSC) is the lead regulatory agency and will be conducting a public hearing, associated with determining the scope of the EA, in Kincardine on October 23, 2006. Subsequent to this hearing, the EA Scoping Document will be finalized. As well, the EA track (Comprehensive Study or Panel) will be determined.

A consulting company has been hired to conduct technical studies and to document the EA with a planned submission date to the CNSC in December 2008.

Following EA approval, a process will be initiated to obtain a Site Preparation/Construction Licence from the CNSC, targeted for receipt in the 2011/2012 timeframe.

5.2 Geoscientific Site Characterization

To date the geologic setting at the site has been predicted through regional geology studies as well as from deep boreholes drilled in the vicinity of the site for other purposes.

A detailed Geoscientific Site Characterization Plan (GSCP) has been prepared with the assistance of a specialized consulting company supported by a number of international experts and an independent, OPG-established Geoscience Review Group with members from Canada, France, Switzerland and United States. The Plan is available on the project website.

Phase 1 of the Plan has been developed to support the project EA. In Phase 1, 18 km of 2-D seismic lines will be shot. Two boreholes will also be drilled and tested. One will be about 400 m in depth and will be used to investigate the upper stratigraphy. The second will be drilled through to the Precambrian basement at about 800 m depth and will be used to investigate the lower Ordovician-age shales and limestones. The seismic survey field work is expected to start in late September 2006, and the borehole drilling is expected to start in early November 2006.

The same consulting company that led the preparation of the GSCP has been hired to coordinate the execution of all GSCP field and laboratory activities. A descriptive geosphere model report based on the results of these investigations is expected to be available in late 2007.

Phases 2 and 3 of the GSCP will include additional boreholes and will support the process to obtain a Site Preparation/Construction Licence for the facility.

5.3 Geosynthesis

A second geoscientific consultant is being hired to lead the preparation of a Geosynthesis report. This report will integrate the results of the geoscientific site characterization program, as well as studies in the areas of long-term climate modeling, geologic framework studies, regional hydrogeochemical and geomechanics assessments, regional hydrogeologic modeling; and site-specific geochemical, geomechanical and hydrogeologic modeling. The Geosynthesis report will provide a multi-faceted, reasoned argument why the geologic conditions at the Bruce Nuclear Site will ensure the safe containment and isolation of the L&ILW to be emplaced in the repository. This report is expected to be available in mid-2008.

5.4 Engineering

The current conceptual design was developed in 2003/2004 to support the joint options study conducted with the Kincardine Municipality. A consulting company has recently been hired to further study and develop the concept. Particular areas of review will be:

- shaft access versus ramp access,
- concept for shielding of intermediate-level wastes,
- waste room design,
- underground waste handling, and
- room seal and shaft seal design.

Rock stress conditions and geomechanical properties of the host rock obtained from the geoscientific site characterization activities will be an important input into the design activities.

An updated conceptual design is expected to be available in late 2007.

5.5 Safety Assessment

The potential dose consequences of the DGR were estimated in preliminary safety assessments conducted in support of the joint options study with the Municipality of Kincardine. These results predicted very low public doses due to the expected diffusion-controlled deep geologic setting.

The planned site characterization activities will provide more detailed information on which to verify the assumptions made in the preliminary safety assessment studies, and to conduct more detailed modelling of the safety of the proposed DGR.

Two separate consulting companies have been hired to conduct the pre-closure and post-closure safety assessments.

The pre-closure safety assessment will cover normal operations and accident conditions. Impacts to workers, public and biota will be considered. The post-closure safety assessment will cover potential future impacts to people and to biota after the repository is closed. The assessment will generally follow the IAEA Integrated Safety Assessment Methodology (ISAM). Two iterations of these assessments are anticipated during the Environmental Assessment stage, to allow for inclusion of more detailed information from other work programs.

The safety of the DGR will be assessed in accordance with the CNSC regulatory guide G-320. The safety case will include both quantitative safety assessments as well as complementary evidence for long-term safety, including in particular evidence from the site geology. Scenarios will be developed that account for normal (or expected) evolution of the site and facility with time (groundwater and gas scenarios), disruptive events such as human intrusion, and hypothetical “what if” scenarios to test the robustness of the repository system.

Pre- and post-closure safety assessment study reports are expected to be available in early and mid-2008, respectively.

Following completion of Phases 2 and 3 of the GSCP and further preliminary design work, the safety assessment studies will be updated to support the Site Preparation/Construction Licence process.

5.6 Community Communications

OPG, and its predecessor Ontario Hydro, have a long history of working in partnership with the Bruce Community. The community relationship has been built on trust and transparency and channels of information exchange are well established.

Within this context, communications on the DGR proposal have been extensive over the last four years and the plan is to continue this throughout the regulatory approvals phase. A multi-tactical communication plan has been designed to engage all stakeholders in support of the environmental assessment process. The plan includes displays at community events, advertising, media events, newsletters, key stakeholder briefings, open houses, speaking engagements and public attitude research.

6. Summary

The regulatory approvals phase for OPG's Deep Geologic Repository project is fully underway with a number of specialist consulting companies hired and technical studies are in progress. Key to the approvals phase is the successful execution of the Geoscientific Site Characterization Plan to confirm our belief that the Bruce Nuclear Site is an ideal location for a deep geologic repository for low- and intermediate-level radioactive waste.

The next major milestone in the project is the submission of the Environmental Assessment report, scheduled for late 2008. Earliest predicted in-service date for the repository is 2017.

OVERVIEW OF THE CURRENT AND PLANNED ACTIVITIES IN THE FRENCH UNDERGROUND RESEARCH LABORATORY AT BURE

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ABSTRACT

In November 1999 Andra began building an Underground Research Laboratory (URL) on the border of the Meuse and Haute-Marne departments in eastern France. The research activities of the URL are dedicated to reversible, deep geological disposal of high-activity, long-lived radioactive wastes in an argillaceous host rock. The studies covered four complementary aspects: acquisition of data (waste packages, material behaviour and clay medium), repository design and reversibility studies, analysis of the long term behaviour of the repository, safety analyses. For the next phase starting in 2007, Andra will carry out integrated tests of a technological scope, i.e. trial drift, demonstrator of current drift. The results should make it possible to assess the safety of a disposal over several tens and even hundreds of thousands of years and submit in 2015 a file for permission request for the HLW and ILW deep disposal.

1. Introduction

In November 1999 Andra began building an Underground Research Laboratory (URL) in eastern France. The research activities of the URL are dedicated to reversible, deep geological disposal of high-activity, long-lived radioactive wastes in an argillaceous host rock. The objectives of the URL for the 1999-2005 years were mainly the *in situ* characterization of the physical and chemical properties of this rock. The results of this research are presented in the file “2005 Argile” [1].The studies covered four complementary aspects:

- Acquisition of data concerning the waste packages, material behaviour and clay medium,
- Repository design: waste conditioning, repository architecture and integration in a geological site, operating modes and reversibility,
- Analysis of the long term behaviour of the repository and modelling of its thermal, mechanical, chemical and hydraulic evolution,
- Long term safety analyses.

For the next phase starting in 2007 the following activities will be carried out:

- Consolidate data acquired over the period 2002-2005 and conduct long term experiments,
- Carry out integrated tests of a technological scope, i.e. trial drift, demonstrator of current drift, demonstrator integrating clay core, concrete plug and buffer material, prototype of disposal vaults for ILW and HLW,
- Quantify more precisely the safety margins through development of modelling, flow-migration coupling (water-gas), reactive migration in desaturated environment, management of uncertainties and probabilistic methods.

2. URL Site Overview

The target horizon for the URL is a 130-m-thick layer of argillaceous rocks that lies between about 420 and 550 m below the ground surface. Stratigraphically speaking, the depositional period straddles the Callovian and Oxfordian subdivisions of the Middle to Upper Jurassic. Argillaceous rocks contain a mix of clay minerals and clay-sized fractions of other compositions. The clays, constituting 40 to 45% of the Callovo-Oxfordian argillaceous rocks, isolate the groundwaters. Silica and carbonate-rich sedimentary components reinforce the rock and ensure stability for underground construction.

The URL location lies in the eastern portion of the Paris Basin, which covers a major portion of Northern France. The beds are nearly flat-lying with a slight dip of less than 1.5° westwards towards the centre of the Basin. The deep-water depositional environment of the Callovo-Oxfordian argillaceous rocks created a homogeneous layer that is continuous over most of the Paris Basin. The stratigraphy of the URL site consists of Jurassic limestones, marls and argillaceous rocks (Figure 1). The major overlying limestone units are the Tithonian Barrois limestones, forming a surface veneer over the URL site, and the Oxfordian limestones from about 150 to 400 m in depth. Between the Tithonian and Oxfordian limestones is a 150-m-thick sequence of mixed Kimmeridgian argillaceous rocks, marls and limestones. Underlying the Callovo-Oxfordian argillaceous rocks are the Bathonian and Bajocian Dogger limestones and dolomitic limestones [9].

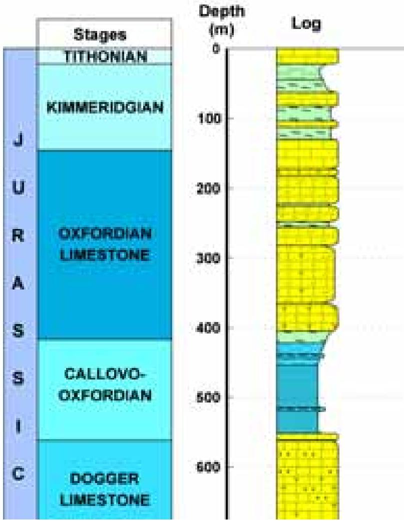

Figure 1 Stratigraphic column at the URL site

3. Construction of the URL

Two 500-m-deep shafts provide access from the surface to the argillite host rock. The main shaft has a 5-m diameter and allows access for personnel and equipment, material extraction and ventilation. The 4-m auxiliary shaft located 100 m away from the main shaft serves the ventilation system and provides not only mine safety, but also a second access for lowering equipment. From the shafts, the laboratory has two levels of access drifts at depths of 445 and 490 m (Figure 2). The upper drift will have a simple T-shaped configuration and a total length of about 4 m. It provides access to boreholes in order to monitor shaft-sinking effects through the argillaceous host rock [5]. The five hundred metres of drifts at the 490-m level constitute the key experimental level of the laboratory. Experimental zones are located in a specific area in order to allow construction and drift-fitting work to take place at the same time.

3.1. Shaft-sinking method

The choice of a suitable shaft-sinking method was limited to drilling and blasting. The approach was chosen over shaft-drilling methods for several reasons, including the lack of experience with shafts as large as the laboratory’s. With a view to saving time, raise-boring was ruled out because both shafts are being sunk in parallel from the surface. The most important consideration, however, was the need to conduct scientific activities and observations in the shaft during construction, which would have been very difficult in a shaft-drilling operation. The selected shaft-sinking method uses a multistage platform which supports all shaft-construction operations, including drilling and blasting, mucking and applying the concrete liner.

The support system was installed directly and immediately after excavation. It consists of bolts and wire mesh covered with shotcrete in order to prevent spalling. The final lining consists of concrete poured in 3-m sections at a time. The thickness of the concrete ring is approximately 30 and 45 cm in carbonates and argillaceous rocks, respectively. In addition, the stress within that lining is recorded by vibrating wires that are fitted while pouring the concrete ring.

3.2. Drift-opening methods

Due to the building requirements of the Bure URL, the drill-and-blast method was applied at a depth of 445 m and the pneumatic-hammer method to open drifts at a depth of 490 m. Supports consist of 2.4-m bolts and sliding arches. Down at 490 m, the floor is reinforced with bolts. A shotcrete lining sprayed over wire mesh prevents spalling.

Since one of the purposes of geomechanical measurements is to assess potential convergences and stresses on a final lining, specific zones have been instrumented for convergence measurements and measurements on supports. The observed convergences depend on the orientation of the drifts and on the excavation and support methods being used. After one year, the measured convergences are in the order of approximately 10 cm and deferred deformations (creeping) are observed.

4. Experimental programme carried out in 2004-2006

Studies and experimental work cover three major aspects in the URL drifts:

- Containment capability of the host formation
  This containment capability comes from the specific physical characteristics of the rock and the physico-chemical characteristics of the interstitial fluids and their interaction with the rock. The fundamental physical characteristic is permeability. This property is studied through various specific tests [4]. The chemical characteristics of the interstitial fluids condition the mobility of the various radionuclides likely to be found in the natural environment [6]. The studies focus on knowledge of the geochemistry of the interstitial fluids in equilibrium with the minerals in the rock and on the diffusion and retention capabilities of the radionuclides.
- Creation of damaged and disturbed zones associated with drift excavation
  The main purpose of the studies on this topic is to investigate how the rock reacts to the excavation of shafts and drifts, and the associated development of the damaged and disturbed zone [8]. Several techniques and methodologies used at Bure URL had been previously developed at Mont Terri Rock laboratory [7]. Damaged zone (EDZ) and disturbed zone (EdZ) were characterized during monitoring of the shaft and excavation of the experimental drift at 445 m (Figure 2).
  Measurements are grouped in the drift sections and within specific shaft excavation monitoring experiments. These experiments include a set of boreholes with instrumentation installed in advance in the drift.

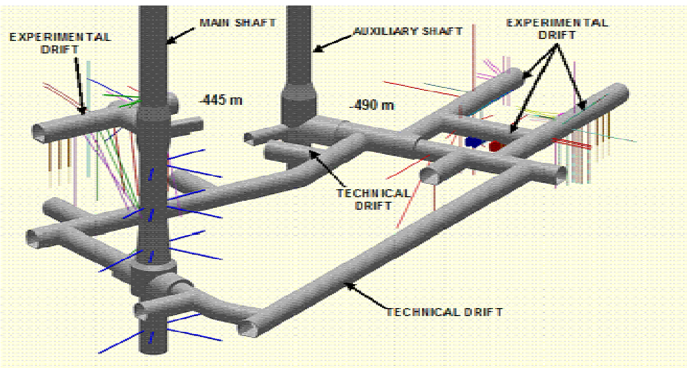

Figure 2 Detailed view of experimental drifts

- Assessment of sealing zone concept
  The sealing of a drift is a major issue when considering the disposal construction options. It involves designing systems to re-establish the original low permeability of the formation by overcoming potentially negative effects from the damaged zone surrounding the drifts and shafts. The system studied for the Andra project is called "anchor key". An anchor Key is a 30 to 40 meter long drift, filled with swelling clay (bentonite). Every five to ten meters a slot is made around the drift with a saw. This slot is filled with swelling clay and this device interrupts a potential flow along the drift. Experiments on the feasibility of an anchor key have been conducted, firstly in the Mont Terri Rock Laboratory [3] and subsequently in the KEY drift at the main level of the Bure URL.

5. Repository design and 2007- 2015 Programme

Through its construction and experimental activities, the laboratory has helped Andra to develop a concrete approach with a view to proposing suitable architectures and management methods for a repository [2].
The future work in the laboratory will include the life-size construction of the different components of a disposal facility, such as the cells or the plugs for cells and drifts. Once specifications will be set, it will be possible to draw a concrete preliminary design integrating the specific characteristics of the selected zone for the implementation of the repository.

5.1. Main repository features

The design of the repository is regulated by the safety approach that will lead to the sizing and specification of containment barriers with a view to:

- preventing water circulations (low permeability of the geological environment and of sealing and repository structures);
- immobilising radionuclides at the package level by creating or maintaining favourable physicochemical conditions for that retention;
- retarding and mitigating any potential migration of radionuclides outside disposal cells.

Hence, in the framework of the Meuse/Haute-Marne project, investigations have led to the proposal of a repository with the following features [2]:

- it is located at the centre of the layer in order to maximise the thickness of the impermeable geological formation and to ensure the best containment possible;
- disposal areas for that waste category are compartmentalised in order to reduce intrusion risks or failure consequences. The different waste categories are emplaced in separate disposal areas in order to simplify their safety assessments and to ensure the thermal independence of the different areas;
- structures have a simple geometry with circular profiles that are considered as the most stable and are lined in such a way to be stable for 100 years;
- materials coming in contact with the rock, whether they are natural (clay rock) or man-made (concrete, steel, plugging or backfill materials) help to maintain the physicochemical conditions retarding package alteration and degradation in order to limit the release of radionuclides in the biosphere.

5.2. Planned activities for the 2007-2015 period

For the next phase starting in 2007 the following activities will be carried out:

- Consolidate data acquired over the period 2002-2005 and conduct long term experiments (diffusion, porewater characterization, hydrothermal coupled phenomena), i.e.: hydromechanical evolution of the shafts and drifts, continuation of installed experiments beyond 2006, setting up experiments on rock/materials, diffusion experiment in the long-term (2007). These studies will lead to quantifying more precisely the safety margins through development of modeling (Flow-migration coupling (water-gas), reactive migration in desaturated environment, management of uncertainties and probabilistic methods

- Carry out integrated tests of a technological scope, i.e. trial drift (2007- 2008) and demonstrator of current drift (2008), construction of a lasting concrete cladding (2009-2010), demonstrator integrating clay core, concrete plug and buffer material (2007-2009), prototype of a disposal vault for ILW wastes (e.g. : l=80m) (2008-2009), 2 HLW horizontal demonstrator vaults (e.g. : l=40m)(2008-2010).
  These tests will include detailed studies of individual components, i.e. ILW and HLW disposal packages, handling equipment of packages through an international program (ESDRED), sealing of drifts, interfaces packages/handling equipment/vault and construction and closing of the vaults. In addition, detailed studies of bodies of architecture will be carried out. The studies will focus on shafts, infrastructures of the shafts zone, thermal dimensioning of the HLW area and layout and dimensioning of nuclear surface installations.

6. Conclusions

The experimental programme of the URL addresses the two major issues of demonstrating the natural isolation capability of argillite and the feasibility of constructing and operating a repository without compromising those isolation properties. Through its construction and experimental activities, the laboratory has helped Andra to develop a concrete approach with a view for proposing architectures and management methods for a repository.
The future work in the laboratory will include the life-size construction of the different components of a disposal facility, such as the vaults or the sealing for vaults and drifts. At the end of the 2006 law on nuclear waste management, it will be possible to submit to the safety authorities a concrete design integrating the specific characteristics of the selected zone for the implementation of the repository.

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UNDERGROUND CHARACTERISATION AND RESEARCH FACILITY ONKALO

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ABSTRACT

Posiva's repository for geological disposal of the spent fuel from Finnish nuclear reactors will be constructed at Olkiluoto. The selection of Olkiluoto was made based on site selection research programme conducted between 1987-2001. The next step is to carry out complementary investigations of the site and apply for the construction license for the disposal facility. The license application will be submitted in 2012. To collect detailed information of the geological environment at planned disposal depth an underground characterisation and research facility will be built at the site. This facility, named as ONKALO, will comprise a spiral access tunnel and two vertical shafts. The excavation of ONKALO is in progress and planned depth (400 m) will be reached in 2009. During the course of the excavation Posiva will conduct site characterisation activities to assess the stucture and other properties of the site geology. The aim is that construction will not compromise the favourable conditions of the planned disposal depth or introduce harmful effects in the surroundig bedrock which could jeopardize the long-term safety of the geological disposal.

1. Introduction

The geologic disposal of spent fuel from the Finnish nuclear power plants – Loviisa 1&2, Olkiluoto 1&2 and later Olkiluoto 3 – will be executed by Posiva Oy, a company owned jointly by the two nuclear power producers Fortum Oyj and Teollisuuden Voima Oy. Preparations for nuclear waste management were started already in the 1970s when the first power plants were still under construction. In 1983, the Finnish Government confirmed a target schedule for nuclear waste management, in which the construction of the disposal facility was scheduled for the 2010s and the start of actual final disposal for the year 2020.

Potential sites for the disposal of spent fuel were screened in the 1980s, followed by detailed site investigations in the 1990s and an environmental impact assessment in late 1990s. In 1999, Posiva submitted an application to the Government for a Decision-in-Principle to choose Olkiluoto as the site of the final disposal facility. After the mandatory local consent was received and a favourable safety statement issued by the regulator, the Finnish Government issued a Decision-in-Principle in favour of the project in December 2000. The Finnish Parliament approved the decision by 159 votes in favour and 3 against in May 2001.

The disposal project has progressed to the next stage – constructing an underground characterisation facility, known as ONKALO, at Olkiluoto. Work on the entire disposal project is progressing so that disposal can commence in 2020. ONKALO will be used to obtain further information to plan the repository in detail and to assess long-term safety and construction engineering solutions. ONKALO will also enable final disposal technology to be tested under actual conditions. ONKALO is not intended solely for research premises, but has also been designed to serve as an access route to the repository when constructed. ONKALO will take about 10 years to complete. Construction is scheduled for 2004–2014 and investigations will be made from the start of construction in conjunction with excavation. Once ONKALO has been completed, work will start on building the encapsulation plant and final disposal repository in the 2010s /1/.

2. ONKALO

The site characterisation programme already included the assumption that an underground rock characterisation facility would be required at the site confirmation stage to allow a detailed repository design to be developed and the preliminary safety assessment to be produced. The plans of the facility were realised after the Decision in Principle was issued. A decision was made to excavate the underground rock characterisation facility, ONKALO, at Olkiluoto. The approach adopted was that ONKALO facility shall be constructed in such a manner that it allows further characterisation and research work to be carried out without jeopardising the long-term safety of the repository site. In addition, it should be possible later to link the ONKALO to the repository so that they are integrated (Fig. 1).

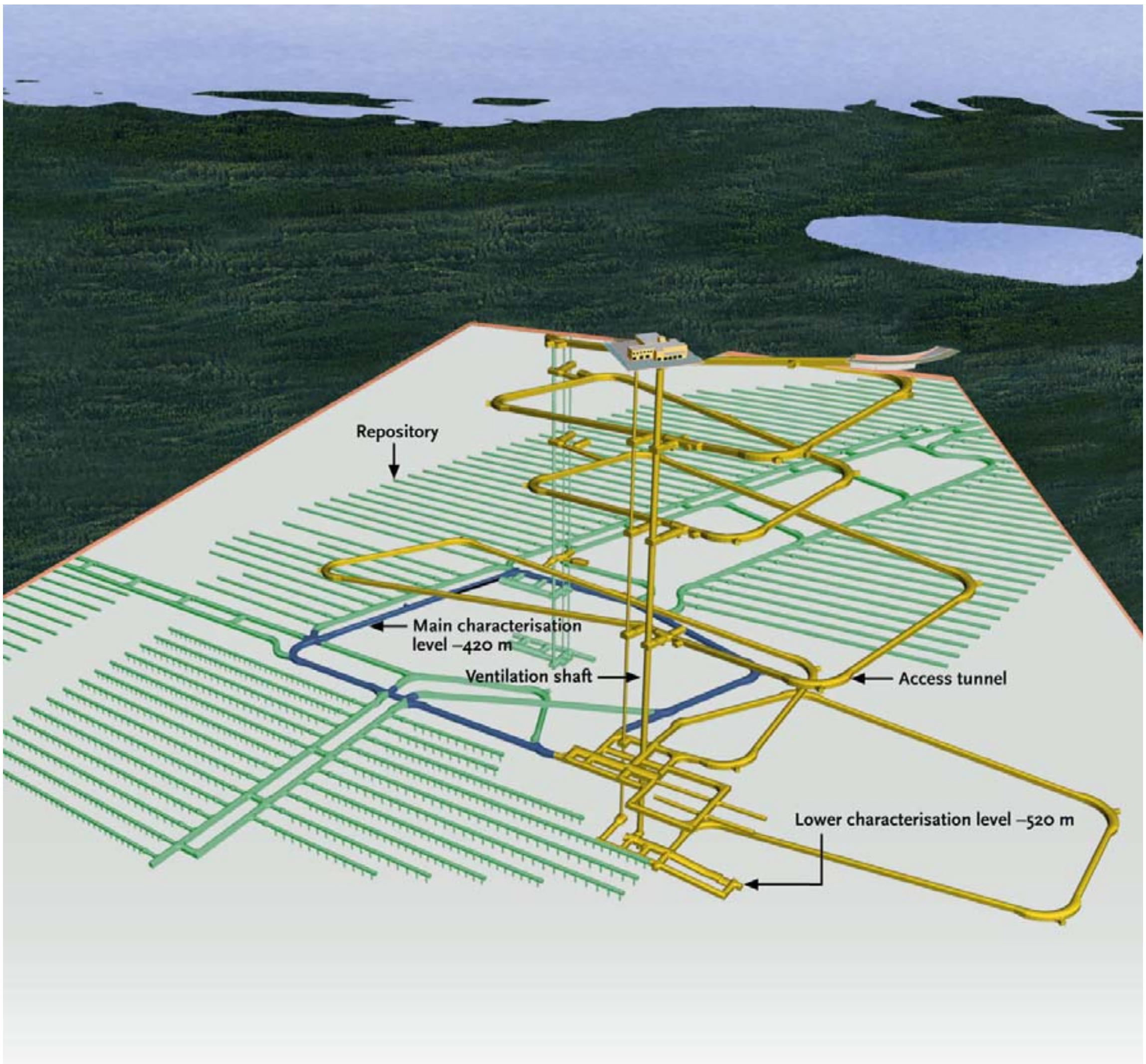

Fig. 1. *Concept design for deep repository in Olkiluoto. The ONKALO is marked with yellow.*

The location of the tunnel entrance is in the central part of the Olkiluoto island, some two kilometres away from the Olkiluoto nuclear power plant near the southern border of the existing site investigations area (Fig. 2). The location was decided based on comparison between a number of alternatives. In this comparison one of the main criteria was the anticipated disturbance to geological environment of the repository. In particular, the inflow of groundwater to the tunnel was to be kept to the minimum. After the systematic comparison of various alternative concepts, the decision was made in 2002 that the access to the repository depth would be provided by a combination of an access tunnel and a vertical shaft attached to it. The main aspects in favour of the combined tunnel-shaft concept were the increased flexibility as regards the planned future use of the facility as a part of the planned repository, the logistic benefits as well as the greater opportunities for characterisation work during construction.

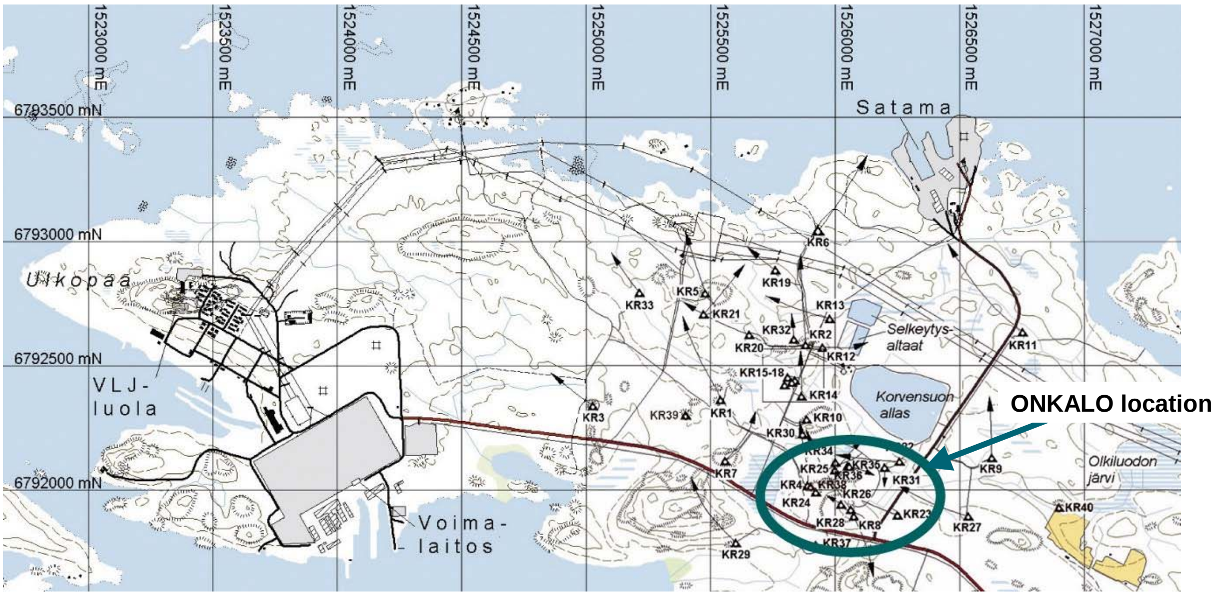

Fig. 2. Map of Olkiluoto site. Location of ONKALO is marked together with the location of deep investigation boreholes.

The present design of the ONKALO is presented in Fig. 1 The main characterisation level is at the depth of 420 metres below the sea level. The lower characterisation level is 100 metres beneath the main level. The inclination of the tunnel is 1:10, which means that the length of the access tunnel will be approximately 5.5 km. The total length of the tunnels and shafts will be about 9 km. A total of 365 000 m^3 of rock will be excavated.

The site preparations for the facility were started in 2003 and the actual excavation work began in September 2004. The tunnelling work is carried out by traditional drill & blast techniques. Raise boring method has been used for the first section of the exhaust shaft.

By the end of August 2006, the excavation of ONKALO had proceeded 1350 metres to a level of -126 m. The excavated tunnel meets the specified quality requirements, the management of leakage waters being one of the most significant requirements (Fig. 3). Due to the fractured nature of the surface rock, quite extensive grouting of the rock has been necessary.

The infrastructure of the site is almost completed. The concrete walls of the tunnel entrance, the washing hall, the fuel distribution station and the asphalting of the machine field and roads are completed. The site office has been built, the site perimeter has been fenced and site surveillance has been organised.

3. Underground characterisation and research

A programme for the underground characterisation and research (UCRP) to be carried out in the ONKALO has been established /2/. What Posiva aims at achieving with the activities proposed for the ONKALO is, of course, that the general suitability of the site will be demonstrated. It is only with such confirmation that it will be possible to proceed to the application for a construction licence for the repository. The programme during the tunnelling stage includes mapping of the tunnel faces, drilling of pilot and characterisation holes with subsequent rock mechanical, geological, geophysical and geohydrological studies, hydrogeochemical sampling and measurements, determination of fracture and flow data plus various rock-mechanical tests and measurements.

Investigations in ONKALO started in September 2004 at the same time with the excavation work by geological mapping of the tunnel roof and walls. The first stage of geological mapping is done right after excavation under last round and basic information about rock quality mainly for the rock support and grouting planning is collected. The systematic mapping follows the construction work about 100 to 200 meters behind the tunnel face. During this second stage the rock is investigated more carefully and this is actually the real rock investigation phase. Systematic bedrock sampling is also done for mineralogical studies.

Probe holes, which are bored to tunnel face after every fourth excavation round and pilot holes, which are core drilled boreholes inside the tunnel periphery makes possible to do different boreholes investigations during excavation work. From probe holes flow measurements are made and the results are used in the grouting planning and later also in the hydrogeological modelling. Geophysical logging, borehole imaging, flow measurements and groundwater sampling are standard investigations in the pilot holes. The results from the pilot hole drilling and related studies are used in excavation planning and in different modelling exercises. Results from the pilot hole studies are an important part of the prediction outcome studies. Prediction-outcome process is used to predict the rock conditions further along the tunnel line. These predictions assist in the further design of the facility and make an important part of the learning process built into the whole ONKALO programme. So far five pilot holes have been drilled and investigated.

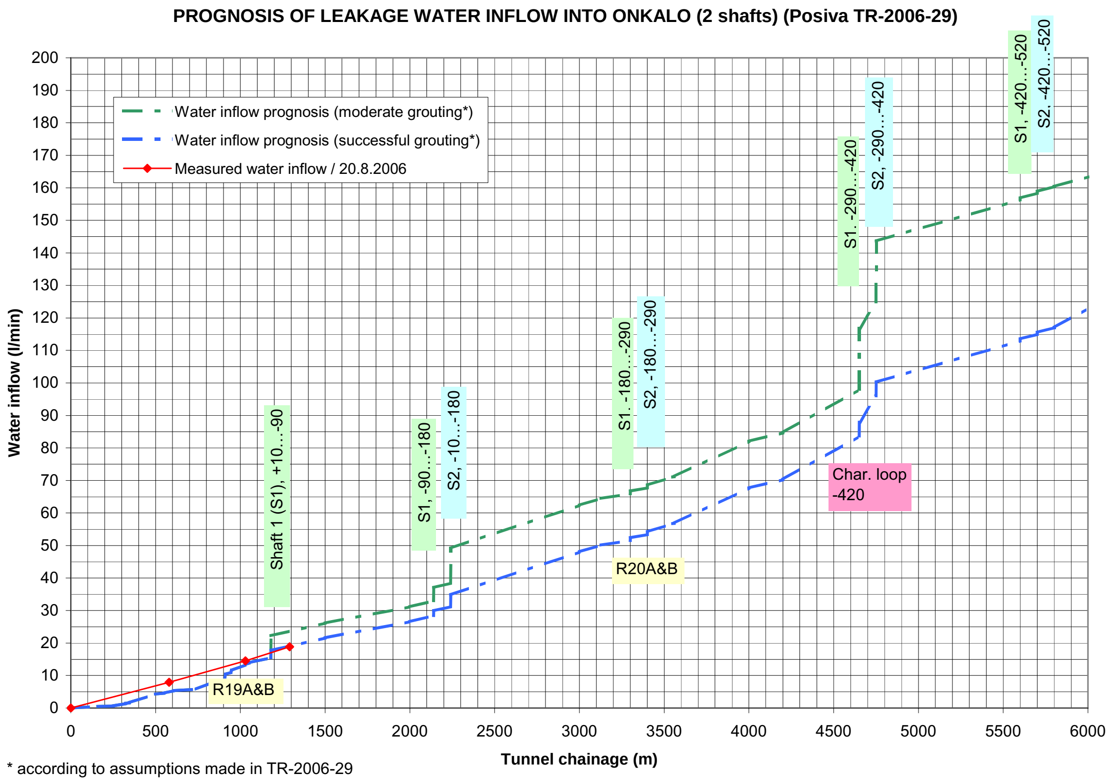

__Fig. 3.__ Prediction of leakage water into ONKALO (green, blue) and measured water inflow 20.8.2006 (red). Location of two major fracture zones (R19A&b, R20) are shown in yellow.

The fact that the ONKALO is planned to become a part of the repository means that it has to be designed and constructed according to the rules and requirements for nuclear facilities, for example, the quality assurance criteria posed by STUK. One very important issue from the long term safety point of view is to limit the groundwater inflow to the ONKALO. The amount of leaking water is followed regularly with flow mapping and with measurements done from the measuring weirs. Later on the measuring weirs will be put into the automatic monitoring system so the amount of leakage can be followed on line. At this moment the total leakage to tunnel is at chainage 1290 about 19 l/min (See Fig. 3). Also control of the excavation damages (EDZ) during excavation work is essential. Studies of the EDZ from the ONKALO tunnel wall, floor and roof started in spring and results will be reported by the end of 2006. At the same time procedures how to investigate EDZ are under development.

Groundwater quality is monitored to see the possible changes caused by the ONKALO construction. For this purpose permanent monitoring points will be put into the tunnel. Basically the monitoring point called groundwater station is core drilled borehole, which length is some tens of meters. Station has been equipped double packer system and with automatic monitoring equipment, which measures pH, electrical conductivity, dissolved oxygen and redox potential of the groundwater on line. By groundwater quality monitoring is also possible to investigate the influence and properties of the used cements. Experiment to investigate low pH cement has already started in ONKALO.

Long term experiments need to have place for the set up in the tunnel. For that purpose investigation niches, small tunnel ends, are planned to construct. The first one will be excavated during autumn 2006 and first experiments to start will be rock stress measurements and small-scale hydrogeological interference test. These tests will be followed with geochemical and geomicrobiological studies.

In parallel with the construction of the ONKALO the fieldwork at surface continues, consisting of deep drillings (43 deep drillholes by the end of 2006), groundwater sampling, geophysical and geohydraulic measurements, geological mapping and various monitoring networks. The fact that the site investigations are now focused on Olkiluoto makes it possible to employ new efficient methods for data gathering, e.g., investigations trenches, which nicely complement the lithological and fracturing data so far obtained only from the rather rare outcrops on the Olkiluoto island.

Because the construction of the ONKALO and later final disposal facility will affect the surrounding rock mass, a monitoring programme was established /3/. During ONKALO construction rock mechanical, hydrological, chemical and environmental monitoring is carried out both from surface and from ONKALO and the quality and quantity of foreign materials used in ONKALO are measured and registered. The main aim is to observe possible changes in the host rock and obtain information on responses of the host rock to the excavation. The results of the monitoring are compared to the baseline values presented in 2003 /4/.

All the data achieved with different investigation methods will pass several different modelling steps. The interpretation and modelling of the field investigations data aim at building a consistent picture of the site. A special effort is made to integrate the differerent disciplines of the site knowledge by a specific Olkiluoto Modelling Task Force. The purpose of the Task Force work is to coordinate the work of different disciplines in such a way that a coherent picture of the site can be produced.

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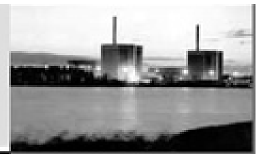

POSTER SESSION

MODELLING OF RADIONUCLIDE RELEASES FROM THE NEAR FIELD OF THE GEOLOGICAL REPOSITORY IN CRYSTALLINE ROCKS FOR RBMK-1500 SPENT NUCLEAR FUEL

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ABSTRACT

During 2002-2005 the assessment of possibilities for disposal of spent nuclear fuel (SNF) in Lithuania was performed with support of Swedish experts. Potential geological formations for disposal of SNF were selected, disposal concept was developed and preliminary generic safety assessment was performed. Performing safety assessment the analysis of radionuclides migration from the repository as well as their impact to human and environment were also very important issues.
In this paper results on the analysis of the radionuclide releases from the near field of the geological repository for RBMK-1500 SNF in crystalline rocks in Lithuania are presented. Radionuclide migration in the near field region was assessed using computer code COMPULINK7 (SKB, Sweden) The results of analysis show that most of safety relevant radionuclides of RBMK-1500 SNF are effectively retarded in the near field region. The release is dominated by ^{129}I and ^{59}Ni firstly while after app. 50 thousand years ^{226}Ra is dominating. The sensitivity analysis was also performed for the parameters that have direct influence on releases.

1. Introduction

During 2002-2005 the assessment of possibilities for disposal of the SNF in Lithuania was performed with the support of Swedish experts. Potential geological formations for disposal of the SNF were selected, disposal concept was developed and preliminary generic safety assessment was performed. Disposal concept for RBMK-1500 SNF in crystalline rocks in Lithuania is based on Swedish KBS-3 concept with SNF emplacement into the copper canister with cast iron insert. The bentonite and its mixture with crushed rock are also foreseen as buffer and backfill material. Detailed description of the disposal concept for RBMK-1500 SNF in crystalline rocks in Lithuania is presented in the paper presented to this conference [1].

The radionuclide transport analysis was performed for the RBMK-1500 SNF with initial enrichment of 2.8 % ^{235}U and 0.6 % Er_2O_3. The SNF burnup is app. 29 MWd/kgU, the radionuclide inventory was assessed by computer code SAS2H (computer code system SCALE 5) [2]. Radionuclide transport calculations were performed using computer code COMPULINK7 (Sweden) for the safety relevant radionuclides for RBMK-1500 SNF that were identified in [3]. For sensitivity analysis, each parameter value influencing the radionuclide release directly was varied by its variability factor and the relative change in the maximum release value indicates the sensitivity of releases to parameter uncertainties.

2. Processes in the near field

The evolution of a failed canister is complex and depends on a number of uncertain factors. Water is likely to intrude into the canister, causing corrosion of the cast iron insert with hydrogen gas generation. The build-up of gas pressure in the canister can be considerable and lead to the suppression of further water entry and also to gas release through the buffer. As corrosion proceeds, corrosion products will exert mechanical pressure on the copper canister, corrosion also causes a weakening of the cast iron insert and this could also lead to expansion of defects [4]. The evolution will also be influenced by external factors like the external mechanical load on the canister and by the thermal conditions. When the initial defect becomes larger water can also enter freely and a continuous water pathway is expected to form. The materials of fuel claddings are stable in water and corrode only very slowly, but microscopic cracks could develop eventually, forming passages through the tube walls. As there is some current discussion as to how much short term credit can be taken for the tubes as part of the multibarrier system, usually it is considered that the lifetime of the cladding is very short under anoxic repository conditions and the claddings do not provide a barrier for the transport of radionuclides released from SNF matrix. As the groundwater comes into the contact with SNF its alteration/dissolution processes and the release of radionuclides begin. Key parameters influencing SNF matrix alteration/dissolution are specific activity of SNF, primary radiolytic yields, temperature, specific surface of SNF, iron and H_2 concentration, concentration of carbonates, initial oxidation state of SNF matrix, etc. [5]. Depending on the heterogeneous distribution of radionuclides in the structure of SNF their release to groundwater is classically described by the contribution of two terms: an instantaneous and a slow long-term release of the radionuclides. As SNF matrix degrades, the radionuclides embedded in it are congruently released into the volume inside the canister. Solubility limits for dissolved radionuclides are applied and in case of radionuclide concentration exceeds solubility limit they start to precipitate. The release of dissolved radionuclides from the waste package by diffusion and/or advection depends on the conditions to be expected in the repository as well as on the properties of engineering barriers. Buffers based on the clay have very low permeability to water flow. The advective transport of the dissolved radionuclides is generally considerably smaller than that due to molecular diffusion. Dissolved radionuclides will diffuse in the water existing in the clay pores and could interact with the pore surface. Sorption term is general and includes contributions from all heterogeneous reactions of dissolved contaminants with solid surfaces: chemisorption and physisorption, precipitation, as well as ion exchange and isomorphic substitution.

3. Modelling of radionuclide transport

The radionuclide transport calculations were performed for the canister defect scenario. Taking into consideration the similarities in the repository environment and repository concept, the selection of scenario is based on experience from the safety assessment performed in Sweden [6]. For Lithuanian case it is assumed that one canister out of total 1400 will pass through quality inspection with penetrating defect of 1 mm^2 in size. For this preliminary study only the main processes determining the repository safety in the case of SNF disposal in the crystalline rocks are taken into account. Specific features and processes related to KBS-3H design indicated in [7] (e. g. a loss or redistribution of buffer mass, eventual gas bubble transport and transport of volatile radionuclides, the accumulation of gas along the top of the drift and its effects on groundwater transport, the eventual chemical alteration of bentonite, effect on rhelogical properties, swelling and hydraulic conductivity of bentonite due to steel components) have not been addressed in this preliminary study.

Due to very low permeability of the bentonite buffer, the transport of released from SNF matrix and dissolved radionuclides in the near field is going to be diffusion dominated. For radionuclide transport assessment the methodology presented in [8] was used. According to this methodology the barrier system is discretized into compartments. The material balance over the compartment connected to some other compartments for n radionuclide is expressed as follows:

$$V_i \varepsilon_i R_i^n \frac{dC_i^n}{dt} = V_i \cdot S^n + V_i \varepsilon_i R_i^{n-1} C_i^{n-1} \lambda^{n-1} - V_i \varepsilon_i R_i^n C_i^n \lambda^n - \sum_j \left(\frac{AD_e}{d} \Delta C \right)_{i,j}^n \quad (1)$$

where V - compartment volume (m^3), C - radionuclide concentration (mol/m^3), ε - material porosity (m^3/m^3), R - retardation factor (-), S - general source term, λ - decay constant (1/yr), A – compartment cross section area (m^2), D_e - diffusivity in material (m^2/yr), d - diffusion length (m). Mass transfer of dissolved species from the stagnant porewater in the bentonite into the groundwater flowing in a fracture intersecting the deposition tunnel is limited by the boundary layer resistance. The assessment of this transfer is handled through the fictitious equivalent flow rate Q_{eq} [9]:

$$Q_{eq} = 2\pi \cdot r_2 \cdot 2b_v \sqrt{\frac{4D_w u}{\pi^2 \cdot r_2}}, \quad (2)$$

where $2b_v$ is the volume aperture of the fracture (m), r_2 is the radius of the deposition tunnel (m), D_w is the diffusivity in water (m^2/yr), *u* is the velocity of water in the fracture (m/yr).

As there are no yet an experimental data on RBMK-1500 SNF dissolution under reducing conditions as well as data on instant release fraction for some nuclides, the data compiled for other UO_2 type SNF were used instead. Based on the data available in [4, 10, 11] the rate of SNF alteration/dissolution rate of 10^{-7}/yr is assumed as reference value. The structural parts of SNF assembly are assumed to be fully corroded and radiouclides only present in structural parts of SNF assembly are assumed to be fully available. It is also assumed that after the defect becomes larger the entire void in the canister at closure, approximately 0.5 m^3, will be filled with water and radionuclides released instantaneously or during SNF alteration/dissolution dissolve in the water inside in the canister. If the concentration of nuclides exceeds its solubility limit, they start precipitate. Sorption on the internal parts of the canister is neglected. Thus the solubility of radionuclides limits the concentration in the canister and the release rate of the species escaping from the canister. The solubility depends on the composition of water entering the canister which is influenced by chemical processes in the buffer material and inside the canister as well. Solubility limits may also be altered by possible changes in the redox potential in the repository by radiolysis of the water entering the canister. Since the solubility limits also depends on the conditions in the repository environment and they have not been determined for Lithuanian repository yet, the radionuclide solubility values were used from [6]. Because of the conditional nature of diffusivity, sorption coefficients, porosity, particularly of K_d, they have to be determined for the conditions expected to be relevant for the repository environment. Since these site specific parameters have not been determined yet it was assumed that the conditions of the repository environment is similar to that for other granitic rocks. Data on diffusivity, sorption that were derived based on systematic datasets available in the literature and/or on thermodynamic models and presented in [12] were used in the calculations. In the present analysis the hydraulic gradient around the deposition tunnel is assumed to be 1 % after the closure of the repository and a reference fracture intersecting tunnel is assumed to have a transmisivity of 10^{-8} m^2/s and a volume aperture of 250 μm based on [9].

4. Results

As could be seen in Fig. 1, most of identified safety relevant radionuclides for RBMK-1500 SNF disposal are effectively retarded in the near field region. The release to the far field is dominated by ^{129}I and ^{59}Ni firstly while after app. 50 thousand years ^{226}Ra is dominating. ^{226}Ra is formed in the decay chain of ^{238}U and it is more mobile than its predecessors. Detailed study of the output shows that the concentrations of the following nuclides reach their solubility limits in the canister: ^{242}Pu, ^{239}Pu, ^{238}U, ^{236}U, ^{235}U, ^{234}U, ^{233}U, ^{232}Th, ^{230}Th, ^{229}Th, ^{237}Np, ^{231}Pa, ^{79}Se, ^{93}Zr, ^{99}Tc, ^{107}Pd, ^{126}Sn.

The sensitivity analysis was also performed for the parameters that have direct influence on releases (SNF matrix dissolution rate, solubility, diffusivity, sorption coefficient). For the parameter sensitivity analysis the same methodology as presented in [13] is applied. Firstly the radionuclide release from the near field region is assessed using the reasonable values of parameters. Secondly, by varying each parameter individually by a factor according to its variability, the relative change in the performance of the reference case is calculated as the ratio of maximum release rates. For the sensitivity analysis few radionuclides have been chosen, considering the half-life, the capacity of sorption and diffusion in the backfill material (^{135}Cs, ^{129}I, ^{239}Pu, ^{238}U, ^{59}Ni, ^{226}Ra).

The lower and upper values of sorption and diffusivity coefficients reported in [12] indicate a large range for some radionuclides (e. g., actinides). A variability factor of about 5 is selected to test the parameter sensitivity. Since the range of solubility variation is of some orders of magnitude, a rough estimation of the uncertainty in the solubility data may result in the variability factor about 10 [13]. The equivalent flow rate is assumed to vary by factor 5 that could correspond to the increase of the fracture transmissivity by one order of magnitude.

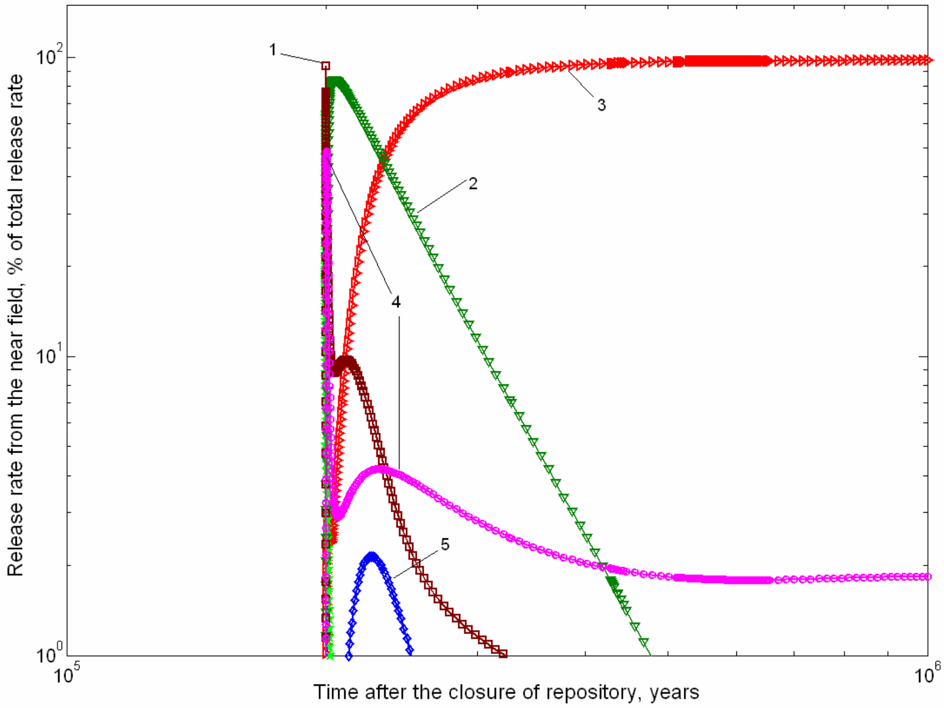

Fig. 1. Release rate from the near field (1 - ^{129}I, 2 - ^{59}Ni, 3 - ^{226}Ra, 4 - ^{135}Cs, 5 - ^{94}Nb)

Uncertainties in the fuel matrix dissolution have influence on the release of radionuclides whose concentration in the canister is not controlled by solubility limit (maximum release rate increased 4 times for ^{226}Ra). For radionuclides with instant release fraction the significance of increased fuel matrix dissolution depends on what (instant release fraction or congruent release from SNF matrix) governs the maximum release from the near field. In this reference case, increased fuel dissolution rate by factor 10 results in increased maximum release from the near field for cesium (about 4 times), while for iodine the release caused by instant release fraction (IRF=2%) becomes only comparable with part congruently released from the SNF (the maximum release increased only app. 24 %). For radionuclides whose release from the canister is controlled by solubility, the uncertainties in solubility limit have significant influence on the release (maximum release increased about 10 times for ^{238}U and about 7 times for ^{239}Pu).

The influence of uncertainties in the radionuclide diffusion coefficient is more significant when the concentration of dissolved radionuclides in the canister remains constant (e. g. limited by solubility). In this case decreased buffer resistance and increased concentration gradient results in higher release from the near field (5 times for ^{238}U, and app. 250 times for ^{239}Pu in this study). In other case increased diffusivity could result in increased release to the far field until it is controlled by buffer resistances and not by the concentration of radionuclides available to migrate (as for Ra). As diffusivity for iodine is low compared with the diffusivity for cesium, increased iodine diffusivity had an influence on increased its release from the near field. But this increase is less significant (maximum release of iodine increased 3 times) than for solubility limited radionuclides, as more rapid release from the canister results in decreased concentration and smaller concentration gradient.

Uncertainties in the sorption coefficient have a strong influence on the release of more sorbing radionuclides. Due to increased sorption sorbing radionuclides remain longer in the near field region and they have more time to decay to insignificant concentrations. Sorption capacity of the bentonite surrounding the canister and the hydraulic properties of fractured (through equivalent flow rate) rock nearest to the deposited canister have to be accurately determined in order to minimize uncertainties in the release to the far field. In case of increased sorption the maximum release of radionuclides decreased 10^1-10^4 times, in case of increased fracture transmisivity (subsequently Q_{eq}) the maximum release of the radionuclides increased app. 3-4 times. On the other hand, for nonsorbing long-lived iodine, with low diffusion coefficient uncertainties in the hydraulic properties have a very small effect on the release.

5. Summary

Preliminary analysis of radionuclide transport from possible repository for disposal of SNF in Lithuania was performed. The main features and processes important for radionuclide migration from the repository in case of SNF disposal in crystalline rocks were analyzed and available RBMK-1500 specific data were used. The possible impact of the specific features and processes related to KBS-3H real design were not taken into account.

The results of radionuclide transport analysis show that the most of identified safety relevant radionuclides for RBMK-1500 SNF disposal are effectively retarded in the near field region. The release to the far field is dominated by ^{129}I and ^{59}Ni firstly while after app. 50 thousand years ^{226}Ra is dominating. The sensitivity analysis of the results obtained was also performed.

6. Acknowledgments

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ALTERATION OF THE CEMENTITIOUS MATERIAL UNDER THE SALINE ENVIRONMENT

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ABSTRACT

Leaching experiment of ordinary and fly ash mixed cement were carried out by using the artificial sea water and deionized water as leachates. The pH values of saline leachates were decreased at the lower solid/liquid ratio in comparison with the case of deionized leachates. $Mg(OH)_2$ and ettringite were observed only in the case of saline water. The results of the geochemical calculation with inputs of $Mg(OH)_2$, ettringite and hydrotalcite well represented in the change of pH value and the mineral-composition with those of the experiment. However, in the latter stage of the fly ash mixed cement case, calculated concentrations of Si and Al of the leachate weren't consistent with the observed values. These results showed that substitution of Mg for Ca and influences of sulfate ion should be considered in the modeling of the cement alteration in the saline water case, and Al-substituted C-S-H might be considered in the alteration of fly ash mixed cement.

1. Introduction

Because candidate site for geological disposal is not yet selected in Japan, the applicability to wide range of disposal environment, such as salinity of the ground water, of every material and those assessment models for EBS system should be demonstrated. Many researches on the alteration of the cementitious material have been carried out and some serviceable alteration models are introduced [1]. Sugiyama's model [2] has been used in the 2nd progress report of TRU-waste disposal in Japan [3]. However, almost of these alteration models are considering only with fresh groundwater but with saline groundwater. Because the alteration of the cementitious material arise the alteration of other barrier such as bentnite buffer and host rock, the validation of the applicability of these alteration models for saline environment is necessary to use the geochemical simulation or performance assessment. However, such a validation of cement alteration for saline environment hadn't been carried out.

Because, form the point of view of the chemical durability, mixed cement such as fly ash mixed cement (FAC) may be used for the construction of the disposal vault as liner, structure, filling material,

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grout and so on, the alteration model for mixed cement is also needed.

In this study, we report the results of a trial to present the applicability of alteration model of ordinary portland cement (OPC) for the geochemical simulation of alteration of FAC and of the batch alteration experiment of OPC and FAC in the artificial sea water (ASW). Results of the geochemical simulation under saline environment reflected the results of the experiment are also reported.

2. Experiment

2.1 Preparation of hydrated cement pastes

Because re-saturation of groundwater will take long time in the actual disposal facility, the cementitious materials will be well hydrated before beginning of the alteration cause by the reaction with groundwater. Therefore, we made well hydrated cement paste by the hot water curing to accelerate the hydration. Because the hydrated minerals in the high temperature are different from those in the lower temperature, the curing temperature was set 323K by following the previous research [4]. Fly ash/OPC ratio in FAC was 30wt%. Cement and water were mixed (water/cement ratio=0.6) and cured for 7 days. The hardened cement were crushed into 5mm or smaller diameter. The crushed cement were dried under vacuum to make easy to fine grinding. The dried crushed cement grinded into 250 μm. Thereafter, the powdered cement were cured again for 91days with water to hydrate sufficiently. The hydrated cement pastes were dried in vacuum and grinded into 100 μm.

2.2 Leaching experiment

Immersion experiment [5, 6] and flow-through experiment [7] used to be carried out to evaluate the degradation mechanisms of cementitious materials. We chose the immersion method with various water/solid ratio to identify the minerals in the hydrated and degraded cement in equilibrium with groundwater. Hydrated OPC and FSC pastes were put into PTFE bottles. Deionized water (DIW) or ASW were put into the bottles with water/cement ratio of 10, 100, 200, 1000 and 2000 (cm^3-water/g-cement). The composition of the ASW is shown in Table 1. These process of immersion were carried out in a globe box of Ar atmosphere and these bottles were stored in the globe box. Immersion period was set 6 months.

species	pH	Na	Ca	K	Mg	Al	C	S	Cl	Si
concentration (mol/litter)	8.2	$0.45*10^{-1}$	$9.4*10^{-3}$	$1.9*10^{-2}$	$5.1*10^{-2}$	$1.8*10^{-6}$	$2.4*10^{-3}$	$2.9*10^{-2}$	$5.3*10^{-1}$	N.D.

Table 1: Composition of artificial sea water

2.3 Analyses

Identification of minerals in hydrated and degraded cement pastes was carried out by powder X-ray diffraction method (XRD). The pH values of leachate of every bottle were measured by standard glass electrode. Concentration of Ca, Si, SO_3^{2-}, Mg, Fe, Na and K were measured by ICP method. Before the ICP measurement, the leachate were filtered by using membrane filters of 45 μm under the Ar atmosphere.

3. Geochemical simulation

3.1 Preconditions

Geochemical calculation to simulate the composition of minerals in the degraded cement and species in the leachate was carried out by using PhreeqC [10]. Thermodynamic data of the primary and secondary minerals was set by JNC-TDB.TRU [11]. Atkinson's alteration model [12] of C-S-H system was selected to calculate C-S-H dissolution[13]. List of the minerals considered in calculation were shown in Table 2.

	Primary minerals	Secondery minerals
All Case	$Ca(OH)_2$, C_2AH_6, ettringite, C-S-H, Na_2O, K_2O (Na_2O and K_2O was considered as the liquidous ion)	C_3ASH_4,C_2ASH_8,Caolinite, Pyrophilite, Monosulfhate, Calcite, Analcime, Roemontite
Set in ASW case		Friedel's Salt, $Mg(OH)_2$
Set as the parameters for calclations		C_4AH_{13},C_4AH_{19}, Sepiolite, Hydrotalcite

Table 2 List of the minerals considered in the calculation

	OPC-DIW					OPC-ASW				
Liquid/Solid ratio	initial	10	200	1000	2000	initial	10	200	1000	2000
Portlandite	○	○	△			○	○			
Brucite							○	○		
C-S-H gel	○	○	○	○	○	○	○	△	△	△
Gypsum							○	○		
Ettringite	○	○	○	○		○	○	○		
Monosulfate										
Katoite	○	○	○	○	○	○	○	△		
	FAC-DIW					**FAC-ASW**				
Liquid/Solid ratio	initial	10	200	1000	2000	initial	10	200	1000	2000
Portlandite										
Brucite										
C-S-H gel	○	○	○	○	○	○	○	△	△	△
Gypsum										
Ettringite		○	○				○			
Monosulfate	○	○				○				
Katoite	○	○	○	○	○	○	○			

Table 3 Results of the XRD experiment

○ :Well Detected △ :Slightly detected Blunk:Not detected

4. Results and discussion

Table 3 shows the dominant results of XRD analyses. In the initial system of FAC, C-S-H gel, monosulfate and katoite were observed as hydrated minerals and mullite and quartz were observed as inert minerals. Because of lower sulfate concentlation in the FAC system, monosulfate seemed to be formed instead of ettringite formation. In addition, the minerals in the alteration of FAC were almost the same as those of OPC, except the absence of portlandite and ettringite. This result suggests that the alteration mechanism of FAC might be the same as that of OPC. Precipitation of brucite ($Mg(OH)_2$) and dissolution of portlandite ($Ca(OH)_2$) and C-S-H gel were detected in the early stage of the alteration of OPC-ASW case. The color of the cement paste was changed into yellow by the degradation in ASW. Figure 1 shows the change of the pH values of leachates. In the ASW case, the pH value fell under 11 in the small liquid/solid ratio in comparison with IEW case. The value of pH at the equilibrium of brucite is significantly lower than that of portlandite. Figure 2 shows the change of Ca and Mg concentration of the leachate. Ca concentration in the ASW cases were higher than those of DIW cases in all stages of alteration. These result seem to show that Mg substitutes to Ca of portlandite in the hydrated cement

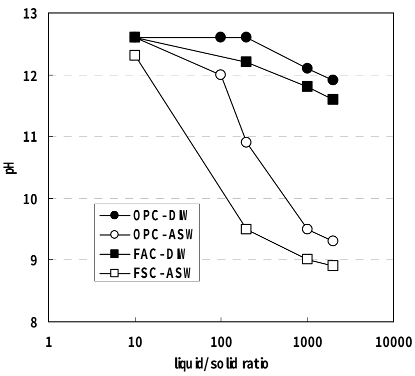


Fig.1 Change of pH of leachate with liquid/solid ratio

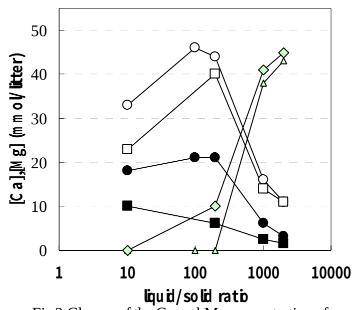
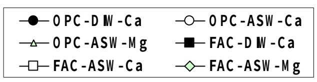
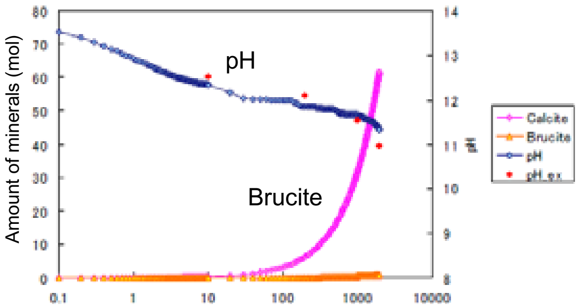


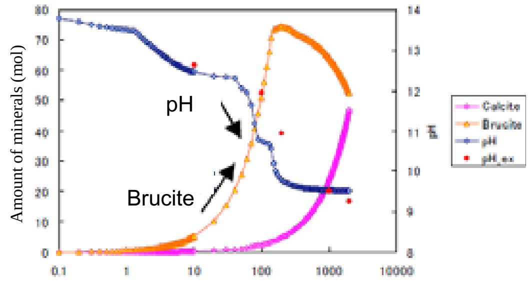
Fig.2 Change of the Ca and Mg concentration of leachate





Liquid/solid ratio

Fig.3 Results of the geochemical simulation of alteration of FAC in DIW



Liquid/solid ratio

Fig.4 Results of the geochemical simulation of alteration of OPC in ASW

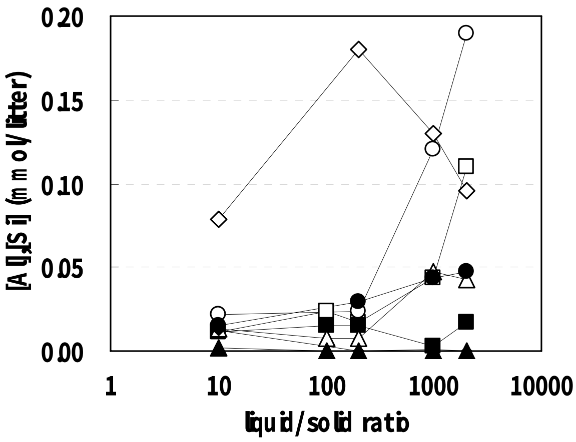
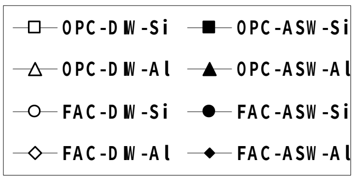
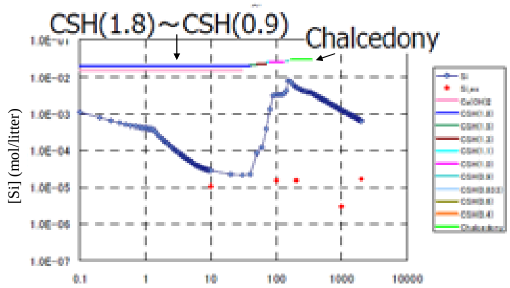


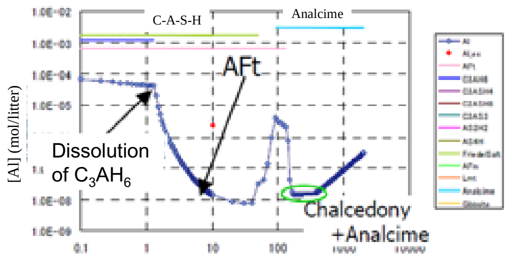
Fig.5 Al and Si concentration of leachate





Liquid/solid ratio

Fig.6 Results of the geochemical simulation of alteration of OPC in ASW (change of Si concentration)



Liquid/solid ratio

Fig.7 Results of the geochemical simulation of alteration of OPC in ASW (change of Al concentration)

system in early stage of alteration and, in the latter stage, C-S-H gel would attacked by Mg. Precipitation of gypsum observed in the OPC-ASW might be caused by the SO_4^{2-} released from ettringite by Cl substitution. Figure 3 shows the results of the calculation of the change of pH value in the case of FAC-DIW. In this case, the change of pH value (experimental value were shown as the closed dots in those figures) and the precipitation-dissolution behavior of brucite might be well represented. From this, in the DIW case, the applicability of the alteration model of Ca-Si-H system for OPC thought to be applicable to the FAC alteration. Figure 4 shows the results of the calculation of pH and Brucite concentration in OPC-ASW case. These phenomena were also well represented in ASW case. Figure 5 shows Si and Al concentration. The increase of the Al concentration in the DIW cases would be dissolution of ettringite. In the ASW case, precipitation of gypsum or other sulfo-alminate phase might cause the lower concentration of Al. Figures 6 and 7 show the calculation results of Si and Al concentration in the OPC-ASW case. Calculated values of Si and Al were different form those of experimental values. The difference is thought to be caused by the imperfection of the model for incongruent dissolution of sulfo-almino-silicate system such as ettringite, monosulfate and hytrotarcite. In addition, because there are no thermodynamic databases for Al substituted C-S-H gel, dissolution of such C-S(A)-H weren't considered in these calculations. Therefore dissolution behavior of Al and Si in the latter stage that such C-S(A)-H would dissolve might not be able to represent accurately.

5 Conclusions

From the results of the leaching experiment of OPC and FAC using the ASW and DIW and geochemical calculation of cement-water reaction, followings are suggested.

a. The alteration model of Ca-Si-H system for OPC thought to be applicable to the FAC alteration
b. In the early stage of the alteration, the alteration calculation considered the Mg containing secondary mineral such as brucite will be applicable.
c. Thermodynamic data for Al substituted C-S-H gel might be necessary to accurately calculate the latter stage of cement alteration in the saline water.

Acknowledgment

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UPGRADING OF RADON'S-TYPE NEAR SURFACE REPOSITORY IN LATVIA

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ABSTRACT

In 1959, the Soviet government decided to construct the near surface radioactive wastes repository "Radons" near the Baldone city. It was put in operation in1962. The changes in the development of the repository were induced by the necessarity to upgrade it for disposal of radioactive wastes from the decommissioning of the Salaspils Research Reactor (SRR). The safety assessment of repository was performed during 2000-2001 under the PHARE project for necessary upgrades of repository. The outline design for new vaults and interim storage for long lived radioactive wastes was elaborated during 2003-2004 years. The Environmental Impact Assessment (EIA) for upgrade of Baldone repository was performed during 2004-2005 years. It was found, that additional efforts must be devoted for solution of social aspects o for successful operation and upgrade of repository. It was shown by EIA, that the local population has a negative opinion against the upgrade of repository in Latvia. The main recommendations for upgrades were connected with increasing the safety of repository, increasing of PR activities for education of society and developing of compensation mechanism for local municipality.

1. Introduction

The national radioactive wastes repository "Radons" is located in Baldone site near the capital of Latvia – Riga. It was put into operation on 1962. The repository was originally built according to former USSR design as a near surface "Radons" –type repository with common vaults.

Since 1995, after introduction of new technology with the possibility of retrieval of containers with radioactive wastes, the new 7-th vault was put in operation.

In May 16 of 1995, the Cabinet of Ministers had made the Order No. 263 to shut down the Salaspils Research Reactor. SRR was shutdown in June 19 1998. According to the Order No. 57 of Cabinet of Ministers in October 26 1999, this accepts the option to direct dismantling of SRR to "green field", the upgrade of national radioactive wastes repository was initiated.

The national strategy for radioactive wastes management development comprises a series of 13 actions (together with budgetary implications) that should guarantee safe management of radioactive waste in Latvia up to 2010, hence up to the complete dismantling of the Salaspils research reactor. This strategy largely relies on the recommendations of the EC-funded study that was completed in 2001 [1], as well as several studies for decommissioning of Salaspils research reactor [2-4] The decision of the Government of Latvia in 26 June 2003 defined to start the upgrade the Baldone repository.

The outline design for additional vaults and interim storage for long lived radioactive wastes was elaborated during 2003 – 2004 years under EC-funded project. To fulfil all demands of national regulations, the EIA studies were performed during 2004-2005 years.

2. The short description of repository

Radioactive wastes repository "Radons" occupies 7 ha territory. It consists on 2 parts –supervision part and control area with the vaults (Fig. 1). The environmental laboratory, decontamination building, garage building is located at territory of repository. The emergency group of the Hazardous wastes management state agency is based on the infrastructure of the repository. There are 7 vaults at the control area of Baldone repository. Three of them are concrete, underground 200 m^3 vaults (1, 3, 6), 2 - concrete underground 40 m^3 vaults (4, 5) and one vault is a 200 m3 stainless steel underground

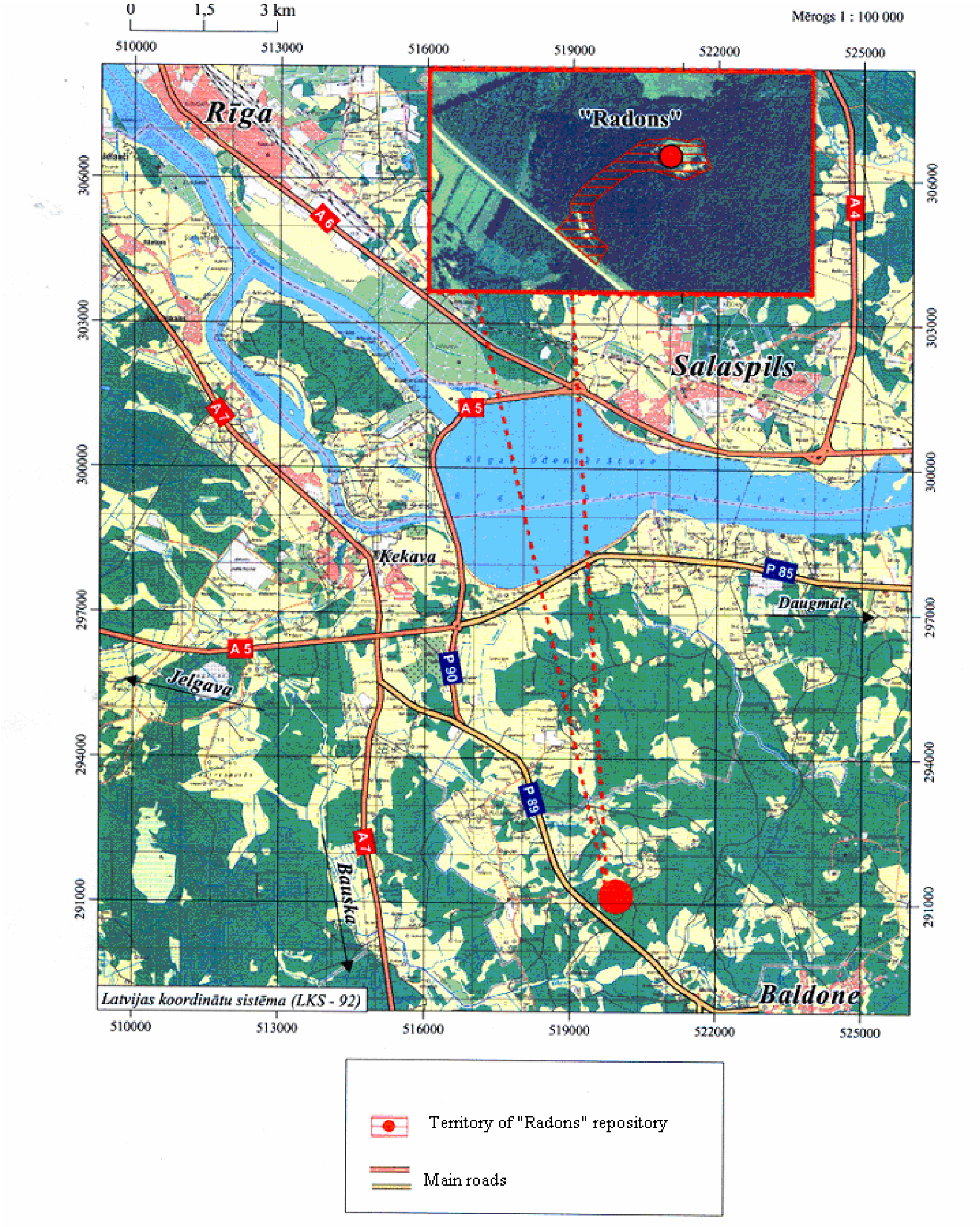

Fig.1. Radioactive wastes repository "Radons".

tank used for liquid waste (2), but now the waste from it is removed and the tank was cleaned up. As vaults for solid waste were filled up, a new 1200 m^3 vault was constructed (7). It is in maintenance from the end of 1995. The main functions of repository are following:

1. safe maintenance of disposed and temporary stored radioactive wastes with total amount 400 TBq;
2. transportation of radioactive wastes at territory of Latvia;
3. decontamination of contaminated materials, soil and different objects including vehicles;
4. management of accidents with radioactive and hazardous materials;
5. implementation of area monitoring programme.

3. Development of repository during 2000- 2005 years

It was shown [1-6], that the decommissioning of Salaspils research reactor causes significant changes in radioactive wastes management system of Latvia. The following upgrades were performed at repository:

1. Security systems (2002-2004);
2. Radiation protection upgrades (2003-2004);
3. Upgrade of the 7-th vault 2003-2004);
4. Transport systems upgrades (2003-2005);
5. Radioactive wastes packages upgrade, including tests (2000-2004);
6. Emergency group upgrade (2004-2005).

The following studies were performed for improving of radioactive wastes management system in Latvia and Hazardous wastes management agency:

1. Safety assessment for planned upgrades of capacity of repository – PHARE project (2000-2001);
2. Preparation of outline design for additional vaults and interim storage of long lived radioactive wastes- PHARE project (2003-2005);
3. Environmental Impact Assessment studies for upgrade of repository (2004-2005) [5].

4. Interactions with the local municipality.

Operational activities of repository are connected with the interactions with the local municipality of Baldone. The attention of population of Baldone municipality to repository is shown on Fig. 2.

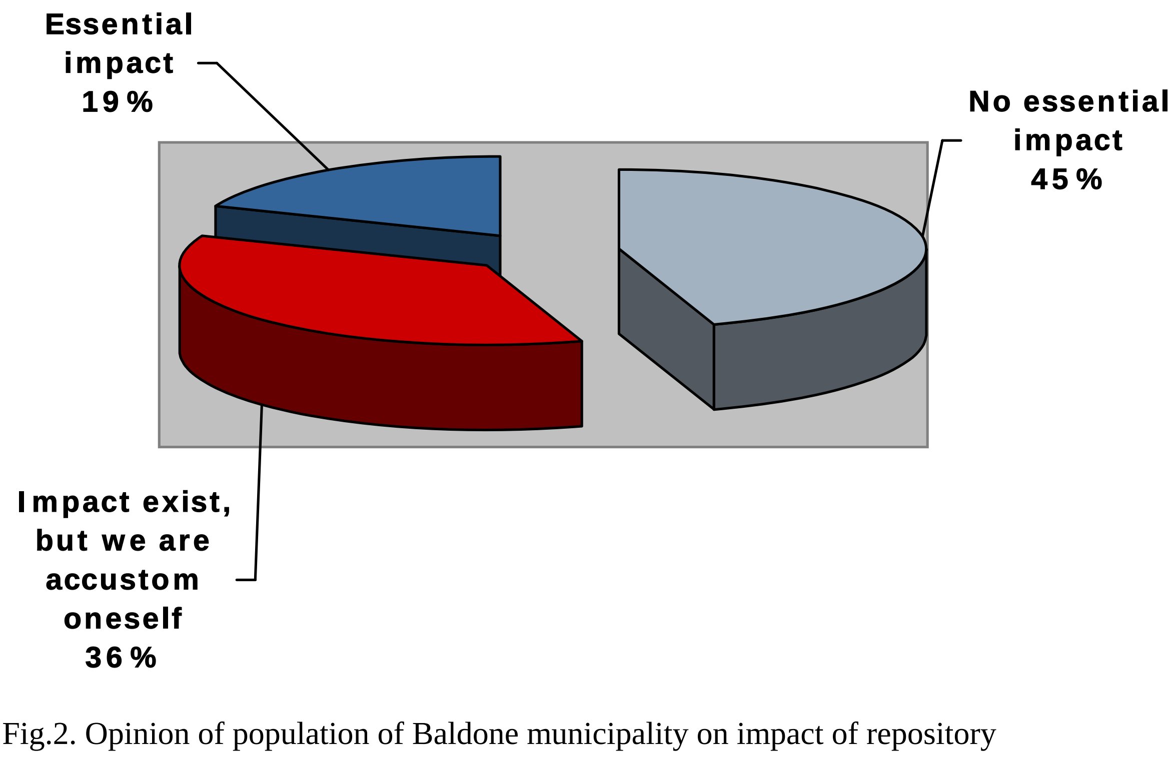

Fig.2. Opinion of population of Baldone municipality on impact of repository

The main problems according to point of view of population are summarized in the Table 1. The point of view of population was clarified during studies of EIA and presentation of outline design for upgrading of repository's "Radons" site, which is connected with construction of two additional vaults and one temporary storage for long lived radioactive wastes, which cannot be acceptable (do not fulfill waste acceptance criteria) for disposal in repository. The table indicates the main factors referred by population which can be related with the repository.

Table 1

Main problems created by repository for population of Baldone municipality (opinion of population).

Factor	Value,%
Impact on health	51
Unclear impact of radiation	31
Psychological discomfort	18
Impact on nature and animals	12
Lack of information on repository	10
Impact on economy	8

EIA studies show that about 62 % of population is against to upgrade of repository. Main reasons are connected with “fear factor”, leak of information and previous problems in communication with the Government during reconstruction of site in early 90-ties. Main recommendations of EIA studies are:

1. Increase safety of repository;
2. Develop PR activities for education of society;
3. Develop the compensation mechanism for local municipality.

To develop the positive co-operation between the local municipality and repository, the following measures are performed:

1. Preparation and submission of 3 months activities report for local municipality;
2. Preparation and submission of annual environment monitoring report;
3. Participation in the renovation activities of the middle school of Baldone;
4. Support of different projects of Baldone municipality;
5. Developing of wastes minimization program for decommissioning of Salaspils research reactor.The last issue is connected not only with the protection of population of Baldone municipality, but also includes the measures for protection of environment by using of modern technologies for conditioning of radioactive wastes at Salaspils site, which is more suitable for it purposes.

5. Conclusions

1. The national near surface disposal site for radioactive wastes exist in Latvia.
2. The decommissioning of Salaspils research reactor causes upgrades of Baldone repository.
3. The additional efforts must be performed for development of co-operation with local municipality to receive the support for development of radioactive wastes management system in Latvia.
4. The education of society is necessary for further development of radioactive wastes management system in Latvia.

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Triumf - The Swedish data base system for radioactive waste in SFR

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ABSTRACT

All short lived LLW/ILW from the operation and maintenance of all Swedish Nuclear Power Plants are disposed in SFR, the Swedish final repository for radioactive operational waste. It is important to save all the information about radioactive waste that is needed now and in the future. To be secure that, we have developed a database system in Sweden called Triumf, consisting information about all the waste packages that are disposed in SFR. The waste producers register data concerning individual waste package during production. Before transport to SFR a data file with all information about the individual waste packages is transferred to Triumf. When transferred, the data are checked against accepted limitations before the waste can be loaded on the ship for transport to SFR. After disposal at SFR the deposition location in the repository is added to the database for each waste package.

1 Introduction

SKB (Swedish Nuclear Fuel and Waste Management Co) is owned by the Swedish Nuclear Power utilities and has been appointed as responsible for the management of Sweden’s radioactive waste. The final repository for radioactive operational waste, SFR, has been in operation since 1988. All the short-lived waste; low-level waste (LLW) and intermediate-level waste (ILW) from the operation and maintenance of the nuclear power plants is disposed in SFR, along with radioactive waste from medical use, industry and research.

SFR has five different rock chambers for disposal of different kind of waste. The most active waste is disposed in a concrete silo surrounded by a clay buffer. The other four chambers consist of one cavern for LLW (BLA), two caverns for concrete tanks with dewatered ion exchange resins (BTF1 and BTF2), and one cavern for ILW (BMA). BMA and the silo are for intermediate-level waste and the three other caverns are for low-level waste. Today about 30 000 m^3 of Low- and intermediate level waste has been disposed in the rock chambers at SFR and the total capacity of the repository is 63 000 m^3.

For a waste management system it is important to be sure to save all relevant information about the radioactive waste that is disposed in the repository. To secure that all information needed now and in the future about the radioactive waste is stored, it is important to document as much as possible. It is also important that the information about the waste is kept in a safe way, not only now but also in the future. Because of this we have developed a database system in Sweden called Triumf. Triumf consists of information about all the waste packages that are disposed in SFR.

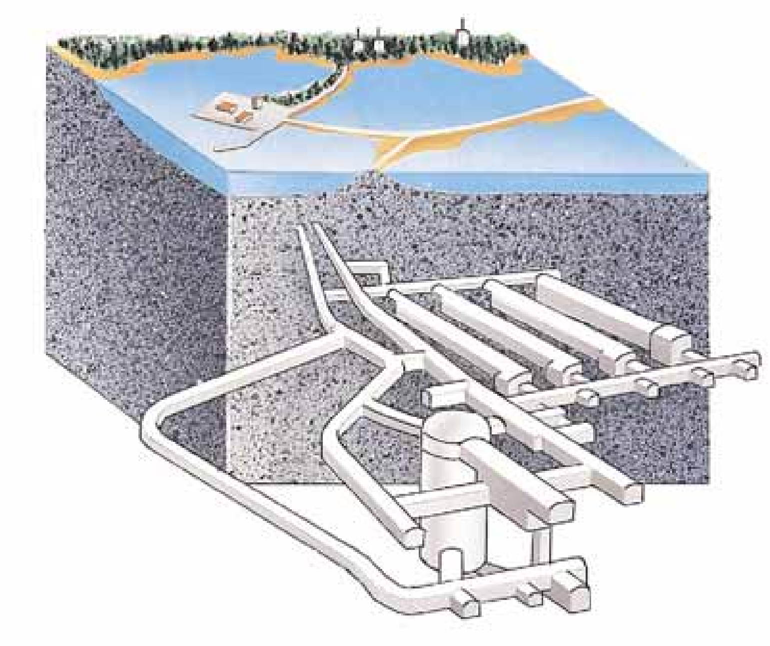

Fig 1. SFR

2 Radioactive waste management system

The Swedish waste system includes a lot of participants; waste producers, transport ship operators, final repository owners, repository operators and authorities.

In Sweden the waste is divided into waste types depending on what kind waste it is, how it is treated and who the producer is. Every waste type has its own designation and a document that describes the waste type. This document, called Waste type description, gives the criteria for the waste type. The waste type also gives information about for example the physical and chemical properties of the waste package

The waste type description has to have an approval from the authorities before the waste type can be transported to and disposed in SFR.

Handling of radioactive waste is strictly regulated and the system consist of different types of rules like waste acceptance criteria, transportation rules and deposition rules. To be sure that the rules are followed, they are checked by the staff and by the computer system. Triumf consists of all applicable rules and they are used as a first step to check incoming waste data.

The waste producers register data concerning individual waste packages during production and they are responsible to attend to that the information follows with the waste package to the final repository, where the information will be kept forever.

The Triumf system helps us to both store the information in a safe way for the future and to make sure that all the rules in the waste management process are fulfilled.

3 Information of the waste registered in Triumf

Before transport to SFR a data file with all information about the individual waste packages is transferred to Triumf.

When the information of the waste package has been transferred to Triumf, the system does a first check of the information to see that the data file comes from the right sender and that all the information is in the right format. Triumf also checks that the information of the packages is in the range of acceptance for that waste type.

After that a competent person at SKB does a last check of the incoming data and sends an approval to Triumf. Before this approval has been transferred to Triumf the overhead cranes in the cavern for ILW and the silo are locked for deposition of those waste packages. When the approval has been transferred to Triumf the waste producer are allowed to load the waste containers on the ship and the overhead cranes at SFR can dispose the waste when it arrives to the repository.

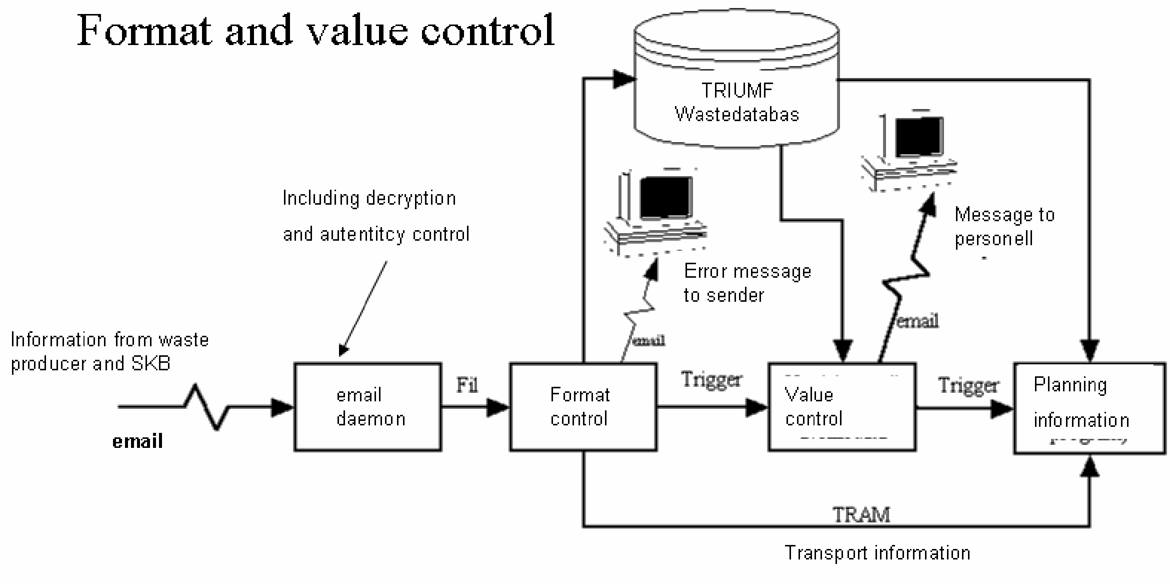

Fig 3 Format and value control in Triumf

The following information of each waste package is transferred to the Triumf database before transportation to SFR:

- *Waste package identification* - Each package has it's own unique identification number. Each package must have a permanent label with this identification number. This label is a link between the package and the record.

- *Waste type* - The information on which waste type the waste package belongs to is registered in Triumf. To have all information of the waste, you need the information both in Triumf and from the Waste type description.

- *Package type code, waste category code and conditioning method code* - A number system gives the information of the conditioning method of the waste, what package is used and which waste category the waste belong to.

- *Package weight* - The package weight is important information for the handling in SFR.

- *Date of production* – This information can for example give you information about the production process if it has changed in time.

- *Nuclide content and total activity* - The nuclide content is registered, both the nuclide specific and the total activity.

- *Surface doserate and doserate at 1 meter* – Is interesting under the different handling steps.

- *Measuring date* - The measuring date is necessary information for conversion of the activity of the waste package which is done automatically by the system. You just give a date for calculation and get the correct activity both nuclide specific and the total activity of the package.

After the waste package has been disposed in the rock chambers at SFR the information of the position and date of deposition of the waste are registered in Triumf. The control system for the overhead crane gives this information automatically to Triumf, the waste disposed in the caverns for LLW has to be registered manually buy the operator.

Fig 3. Unloading of a transport container in BMA

4 Reports from the Triumf system

From the Triumf system it is possible to search for the information you are interested in by using a data browser. You give your selection rules to the browser and get the information you want. For example this is useful for reports on how much waste has been disposed in different caverns and what the activity of a specific nuclide is at the moment in a cavern.

To the Triumf system a newly developed data program is coupled. This program called Prosit is used for short- and long rang forecasts to find out if it's enough space in SFR for the waste that is planned to be produced under the life cycle of the Swedish power plants. Also the radioactivity and different materials in the waste is important to forecast so the limit in the license for SFR don't get exceeded. To do these forecasts standardized referens waste packages for every waste type are registered in the system with average activity, material content and so on.

The program is also used for producing annual reports for the repository.

The Tunnel Sealing Experiment: The Construction and Performance of Full Scale Clay and Concrete Bulkheads at Elevated Pressure and Temperature

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ABSTRACT

Concepts for deep geologic disposal of radioactive waste, as proposed by many international organizations, include bulkheads or plugs in the shaft, or at the entrances to disposal rooms, or both. The seals are primarily to prevent groundwater transport of radioisotopes along underground openings but also provide a measure of security by restricting tunnel access. The safety of the respective disposal systems relies on the combined performance of the natural barriers (host rock) and engineered barriers (the waste form, the waste container, the buffer barrier, the room, tunnel and shaft backfill and sealing materials). To understand the functionality of these systems it is important to study them in whole or in part at full scale. One such study was the Tunnel Sealing Experiment (TSX), a full-scale tunnel seal component study. The TSX showed it is possible to construct tunnel seals that limit axial flow under high hydraulic gradient and elevated temperature. The clay and concrete bulkheads had seepage rates of 1 mL/min and 10 mL/min at ambient temperature. Elevated temperatures caused a further decrease in seepage past the concrete bulkhead to approximately 2-3 mL/min.

1. INTRODUCTION

The TSX had two bulkheads (Fig. 1). One was made of low heat high performance concrete (LHHPC) developed at Atomic Energy of Canada Limited (AECL) [1] and the second was made of approximately 9000 highly compacted bentonite-sand material (clay) blocks. The swelling of the clay bulkhead was confined by sand in the test chamber on one side and by a structural steel restraint on the other. In the first phase of the TSX, the central 12-m-long sand-filled test chamber was pressurized to 4 MPa by means of a static water head. A circulation pump and heaters were added for a second thermal phase that involved heating the water in two steps to 65°C at the face of each bulkhead. At the conclusion of heating, a three-month cooling period was followed by depressurization of the tunnel. Samples were then taken to measure the post-test conditions in terms of density, water content, structure, chemistry and strength. The first phase of the TSX [1] was conducted jointly at the URL by Japan Atomic Energy Agency (JAEA), Agence Nationale pour la Gestion des Déchets Radioactifs (ANDRA) of France, the United States Department of Energy (through the science advisor for Waste Isolation Pilot Plant) and AECL to demonstrate technologies for construction of bentonite and concrete bulkheads, to quantify the performance of each bulkhead and to document the factors that affect performance. The second phase was conducted by JAEA, ANDRA and AECL [2].

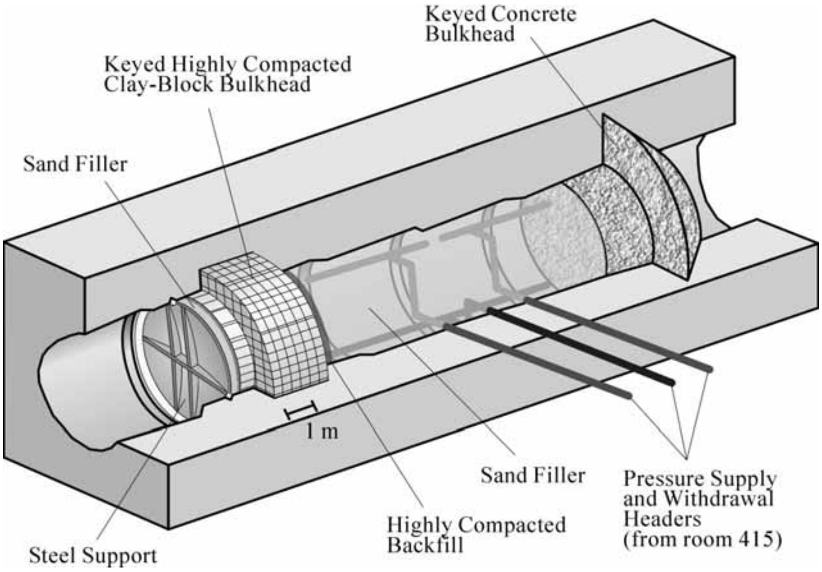

Fig. 1. Configuration of the TSX. Clay Bulkhead is 2.6 m thick; Concrete Bulkhead is 3.5 m thick

2. CONSTRUCTION AND OPERATION

The tunnel was excavated parallel to the trend of the maximum principal stress (σ_1 = 60 MPa (trend/plunge 145.0°/14.6°), σ_2 = 48 MPa (53.5°/5.8°) and σ_3 = 11 MPa (302.6°/74.2°), had a 3.5-m-high by 4.4-m-wide elliptical cross section and was 40 m in length using careful full faced drill and blast techniques. This was followed by excavation of the keys using a rock excavation technique developed at the URL called perimeter line drilling and rock splitting. Drilling delineated the perimeter of the rock area to be removed and the central mass was drilled and then hydraulic splitters were used to remove the rock. The rock mass was unfractured granite and granodiorite.

The 67 m^3 clay bulkhead was installed first along with its restraint system. In a repository, swelling clay would be restrained by concrete or backfill material. The restraint system, which was designed to withstand 4 MPa of hydraulic pressure and 1 MPa of swelling pressure, also permitted seepage collection via a geomembrane configured into different measurement zones. The system consisted of a rock bolted concrete bearing ring supporting an elongated hemispherical steel dome that had the loading evenly distributed via 1 m of silica sand confined by a stainless steel plate adjacent to the clay. The 2.6 m long clay bulkhead was comprised of 70% Kunigel V1 bentonite clay and 30% graded silica sand blocks with nominal dimensions of 0.1 m x 0.36 m x 0.20 m. The blocks were fitted together and crushed block material was used for gap fill. Before placement of the blocks, the walls of the clay key were pneumatically covered with 5 to 60 mm of shotclay material. The shotclay material was fabricated by first air-drying and crushing compacted clay blocks into particles of 10-mm-diameter or smaller, and then returning the material to the mixing machine to "round" off the corners of the particles. As the clay bulkhead was being constructed, a 0.3 m thick sand-clay backfill wall was built on the test chamber side of the tunnel to serve as both support and erosion control for the clay bulkhead. The central portion of the tunnel was filled with sand after the clay bulkhead was built, as internal restraint and to reduce the volume of water used.

On the upstream side of the 76 m^3 concrete bulkhead, a 250-mm-thick wall was first cast to provide an inner form against which the concrete bulkhead could be poured and to act as a buttress for the remaining sand placement. The concrete wall and bulkhead were composed of LHHPC. In LHHPC a substantial part of the Portland cement is substituted with pozzolanic silica fume and non-pozzolanic silica flour. A naphthalene-based superplasticizer was used to enhance workability of the concrete. These substitutions lower the heat of hydration and also reduce the pH of the cured concrete to the range of 10.6 [1]. The cement, silica fume and silica flour (in a 1:1:2 ratio) were blended, batched and bagged for use in pre-weighed quantities. The dry aggregates were similarly and separately prepared.

Both the fine and coarse aggregates were derived from a glacial deposit and were mostly of granitic origin. Once the concrete bulkhead was cured, initial pressurization of the tunnel to 300 kPa showed high flow rates. Pressure was reduced and the concrete-rock interface was grouted through pre-installed grout tubes. Subsequent pressurization showed substantially reduced flow along the interface.

The pressurization system was installed in parallel with the seal construction. It supplied pressurized water by means of a standing water head; a pressure-reducing valve permitted the water pressure in the tunnel to be increased to a maximum of 4 MPa. Pressurization was planned for four months, but took place over a period of 19 months due to initial flow events past the clay bulkhead that showed time was required for hydration and associated swelling of the clay bulkhead to take place. In 2002 a heater loop was added to allow the temperature in the tunnel to be increased and the water circulated through the tunnel was heated until 2003 November reaching a peak temperature of approximately 65°C at the centre of the upstream ends of the bulkheads. The experiment was decommissioned in 2004 with samples taken from both bulkheads to confirm readings and determine physical conditions.

The TSX was monitored with over 900 instruments. Instrumentation recorded hydraulic pressure in and around the clay, concrete and rock; loading pressure in and around the clay; moisture content in the clay and concrete; damage development in the rock and concrete; strain in the concrete; displacement of the concrete and clay bulkheads; water flow into the tunnel; and temperature of the tunnel, concrete and clay. Seepage was recorded manually.

3. OBSERVATIONS

Initially there were high flows past the clay bulkhead, however, hydration and subsequent swelling of the clay reduced the seepage rate. Flow was primarily at the shotclay and clay-rock interface for the clay and at the interface for the concrete bulkhead. High seepage also occurred at the concrete-rock interface, requiring remedial cement grouting. Ultimately, at 4 MPa hydraulic pressure across these bulkheads, the grouted concrete bulkhead and the saturated clay block bulkhead had effective hydraulic conductivities of 10^{-10} m/s (~10 mL/min seepage) and 10^{-11} m/s (~1 mL/min seepage) respectively (Fig. 2). At elevated temperatures the rate was unchanged for clay but decreased to 2-3 mL/min for the concrete bulkhead. Tracer tests indicated a 19-hour transit time past the concrete bulkhead and 816 hours past the clay bulkhead at ambient temperature and 4 MPa pressure.

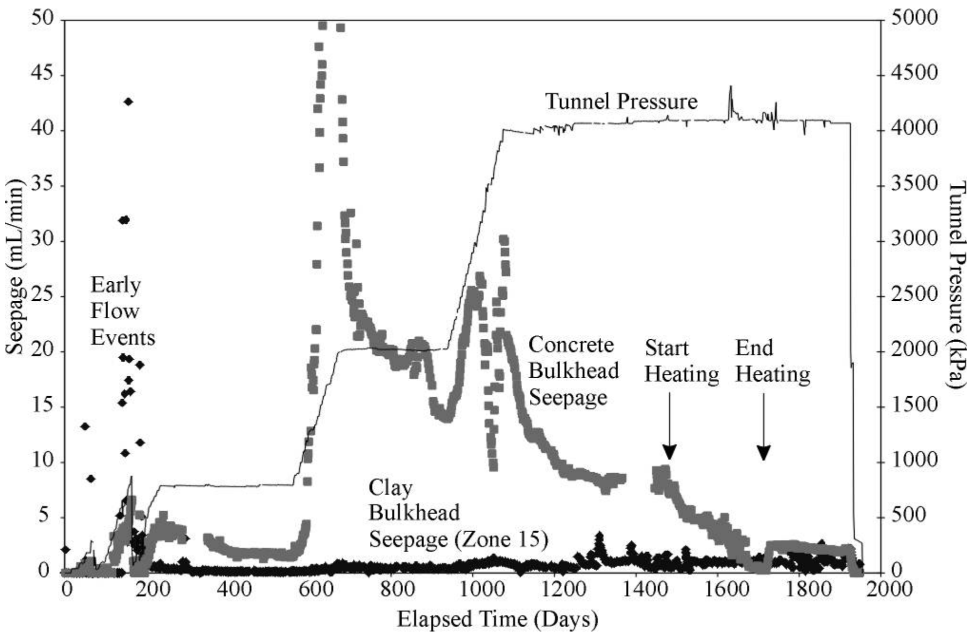

Fig. 2. Clay and concrete bulkhead seepage

The clay bulkhead saturated from the edges inward to the core, which was confirmed by suction and hydraulic pressure measurements and post-testing sampling. During heating more permeable regions showed no temperature related pressure increase while those in the core showed an increase indicating that water movement was more restricted in the clay bulkhead core.

The clay bulkhead showed a total of 54 mm of downstream displacement as a result of clay bulkhead and restraint system compliance. The displacement was confirmed by sampling that showed the clay near the upstream end had a lower density, indicating that the clay bulkhead had expanded to maintain contact with the rock and the backfill wall. The dry density and gravimetric water content of the clay blocks used in TSX construction were approximately 1.93 Mg/m^3 and 14.7% respectively. At the end of the test, block density ranged from 1.85-2.0 Mg/m^3 and water content ranged from 13-17%. The lower density (higher water content) material was near the upstream face and the higher density (lower water content) was located near the downstream face.

The shotclay materials placed in the region between the blocks and the rock had an estimated as-placed dry density of 1.3 Mg/m^3 and gravimetric water content of 18.5%. At the end of the test, on the upstream side of the keyed section of the bulkhead, the shotclay had dry densities ranging from 1.0-1.5 Mg/m^3 and water contents of 30-60%, indicative of swelling to maintain contact with the rock surface. On the downstream end of the keyed section, the density was approximately 1.8 Mg/m^3 and the water content was approximately 18%.

The backfill wall was also fully saturated at the end of the test. The lower compacted backfill was placed at 2.1 Mg/m^3 and 5% water content and was approximately 2.2 Mg/m^3 and 8.5% at the end of the test. The upper, pneumatically placed portion was placed at 1.5-1.9 Mg/m^3 and 14-28% water content and was approximately 2.2 Mg/m^3 and 8.5% at the end of the test. Some expansion of the backfill material occurred into the gap at the crown of the sand-filled pressure chamber, causing decreased backfill density near the roof of the tunnel.

Three unusual features were noted during decommissioning of the clay bulkhead: the first was small, unconnected relict bands or pockets of eroded material in the crown region of the keyed portion of the bulkhead and extending perhaps 1/3 of the way down the wall of the clay key. The second was a series of very thin, unconnected, relict flow channels along the contact between clay blocks closest to the sidewall of the clay key. The third was a region that contained air bubbles in the uppermost portion of the upstream face of the clay key where the shotclay was least dense and escape of trapped air was impossible. Given the observed seepage rates in the clay bulkhead, none of these features is believed to have substantially affected clay bulkhead performance.

The concrete bulkhead showed only 0.2 mm of axial displacement from hydraulic pressure loading and 0.7 mm of axial thermal expansion. The concrete bulkhead was monitored by an array of acoustic emission (AE) sensors and showed that 84% of the AE activity occurred within a few weeks of pouring due to formation of three internal cracks. These formed due to internal stresses related to shrinkage associated with curing and the shape of the bulkhead. The cracks were filled during grouting of the interface. The remaining AE activity mainly occurred once the full 4 MPa tunnel pressure was reached (10%) and during heating (4%). The remaining events occurred during pressurization and end of test depressurization, suggesting the concrete bulkhead acted as one stable mass once it had cured, and had been grouted.

At the concrete bulkhead, piezometers indicated that the grout injection location acted as a gasket. Pressure in the interface upstream was similar to the tunnel pressure, while the pressure downstream from the injection location was essentially zero.

The concrete mass appeared homogeneous on the macroscopic scale in terms of material distribution, however, the measured properties of the concrete indicated some spatial differences in the concrete. Porosity measurements ranged from 11% on the downstream end to 6.5% on the upstream end. The upstream values are similar to those found in high performance concrete (such as LHHPC), while the downstream end porosity value is similar to that found in regular concretes. Sonic velocity ranged from 4300 to 4800 m/s, with lower velocity near the downstream end of the concrete bulkhead and higher velocities near the centre of the bulkhead, however, the sonic velocity measurement closest to the downstream face had a magnitude similar to that of the bulkhead core. Water supplied to the bulkhead face on the inside of the downstream formwork may have aided in decreasing void formation during curing for a shallow depth of concrete. The transmissivities from the centre of the bulkhead were on the order of 10^{-15} m/s based on *in situ* testing. Near the upstream and downstream ends the transmissivities were approximately one to two orders of magnitude higher. The concrete-rock interface and cracks returned magnitudes on the order of 10^{-12} m/s. Scanning electron microscope images showed the edges of the concrete to have more microcracks and some ettringite formation. The compressive strength of the bulkhead averaged 77 MPa, indicating that on average, the concrete did not gain much strength beyond its predicted 28-day strength of 70 MPa. However, the upstream strength value was approximately 100 MPa, suggesting a lack of water in the bulkhead core and downstream locations may have inhibited strength increase. The measurements indicate a less permeable core with some perimeter microcracking that likely resulted from drier conditions at the perimeter.

4. SUMMARY

The TSX allowed the testing of candidate repository materials at full scale. The TSX showed it is possible to construct functional clay and concrete bulkheads to seal tunnels and limit axial flow. Prior to the experiment it was believed that the EDZ would be the primary pathway for water flow around the bulkheads but the keyed seals cut-off or reduced flow through the EDZ and the primary pathways were actually the concrete-rock interface and clay-rock interface.

The swelling clay bulkhead also demonstrated the ability to self heal and to adjust to differential displacements in its own mass without developing leaks but only if the water pressure was increased gradually or flow past it can be controlled. In the TSX a structural steel restraint system was installed providing mechanical support but not seepage resistance. The concrete bulkhead was able to withstand the loading from hydraulic pressure with minimal offset and once grouted provided considerable hydraulic resistance. This suggests concrete would make a suitable restraint for a swelling clay component of a seal.

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HYDROGEOCHEMICAL INVESTIGATIONS AT THE ANDRA MEUSE/HAUTE-MARNE UNDERGROUND RESEARCH LABORATORY

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ABSTRACT

In November 1999 Andra began building an Underground Research Laboratory (URL) in eastern France. The geological formation selected for this laboratory is a 130-meter thick argillaceous rock level. This clay rich layer is located at a 400 to 600 meter depth. To characterize the confining properties of the clay, pore water composition had to be studied. For this purpose an innovative device was designed for gas equilibration and direct sampling of the pore water. The experimental device consists of a vertical ascending borehole with a 5 meter long test interval at its far end in which a gas circulation is established. After a few weeks, due to the hydraulic gradient between the test interval and the rock formation, the water flows freely at a rate of 0.5 to 1.3 litters per month in the borehole and it is sampled. The chemical composition of this water is compared with a theoretical composition deduced from core analyses and thermodynamic modelling.

1 Introduction

The Callovo-Oxfordian formation of the eastern Paris Basin is a 130 m thick clay rich sedimentary sequence. Its water content is around 8 %wt and its hydraulic permeability is below 10^{-11} m.s^{-1} (10^{-18} m^2). Since 1994, Andra (the French Agency for nuclear waste management) has been studying the feasibility of a high-level long-lived radioactive waste disposal in this formation. The geochemical composition of pore water is one of the elements required to assess the confining properties of this argillaceous rock. Two aspects related to pore water composition are being studied to this effect. The first one concerns the knowledge of the hydrogeochemical mechanisms governing this composition to study the interaction of the water with the various barriers surrounding the waste canisters. The second aspect concerns the distributions of non reactive natural tracer concentrations. Indeed, the interpretation of the observed distributions will help to evaluate the history of solute transfers.

From the outset, the Callovo-Oxfordian pore water composition has been studied through measurements performed on core samples from deep boreholes. A conceptual model of the pore water composition based on thermodynamic equilibrium has been built and a consistent set of data has been obtained on natural tracers such as water stable isotopes, noble gases and chloride [1].

In 2004, in situ direct sampling of the water was carried out for the first time in this type of formation and has been performed continuously since then. For that purpose, two short (15 m long) boreholes were drilled in two drifts at 445 m and 490 m depth (Fig 1). In the open section of these boreholes (test interval), a gas is circulated at a total pressure of about 1.2 bar. Due to the hydraulic head gradient between the test interval and the surrounding rock, the water flowing freely into the interval is pumped out. These two experiments have made it possible to “pump” the pore water at rates between 0.5 and 1.3 L per month for the past two years. This paper presents the design and main results of this innovative experiment.

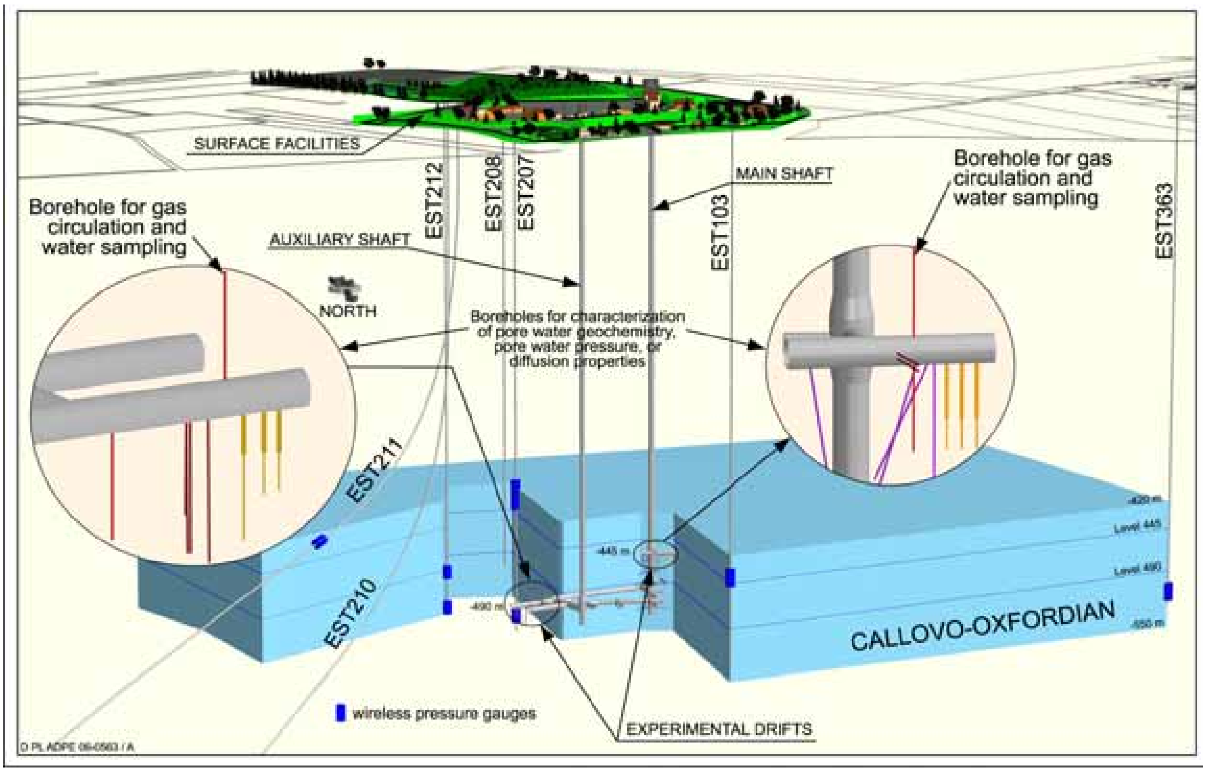

Fig 1. Localization of deep and short boreholes used for Callovo-Oxfordian pore water characterization on the URL site

2 Experimental design

2.1 Drilling phase

The two 15 m long boreholes dedicated to gas circulation and water sampling are vertically ascending. The top 5 meters constitute the test intervals. Taking into account the feedback from the Mont Terri project [2, 3], these boreholes were drilled with nitrogen to avoid rock oxidation. The borehole equipment was installed as quickly as possible after the drilling. Great care was taken to restrict the development of bacterial activity in the boreholes by cleaning the drilling tools and implementing a specific protocol for their handling. The cores provided samples for physical, mineralogical, chemical and microbiological analyses on solids, interstitial fluids and dissolved gases.

2.2 Equipment

The borehole equipment allows the circulation of the gas in contact with the rock in a closed circuit, the sampling of formation water produced in the test interval, and the monitoring of gas and water pressures in the open section. The equipment comprises packers isolating the test interval and 6 gas, water and pressure control lines that link the test interval to the controls in the drift (Fig 2). A slight overpressure of inert gas (nitrogen or argon) was applied to the borehole during the installation of the equipment to prevent oxygen from entering into the test interval. Once the installation completed, the open section and control lines were filled with argon.

The control lines are linked to the so-called “surface” equipment (controls) located in the drift. This equipment consists of pressure gauges connected to a data acquisition system and two modules: a water sampling module and a gas circulation module. The water sampling module allows pumping of the water at a controlled flow rate, online monitoring of the pH, Eh and electrical conductivity of the pumped water and conditioning of the sampled water to avoid contact with the atmosphere. The gas circulation module allows monitoring the evolution of the gas composition due to exchanges with the rock surface of the open section. Monitoring is performed through online infrared spectrometry and gas sampling.

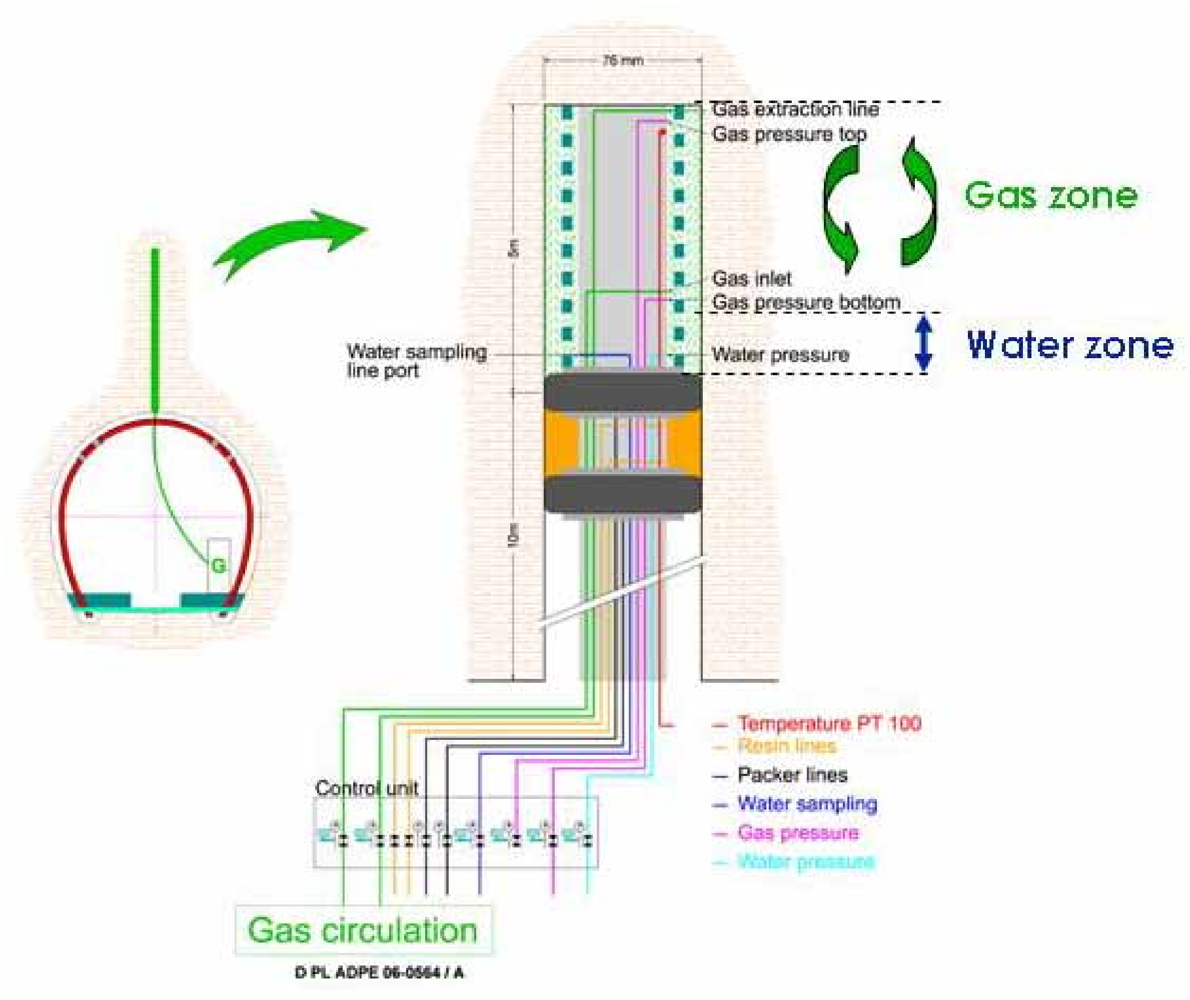

Fig 2. Experimental design for gas circulation and water sampling

2.3 Monitoring and control of the experiment

During the experiment, the water pumping flow rate is adjusted to maintain the water level between 10 and 40 cm at the bottom of the test interval. Above this section, where the water accumulated transiently is pumped out, the gas is circulated in contact with the accumulated water surface and the rock over more than 4.5 m height. Gas pressure was set from the outset and has been maintained throughout between 1.1 and 1.2 bars. These values have induced a high pressure gradient around the test interval.

3 Main results

3.1 Hydraulic properties

The water production in the two boreholes is about 0.5 L per month at -430m (center of the interval of the borehole drilled at -445 m) and 1.3 L per month at -475 m (center of the interval of the borehole drilled at -490 m), respectively. A network of pore pressure gauges measures the natural hydraulic head around the boreholes. Results of calculations obtained through this data show hydraulic permeabilities of about 10^{-13} m/s. These values are coherent with other available measurements at these levels.

3.2 Gas composition

The composition of the circulating gas evolves from pure argon. The main gases are carbon dioxide and methane. The molar fractions of carbon dioxide, methane, and other alkanes up to C5 (pentane), as well as of other gases coming from the rock and its pore water have increased. After more than a year, the gas composition has not stabilized as yet, but the maximum partial pressures observed are close to 2 mbar for both carbon dioxide and methane in the two boreholes.

3.3 Water composition

Fig 3 presents online measurements of pH and Eh carried out on the water pumped in the borehole located in the drift at -490 m (center of the test interval at -475 m). The pH is very stable at 7.3 ± 0.1. The Eh maximum positive values correspond to calibration phases. Stabilization takes a long time following a calibration phase. Minimum values below -200 mV SHE are considered to be the most representative of the pore water.

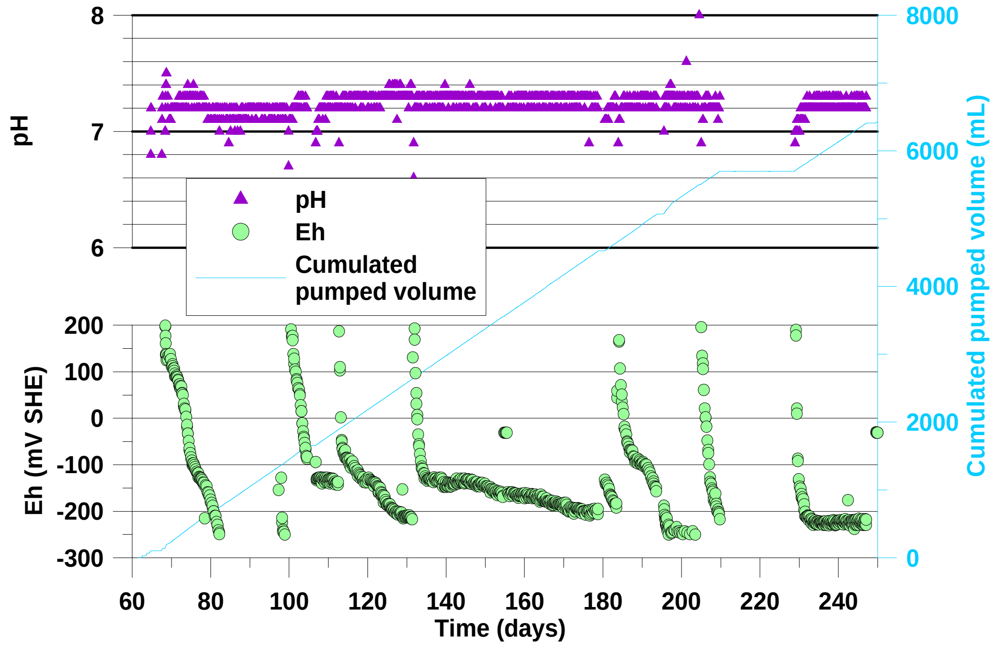

Fig 3. pH and Eh evolution of the pumped water and cumulated volume

Chemical analyses have been performed on the sampled water. Fig 4 shows the results for major species in a Schoeller logarithmic scale diagram. Two of these results (-475 and -430 m) come from the boreholes dedicated to gas circulation and water sampling. Additional analytical results of direct in situ sampling at -490 m were obtained from a borehole dedicated to pore pressure measurements. In this borehole, water inflow was observed for a short period of time following the installation of the multipacker completion. The observed compositions at these three depths are very similar; the main difference concerns chloride concentration. The study of this concentration on cores revealed that it varies slightly with depth [4, 5]. The chloride concentrations observed in the water collected in situ should help refine the values deduced from core measurements and bring new data on the porosities accessible to this species.

The fourth curve of Fig 4 corresponds to the composition worked out through thermodynamic equilibrium modelling before the first in situ collection of water samples. This computation relies on the hypothesis of an equilibrium with the minerals identified in the rock in neoformed states, of reactions of cation exchanges at the surface of the argillaceous minerals and on the average chloride concentration deduced from the core measurements. The constants of these ion exchange reactions were deduced from core tests. [1].

Comparison between the observed compositions and the calculated composition shows a discrepancy bellow a factor of two for all the species except potassium. For this species an acceptable modification in the model of the potassium and sodium exchange constant leads to the observed concentration.

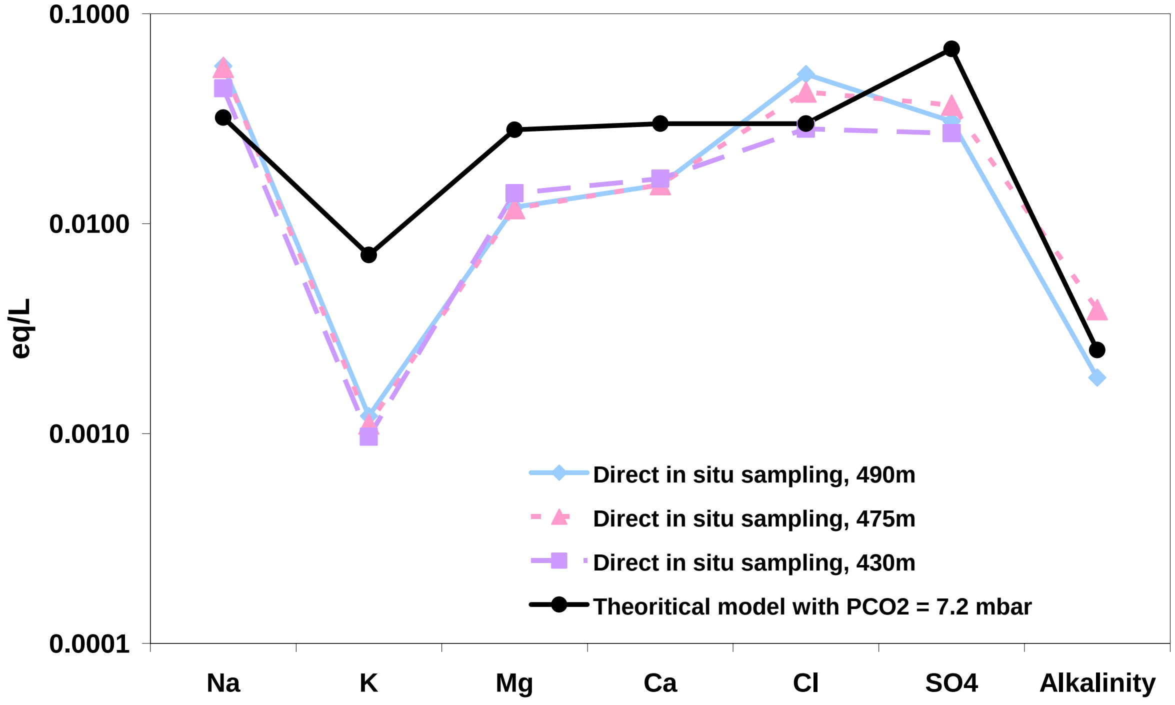

Fig 4. Concentration of major species measured in the pumped water and the composition obtained through thermodynamic modelling

4 Conclusion

The gas circulation and water sampling experiments implemented in the Andra laboratory made it possible and analyse for the first time Callovo-Oxfordian pore water. pH stability over several months demonstrate that the water samples collected through dedicated experiments remain unaltered. The model provided satisfactory prediction for the pore water below a factor 2 for most of the major species. The compositions observed on the water samples should help improve the conceptual model. They confirm the vertical variations of the chloride concentrations and should help refine the results obtained on cores for this species over the entire profile.

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SCIENTIFIC INVESTIGATION IN DEEP BOREHOLES AT THE MEUSE/HAUTE MARNE UNDERGROUND RESEARCH LABORATORY, NORTHEASTERN FRANCE

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ABSTRACT

From 1994 to 1996, the preliminary investigation carried out by Andra, identified a sector favourable for hosting a laboratory in argillaceous Callovo-Oxfordian formation which has a thickness of 130 m and lies more than 400 m below ground level. In November 1999 Andra began building an Underground Research Laboratory (URL) with a 3D seismic survey over 4 km². From 2000 to 2004, large programs of boreholes were carried out on site and on the sector in order to define the characteristics of formations, to improve the regional geological and hydrogeological knowledge and to provide an accurate definition of structural features in Callovo-Oxfordian argillites and Dogger limestones.

These drilling programs have provided a fine characterization of the argillites on the laboratory area and a good correlation of geological properties at a sector scale.

1 Choice of Eastern France

ANDRA is in charge of analyzing the possibility of implanting a reversible nuclear waste disposal in deep geological formations. With this aim it has undertaken the construction of an underground laboratory in the eastern part of the Paris Basin. This region, with a geological history running over 365 million years and historically known to be stable, had been generally identified (Figure 1).

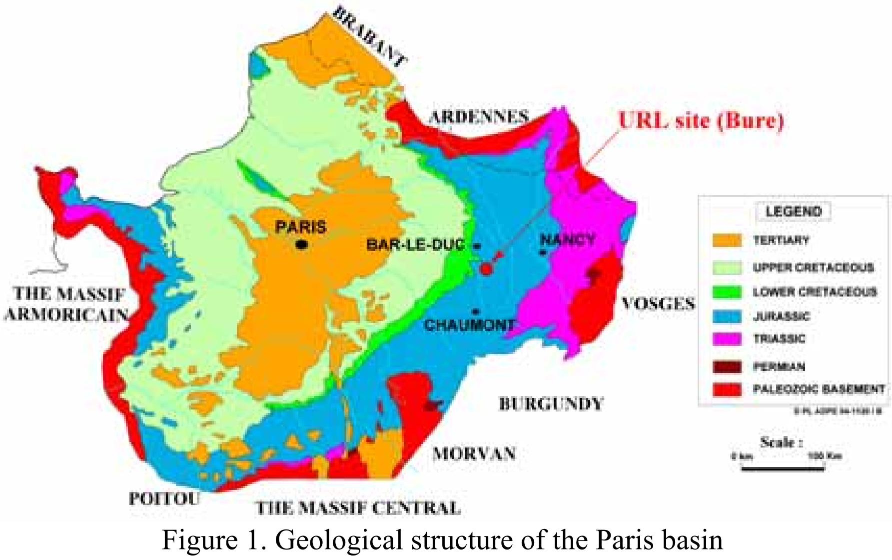

Figure 1. Geological structure of the Paris basin

The building of the Meuse/Haute-Marne underground research laboratory is conducted through the implementation of a scientific and technical approach, based on the knowledge acquired during the preliminary borehole drilling phases.

2 Research activities in the sector between the Meuse and the Haute-Marne (1994 to 1996)

The objectives of the preliminary investigation phase were to verify the characteristics of the selected host formation (argillites of Callovo-Oxfordian age) and the parameters of the surrounding rocks, particularly the hydrogeological parameters of the Oxfordian and Dogger formations. This investigation was carried out over a sector identified by an exhaustive analysis of prior geological survey data, 2D seismic profiles and borehole drilling operations.

In late 1994 and early 1995, two deep cored boreholes were drilled (Figure 2). Borehole HTM102 (1100 m deep), entirely cored from the Kimmeridgian to the Lias, was extended (rotary) to attain the first levels of the Triassic formation. In borehole MSE101 (922 m deep), located 15 km northwest of borehole HTM102, only the argillites formation has been cored.

From mid-1995 to mid-1996, 4 deep boreholes were drilled for hydrogeological (rotary method) and geomechanical (cored) investigations, three in the laboratory site and a fourth one 3 km southeast of the site.

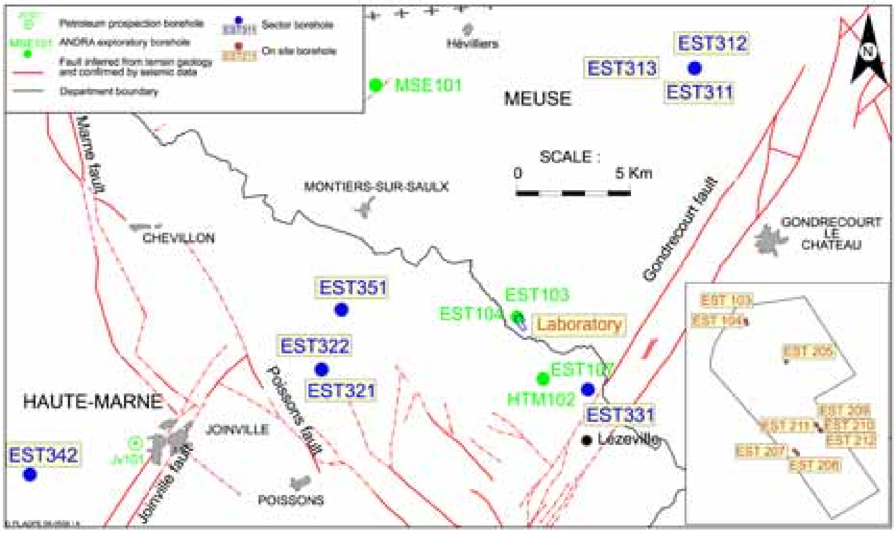

Figure 2. Location of the preliminary, sector and on-site boreholes

From 1994 to 1996, Andra identified through its preliminary investigation an area favourable for hosting a laboratory on the sector south of the Meuse and north of the Haute-Marne departments. In this sector the 130 m thick argillaceous Callovo-Oxfordian formation, lying at more than 400 m depth, has a very low permeability.

3 On-site boreholes (2000)

In March 2000, to monitor the influence of the drainage on the Kimmeridgian and Oxfordian formations of the shaft access, hydrogeological boreholes were drilled on the laboratory site using an inverse circulation method

For the Oxfordian formation, two boreholes (EST201 and EST203 - 420 m deep) have been equipped with five measuring chambers, thereby supplementing boreholes EST103 and EST104 located north of the laboratory site. For the Kimmeridgian formation, one borehole (EST202, 150 m deep) has been equipped with three measuring chambers (Figure 2).

In mid-2000, two geological exploratory boreholes were cored in the main and auxiliary shafts axis in order to define the characteristics of the formations to be taken into account for shaft sinking. A large program of scientific, hydrogeological and geomechanical measurements was therefore undertaken. In the EST205 borehole, an oil base mud was used to core the Callovo-Oxfordian argillites in order to achieve a better core recovery and carry out geomechanical tests (micro-hydraulic fracturing tests between 460m – 500m)

4 Scientific boreholes at sector scale (2003)

A drilling program was carried out on a large sector (about 2 000 km²) around the laboratory in order to improve the regional geological and hydrogeological knowledge, enhance the hydrogeological model, identify the lateral variation of the host formation and perform permeability measurements.
Throughout this sector (Figure 2), 8 boreholes were drilled on 5 platforms chosen for the acquisition of permeability and cinematic porosity fields in the rock structures surrounding the Callovo-Oxfordian, Oxfordian and Dogger formations.
The inverse circulation drilling method was used to drill the Oxfordian and Dogger carbonated formations, above and under the argillites formation, and more specifically for coring in the argillites.

The scientific program included:

- a detailed geological survey to acquire a lithological profile and identify possible sedimentation gaps in the Callovo-Oxfordian formation,
- hydrogeological and geochemical measurements in the carbonated formations (Oxfordian, Dogger) to obtain uncontaminated water and gas samples.

This information was used to define an equivalent transposition zone relevant for applying the results obtained.

5 Formation exploration borehole campaign (2003-2004)

The 2003-2004 on site drilling program consisted of seven deep boreholes, four of them deviated (up to 10° from the horizontal plane) to obtain an accurate definition of structural features in the Callovo-Oxfordian argillites and Dogger limestones, and the geological variability of the host formation at the laboratory scale (Figure 3). The objectives of the other drillings were to identify the natural stress properties of the host layer and to perform the first in-situ measurement of the diffusion of radioactive elements in the argillites. The drilling methods implemented were the inverse circulation drilling or polymer mud coring in the carbonate formations, the oil base mud coring in argillites and the air or nitrogen coring for the diffusion test in the argillites (EST208).

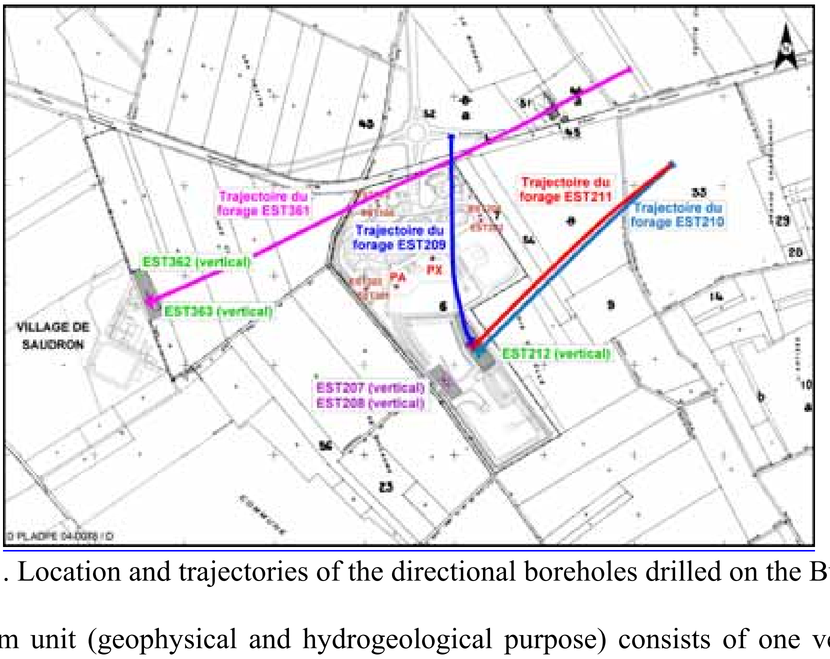

Figure 3. Location and trajectories of the directional boreholes drilled on the Bure site

The first program unit (geophysical and hydrogeological purpose) consists of one vertical borehole EST212 and two oblique directional boreholes (EST210 and EST211). Two EPG electromagnetic transmission pressure sensors were installed in EST212 and one in EST211.
The second program unit (geological and geomechanical purpose) consists of one oblique directional borehole (EST209) and one 1494 m-long sub horizontal borehole (EST361). A third borehole EST363 (vertical) was equipped with an EPG sensor.
The third program unit (measurement of the diffusion of radioactive elements) consists of two vertical boreholes, EST207 and EST208.

6 Hydrogeological results

The hydrogeological properties were acquired through specific tests, using innovative techniques developed in collaboration with our Finnish partners.
Head disturbances in the Oxfordian formation caused by shaft sinking began on 26 November 2001 when a drilling, intersected a porous and identified level 24 m from the sole of the main shaft, thereby triggering the drawdown process.

The permeabilities of the Callovo-Oxfordian formation (Figure 4) measured in boreholes EST212, EST211 and EST363 confirm the values determined for the host formation during previous campaigns, namely its very low permeability (5.10-13 m/s).

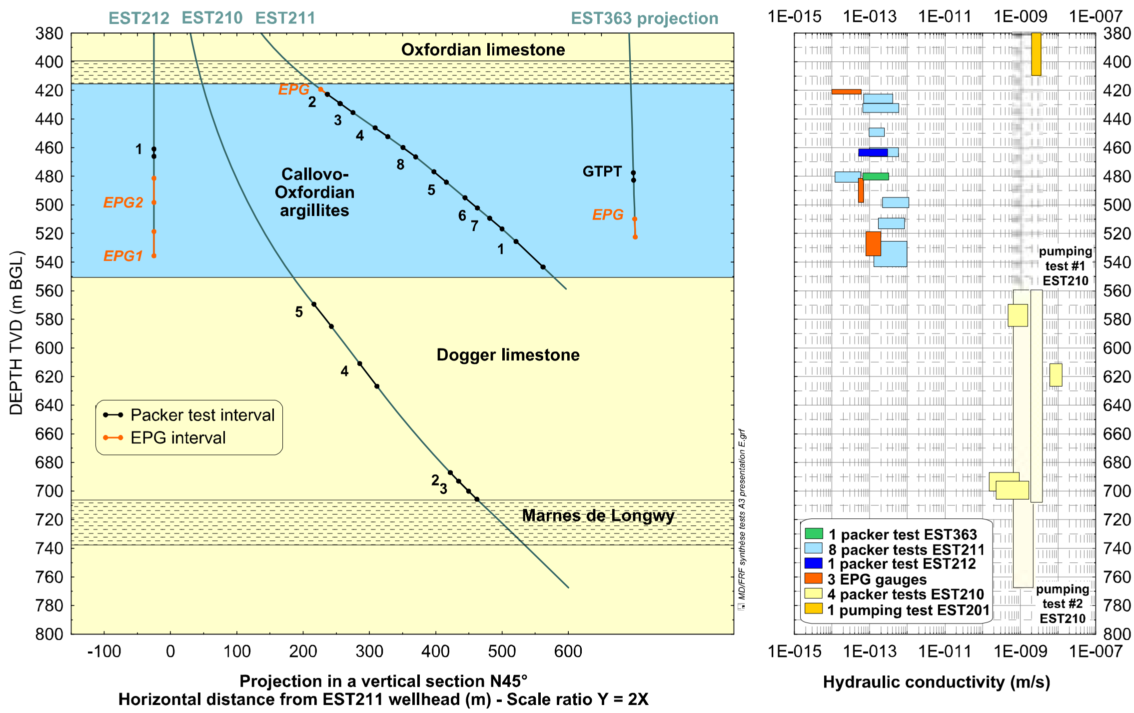

Figure 4. Results of permeability measurements in the host formation

7 Geological results

In the sector identified, the results of the geological survey and the interpretation of the intersected facies lead to the following conclusions regarding the geometry of the formations:

- The Oxfordian formation displays a slight decrease in thickness towards the northeast, in accordance with the regional palaeographic scheme. The transition from a carbonated platform system (towards the east) to a more open basin (towards the west) during the lower Oxfordian is confirmed.
- The Callovo-Oxfordian formation displays an increase in thickness (Figure 5) from the southwest towards the northeast (100 to 160 m, respectively), ranging from 135 to 138 m on the site.

From the structural point of view, no structures were intersected in the cored sections of the Callovo-Oxfordian formation (only joints or fissures were encountered in the carbonated facies).

The mineralogical results for the various boreholes confirm the mineralogical composition of the host formation, which contains 30 to 55% clay minerals, associated with 20 to 30% carbonates, 20 to 35% quartz and a small percentage of subordinate minerals. The lateral variation of the mineralogical composition is very low.

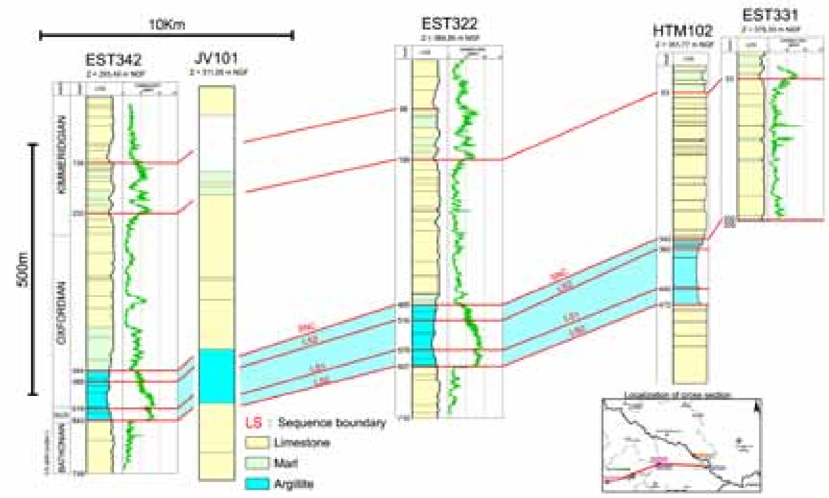

Figure 5. East-West lithological cross section of the sector

8 In situ stress profile

The various geomechanical measurement campaigns conducted in the framework of the survey borehole campaign in the Callovo-Oxfordian formation (boreholes EST209 and EST361) and Dogger formation (EST210) have (Figure 6):

- confirmed the amplitude of the minor horizontal component (σh) in the argillites
- made it possible to measure directly the amplitude of the minor horizontal stress in the Dogger formation
- yielded the anisotropy ratio of the horizontal stresses in the argillites
- shown that the maximum principal stress corresponds to the major horizontal stress (σH) in the Callovo-Oxfordian formation

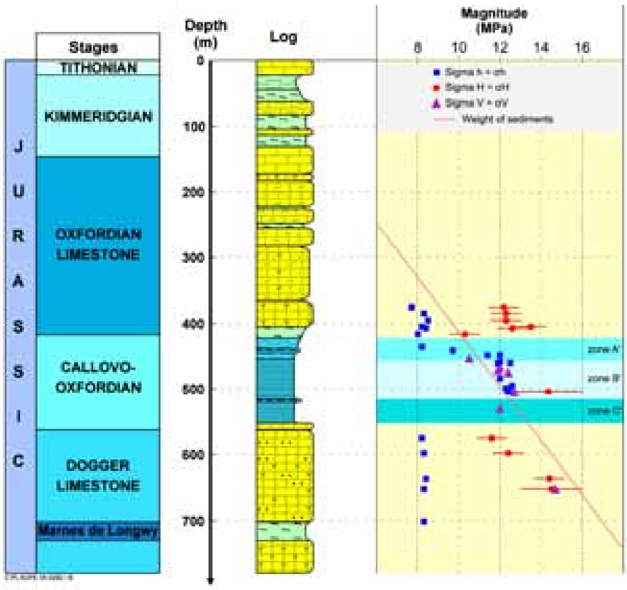

Figure 6. Stress values measured in-situ in the boreholes

9 Conclusions

These drilling programs (27 boreholes, 11 873 m of drillings, 4432 m cored) have provided a fine characterization of the physical and chemical properties of the argillites on the laboratory area and a good correlation of geological properties at sector scale. Based on these results, a transposition zone of 200 km² with similar characteristics has been delimited. During the next phase starting in 2007, a drilling survey will be carried out over the transposition zone to acquire the same accurate knowledge over the entire area of interest to ANDRA .

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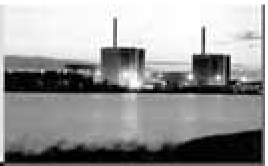

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