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Geological Challenges in Radioactive Waste Isolation

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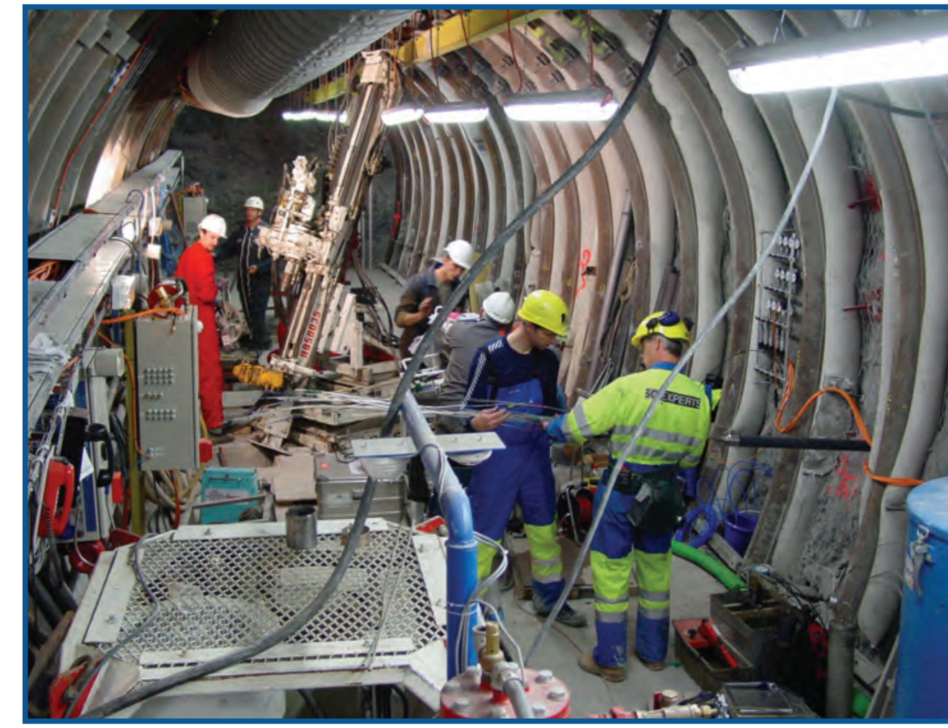
Fourth Worldwide Review



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Acknowledgments

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The cover illustration shows monitoring equipment for different experiments being carried out in the Callovo-Oxfordian argillites at the 445 m level in the French Meuse/Haute-Marne underground research laboratory. It is used with the permission of Agence Nationale pour la Gestion des Dechets Radioactifs.

Finally, we would like to express our gratitude to the scientists and engineers who contributed to this review. We are grateful for the considerable time and effort these authors took in preparing their contributions, and for their kind cooperation during the process of bringing this effort to completion.

Chapter 1

Introduction to *Geological Challenges in Radioactive Waste Isolation: Fourth Worldwide Review*

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1.1. INTRODUCTION

Over the past forty years, the development of the technology needed to isolate radioactive waste in underground rock systems has been found to be a formidable problem. This is especially the case in connection with high-level waste (HLW) after its removal from operations in nuclear power plants. There is also the additional problem of isolating low- and intermediate-level waste (LILW). Significant quantities of LILW are generated from various sources, and while they are not as long-lived in terms of radioactivity and do not pose the same level of difficulty as HLW, they constitute another problem for the nuclear industry.

The first investigations on isolating radioactive waste underground were started in the United States in the 1960s, in a salt mine near Lyons, Kansas (Witherspoon and deMarsily, 1991), and in West Germany in 1965 using an underground research laboratory (URL) in the Asse salt mine (Langer et al., 1991). Other countries began to focus on different rock types, and the first effort to study the problems of isolating HLW in granitic rock was initiated in 1977, when another URL was developed in a large body of granite at Stripa, Sweden, using the underground works of an abandoned iron ore mine to provide access to the laboratory (Witherspoon and Degerman, 1978). In 1980, the Belgians started their HADES project at Mol with the construction of a vertical shaft, ~220 m deep, from which they developed a URL in the Boom clay (Neerdael, 1996). This was the first underground research laboratory in clay. In 1983, Nagra in Switzerland started their Grimsel rock laboratory in crystalline rock some 450 m below the summit of the Juchlistock mountain in the Bernese Alps (McCombie and Thury, 1991). This early development of URLs captured the attention of those involved in waste isolation problems in other countries, and a significant number of additional URLs have now been put into operation or are being planned. The variations in rock conditions in these URLs and the wide variety of research investigations conducted in them have produced another benefit. Joint projects have been organized in which one or more countries could participate, through appropriate agreements, in some technical project with the operator of a URL. The Swiss waste program, although relatively small in terms of budget and manpower, has developed a surprisingly wide international scope of operations in their joint projects at Grimsel (in crystalline rock) and at Mt. Terri (in Opalinus clay).

To keep everyone abreast of the results from the wide variety of investigations that have been carried out in developing the technology needed for radioactive waste isolation, we collected reports from 19 countries and published the *First Worldwide Review* in 1991 (Witherspoon, 1991). We obtained reports from 26 countries and published the *Second Worldwide Review* in 1996 (Witherspoon, 1996), and we published the *Third Worldwide Review* with reports from 32 countries in 2001 (Witherspoon and Bodvarsson, 2001).

Some very interesting and significant results have been compiled in this *Fourth Worldwide Review* on radioactive waste isolation. A summary of pertinent data on the developments and plans from the reports on 24 countries is given in Table 1.1 at the end of this chapter.

1.2. *FOURTH WORLDWIDE REVIEW:* SOME HIGHLIGHTS

1.2.1. Is Clay the Best Rock Type for Long-Term Geologic Storage of High-Level Radioactive Waste?

Various rock types are being investigated in the different countries with geologic storage programs. Many countries are considering fractured granitic rock; others clay, salt, or sedimentary rock. The experience at Yucca Mountain, Nevada, U.S.A. has shown that the characterization of fractured volcanic tuffs is very difficult, particularly because the rock there is unsaturated, and unsaturated flow and transport is poorly understood. Certain important parameters such as fracture volumetric porosity, unsaturated fracture characteristic curves (relative permeabilities and capillary pressure as a function of liquid saturation), and fracture-matrix interaction are very difficult to measure either *in situ* or in the laboratory. Currently there are no such measurements for the unsaturated fractured rock at Yucca Mountain, although some values have been indirectly estimated.

A similar situation arises in the saturated fractured granite that is a likely candidate rock type for geologic repositories in many countries. Granitic rock is typically sparsely fractured, with a fairly random fracture system. Evaluation of fracture porosity and the effective fracture-matrix surface area for advective flow and matrix diffusion is extremely difficult. This will also be the case for any other rock types where fracture flow dominates. In contrast, the clays being investigated in Belgium, France, Switzerland, and other countries seem to have very low permeability, such that diffusion will be the main transport mechanism. The characterization of the diffusive processes in clay is much more straightforward and reliable than the processes involved in low-permeability rock (such as volcanic tuffs or granites)—which include advection, matrix diffusion, and sorption. Thus, clays may be the rock type of choice, and below we summarize the experience gained in Belgium, France, and Switzerland.

It has been found in several different countries that the hydraulic conductivity of argillaceous rocks can be extremely low (on the order of 10^{-12} m/s). This was first reported by the Belgians (Bonne, 1991) from their investigations on the Boom clay at a depth of 225 m in their underground research laboratory (URL) at Mol. More recently, the French have been investigating the Callovo-Oxfordian argillites at a depth of 445 m in their Meuse/Haute-Marne URL and have obtained hydraulic conductivities, from borehole and sample measurements, varying from 10^{-12} to 10^{-14} m/s (Lebon et al., 2006). They describe experiments on argillite samples in the laboratory that suggest that the permeability of the excavation-disturbed zone tends to reduce with the creep of the argillites during their resaturation and stress release, thus causing a gradual closing of fractures. After a long period of time, the stresses return to a natural equilibrium, closing the fractures so strongly that the permeability of the disturbed zone returns to that of the undisturbed argillite (Lebon et al., 2006).

The Swiss have also been investigating the properties of clay and have obtained results that are in good agreement with those of the French. They have concentrated on the Opalinus Clay, which is about 100 m thick at a depth of ~500 m in Northern Switzerland (Zuidema et al., 2006). They have confirmed the very low hydraulic conductivity of this clay, and also of surrounding sediments, which together with the Opalinus Clay form a nearly 300 m thick low-permeability layer. They have obtained geochemical and isotopic evidence that solute transport within the clay occurs predominantly by diffusion (Zuidema et al., 2006). The Swiss have concluded, along with the French, that clay layers can provide an extremely effective geological barrier to solute migration.

During their investigations of the Opalinus Clay, the Swiss observed an auxiliary problem that needs special attention. This involves the evolution of gas within the repository near field that consists, predominantly, of H_2 produced by anaerobic corrosion of steel. From detailed investigations, the Swiss have concluded that "a strong case can be made that even conservative gas scenarios do not compromise the safety of the repository" (Zuidema et al., 2006).

1.2.2. Public Acceptance of the Development of Waste Isolation Projects

The reactions of the public to the development of radioactive waste isolation projects have varied considerably, and this has led to several different methods of handling these problems. In 1993, an extensive site-selection procedure by Nagra in Switzerland resulted in the nomination of a site at Wellenberg in Central Switzerland as the preferred location for an LILW repository. The principle of using Wellenberg as a repository site was accepted in public referenda in the local community, but in 1995, the action was blocked by a narrow margin at the cantonal level (Zuidema et al., 2006).

Between 1996 and 2001, Wellenberg was evaluated again, and a modified disposal concept to proceed with an exploratory drift received the necessary concession from the local government in September 2001. Despite the fact that many of their requirements were fulfilled, the 1995 opponents decided to fight the concession in the weeks leading up to the cantonal vote in September 2002. The result was a higher vote against the project than was obtained in 1995, and the site had to be abandoned for political reasons. The definition of a site selection procedure, including site selection criteria, is now under development by the Swiss Federal administration. Nagra is currently compiling technical-scientific material, mainly geological information, to support the site selection process (Zuidema, et al., 2006).

As was discussed in the *Third Worldwide Review*, Nirex in the United Kingdom had developed a favorable site for underground investigations at Sellafield in northwest England. They applied for permission to construct a URL at the site, but the application was rejected in 1997 by the local county council. When the rejection was upheld in a public inquiry, Nirex terminated the project and abandoned the site (Hooper, 2006).

Nirex reviewed what went wrong and decided to shift from being an entirely scientific organization to one that also encompasses social considerations. At the same time, changes have been made to make Nirex more accountable and transparent. After other organizations and the Government reviewed this situation, a new independent committee was established in November 2003—the Committee on Radioactive Waste Management (CoRWM)—to oversee the second phase of the consultation exercise. CoRWM is expected to recommend to Government its preferred long-term waste management options by July 2006 (Hooper, 2006).

A somewhat similar situation has developed in Canada. For almost 30 years, the Atomic Energy of Canada Limited (AECL) in Canada has been developing the concept for emplacement of nuclear fuel wastes in a deep geologic repository excavated in the plutonic rock of the Canadian Shield. In 1994, AECL submitted its Environmental Impact Statement on the concept to a federal Environmental Assessment Panel (EAP) for review. Public hearings associated with the review took place during 1996 and 1997. In 1998, the federal government completed its review of the concept and found it to be technically safe, and in compliance with current regulatory requirements. However, the review also concluded that there was not sufficient public support at that time to implement a repository siting program.

The government of Canada responded to the recommendations of the EAP and issued the Nuclear Fuel Waste Act (NFWA). In response to this Act, the Nuclear Waste Management Organization (NWMO) was established, October 1, 2002, by the nuclear utilities in Canada to study options and to recommend an approach for long-term management of Canada's used nuclear fuel. The NWMO was tasked to study approaches based on at least three methods:

1. Deep geological disposal in the Canadian Shield
2. Storage at nuclear reactor sites
3. Centralized storage, either above or below ground

The NFWA required the NWMO to submit its study to the Minister of Natural Resources Canada by November 15, 2005, after which the federal government would decide on the preferred approach for Canada. NWMO has not been able to proceed with a long-term solution that gives Canadians confidence that such an approach reflects and responds to their values, issues, and concerns. For this reason, NWMO adopted a study process designed to develop, collaboratively with the public, a management approach for the long-term care of used nuclear fuel that is socially acceptable, technically sound, environmentally responsible, and economically feasible.

During almost three years of dialogue with Aboriginal peoples, the public, and specialists, NWMO received very specific direction on both the way in which management approaches should be conducted, and the advantages and limitations of each as judged by interested Canadians. After reviewing each of the three options identified for study by NFWA, many Canadians suggested that an additional option should be considered, an option that would attempt to capitalize on the advantages of the other three approaches. This led NWMO to develop the Adaptive

Phased Management (APM) approach, and to launch a dialogue with Canadians about its appropriateness in the period June through August 2005. Overall, the majority of Canadians who engaged in these dialogues considered APM an appropriate approach for Canada, and the final report that was submitted by NWMO to the Government of Canada in November 2005 recommended this approach (Russell and Facella, 2006).

As was discussed in the *Third Worldwide Review*, the Nuclear Waste Management Organization of Japan (NUMO) was established in October 2000 as the implementing organization for HLW disposal. Its assigned activities include repository site selection, preparing relevant license applications, and construction, operation, and closure of the repository. As the first milestone in the siting process, NUMO announced the start of open solicitation, to the public, of volunteer municipalities for preliminary investigation areas (PIAs), with four documents published together as an information package on December 19, 2002. The information package, which provides basic information for supporting and promoting discussions by municipalities as to whether the repository plans could be accepted, has been sent to all (3,239) municipalities in Japan (Kitayama et al., 2006).

NUMO has chosen a volunteer approach to site selection in the belief that the support of local communities is essential to the success of this highly public, long-term project extending over more than a century. It is particularly important to promote public understanding of geological disposal, and to obtain and maintain public trust. To insure transparency, NUMO makes available diverse information relevant to its siting activities through the publication of documents, websites, etc., and will provide opportunities for residents in the vicinity of PIAs to voice their opinions. This approach presents particular challenges for the repository concept development program. With this in mind, NUMO intends to provide regular updates on how to pursue and apply repository concept development and site evaluation techniques (Kitayama et al., 2006).

1.2.3. A Successful Program of Site Selection in Finland

Finland has developed a very successful site-selection program, one that has effectively included public participation. Over 15 years ago, when the first fieldwork for an HLW repository was about to begin, Posiva set up a cooperation group with the residents of each community where investigations were to be undertaken. They started with four different sites, and they established four separate cooperation groups. Several meetings with each group, including field trips, were arranged annually, and at the end of each year, a written report was given to each group that summarized all results and future plans.

Before any significant commitment to a nuclear facility is made, Finnish law requires the passage of a Decision-in-Principle (DiP) by the government, which must include municipal approval. When all fieldwork at the four sites was completed, the final selection was based on the outcome of an environmental impact assessment carried out in 1997–1999. The final selection was a site at Olkiluoto in the municipality of Eurojoki, whose residents had rendered a strong vote of approval for the proposed site. An LILW repository has been in operation at Olkiluoto since 1992, and the local people are well aware of the procedures that were followed in getting permission to construct this facility (Ryhanen, 1991).

In December 2000, the Finnish government approved an application for the DiP that had been made, and on May 18, 2001, the Parliament ratified the decision. Finland is the first country in Europe to obtain this kind of governmental approval for an HLW repository site. The next step in the program is the construction of an underground research laboratory (URL) at the repository site. The facility is intended for final assessment of the previous conclusion on the suitability of the Olkiluoto site for a safe geologic repository. A positive outcome of this assessment should make it possible to proceed to submission of an application for the construction license. According to government guidelines, this submission would be planned for 2010. The construction of the URL has now started. Several hundred meters of tunnel have been excavated, and the implementation of underground investigations is under way (Vira, 2006).

1.2.4. The Swedish Program of Combining Development of Repository Technology with Site Investigations

The Swedish alternative for a deep geological repository to isolate HLW is called the KBS-3 method, and involves encapsulating the waste in copper canisters with cast iron inserts and then embedding the canister, surrounded by bentonite clay, in the bedrock at a depth of about 500 m.

The Swedish program constructed the Äspö Hard Rock Laboratory (HRL) on the island of Äspö outside Oskarshamn, in which extensive investigations have been carried out in deciding what a repository will look like, and what materials, special equipment, and technology will be used in its development (Lundqvist, 2001). In investigating this concept, SKB has been working on the design and fabrication of a copper canister that is nearly five meters long and weighs about 25 tons. The large size of this canister and the necessity of having to manoeuvre it around within a five-meter tunnel have necessitated the development of specialized emplacement equipment. In addition, SKB has been conducting research at Äspö on backfilling and plugging tunnels, retrieving canisters from sealed deposition boreholes, and creating a prototype repository to simulate the integrated function of the repository components. An overall goal for SKB is that the first stage of the deep repository should be ready for initial operation in 2017. Regular operation should then commence in 2023 (Lundqvist, 2006).

The siting process for the deep repository has now reached the site investigation phase. Two municipalities have been selected from an original list of eight. These are Oskarshamn in southeast Sweden and Östhammer in the region of northern Uppland. SKB is now compiling the material needed to choose the most suitable location, as well as to apply for a permit for an encapsulation plant in 2006 and for the deep repository in 2008. This work includes safety analyses based on site-specific repository designs and environmental impact statements (Lundqvist, 2006).

1.2.5. License Application for Yucca Mountain in the United States

As stated in the *Third Worldwide Review*, after reviewing the results of the site characterization process for Yucca Mountain, Nevada, the U.S. Secretary of Energy concluded that a repository at Yucca Mountain would perform in a manner that protects public health and safety. The Secretary recommended the site to the President in February 2002; the President agreed and recommended to Congress that the site be approved. The Governor of Nevada submitted a notice of disapproval, but both houses of Congress acted to override the disapproval. In July 2002, the President's approval allowed the Department of Energy (DOE) to begin the process of submittal of a license application for Yucca Mountain as a site for the nation's first repository for spent nuclear fuel and high-level radioactive waste (Arthur and Voegele, 2006).

The safety of a repository at Yucca Mountain will be assured by the performance of the natural and engineered features of the site, acting in concert, to prevent or delay the transport of radioactive materials to where the public could eventually be exposed to them. Existing U.S. Environmental Protection Agency (EPA) standards for Yucca Mountain after permanent closure address all potential pathways of radiation exposure and limit an individual's annual radiation exposure from all pathways to a level consistent with what the EPA considers an acceptable risk to society for 10,000 years following closure. The EPA also sets a standard by which to protect the groundwater around Yucca Mountain. This standard sets specific limits for the concentration of different types of radioactive particles in the groundwater. Further, the DOE must assess the consequences of a potential inadvertent human intrusion into the repository at some time in the future.

Calculations performed to date indicate that a repository at the Yucca Mountain site will likely meet all of the EPA requirements for 10,000 years following permanent closure. The standard also includes a requirement to assess the performance of the Yucca Mountain repository at time of peak dose. The evaluation has been done, and the results have been reported in the Environmental Impact Statement. However, since that time, a Federal Court has vacated that standard to the extent that it only required compliance for 10,000 years. The EPA and the Nuclear Regulatory Commission (NRC) are currently assessing paths forward; their potential actions will have a bearing on the license application process.

Before a repository can be constructed and operations can begin, the DOE first must apply for and receive construction authorization from the NRC, an independent agency of the federal government. This construction authorization must then be followed by an application for, and receipt of, a license to receive and possess waste. An amendment to the license will be needed to close the inquiry. The Department of Energy is in the process of preparing the license application (Arthur and Voegele, 2006), and it is expected to be submitted to the NEC in 2008, with repository operations to be started in 2020.

1.2.6. Development of Multinational Repository Concepts

The wide collaboration of many different organizations and countries in pursuing joint research projects in the URLs of Europe has led to the idea of developing multinational organizations for shared repositories. IAEA published an overview on the topic, emphasized the potential advantages, and also initiated a high-level international group of experts to look into this matter.

The European Commission (EC) also included the topic of shared regional European repositories in the Nuclear Package of legislation that it has been trying to put through Parliament. The EC has provided direct support for the multinational project SAPIERR, which involves a working group of representatives from 14 nations in a pilot project on regional repositories in Europe. Further international cooperative efforts have been organized by the not-for-profit association Arius, which currently includes members from eight countries. In a slightly different approach to the “partnering “ concept, Russia has expressed interest, at the governmental level, in possibly hosting an international repository, and two international meetings took place in 2002 to discuss this initiative further (McCombie and Chapman, 2006).

1.2.7. International Training in Geological Disposal

The international community in nuclear energy has been studying the geological disposal concept for almost 50 years. For the last 30 years, there has been an intensive effort to answer outstanding questions, develop the technology to a practically implementable level, and bring the concept to fruition. The success picture is far from uniform. Some disposal programs are on the threshold of repository construction. Others have almost reached this point, then suffered total setbacks that have taken them back to the options comparison stage of the 1970s. Not only have these projects lost all momentum, but they have lost most of the skills and experience built up over decades (Chapman et al., 2006). Programs in many other countries are just in their initial site characterization phase. (Witherspoon and Bodvarsson, This Volume).

Ensuring that we have sufficient skills internationally over many decades into the future is thus regarded as an important challenge to the current generation of managers. It is particularly relevant today, when many of those involved in developing the geological disposal concept are nearing retirement (Chapman et al., 2006). The experience of what works and what does not, both technically and in terms of stakeholder involvement, are vital knowledge if national programs wish to maximize their efficiency and avoid some of the pitfalls of the past.

With this background in mind, there have been several initiatives started in the last two to three years to ensure proper provision of education and training in radioactive waste management—in particular, geologic disposal. Notable among these initiatives are:

- The IAEA Network of Centres of Excellence, *Training in and Demonstration of Waste Disposal Technologies in Underground Research Facilities*
- The establishment of the International Training Centre (ITC) School of Underground Waste Storage and Disposal in Switzerland
- The CETRAD project in the European Union
- The establishment of the World Nuclear University by the World Nuclear Association.

These initiatives are briefly reviewed by Chapman et al. (2006).

1.2.8. Need for a Long-Term Science and Technology Component in Radioactive Waste Isolation Programs

At the end of this volume, Budnitz (2006) raises an interesting issue, that of the need for a long-term science and technology component in radioactive waste isolation programs. The fundamental rationale for such a program is to provide a dedicated focus for longer-term science and technology activities that ultimately will benefit the whole repository mission. Such a program, separately funded and with a dedicated staff (separate from the “mainline” activities to develop the repository, the surface facilities, and the transportation system), can devote itself exclusively to the development and management of a long-term science and technology program.

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Table 1.1. Developments in radioactive waste isolation from 2006 *Fourth Worldwide Review*   TBD—To be determined

Country	Lead Organization	Potential Sites	Prospective Rock Types	Status of Site Characterization	Prospective Design/Engineered Barrier	Near-Term Plans
Belarus	Institute of Geological Sciences	Central Belarussian Massif and Pripyat Graben	Granite Palygorskite Clay	Geologic and geophysical investigations of region, numerical models of radionuclide migration developed	TBD	TBD
Brazil	Brazilian Nuclear Energy Commission (CNEN)	TBD	TBD	TBD	LLW in near surface repository, HLW stored at reactor	TBD
Bulgaria	Geological Institute of Bulgarian Academy of Sciences	Clayey formations near Kozloduy NPP	Neogene clayey formations	Drilling to 4,300 m. Investigating seismic profiles, geochemical studies, and hydrogeological conditions.	LILW in pre-stressed concrete cubes at NPP; HLW in deep repository.	LILW and HLW repositories in clay near Kozloduy NPP
Canada	Nuclear Waste Management Organization (NWMO)	TBD	Granite if deep geological disposal selected	Since 1978, URL and other facilities used in site characterization studies and technology development.	Depends which option selected—deep rock disposal, storage at NPP, or centralized storage	Fourth option of Adaptive Phased Management recommended by NWMO
China	Beijing Research Institute of Uranium Geology (BRIUG), China National Nuclear Corporation (CNNC),	Jiujing, Xinchang Xiangyangshan, and Yemaquan blocks in Beishan region of Gansu Province	Monzonitic granite	Jiujing and Yemaquan sections have had surface geology, hydrogeological and geophysical surveys, and borehole drilling. Bentonite buffer and radionuclide migration studies performed.	TBD	Similar investigations in other two sections. URL will be designed and constructed in the period 2015–2030.
Czech Republic	Radioactive Waste Repository Authority (RAWRA)	LLW at Dukovany NPP in near surface repository, TBD for HLW	Granite for HLW	Six granite sites subjected to site characterization studies. Results by October 2005. Program will be stopped until 2009.	Large physical model based on KBS-3 concept of Sweden being used at Czech Technical Univ. to investigate behavior.	Two sites to be selected for deep geological repository by 2015. Confirm one site by 2025. Prepare for construction of URL and long-term experiments by 2030. Operation of repository by 2065.
Finland	Posiva Oy	Olkiluoto near NPP at Eurajoki, with existing repository for LILW and new repository site for HLW	Granite	Construction of ONKALO characterization facility started. Reach 420 m level by 2010.	KBS-3 design of copper-steel canister, bentonite buffer, and backfill.	Carry out comprehensive investigations while maintaining required safety levels and complying with plans for repository. Plan to submit application for repository construction license in 2012.

Country	Lead Organization	Potential Sites	Prospective Rock Types	Status of Site Characterization	Prospective Design/Engineered Barrier	Near-Term Plans
France	Agence Nationale pour la Gestion des Dechet Radioactifs (ANDRA)	Meuse/Haute-Marne clay site and a nonspecified granite site	Argillite at Meuse site and granitic batholiths	URL shaft at Meuse site to 490 m with horizontal drift at 445 m in Callovo-Oxfordian argillite. Participating in 5 granitic experiments in foreign URLs.	Investigating preliminary concepts for designs of disposal cells in clays, using tunnels and caverns	Measured permeabilities of clay are so low that diffusion may be the controlling feature. Additional measurements to confirm first results. Characterization of disturbed zone is needed. Testing of plugging methods and thermal response of argillite.
Germany	Federal Office for Radiation Protection (BfS)/Federal Institute for Geosciences and Natural Resources (BGR)	Gorleben salt dome, Konrad iron ore mine	Rock Salt	Permission to emplace waste with negligible heat generation in Konrad mine approved in 2002, but objections have stopped process. Gorleben has been found suitable for repository, but report of investigations has not been published.	Steel container, backfill material depending on waste type and host rock	Investigations in clay and granite are being carried out in URLs of other countries.
Hungary	Public Agency for Radioactive Waste Management (PURAM)	Üveghuta site for LILW. Boda site for HLW	Granite at Üveghuta site, Boda claystone at Boda site	Üveghuta investigated (1990s), then selected as LILW repository. IAEA agreed. Further confirmation work (2002–2003) approved by Hungarian Geological Society. In 2004, exploration program for HLW repository at Boda restarted.	At Üveghuta site, waste drums and disposal containers to be emplaced in tunnels with clay in backfill material. TBD at Boda site	Site characterization and repository design at Üveghuta to continue. Public outreach program to continue. For HLW, start R&D work at Boda (2005–2008), start construction of URL and elaborate R&D work (2009–2012).

Country	Lead Organization	Potential Sites	Prospective Rock Types	Status of Site Characterization	Prospective Design/Engineered Barrier	Near-Term Plans
Japan	Nuclear Waste Management Organization of Japan (NUMO), Japan Nuclear Cycle Development Institute (JNC) Radioactive Waste Management Funding and Research Center (RWMC)	TBD	Crystalline or sedimentary rocks. URLs at Mizunami and Horonobe	Siting process for HLW repository developed in 3 stages. At each stage, NUMO solicits opinions of local residents, governors, and mayors. In 2002, information package supporting participation by public sent to all (3239) municipalities in Japan.	Vitrified waste in steel over-pack embedded in bentonite and emplaced either in tunnels or in vertical holes drilled from bottom of tunnels	NUMO is promoting R&D activities aimed at providing scientific background of siting factors for selecting PIAs (preliminary investigation area). JNC is pursuing site characterization methodologies at URL at Mizunami in crystalline rock and at URL at Horonobe in sedimentary rock. RWMC is pursuing R&D activities related to sociological issues and advanced technological options.
Korea	Korea Hydro & Nuclear Power Co. (KHNP) for LILW. Korea Atomic Energy Research Institute (KAERI) for HLW	Wolsung site for LILW	Granite for LILW repository. Plutonic rocks screened as primary host rock for HLW repository.	Site suitability study for Wolsung site was completed in 2005 for LILW repository. Detailed site study will be started in 2006.	Rock cavern or cement grouted vault for LILW. Encapsulate HLW in corrosion resistant containers in boreholes with bentonite buffer, drilled in tunnels at depth of ~500 m.	Safety analysis and design will be started in 2006 for LILW repository. Next step for HLW repository is further development of repository concept using field data from in situ investigations at specific sites.
Latvia	Radiation Safety Centre	TBD	TBD	RW from Salaspils Res. Reactor, being closed, will be disposed of in near-surface LILW-SL repository at Baldone. SF and long–lived RW to be moved out of Latvia as per USA-IAEA-RUSSIA cooperation project.	TBD	Latvia is aware of recent development of regional disposal concept and is evaluating the possibilities of using this option.
Lithuania	Lithuanian Energy Institute	Existing solid LILW storage facilities at Ignalina NPP.	For deep repository: crystalline clay, anhydrite, rock salt.	Based on 4 year period of investigations: Lower Cambrian and Lower Triassic clayey formations are potential alternatives to crystalline rocks. Crystalline basement rock one of best prospects for repository.	Proposed repository design is based on the KBS-3 concept developed by SKB in Sweden.	Activities related to implementation of landfill repository for very low level waste at the Ignalina NPP site were started in 2003. The Lithuanian Energy Institute, with support of Swedish experts, is working on the development of the repository concept and the generic safety assessment of a repository in crystalline rock in Lithuania.

Country	Lead Organization	Potential Sites	Prospective Rock Types	Status of Site Characterization	Prospective Design/Engineered Barrier	Near-Term Plans
Mexico	Instituto Nacional de Investigaciones Nucleares (ININ)	TBD	TBD	TBD	TBD	The need exists for people within the nuclear industry in Mexico to provide definitive solutions within the next 10 years concerning disposal of LILW generated at Laguna Verde NPP and at other facilities.
Russia	All-Russia Designing and Research Institute of Production Engineering	Krasnoyarsk Territory, Kola Peninsula, Mayak Combine	Sand and sandstone for liquid wastes. Hard rock for solid wastes.	Liquid waste isolation has worked satisfactorily since 1963. Isolation of solidified waste in hard rock massifs, mined out areas, and permafrost have been investigated.	TBD	Disposal of liquid radioactive waste to be completed by 2015 and projects will be shut down. Solidified waste will be stored at surface while geological repositories are being researched and constructed, with operation after 2020.
Slovak Republic	DECOM Slovakia	Granitic rocks in Tribec, Veporske vrchy, Stolicke vrchy Mountains, Argillaceous formations in Ceyova vrchovina Upland and Rimavska Kotlina Basin	Granites, siltstones, and claystones	Five localities have been selected as prospective sites for a deep repository. Three are in granite rocks, and two are in argillaceous rocks. Further reduction of sites expected, but two should be considered up to 2005. Decision on selection of candidate sites expected around 2010. Commissioning of deep geological repository by 2037. Need for URL is considered to be appropriate.	Proposed disposal container with 7 WWER-440 SNF assemblies to have outer wall of carbon-steel coated with nickel and inner wall of stainless steel. The inner cask would be an aluminum alloy.	After a period of project stagnancy, restarting of research and development activities is expected in near future. Siting, safety analyses, and coordinating activities will be primary areas. This should lead to an acceptable candidate locality and demonstrate feasibility of construction, operation, and closure of the repository. Slovakia has also taken a very active role in international cooperation related to geological disposal, and these activities will continue.
Slovenia	Agency for Radwaste Management ARAO)	LILW site should be identified in 2005 and site characterization completed in 2007.	TBD	Site selection process requires participation by public and a bidding process to make a selection. If process is completed successfully in 2005, repository will be in operation by 2013. SF from Krsko NPP stored at plant, to be disposed of in a deep repository. Any action to be postponed until end of NPP operation.	It is assumed that SNF will be encapsulated according to the Swedish concept.	LILW site suitability investigations to continue in accordance with the requirements for appropriate participation by public.

Country	Lead Organization	Potential Sites	Prospective Rock Types	Status of Site Characterization	Prospective Design/Engineered Barrier	Near-Term Plans
South Africa	South Africa Nuclear Energy Corporation (Necsa)	Pelindaba dry-storage site. Vaalputs RW disposal facility	Granitic	Drilling in Vaalputs area (1996) found excellent granitic rock, but work was stopped. Recently, Necsa has established a collaborative program with a number of outside researchers to investigate the geology, tectonics, and environmental aspects of the Vaalputs site.	TBD	If geologic disposal is to be part of national policy, all stakeholders are to be involved. International cooperation is essential, various options (including regional repository) to be included.
Spain	Enresa National de Residuos Radioactivos (ENRESA)	TBD	Clay, granite	Spain has been pursuing the development of generic technology for deep geologic disposal (DGD) of HLW in clays and granites by way of a highly selective participation in international R&D programs, mainly those of the EU. Results of this effort are geo-referenced in a GIS system.	Carbon-steel canisters embedded horizontally in bentonite buffer spaced 2 m apart in drifts	The change in government that occurred in 2004 has led to the modification of the management strategy, which will be reflected in a new General Radioactive Waste Plan to be approved in 2005. This will establish, as the top and most urgent priority, the availability of a Centralized Temporary Storage (CTS) facility, to be operable by 2010. Consequently, with CTS in operation, the DGD of HLW will be placed on a secondary level.
Sweden	Swedish Nuclear Fuel and Waste Management Co. (SKB)	Forsmark site in Oskarshamn municipality, Southeast Sweden. Laxemar site in Osthammar municipality, Northern Upland.	Metagranitoids Monzodiorite	Characterization of the surface geological and ecological conditions of Forsmark site have been completed. Drilling of ten boreholes (1,000 m) and shallower percussion holes (depth of ~200 m) have been completed. Comprehensive preliminary site description has been prepared. Communication program with nearby residents, the public, the municipality, and other local stakeholders has been established. Laxemar site characterization program is ongoing. It was determined that hydrogeologic and geochemical conditions are favorable for deep geological disposal.	Encapsulating fuel in copper canisters with cast-iron inserts. Embedding each canister in a tunnel (filled with bentonite) at a depth of about 500 m.	SKM is now (a) compiling the material needed to choose the most suitable site location and (b) going to apply for a permit for the encapsulation plant in 2006 and for the final repository in 2008. The work includes safety analyses based on site-specific repository designs and environmental impact statements.

Country	Lead Organization	Potential Sites	Prospective Rock Types	Status of Site Characterization	Prospective Design/Engineered Barrier	Near-Term Plans
Switzerland	National Cooperative for the Disposal of Radioactive Waste (NAGRA)	Wellenberg repository site was accepted in public referenda in the local community, but blocked at the cantonal level in 1995. Revised concept was blocked in second cantonal vote in 2002, and site was abandoned for political reasons. Zürcher Weinland site is considered now for a geological repository of SNF/HLW/ILW. The definition of a site selection procedure is now under development by the Federal Office of Energy.	Three types of host rock are considered, the crystalline basement or one of the two overlying, low-permeability sediment layers.	The Opalinus Clay has been investigated at the Mont Terri URL in northern Switzerland. A potential siting area in the Opalinus Clay of the "Zürcher Weinland" has been investigated for its structural, hydro-geological and geochemical properties. Field investigations at the Grimsel Test Site (crystalline rock) are ongoing.	The multibarrier conceptual repository design with deep disposal (about 500 m to 1 km below the surface) in a specially constructed facility. In-tunnel emplacement of SNF/HLW waste packages in sediment or crystalline basement. Engineered barriers will be designed in addition to the vitrified waste.	Nagra is currently compiling technical-scientific material to support the site selection process. The major next step in Nagra's program, site selection, is expected to commence at end of 2006/ beginning of 2007, after the legislative and political framework for radioactive waste disposal currently under discussion is finalized.
United Kingdom	DECOM Slovakia	Granitic rocks in Tribec, Veporske vrchy, Stolicke vrchy Mountains, Argillaceous formations in Ceyova vrchovina Upland and Rimavska Kotlina Basin	Granites, siltstones, and claystones	Nirex had developed a favorable repository site at Sellafield to construct URL at that site was rejected in 1997. When rejection was upheld in public inquiry, Nirex terminated the project and abandoned the site. Nirex reviewed what went wrong and decided to shift from being an entirely scientific organization to one that also encompasses social considerations. New, independent CoRWM established in 2003 to oversee second phase of consultation exercise.	The current Nirex Phased Geological Repository Concept considers an option of not backfilling the waste-filled caverns for as long as 300 years, during which waste could be monitored and retrieved.	CoRWM will recommend to Government its preferred long-term waste-management options by July 2006.

Country	Lead Organization	Potential Sites	Prospective Rock Types	Status of Site Characterization	Prospective Design/Engineered Barrier	Near-Term Plans
United States	United States Department of Energy (DOE)	Yucca Mountain, Nevada	Volcanic tuff	Yucca Mountain site characterization has been completed. In February 2002, U.S. Secretary of Energy recommended the site to the President, who agreed and recommended to Congress that site be approved. Governor of Nevada submitted notice of disapproval, and both houses of Congress acted to override disapproval. In July 2002, the President's approval allowed DOE to begin the process of license-application submittal for Yucca Mountain.	Waste packages will consist of two concentric cylinders (the inner cylinder made of stainless steel, and the outer one made of a corrosion-resistant nickel-based alloy), and will have three lids to provide a leak-tight closure. Waste packages will be placed horizontally in the emplacement drifts. Titanium drip shields will be installed to protect waste packages from potentially dripping water and rockfall. Safety of repository will be assured by the performance of the natural and engineered features of the site, acting in concert, to prevent or delay the transport of radioactive materials to where the public could eventually be exposed to them.	Before repository can be constructed, DOE must first apply for and receive construction authorization from NRC. This authorization must then be followed by an application for, and receipt of, a license to receive and possess waste. An amendment to the license will be needed to close the inquiry. DOE is in the process of preparing the license application and expects to submit to NRC in 2008. Repository operations are expected to commence in 2020.

Chapter 2

Progress Towards a Deep Geological Repository in the Republic of Belarus

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ABSTRACT. Granite (located in the Central-Belarussian Crystalline Massif), salt deposits, and palygorskite-bearing clayey rocks of the Pripyat Graben have previously been defined as the best geological rock for a radioactive waste repository (RWR) (Kudelsky, 2001). In this report, we present the results of a more detailed assessment of the palygorskite zone, as a further step towards finding a site for an RWR. Palygorskite is a zeolite-like clay mineral and a well-known natural sorbent (like sepiolite and montmorillonite). The palygorskite bed is situated in the southern part of Belarus in a relatively thinly populated area, at a depth of 90–120 m, with a thickness of 250 m. This zone coincides with the upper marl series in the lithologic profile of the Upper Devonian, lying over salt deposits. Besides palygorskite, illite 1 Md and chlorite-smectite are also present in the clayey fraction of the rock. This palygorskite-bearing series is a plunging geological formation in the section from west to east, and some specific profile types may be observed. This central zone is the most suitable one for an RWR. No sandstone layers are located in this zone, and the permeability is quite low. A clay tuff layer, with a thickness of 0.2–1.7 m, lies at the bottom of the palygorskite zone and extends over the entire area. This layer provides additional isolation for further groundwater protection. The palygorskite-bearing rock lying over salt deposits provides good characteristics for an RWR.

2.1. INTRODUCTION

No nuclear power plants (NPP) exist in Belarus at this time. However, the search for suitable geological beds for radioactive waste disposal remains an important task, because the construction of an NPP is planned in the future. The main problems with regard to radioactive waste disposal and the best geological zones for intermediate-level waste (ILW) and high-level waste (HLW) disposal in Belarus were discussed in Kudelsky and Yasoveyev (1991) and Kudelsky (2001). Granitic rocks of the crystalline basement, salt diapir domes, and the palygorskite-bearing zone of the Pripyat Graben have been defined as the best prospective sites for an RWR. More detailed characteristics of the palygorskite zone are considered in this study as part of a further step toward constructing a deep radioactive waste repository.

Palygorskite is a well-known zeolite-like clay widely used for ionic and molecular purification of liquids and gases. Upper Devonian sedimentary rocks of the Pripyat Trough, which contain palygorskite, show the best RWR characteristics in terms of environmental protection. Palygorskite-zone location, its lithologic-stratigraphical section, rock parameters, clay mineral characteristics, and the best places for RWR disposal are the focus of this report.

2.2. PALYGORSKITE ZONE LOCATION

The Pripyat Trough is situated in southern Belarus. The palygorskite zone in this region is 120 km long and 40 km wide. An Upper Devonian palygorskite-bearing zone covers an area of about 2,000 km^2, with a thickness reaching 250 m. This zone is found in the northwestern part of the trough, near the Starobin salt field (Figure 2.1).

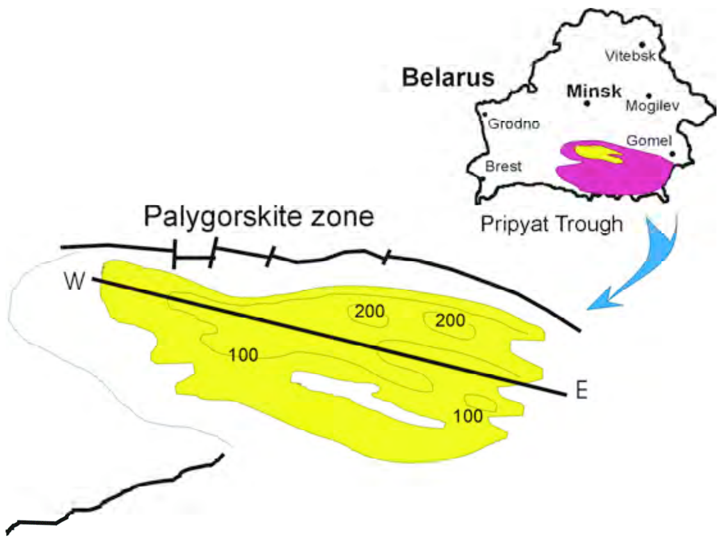

Figure 2.1. Location of palygorskite zone in the northwestern part of the Pripyat Trough

2.3. GEOLOGICAL PROFILE

The Upper Devonian consists of three lithologic series: a bottom sulfate-marl series lying on the salt-bearing sedimentary rocks, an intermediate dolomite-marl series, and an upper marl series. In some areas, the carboniferous sediments cover the Upper Devonian deposits. Marl, dolomite, sulfite (gypsum and anhydrite), clay, limestone, and oil shale are among the major rocks, with marl making up more than 50% of the total profile. Significantly, there are no sandstone beds in the profile, so the hydraulic permeability of the rock should be low. There are three main clay minerals present: illite 1Md, palygorskite, and a chlorite-like mineral; the illite 1Md provides some correlation between the lithological series and clay mineral series sequences. The bottom clay, Subzone I (thickness 100–180 m), contains an illite 1Md-chlorite association and correlates with the sulfate-marl lithological series. Chlorite is a common admixture within Subzone I, and its concentration varies from 18 to 80%. Moving upwards, the next clay, Subzone II, contains a casual chlorite appearance (thickness 20–85 m), with illite 1Md prevailing in this subzone. Subzone II correlates with an interval having a gradual transition from the lithological sulfate-marl series to the dolomite-marl series. Subzone III contains illite 1Md exclusively, and correlates with the dolomite-marl series. Subzone IV is a palygorskite-bearing clay that contains an illite 1Md and chlorite admixture (Figure 2.2).

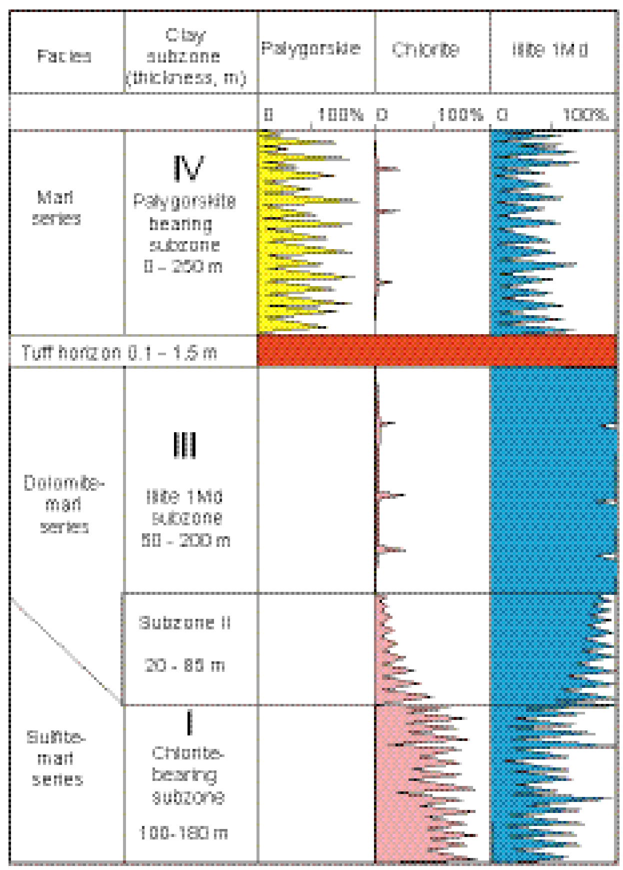

Figure 2.2. The lithologic series and clay subzones in the geological profile

2.4. CLAY MINERALS

Dioctahedral illite 1Md prevails in the profile, with an average crystal-chemical formula of:

$$K_{0.55}\ (Al_{1.04}\ Fe_{0.28}\ Mg_{0.57})_2\ \ (Si_{3.66}\ Al_{0.34})4\ \ O_{10}\ (OH)_2 \quad \text{(Eq. 1)}$$

The chlorite-like minerals are the tri-octahedral, disordered, mixed-layered chlorite-smectite clay minerals. Their average formula is:

$$(Fe_{0.70}\ Al_{0.65}\ Mg_{4.55})_6\ (Si_{3.70}\ Al_{0.30})_4\ O_{10}\ (OH)_8 \quad \text{(Eq. 2)}$$

Illite and chlorite are phyllosilicates with isometric sheet-like microcrystals. Palygorskite is a magnesium alumo-silicate clay mineral with a fibrous microcrystal morphology and chain structure. It is a zeolite-like mineral with microchannels (Section 4Å x 6.5Å) in its structure, so that ionic and molecular absorption processes take place within it. This mineral is widely used for gas and liquid purification in various technological processes. Smectite and palygorskite clays are known as the best linear barriers for RWRs.

2.5. DESCRIPTION OF THE PALYGORSKITE DEPOSIT

The Palygorskite subzone extends over the northwestern part of the Pripyat Trough. Its depth and thickness vary greatly from west to east. The top lies at a depth of 90–120 m; the bottom depth varies from 100 to 850 m. Four profile types may be observed in the palygorskite zone (Figure 2.3). This zone is completely eroded west of the ore deposit, but in the second profile its thickness is ~30–50 m, and then in the third profile its thickness increases to 170–250 m. The zone is very unstable east of the ore deposit. Palygorskite concentrations vary from 0 to 70% in the thin seams because of the variation in sedimentation conditions. Thus, the second and third profile types are the most suitable for an RWR.

There are some tuff layers in the Upper Devonian overlying the salt formation. The most consistent tuff layer lies at the bottom of the palygorskite-bearing bed, indicating that this lithological series has been formed under the influence of regional volcanic activity. The tuff layer extends over the entire palygorskite zone, and initial deposits of tuff in the Upper Devonian formation have been transformed into clayey bentonite-like layers,

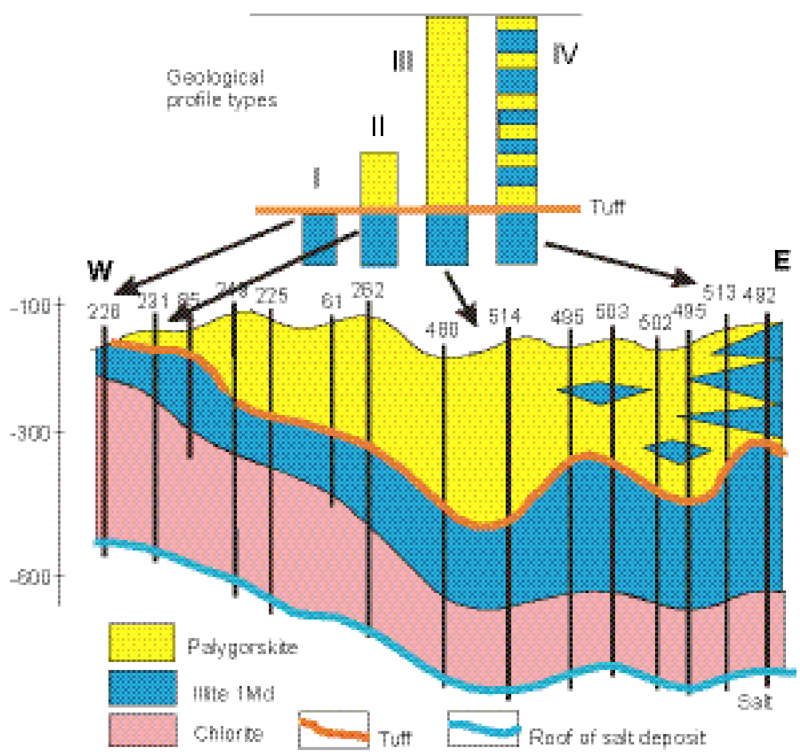

Figure 2.3. Palygorskite deposit section

which are white and slick. Nevertheless, they consist, not of smectite, but mostly of illite 1Md, which has the structure of ordered di-octahedral illite with lash-like microcrystals.

This tuff layer has a specific structure, with the bottom showing a sharply cut-off border. Its upper border shows a gradation from clear tuff to marl, such as in laminated rocks. This tuff layer inhibits the migration of moisture (e.g., CaO concentration in tuff decreases to 0.1%) because of its hydraulic insulating properties, and thus it could be used as a natural barrier. A repository site could be located above the tuff horizon, so that groundwater pollution may be prevented in case there is some leakage of radioactivity.

The best sites for RWR disposal may be located in the central area of the palygorskite-bearing zone at a depth of 100–120 m (Figure 2.3). The Palygorskite shows a consistent distribution and stable presence within the entire profile. The main rock type is clayey marl, which contains calcite (20–40%), palygorskite (40–60%) and illite 1Md (20–50%)(Figure 2.4). There are no sandy beds in the palygorskite zone, and the rock hydraulic permeability has low values.

2.6. CONCLUSIONS

An Upper Devonian clay bed could be used for deep RWR disposal in Belarus. The site could be constructed at an optimal depth of 100–120 m. Marl is the prevalent rock, with a palygorskite concentration of 40–60%. Palygorskite is a natural zeolite-like material, and consequently is a good prospective geological layer for radioactive waste storage. A clayey tuff layer at the bottom of the palygorskite subzone has low hydraulic permeability, and thus may be used as a natural barrier.

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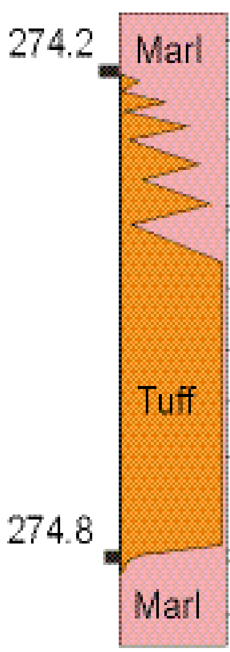

Sample	CaO	SiO_2	Al_2O_3	MgO	Fe_2O_3	K_2O
12m	9.6	45.6	8.4	8.7	4.3	4.2
11t	0.6	55.3	10.8	11.0	5.3	4.1
10m	2.6	54.4	13.1	13.7	4.7	3.3
9t	0.8	55.6	10.2	10.2	4.9	4.8
8m	2.2	54.9	11.6	13.1	4.6	3.8
7t	0.6	56.6	9.3	12.8	4.5	5.1
6t	0.3	57.8	11.7	10.2	3.5	5.3
5t	0.5	57.2	10.1	10.2	5.0	4.6
4t	0.1	56.8	9.9	7.8	4.5	5.1
3t	0.7	57.6	11.2	9.1	4.8	4.7
2t	1.1	58.7	11.3	8.7	4.4	4.8
1m	19.9	37.5	6.9	6.8	1.9	3.3

Figure 2.4. Tuff layer in Borehole 64 at a depth of 274.2–274.8 m

Chapter 3

Radioactive Waste Management in Brazil

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3.1. INTRODUCTION

The Brazilian nuclear program involves the operation of several nuclear and radioactive installations. Two nuclear power plants are in operation and one is under construction. Angra 1 is a 657 MWe gross/626 MW net, 2-loop PWR and Angra 2 is a 1,345 MWe gross /1,275 MWe net, 4-loop PWR), both located in Angra dos Reis, in the state of Rio de Janeiro. A third facility, Angra 3, was a 1,312 MWe gross/1,229 MW net, 4-loop PWR, whose construction was interrupted in 1991 and has not yet resumed. The two now-functioning reactors are operated by the engineering company Eletrobras Termonuclear S. A. (ELETRONUCLEAR).

At present, there are two functioning uranium mine and milling facilities operated by the Brazilian Nuclear Industry (INB). The first such mine was operational from 1982 until 1991 (in Poços de Caldas, located in the state of Minas Gerais). All the economically recoverable uranium has been extracted from that mine, and currently no mining activity is going on there. However, the uranium treatment facility there is still operational and has been used to process other source material from the second Brazilian mine, located in Caetité, in the state of Bahia. This second mine has the capacity for treating 100 t/yr of U_3O_8, as well as the production of lanthanum chloride and cerium hydroxide. A new mining facility, Catetité, operational since 2000, includes reserves of 100,000 tons of U_3O_8 and a capacity of 400 t/yr of yellow cake (U_3O_8), which can be expanded to 800 t/yr. Moreover, INB plans to open a fuel element complex (FEC) located in the state of Rio de Janeiro, which will include a reconversion and fuel-fabrication plant. This enrichment plant is expected to be in operation by end of year 2005.

In addition, Brazil has four research reactors, all of which are owned and operated by CNEN. The first (and oldest in Latin America), IEA-R1, built in 1956 within the U.S. Atoms for Peace program, is located at the Nuclear and Energetic Research Institute (IPEN), on the São Paulo University campus at São Paulo. With a maximum power of 5 MW, it is used for physics experiments and radioisotope production, such as for I-131, Sm-153, and Mo-99. The second reactor, IPR-R1, a 100 kW Triga model, operating since 1960 at the Centro de Desenvolvimento de Tecnologia Nuclear (CDTN), is located on the campus of the Federal University of Minas Gerais, in Belo Horizonte, and is used mainly for research work. The first fuel-assembly replacement of the reactor is expected to occur in 2010.

The third Brazilian Nuclear Research reactor, ARGONAUTA (built in 1965), located at the Institute of Nuclear Engineering (IEN) on the campus of the Federal University of Rio de Janeiro, can operate at a maximum power of 1 kW/h. This low reactor-burnup rate (approximately 0.25 MW/day—similar to IPR-R1) allows for efficient storage of spent fuel. This reactor is used for training, including sample irradiation. The last reactor, IPEN MB-01, also located at IPEN, is the result of a national joint program developed by CNEN and the Navy, and is a basic water-tank-type critical facility rated 100 W. This reactor is mainly used for simulation of low-level radioactive waste (LRW) and research in reactor physics (with a burnup rate of below 0.1 MW/day).

Brazil's nuclear program also includes:

- Pilot-scale fuel-cycle facilities, including one plant for converting uranium to UF_6 and another for uranium enrichment
- 3,248 medical, industrial, and research facilities
- One industrial facility for processing of monazite sands

3.2. WASTE CLASSIFICATION

Brazil has adopted the radioactive waste classification system established by the International Atomic Energy Agency (IAEA), which organizes radioactive wastes into three categories, as shown on Table 3.1. (Note that those radionuclides designated "short-lived" are those with half-lives of ~30 years, such as Co-60, Sr-90, and Cs-137.).

The major sources of radioactive waste in Brazil are, at present:

- The Angra I and II Nuclear Power Plants

Figure 3.1. The Angra I and II Nuclear Power Plants

- The Monazite Processing Industry, which is being decommissioned
- The 3,500 m^3 of low-level waste resulting from the decontamination work performed in Goiânia, following the 1987 accident that involved a 1375 Ci therapy source
- The 3,248 medical, industrial, and research facilities

Table 3.1. IAEA Waste Classification

Category	Characteristics	Disposal Options
Exempt Waste (IAEA-1987)	Activity levels equal to or below exemption limits based on a maximum impact dose of 0.01 mSy/y for the public	No radiological restriction
Low- and Intermediate-Level Waste:	Activity levels above exemption limit and heat generation equal to or below 2 kW/m^3.	
a. Short-lived waste	Long-lived alfa-emitter content equal to or below 4,000 Bq/g and an average specific activity of all radionuclides in the package (immobilized) of below 400 Bq/g).	Near-surface or geological repository (Lavie-1984, IAEA-1985, INTERA 1987, AECB-1985, AECB-1987, AECB-1998)
b. Long-lived waste	Radionuclide alfa-emitter concentration above limits cited before for short-live radioactive waste	Geological repository (IAEA 1981, IAEA-1989)
High-Level Waste	Heat generation above 2 kW/m^3 and alfa-emitter concentration above the limits allowed for low- and intermediate-level waste—short lived (a)	Geological repository (IAEA 1981, IAEA-1989)

Waste generated by the Uranium Mine and Milling Industrial Complex, although significant in volume, is kept at the site, in a dam especially built for this purpose .

3.3. RADIOACTIVE WASTES FROM THE NUCLEAR POWER PLANT

From 1981 to 2003, the first Brazilian Nuclear Power Plant, Angra I—a Westinghouse two-loop pressurized water reactor of 626 Mwe (Figure 3.1)—generated a total of 1,953.3 m^3 of solid/solidified waste, with an accumulated activity of ~304 TBq. Table 3.2 shows the quantities and percentages of low-level nuclear waste (LLW) and intermediate-level nuclear waste (ILW) generated by Angra I-NPP.

Since the Brazilian reprocessing program has not yet been clearly defined, the Angra I spent fuel stored at the on-site reactor basin (containing 546 spent fuel elements) poses by far the most difficult disposal problem, both technically and politically.

3.4. RADIOACTIVE WASTE FROM FUEL CYCLE AND MONAZITE PROCESSING FACILITIES

The uranium mining and milling industrial complex (CIPC), located at the Poços de Caldas plateau in the Brazilian state of Minas Gerais, has produced, from 1982 to 1991, 1,170 tons of ammonium diuranate. The waste generated from this process is kept in a 29.2 hectar dam system, with an actual volume capacity of 1 million m^3. It is estimated that 0.8 TBq (130 Ci) of U-238,

Table 3.2. Nuclear Power Plant Radioactive Waste Inventory

	Number of Packages					
Year	**Cartridge Filters 208l**	**Evaporator Concentrates 208l**	**Noncompressible (1208l and 208 l)**	**Spent Resins 208l**	**Compressible 208l *—****	**Inactives 208l**
1982	14	41	---	---	74—	4
1983	17	14	6	---	272—6	---
1984	8	---	2	73	85—4	3
1985	10	23	27	60	132—9	18
1986	22	52	63	2	341—23	27
1987	11	129	111	---	136—11	20
1988	12	155	118	109	344—29	11
1989	8	116	30	1	203—19	21
1990	13	179	24	0	113—12	21
1991	3	68	9	28	86—3	12
1992	16	485	14	---	56—17	---
1993	18	315	70	121	448—445	8
1994	16	279	33	---	—196	8
1995	8	129	12	---	—36	9
1996	**37**	**194**	**31**	**12**	**—495**	**10**
1997	**36**	**197**	**82**	**205**	**—239**	**8**
1998	**33**	**108**	**13**	**23**	**—125**	**8**
1999	**5**	**72**	**7**	**72**	**—81**	**5**
2000	**15**	**19**	**16**	**56**	**—148**	**4**
2001	**9**	**21**	**11**	**35**	**—113**	**2**
2002	**22**	**62**	**15**	**29**	**—113**	**1**
2003	**16**	**52**	**11**	**125**	**—108**	**1**
Total	**349**	**2710**	**705**	**951**	**—2232**	**192**
m^3	**72.60**	**733.20**	**352.60**	**286.5**	**—464.30**	**44.20**
Bq	**1.41×10^{13}**	**5.72×10^{13}**	**5.49×10^{12}**	**2.26×10^{14}**	**1.58×10^{12}**	**-----**

* *Before re-opening the drums for a better segregation*
** *After the new segregation*

Figure 3.2. Waste dam at CIPC

15 TBq (405 Ci) of Ra-226, and 4.2 TBq (112 Ci) of Ra-228 were disposed of at this site (Figure 3.2).

There are at present about 600 metric tons of "mesothorium," with an estimated Ra-228 activity of 1.85 TBq (50 Ci) disposed of in a trench at CIPC, and 0.2 TBq (6 Ci) stored in a shed at USAM in São Paulo (78 m^3). Although not formally classified as waste, the material with thorium hydroxide, separated from the rare earth elements during monazite processing, is also in storage at many installations in Brazil. The waste volume generated by the fuel-elements assembly unit, as well as by all the other pilot-scale fuel-cycle facilities, is negligible compared to the above-mentioned figures.

3.5. RADIOACTIVE WASTE FROM MEDICAL, INDUSTRIAL, AND RESEARCH APPLICATIONS

There are at present 3,248 radioactive facilities in Brazil that use several radionuclides. Table 3.3 shows the number and percentage of these facilities, by field of application. It can be seen from Table 3.3 that most of the radionuclides have medical (~39%) and industrial (~30%) applications.

Table 3.3. Radioactive installations

Field	Number	Percent
Medical	1,261	38.8
Industrial	984	30.3
Research	694	21.4
Distribution of Radionuclides	61	1.9
Services	248	7.6
Total	**3,248**	

In 1989, the Brazilian Regulatory Body suspended the authorization given to several manufacturers to use radioactive sources in lighting conductors. As a consequence, an estimated 75,000 of these devices installed throughout the country, with an overall activity on the order of 3.7 TBq (100 Ci) of Am-241, had to be recalled and stored by CNEN over the next five years.

Most of the soluble radioactive wastes produced as a result of using short-lived radioisotopes in medical institutions and research laboratories can be discharged into sanitary sewerage systems, after a given decay period, with concentrations and total activities not exceeding the limits specified in the Brazilian regulations based on USA regulation limits (Code of Federal Regulation, 1987). To be on the conservative side, it was assumed that whole flask contents were not utilized. Moreover, in determining the recommended decay period, we had to take into account the time required for the specific activity of the contaminated empty flask (assumed to have 23 g and a remainder of 2% of the initial activity) to reach a value of 2 nCi/g (74 Bq/g), which is acceptable for dustbin disposal.

Figure 3.3 shows the approximate percentages of the total spent sources (47,062) generated by several Brazilian states and stored at CNEN's research institutes, in rounded-off percentages (2,286 at the IEN-RJ institute, 4,756 at CDTN-MG, and 40,020 at IPEN-SP).

3.6. RADIOACTIVE WASTES PRODUCED AS A RESULT OF THE GOIANIA ACCIDENT

The violation of a teletherapy source in Goiânia, Brazil, at the end of September 1987, with a subsequent spread of most of its radioactive contents (i.e., 1,375 Ci of Cs-137), over a large urban area, brought about the need to estimate the quantities recovered during the decontamination work performed by CNEN (Figure 3.4). Approximately 3,500 m^3 of wastes were generated, with an estimated overall activity lying between 47 TBq (1,270 Ci) and 49.6 TBq (1340 Ci). Accounting for the

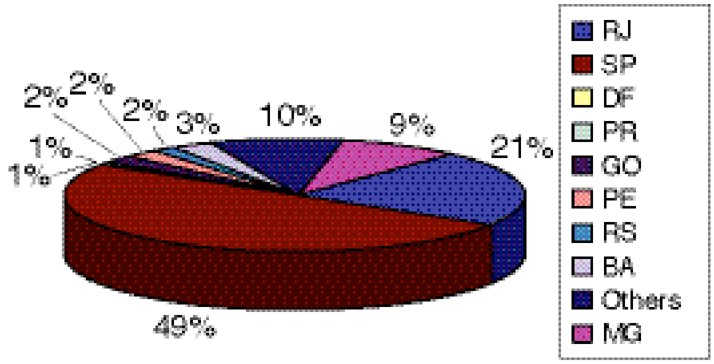

Figure 3.3. Number of spent sources generated per state

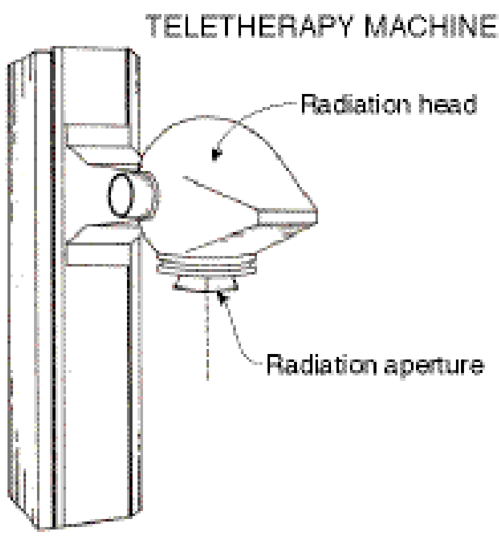

Figure 3.4. Teletherapy equipment

decay period necessary for the contents of all packages to reach a Cs-137 concentration level not greater than 87 Bq/g, it was possible to classify the drums and metal boxes into five groups, as described in Table 3.4.

The following packages were also used in Goiânia:

- 1 metal package for the headstock with the remaining source (4.4 TBq and with 3.8 m^3 from Group 5);
- 10 ship containers (374 m^3 with 0.4 TBq from Group 1), and;
- 8 special concrete packages (11.4 m^3 and 0.7 Bq from Group 5).

The waste was temporarily stored in open-air concrete platforms, occupying an area of about 8.5 x 10^6 m^2 at a site near the village of Abadia de Goias and 23 km away from the center of Goiânia, a city with a population of ~1 million (Figure 3.5).

Taking into account the decay period necessary for the contents of all packages to reach a Cs-137 concentration level not greater than 87 Bq/g, it was possible to classify the drums and the metal boxes into five groups, as described in Table 3.3. According to the IAEA classification method, all radioactive waste collected in Goiânia fell into the category of low-level, short-lived waste, and the IAEA disposal option for such waste is emplacement at shallow depths, in engineered storage facilities. It can be seen from Table 3.4 that approximately 33% of the waste volume (Group 1) has a specific radioactivity not greater than 87 Bq/g. Furthermore, most of the recovered activity is distributed over only 16.5% of the total volume, and will require a decay period greater than 150 years to reach acceptable concentration levels (Groups 4 and 5) (Heilbron et al., 1993).

The remaining 40.8% of the waste volume (Groups 2 and 3) were recently placed in concrete containers—to improve its condition, as well as to provide an additional engineered barrier in the near-surface repository that would be constructed. Moreover, although the specific radioactivity of the waste classified as Group 1 is less than the value established in the regulation for dust-bin disposal of solid wastes by users of radioisotopes, this group will not be considered exempt from control. The Brazilian Regulatory Body understands that the above-mentioned exemption criteria were established for solid wastes generated by facilities that handle small quantities of radioactive materials. It is also understood that care should be taken to avoid the deliberate fractioning and dilution of wastes, so as to achieve compliance with disposal regulations.

One near-surface repository (CGP) was constructed in 1995 (Heilbron et al., 1994)—after exemption from the Brazilian Licensing Body (IBAMA)—for all the radioactive waste below the 87 Bq/g limit (Group 1); see Figure 3.6. Another repository was constructed in 1996 for radioactive waste from IAEA Groups 2–5 after licensing by IBAMA, based on the assessment of a specific Safety Analysis Report (based on U.S. 1987, U.S. 1988 documents); see Figure 3.7 (Heilbron et al., 1992).

3.7. WASTE MANAGEMENT POLICY

As decreed by Brazilian legislation, the National Nuclear Energy Commission, CNEN, is the governmental body responsible for the receiving and final disposal of radioactive waste for all of Brazil. CNEN also holds national responsibility for the promulgation of regulations concerning waste management and disposal (CNEN-NE-6.05,1985, CNEN-NE-3.01,1988, CNEN-

Figure 3.5. Provisional storage at Abadia de Goiás

Table 3.4. Goiânia waste inventory

Group Time in years	Quantity metal box	Volume m^3	Quantity drums	Volume m^3	Total Activity TBq	Total Volume m^3
(1) t=o	404	686.8	2,710	542	0.06	1230.8
(2) < 0 <=90	356	605.2	980	196	0.76	801.2
(3) <90 <=150	287	487.9	314	62.8	1.44	550.7
(4) <150 <= 300	275	467.5	217	43.4	13.67	510.9
(5) t>300	25	42.5	2	0.4	30.064	42.9
Total	**1,347**	**2,289.9**	**4,223**	**844.6**	**45.71**	**3,134.5**

NE-6.02,1984, CNEN-NE-1.04,1984, CNEN-NE-6.06, 1990)

It is worth mentioning that political and psycho-social aspects related to the subject of radioactive waste disposal (the "not in my backyard" syndrome) contribute enormously to the difficulties faced by the Brazilian Government in establishing a national waste management policy.

3.7.1. Goiânia Wastes

Because of the situation described above, CNEN con-

Figure 3.6. CGP waste disposal

Figure 3.7. Second repository-waste disposal

structed two repositories for the Goiânia wastes. The first repository was opened in 1995 for waste classified as Group 1. A safety analysis report (SAR) was prepared for this repository, based on a simple and robust model showing that the radiological impact caused by disposal would be negligible. This report was submitted to IBAMA (the Brazilian organization responsible for environmental licensing) in order to obtain an exemption from submitting an environmental impact report (EIR).

The second repository, for waste from Groups 2, 3, 4, and 5, was constructed in 1996, very near the first one. For this second facility, the State of Goiás prepared an EIR and submitted it to IBAMA for approval. A special SAR was prepared by a consultant organization and was submitted to CNEN for evaluation. The mesothorium stored at CIPC, however, was placed in the local dam, since the studies conducted by CNEN showed that the environmental impact caused by this type of disposal was negligible.

3.7.2. Angra I and II Waste and Other Radioactive Facility Waste

For the waste resulting from the operation of Angra-I and -II related to isotopic medicine, industry, and research, a single national repository is planned. Its location and design have not yet been established. For the present, Angra I and II wastes are being stored at the nuclear power plant site.

Figure 3.8. Waste inside building

As mentioned above, wastes from the radioactive facilities are stored at three of CNEN's Research Institutes (Figure 3.8).

3.7.3. CNEN Safety Assessment Projects

Two projects are being conducted by CNEN that are concerned with safety assessment of final disposal facilities. The first project has the support of the IAEA, with experts from many laboratories in the U.S. Department of Energy (DOE); and the second project is being conducted within the Federal University of Rio de Janeiro. The University of Rio de Janeiro developed a national code for radionuclide migration and dose calculations for near-surface repositories; this work was completed at the end of 1998.

The project with IAEA aims at establishing a national capability for assessing the safety of waste disposal facilities. For this purpose, a multidisciplinary group of experts will be drawn from CNEN institutes and trained in safety-assessment methods, including the best use of relevant computer codes and the proper techniques for laboratory and field measurements.

In November 2002, the IAEA launched a new coordinated safety-assessment research project ASAM (Application of Safety Assessment Methodologies for Near Surface Waste Disposal Facilities) with the participation of Brazilian experts. The primary objective of the project is to investigate the application of safety-assessment methodologies used for postclosure safety assessment—and in particular the methodology developed under the IAEA's ISAM project (Figure 3.9)—to a range of near-surface disposal facilities; and to develop practical approaches for assisting regulators, operators, and other specialists in their review of such safety assessments.

3.8. CONCLUSIONS

The Brazilian Nuclear Program oversees the operation of several nuclear and radioactive installations over the country. The radioactive waste management policy is the responsibility of the Brazilian Nuclear Energy Commission, which is the regulatory body responsible for the licensing and control of the entire nuclear fuel cycle, as well as installations that use radionuclides for medical, industrial, or research objectives in the country.

The management of LLW does not represent a technical problem anymore, once the Brazilian Nuclear Energy Commission approved the capacity, construction, and licensing of two near-surface repositories for the safe disposal of radioactive waste generated by the Goiânia

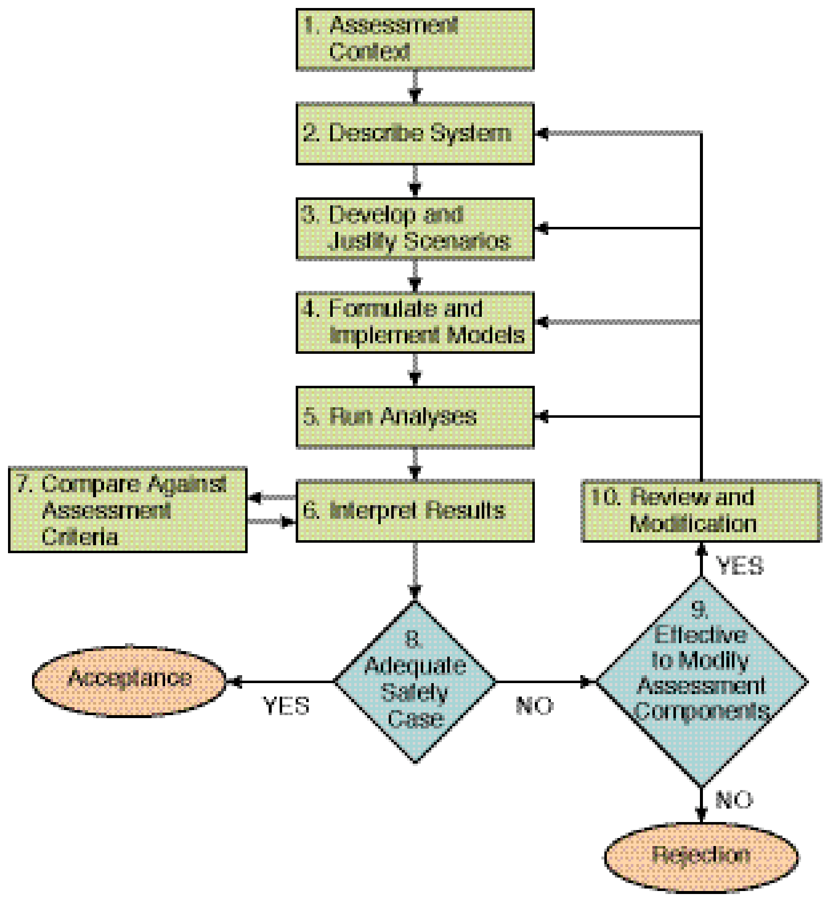

Figure 3.9. ISAM Methodology

accident. Unfortunately, there still is a public acceptance problem, resulting from the many political and social concerns over the subject of radioactive waste disposal. The "not in my backyard" syndrome contributes enormously to the difficulties faced by the Brazilian government in establishing a national waste management policy.

Since the Brazilian reprocessing program has not been clearly defined, the spent fuels from the nuclear power plants are stored in pools at the on-site reactor basin. This poses, by far, the most difficult problem to solve, in both its technical and political aspects.

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Chapter 4

Nuclear Waste Disposal in Bulgaria: Possibilities for Radioactive Waste Disposal in Kozloduy Nuclear Power Plant Area

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ABSTRACT. The vicinity of the Kozloduy Nuclear Power Plant (NPP) has been comprehensively investigated from a geological point of view. A number of boreholes have been drilled within a zone of about 25 km from the NPP, 60 of them with a depth ranging from 1,000 to 4,300 m. The NPP is situated in the Lom Depression, which is part of a stable tectonic platform. Predominantly, argillaceous sediments with a thickness exceeding 1,000 m were deposited in the Depression during the Paleogene and the Neogene. An analysis of the existing information provides reason to expect that the Neogene clayey formations near the NPP could host a repository for geological disposal of long-lived radioactive waste (RAW). One of the most promising sites for a surface-type repository (for short-lived RAW) is situated in one of the Neogene formations. The results from the investigations performed at this site, described in this paper, confirm its potential suitability: The site is characterized by simple hydrogeological conditions, including a thick zone of aeration (20–21 m) below which a poor local aquifer of modest thickness (1.5–2.0 m) is situated, and a well-defined zone of discharge. In addition, this site has geological and geotechnical features suitable for a repository. Such a repository would be founded on clay sediments with good retardation properties and sufficient bearing capacity. The possibility thus exists for containing both types of waste in close proximity to the NPP, with all the advantages ensuing from such a decision.

4.1. INTRODUCTION

The late-1990s site selection process for a Bulgarian national radioactive waste (RAW) repository was described in the *Third Worldwide Review* (Evstatiev and Kozhukharov, 2001). Since the end of the last decade and especially the beginning of the present one (2002–2003), analyses and explorations in the 25–30 km zone around the NPP have been carried out (through the financing of the Kozloduy NPP) with the aim of evaluating the geological conditions for RAW disposal. Analysis and reassessment were performed, using the available information, related to the geological and tectonic settings of the region, the seismic conditions, the deep hydrogeology, the processes of geological hazards, etc. The main focus was on the selection of potential sites for low- and intermediate-level radioactive waste (LILW) disposal. Using multi-attribute decision analysis according to 28 criteria, the potential sites have been determined. Preliminary geological, geomorphological, geological engineering, and hydrogeological explorations were carried out at the best prospective site, situated a few kilometers west-southwest of the NPP. The analyses also found good conditions for eventual deep disposal of long-lived RAW within the region (Evstatiev, ed., 2003). Thus, the favorable possibility arises of building repositories for both types of waste in one place or in close proximity.

The present report considers the more important results from recent investigations. First, the deep geological structure of the region around the Kozloduy NPP and the possibilities for deep disposal of long-lived RAW are described. Then, the main results from explorations

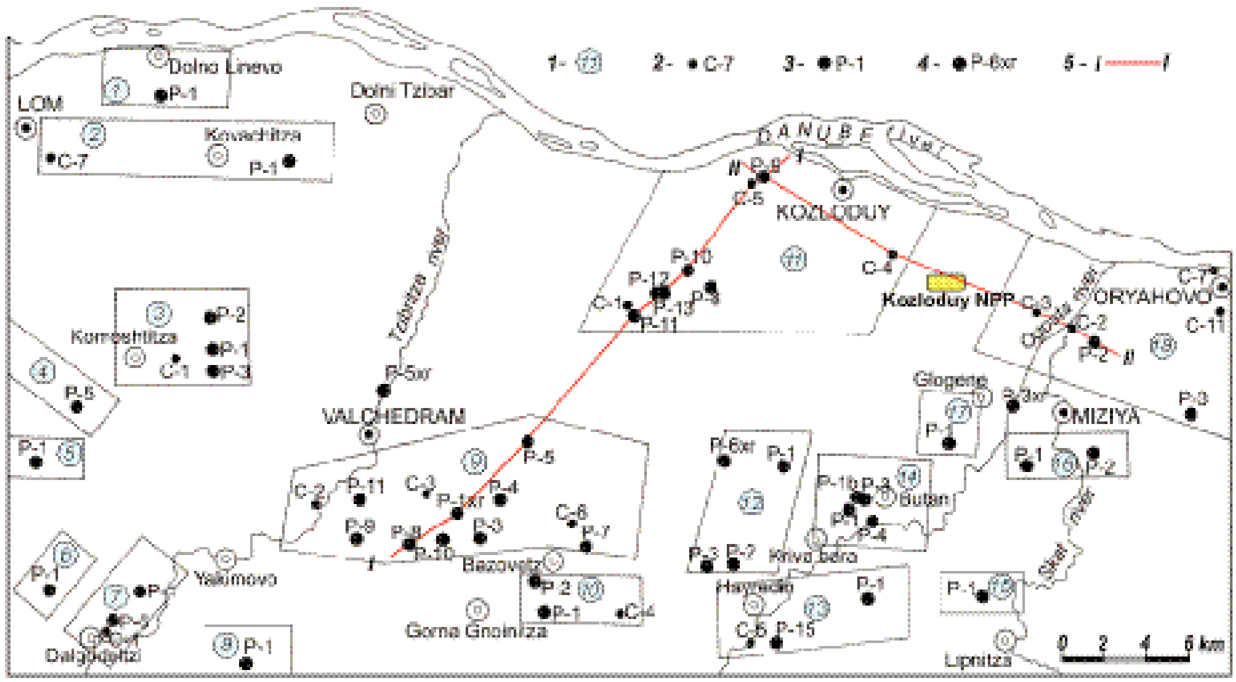

Figure 4.1. Disposition of the areas and boreholes from previous oil and gas explorations: (1) area number; (2) boreholes with depths <1,000 m; (3) boreholes with depths > 1,000 m; (4) hydrogeological boreholes; (5) disposition of the Geological Cross Section I–I.

into construction of an LILW surface-type repository at the potential site are discussed.

4. 2. DEEP GEOLOGY AND POSSIBILITIES FOR GEOLOGICAL DISPOSAL OF HIGH-LEVEL RADIOACTIVE WASTE (HLW)

4.2.1. Lithology and Stratigraphy

The Kozloduy NPP site has a thoroughly investigated deep geological structure, owing to extensive oil and gas explorations previously carried out there. Over the years, 60 deep boreholes (Figure 4.1) have been drilled to a depth of 4,300 m, and several deep seismic profiles have been made within a 25 km zone around the plant (Evstatiev, ed., 2003; Evstatiev et al., 2004). The available data show that sediments with small discontinuities from the Quaternary to the Lower Triassic ages are represented in the geological profile (Figure 4.2). The most interesting sediments from the perspective of HLW geological disposal are the thick, spatially homogeneous clayey-marls, situated at a depth of about 200–300 m to 1,100–1,300 m from the surface. These sediments include the Middle and Upper Paleogene and the Neogene formations—the Deleina, Krivodol, and Smirnenski.

The Paleogene sediments consist of three formations situated one over the other: *lower—marl-sandstone-lime - stone, middle—marl, and upper—silty-clay*. The middle and upper formations could be considered potential host media for geologic disposal. Their total thickness is 300-400 m, and clay and marls with sandstone intercalations predominate in both groups. If the repository were situated in the overlying Neogene sediments, the deposits of the two upper groups would be part of an impervious barrier preventing eventual radionuclide migration.

The Neogene, at a depth in the region of the NPP exceeding 1,000 m (Figure 4.3), is represented by sediments of the Miocene (Delein and Krivodol Formations) and the Pliocene (Smirnenski, Archar, and Brusarci Formations).

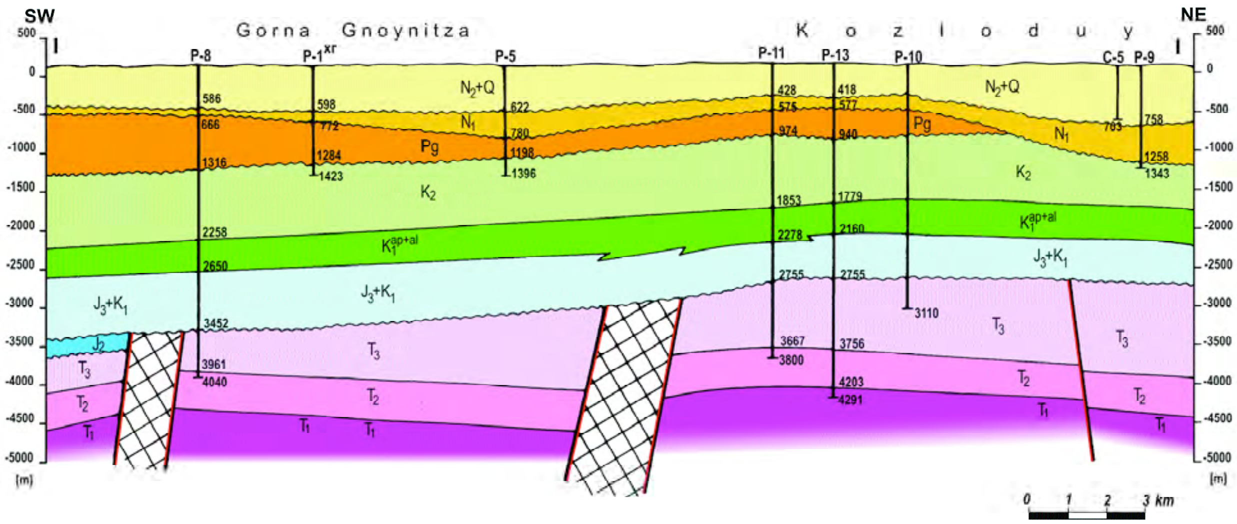

Figure 4.2. Geological Cross Section I–I: (1) Lower Triassic; (2) Middle Triassic (3) Upper Triassic; (4) Middle Jurassic; (5) Upper Jurassic-Lower Cretaceous; (6) Lower Cretaceous–Aptian-Albian; (7) Upper Cretaceous; (8) Paleogene; (9) Miocene; (10) Pliocene-Quaternary; (11) discordant boundary; (12) a. fault zone, b. fault

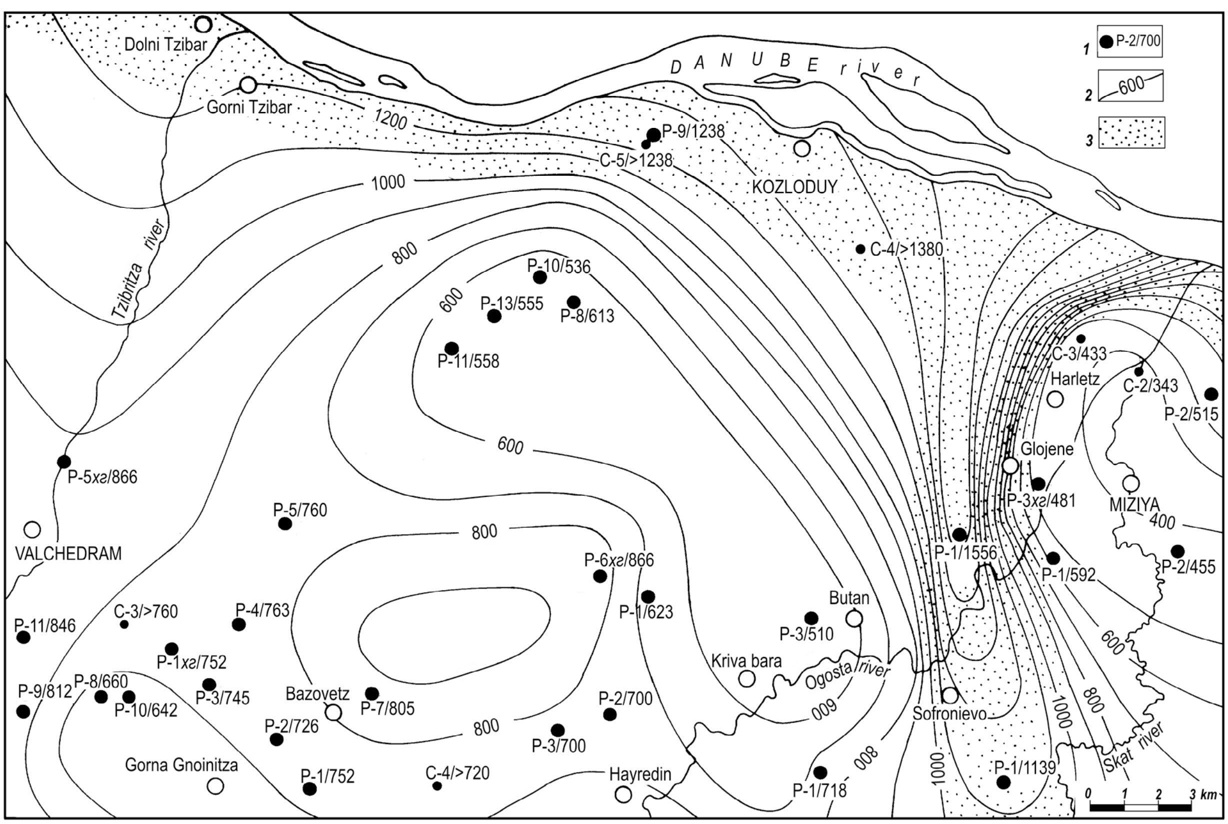

Figure 4.3. Isopach map of Neogene sediments: (1) borehole number/thickness of the Neogene sediments; (2) an isopach of the proper thickness; (3) cumulative Miocene paleovalley. Numbers here refer to the numbers along cross-section line in Figure 4.1.

The Delein Formation is composed of gray-bluish clays, among which single intercalations of clayey limestones, silty clays, and sandstones are encountered. A layer of gypsum and anhydrite with a thickness ranging from 15 to 30 m has developed in the middle of the section. Close to the Kozloduy NPP, the thickness of the formation is from 200 to 440 m. Its age is Badenian. This formation may be considered as one of the best prospects for further investigations related to geological disposal of HLW.

The Krivodol Formation consists mainly of gray and gray-bluish, stratified silty and calcareous clays, with intercalations of marls, dense clayey limestones, and sandstones. In the area near the Kozloduy NPP, the formation consists mainly of clayey facies; while beyond this immediate region, it is more coarse grained, with limestone sediments taking up much of the formation. The thickness of the Krivodol Formation in the vicinity of the NPP is 120–140 m. Its age is Sarmatian.

The Smirnenski Formation consists mainly of gray and gray-greenish, low-calcareous, silty clays with clayey limestone and marl intercalations. Sandstones with a thickness of up to 5–10 m are encountered in the lower and middle part of the section. In the top portion of the section, the formation contains less silty and limestone components. The formation thickness in the region of the NPP is 200–250 m. Its age is Meotian-Lower Pontian.

The Archar Formation, the only Neogene formation, is entirely sand and was deposited at a shallow depth (80–100 m) in the region. The formation crops out in the near vicinity of the area under consideration. In the lower levels, the sands are intercalated by dark gray, low-calcareous, thin-layered clays. The age of the formation is Upper Pontian.

The Brusarci Formation is situated close to the surface and consists mainly of clays with sand intercalations, which increase in thickness in the lower part of the section immediately above the Archar Formation. Its thickness varies from 50 to 200 m, and its age is Dacian-Romanian. This formation is not suitable for geological disposal, but as will be shown later, it has some good conditions for the construction of a near-surface LILW repository. The sediments of the Brusarci Formation are covered by Quaternary loess deposits.

4.2.2. Tectonics

The investigated region is in the western part of the Moezian Platform, which is located at the end of the Euroasian tectonic plate. The major tectonic activities in

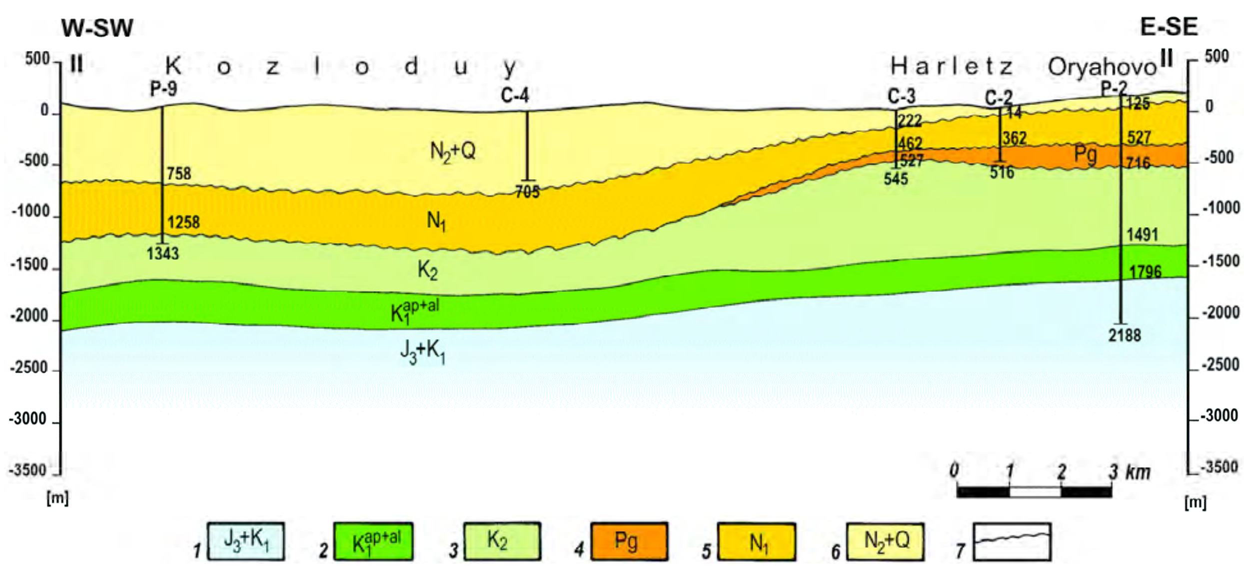

Figure 4.4. Geological Cross Section II–II: (1) Upper Jurassic–Lower Cretaceous; (2) Lower Cretaceous–Aptian-Albian; (3) Upper Cretaceous; (4) Paleogene; (5) Miocene; (6) Pliocene-Quaternary; and (7) discordant boundary

the platform ceased during the Jurassic, i.e., 150 million years ago (Figures 4.2 and 4.4). Since then, the Moezian Platform has been subjected mainly to fluctuating movements. The region falls within the eastern marginal part of the Lom Depression, which was formed during the Sarmatian. A large closed basin was formed in this depression, which was subsequently filled primarily with thick clayey deposits. After several transgressions and regressions, the basin water withdrew from the region about 2.60 Ma years ago. During the Glacial Pleistocene starting ~0.94 Ma, the southern parts of the depression, to which the region of the Kozloduy NPP belongs, were subjected to elevation, in contrast to the northern parts of Romania, where sinking continued during the Quaternary. The fluctuating neotectonic movements provoked faulting beyond the boundaries of the region.

In the beginning of the Glacial Pleistocene, about 0.80 Ma ago, the Danube River cut through the Panonian Basin (in Hungary), and began the formation of its new river terraces. The process of aeolian loess accumulation took place simultaneously with the formation of terraces.

4.2.3. Earthquake Hazard (Seismicity)

The seismicity of the region was thoroughly investigated in connection with the construction of the Kozloduy NPP. The main sources of the seismic hazard are earthquake zones outside the region. The most important of these is the Vrancea seismic focus in neighboring Romania, which has generated earthquakes of magnitude M=7. The local foci have M <4. According to maps that forecast seismic events for a period of 1,000 and 10,000 years, the region may be subjected to earthquakes of VII degree intensity, according to the MSK-64 scale and assuming a seismic coefficient of $K_s = 0.15$ (in compliance with the national norms for constructions of higher importance, as RAW repositories are).

From the seismotectonic point of view, the Moezian platform, and the Kozloduy region in particular, appears to be one of the calmest areas compared to other parts of Bulgaria. The rest of the country falls within the Alpine-Himalayas orogeny, where there is a much greater seismic risk.

4.2.4. Deep Hydrogeology

The deepest aquifer horizons are situated in the Middle Triassic and Upper Triassic deposits at a depth of more than 4,000 m. The Lower-Middle-Jurassic and the Upper Jurassic-Lower Cretaceous horizons are located above them, also at a great depth (about 3,000 m). The Paleogene aquifer complex, situated at a depth of 800–1,000 m, is developed in the lowest marl-sandstone-limestone formation.

With respect to deep geological RAW disposal, the aquifer horizons in the Neogene sediments are of greatest importance. The Krivodol Formation in the NPP region is represented by clayey facies with no aquifers. The more water-abundant layers in this formation have been observed in boreholes to the east of the region of interest.

The main aquifer horizon is in the Archar Formation (Upper Pontaian) at a depth of 80–100 m from the surface. It is formed in sands with rare clay intercalations and is spread throughout the Kozloduy region. It is charged by atmospheric precipitation in a zone of sands outcropping in the southern direction. In the vicinity of the Kozloduy NPP, the water conductivity is as much as 100–150 m^2/d. The natural water resources of the aquifer horizon are estimated to be 1 m^3/s. The flow in this aquifer is from the south to the north towards the Danube River, where it has a hydraulic connection with an alluvial aquifer on the lowest river terrace.

In the Kozloduy region, clayey sediments predominate in the near-surface Brusarci Formation. Although not always spatially homogeneous, these sediments act as a water barrier between the alluvial (Quaternary) deposits, if present, and the sands of the Archar Formation.

4.3. INVESTIGATING POTENTIAL SITES FOR AN LILW REPOSITORY

4.3.1 Location and Geomorphology

The potential LILW site is located in the western part of the Danubian plain, approximately midway between the Tsibritsa and Ogosta rivers (tributaries of the Danube River), on a gentle slope above a small valley without a permanent water stream. The site is about 2.5 km west-southwest of the Kozloduy NPP.

The following geomorphological forms can be distinguished in the region of the site: *a loess plateau*, incised by the tributaries of the Danube, the Tsibritsa, and Ogosta rivers, of *old denudational surfaces and*

Pleistocene river terraces. This loess plateau, the starting point for the formation of contemporary relief, is genetically related to the old erosional-accumulative level of lacustrine-fluvial genesis, formed at the end of the Pliocene and beginning of the Pleistocene. This level has been dry land from 0.8 Ma. In it, the formation of the small tributary valley and its branching began, as a result of erosion activity, accompanied by alluvial and aeolian sedimentation.

The gentle slope on which the site is located (elevation of 90–100 m above sea level) was formed of Brusarci Formation clays after some denudation of their top part. This Pliocene denudational surface is slightly inclined to the east-southeast. Loess sediments deposited during the Pleistocene were also subjected to surface erosion, and as a result, the thickness of these sediments varies from 1–2 m in the western part of the site to 12–14 m in the eastern part.

Comparisons between the hypsometric levels at the basement of the geomorphological forms (of equal age) for the site, and the same forms in the eastern parts of the Danubian plain, show that the neotectonic movements throughout this part of northern Bulgaria are positive and relatively of almost equal magnitude. This is evidence for the absence of active faults.

Moreover, the site is not endangered by inadmissible erosional processes. In the beginning of the Holocene, the erosion incision attenuated, and since then, accumulative processes predominate in the valley. Comparisons between the thickness of the recent soil and the soil thickness of the closely situated plateau show that surface erosion since the beginning of the Holocene is negligible and could be neglected as a factor threatening the safety of the eventual repository.

The region of the site is characterized by a moderate continental climate, formed under the influence of inrushing humid oceanic air from the west and northwest and the continental atmospheric masses coming from the north-northeast. The average annual temperature of the air is about +11°C, with an average annual amplitude of 24°C and an average annual precipitation of 550 mm. The number of days with snowfall averages 21, and the average annual potential evaporation is 440 mm.

Surface water drainage in the region is through the Danube River, its tributaries (Tsibritsa and Ogosta), and five to six other tributaries of these rivers without constant (permanent) runoff. The Danube River is the main drainage artery, with an average annual runoff of 5,700 m^3/s. Runoff of the Ogosta River at its mouth is 23.8 m^3/s, and runoff for the Tsibritsa River is 1.97 m^3/s.

4.3.2. Geotechnical Characteristics

The deep geological and hydrogeological setting, as well as the tectonic and seismic conditions of the site, are as described in Section 4.2. The preliminary investigations of the potential site show that the ground base is built from the sediments of the two complexes: the Quaternary and Pliocene (Figure 4.5).

The Quaternary deposits contain two layers: loess and

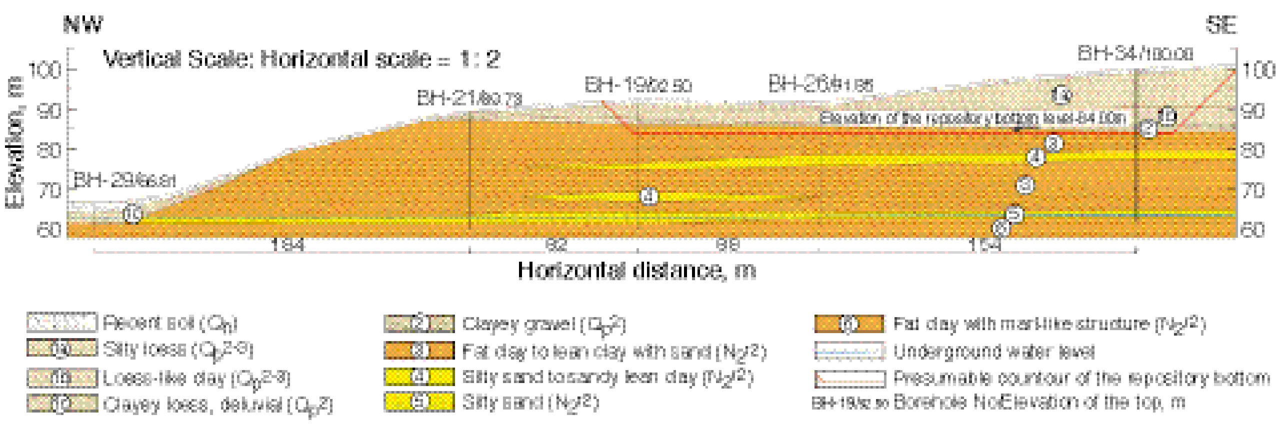

Figure 4.5. Geological profile through the potential LILW site

loess-like clays (Layers 1a and 1b) with a thickness between 5 and 14 m, and clayey gravels (Layer 2) with a thickness of 0.5–3.5 m. The Pliocene deposits are represented by the sediments of the Brusarci Formation, with a total thickness of about 80 m. In the site zone they consist mainly of clayey sediments containing spatially inhomogeneous sand intercalations. According to the conceptual plan, the repository will be of the near-surface type, built on the Brusarci Formation clays, with loess deposits and clayey gravels removed. In this way, the difficulties connected with the collapsible loess soils will be avoided.

According to their geotechnical properties the Pliocene deposits are divided into four layers (Figure 4.5), as follows:

Fat clay to lean clay with sand (Layer 3). This is the main layer because the foundations of the future repository will be made in it, and it would be the main barrier against radionuclide migration. The total thickness of the layer is from 16.5 to 18.0 m. It is characterized by the following averaged parameters: bulk density ρ = 2.03 g/cm^3, water content w = 19.2%, density of solids ρ_s = 2.74 g/cm^3, void ratio e = 0.61, liquidity index I_L = 0.01, degree of saturation S_r = 0.86, unconfined compressive strength q_u = 450 kPa; shear strength parameters -φ_u =18 deg, c_u = 53 kPa and φ' = 21 deg c' =38 kPa, stress-strain modulus at pressure range 200–400 kPa -E_{300} = 13.0 MPa, coefficient of consolidation (at p = 400 kPa and Δp = 200 kPa) -c_v = n.10^{-3} ÷ n.10^{-2} cm^2/s, SPT data are N_{70} = 25 and N_{60} = 30 (estimated consistency - very stiff), the compressional and transverse seismic wave velocities determined by borehole seismic logging – V_p = 750–960 m/s and V_s = 335–490 m/s respectively. The sediments of Layer 4 are incorporated into the sediments of Layer 3 in the form of thin intercalations or lenses, thus dividing the latter in two or three parts.

Silty sand to sandy lean clay (Layer 4). It is characterized by the following averaged parameters: w = 12.7%, ρ = 1.87 g/cm^3, ρ_s = 2.67 g/cm^3, e = 0.60, S_r = 0.56, φ' = 33 deg and c' = 4.9 kPa. According to the SPT data - N_{70} = 23 and N_{60} = 27, the sand is classified as medium dense to dense. The velocities of the seismic waves are V_p = 725–730 m/s and V_s = 315–375 m/s.

Silty sand (Layer 5). The layer extends throughout the site at a depth of 26–37 m. It is embedded almost horizontally, with a slight inclination to the northeast, with thickness between 1.5 and 3.0 m. With respect to grain-size composition, it is similar to Layer 4, but it is regarded as a separate layer because of the small local aquifer within it. The established piezometric level is at elevation of 63.5–64.5 m (i.e., at 26–35 m from the surface). According to the SPT data N_{70} = 21 and N_{60} = 24, the sand is classified as medium dense. Using the relationship of Skempton (1986), the relative density Dr of the sand has been calculated to be 0.63–0.66. An angle of internal friction of 34 deg has been determined after the average value for N_{70} (Bowles, 1984, 1996). Although Layer 5 is situated at a depth of more than 15 m, an assessment of the liquefaction potential under seismic impact has been made, according to the methodology of Eurocode 8 (Part 5). This assessment found that the sand cannot liquefy even in the case of maximal seismic acceleration of 0.20 g, caused by an earthquake with magnitude M = 7.5.

Fat clay dense stiff with marl-like structure (Layer 6). The layer is situated about 28 to 37 m from the surface, and its upper surface is slightly inclined to the northeast. The entire layer thickness is about 60 m, with the rusty-brown color of the clay transformed to grayish blue at depth. The layer plays the role of an impervious barrier for the local aquifer formed in Layer 5. The upper part of the layer possesses the following average geotechnical characteristics: w = 18.8%, ρ = 2.15 g/cm^3, ρs = 2.77 g/cm^3, e = 0.54, I_L = 0.002, q_u = 750 kPa, φ_u = 12.2 deg, c_u = 188.8 kPa and φ' = 12.4 deg, c' = 189.3 kPa, E_{300} = 27.7 MPa, at p = 400 kPa and Δp = 200 kPa c_v = n x 10^{-3} cm^2/s, SPT data – N_{70} = 28 and N_{60} = 33 (estimated consistency is very stiff). Seismic wave velocities are V_p = 1,850 m/s and V_s = 500–700 m/s.

Preliminary investigations, based on these data, show that the soil base possesses sufficient bearing capacity for the construction of a near-surface repository at reasonable cost.

4.3.3. Geochemical Characteristics

Conditions That Created Brusarci Formation Sediments

The deposition of the sediments of the Brusarci Formation took place under changing climatic and geodynamic circumstances, including the changing depth of the lacustrine basin (Evstatiev, ed., 2003). This has provoked the formation of clay, sandy clay, and (to a lesser extent) clayey sand and sand sequences; along with

Table 4.1. Content of oxides and ratios between them

Layer	Soil type	SiO_2	SiO_{2+} R_2O3	SiO_2/ R_2O_3	Na_2O+ K_2O	K_2O/ Na_2O	CaO+ MgO	MgO/ CaO
		% by weight	% by weight	-	% by weight	-	% by weight	-
3	clay	58.09	85.60	2.1	3.08	3.9	2.76	2.2
4	sand	78.15	91.17	6.0	2.73	1.2	1.99	0.36
3	clay	65.75	88.84	2.8	3.00	3.0	1.60	1.6
5	sand	80.78	92.39	7.0	3.03	1.5	1.51	0.35
6	clay	61.00	85.05	2.5	2.86	3.08	2.00	1.4

intercalations of high carbonate content and coarser terrigenous material. During a significant period of their history, the sediments were subjected to erosion and weathering that led to reduction of the geological overburden load, emergence of chemical processes accompanying the weathering, and migration of soluble substances. The clays are predominant in the geological profile despite the variable sedimentation conditions. Under the effect of weathering in their topmost 18–20 m, oxidation processes took place, the evidence for which is their rusty-yellow color. With depth, under the first aquifer horizon, the clays become grayish-green—evidence for the existence of reducing conditions.

Chemical Composition

The chemical composition of representative samples from the clays and sands of the Brusarci Formation is expressed as oxide content in weight % in Table 4.1. The amount of silicon dioxide varies from 58% to 66% in the clays and 78% to 81% in the sands. The SiO_2:R_2O_3 ratio for the sands is 6-7. It is lower for the clays (2.1–2.8), caused by weathering. The so-called silicon coefficient SiO_2:Al_2O_3 is also a geochemical assessment tool: it takes into account the dialuminum trioxide, which is more sensitive to geochemical changes. The relatively lower values of this coefficient for the investigated clays (2.8–4.0) indicate that weathering took place under arid-climate conditions.

The amounts of alkaline oxides are secondary; K_2O is dominant in all samples. With respect to earth alkalines, CaO is predominant in sands and MgO in clays. Only the more calcareous clays and the clay intercalations with higher calcite-concretion content exhibit CaO amounts higher than that of MgO. Moreover, the organic-substance content calculated as humus is 0.06–0.18% in clays; it is only 0.002% in sands. The amount of carbonates in the investigated samples is low because they were extracted outside the local carbonate zones in the upper part of the massif, where it reaches up to 20% and more.

The total quantity of water-soluble salts in the investigated soil samples is less than 0.1%. The water extract is slightly alkaline, and the pH value varies from 7.8 to 8.1. All three types of water-soluble salts—hydrocarbonates, chlorides, and sulphates—are present, although in very small amounts, in all the samples. Salinization of the soils is hydrocarbonate, with the quantity of HCO_3^- exceeding 90% of the total anion sum (expressed in meq/dm^3). The dominating cation is Na^+, followed by Mg^{2+} in clays or Ca^{2+} in sands and more calcareous clayey intercalations.

Mineral Composition and Sorption Properties

The investigated clays contain clayey, silty, and sandy fractions. For this reason, except for the clayey and mica minerals, there are also quartz, plagioclase, K-feldspar, goetite, and calcite in their composition.

The content of clay minerals, represented by smectite, vermiculite, illite, chlorite and kaolinite, varies from 41 to 60%. Smectite is dominant in half of the investigated samples—from 27 to 37% of the total mineral content. Smectite is also predominant in the clayey fraction (less than 0.005 mm) at 38–41%, followed by illite at 24–25%, and kaolinite at 16–17%. The rest of the finely dispersed minerals amount to less than 20%, the highest amounts belonging to goetite and quartz. The sorption complex of smectite contains two-valency cations.

The ion-exchange capacity and sorption with respect to Cs, Sr, Cs^+, and $+Sr^{+2}$ have been determined for four representative clay samples. Our determination is that the clays possess good ion exchange capacity—18.88 and 30.05 meq/100 g. Sorption with respect to Cs^+ is between 6.54 and 7.46 mg/g; with respect to Sr^{+2}, it is between 1.05 and 2.44 mg/g. Sand layers consist of quartz (53–55%), albite (13–16%), orthoclase (7–11%), and illite (6–10%). Kaolinite, smectite, goetite, and calcite are also present in smaller amounts.

SEM investigations showed that the structure of clays from the reduction zone (*Layer 6*) is more homogeneous and consists of more fine-grained aggregates than the clays from the oxidation zone (*Layer 3*). Microcracks are observed in the clays close to the surface, but they are not found at greater depths.

4.3.4. Hydrogeological Conditions

Main Hydrogeological Units

With respect to radionuclide migration, the three main hydrogeological units can be divided as follows:

1. Unsaturated zone (zone of aeration), with a thickness of 20–21 m
2. Local aquifer, formed in Layer 5 with a small thickness of 1.5–2.0 m and slightly water abundant
3. Impervious barrier of the aquifer (Layer 6).

Zone of Aeration

The zone of aeration extends throughout *Layer 3* and *Layer 4*. The unsaturated zone is developed in an oxidation medium, which results in a certain inhomogeneity of the hydraulic parameters. The coefficient of filtration k determined by laboratory tests varies from $n \times 10^{-10}$ m/s for the clay of *Layer 3* up to $n \times 10^{-6}$ m/s for the sand of *Layer 4*. In addition, the results from a borehole filtration test in this zone provided an integrated value of $k = n \times 10^{-8}$ m/s.

Local Aquifer Horizon

Our hydrogeological investigations (piezometric measurements and pumping tests) have provided the following characteristics for the local aquifer:

- It is fed by rain water in the southern area.
- The groundwater flow direction is from southwest to northeast.
- The piezometric level is between an elevation of 66 m in the southern end and 62 m in the northern end of the site.
- The aquifer drains into the lowest part of the valley at a distance of about 300–350 m northeast of the site and afterward into a terrace on the Danube.
- The hydraulic gradient of the aquifer is 0.010–0.016 above the site, 0.004–0.005 in the central zone of the site, and 0.010–0.013 below the site.
- The coefficient of filtration is $k = 1.3 \times 10^{-5}$ m/s, the transmissivity of the layer is from 1.5 to 2.0 m^2/d, and the groundwater discharge in the site vicinity, along a length of 500 m, is only about 10 m^3/d.

Impervious Barrier

The impervious barrier of the aquifer is *Layer 6*, the yellow-rusty brown fat clay with marl-like structure, which gradually becomes gray-bluish with depth and is very dense and stiff. The coefficient of filtration is $k \leq 1 \times 10^{-11}$ m/s. This impervious barrier coincides with the upper part of the reduction zone of the Brusarci Formation. The thickness of the impervious layer is about 60 m, excluding any connection with the most important aquifer in the region, in the Archar Formation.

4.4. CONCLUSIONS

The investigations carried out allow the following conclusions to be made:

- The geological aspects of the prospective site have been investigated comprehensively. On the basis of data from 60 deep boreholes, the geological-tectonic structure has been established to a substantial depth (4,300 m), for a long time interval (250 million years), and for a sufficiently large area (about 800 km^2).
- The region is part of a stable tectonic plate, in which only fluctuating vertical movements have occurred since the Jurassic. These movements were predominantly negative from the Sarmatian to the Quaternary, and then became mainly positive during the Quaternary. The seismicity of the region is one of the lowest in Bulgaria, and no active faults have been established.
- The thick (up to 800 m) and spatially homogeneous clayey deposits of the Neogene formations are most suitable, as a host medium, for the eventual geological disposal of HLW. They possess very favorable

isolation properties. The aquifer horizon in the Archar Formation, located 80–100 m from the surface, might create difficulties for mining construction, but no hazards are expected with respect to radionuclide contamination.

- A potential LILW site is located very near (about 2.5 km) to the Kozloduy NPP, in Pliocene clay sediments of the Brusarci Formation. These sediments are characterized by favorable geotechnical and geochemical properties: the site is not affected by hazardous geological processes, flooding, or other hazards; and the surface erosion will be insignificant during the lifetime of the repository. The site is characterized by a thick zone of aeration of about 20–21 m, below which a poor local aquifer is located. The hydrogeological settings of the site are simple, and a well-defined modeling of the radionuclide migration will be performed.
- The groundwater discharge in the zone of the site is very small (about 10 m^3/d), which will permit effective groundwater control. Furthermore, the local aquifer could act as a natural horizontal drainage system beneath the entire future repository.

The main conclusion is that in the vicinity of the Kozloduy NPP, the possibility exists for the same site to be developed as a surface LILW repository and a deep geological HLW repository. If further investigations confirm this conclusion, the site will offer very important advantages from the viewpoints of safety, waste transport, infrastructure, and local population support.

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Chapter 5

Progress Towards Long-Term Management of Used Nuclear Fuel In Canada

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5.1. INTRODUCTION

Since the 1970s, Canadians have benefited from nuclear power, and have been developing the technology and expertise to manage used nuclear fuel for both interim and long-term management. In 1978, the governments of Canada and Ontario established the Canadian Nuclear Fuel Waste Management Program as a step towards the safe and long-term management of used fuel. Atomic Energy of Canada Limited (AECL) and Ontario Hydro (now Ontario Power Generation Inc.) were the organizations primarily responsible for developing storage, transportation, and deep repository technology in Canada. In 1994, AECL submitted its concept for placement of nuclear fuel wastes in a deep geological repository (DGR) excavated in the plutonic rock of the Canadian Shield. The concept underwent a federal environmental review with public hearings (CEAA, 1998). The government of Canada responded to the recommendations of the environmental assessment panel and issued the *Nuclear Fuel Waste Act* (Canada 2002).

In response to the *Nuclear Fuel Waste Act*, the Nuclear Waste Management Organization (NWMO) was established by the nuclear utilities in Canada (Ontario Power Generation Inc., Hydro-Québec, and NB Power Nuclear) to study options and to recommend an approach for long-term management of Canada's used nuclear fuel. The NWMO was tasked to study approaches based on at least three methods:

1. Deep Geological Disposal in the Canadian Shield
2. Storage at Nuclear Reactor Sites
3. Centralized Storage, Either Aboveground or Belowground

The *Nuclear Fuel Waste Act* required the NWMO to submit its study to the Minister of Natural Resources Canada by November 15, 2005, after which the federal government will decide on the preferred approach for Canada.

As the technical foundation for its study, NWMO recognizes that much work has been done over the past 20 years with advancing the technology for both storage and deep geological repository concepts in Canada and internationally. Since the *Third Worldwide Review* was issued in December 2001 (Witherspoon and Bodvarsson, 2001), significant technical progress has been made in siting geoscience, safety assessment, and repository system engineering. Recently, the technical program in Canada has been managed by Ontario Power Generation Inc. (OPG), and the work has been summarized in a series of annual reports (Gierszewski et al., 2002; 2003; 2004; Hobbs et al., 2005). In addition, conceptual designs and cost estimates have been developed for extended used-fuel storage at nuclear reactor sites, for centralized storage and deep geological disposal (CTECH 2003a; 2003b; 2002), and for the transportation system to a central facility (Cogema, 2003).

Similar to several other countries, despite the technical progress over the past decades, we have not been able to proceed with a long-term solution that gives Canadians confidence that the approach reflects and responds to their values, issues, and concerns. For this reason, the NWMO adopted a study process designed to develop collaboratively with Canadians a management approach for the long-term care of used nuclear fuel that is socially acceptable, technically sound, environmentally responsible, and economically feasible.

In this paper, we will outline what we have learned from our dialogue with Canadians, describe the approaches considered in our study, and outline our recommended approach included in our Final Study Report *Choosing a Way Forward: The Future Management of Canada's Used Nuclear Fuel* (NWMO, 2005b) which was submitted to the Government of Canada in November 2005.

5.2. LISTENING TO CANADIANS

In developing a framework to assess the approaches, NWMO understood that the challenge was to ensure that this framework is driven by the values of Canadian society as a whole. The approach embraced by the NWMO in developing and implementing the framework was aimed at ensuring a transparent and inclusive process for:

- Shaping the questions which ought to be asked and answered in the study, and key issues to be addressed in the assessment of the management approaches
- Determining the range of technical methods which ought to be considered
- Identifying and assessing the risk, costs, and benefits of each management approach
- Designing the overarching management structure and implementation plans for each management approach

The NWMO designed its three-year study as a dialogue conducted iteratively over four phases, each phase supported by a series of milestone public discussion documents designed to:

- Share what we had heard from Canadians
- Describe how it was being applied in the study
- Solicit input to shape and direct subsequent steps in the study

In our first discussion document, *Asking the Right Questions?* (NWMO, 2003), we asked Canadians if we were capturing the key questions that should be asked and answered in the analysis and study of potential methods for the long-term management of used nuclear fuel. In our second discussion document, *Understanding the Choices* (NWMO, 2004), we reported back on the direction we had received from our engagement process and research activities to date, and presented the results of a preliminary comparative analysis of the options as a foundation for further dialogue. In our third discussion document, our Draft Study Report *Choosing a Way Forward* (NWMO, 2005a), we again reported back on the direction emerging from dialogue and research activities, and outlined the recommendation that we planned to make for broad review and discussion.

Reflecting on the comments of Canadians, it became apparent early in the study that although we share certain values and objectives that should inform the NWMO's study, the assessment of management approaches necessarily involves difficult decisions about priorities and the conditions under which trade-offs among objectives would be appropriate. The dialogue brought into focus some of the difficult choices and trade-offs that will need to be reflected in the assessment of approaches.

5.3 PRELIMINARY ANALYSIS OF THE OPTIONS

Early in the study process, the NWMO identified 14 potential methods for managing used nuclear fuel. For the most part, Canadians agreed that our focus should be on the three methods requiring study under the *Nuclear Fuel Waste Act* as those which have the most promise. In 2004, the NWMO assembled a multidisciplinary group of individuals as an Assessment Team to translate the ten questions presented in the first discussion document into an assessment framework, taking into account the comments of Aboriginal peoples, the public, and specialists on those questions, and to conduct a preliminary assessment of alternative approaches (Ben-Eli et al., 2004). The Assessment Team's methodology was deliberately selected to allow for a holistic assessment – one that systematically integrates social and ethical dimensions with technical, economic, financial, and environmental considerations.

At the core of the assessment were eight objectives and influencing factors. The eight objectives were fairness, public health and safety, worker health and safety, com-

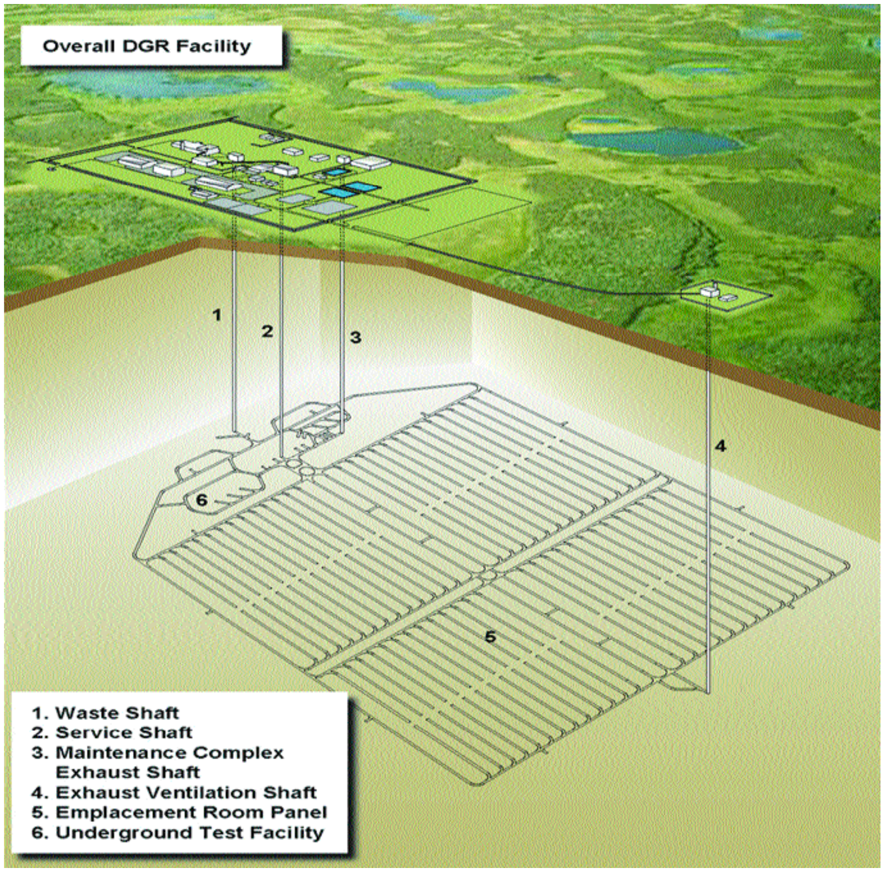

Figure 5.1. Overall design and layout of a deep geological repository

munity well-being, security, environmental integrity, economic viability and adaptability.

By using a framework reflective of the values and concerns of Canadians, the Assessment Team created a preliminary description of the strengths and limitations of the management approaches for consideration and dialogue among Canadians. In advancing our understanding of some of the distinguishing features of the options, this framework provided the context for a substantive discussion with Canadians on how to consider the relative risks, costs, and benefits of the alternative management approaches.

Here we summarize the approaches under examination and the preliminary findings of the Assessment Team concerning the advantages and limitations of each approach. This description was included in our second discussion document and was the subject of a broad dialogue among citizens as well as further work by specialists.

5.3.1. Deep Geological Disposal on the Canadian Shield

Deep geological disposal would involve siting and developing an engineered repository located deep within the stable plutonic rock of the Canadian Shield. Conceptual designs have been prepared for the used fuel packaging facility and associated infrastructure, as well as the underground repository constructed at a depth of 500 to 1,000 m (CTECH, 2002). The overall design of the deep repository is illustrated in Figure 5.1.

The siting, design, licensing, approval, and construction of a central DGR and transportation system could take up to 30 years to complete. Used fuel would need to be transported from existing interim dry storage facilities at the seven reactor sites to the central used-fuel packaging facility. The fuel bundles would be placed into long-lived containers, such as steel-lined copper containers, and surrounded by clay-based sealing materials in the vault (see Figures 5.2 and 5.3). The used fuel container

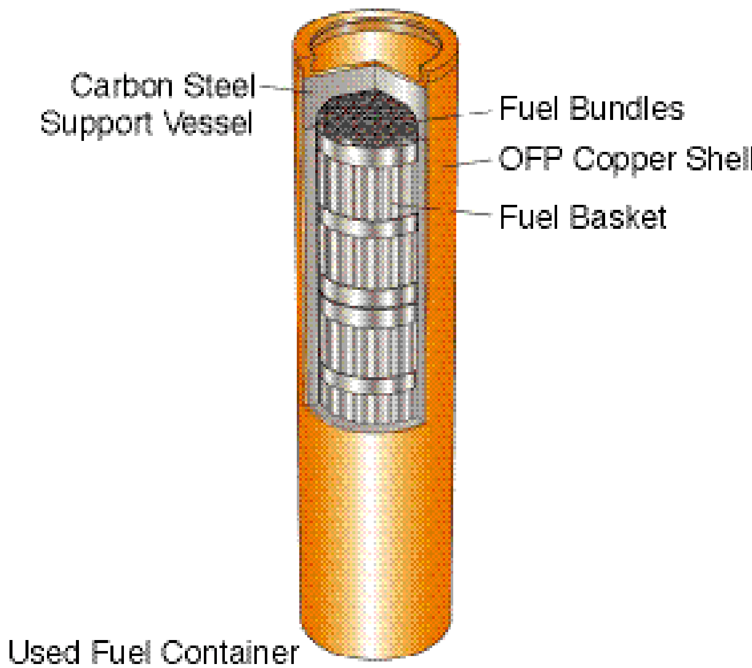

Figure 5.2. Used fuel container for a deep geological repository

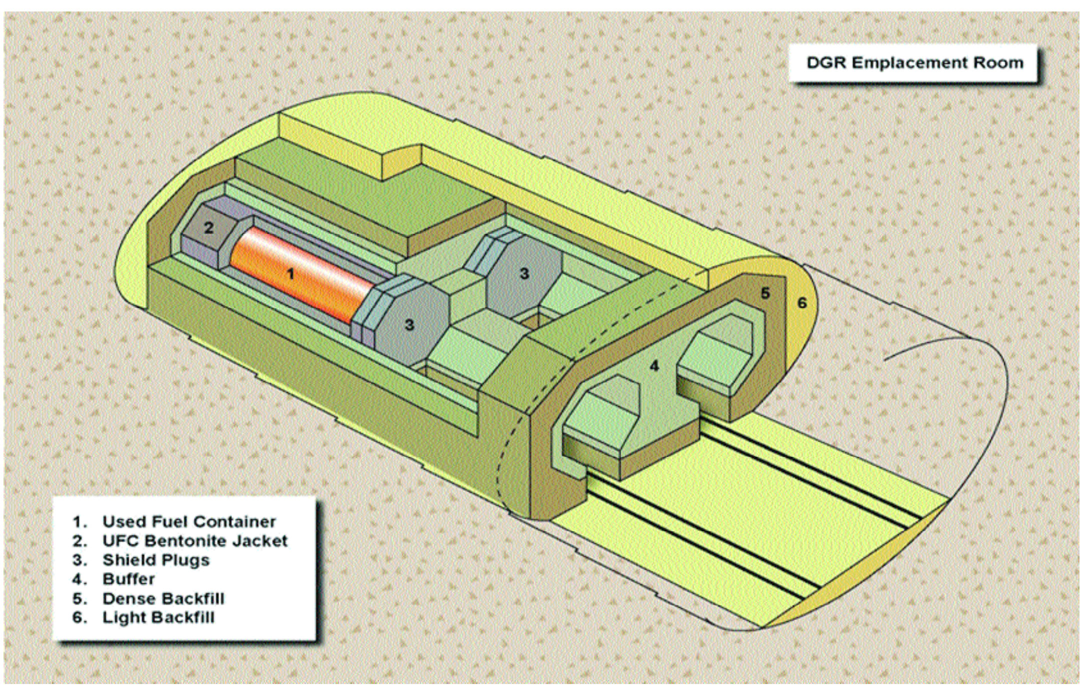

Figure 5.3. Placement room layout in a deep geological repository

The reference conceptual design and cost estimate developed by CTECH (2002) for deep geological disposal on the Canadian Shield was for the in-room placement of used fuel containers. Other conceptual designs are being developed for the Canadian concept; they include the in-floor borehole concept and the long horizontal borehole concept (Gierszewski et al., 2004; Russell and Simmons, 2004). Selection of the most appropriate placement method would depend on specific conditions at a site, and on development and demonstration of engineering technology, among other factors.

Analysis of deep geological disposal on the Canadian Shield has shown that the health risks are small and less susceptible to upset conditions, due to the passive safety nature of the system (Garisto, 2004). Transportation risks are similar to the centralized extended storage concept.

The most recent postclosure safety assessment of the deep geological repository concept can be found in the Third Case Study prepare by OPG (Garisto et al., 2004). This study builds on the two previous safety analyses by Atomic Energy of Canada Limited (Goodwin et al., 1994; 1996) and uses the most recent conceptual design of a deep geological repository at a generic location on the Canadian Shield (CTECH, 2002). The calculated public dose from the assumed undetected early failure of several used fuel containers in the repository for the reference scenario is illustrated in Figure 5.4. The postclosure dose was modeled for the most exposed member of the critical group, a self-sufficient farmer residing near the surface discharge location. The peak dose of about 10^{-7} Sv/a occurs after nearly 500,000 years and is dominated by the long-lived and relatively mobile radionuclide iodine-129.

The Third Case Study public dose predictions are several orders of magnitude below the International Commission on Radiological Protection reference dose constraint of 0.3 mSv/a (ICRP, 2000) and the average Canadian natural background dose of 1.7 mSv/a (Grasty and LaMarre, 2004).

Advantages: Deep geological disposal in the Canadian Shield results in the eventual permanent long-term iso-

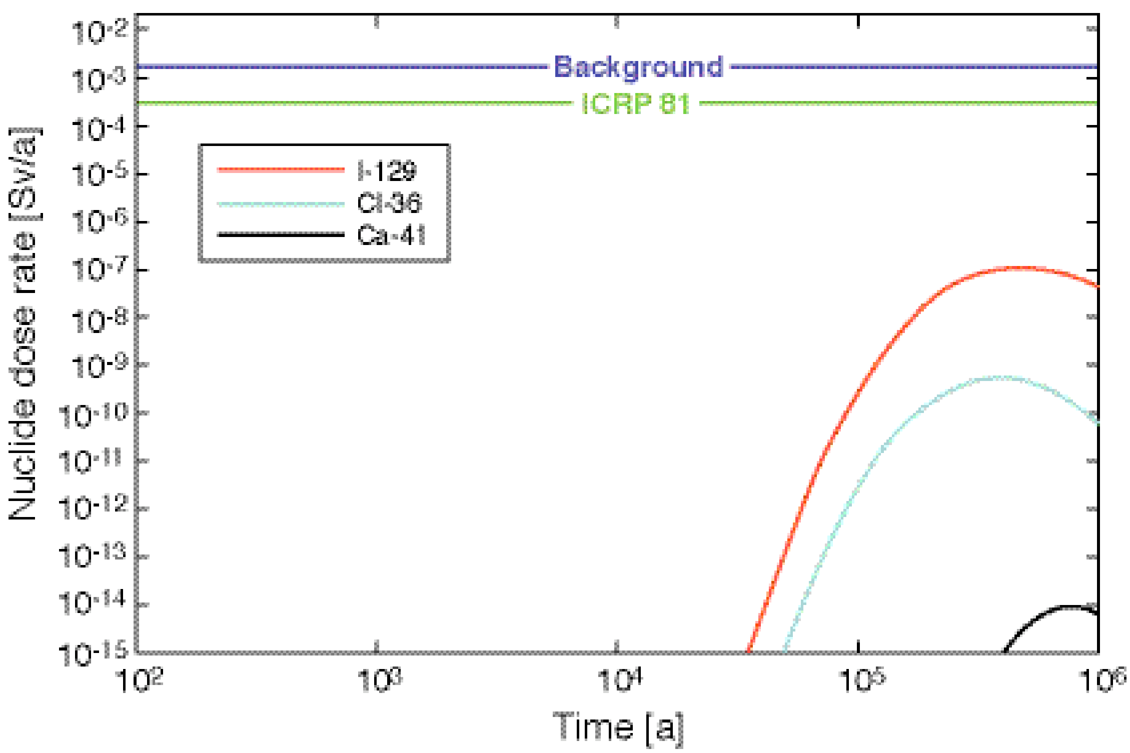

Figure 5.4. Postclosure safety assessment of a deep geological repository

lation of used nuclear fuel, which reduces or may eliminate the necessity for institutional and operational continuity and financial surety. As a consequence, after placement and closure, provision of long-term resources and funding would not be required, although further actions (such as postclosure monitoring) are not precluded. The site is chosen with specific features as a requisite and, if done well, can be achieved with community participation. The intrinsic geologic, hydrologic, and other features of the site, in combination with engineered features such as long-lived waste packages and material buffers, isolate the used nuclear fuel from the accessible environment for the very long time periods that they remain hazardous. Deep placement reduces security concerns, both before and after closure.

Limitations: Advance "proof" that such a system works is not scientifically possible because performance is required over thousands of years. However, detailed scientific studies, natural analogues, models, and codes form the foundation for the assurances of performance provided to regulatory authorities and interested organizations and individuals. Monitoring becomes more difficult as the used nuclear fuel is placed deep underground and as the site is backfilled and closed. At this stage, adaptability and flexibility are also reduced as retrieval of the used fuel, for example, becomes much more difficult, costly, and hazardous compared with storage. Siting for a deep repository in the Canadian Shield must pay particular attention to intrinsic geological features, perhaps limiting options more than for storage alternatives. As with centralized storage, community participation in regard to siting could be contentious, and transportation of the used nuclear fuel will be required.

Figure 5.5. Dry storage container at the Pickering Waste Management Facility in Ontario

5.3.1. Storage at Nuclear Reactor Sites

Permanent or indefinite storage at the nuclear reactor sites would involve expansion of existing interim dry storage facilities or the establishment of new long-term dry storage facilities at each of the seven existing nuclear reactor sites in Canada. These interim storage facilities have been licensed by the Canadian Nuclear Safety Commission. The siting, approval, design, and construction of any new storage facilities could take up to 10 years to complete.

Examples of Canadian dry storage facilities are illustrated in Figures 5.5 and 5.6. Current dry storage containers are designed to last at least 50 years, but their life expectancy is expected to be 100 years or longer. Conceptual designs for long-term dry storage facilities for used fuel in casks, vaults, or silos have been developed by CTECH (2003a). This option would require ongoing maintenance, inspections, monitoring, and security, and a cyclical program to refurbish or replace storage buildings and repackage used-fuel storage containers. A complete cycle of refurbishment and repackaging of the nuclear fuel storage system is approximately 300 years.

Analysis of facilities for storage at nuclear reactor sites has shown that there are no potentially significant pub-

Figure 5.6. Concrete silos at the Whiteshell Laboratory in Manitoba

lic health or ecological impacts associated with normal operation of the facilities (Garisto, 2004). The health risks are small and similar to the conventional risks typical of other large industrial projects.

Advantages: No off-site transportation of used nuclear fuel would be required because the used fuel would remain next to where it is generated. Each of these sites already houses nuclear installations, so there is nuclear expertise on site and in the existing communities. These communities are familiar with the presence of nuclear facilities, including storage of used nuclear fuel. Further, the ability to monitor the performance and the flexibility to adapt to changing conditions should be facilitated. The science and technology required are well in hand

Limitations: The key disadvantage, shared with centralized storage, is the need for continuing administrative controls and operations, including the necessary funding, for the thousands of years the used nuclear fuel remains hazardous. Unlike centralized storage, storage at nuclear reactor sites means continued management at a number of sites, each of which has, as its primary focus, the production of power, not the long-term management of used nuclear fuel. These reactor sites were selected for their suitability for reactor operation, not for very long-term storage of used nuclear fuel. The used nuclear fuel will remain hazardous well beyond the almost-certain shutdown and ultimate abandonment of the nuclear reactor sites. Storage at nuclear reactor sites would result in very long-term used nuclear fuel management at a number of sites located next to important bodies of water. This raises security, environmental, and safety issues and adds significant uncertainty, given the potential for changes in institutions and governance, and the likelihood of extreme natural and human-induced events over such an extended time.

5.3.3. Centralized Storage

Long-term storage of used nuclear fuel would involve siting and developing a new, long-term dry storage facility at one site in Canada. While a central facility could be built at an existing nuclear site, for assessment purposes it was assumed that the facility would be sited at a new location. Designs have been prepared for both aboveground and belowground (CTECH, 2003b). One of the aboveground designs is storage in casks and vaults in storage buildings (see Figure 5.7). An example

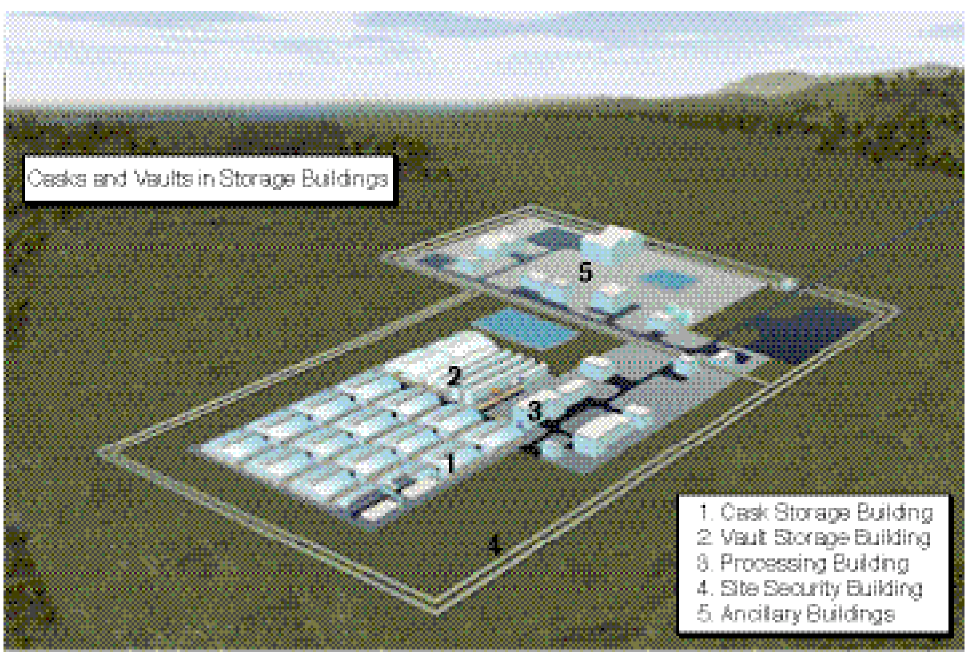

Figure 5.7. Example of centralized aboveground storage—casks and vaults in storage buildings

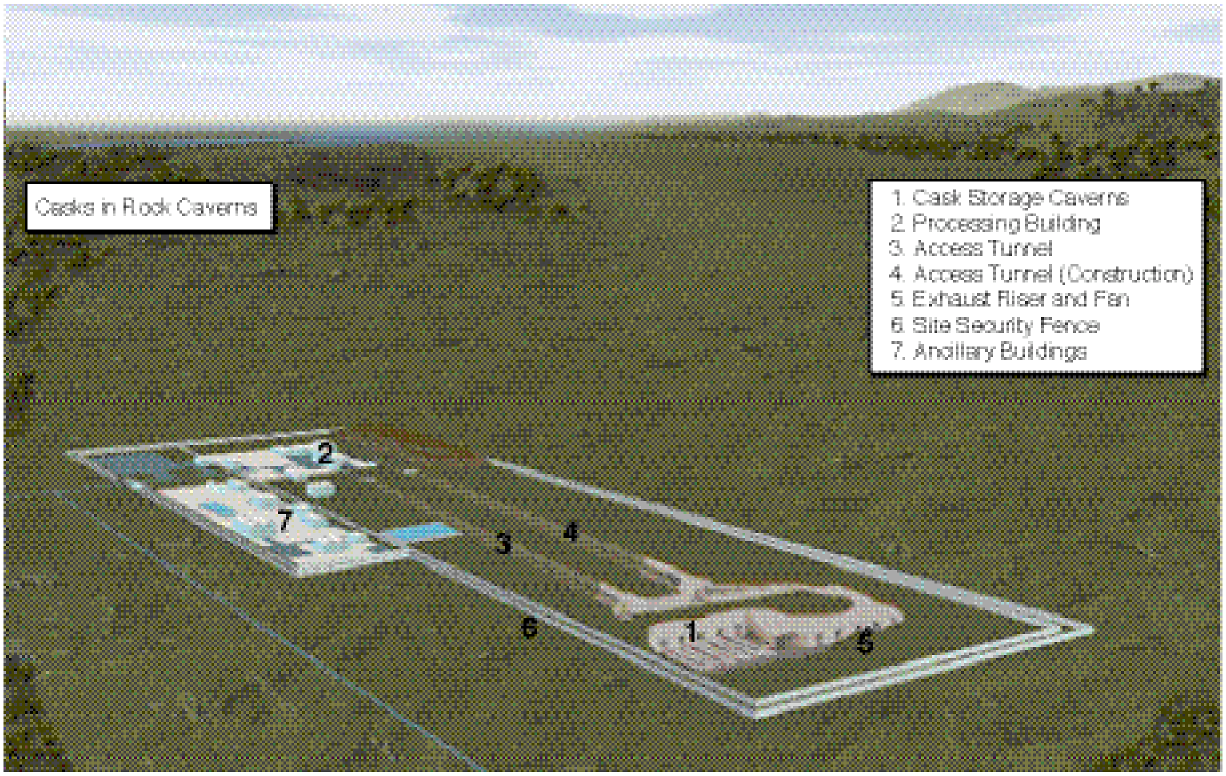

Figure 5.8. Example of centralized belowground storage—casks in rock caverns

of belowground storage is casks in rock caverns excavated to a depth of 50 m in competent rock (see Figures 5.8 and 5.9). The siting, approval, design, and construction of a central storage facility and transportation system could take up to 20 years to complete. Used fuel would need to be transported from existing interim dry storage facilities at the seven reactor sites to the central facility. Once all the used fuel is transferred to the central site, ongoing maintenance, inspections, monitoring, and security, as well as a cyclical program to refurbish or replace storage buildings and repackage used fuel storage containers, would be required. As with reactor site storage, a complete cycle of refurbishment and repackaging of the nuclear fuel storage system is ~300 years.

Analysis of centralized extended storage facilities has shown that the health risks are small and similar to reactor-site extended storage (Garisto, 2004). The most significant changes in health risks are those associated with transportation of used fuel from the reactor sites to the central facility. However, stakeholder input and mitigation measures can be adopted to improve transportation safety, as required.

Advantages: Centralized storage, either aboveground or shallow belowground, would allow for the site selection solely on the basis of used nuclear fuel management. If done well, siting can be achieved with community participation. These are both key potential advantages compared to at-reactor storage and apply to the siting of a deep geological repository as well. Such a site could be either at an already existing nuclear site, if suitable, or at a different site should that prove more advantageous. With the option of shallow belowground storage, some of the security concerns can likely be abated. As with storage at nuclear reactor sites, the required science and technology are well in hand.

Limitations: Centralized storage shares with the storage-at-nuclear-reactor-sites option the key disadvantage of requiring effective and continuing administrative controls and operations, including the required funding, for thousands of years. It also would require the identifica-

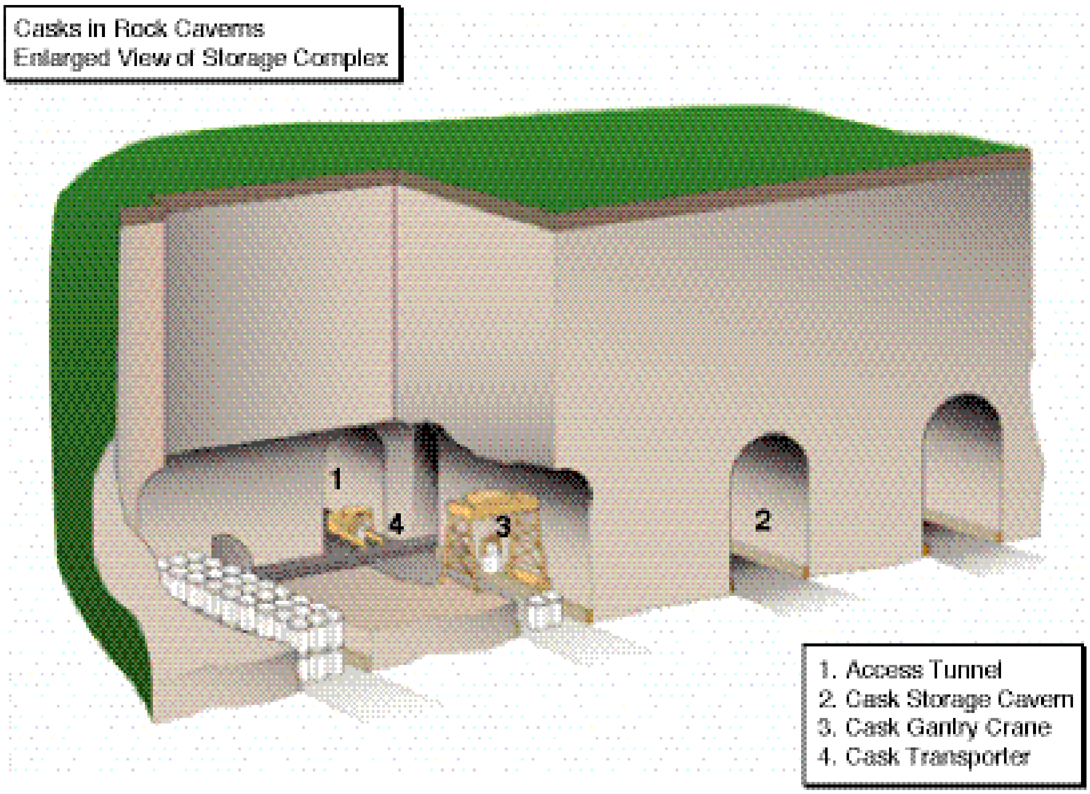

Figure 5.9. Underground caverns for centralized storage—enlarged view

tion and development of a site with potentially contentious community involvement. Transportation of the used nuclear fuel to the site would be required, with its attendant risks and costs.

5.4. A FOURTH OPTION EMERGED–ADAPTIVE PHASED MANAGEMENT

During almost three years of dialogue with Aboriginal peoples, the public, and specialists, the NWMO received very specific direction on both the way in which we should assess the management approaches, and the advantages and limitations of each as judged by interested Canadians. After reviewing each of the three options identified for study in the *Nuclear Fuel Waste Act*, many Canadians suggested that an additional option should be considered, an option that would attempt to capitalize on the advantages of the other three approaches. The NWMO also heard that the way in which a management approach is implemented is as important to its acceptability as the technology used, and we received very specific direction on the requirements of an appropriate implementation plan.

In response to the interest in this fourth option, which combines the strengths of each of the other three options, the NWMO developed the Adaptive Phased Management approach and launched a dialogue with Canadians about its appropriateness in the period June through August 2005. Adaptive Phased Management consists of both a technical method and a management system. The key attributes of the approach are (NWMO, 2005b):

- Centralized containment and isolation of the used fuel in a deep geological repository within a suitable rock formation, such as the crystalline rock of the Canadian Shield or Ordovician sedimentary rock
- Flexibility in the pace and manner of implementation through a phased decision-making process, supported by a program of continuous learning, research and development

- Provision for an optional step in the implementation process in the form of shallow underground storage of used fuel at the central site, prior to final placement in a deep repository
- Continuous monitoring of the used fuel to support data collection and confirmation of the safety and performance of the repository
- Potential for retrievability of the used fuel for an extended period, until such time as a future society makes a determination on the final closure, and the appropriate form and duration of postclosure monitoring

The NWMO would implement this comprehensive approach in compliance with the *Nuclear Fuel Waste Act*, and would:

- Meet or exceed all applicable regulatory standards and requirements for protecting the health, safety and security of humans and the environment.
- Provide financial surety through funding by the nuclear energy corporations (currently Ontario Power Generation Inc., Hydro-Québec and NB Power Nuclear) and Atomic Energy of Canada Limited, according to a financial formula as required by the *Nuclear Fuel Waste Act*.
- Seek an informed, willing community to host the central facilities. The site must meet the scientific and technical criteria chosen to ensure that multiple engineered and natural barriers will protect human beings, other life forms, and the biosphere. Implementation of the approach will respect the social, cultural, and economic aspirations of the affected communities.
- Focus site selection for the facilities on those provinces that are directly involved in the nuclear fuel cycle.
- Sustain the engagement of people and communities throughout the phased process of decision and implementation.
- Be responsive to advances in technology, natural and social science research, Aboriginal Traditional Knowledge, and societal values and expectations.

Among the key insights which emerged from the NWMO study are that any management approach, no matter how well conceived, will fail if it is not also well executed. The process by which a management approach is implemented, and the institutions and systems which are put in place, will be important determinants of the overall effectiveness of the approach and the extent to which it is and continues to be responsive to societal needs and concerns. The process of implementation is very important to social acceptability.

Over the course of dialogues with Aboriginal peoples, the public and specialists, many focused their comments on the features they believe should be part of the implementation plan that accompanies the management approach selected by the Government of Canada. Much of the common ground uncovered in the NWMO study relates to principles and expectations for how decisions will be taken, how citizens will be involved, and how any management approach will be implemented and monitored over time. The NWMO recommendation is accompanied by a commitment to an implementation process that includes a recognition of the need for continuous learning, and a commitment to collaboratively define and periodically assess indicators of progress as a means of facilitating adaptation to evolving conditions.

Adaptive Phased Management is a staged management approach with three phases of implementation, which are briefly summarized below (NWMO, 2005b).

5.4.1. Phase 1: Preparing for Central Used Fuel Management

This phase sets the necessary building blocks for establishing the facilities and infrastructure for long-term management of used fuel. While much has been done to advance the technology for used fuel management in Canada, more research and development work needs to be completed. Adaptive Phased Management would enable us to take the time required to gain greater certainty in the performance of used fuel storage, transportation, containment and isolation technologies–and Canadians would have the opportunity to participate in the radioactive waste management programs in other countries with similar concepts and geographic features.

Phase 1 is expected to take approximately 30 years to implement and would focus on preparing for central used fuel management over the long term. Phase 1 activities would include:

- Maintain storage and monitoring of used fuel at nuclear reactor sites.
- Develop with citizens an engagement program for activities such as design of the process of choosing

a site, development of technology and key decisions during implementation.
- Continue engagement with regulatory authorities to ensure pre-licensing work would be suitable for the subsequent licensing processes.
- Select a central site that has rock formations suitable for shallow underground storage, an underground characterization facility, and a deep geological repository.
- Continue research into technology improvements for used fuel management.
- Initiate licensing process, which triggers the environmental assessment process under the *Canadian Environmental Assessment Act*.
- Undertake site characterization, safety analyses, and an environmental assessment for the shallow underground storage facility, underground characterization facility, and deep geological repository at the central site, and to enable transport of used fuel from the reactor sites.
- Obtain a license to prepare the site.
- Develop and certify transportation containers and used fuel handling capabilities.
- Obtain a license to construct the underground characterization facility at the central site.
- Decide whether or not to proceed with construction of the shallow underground storage facility and to transport used fuel to the central site for storage.
- If a decision is made to construct the shallow underground storage facility, obtain a construction license and then an operating license for the storage facility.

A generalization of Phase 1 of Adaptive Phased Management is illustrated in Figure 5.10.

5.4.2. Phase 2: Central Storage and Technology Demonstration

Phase 2 of Adaptive Phased Management is expected to take an additional 30 years to implement and would focus on central storage and demonstration of long-term containment and isolation technology at the under-

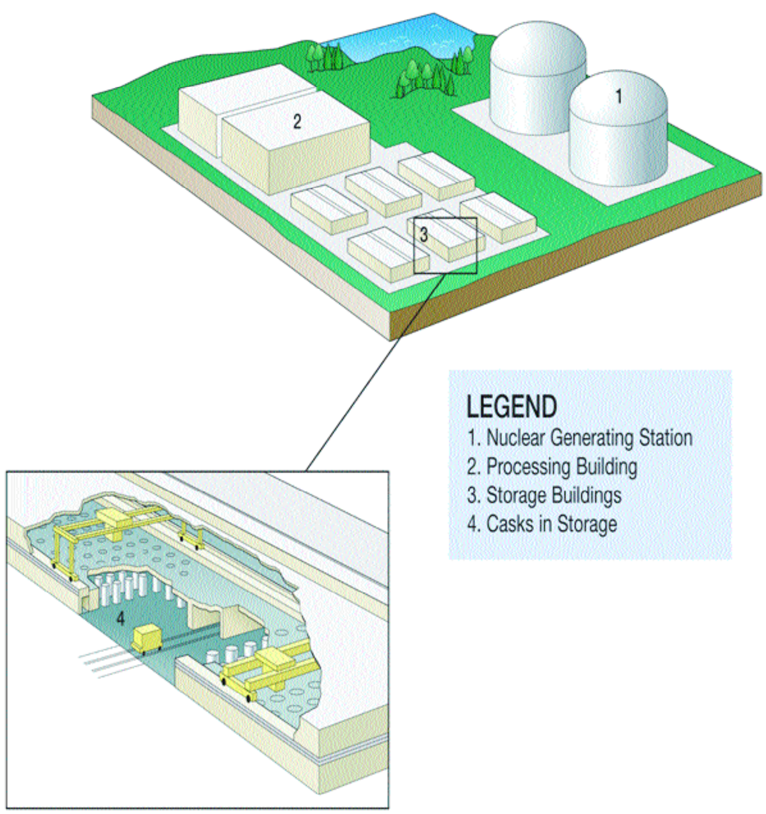

Figure 5.10. Phase 1: Preparing for central used fuel management

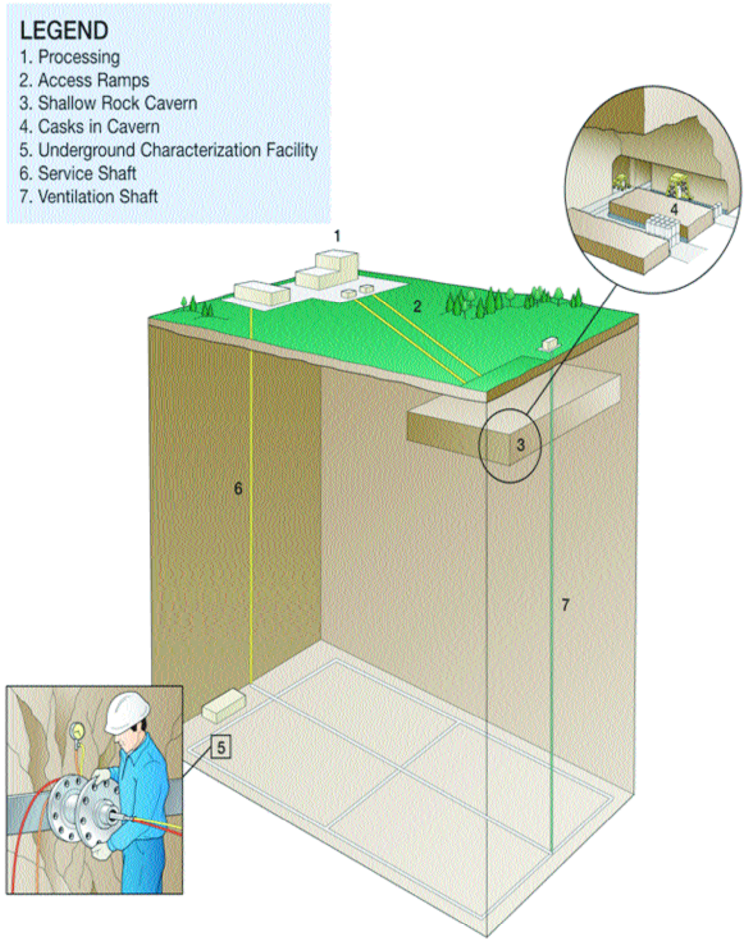

Figure 5.11. Phase 2: Central storage and technology demonstration

ground characterization facility at the central site. Phase 2 activities would include:

- If a decision is made to construct shallow underground storage, begin transport of used fuel from the reactor sites to the central site for storage.
- If a decision is made not to construct shallow underground storage, continue storage of used fuel at reactor sites until the deep repository is available at the central site.
- Conduct research and testing at the underground characterization facility to demonstrate and confirm the suitability of the site and the deep repository technology.
- Engage citizens in the process of assessing the site, the technology, and the timing for placement of used fuel in the deep repository.
- Decide when to construct the deep repository at the central site for long-term containment and isolation.
- Complete the final design and safety analyses to obtain the required operating license for the deep repository and associated surface handling facilities.

A generalization of Phase 2 of Adaptive Phased Management is illustrated in Figure 5.11.

5.4.3. Phase 3: Long-term Containment, Isolation and Monitoring

Phase 3 of Adaptive Phased Management is expected to extend beyond the first 60 years to implement and would focus on long-term containment, isolation, and monitoring. Based on current scientific knowledge, the best way to ensure long-term containment and isolation of used nuclear fuel is to put it in engineered systems underground in a deep geological repository that would keep it isolated from humans and the environment for a

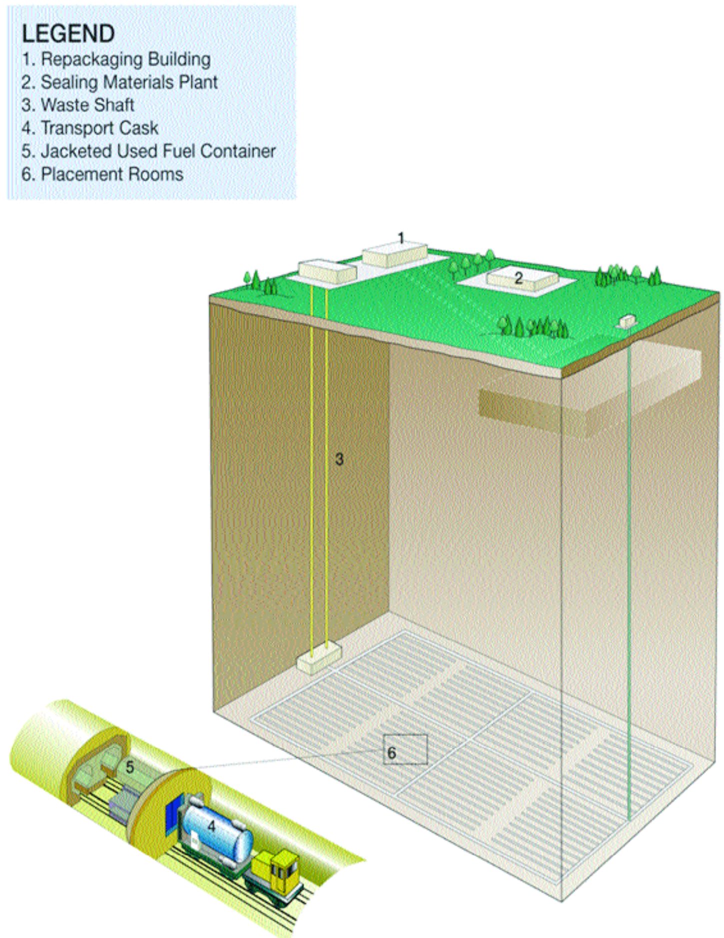

Figure 5.12. Phase 3: Long-term containment, isolation and monitoring

very long time. We have substantial geotechnical evidence, natural analogues, and safety analyses in Canada and internationally to support the deep geological repository concept.

Phase 3 activities would include:

- If used fuel is stored at a central shallow underground facility, retrieve and repackage used fuel into long-lived containers.
- If used fuel is stored at reactor sites, transport used fuel to the central facility for repackaging.
- Place the used fuel containers into the deep geological repository for final containment and isolation.
- Decommission the shallow underground storage facility.
- Continue monitoring and maintain access to the deep repository for an extended period of time to assess the performance of the repository system and to allow retrieval of used fuel, if required.
- Engage citizens in ongoing monitoring of the facility.
- A future generation would decide when to decommission the underground characterization facility and any remaining long-term experiments or demonstrations of technology, and when to close the repository, decommission the surface handling facilities, and what the nature of any postclosure monitoring of the system would involve.

A generalization of Phase 3 of Adaptive Phased Management is illustrated in Figure 5.12.

5.5 DISCUSSION OF MANAGEMENT APPROACHES

Over the course of an iterative dialogue with Aboriginal peoples, the public, and specialists, the NWMO has undertaken a comparison of the benefits, risks and costs of each management approach with those of the other approaches, taking into account the economic regions in which the approach might be implemented, as well as the social and economic considerations associated with it. This comparative analysis of approaches has been documented in a series of reports during the NWMO study process (Ben-Eli et al., 2004; Golder Associates Limited and Gartner Lee Limited 2005a; 2005b; 2005c; NWMO, 2005b).

The storage options, Option 2, Storage at Nuclear Reactor Sites and Option 3, Centralized Storage, are expected to perform well over the near term. However, the NWMO believes that the risks and uncertainties concerning the performance of these storage options over the long term are substantial in the areas of public health and safety, environmental integrity, security, economic viability, and fairness. A key contributing factor to this expected performance is the extent to which the storage options rely on strong institutions and active management to ensure safe and effective performance of the management system. The NWMO expects that these institutions and capacity for active management will be strong over the foreseeable future, but uncertain over the long term.

Option 1, Deep Geological Disposal in the Canadian Shield, is judged to perform well against the objectives in the very long term because of the combination of engineered and natural barriers to isolate the used fuel. A key weakness is its adaptability, which is an important objective for Canadian citizens. Over the short term, the approach is judged to be less flexible in responding to changing knowledge or circumstances either concerning the performance of the system itself over time, or more broadly to innovations in waste management technologies.

Option 4, Adaptive Phased Management is different from Option 1, Deep Geological Disposal in the Canadian Shield, in several respects:

- Adaptive Phased Management is both a technical method and a management system with an emphasis on adaptability. Through a phased implementation process with explicit decision points, new knowledge and technology can be accommodated, as can the societal change that will be inevitable over the time period of implementation.
- The inclusion of sedimentary rock as a potential host rock formation for a deep geological repository is based on international experience and work in Canada, which identified several independent geoscientific arguments suggesting that Ordovician shales and limestones would provide a highly suitable environment for a repository (Mazurek, 2004). This analysis followed previous studies in Canada on the potentially suitability of sedimentary rock for a deep geological repository for used fuel (Aikin et al., 1977; Russell and Gale, 1982; Heystee, 1989). This feature of the approach expands the possible geotechnically suitable sites for a central facility and will provide a greater opportunity to balance a wide range of societal objectives without compromising safety.
- Contingencies against unforeseen events, either natural or human-induced, are built in and funded to ensure that it is this generation of Canadians that is assuming financial responsibility. The optional step of providing shallow underground storage at the central site could respond to calls for enhanced security or the need or wishes of reactor-site communities to move the used fuel more quickly to a central location.
- And finally, these two management approaches were derived in very different ways. The disposal option was developed almost exclusively by scientific, technological, and engineering specialists. Adaptive Phased Management evolved through a process of engagement with citizens, including specialists. Consequently, the approach is built to respond to a broader set of considerations and values, to recognize common ground, and to balance competing objectives.

Overall, the majority of Canadians who engaged in our dialogues considered the Adaptive Phased Management approach to be appropriate and reasonable for Canada (Stratos, 2005; Navigator, 2005). The approach contains a number of design elements that provide people with the comfort they need to accept Adaptive Phased Management as an appropriate approach for Canada.

Dialogue with Canadians has highlighted that an optimal balance must be found between flexibility in the near term, which allows for new learning, and the implementation of an approach that contains and isolates the used fuel in a way that does not require active care by people over the very long term. The NWMO believes that Adaptive Phased Management achieves such a balance.

5.6. CONCLUSION

The NWMO has studied, with Canadians, approaches for long-term care of Canada's used nuclear fuel and has recommended Adaptive Phased Management, a risk management approach that is responsive to what we understand to be the values and expectations of Canadians in providing safe and secure isolation of used fuel for the very long term, and responsible in that it commits this generation of Canadians to take the first steps now to manage the used fuel that we have created over the long term.

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Chapter 6

Deep Geological Disposal of High-Level Radioactive Waste in China: A Three-Step Strategy and Latest Progress

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6.1. THE LATEST NUCLEAR POWER DEVELOPMENT PLAN—30 MORE NUCLEAR REACTORS

As a result of the recent, rapid economic growth in China, electricity shortages have become a major problem over the past few years in some coastal provinces. This situation has made the Chinese government change its policy regarding nuclear power plant development. In 2003, to meet the strong demand for electricity, the government decided to pursue "active development of nuclear power." It has determined that the installed capacity of nuclear power plants (NPP) will reach 40 GW by 2020, and that the electricity produced by these plants will provide only 4% of the total electricity production. This finding means that about 30 more nuclear reactors (1,000 MW-grade) must be constructed before the year 2020.

At present, there are five NPPs in operation on the Chinese mainland: Qinshan, Qinshan-II, and Qinshan-III in eastern China (Zhejiang province), and Daya Bay and Lin'ao in southern China (Guangdong province). Together with the Tianwan NPPs (2 units) under construction, the total electrical capacity produced by NPPs will be 9.13 GW by 2005. This will account for 1.6% of the total installed electricity capacity in China, while the electricity produced accounts for 2.29%.

To safely dispose of high-level radioactive waste generated from the current/future nuclear power plants and other nuclear facilities, China has proposed a long-term program and has been conducting research for the final disposal of high-level radioactive waste since 1985. The China Atomic Energy Authority (CAEA) is the government organization in charge of radioactive waste disposal, whereas the State Environment Protection Administration (SEPA) is the regulatory body. The implementation activities related to radioactive waste disposal are currently managed by the China National Nuclear Corporation (CNNC), while the Beijing Research Institute of Uranium Geology (BRIUG) is the lead institute at present.

It is estimated from the Chinese nuclear power plan that the accumulated spent nuclear fuel will be 1,000 tons by 2010, and 2,000 tons by 2015. After 2020, about 1,000 tons of spent fuel will be produced each year.

The spent nuclear fuel in China will be reprocessed first, followed by vitrification and final geological disposal of high-level radioactive waste. The repository will be a shaft-tunnel model, located in saturated zones in granite.

China has also been building a pilot reprocessing plant, which will be put into operation in 2008. The siting for a commercial reprocessing plant will begin soon, and this plant is scheduled to be built in about 2020.

6.2. STRATEGY FOR HIGH-LEVEL RADIOACTIVE WASTE DISPOSAL AND LONG-TERM PLAN

In 1985, CNNC proposed an R&D program for the deep geological disposal (DGD) of high-level radioactive waste (HLW) (Yang, 1992). The program was divided into four phases: (1) technical preparation, (2) geological study, (3) *in situ* testing, and (4) repository construction. The objective of the program is to build a granitic national geological repository by 2040 that can isolate vitrified waste, transuranic waste, and HLW. The DGD program was preliminary, and it has been revised as the program progresses.

In earlier years, owing to a shortage of experience, there was no exact technical strategy for the development of an HLW repository. However, a significant effort has been made on fundamental research, and the importance of a generic underground research laboratory (URL) has been recognized. URLs have been used only for methodological studies, such as the Stripa Mine and the Aspo URL in Sweden, the Asse Salt Mine in Germany, the Grimsel Test Site and Mt. Terri URL in Switzerland, the URL in Canada, the HADES facility in Mol, Belgium, and the Tono Mine and Kamaishi Mine in Japan.

Based on the experience obtained in other countries, and considering the situation in China, a preliminary three-step development strategy for HLW disposal in China is proposed:

Step 1. Site selection and site characterization
Step 2. Construction of a URL at the potential site for the HLW repository
Step 3. Construction of the HLW repository

Fundamental and supporting research, such as on performance assessment, backfill material, radionuclide migration, natural analogues, and the technology for construction, sealing, and closure of the repository, are all conducted at the same time.

During Step 1, nationwide site screening, regional screening, subregional selection, deep geological environment studies, and site characterization will be carried out, with the objective of finding and confirming the final site. The Beishan area, located in northwest China's Gansu province, has been selected as the area most suitable for an HLW repository. Within the Beishan area, three granitic sections (Jiujing, Xiangyangshan-

Table 6.1. The long-term plan for the implementation of China's high-level radioactive waste repository

Activities	Phase 1: 2001-2005 Site Selection and Site Confirmation	Phase 2: 2015-2030 URL Construction & in situ Tests	Phase 3: 2030-2050 Repository Construction
Site selection and site characterization	Area/site selection, surface investigation, borehole drilling and testing, complete site confirmation	Supplementary work for site characterization	Monitoring of the site
Underground Research Laboratory (URL)	Feasibility study, URL site recommendation, Design of URL	Construction of URL, *In situ* tests & demonstration of disposal technology	Part of *in situ* tests, monitoring of repository
HLW Repository	Conceptual design	Preliminary design and detailed design	Construction completed around 2050
Research and Development	Studies on radionuclide migration, engineered barriers, performance assessment methodologies	Studies on *in situ* tests, radionuclide migration engineered barriers, construction technologies	Studies on repository closure, monitoring, etc.

Xinchang, and Yemaquan) are considered as the most suitable sections. During 1999–2004, surface geological, hydrogeological, and geophysical surveys and borehole drilling have been conducted in the Jiujing and Yemaquan granitic sections, while over the next 5 years (2006–2010), similar work will be conducted on the other sites. During Step 2, a URL will be built at the site selected during Step 1. This URL will serve as a methodological laboratory and for site evaluation. During Step 3, a final repository will be built at the site. It might also be built based on results from a previous URL.

This three-step strategy could serve as a time- and money-saving strategy, because a wealth of methodological data from foreign URLs is now available, and thus a generic URL may not be truly necessary for China's program.

A preliminary long-term plan for the implementation of China's high-level radioactive waste repository is outlined in Table 6.1. It is estimated that a site-specific URL at Beishan will be constructed between 2015 and 2020, and a national repository will be built by ~2050 (Zhang, 2004).

6.3. PROGRESS IN SITE SELECTION AND SITE CHARACTERIZATION

6.3.1. Site Selection

Site selection, which started in 1986, has been an important part of China's HLW disposal program. The entire siting process was divided into four stages: nationwide screening, regional screening, area screening, and site confirmation. During the siting process, the following have been considered: socio-economic and natural factors (including population), economic potential, plant/animal resources, mineral resources, land use, local public attitude, geological/ hydrogeological conditions, and engineering conditions. Since 1985, the following site-selection activities have been conducted.

- Nationwide screening (1985–1986): Following the preliminary siting criteria, five regions were selected as potential repository sites: southwestern China, eastern China, Inner Mongolia, southern China, and northwestern China.
- Regional screening (1986–1989): Based on results from the previous stage, further investigations resulted in the selection of 21 candidate areas. In the northwestern China region, the Beishan area in Gansu Province was found to be the most promising area.
- Area screening (1990–present): Since 1990, most of the site selection effort has been concentrated on the Beishan area. A number of studies have been conducted there, on regional crust stability, tectonic evolution, lithology, and hydrogeology, as well as preliminary geophysical surveys. At the same time, possible host-rock types for the repository were also investigated, with the conclusion that granite would be the most suitable medium for China's repository.

6.3.2. Site Characterization

Since 1993, our efforts have been concentrated on the Beishan area. Investigations in this area have included studies of the regional geologic setting, crustal stability, geological characteristics, hydrogeology, and site characterization methodology. Also during this time, the International Atomic Energy Agency (IAEA) collaborated on China's site characterization program through technical cooperation projects (Project Number: CPR/9/026 and CPR/4/024). Eight granitic sections in this area were selected as candidates for HLW. Among them, three sections (Jiujing, Xiangyangshan-Xinchang, and Yemaquan) were subsequently chosen as the best prospects.

During 1999–2004, the Beijing Research Institute of Uranium Geology conducted site characterization studies at the Jiujing and Yemaquan sections in the Beishan area (Wang, 2001). Field work on surface geology, hydrogeology and geophysical investigations and the drilling of four boreholes (BS01, BS02, BS03, and BS04) were carried out. A series of borehole tests—such as pumping tests, injection tests, borehole televiewer/radar surveys, sample-collection, and geostress measurements—have been conducted. Favorable results were obtained, which provide important data for evaluating the suitability of the Jiujing and Yemaquan sections. We have also gained valuable experience in the methods for evaluating sites in arid, fractured-granitic areas (Wang, 2004b).

Once it is verified that the Beishan area is suitable, a URL will be built there, and more detailed site evaluation, *in situ* tests, and underground experiments will be carried out. The URL will serve as both a methodological laboratory and site confirmation tool, and could be further developed into an actual HLW repository.

6.3.3. Geology of Beishan Area

The Beishan area, in Gansu province, is the preselected area for China's high level radioactive waste repository. The crust in this area ranges in thickness from 47 to 50 km. The seismic intensity of the area is less than 6.0, and no earthquakes with Ms > 4.75 have occurred. The topography of the area is characterized by the flat Gobi Desert and small hills, with elevations above sea level ranging between 1,000 m and 2,000 m. Deviations in height are usually several tens of meters. Since the Tertiary, this has been a slowly uplifting area without obvious differential movements. The geological characteristics of the area show that the crust is stable and has a great potential for the construction of a high-level radioactive waste repository.

Eight granitic sections have been selected as potential host sites for the future URL and HLW repository:

1. Jiujing (monzonitic granite and tonalite)
2. Xinchang (granite)
3. Xiangyangshan (diorite)
4. Yemaquan (diorite)
5. Qianhongquan (granite)
6. Yinmachang-beishan (granite)
7. Xianshuijing (diorite)
8. Baiyuantoushan-Heishantou (granite)

Among them, three sections (Jiujing, Xiangyangshan-Xinchang, and Yemaquan) have been chosen as sites with the highest potential, and detailed work is to be concentrated on them.

6.3.4. Future Plan for Site Selection

Two of these sections (Xiangyangshan-Xinchang and Yemaquan),will be investigated in the coming 5-year period (2006–2010). These investigations will include surface geological/ hydrogeological mapping, geophysical surveying, and borehole studies. In about 2015, one or two candidate sites (each about 4 x 4 km^2 in area) will be proposed for a site-specific underground research laboratory, where further work will be conducted continuously. Based on the findings and water/rock samples collected, substantial laboratory investigations will be carried out, including radionuclide migration experiments (sorption and diffusion of radionuclides on the granite samples), water-rock interaction, water-bentonite interaction, modeling of the site, and preliminary performance assessment.

6.4. MAJOR ACHIEVEMENTS OF SITE CHARACTERIZATION DURING 2002–2004

6.4.1. Geological Mapping in the Yemaquan Section

Yemaquan, one of the candidate sections for a future HLW repository in the Beishan area, is located in the central part of Beishan, 150 km north of Jiayuguan, Gansu province. No permanent inhabitants live in the area. The district is in the arid Gobi Desert, with rocky outcrops and an elevation ranging between 1,400 m and 1,600 m. The mean annual temperature is between 4–7 ºC. Average precipitation is 70 mm/a, while evaporation is about 3,000 mm/a. No year-long stream or other surface water body exists in the area.

In 2002, a geological investigation, at the scale of 1:50,000, was carried out in the Yemaquan area, resulting in the completion of a geological map (Figure 6.1). In this region, the candidate host rock is Yemaquan granite, over an area of 116 km^2, which is composed of three units: the Hongqiquan, Fanxiushan, and Dongtanyaojing, with isotope ages of 290 Ma, 278 Ma, and 200 Ma, respectively. There are three fault groups: an east-west striking group, a north-west striking group, and a north-east striking group, all of which are evident in a surface electromagnetic survey.

6.4.2. Hydrogeological Investigation of the Yemaquan Section

The water-bearing media in the Yemaquan district include fractures of the crystalline basement, fractures, pores of sedimentary rocks, and pores of the Quaternary system. Three groundwater units can be classified in the area: upland rocky fissure, valley and depression pore-fissure, and basin pore-fissure. The upland rocky-fissure unit is the most widely distributed in the Yemaquan district. Water in this unit is mainly phreatic water contained in the weathering and structural fractures. Groundwater recharge is mainly from precipitation infiltration, whereas the discharge is mainly controlled by evaporation and flows into the fracture water-bearing

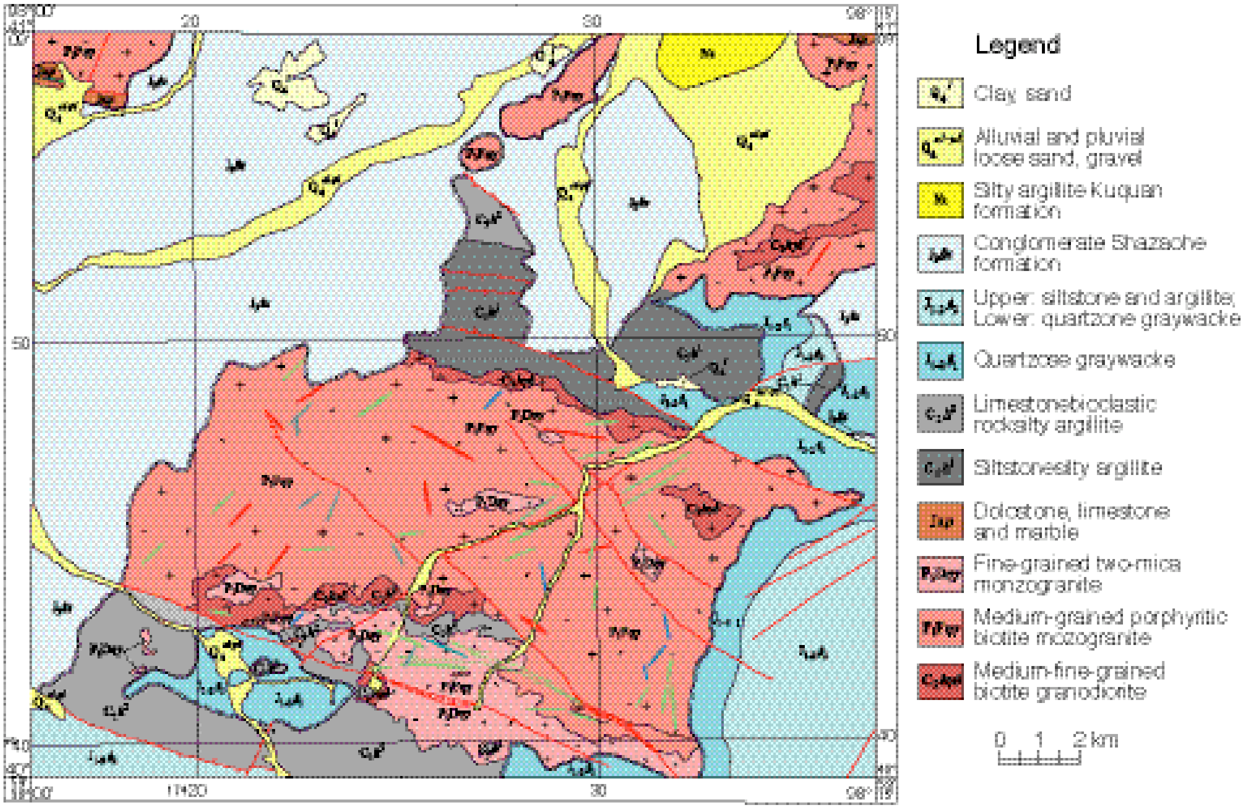

Figure 6.1. Map showing the geology of the Yemaquan section in Beishan, Gansu Province, northwestern China

zone, and underground intermountain runoff. The depth of the water table is generally more than 15 m.

The water content is obviously controlled by topography and geomorphologic landscape. Valley and depression pore-fissure groundwater is mainly recharged by the infiltration of rainfall and temporary flooding, and the main discharge relies on evaporation and runoff towards the basins around the pore-fissure unit. Water table depth for this type is generally less than 5 m. Pore-fissure water is mainly distributed in the basins around the Yemaquan granite body. These basins are mainly composed of Jurassic, Tertiary, and Quaternary sedimentary rocks. The flow of the basin groundwater varies within a wide range, from 10 m^3/d to 1,000 m^3/d, the variation depending mainly on conditions of basin scale, lithology of the aquifer, and structure.

The regional groundwater flow direction is from west to east, but within the Yemaquan granite body, the flow direction is from the center to the periphery of the basin. This groundwater is mainly saline water with total dissolved solids (TDS) > 2 g/L. Water chemical types are mainly Cl^--Na^+, $Cl^-.SO_4^{2-}$–Na^+, and $SO_4^{2-}.Cl^-$-Na^+, indicating that evaporation is the primary mode of discharge.

Thirty-two shallow groundwater samples were collected for hydrogen and oxygen isotope measurement. The major results are: for $\delta^{18}O$: -7.0‰ to -11‰; for δD: -53‰ to -66.5‰; for ^3H: 5.08 to 38.5 TU. The δD-$\delta^{18}O$ plot shows that the groundwater samples fall along a very similar meteoric water line, which indicates that the groundwater is of meteoric origin. In addition, from the groundwater tritium data of the district, we can also conclude that the shallow groundwater system is relatively open, and water circulation in this system is greater than in the deep groundwater.

6.4.3. Drilling of Boreholes BS03 and BS04

In 2003, two more boreholes were completed: BS03 in Jiujing section and BS04 in Yemaquan section. The depth of both boreholes was 500 m. Water samples from borehole BS04 show δD ranging from -73.26.0‰ to -75.9‰, $\delta^{18}O$ from 5.78 to 5.82‰, and ^{3}H of 7.7 TU, reflecting different features from that of the shallow groundwater.

6.4.4. Borehole Hydrogeological Testing

1. Pumping tests. Pumping tests were carried out in BS04 after the end of the drilling. Hydraulic conductivities based on the testing are shown in Table 6.2. From the table, we can see that the water yield was only from 0.00165 to 0.024 L/s when the drawdown varied from 11.04 to 37.04 m. Correspondingly, the hydraulic conductivities were in the 10^{-9} m/s order of magnitude. This indicates that the area around borehole BS04 is very low in water-yield properties.

2. Injection tests: After drilling was completed in BS03, 18 intervals with fractures were chosen to carry out injection tests using a double packer system. The results are shown in Table 6.3; and they indicate that the hydraulic conductivity of the fractured granite ranges from 7.71 x 10^{-9} to 1.14 x 10^{-7} m/s.

6.4.5. Geostress Measurements in BS03

A hydraulic fracturing method was used to measure the geostress field in BS03. Nineteen intervals were successfully tested in borehole BS03. The results show that the maximum horizontal principal stress is 16.92 MPa at a depth of 461.73 m, which belongs to the middle stress

Table 6.2. Pumping test results for borehole BS04 after drilling

Water yield Q (L/s)	Drawdown (m)	Hydraulic conductivities (m/s)
0.00165	11.04	1.278 x 10^{-9}
0.0024	23.04	1.088 x 10^{-9}
0.024	37.04	9.606 x 10^{-9}

Table 6.3. Injection test results for Borehole BS03

Interval	Water Yield (L/minutes)	Hydraulic conductivities (m/s)
127.5~134m	1.0	1.28 x 10^{-8}
135~141.5m	0.6	7.71 x 10^{-9}
148~154.5m	13.6	1.14 x 10^{-7}
156~162.5m	14.6	1.75 x 10^{-7}
171.5~178m	9.0	1.16 x 10^{-7}
187.5~194m	9.4	1.22 x 10^{-7}
195.5~202m	1.2	9.43 x 10^{-9}
209.9~216.4m	6.8	5.37 x 10^{-8}
223~229.5m	14.0	1.82 x 10^{-7}
252~258.5m	5.2	4.12 x 10^{-8}
276~282.5m	8.2	6.50 x 10^{-8}
313.5~320m	6.5	5.17 x 10^{-8}
384.7~391.2m	4.4	3.53 x 10^{-8}
392.2~398.7m	3.6	2.90 x 10^{-8}
402~408.5m	4.2	3.38 x 10^{-8}
455~461.5m	13.1	1.79 x 10^{-7}
467~473.5m	13.8	3.50 x 10^{-7}
487~493.5m	2.6	2.00 x 10^{-7}

region. The minimum horizontal principal stress is 4.83 MPa at a depth of 152.70 m. The direction of maximum horizontal principal stress ranges from NE 65° to NE 71.5°.

6.4.6. Borehole Radar Measurements in BS03

Borehole radar measurements were made in BS03 and BS04. The instrument used was produced by Mala Geosceince Company in Sweden, and the following are the parameters during the measurements:

- Antenna frequency: 250 MHz
- Distance of transmitting or intercepting: 1.9 m
- Sampling frequency: 5316 MHz
- Sampling space: 0.1 m
- Surposition times: 64
- Time window: 141 ns and 188 ns

Figure 6.2 provides an example of radar measurement imaging in BS03. An interpretation of the results shows that there are 21 fractures and 1 punctual reflector along the borehole. The results demonstrate the effectiveness of this kind of measurement.

6.5. PROGRESS IN BUFFER/BACKFILL MATERIAL STUDIES

The concept of geological disposal of high-level radioactive waste in China is based on the concept of a multibarrier system, which combines an isolating geological environment with an engineered barrier system. The buffer/backfill material is one of the main engineered barriers for the repository in China, and bentonite has been selected for this material.

The basic requirement of the buffer is to: retard radionuclide migration by restricting groundwater movement; provide a high sorption capacity for dissolved nuclides; and act as a filter for radionuclide-bearing colloids. To ensure safety over the long time periods of interest, we

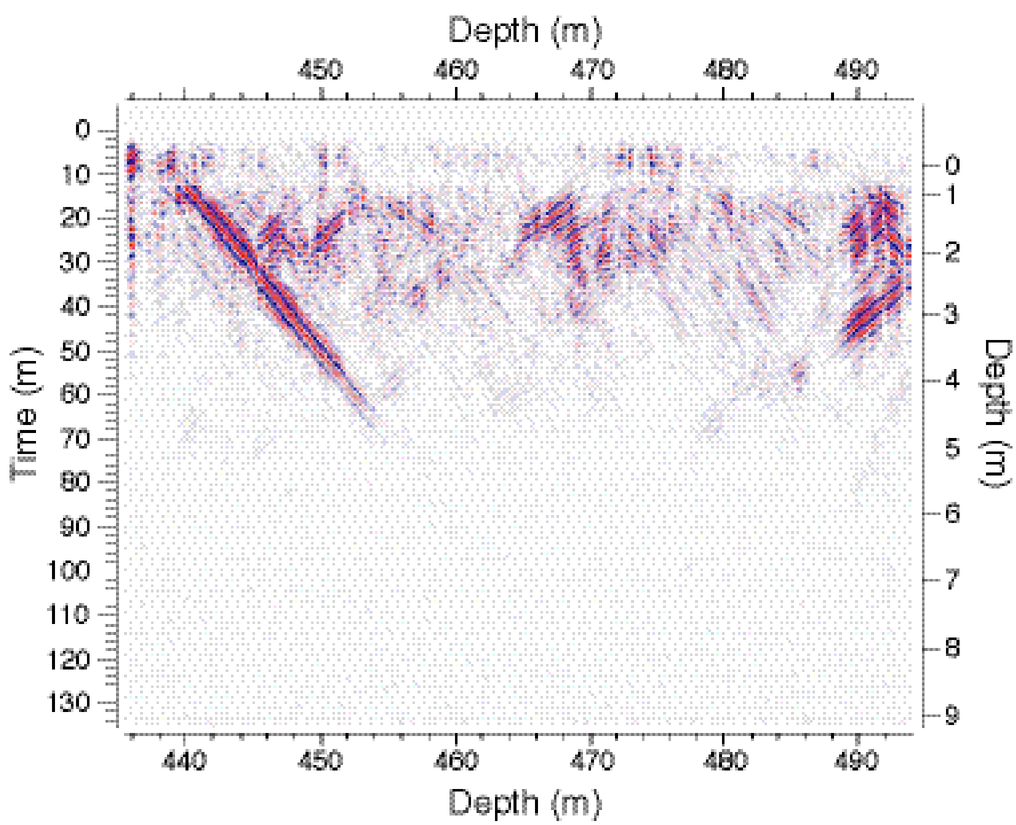

Figure 6.2. Borehole radar measurement image in BS03 (depth at 436 m and 494 m)

must demonstrate that no significant detrimental impact on the physical properties of the buffer material can occur. In parallel, we must also demonstrate the feasibility of manufacturing and installing the buffer. Natural clay is a material that can satisfy all the above functions, to a greater or lesser extent. Among the types of natural clay, bentonite, when compacted, is considered a superior barrier because: (1) it has an exceptionally low water permeability; (2) it fills void spaces in the buffer and fractures in the host rock as it swells upon water uptake; and (3) it has the ability to exchange cations and to adsorb cationic radioelements.

A comprehensive investigation has been carried out to locate suitable deposits of bentonite in China. First, screening criteria for the deposits were proposed:

1. Scale—the candidate bentonite deposit should be large enough to meet the demand for a future HLW repository built 30 or 40 years from now.
2. Quality—as the reference buffer material in an HLW repository, the bentonite must have extremely low permeability, high durability, high swelling properties, suitable thermal conductivity, and reasonable stability, to ensure safety over the long time scales of interest. Given these considerations, Na-bentonite, with high montmorillonite content, is desirable because of its swelling properties and (in general) higher cation exchange capacity.
3. Economic considerations—apart from large reserves, the bentonite mine should also satisfy mineability and economic feasibility requirements. In other words, the candidate mine should be a thin superstratum that would accommodate open-pit mining.
4. Location—transportation of bentonite from the mine to the candidate repository should be convenient, and the cost should be as low as possible. The investigation found 84 major deposits of bentonite in China, with Gaomiaozi (GMZ) bentonite deposits (more fully described in the following paragraph) selected as the candidate buffer material for China's HLW repository.

The GMZ bentonite deposit is a large-scale deposit located in the northern Chinese-Inner Mongolia autonomous region, 300 km northwest of Beijing. Transportation from this mine to the Beishan region would be very convenient. The deposit, with bedded ores, was formed in the upper Jurassic period. Ore minerals include montmorillonite, coexisting with illite. Gangue minerals include feldspar, quartz, calcite, zeolites, cristobalite, and unaltered volcanic glass. The mine area is about 72 km^2 and contains a reserve of about 160 million tons with 120 million tons of Na-bentonite reserves. The major ore body of the deposit extends about 8,150 m, with a thickness ranging from 8.78 m to ~20.47 m.

A preliminary study of the basic properties of GMZ bentonite shows that this bentonite is characterized by a high content of montmorillonite (>70%) and low impurities. Research conducted on its swelling, mechanical, hydraulic, and thermal properties has shown GMZ bentonite to be a good buffer/backfill material (Wang, 2004b). Comprehensive studies, including an investigation of the coupled thermal-hydraulic-mechanical behavior of the bentonite, are still ongoing.

6.6. PROGRESS IN RADIONUCLIDE MIGRATION STUDIES

The migration behavior of radionuclides is critical and fundamental for HLW disposal, and comprehensive studies have been conducted in this area. The radionuclides studied include: Pu-239, Np-237, Am-241, Tc-99, I-131, Cs-134, Cs-137, Co-57, Se-75, and Sr-90. The media studied include granite, bentonite, granite fractures, and some metallic minerals. The important studies include:

1. The sorption and diffusion of Tc-99, Np-237, Pu-239, and Am-241 on bentonite (Wen, 1997; Yao, 2004);
2. The sorption of Tc-99 and I-129 on stibnite, tiemannite, jamsonite, and active carbon of apricot-shell (Zhuang, 2004). The results show that the sorption of iodine for tiemannite and active carbon of apricot-shell are on the order of 10^3 mL/g, while the sorption ratio of TcO_4^- for jamsonite is on the order of 10^4 mL/g. Also, the sorption ratio of Tc-99 for the active carbon of apricot-shell is on the order of 10^4 mL/g. These mineral materials could be considered as additives to bentonite, to prevent the migration of active radionuclides such as Tc-99 and I-129.
3. Sorption of Np-237, Pu-239, and Am-241 for stibnite: our studies show that the sorption ratio of Np-

237, Pu-239, and Am-241 for stibnite can reach 1.7 x 10^3, $\geq$4.2 x 10^4 and 1.4 x 10^4 mL/g respectively.

4. Establishment of a small-scale RADionuclide MIGration study device (RADMIG), which can simulate the conditions of a repository. The specification of the device is T=100 °C, P=5 MPa, Eh < -200 mV.
5. The migration behavior of Np, Pu, and Tc in the Beishan granite, including the measurement of sorption ratio and diffusion coefficient. The experiments were conducted in low-oxygen glove boxes. Those results are site-specific, which are quite useful for performance assessment.
6. The migration behavior of Tc-99 in a natural single granitic fracture: a 2-D model was established to describe the concentration distribution of Tc-99.
7. The studies on the speciation of Np, Pu, Tc, Am in groundwater, especially the groundwater from the boreholes at the Beishan site (Zhang, 2004).
8. The colloids of Np, Pu, and Tc in groundwater, and its impact on the migration of those radionuclides.

6.7. ANALOGUE STUDIES

The following natural analogue studies and anthropogenic studies have been conducted (Wang, 2004b):

1. A natural analogue study of the Lianshanguan uranium deposit. The content of Pu-239 in uranium ores and host rocks was analyzed. Results showed that there has been no Pu-239 migration from the uranium ores since they were formed 1,900 Ma.
2. The geochemical behavior of Iodine-129 at the Lianshanguan uranium deposit (Chen, 1998). The content of I-129 in uranium ore and groundwater was analyzed by accelerator mass spectrometry (AMS). Results showed that the groundwater has leached out I-129 from the uranium ores, indicating that I-129 is a type of active radionuclide.
3. Uranium-series radionuclide and element migration around the Sanerliu granite-hosted uranium deposit in Southern China (Min, 1998). The results show that the migration of U, Th, and most trace elements in the uranium ores and host granite are limited, often less than 30–35 m over 51 Ma. Due to the existence of clay minerals in fractures, the migration of uranium-series radionuclides along the fractures is very limited.
4. Migration of some elements and radionuclides across a granite-granite contact zone (Luo, 1999). A granite-granite contact zone was used for a natural analogue study; results show that the migration distance of major elements, trace elements, and uranium-series is less than 1–2 m.
5. The corrosion layers of unearthed bronze relics dating back to the Xizhou Dynasty, China. These relics, dating back 3,000 years, were used to study corrosion mechanisms, with the results showing that the depth of corrosion is less than 1 mm over the past 3,000 years.

6.8. CONCLUSIONS

High-level radioactive waste disposal is a challenging task, one that is crucial to the sustainable development of the nuclear industry in China. Necessary resources have been arranged for the final geological disposal of high-level waste. A technical strategy and long-term plan have been proposed. Since 1985, progress has been made in site selection and site characterization, backfill material studies, radionuclide migration, and analogue studies. Beishan, located in northwest China's Gansu Province, has been selected as the most promising site for China's high-level radioactive waste repository. During the period 2000–2004, four boreholes were drilled in the Jiujing and Yemaquan sections within the Beishan area; the findings from that activity have shown the advantages of the site. A continued effort will be concentrated on the Beishan area, with associated laboratory research to be carried out in the coming years. The purpose is to prepare for the construction of a high-level radioactive waste repository in China by ~2050.

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Chapter 7

Current Status of the Czech Deep Geological Repository Program

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7.1. INTRODUCTION

Radioactive waste and spent nuclear fuel are generated in the Czech Republic as a consequence of the peaceful use of nuclear energy and ionizing radiation in many industries, particularly in health care, research, and agriculture. The role of nuclear energy in the Czech Republic is determined by the Energy Policy of the Czech Republic, as approved by the government in 2000. In 2004, nuclear power is expected to produce about 37% of the total electricity consumed in the country. This amount shows the importance of Czech nuclear power plants in the energy balance for the country. At present, four pressurized water reactor units of the VVER 440/213 type are in operation at the Dukovany nuclear power plant (NPP), with a total installed capacity of 1,760 MWe. Two other pressurized water reactor units, of the VVER type with a capacity of 1,000 MWe each, are currently in operation at the Temelin NPP. In addition, other nuclear facilities exist throughout the republic, including research reactors, spent fuel storage facilities, and radioactive waste repositories.

The radioactive waste management system in the Czech Republic has a long history. Significant amounts of radioactive waste were produced as early as the beginning of the 20th century, when the exploitation of uranium and radium began. The waste management system included three near-surface low/intermediate-level waste repositories (LILW) intended for institutional waste: Hostim (now closed), the Richard repository near the town of Litomerice, which opened in 1964, and the Bratrstvi repository, near the town of Jachymov, which opened in 1974. Since 1995, LILW from the operation of nuclear power plants has been disposed of at the regional repository located at Dukovany. This repository is also designed for the LILW that will be created by the future decommissioning of both nuclear power plants.

The radioactive waste management system in the Czech Republic changed significantly in 1997, when a new law on the peaceful utilization of nuclear energy and ionizing radiation was passed (the Atomic Act). Since that time, the Czech Republic has had all the preconditions in place to achieve a level of security, reliability, and economic efficiency regarding its nuclear facilities and waste management system, comparable to that of other European Union countries. Upon adoption of the Atomic Act, the Czech government took over the responsibility for radioactive waste disposal; and the Radioactive Waste Repository Authority (RAWRA) was established as a state organization to provide for activities pertaining to the disposal of radioactive waste and spent nuclear fuel. On January 1, 2000, in compliance with the Atomic Act, existing radioactive waste repositories became the responsibility of RAWRA. As a result, the state has taken over full responsibility for the disposal of radioactive waste. The processing of radioactive waste into a form suitable for disposal, except for the processing of spent nuclear fuel, which is to be carried out by RAWRA, remains the responsibility of radioactive waste producers. In 2001, the Ministry of Industry and Trade, which coordinates activities in the nuclear field as part of the government's economic and energy policy, published a key document, *The Concept of Radioactive Waste and Spent Nuclear Fuel Management in the Czech Republic*.

7.2. THE CONCEPT OF RADIOACTIVE WASTE AND SPENT NUCLEAR FUEL MANAGEMENT IN THE CZECH REPUBLIC

The Czech Government finally approved the *Concept* after the conclusion of discussions on the environmental impact assessment portion of the document, in May 2002. The *Concept* serves as a basic document upon which the government's long-term strategy will be built, both with respect to those organizations generating radioactive waste, and those bodies and institutions otherwise involved in radioactive waste and spent nuclear fuel management. Hence, in the long term, the *Concept* also provides guidance for the activities of RAWRA.

It is assumed that the technical equipment essential for the safe management of LILW generated in the Czech Republic is available. Radioactive waste repositories are operated in compliance with current legislation (requirements take the form of limits and conditions) and as per international practice, the performance of disposal systems is periodically assessed. Long-term, low- and medium-level wastes form a relatively small proportion of the total waste produced. However, such waste cannot be accepted by currently operated near-surface repositories. For the time being, waste of this kind is mainly stored unprocessed by the generator, with a small amount being stored by RAWRA.

The conceptual recommendations for high-level waste and spent nuclear fuel management set out by the *Concept* are as follows:

- The permanent storage of high-level waste and spent nuclear fuel would require permanent supervision at storage facilities and the continual replacement of basic storage and supporting system components. The only advantage lies in the availability of spent nuclear fuel as a possible future source of energy. On the other hand, such an availability considerably increases the risk of misuse of these fissile materials. Therefore, permanent storage of high-level waste and spent nuclear fuel is rejected as unrealistic.
- Any final decision concerning the options for managing high-level waste and spent nuclear fuel depends on the feasibility of establishing a deep geological repository (DGR) in the Czech Republic. Hence, a program for the development of a DGR will continue with the selection of its location, confirmation of its suitability, and the design of the whole repository system. Therefore, storage of spent nuclear fuel and high-level waste should be provided for until the DGR is put into operation.
- The option of high-level waste and spent nuclear fuel disposal in an international regional repository has not been excluded, although for the time being, it remains unrealistic. However, should such a project become feasible at a future date, the knowledge acquired in developing a DGR in the Czech Republic should prove invaluable in the construction of a regional repository.
- Advanced separation methods of processing spent nuclear fuel make it possible to dispose of or utilize minority actinides (e.g., by their transformation in new types of reactors). In so doing, usable energy can be generated, thereby fully realizing the energy potential of spent nuclear fuel. Studies of such methods will be financially and scientifically supported.

The *Concept* sets out the basic aims and the direction for the development of the radioactive waste and spent nuclear fuel management system. A full evaluation of

Table 7.1. The targets given by the *Concept* for the high-level waste and spent nuclear fuel management

Target	Date
To select sites with proper geological conditions, taking into account local developments at proposed sites. After evaluation of relevant results, then select two sites into land-use plans (main and reserve) for DGR.	2015
On the basis of geological work performed and complex data analysis, to confirm the suitability of one site for a repository.	2025
To prepare the necessary documentation for construction of an underground research laboratory and performance of long-term experiments for confirmation of safety of deep geological repository.	2030
Operation of deep geological repository.	2065

the aims of the *Concept* is expected around 2010. Essential for its implementation is a speedy verification of the feasibility of a DGR in the Czech Republic (i.e., to identify and confirm a suitable site), or a demonstration that transmutation or spent-nuclear-fuel recycling has been successfully developed.

The *Concept* assumes 3,800 tHM of spent nuclear fuel and ~20,000 m^3 (after conditioning) of high-level (HLW) and other waste which cannot be accepted at existing repositories. These figures exclude new nuclear sources, if any.

The following specific targets provide a framework for fulfilling the aims of the high-level waste and spent nuclear fuel management concept.

In response to a request from the State Office for Nuclear Safety (SONS) in December 2003, the International Atomic Energy Agency (IAEA) put together a team of four international experts to review the status of the Czech Republic's Deep Geological Repository Development Program, implemented by RAWRA under the auspices of the IAEA's Waste Management Assessment and Technical Review Program (WATRP). In May 2004, the team held a review meeting in Prague with RAWRA personnel, board members, subcontractors, and other stakeholders such as SONS and the mayors of towns located near the selected sites. Important issues that had been raised by team members were discussed in detail. The group concluded that the development program is both comprehensive and appropriate. Their conclusions and recom-

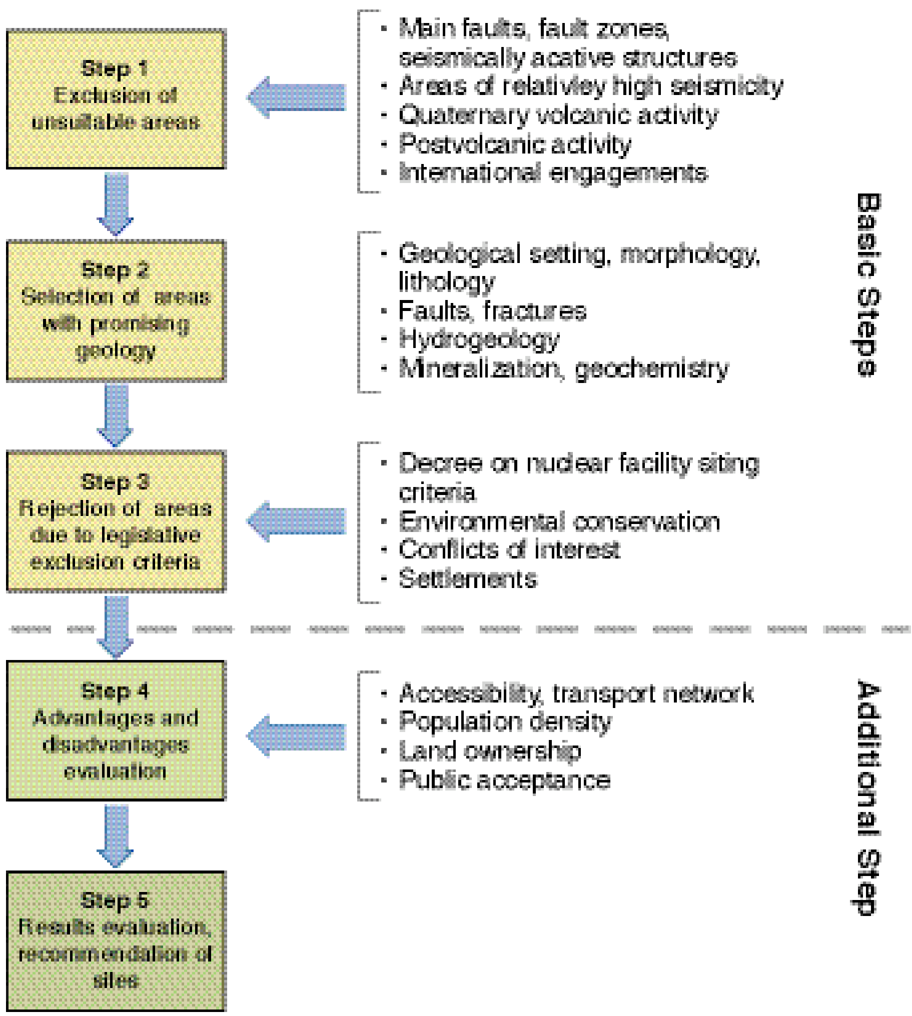

Figure 7.1. Scheme for stepwise site selection during the area survey stage

mendations have been taken into consideration and are now incorporated into the program.

7.3. PROGRESS IN DEEP GEOLOGICAL REPOSITORY SITE SELECTION

7.3.1. Revision of the Area Survey Stage

A number of studies aimed at locating a site for a future DGR were carried out prior to the establishment of RAWRA. These studies were rejected by the Authority, who cited a lack of complexity (only geological information was considered), low transparency, and potential subjectivity. The main thrust of such studies was to collect and evaluate existing geological information relevant to the selection of some promising sites. Subsequently, eight sites were recommended for further consideration.

In 2001, RAWRA commissioned its own area survey. The project took two years to complete and was divided into five steps. The entire geographical area of the nation was included in the survey. Individual steps together with relevant supporting information are shown in Figure 7.1.

Following the completion of Steps 1 to 3, 11 sites were identified as suitable for inclusion in the site characterization stage. Step 4 decreased the number of sites to eight. This evaluation, over the entire nation, was made on the basis of existing information only. No new data were obtained.

The exclusion of unsuitable areas in Step 1 was based on the conditions uncovered through the investigations shown in Table 7.2. In Step 2, the criteria listed in Table 7.3 were used in selecting areas with promising geology.

Table 7.2. Exclusion of unsuitable areas

Main faults, fault zones, seismically active structures	5 km thick strips either side of a fault (fault zone) were excluded (based on expert judgment) to avoid any uncertainty in the location and dip of the fault, particularly concerning depth, and to avoid the effects of dilatancy and a possible increase in seismicity
Areas of relatively high seismicity	An exceptional value I = 7° MSK-64 (IAEA, 1979) was set as the excluding limit. Around deep (more than 50 m) and extended water dams, a zone 5 km wide was excluded to avoid the possible artificial seismicity caused by a sudden change in water level. The connection of geotectonics to old fault systems is known in a few places. Affected areas were excluded.
Quaternary volcanic activity	Areas in which volcanoes of Quaternary age are known to have been active were excluded.
Postvolcanic activity	Areas with occurrences of thermal water and gas release (CO^2) were excluded.
International engagements	With regard to Principle 3 of Safety Series No. 111-F (IAEA 1995), a zone 15 km wide along the state border was excluded. This is considered to be the maximum expected width of the emergency planning zone of a DGR.

Table 7.3. Selection of areas with promising geology

Geological setting, morphology, lithology	The rock body must be of adequate areal extent and thickness. Existing data should support the assumption of a simple structural setting and lithological homogeneity. The number of veins and intercalations of different composition must be very low. Rock properties must not impede repository construction, and the morphology must not block accessibility to repository construction. There must be no old mining activity. Rock properties favorable to the construction of a DGR are essential.
Faults, fractures low	No distinct fault may be present, and existing information should support the assumption of a degree of fracturing.
Hydrogeology	Low flow velocity and hydraulic conductivity should indicate the presence of very old groundwater. Favorable hydrogeochemical properties of water may be assumed.
Mineralization, geochemistry	No potential mineral deposits exist. Available data should indicate that hydrothermal and other rock alteration, on a large scale, are improbable. Favorable geochemical properties of the rock are expected.

Table 7.4. Rejection of areas due to legislative exclusion criteria

Decree on nuclear facility siting criteria	SONS decree No. 215/1997 on the criteria for siting nuclear facilities contains a number of excluding and implicating conditions. Some criteria are relevant to both the surface and subsurface parts of the DGR.
Environmental conservation	Act No. 114/1992 on nature and landscape conservation prohibits or limits certain activities in protected areas (national parks, nature reserves, wildlife refuges)
Conflicts of interest	The most important conflicts of interest result principally from provisions contained in the fol lowing acts: The Waters Act (No. 254/2001), The Spas Act (No. 164/2001), The Construction and Land-Use Planning Act (No. 50/1976), The Mining Act (439/1992)
Settlements	The surface facilities for the deep geological repository must not interfere with settlements, and subsurface works must not have any negative impact on settlements.

The geology of the Czech Republic limits the selection of promising rock types. No salt domes or beds are present, and soft clay formations, tuff layers, and basalt flows are of limited extent. These conditions are not favorable to the siting of a deep geological repository. Sites consisting of claystone are exceptional. Granitoid and metamorphosed rocks (gneiss) make up the most promising rocks types in the Bohemian Massif. To our knowledge, the granitoid bodies are less complicated in terms of their structural and lithological setting.

In Step 3, the legislative norms listed in Table 7.4 were applied.

Step 4, described above as an additional step, evaluates the advantages and disadvantages of the individual sites selected in Steps 1 through 3. On the basis of accessibility, transport infrastructure, population density, land ownership, and public acceptance, it is possible to compare individual selected sites from the point of view of expected complications, before and during construction, as well as from that of costs in the preconstruction preparatory stage.

Based on the new area survey stage (Simunek et al., 2003), the results of which are outlined above, RAWRA decided to begin the site characterization stage at six sites, all of which are located in granitoid bodies. The sites are shown on Figure 7.2, together with a simplified geology and shaded relief.

7.4. PRELIMINARY SITE CHARACTERIZATION STAGE

The preliminary site characterization stage began at the above-mentioned six sites in 2003, the aim being to decrease the areal extent of existing sites (~40 km^2 each) and to recommend the optimal area for detailed site characterization.

The following activities were planned:

- Existing geological information updating
- Conflict of interest updating
- Remote sensing
- Helicopter-borne geophysical measurements
- Field reconnaissance
- Very low frequency (VLF) measurements
- Pre-feasibility study
- Final evaluation and suggestions

A geographical information system (GIS) for RAWRA will be implemented at the same time.

The first and second items on the above list involve the updating of various databases containing information essential to an objective evaluation of each site. Remote sensing involved RADARSAT, Landsat ETM+, and QuickBird satellite imaging. In addition, ZABAGED digital orthophoto maps and stereoscopic aerial photographs were taken. A digital terrain model (DTM) was used in contour line form at a scale of 1:10,000, and in shaded relief form at 1:25,000. Morphotectonic analysis was performed for each site according to standard practice. Standard computer codes were used for data processing and visualization.

McPhar Geosurveys Ltd carried out a helicopter-borne geophysical survey during November 2003 (McPhar 2004). Survey lines, 1,845.1 km in length, were flown in a grid, measuring 200 x 500 m, over all six sites. The magnetometer system (M), gamma-ray spectrometer (GS) and the HummingBird electromagnetic system (EM), with four transmitter-receiver coil-pairs, were

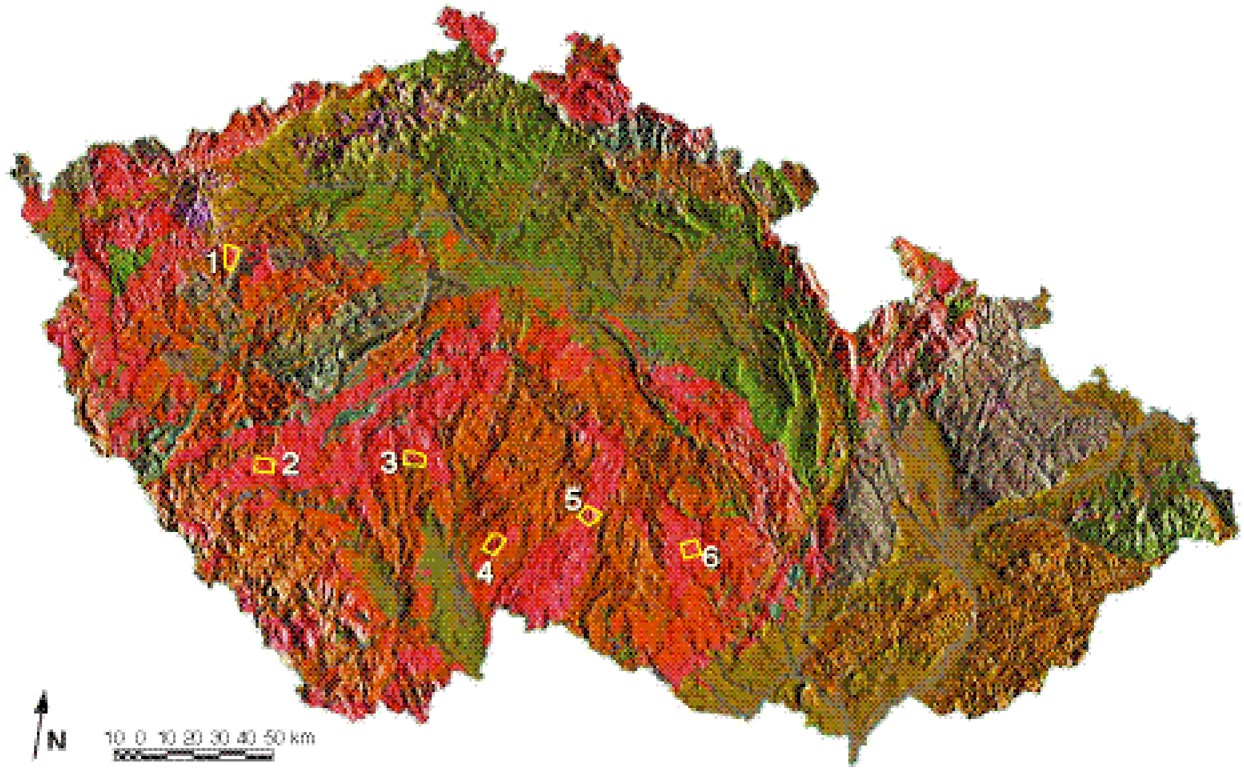

Figure 7.2. Map showing locations selected for site characterization: (1) Lubenec–Blatno, (2) Pacejov Nadrazi, (3) Bozejovice–Vlksice, (4) Lodherov, (5) Rohozna, (6) Budisov

employed in the process. PC-based systems, with large-volume discs, were used for data acquisition. In addition, the helicopter was equipped with all the necessary navigation systems, altimeters, and a digital video imaging system. The helicopter's nominal survey speed was 25 to 30 m/s, the survey height was 60 m, and the height of the "bird" with M and EM was 30 m. Magnetometer and EM data were recorded at intervals of 0.1 second (every 3 m), and GS and navigation data were recorded at intervals of 1.0 second (every 30 m).

Remote-sensing analysis and airborne geophysical measurements have provided very interesting and useful results. Despite an occasionally discouraging prognosis, they have clearly demonstrated the applicability of results obtained (even in the vegetation and demographic conditions of Central Europe).

Individual profiles, in two perpendicular directions, were measured at each site using very low frequency (VLF) measurements. The aim was to evaluate how frequently conductive zones (tectonics, fracturing) occurred. Field reconnaissance concluded these activities.

Currently, all the data are being evaluated, and a prefeasibility study for each site is being prepared. These studies should eventually provide us with some realistic possibilities for the location of both the surface and subsurface components of the DGR and their interconnection.

A final report containing all the results, as well as a recommendation of sites for further consideration, will be submitted to RAWRA at the end of October 2005. After this date, all siting work will be stopped. According to Czech Government decree No. 550, June 2, 2004, all siting activities will be postponed until 2009.

7.5. SUPPORT ACTIVITIES, PROGRAMS, AND PROJECTS

The Czech deep geological repository development program also includes the following activities:

- Test site activities
- Natural analogue studies
- Engineered barriers study

7.5.1. Test Site

The selection of individual polygons exhibiting characteristic geologic conditions, for the testing of individual techniques and methods, was completed during 2003 (Woller and Nachmilner, 2001). Current activities in the Melechov Massif involve the optimization of geophysical measurements and the checking of geophysical results using boreholes. The testing of equipment for hydrogeological measurements in boreholes will be continued in the near future.

7.5.2. Natural Analogue Studies

An investigation of montmorillonite clay, intercalated with organic rich clay containing uranium, is under way at the Ruprechtov site in the northwest part of the country. The sediments are of Oligocene–Miocene age. This study, which began in 1997, is being carried out by the Nuclear Research Institute (Rez) in close cooperation with Gesellschaft für Anlagen und Reaktorsicherheit mbH (GRS, Germany). During the last few years, geochemical and hydrologic models have been variously employed and their results verified with field measurements in new boreholes. A detailed sedimentological study of cores produced the results expected concerning the geological history of sediments.

Research continues on glass colored by uranium-bearing pigments (Woller and Nachmilner, 2001). Samples from several glass factories have been obtained, and these samples have been used in an investigation of how variations in the composition of the glass from different batches impact results. The research has been extended to include the study of slag from the Pribram smelting works, which in the past often used polymetalic ore containing a significant amount of uranium. Preliminary results concerning degradation of the glass matrix and the migration of uranium and other elements are very promising.

A new project involving the study of an anthropogenic analogue began in 2003. The subject of this study is a tunnel excavated for a water pipeline in the granitoids of the Krkonose–Jizera Massif in the northern part of the Republic. The tunnel, 2,593 m in length, was constructed between 1981 and 1984 using two different techniques: a tunnel boring machine, and drill and blast. This provided an opportunity to study and compare the frequency and character of newly formed fractures and the character and extent of the disturbed zone. In addition, precipitated minerals, microbiology, and concrete degradation are being investigated. This project is being carried out by the Czech Geological Survey in conjunction with a number of other organizations and institutions.

7.5.3. Engineered Barriers

The testing of montmorillonite clay from Czech deposits and a comparison of its properties with those of bentonite have been concluded. Various geomechanical properties of these materials were investigated under different conditions, as well as their mineralogical and geochemical properties, including the sorption of different radionuclides. The results confirm that a certain amount of montmorillonite clay can be used as a raw material for the backfill mixture. This project was carried out in cooperation with POSIVA (Finland) and SKB (Sweden).

As part of the ongoing bentonite-barrier research project, a large physical model, the so-called Mock Up CZ, has been constructed at the Centre for Experimental Geotechnics in the Czech Technical University, Prague. This experiment simulates the vertical placement of a radioactive waste container, according to the Swedish KBS-3 concept. The model basically consists of a barrier of bentonite blocks, a heater, and a watering system. The heater, substituting as a thermal source for the spent nuclear fuel, is placed inside the barrier. Water, seeping through the barrier, simulates the invasion of groundwater. The entire experiment is enclosed in a cylinder constructed to withstand the high pressure caused by the swelling of the bentonite. A number of sensors have been installed within the bentonite barrier to monitor changes in pressure, temperature, and moisture. This experiment, supported by RAWRA, is being carried out in conjunction with Prof. R. Push and SKB Sweden.

7.6. INTERNATIONAL COOPERATION

International cooperation is crucial to the successful development of the Czech DGR program, and the increasing amount of international cooperation is seen as a positive sign. In addition to collaboration with

Gesellschaft für Anlagen und Reaktorsicherheit mbH (GRS Germany), POSIVA (Finland), and SKB (Sweden), Czech institutions and companies are currently involved in the following Sixth Euratom Framework Program projects:

- RED-IMPACT
- NF-PRO
- ACTINET-6
- EUROPART
- CETRAD

7.7. CONCLUDING REMARKS

The Czech deep geological repository program is progressing well. The program is in compliance with the approved *Concept of Radioactive Waste and Spent Nuclear Fuel Management in the Czech Republic*, according to which a geological repository will come into operation in 2065. The recent suspension of siting activities by government decree (mentioned above) will have no fundamental impact on the progress and basic milestones of the project. Indeed, the suspension of siting activities will accelerate other parts of the program, and, we hope, help us to improve public acceptance of the overall program in the long term.

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Chapter 8

Beginning of the Underground Characterization for a Spent Nuclear Fuel Repository in Finland

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8.1. INTRODUCTION

In the year 2001, the Finnish Parliament ratified the decision-in-principle related to the disposal of spent fuel from Finnish nuclear power reactors. According to the decision, the repository would be located at Olkiluoto, in the municipality of Eurajoki, on the western coast of Finland, and the disposal would be based on the KBS-3 concept.

As explained in the *Third Worldwide Review* (Witherspoon and Bodvarsson, 2001), the next step of the program was the construction of an underground rock characterization facility on the repository site. The facility would be intended for final assessment of the previous conclusions on the suitability of the Olkiluoto site for a safe geologic repository. A positive outcome of this assessment would make it possible to proceed to submission of the application for the construction license. According to the Government guidelines, this submission would be planned for 2010.

The construction of the underground rock characterization facility (ONKALO) has now been started (as foreseen in 2001). Several hundreds of meters of tunnel length have now been excavated, and the implementation of the underground investigations program is under way. The target depth of the main characterization level is 420 m, but the access tunnel will be built down to a depth of 520 m. The entire facility would be ready in 2010. Allowing for the time needed to do the ONKALO construction work, the Ministry of Trade and Industry (KTM) has moved the target year for submitting the construction license application back two years, to 2012.

In the period that has passed since the *Third Worldwide Review*, the focus of the Posiva program has been on the design and planning of the ONKALO construction and characterization work. One of the important planning and design requirements is that it should be possible to use the ONKALO later as part of the repository. In practice, this means that the facility should be built in compliance with the requirements for nuclear facilities.

In parallel with the ONKALO design, progress has been made in construction, investigation, and development of the technology needed for encapsulation and disposal of the spent nuclear fuel (SNF). Most of the work is now organized into joint projects with the Swedish Nuclear Fuel and Waste Management Company (SKB), pursuant to the agreement signed between SKB and Posiva in 2001. The cooperation agreement grants Posiva access to the Oskarshamn Canister Laboratory and Äspö Hard Rock Laboratory in Sweden. Because of SKB program priorities before 2006, considerable attention has been paid to final testing of the technologies for canister fabrication, sealing, and inspection.

The first plans have also been made for the Safety Case that is needed as part of the application for the construction license. According to the plan, a step-wise process is launched to develop a portfolio of reports that will make the Safety Case. The first version of the port-

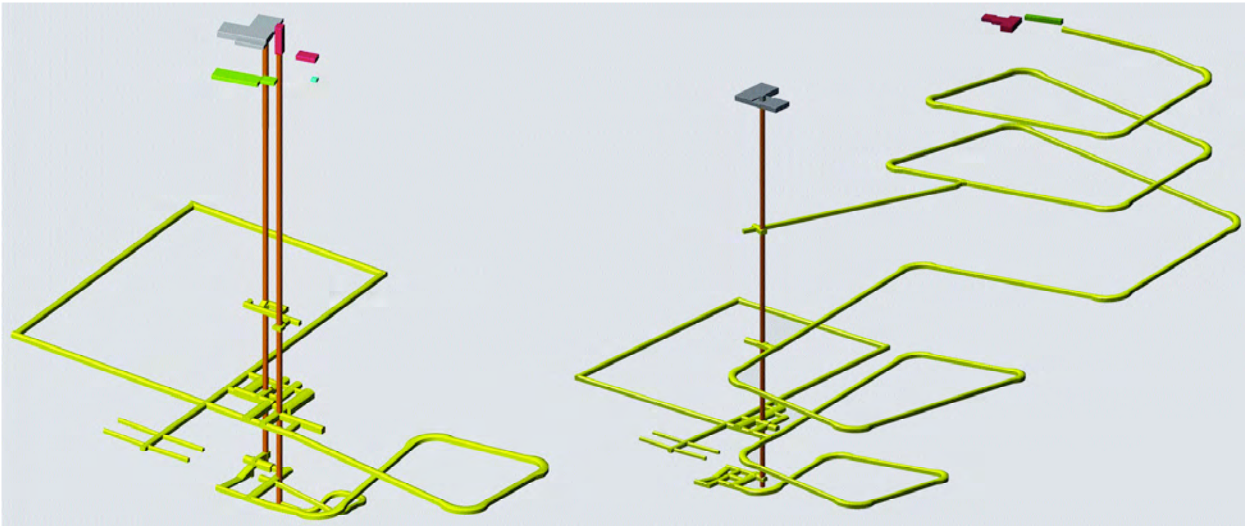

Figure 8.1. The main conceptual design alternatives for the underground rock characterization facility

folio package should be ready for discussion in 2009.

Here, we summarize the progress made in Finland's nuclear waste disposal program since the *Third Worldwide Review* in 2001 (Witherspoon and Bodvarsson, 2001) and outline the program for the near future. A recent account of progress made since 2001 can also be found in the three-year research and technological development program (RTD Program), published in 2003 (TKS-2003, Posiva, 2003a).

8.2. ONKALO

The early design work for the underground rock characterization in the 1990s was based on the idea of vertical access shafts, plus an investigation tunnel at the target repository depth. However, after a systematic comparison of various conceptual alternatives, the decision was made in 2002 that the access to the repository depth would be provided by both an access tunnel and a vertical shaft. The alternatives to this design (at the final stage of this comparison) were to have two access shafts or to have an access tunnel and one shaft (Figure 8.1). The most important criteria in this comparison included the following:

- The ONKALO buildings and structures must comply with the Olkiluoto land use plan, ensuring the needs of the present operation and future projects in the area.
- The ONKALO systems must be based on proven technology that will help to minimize technical risks and improve working conditions.
- The ONKALO must enable the disposal facility to operate as planned in 2020, according to the decisions taken by the KTM and Posiva's owners.
- The ONKALO must enable the collection of sufficient information and knowledge of the repository host rock and other underground conditions, to ensure the long-term safety of the repository and to meet the requirements of its construction license.
- The occupational and operational safety level must correspond to the safety level of the VLJ repository at Olkiluoto. (In other words, the occupational and operational safety-level requirements must be high.)
- Visits to the ONKALO must be possible during and after its construction.
- The ONKALO must comply with the plans for the repository.
- The construction of the ONKALO must not alter the natural (i.e., baseline) conditions in the bedrock, and the conditions otherwise favorable with respect to long-term safety, more than is absolutely necessary for the purposes of construction.
- The environmental impact of the ONKALO's construction must be kept as small as possible, so as not to affect the natural surroundings or change the local inhabitant's living conditions to an unacceptable degree.
- Costing decisions regarding the construction of the ONKALO must take into account the entire life cycle costs of the repository development project, including the sealing of the facility.

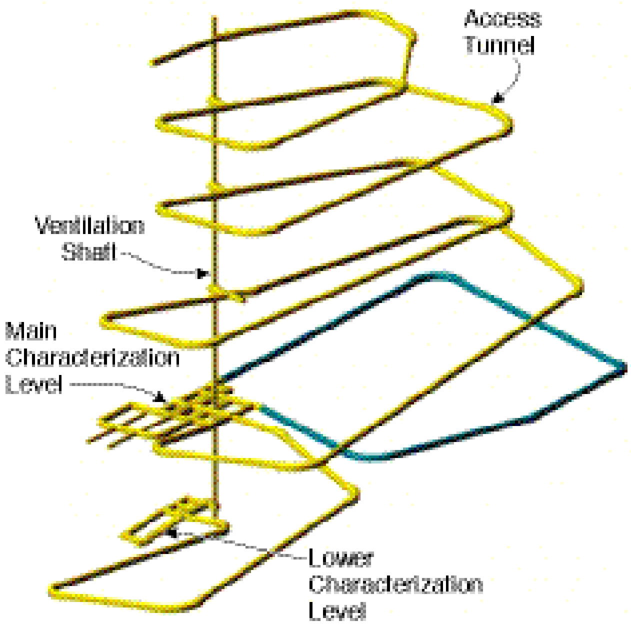

Figure 8.2. The ONKALO design at the Main Drawings Stage

Both of the main alternatives were deemed suitable to meet the primary design criteria. Posiva selected the access tunnel for the next design stage (the main drawing stage) mainly because it provided greater flexibility, was superior in terms of logistics, was associated with a better working environment, and allowed greater opportunities for characterization, especially during its construction.

The final layout was worked out on the basis of the combined tunnel-shaft alternative. In the final version of the main drawings stage, the access tunnel was brought closer to the shaft for efficient ventilation and improved work safety (Figure 8.2). The main characterization level is at the depth of 420 m from the sea level, the lower characterization level is 100 m 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 details of the design are described in Posiva (2003b). A total of 330,000 m^3 of rock will be excavated.

Site preparations for the facility were started in 2003, and the actual excavation work began in September 2004. The tunnelling work has been carried out using normal drill and blast techniques. At the time of writing this report in early May 2005, over 400 m of tunnel length have been excavated. Currently, the design of the facility is being reviewed as to the feasibility of replacing the planned large-diameter shaft with two shafts of smaller diameter.

The tunnel entrance is located in the central part of Olkiluoto Island, near the southern border of the present investigation area (Figure 8.3). An aerial photograph of the ONKALO construction site is shown in Figure 8.4. This location was chosen through comparison of a number of alternatives. In this comparison, one of the main criteria was the expected disturbance to the host rock of the repository; in particular, the inflow of groundwater to the tunnel was to be kept to the minimum. The best way to achieve this was to locate the tunnel in good rock, outside zones of high-groundwater transmissivity. In practice, this meant that the entire tunnel should be built in a well-characterized part of the Olkiluoto area.

The underground characterization and research program (UCRP) to be carried out in the ONKALO is outlined in Posiva (2003c). What Posiva hopes to achieve with the activities proposed for the ONKALO is, of course, that the general suitability of the site will be demonstrated, as only with such confirmation will it be possible to apply for a construction license for the repository. The program during the tunnelling stage includes:

- Drilling of pilot holes
- Probing and mapping of tunnel
- Exploration niches for characterization boreholes and the drilling of these holes
- Hydrogeochemical sampling
- Inflow measurement weirs and other hydrogeological sampling
- Rock mechanics sampling
- Characterization of the ventilation shaft
- Several cored boreholes drilled from the lower parts of the access tunnel to explore rock conditions on the main characterization level.
- Combined and updated modeling description of the entire volume of the rock mass and the development of models for predicting the volume of rock around the main and lower characterization levels. Comparison with earlier predictions.

The first two pilot holes have now been drilled, and the facility is being equipped for gathering data on host rock hydrogeological and hydrogeochemical conditions. An important part of the investigations program is the prediction-outcome process, in which models are used to

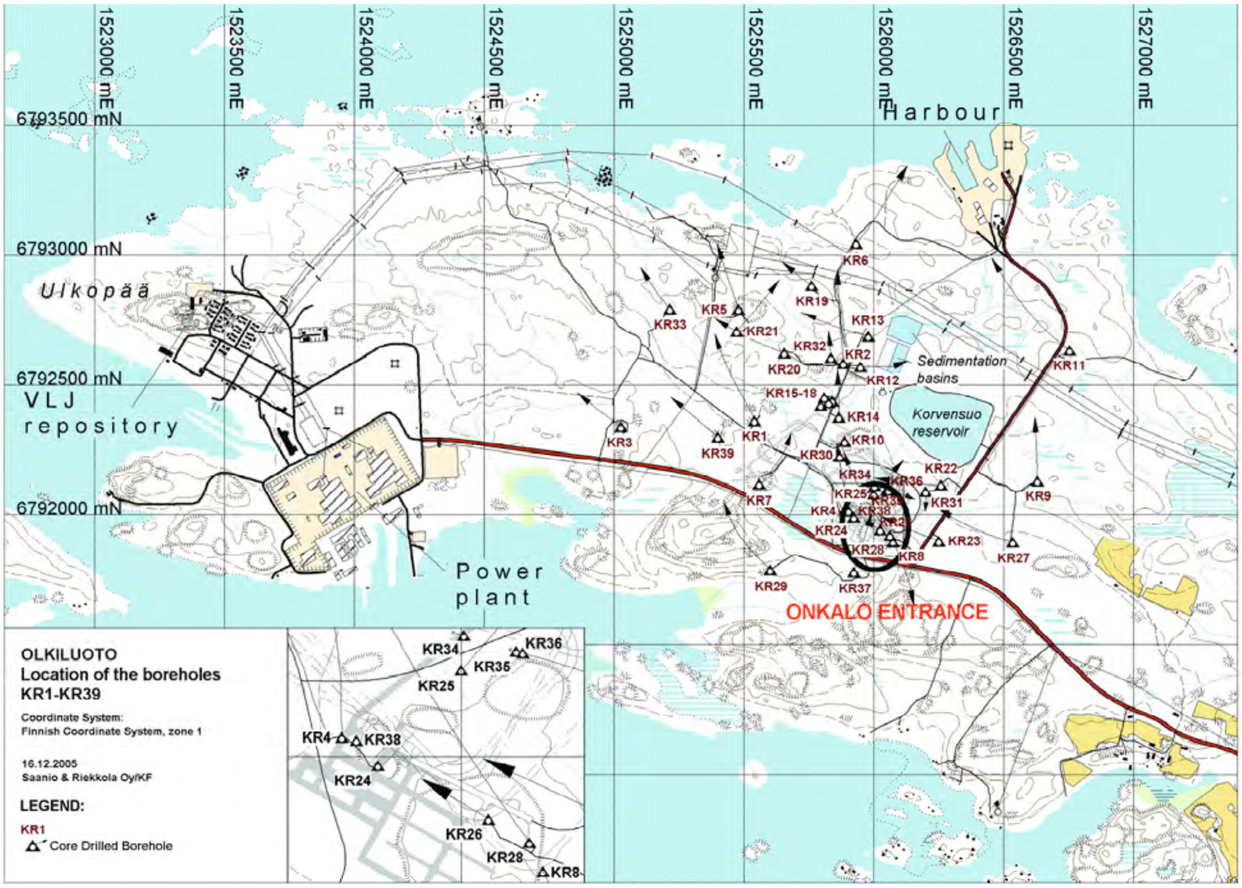

Figure 8.3. The location of the ONKALO entrance in the Olkiluoto investigations area. On the left are the Olkiluoto nuclear power plant units 1 and 2; north of them is the location of the VLJ Repository for low- and intermediate-level wastes.

Figure 8.4. A general view of the ONKALO site in March 2005

predict the conditions ahead. These predictions assist in the further design of the facility and build an important part of the learning process directly into the entire ONKALO program.

The fact that the ONKALO may become a part of the repository means that it has to be designed and constructed according to the requirements for nuclear facilities. Specifically, this means that the construction must comply with the quality assurance criteria posed by STUK, the Radiation and Nuclear Safety Authority in Finland. A specific graded QA program for ONKALO, based on regulatory guidelines and IAEA safety guides, has been launched to complement the ISO 9001-based QA applied by Posiva for its normal RTD activities.

8.3. SITE INVESTIGATIONS

With the construction of the ONKALO, part of the site investigations at Olkiluoto is now moving underground. However, field work at the surface continues as well, consisting of deep drillings, 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., investigation trenches, which nicely complement the lithological and fracturing data so far obtained only from the rather rare outcrops on the island.

The number of deep-cored boreholes at Olkiluoto is 33 (see Figure 8.3), which is before the drilling campaign of 2005. Another 5–6 boreholes will be made during 2005, mainly to assist in the further design of the ONKALO, to check various indications of new fracture zones, and to start the exploration for an extension of the investigation area.

A great deal of emphasis is being given to correctly interpreting the field-investigations data, to build a consistent picture of the site. We are making a special effort to integrate site knowledge through the establishment of the Olkiluoto Modeling Task Force (OMTF). The purpose of the OMTF work is to coordinate and combine the expertise in different disciplines in such a way that a coherent picture of the site can be produced (Figure 8.5). First results from this work have just been published in the 2004 Site Description of Olkiluoto (Posiva 2005).

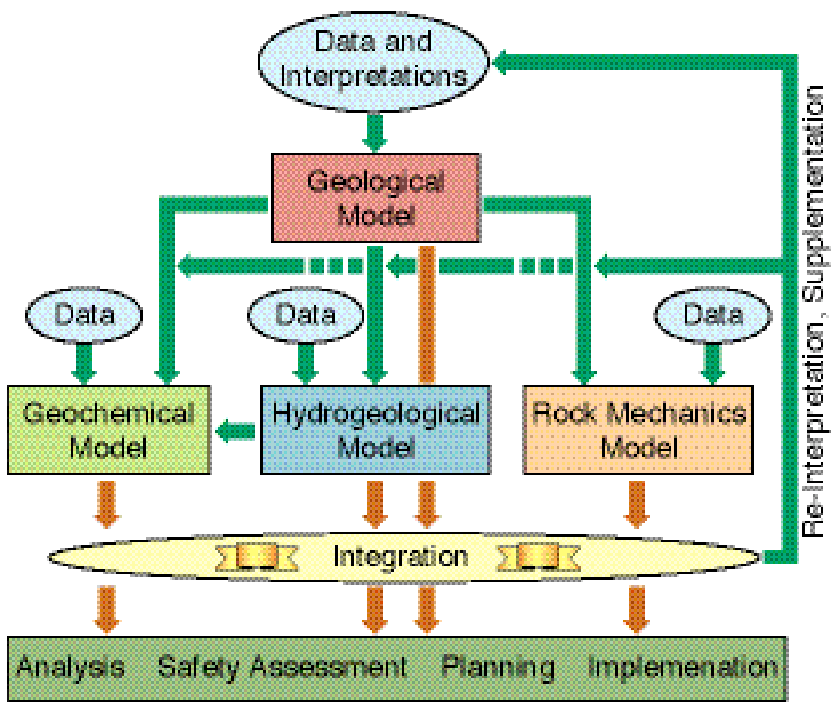

Figure 8.5. The integration of disciplinary models for a coherent site desciption. The geometrical framework of the geological description is the key to this integration (from Posiva, 2003b).

One main objective for the site investigations is to develop suitability criteria for different parts of the repository (access tunnels, deposition tunnels, disposal holes). A three-phase project has just been developed for a first cut at a host-rock classification system on which to base such a suitability assessment (Hagros et al., 2005). The idea has been to look at both the requirements for the host rock and methods to verify them in reality. The plan is to use the ONKALO as a testing ground for further development of this system.

8.4. PROGRESS IN TECHNOLOGY

The development of the encapsulation and repository technology continues, based on the KBS-3 concept. In 2003, a detailed description of the disposal facility was published (the English translation in 2004; Posiva 2004). This description is used for many purposes, such as an information source for stakeholders, but above all it presents the current reference design, based on the accumulated knowledge, and how we plan to implement the disposal system at Olkiluoto.

The design premises have been changed following the decision to construct the new reactor at Olkiluoto (OL3), which should be commissioned for commercial operation in 2009. The main areas in the design and technical development program are:

- Encapsulation of spent nuclear fuel
- Canister development
- Design of the repository together with its above ground facilities
- Development of tunnel backfill

Many of the technical development tasks are conducted as joint projects with SKB. With regard to encapsulation, SKB had already made choices regarding the requirements for canister quality, qualification procedures, manufacturing, sealing, and inspection methods—some years ahead of Posiva. Because of this, Posiva is concentrating on the long-term optimization of processes, such as alternative canister manufacturing methods, while SKB is working on shorter-term issues.

The present canister structure consists of a massive nodular graphite cast iron insert with positions for twelve fuel assemblies and a 50 mm thick overpack of copper, as described in Witherspoon and Bodvarsson (2001). The basic canister design is the same for both the BWR fuel elements of the Olkiluoto plant and the VVER 440 fuel elements of the Loviisa plant, and also for the future OL3 unit.

Several methods for fabrication of the copper overpack have now been tested. Possible alternatives are the extrusion technique, forging, or the so-called pierce and draw technique (which allows the manufacturing of the canister bottom together with the body as one piece; see Figure 8.6). The sealing of the canister was previously one of the main technical challenges in the KBS-3 concept. Posiva's own testing has focused on the high-vacuum electron-beam welding technology in cooperation with the Finnish Patria Aviation Oy, but Posiva has also continued the work carried out by SKB on friction-stir welding technology. The recent announcement by SKB concerning the success of this development program means that two feasible alternatives now exist for sealing the copper canister. Besides manufacturing and sealing of the copper overpack, the third main element in the canister development, the NDT inspection technique, is now also approaching maturity. This work has focused on radiography for volumetric inspection and phased-array ultrasound methods for more detailed detection of possible canister defects.

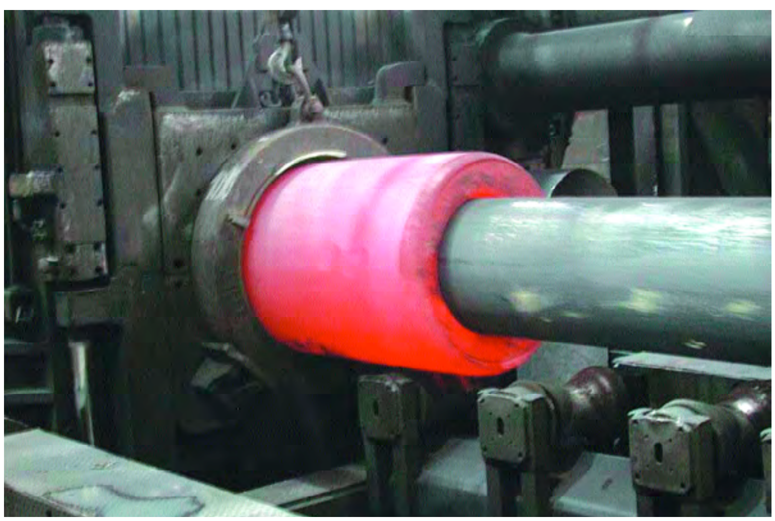

Figure 8.6. Pierce-and-draw method for copper canister manufacturing

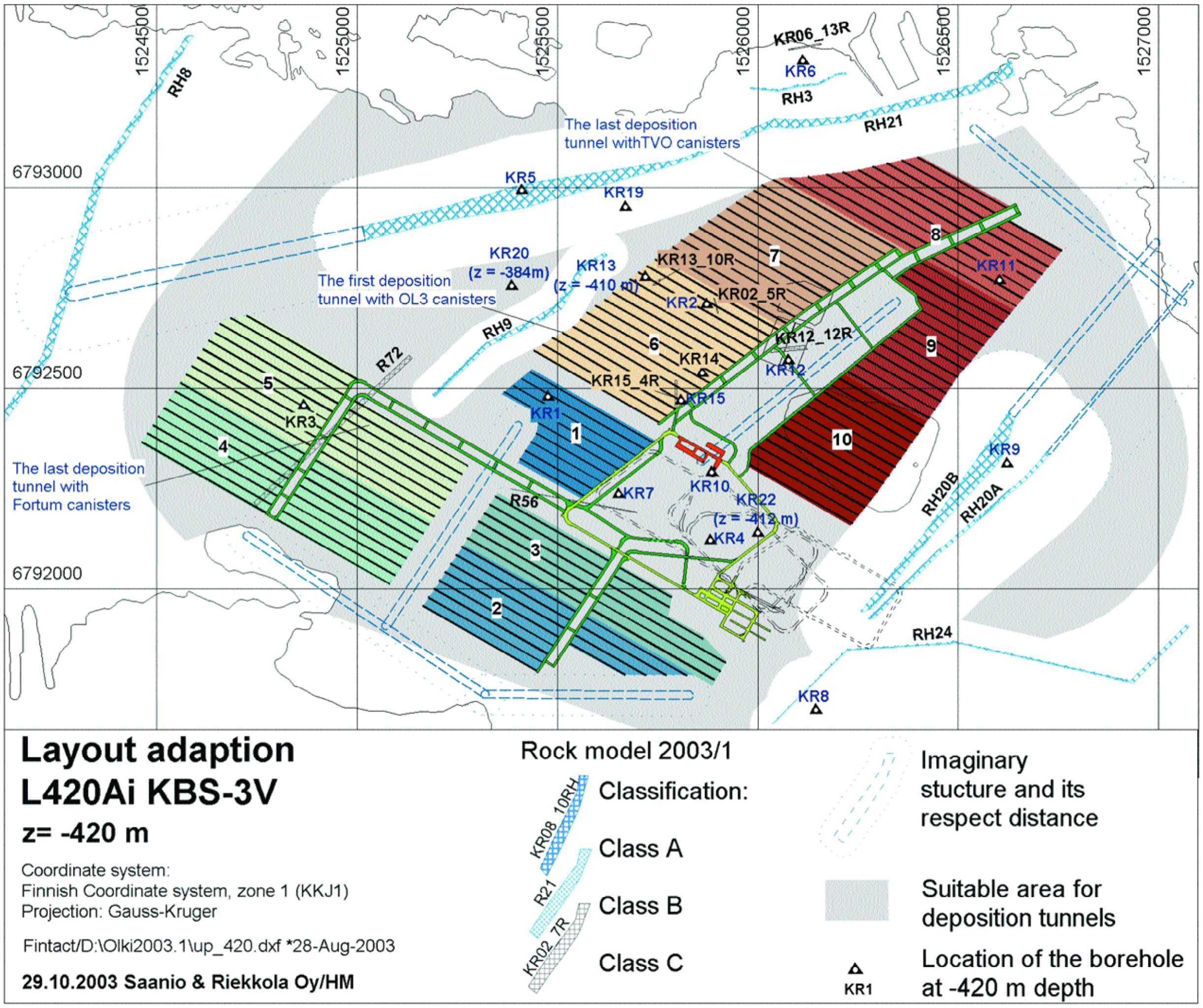

Figure 8.7. Possible layout of tunnels in the Olkiluoto area, based on Rock Model 2003/01

The work on repository technology includes further development of site-specific layout alternatives for the repository panels, more detailed studies of thermal dimensioning of the repository, tests and comparisons of alternative backfilling materials, and design of the technology for canister handling. One important project, in cooperation with SKB and the NUMO of Japan, consists of developing low-pH cement materials for various repository functions.

An example of the current layout thinking is shown in Figure 8.7. The layout is adapted to the known site features according to tentative respective distances to different fracture zones. The requirements for such respective distances will, however, be critically evaluated in the future, together with the development of approaches for suitability criteria.

The plan is to build the repository in several stages, which means that the excavation of the tunnels takes place in parallel with the canister emplacement operations. The separation of the construction and operation activities is facilitated by the new layout, which consists of two parallel main tunnels (Figure 8.8).

An important project is being carried out in cooperation with SKB to assess the feasibility of a horizontal version of the KBS-3 concept. In this concept, the tunnels are

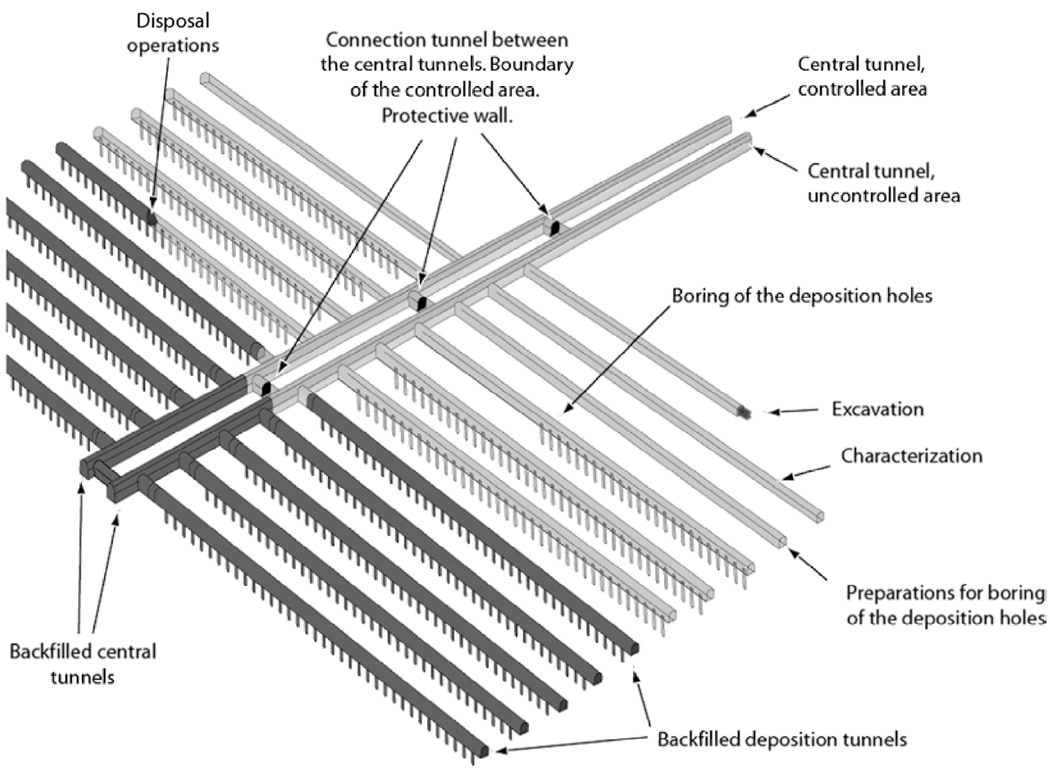

Figure 8.8. Principle of the double central tunnel system

emplaced directly in the tunnels, but now in a horizontal position. The basic components and dimensions remain the same as in the "standard" vertical design. But to facilitate canister and buffer emplacement in their proper positions within the tunnel, they are both put inside an overpack of perforated steel (Figure 8.9). The plan is to complete the present feasibility study by the end of 2007, after which a comparison will be made of its relative advantages and disadvantages vis-à-vis the vertical concept.

8.5. DEVELOPMENT OF THE SAFETY CASE

The main goal of the current program phase is to submit a successful application for the construction license of the disposal facility in 2012. One main ingredient of the application is the preliminary safety analysis report (PSAR), which addresses both the long-term safety and the operational safety of the facility. The long-term safety will be dealt with as a "Safety Case," which is basically a synthesis of evidence, analyses, and arguments that quantify and substantiate the safety, and the level of expert confidence in the safety, of a geological disposal facility for radioactive waste. A Safety Case includes a quantitative safety assessment, which involves the process of systematically analyzing the ability of the disposal facility to provide the safety functions and to meet technical requirements, and evaluating the potential radiological hazards and compliance with safety requirements. The Safety Case broadens the scope of the safety assessment to include the complete range of evidence and arguments that complement and support the reliability of the quantitative analyses.

The Safety Case is built on the safety concept associated with the technical disposal method. According to the current thinking at Posiva, long-term safety will be based primarily on pillars of long-term containment, in particular the isolation provided by the canister (Figure 8.10). However, in conformity with the defense-in-depth, another system of pillars based on retention, retardation, and dilution, provided by barriers other than the container, is there to provide additional protection should some of the canisters fail.

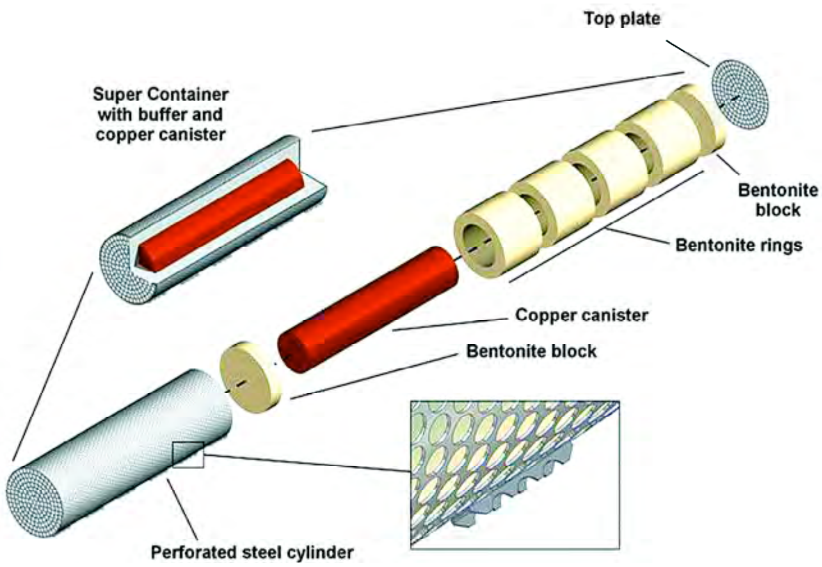

Figure 8.9. The concept of the KBS-3H overpack

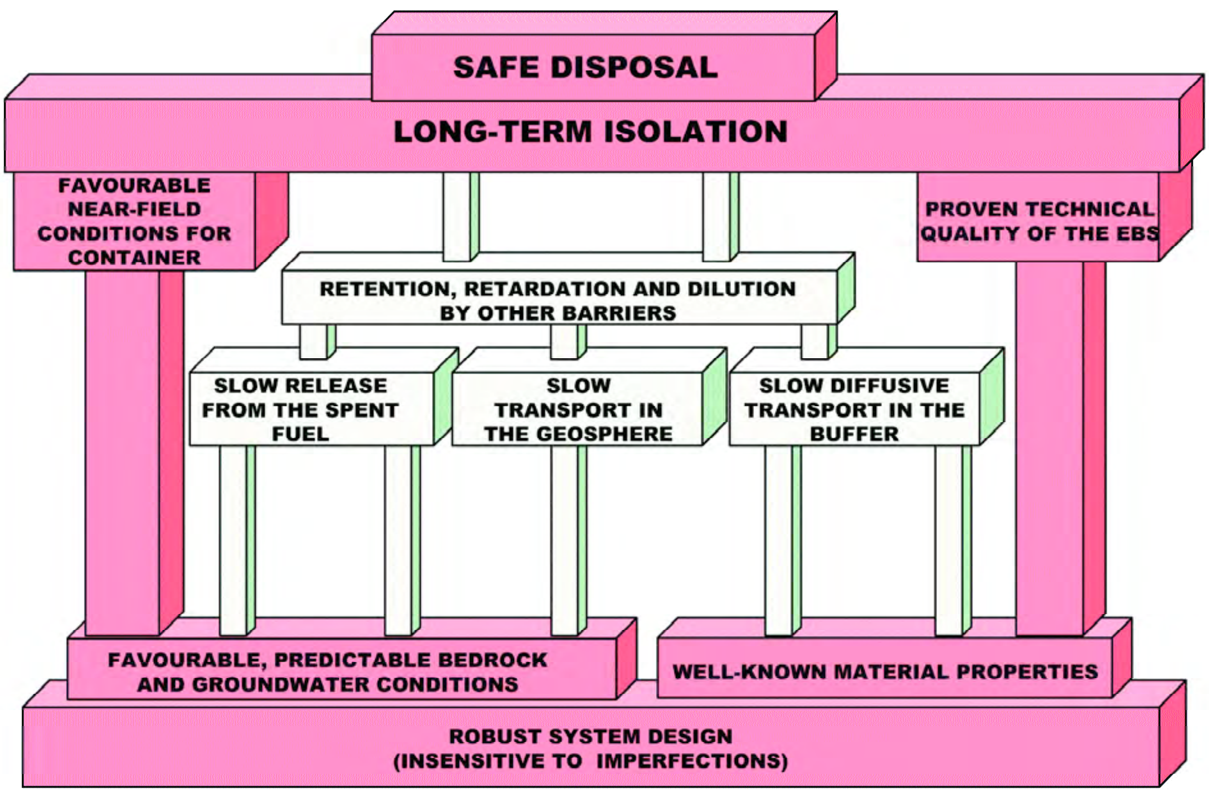

Figure 8.10. Posiva's safety concept

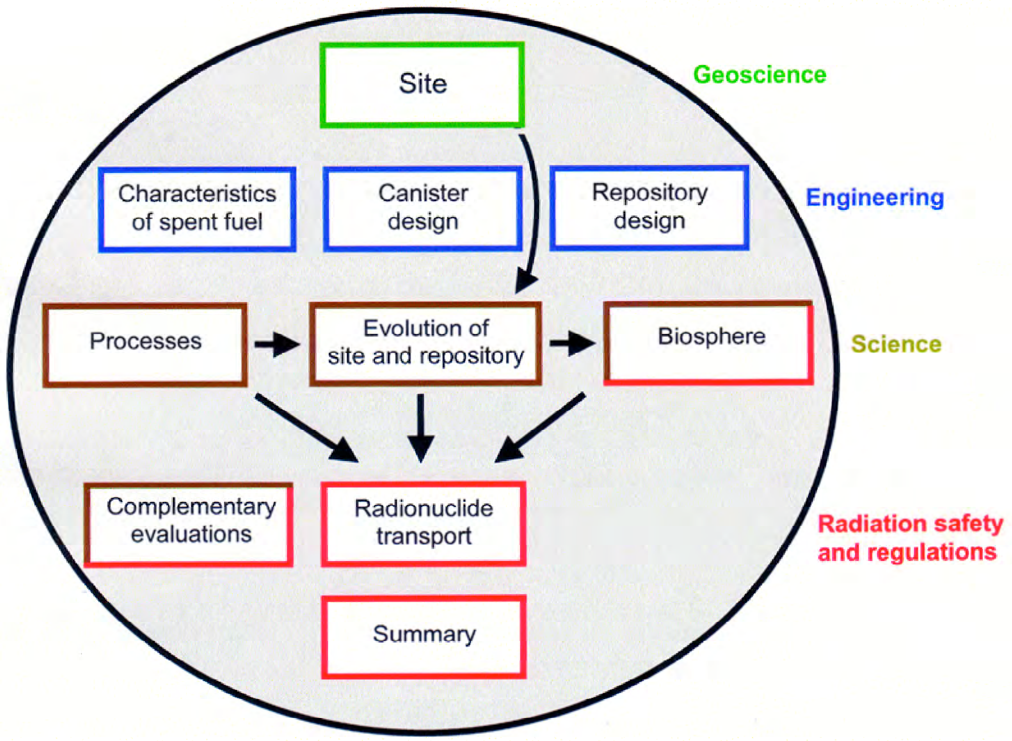

Figure 8.11. Main reports of the Safety Case. The nature of the reports is indicated by the colors of the boxes, and the arrows show the most important transfers of knowledge and data.

The Safety Case will be organized in a portfolio that includes ~10 reports (based on supporting technical reports) that will be periodically updated. The nature of the reports and the most important links between them are illustrated in Figure 8.11.

The site report, describing the present state and past evolution of the Olkiluoto site, forms the geoscientific basis of the Safety Case. The engineering basis is provided by the reports on the characteristics of spent nuclear fuel, canister design, and repository design. The process report containing descriptions and analyses of features, events, and processes potentially affecting the disposal system, and the report on the evolution of the site and repository, form the scientific basis of the Safety Case. The latter report will describe and analyze the evolution of the disposal system, from the emplacement of the first canisters in the repository, over the various transient phases, into the far future. Radiation safety and fulfillment of regulatory requirements will mainly be dealt with in the biosphere assessment, radionuclide transport (safety assessment), and complementary evaluations of safety (e.g., natural analogues) reports. The summary report draws together the key findings and arguments. Interim summary reports will be compiled, and the Safety Case plan will be updated in association with the three-year programs for research, development, and technical design.

8.6. NEXT

With the start of excavation for the underground rock characterization facility, ONKALO, Posiva has taken an important step towards implementation of the plan for final disposal of SNF. The underground characterization phase is required by the current regulations as a necessary step before the construction license and should provide the ultimate proof—or refutation—of the suitability of the site for disposal. In addition, it will provide a great deal of data for design and safety assessment. However, for Posiva, the ONKALO is also an important learning ground for the actual repository work; specifically, it should teach the management how to combine investigations, engineering design, and actual construction work to produce a safe repository. The experience gained so far shows success in terms of many important aspects, such as the management of groundwater inflow. However, our experience also reveals that information about the prospective bedrock is limited, and we must prepare for the unexpected.

Posiva now plans to submit the application for a construction license in 2012. In light of present progress, this goal seems realistic. After the construction license, the next major milestone is the license for operation, scheduled for 2020—this operational phase may take a long time. Given the current plans for nuclear power plant operation, the closure of the disposal facility will likely occur some time in the 2100s.

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Chapter 9

Status of Research on Geological Disposal of High-Level and Long-Lived Radioactive Waste in France

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9.1. LEGAL PROCESS OF RESEARCH

The French Parliament's 30 December 1991 Radioactive Waste Act defined three avenues of research: (1) partitioning and transmutation (inventory reduction), (2) reversible or irreversible disposal in deep geological formations, and (3) waste conditioning and long-term storage. This Waste Act entrusted Andra for research into disposal within deep geological formations, and CEA for both the other avenues. It established a date for a parliamentary decision in 2006 (thus after fifteen years of work, 1991–2006), at which time it would examine how to proceed, based on the results obtained so far.

In December 2001, to prepare for this important milestone, Andra presented a preliminary dossier called the "2001 Argile," as a result of a second iteration loop (1997–2001) in repository design and performance assessment. Besides the review of the work carried out by Andra, this interim report has the status of a methodological test, especially when considering the long-term safety analysis. It aimed also at focusing the research program on the key issues for this safety analysis. This dossier was reviewed in 2002 by the International Review Team under OECD/NEA aegis (Peer Review) at the request of French Research and Industry Ministries. It was also examined by the French regulatory body, at the request of the Safety Authority.

A key input to any management of radioactive waste is the reliable assessment of the waste inventory, taking into account both existing and committed waste. In 1999, Y. LeBars was appointed as Andra president and entrusted with the mission to set up a prospective inventory of radioactive waste, including recoverable nuclear materials. The inventory work was started in 2002 and resulted in the publication of an initial report in October 2004. This document consists of a summary booklet for the general public, a synthesis report that offers readers the possibility of requesting a more detailed set of information, and technical documents that present the waste families in detail, together with the geographical locations of this radioactive waste by major region.

During this 15-year period, hearings have been periodically organized by the French Parliamentary Office for Assessment of Scientific and Technological Options. The transparency of information was ensured by public reports of the National Evaluation Board (CNE), by annual reports of the steering committee on the 3 avenues for research, chaired by the Ministry for Research (COSRAC), and by the opening of the URL (underground research laboratory) site to visitors. An independent local ad hoc committee (CLIS) on the Meuse Haute-Marne site has provided public debate and information. Recently, in 2005, the government has decided to call for a public debate on radioactive waste management orientations, to be organized by the National Committee for Public Debate (CNDP) in September 2005.

The purpose of the 2006 date stipulated by the Waste Act is to define a high-level, long-lived waste management strategy based on the results of research. In this context, both the research steering bodies (Andra and CEA) submitted a scientific-assessment dossier in June 2005. (and will issue a supplemented version in December 2005 that includes updated results from the Meuse/Haute-Marne

Underground Research Laboratory). This "Dossier 2005," on radioactive waste disposal in a geological formation, concerns both the Meuse/Haute-Marne clay site (Dossier 2005 "Argile") and a nonspecified granitic site (Dossier 2005 "Granite"). It represents 15 years of research that involved nearly 100 laboratories, 7 laboratory groups, and partnerships with leading French and foreign research organizations, including European Union-funded research projects.

Dossier 2005 will be assessed by the appropriate bodies (CNE, Nuclear Safety Authority). Specifically, Dossier 2005 Argile (dedicated specifically to the Meuse/Haute-Marne site) will undergo an international peer review under NEA aegis. Following these assessments and public discussion, the French government is expected to propose a bill to Parliament for debate in the first quarter of 2006.

9.2. THE RESEARCH PROGRAM OF THE MEUSE/HAUTE-MARNE URL

Since 1994, 27 specific deep boreholes have been drilled, and 2,300 m of argillite (Callovo-Oxfordian host formation) core samples have been obtained (over 4,200 m of cored boreholes). Andra took more than 30,000 samples (including 7,300 fluid samples) and analyzed 5,300 rock samples. Specific 2-D and 3-D seismic campaigns were carried out at the site, and 1,300 km of additional seismic lines acquired from the oil industry were reprocessed and analyzed.

In 2003, eight boreholes, drilled in five different locations spread over a large surface area around the laboratory, allowed investigators to assess the hydrodynamic characteristics of the surrounding limestone formations (overlying Oxfordian and underlying upper levels of Dogger), notably hydraulic head and permeability. Water samples were taken to analyze krypton^{81} and chlorine^{36} concentrations and to determine the groundwater residence time. These new data consolidated the geological and hydrogeological models of the site.

In 2003–2004, four deviated boreholes, using oil-industry drilling techniques for subhorizontal surveying, confirmed the homogeneity of the host formation and the absence of faults. More than 1,300 m of core samples have been collected from the Callovo-Oxfordian formation, and more than 300 m from the Dogger formation. These cores have very few microcracks in the Callovo-Oxfordian, and the microcracks are located at the top and bottom parts of the host formation and appear to be sealed. They are hectometrically spaced and approximately 1 m long. *In situ* measurements confirmed that the Callovo-Oxfordian argillite has a very low permeability, and that these features have no impact on its hydraulic conductivity. An extensive program was carried out in these boreholes to measure the *in situ* stress field in the Callovo-Oxfordian and its surrounding formations.

Simultaneously with the construction of the URL and the first experimental drifts at the 445 m level, a campaign of vertical deep boreholes allowed important parameters to be measured. For example, Andra carried out: (1) gas injection tests in a 540 m deep borehole to study how gas migrates within the argillite; and (2) a challenging diffusion test, similar to those conducted from the experimental underground drift in a borehole drilled at 500 m.

The construction of the URL started in August 2000. Between 2000 and 2005, the sinking and construction of the two shafts included numerous scientific investigations. These shafts were used in a detailed geological survey of the Oxfordian and the Callovo-Oxfordian, reached in 2004—particularly to study the hydraulic and mechanical impact of the excavations and to assess the extent of the rock damage. Furthermore, since November 2004, 40 m of drifts have been made available for *in situ* scientific experiments, at a depth of 445 m, allowing for the emplacement of 350 sensors (Figure 9.1).

The experimental program in these first drifts (445 m level) consists of:

- A vertical mine-by-test experiment around the shaft (15 boreholes oriented in directions chosen according to the main stress field around the shaft and at different distances from the shaft axis). Since the main shaft sinking was resumed in March 2005 (sinking was stopped during the construction of the drift at –445 m in August 2004), the sensors installed in these boreholes beforehand have monitored the immediate impact as shaft sinking was approaching and reaching the monitored volume of Callovo-Oxfordian, as well as the deferred effects on the Callovo-Oxfordian argillites.
- Two boreholes equipped to sample the water and

Figure 9.1. Monitoring equipment for the different experiments conducted in the first experimental drift (445 m level)

gas dissolved in the argillite, to carry out *in situ* measurements of various chemical parameters and to specify the chemical composition of the interstitial water.

- Two horizontal boreholes fitted out to complement the argillite permeability data (with the resulting data confirming the low permeability of the rock already measured).
- An *in situ* diffusion experiment in three boreholes to measure the diffusion velocity of water (using tritiated water) and of various elements prevalent in the repository (or similar ones that are less radioactive and have a shorter half life, such as iodine, chlorine^{36}, sodium^{22}, or cesium^{134}).

The 490 m URL main level was completed in early October 2005, and several experiments were carried out at the end of 2005:

- Characterizing the excavation-disturbed zone (EDZ)
- Testing the construction and performance of an anchoring key using swelling clay
- Measuring the thermal response of argillite

The water sampling and diffusion experiments already carried out at 445 m are repeated at this main level to complement the measurements. These new data justify the issue of the updated version of Dossier 2005 Argile for the end of 2005. Moreover, the monitoring for the experiment will remain in operation and will continue to provide data in the medium term beyond 2005.

In 1996, Andra joined the Mont Terri project's international consortium, set up to conduct *in situ* experimentation in an exploratory tunnel for future motorway construction (Jura canton and republic in Switzerland). Since 2002, a program has been carried out to validate the full-scale models defined on the basis of data collected by the Meuse/Haute-Marne underground laboratory: diffusion experiments, gas migration tests, geochemistry characterization, and heating tests to assess the mechanical behavior of the rock. It also includes full-scale engineering tests.

Experiments carried out at Mont Terri correspond to two specific objectives for Andra:

1. To prepare the experiments in the Meuse Haute-Marne underground laboratory under the best possible conditions
2. To provide data for validating and upscaling the models developed on a small scale for the larger-scale Callovo-Oxfordien argillites, as far as transferability is possible.

9.3. DOSSIER 2005 ARGILE

Dossier 2005 Argile presents the results of 15 years of research and the outcome of a third iteration loop (2002–2005) in the repository design and performance assessment. It is based on the knowledge acquired since 1991 and collates scientific knowledge, design data, understanding of the total repository system, and safety analyses. The study reaches a positive conclusion concerning the basic feasibility of geological disposal in the Callovo-Oxfordian formation of Meuse/Haute-Marne and sketches out the prospects for subsequent development.

9.3.1. The Waste Inventory

Dossier 2005 takes account of existing and future wastes committed by the French PWR fleet, or some 45,000 tons of heavy metal unloaded from the reactors (40 years of operation per reactor). The future production is estimated using four study scenarios, without anticipating a particular industrial arrangement:

- Reprocessing of all EDF spent nuclear fuels, UOX, and MOX (Scenario 1a)
- Reprocessing of UOX spent nuclear fuels, but no reprocessing of MOX spent nuclear fuel, with continued production of current vitrified packages (Scenario 1b), or production of hotter vitrified-then-concentrated packages: (Scenario 1c)
- End of reprocessing in 2010 and disposal of UOX and MOX spent fuel (Scenario 2)

As an exploratory approach, spent nuclear fuel (SNF) was taken into consideration, although it is not considered to be waste, according to current French law.

The inventory of primary packages, conducted with the waste producers, aims to be as complete as possible. For future waste packages, hypotheses were formulated. This approach leads to a robust inventory, capable of covering various industrial options. A classification of these different packages was made and is used as the basis for the studies.

9.3.2. Main Features of the Meuse/Haute-Marne Site

The Meuse/Haute-Marne site is located on the eastern margin of the Paris Basin, in a stable zone with a simple geological structure. The Paris Basin is a sedimentary basin with flat structural strata. The local seismic activity is very low, and no neotectonic activity has been detected over the entire Meuse/Haute-Marne area. The sediment conditions of the Callovo-Oxfordian formation indicate a geologic environment that has been stable for millions of years. This formation is a homogeneous 130 to 160 m thick argillite stratum, covering a large surface area, with a simple and regular geometry.

The structural framework of the Meuse/Haute-Marne site is well-defined. The newest acquired data have confirmed the geologic structural model as defined in 1994. A very slight fracture density has been identified outside the main regional faults. On the URL site, no vertical fault with throw greater than 2 m has been detected by a 3-D seismic survey over 4 km^2. Four directional boreholes (1,377 m of coring) intercepted only 38 microbreaks, which showed no movement and had no influence on hydraulic properties.

Permeability values obtained from borehole and sample measurements varied from 10^{-12} to 10^{-14} m/s. Indirect permeability assessments result in values over the same range. This very low permeability is explained by the petrofabric of the clay minerals and the pore size (mean value: 20 to 40 nm). The permeability measurements are not influenced by scale effect or anisotropy. This finding is strong confirmation for the homogeneity of the layer and consistency of its properties. It is concluded that diffusion is the dominant transport mechanism, which is confirmed by the distribution of some chemical elements and associated isotopes (e.g., chloride, helium). A comparison of ^{13}C in calcite and interstitial water also indicates that the exchanges have remained very limited

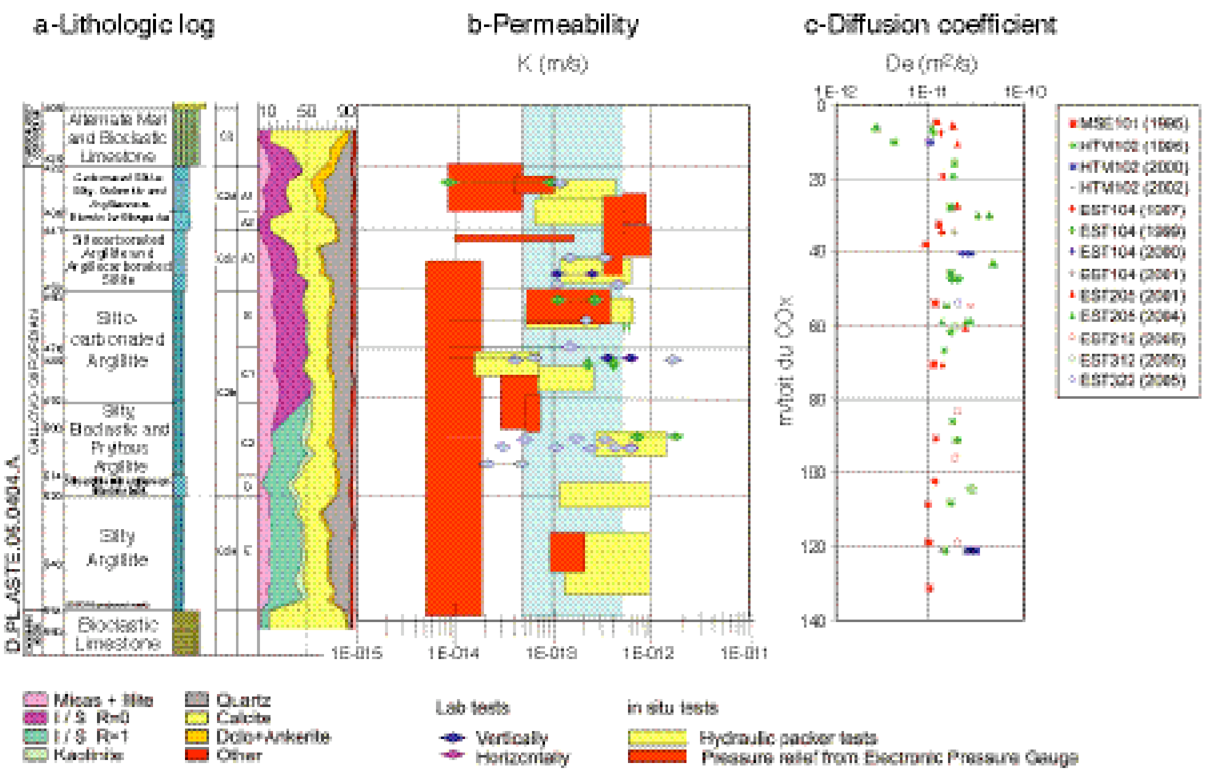

Figure 9.2. Permeability and diffusion coefficient measured within the Callovo-Oxfordian argillites on the Meuse/Haute-Marne site

in the layer for 150 million years (Figure 9.2).

In addition to the very low permeability, the Callovo-Oxfordian provides confinement capabilities through its clay mineral content. This retention capability results in further delaying the migration of elements into the Callovo-Oxfordian; this capability has been tested in a large number of experimental configurations (water chemistry, content of dissolved water).

Experiments on Callovo-Oxfordian argillite samples, and in the Mont Terri laboratory, suggest that the permeability of the EDZ tends to decrease with the creep of the argillites, during their resaturation and stress release, thus causing a gradual closing of the fractures. In the very long term, stresses return to natural equilibrium, closing the fractures so completely that the permeability of the EDZ tends to approach that of the undisturbed argillite. To be on the safe side, Andra has developed a so-called anchoring key concept to limit the flow along drift seals through the EDZ. In addition, a set of criteria has been defined for transposing URL data to a larger area, and a 200 km^2 transposition zone has been identified around the URL.

The hydrogeology of the Meuse/Haute-Marne site is now well known and has been modeled. Because of the low permeabilities of the Oxfordian and Dogger formations, flow velocity is very slow. Hydrogeological modeling of this system is consistent with the results of analyses of chlorine^{36} and carbon^{14} isotopes. The flow velocity of the water is ~1 km per 100,000 years in the Oxfordian formation (and somewhat slower in the Dogger formation).

9.3.3. The Proposed Concept Design

The architecture of the repository presented in the Dossier was chosen for simplicity and robustness, given the current knowledge limits. Construction and operat-

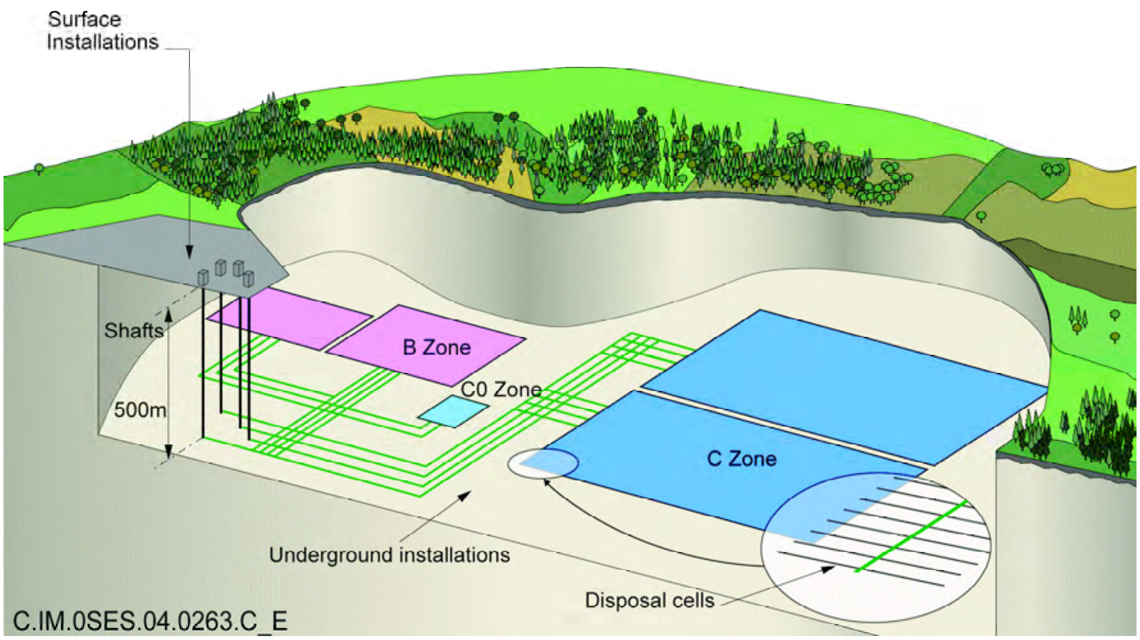

Figure 9.3. General architecture of a potential disposal in the Callovo-Oxfordian formation

ing resources and processes are based on the available technology to be realistic in industrial terms, without aiming at optimization.

The main features of the architecture (Figure 9.3) consist of:

- A single-level layout in the middle of the Callovo-Oxfordian formation
- The separation of the repository into distinct zones depending on the waste nature (B, C, SNF if applicable), to limit interactions and enable management flexibility
- Modularity of the disposal zones, to allow a gradual construction of the repository and to separate modules for safety reasons
- Access through four shafts grouped in the same area and a dead-end architecture of drifts and disposal cells, to avoid a hydraulic short-circuit

In addition to long-term operating safety, the repository design must satisfy the reversibility rationale, tightly linked to the application of the precaution principle, stipulated by the Law of 30 December 1991, and confirmed by the specific request of the French Authorities in 1998 when licensing the URL. Reversibility refers to a "cautious" management in successive phases of a possible repository, leaving the choices open for the coming generations. Adopting this principle supports the possibility of a gradual closure, meaning a gradual reduction in the level of reversibility, towards an increasingly passive configuration. Dossier 2005 describes the mode of action needed to preserve the possibilities of such a choice. Reversibility entails no compromise with regard to the safety objectives, and no additional measures that could significantly affect safety.

For B waste, the disposal packages contain 1–4 primary packages in standardized parallelepiped, stackable concrete containers. The disposal cells are dead-end horizontal tunnels with a pseudo-circular excavated cross section ($\Phi \leq$ 10–12 m). Each cell will be closed by a swelling clay (when the seal material has been chosen).

Each primary C waste package is placed in a sealed unalloyed steel overpack. This overpack has a useful thickness of 55 mm, to ensure its tightness for at least 4,000 years. The disposal cells are dead-end tunnels (Φ 0.7 m, length 40 m). Their spacing must assume a passive dissipation of the canister heat by conduction into the rock and a temperature lower than 90°C. Each cell will be closed (when necessary) by a swelling clay based plug.

SNF packages would be leak-tight unalloyed steel disposal containers. Their leak-tightness is assumed to last 10,000 years. The UOX package has a 110 mm steel thickness and contains four assemblies placed in a cast iron insert. The MOX package is 120 mm thick. There is only one assembly per package. The disposal cells are dead-end tunnels approximately 2.5 m (MOX package) or 3 m (UOX package) in diameter, with a total length of 45 m. An 800 mm thick engineered barrier made of prefabricated swelling clay and sand elements contributes to creating a diffusive environment around the packages. Each cell will be closed by a swelling clay seal.

In support of reversibility, an observation and monitoring program is proposed. Monitoring has the purpose of controlling safety, particularly to protect workers and the environment during the operational phase. Observation aims at monitoring the behavior of the repository, learning about the phenomena occurring and following their evolution, to provide scientific and technical data for the repository's reversible management and for the decision-making process (step-wise approach). Monitoring and observation are closely linked and fulfill the same motivation: increasing confidence in the repository process and control of the repository.

9.3.4. The Performance Assessment of the Repository System

The guideline for performance assessment is the space and time description of the repository and its environment (the concept of phenomenological analysis of repository situations, PARS)–in other words, the history of the repository:

- Analysis data to understand the influence of the various phenomena and identify the key aspects,
- Input data for numerical modeling and simulation of the phenomena and their couplings (representation and simulation tools),
- Support for the safety approach by providing a simple and prudent basic representation of the repository.

The assessment of the geological repository postclosure safety relies on qualitative and quantitative arguments:

- The existence of multiple safety functions, enabling the repository to be maintained in a safe condition even in a degraded situation
- The presence of measures designed to make the repository robust despite uncertainties in knowledge
- The soundness of the scientific basis underpinning the evolution of the repository

A qualitative safety analysis identifies the uncertainties and risks involved in the normal evolution of the repository. It analyzes the effect of these uncertainties and risks on safety functions and the evolution of the repository. It allows investigators to specify the limits of the normal evolution range for sensitivity studies, to describe situations that can diverge from normal evolution, and to check whether they are covered by an altered evolution scenario. This analysis was compared with the approaches employed internationally (FEP 2000 and FEP CAT bases of the OECD/NEA, FEP base of the Nagra).

The scenarios studied comprise one normal evolution scenario and four altered evolution scenarios:

- Container/overpack failure
- Seal failure
- Intrusive borehole drilling
- Severely degraded operation

A safety methodology was set up for examining failure combinations and selecting cautious choices. Nearly 7,000 calculations of safety cases were run, using the Alliances Simulation Platform. If necessary, calculation cases could include unrealistic situations intended to further test the system's robustness ("conventional" and "what if" situations) (Figure 9.4).

9.4. THE RESEARCH PROGRAM ON GRANITIC SITES

To offset not having a URL on a French site, Andra has reinforced its participation in experiments carried out in foreign underground laboratories. At the HRL (in Sweden) the results of the TRUE experiment reinforced our understanding of groundwater flow and solute trans-

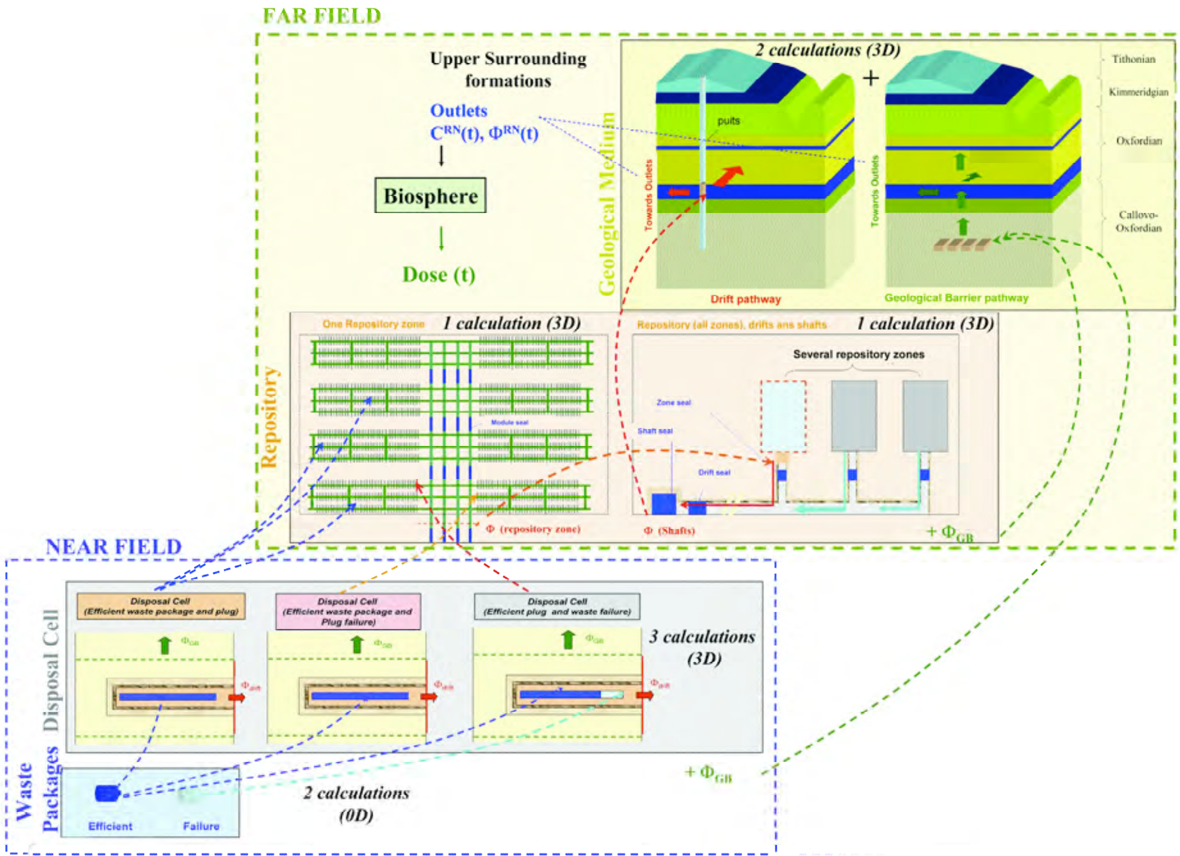

Figure 9.4. Safety calculations architecture (each couple [RN/WastePackage] requires seven 3-D calculations)

port in a fracture network at decametric scale. It offered the opportunity to "benchmark" the numerical codes and models. Andra has also carried out a thermomechanical experiment (TBT) to reproduce the interactions around a vitrified waste cell in a granitic environment. The principle of this experiment is to observe, understand, and model the behavior of deposition hole components under temperatures locally exceeding 100°C. It started in an unsaturated state under thermal transients and ended in a final, saturated state under a stable heat gradient. At the URL (in Canada), Andra had taken part in the Tunnel Sealing Experiment until its dismantlement. This experiment tested, at full scale, the construction method and performance of drift seals in concrete or compacted swelling clay (Figure 9.5).

The phenomenological analysis of repositories in granite has been enriched by transfer-disturbance data from other experiments, particularly those at the Grimsel Test Site (Switzerland) relating to gas migration through fractures (GAM), alkaline plume effects on the rock properties (HPF) and the possibility of radionuclide transfer by colloids (CRR).

Furthermore, Andra and Posiva have tested (in a cooperative framework) different surface reconnaissance methods for outcropping granite on the Olkiluoto site (Finland). Electrical, electromagnetic, and seismic

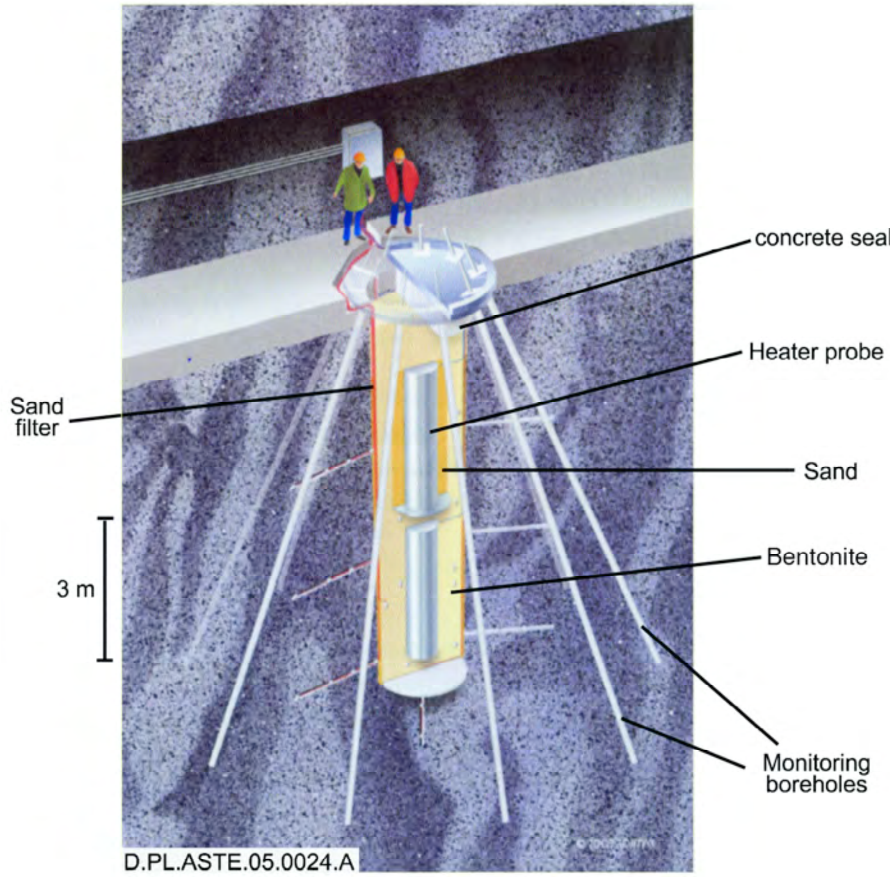

Figure 9.5. Principal design and experimental setup of the Temperature Buffer Test (from Aspö Hard Rock Laboratory Annual Report 2003 TR-04-10)

methods have been compared to detect faults or changes in rock facies. The area investigated has been precisely located for the different steps of this site reconnaissance.

9.5. THE DOSSIER 2005 GRANITE

Dossier 2005 Granite appraises the waste disposal benefits of a granitic site. It identifies and studies the key issues for repository design and performance assessment, to verify that no inhibitory defects exist. It also examines the possible technical options for designing a granitic repository system. Dossier 2005 Granite utilizes the knowledge of granitic rocks and design of disposal concepts achieved in other countries. First, however, the characteristics of French granitic blocks were comprehensively compiled to define ranges of parameter variability for a sensitivity analysis. Based generally on this information, we have generated waste packaging designs, repository architectures, and operation and closure management strategies, taking into consideration basic safety principles (Figure 9.6).

From generic architectures based on the properties of granite, the long-term behavior of different components has been analyzed to understand and model thermal, mechanical, chemical, and hydraulic phenomena interacting within a disposal system. A safety analysis has been carried out in two phases: (1) a qualitative safety

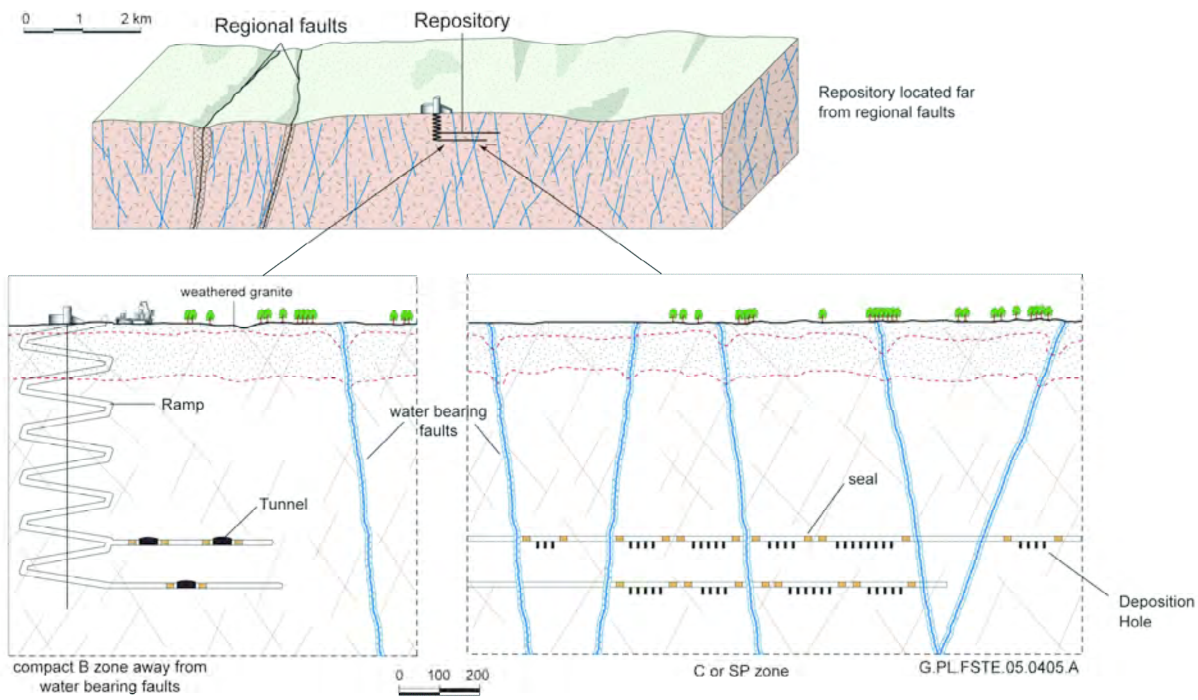

Figure 9.6. Architecture adapted to the fracture pattern within granite blocks

analysis founded on identifying features, events, and processes (FEPs) with regard to the proposed disposal concepts; and (2) a quantitative analysis of three generic geological situations for disposal in granitic rock. This analysis has identified the most important parameters for the performance assessment of such a disposal system and assessed the robustness of the concept options.

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Chapter 10

Current Status of Nuclear Waste Disposal in Germany

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10.1. INTRODUCTION

Germany's recent developments in nuclear waste disposal programs have been characterized by stagnation. On June 11, 2001, the German Federal Government signed an agreement with the energy utility companies to limit the future utilization of the existing nuclear power plants. This agreement is intended to phase out nuclear energy for electricity generation.

Key consequences of the agreement with respect to radioactive waste storage and disposal are as follows:

- Exploration of the Gorleben salt dome as the potential disposal site has been on hold since October 1, 2000.
- Konrad Repository, for all kinds of radioactive waste with negligible heat generation, has been sued at court.
- Interim storage facilities are licensed at 12 reactor sites to minimize transport to the existing central interim storage facilities at Ahaus and Gorleben.

In Germany, radioactive waste disposal is assigned to the Federal Government as a sovereign task. The Federal Ministry for the Environment, Nature Conservation, and Nuclear Safety (BMU) is responsible for nuclear safety and radiation protection. BMU has the authority to issue directions and to supervise the legality and expediency of the acts of those responsible for enforcing the Atomic Energy Act and the Radiation Protection Ordinance.

The Federal Office for Radiation Protection (BfS) is a federal authority in the portfolio of the BMU. BfS implements federal administrative tasks in the field of radiation protection, including radiation protection precaution and nuclear safety, the transport of radioactive substances, and the management of radioactive waste. BfS supports BMU technically and scientifically in these fields. Apart from the BMU, the Federal Ministry of Economics and Labor (BMWA) is responsible for R&D in nuclear waste disposal.

The Federal Institute for Geosciences and Natural Resources (BGR), as chief scientific-technical federal authority in the portfolio of BMWA, advises the German Federal Government in all geoscientific matters. BGR is responsible for the development and performance of the geoscientific investigation programs at the potential repository sites. The Company for the Construction and Operation of Waste Repositories (DBE) acts on behalf of BfS and is assigned to plan, design, construct, and operate the repository.

10.2. RADIOACTIVE WASTE

In Germany, most of the radioactive waste is produced using nuclear energy for electricity generation. Following the phase-out policy of the German government and the 9th amendment of the Atomic Act of 27 April 2002, a limit was defined for the amount of electricity to be generated by nuclear power. As a consequence, the overall volume of radioactive waste generated in Germany is now limited as well. This concerns waste from spent nuclear fuel (SNF) elements reprocessed in France and the UK, the SNF elements themselves, and waste from the operation and decom-

Table 10.1. Radioactive waste inventory up to the year 2040 (in cubic meters)

	Inventory end of 2000	Prognosis 2001-2010	Prognosis 2011-2020	Prognosis 2021-2030	Prognosis 2031-2040	**Total**
Waste with negligible heat generation	76,000	58,000	54,000	76,000	33,000	**297,000**
Heat-generating waste	8,400	9,200	5,700	700	~27	**24,000**

missioning of nuclear power plants. To a lesser extent, radioactive waste, mainly low- and medium-level waste, also arises from research, medical, and industrial applications.

In addition to the classifications of high active waste (HAW), medium active waste (MAW) and low active waste (LAW), we distinguish (in Germany) between heat-generating waste and waste with negligible heat generation. This differentiation was developed with regard to waste disposal in different host-rock formations at different sites. The total amounts of the two waste categories accumulated in Germany projected up to the year 2040 are shown in Table 10.1 (AkEnd, 2002).

10.3. NATIONAL REPOSITORY PROJECTS

10.3.1. The Konrad Mine

The Konrad Mine is an abandoned iron ore mine in the state of Lower Saxony in the northern part of Germany. The iron ore formation at a depth between 800 m and 1,300 m is considered to be the host rock for emplacement of radioactive waste, with negligible heat generation. The great depth and the hydraulic properties of the overburden are the most important advantages of the geological setting at Konrad. It is effectively sealed off from near-surface groundwater by the overburden of several-hundred-meters-thick clay stone and marly formations. The host rock offers only limited availability of water as a transportation medium for radionuclides into the biosphere.

In 1982, an application for initiation of a plan-approval procedure for the Konrad Repository Project was submitted by the PTB (Federal Institute for Physics and Metrology), the legal predecessor of BfS. The licensing documents, the so-called Plan Konrad, were submitted in 1990. According to the "agreement" of June 11, 2001, between the Federal Government and the power utilities, the plan-approval procedure for the Konrad Repository was completed and a resolution adopted on June 5, 2002. The plan-approval decision for the Konrad Repository provides permission for a radioactive waste package volume (with negligible heat generation), limited to approximately 300,000 m^3, to be emplaced in the Konrad Mine. However, operation cannot commence until the court has reached a decision on the objections that have been filed.

10.3.2. The Morsleben Repository

In the former German Democratic Republic (GDR), the abandoned Morsleben Salt Mine was selected as the site of an underground repository for low- and intermediate-level waste with low concentrations of α-emitting radionuclides. The repository is located in a salt structure, formed in the geologi-cal formation of Zechstein/Thuringium (upper Permian). The salt structure consists of folded rock-salt, potash layers, and imbedded anhydrite blocks. The overall thickness of the salt formation is between 350 and 550 m.

The emplacement of low- and intermediate-level waste continued after the reunification of the two German states. Operation of the Morsleben Repository was discontinued by 1998, and with an amendment of the German Atomic Energy Act, all permission to dispose of radioactive waste in the Morsleben Mine became ineffective in 2002. The plan-approval procedure is now limited to backfilling and sealing the repository.

10.3.3. The Gorleben Repository Project

Since 1979, the Gorleben Salt Dome in the state of Lower Saxony has been investigated as the potential repository site for all types of radioactive waste.

Gorleben was selected in 1977 on the basis of geoscientific and economic criteria. Investigation of the Gorleben site from the surface was started in 1979. The exploration program consisted primarily of the following aspects: stratigraphical investigations, hydrogeological investigations, geophysical measurements, seismological surveys, boreholes to the salt leaching surface, deep boreholes into the salt dome, and exploration boreholes for the shafts. After the reunification of the two German states, the investigations were expanded to an area of the former GDR north of the Elbe River. The site investigations carried out from the surface were completed in 1997.

For the underground site investigation and the characterization of the potential host rock, BGR and BfS developed a comprehensive geological exploration program, including boreholes and the application of nondestructive seismic methods. The underground exploration started in 1986 with the sinking of two shafts, followed by the construction of the underground infrastructural area.

In addition to the geological exploration, *in situ* experiments have been performed for the geotechnical characterization of the potential host rock. Results from the *in situ* and laboratory tests provided the database to identify potential areas for disposal. The total results of the geological and geotechnical investigations gained so far show a large volume of potentially suitable host rock in a simple anticline structure of the salt dome. To date, there is no indication that the Gorleben Salt Dome is unsuitable for a repository (Bornemann et al., 2004, in prep.).

Following the agreement between the German Federal Government and the utility companies, exploration of the Gorleben Salt Dome was discontinued in 2000 for at least three and at most ten years, in order to clarify conceptual and safety questions. Thirteen topics have been identified in total, and they should have been processed by BfS by the end of 2004. To date, however, no final results from the investigations have been published.

10.4. CURRENT STATUS OF THE GERMAN WASTE MANAGEMENT POLICY

As part of the phasing-out policy, the new German disposal concept is governed by the following fundamental principles (Nies, 2004):

- Safety is the first priority in the field of radioactive waste management.
- Disposal is the only option that can provide for a sustainable solution to the waste problem.
- Radioactive waste of German origin shall be disposed of in Germany.
- Installation of a disposal facility shall be promoted as soon as possible and must not be postponed or left to future generations.

As boundary conditions for the development of the new waste disposal policy, the Federal Ministry for the Environment specified the following political objectives:

- All radioactive wastes shall be disposed of in deep geological formations in Germany.
- For the final disposal of all types and quantities of radioactive wastes, one repository is sufficient, to be operable in 2030.
- Further repository sites in different host rocks shall be explored; a decision on the repository's site shall be based upon a comparison of alternatives.

To examine further sites in different host rock formations, the Federal Ministry for the Environment had appointed a group of experts (AkEnd) that would develop a new comprehensive procedure for selecting a repository site in Germany. The procedure should be both suitable for safe disposal and acceptable to the public. It was not the task of the AkEnd to implement the procedure, to apply the procedure or the criteria respectively to the selection or judgement on the suitability of the Gorleben site, or to choose or assess other sites.

The AkEnd Committee, as a technical scientific body working independently and free of directives within the framework of the established objectives, started its work in 1999; the final recommendations were published at the end of 2002 (AkEnd, 2002). Following a decision by the German Parliament, at least two sites shall be selected by 2010 for underground exploration, and the repository shall be operable by the year 2030.

According to the recommendations of the AkEnd, political and social agreement on the site selection procedure should take place in a second phase. The Federal

Figure 10.1. BAMBUS II experiment: Dismantling operation of the emplaced dummy canister in the Asse Research Mine (Bechthold et al., 2004)

Ministry for the Environment, however, did not carry out its study of the AkEnd proposal, because the opposing parties in the German Federal Parliament, as well as the energy supply concerns, did not accept an invitation for meetings with a negotiating group in 2003.
In addition to the implementation of the site selection procedure, and in accord with the coalition agreement between the Social Democratic and the Green party, the Federal Government is currently striving for an agreement on financing site exploration. The industry, however, has already declared that they consider further site selection activity unnecessary and therefore are not willing to pay for it (Nies, 2004).

10.5. GERMAN R&D ACTIVITIES FOR NUCLEAR WASTE DISPOSAL

Recent German R&D activities for nuclear waste disposal have been generally determined by the decision of the German government to enhance research activities in clay and granite. This decision, in 2000, required setting new priorities for many current research projects and initiating new projects. Consequently, the importance of international cooperation in joint projects, and particularly the German participation in European underground research laboratories, has increased significantly.

10.5.1. Research in Rock Salt

Since 2000, research concerning nuclear waste disposal in rock salt has been drastically reduced. During the last few years, such activities concentrated on dismantling the Thermal Simulation of Drift Emplacement (TSS) field in the ASSE mine and analyzing the results (BAMBUS II) for topics such as: the compaction behavior and sealing characteristics of crushed salt as backfill, laboratory studies on evolution and long-term behavior of the Excavation Damaged Zone (EDZ), and demonstration experiments on geotechnical barriers.

The BAMBUS II project (Backfill and Material Behaviour in Underground Salt Repositories), undertaken as an international joint project from 2000 to 2003 (Bechthold et al., 2004), was funded by the European Commission as well as national governments and authorities (Figure 10.1). The principal scientific objective of the project was to extend the basis for optimizing the repository design and construction, and for predicting the long-term performance of the barriers in a repository for radioactive waste in rock salt. The work was divided into *in situ* studies, laboratory investigations, modeling studies, and office studies. For use in future repository design and construction, an easily accessible data acquisition system has been developed. The BAMBUS II project was the last *in situ* research activity in the ASSE mine. As a result of the backfilling operations, the ASSE mine cannot be used any further as an underground research laboratory in rock salt.

Crushed salt is primarily intended for use as backfill in rock salt. The sealing capability of crushed salt largely depends on the degree of compaction and the resulting pore volume. Different institutes in Germany perform laboratory studies to evaluate the influence of temperature, moisture, and mineral additives on the performance of crushed salt. Mineral additives are assumed to accelerate compaction. Precompacted crushed salt (bricks) are also tested as backfill material. Some of this research is performed as part of an EC-funded Integrated Project NF-Pro that started in 2004 and will last 4 years.

The long-term evolution of rock salt is also one of the objectives of the NF-Pro project. Laboratory studies are performed to analyze the failure and healing behavior of rock salt. The state of stress and temperature determine whether the salt develops dilatancy or is compacted. In contrast to other host rocks, rock salt is assumed to undergo real healing under compaction. Constitutive laws have been developed to model the long-term behavior of the EDZ.

Shaft, gallery, and borehole seals, at real size, are being tested in operating salt mines. These tests are performed as part of the research concept "Underground Disposal of Chemo-toxic Waste." Also, as part of the EC-funded Integrated Project "Engineering Studies and Demonstrations of Repository Designs (construction-operation-closure)" (ESDRED), emplacement techniques are being tested.

10.5.2. Research in Granite and Clay

The current research on nuclear waste disposal in Germany focuses mainly on clay and granite as host rocks, and on clay as backfill and sealing material in salt, clay, and granite. The research is mainly funded by the German government and the EC. Since no underground research laboratories in granite and clay exist in Germany, *in situ* experiments are conducted, as part of an international cooperation effort, at underground laboratories in Sweden (Äspö), Switzerland (Grimsel, Mont Terri), France (Tournemire, Meuse/Haute Marne), and Belgium (Mol). The work described below refers only to experiments with contributions by partners from Germany.

German Participation in URLs

At the Mont Terri Rock Laboratory (Switzerland), many aspects of nuclear waste disposal in clay stone are covered. In 2004, Germany participated in 15 experiments. Two key issues are considered in these experiments:

1. Development of the EDZ and performance of backfill and buffer
2. Characterization of the Opalinus Clay.

Development of the EDZ and performance of backfill was investigated in nine different experiments. The "Engineered Barrier Experiment" (EC-funded), for example, aims at demonstrating a new concept for construction of HLW repositories in horizontal drifts, in competent clay formations (Figure 10.2). A "Heater Experiment" (EC-funded) was performed to identify and measure the THM response in both the host rock and the buffer, with special emphasis on the host rock/buffer interaction. The effect of repeated desaturation and resaturation is examined in the "Ventilation Experiment," funded since 2004 by the EC as part of the Integrated Project "NF-Pro." The performance of "self sealing barriers of clay-sand mixtures" is tested in boreholes as part of the Integrated "ESDRED" Project.

The characterization of the Opalinus Clay included *in situ* gas-permeability measurements, *in situ* stress measurements, the determination of pore-water chemistry, the equilibrium of gas and pore water, and microbial studies. Results have been integrated into a hydrogeological analysis (Marschall et al., 2003) and a rock-mechanics analysis report (Bock, 2001).

The results from the Mont Terri Rock laboratory are now being applied in several collaborations with Andra at the Meuse/Haute Marne Underground Laboratory in France. Several high-resolution seismic borehole measurements have been performed to investigate the extent of the EDZ around the main shaft and in a niche in the Callovo-Oxfordian clay stone. Gas-permeability measurements have been conducted in the EDZ and the intact rock. In 2005, additional measurements to characterize the EDZ by seismic and geoelectric methods—to determine the gas permeability, pore pressures and *in situ* stress field—are planned.

In the Mol Underground Laboratory (Belgium), measurements to characterize the extent of the EDZ by seismic methods have been performed. As part of an experiment concerning the "Corrosion of Active Glass under *In Situ* Conditions", gas release and permeability have been measured. Gas release from different backfill materials and the Boom Clay resulted in thermal desorption, oxidation of organic compounds, thermal decomposition of minor constituents of the clay, and corrosion of the container material.

German partners have participated in the Full-Scale Engineered Barriers Experiments (Febex I and Febex II) at the Grimsel test site (Switzerland). Two heaters with lengths of 4.54 m each were emplaced in a gallery. The residual volume in the gallery was backfilled with compacted bentonite, and the entrance to the gallery was sealed with a concrete plug 2.7 m long. The diameter of the gallery is 2.27 m, and the diameter of each heater is

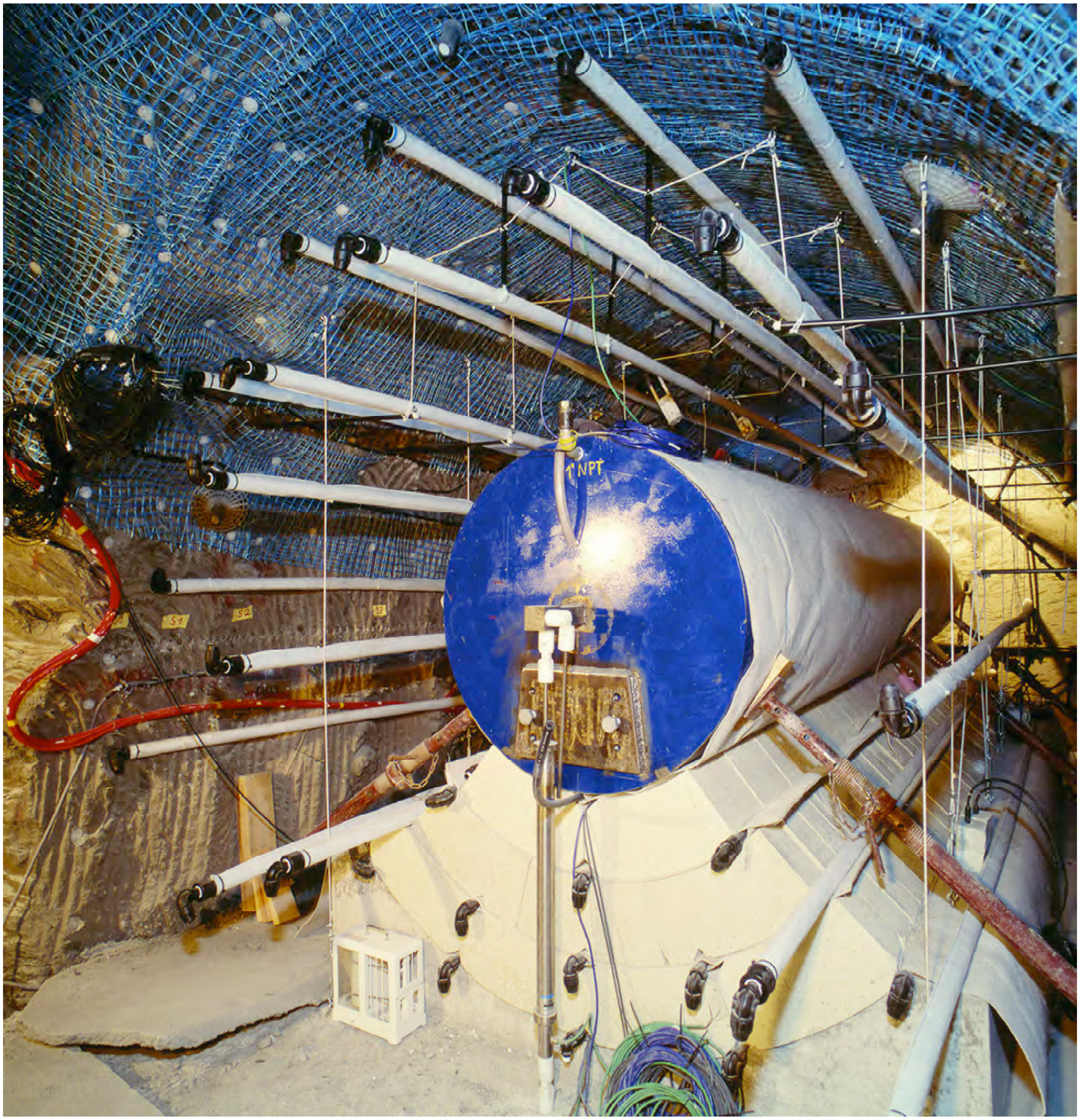

Figure 10.2. Engineered barrier experiment in the Mont Terri Rock Laboratory: emplaced dummy canister on compacted bentonite bricks

0.90 m. Heaters consist of carbon steel, each having a thermal power output of 1,200 W, which leads to a maximum surface temperature of 100°C. German partners measured the gas generation and the gas release in the test field, determined gas permeability in the EDZ after dismantling, and performed THM calculations for the whole system.

In the Swedish underground research laboratory at Äspö, a full-scale replica of the deep repository planned for the disposal of spent nuclear fuel has been constructed. The prototype includes six deposition holes in which full-size canisters with electrical heaters were placed and surrounded by bentonite. The deposition tunnel had been backfilled with a mixture of bentonite and crushed granitic rock. As part of the research activities in the prototype repository at Äspö, electrical resistivity was measured to monitor water uptake in the backfilled drift and boreholes, and desaturation effects around the deposition boreholes were also studied. Results were used as input parameters for thermal-hydrological-mechanical chemical (THMC) modeling of the entire system.

Laboratory Tests

Laboratory research was performed on clay to determine its THM behavior. Particularly as part of the NF-Pro project, the development of damage and the capability to reseal is under research in several laboratories. The development of damage is strongly influenced by the degree of saturation, material anisotropy, and the stress regime.

The physicochemical behavior of clay-based engineered barriers under the influence of a hyper-alkaline plume due to cement degradation has also been investigated. In the case of solution intrusion into the repository, the cemented wastes show a significant change in their structure, which leads to a change in the solution composition and brine pH. The resulting solution can cause changes in the composition of the clay material.

Modeling

Modeling a repository for radioactive waste that is close to reality requires the consideration of thermal (T), hydraulic (H), mechanical (M), and chemical (C) coupled processes. German working groups are involved in the evaluation of constitutive laws to model the THM behavior of clays as host rock and backfill material. As an example, a model to calculate the resaturation and to evaluate the sealing ability of bentonite buffer in final repositories for HLW in crystalline rock is to be developed as part of the prototype repository experiment. The model should include all the relevant physical processes of thermo-mechanically influenced two-phase flow. The approach used in the model, as well as the accompanying laboratory investigations, are to be discussed and modified, if necessary, by the participants of the prototype repository modeling group.

BGR joined the international project DECOVALEX III (Development of Coupled Models and their Validation against Experiments in Nuclear Waste Isolation). This project offers a forum for the improvement of knowledge and modeling of coupled processes. In the course of DECOVALEX, the numerical code RockFlow has been developed as a fully coupled THM code. Currently, the extension to coupling with chemical processes is under development.

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Chapter 11

Current Activities and Future Plans for Geological Disposal of Radioactive Waste in Hungary

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11.1. INTRODUCTION

In Hungary, nuclear power provides a substantial portion of the total electricity produced in the country. The main source of nuclear power, the Paks Nuclear Power Plant (NPP), is comprised of four VVER-440 nuclear reactors, each with a capacity of 460 MW_e. Over the plant's projected life of 30 years, the plant will produce about 13,000 m^3 of low and intermediate level waste (LILW), and nearly 17,000 m^3 will likely be produced by decommissioning the plant. This means that about 30,000 m^3 of LILW from the NPP must be disposed of. Since expansion of the existing near-surface repository such that it could completely accommodate the waste produced by the NPP is impossible, a national program was launched in early 1993 (after some previous unsuccessful attempts) to find a solution for disposal of LILW generated by the NPP.

The number of spent nuclear fuel (SNF) assemblies to be generated during the 30 years' operation is around 13,500. A Hungarian-Soviet Inter-Governmental Agreement on cooperation in constructing and operating the Paks NPP was signed in 1966, and an Additional Protocol was added to it in 1994. In these agreements, which are still in force, Russian authorities agreed to accept delivery of the SNF, while Hungarian authorities agreed to purchase any required new fuel assemblies exclusively from Russia for the entire lifetime of the NPP. According to this Agreement, Hungary was not required to take back radioactive waste or other residuals from the reprocessing of such fuel. Altogether, 2,331 assemblies were shipped back to the Soviet Union (later Russia) between 1989 and 1998. However, in the 1990s, contrary to the terms of the original agreement (but in accordance with international practice), the responsible Russian authorities wished to have Hungary take back the residual radioactive waste and other byproducts created during reprocessing.

At present, Hungary does not have the capability to dispose of high-level long-lived radioactive waste. It was for this reason that the construction, licensing, and operation of an interim SNF storage were classified as a top priority in 1993. In 1997, a Modular Vault Dry Storage facility was commissioned in the immediate vicinity of the NPP. This facility, for the interim storage of SNF, allows for the storage of assemblies for a period of 50 years. Also, it is envisaged that some 100 m^3 of high-level waste will be generated during the operation of the Paks NPP, which is temporarily stored in tube pits designed for this purpose at the NPP, and it is estimated that some 3,700 m^3 of high-level (HLW) and long-lived waste (LL) will be produced during the decommissioning of the plant. Decommissioning of the other two nuclear facilities in Hungary (a training reactor and a research reactor) will also produce some radioactive waste, but to a much smaller extent. These HLW can be disposed of along with the similar waste from the nuclear power plant.

As yet, there is no decision on the back-end of the fuel cycle, but to calculate the future costs of radioactive waste and SNF management, as well as to assure the necessary funding, some assumptions have to be made about disposal. For our purposes here, as a reference scenario, we have postulated the direct disposal of SNF assemblies within Hungary.

11.2. SITING OF A NEW LILW REPOSITORY

In 1993, the Hungarian government launched a national program aimed at selecting a site for disposal of LILW arising from the operation and decommissioning of the Paks NPP. According to the principles that we established, alternative solutions had to be examined, both in terms of location and the mode of disposal. Thus, both near-surface and mined underground repositories down to 300 m depth were considered.

In the surveys of the entire country carried out between 1993 and 1995, about 300 geological formations were identified as being potentially suitable for either near-surface or underground disposal facilities. In the initial phase of site exploration, using exclusion criteria, all areas were ruled out that had to be protected out of political, economic, geological, etc. considerations, or where the disposal site needed to be protected from industrial or natural influences. The next phase was “positive” screening, in which geological prospects were evaluated from the perspective of suitability, meaning the quality of the geological barrier. As a result, about 6,000 km^2 out of the 93,000 km^2 area of Hungary were considered worthy of more research related to near-surface disposal, and about 23,000 km^2 related to subsurface disposal. A suitable site could be expected in areas where the number and density of potential prospects proved to be high. Using this approach, an area of 5,000 km^2 was selected for further exploration. Numerous potential locations were identified—128 for near-surface and 193 for subsurface disposal.

At this stage, another very important issue arose, namely the opinion of the population in the areas under consideration. Public approval was given to just a few dozen out of the potential areas. Of these, four prospective areas (three for near-surface and one for underground disposal) were investigated by field reconnaissance. Boreholes were drilled at two near-surface (loess) sites and one underground (granitic) site. Judging from this comparison, the granite site proved to be more suitable. Based on the first series of investigations, a granite formation in the village of Bátaapáti (in the Üveghuta area) in southwestern Hungary was selected as a potential site for an underground repository. One of the potential near-surface sites was selected as an alternative solution for further investigations, should the investigations in Bátaapáti not meet expectations (Ormai et al., 2001).

In 1997–1999, six boreholes, 300–500 m deep, were drilled at the Üveghuta site (see Figure 11.1). The geological and geophysical investigation of these boreholes resulted in a precise definition of the site.

The preliminary safety assessment did not give rise to any doubts concerning the site’s suitability. Radioactivity concentrations of isotopes calculated for the vicinity of the disposal areas did not significantly exceed the concentrations existing in the natural environment. Results from the preliminary safety assessment of the subsurface disposal facility illustrate that

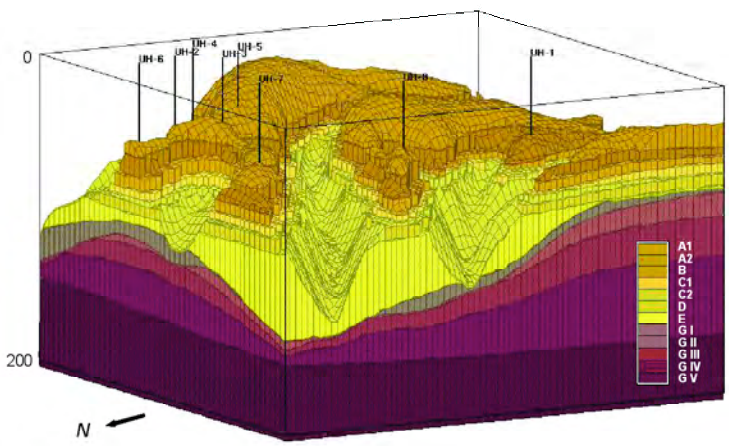

Figure 11.1. Deep boreholes at Üveghuta region

radiological risk to the public is negligible for the post-closure phase (doses to the public are several orders of magnitude lower than the authorized limits for every case considered). This statement is valid for both the normal and altered evolution scenarios. By virtue of the deep location and hydrogeological conditions at the site, the proposed concept of subsurface disposal is not affected significantly by changes in the environment.

In May 1999, because some Hungarian experts expressed reservations concerning the adequacy of the site investigations, the Hungarian Atomic Energy Authority asked the International Atomic Energy

Agency to organize an international peer review of the research on site selection and site suitability for Hungarian LILW disposal. This was undertaken as part of the Agency's Waste Management Assessment and Technical Review Programme (WATRP).

The WATRP team concluded that the process leading to site selection appeared reasonable, and that it had appropriately considered the aspects of both geology and public acceptance. The site appears potentially suitable for disposal of LILW from NPP operational and decommissioning wastes; however, site characterization and repository design should continue. Based on a meeting with local representatives, the team found that an effective and open communication program appears to have been established.

During the 2002–2003 exploration, 23 boreholes were drilled. Eight of them were to a depth between 300–411 m; the rest were less than 101 m deep. Two exploratory trenches (1,490.3 m) were excavated, three dug wells (73.3 m) were deepened, and nine structures for overflow measurements were constructed. Geophysical measurements were performed in the boreholes; geological and tectonic logs and maps were also compiled for the boreholes and trenches, respectively, as well as geological logs for the dug wells. These trenches and dug wells were incorporated into the hydrogeological system. To complement this major ground-based geophysical survey, a wide spectrum of analyses was carried out in various laboratories.

By any standard, the granite of the Üveghuta site can be considered as suitable host rock for the LILW repository: its volume is large enough, the homogeneity is acceptable, the hydraulic conductivity is low, and the infrequent radioactive emissions will be immobilized by the montmorillonite clay in the fissure fillings. Downward-directed flow is deflected laterally by a geological barrier above the repository area, and the freshwater heads in the repository area decrease with depth. The flow system will retard the migration of the infrequently emitted radioactive pollution towards the ground surface and will dilute its concentration.

Perhaps the most important lesson from the exploration of the Üveghuta site is the ambiguity of the relationships between fault tectonics and hydrogeology. Because of argillaceous alteration and filling, the hydraulic conductivity of the major fracture zones is surprisingly low, 5.8×10^{-9} m/s on average, which exceeds the value for the background granite by only one order of magnitude. The biggest influx came from a relatively insignificant fissure, whereas the usual influxes from thick fracture zones were largely the same as those from small fractures (Balla, 2004). A quantitative assessment of these phenomena has not yet been performed, but the preliminary results are promising. On the basis of the integrated interpretation of tectonic and hydrodynamic data, a structural model of the site has been constructed, and it can serve as the basis for further evaluation.

The final report was approved by the Hungarian Geological Survey. In the resolution based on this report, it was confirmed that the site is suitable for an LILW repository, but that determination of the rock volume needed for a waste disposal facility and its protection zone required further underground explorations (see Figures 11.2 and 11.3). Preparations for this stage started in late 2004.

Based on the available investigation results, the repository would be constructed at the outskirts of the village of Bátaapáti, at a depth of 200–250 m below the surface, at 0–50 m above sea level, in granite of Lower Carboniferous age. The exact location of the disposal area will be defined after additional geological investigations and experience gained during the mining exploration. Layout of the subsurface facility is affected by the geological environment and by the amount of waste. At present, a tunnel-type arrangement seems favorable. Both the waste drums and the disposal containers would be placed in the disposal areas, so that any radioactive isotopes escaping from the waste packages over long times would be sorbed by the clay backfill material, either near the waste packages or inside the containers. Thus, the probability of a significant release of radioactivity would be very low, even after several hundred years. It is assumed that the backfill would limit groundwater access to the waste packages. Granite pillars of 10–20 m thickness would support and separate 6 or 10 m wide disposal galleries, ensuring the mechanical stability of the repository. Design of the layout and the characteristics of the disposal areas will need to be refined after further geological investigations.

To initiate activities for constructing the repository, the Parliament's preliminary approval in principle is required. Based on the current plan, the repository could be operational in 2008.

Figure 11.2. Preparations for the underground explorations

Figure 11.3. Excavation of the double incline

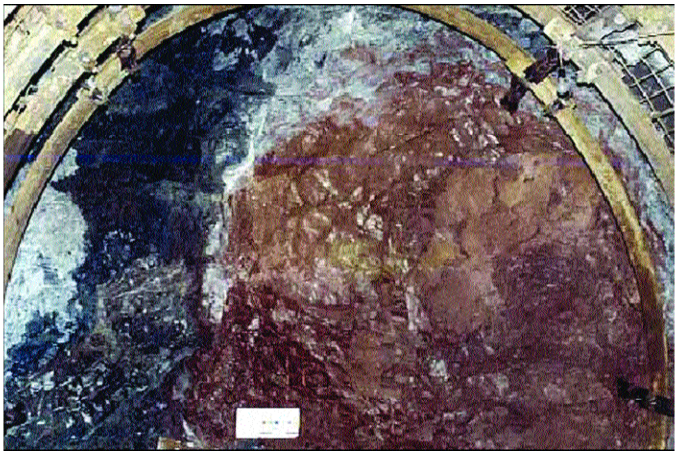

Figure 11.4. The very first picture of the BCF, in an underground opening (1994)

11.3. PREPARATIONS FOR CLOSURE OF THE FUEL CYCLE

Hungary has chosen to delay making a final decision on the option of closure of the nuclear fuel cycle—a position adopted by the Hungarian Atomic Energy Commission in 1993 and confirmed in 1998. The facility for interim storage of SNF allows for the storage of fuel assemblies for a period of 50 years at the site. Hence, it is not necessary to reach an immediate decision on disposal of SNF. This delay offers the possibility of monitoring the disposal experience of other countries, particularly the results of international research on transformation of HLW. However, it is expedient to develop a policy and strategy, as well as a working program, to support R&D activities within Hungary. In this way, we will be able to evaluate and apply international results, in accordance with our own perspective, and reach a decision regarding a solution to the closure of the nuclear fuel cycle.

From the very beginning, it was obvious that all the problems associated with the management of HLW would have to be solved by Hungary on its own, irrespective of whatever solution might be found for the issue of the fuel cycle back-end. This is the reason that preparations for disposal of high-level and long-lived radioactive wastes started in 1993.

The exploration tunnel excavated in the Mecsek Uranium Mine reached the claystone formation in 1994, and the on-site underground data acquisition began in this area, at a depth of 1,100 m (accessible from the former uranium mine). The possibility of implementing *in situ* examinations at this depth is very rare, even worldwide. Its importance is given by the fact that the higher temperatures (50°C), rock stresses (s1»30 MPa), and water pressures (p»9–10 MPa) resulting from the substantial depth ensure the investigation and better understanding of several special effects determining the isolating function of the geological barrier (Short-Term Programme, 1998).

Between 1995 and 1998, a short-term program was launched to characterize the rock mass known as the Boda Claystone Formation (BCF) (Figure 11.4). The program was limited to three years, because of the inevitability of the closure of the mine, planned for 1998: the existing infrastructure of the mine could be economically utilized only during this time period.

Confining performance and long-term stability of the BCF can be summarized as follows: The recent 700-1,000 m thick layers of BCF were settled in an alkaline (or playa) basin under extreme climatic inflow and geochemical conditions, and later they were buried to at least 3.5 to 4.5 km depth. The diagenesis of sediments occurred at high temperature (~ 150–200 °C) and at high pressure (120–150 MPa). This situation resulted in the present overconsolidated, highly indurated character of the BCF. The bulk-porosity and hydraulic conductivity of the intact rock-matrix are very low (0.6–1.4%; 10–15 m/s) (see Figure 11.5). The typical interval for the Young-modulus is between 30-40 Gpa, and the aver-

Figure 11.5. Implementation of a hydraulic test at the borehole

age unconfined strength exceeds 100 MPa. Composition of the 35–50% clay-mineral content also fits into the burial history. According to the mineralogical tests, the dominant clay mineral in unweathered rock types of BCF is illite (25–40%). Chlorite content is 5–15%, and smectite and a chlorite-smectite mixed layer were also detected in every sample. However, with the exception of some special places, the proportion of swelling clay minerals is near the detectable limit (>1%). Thus, self-healing caused by swelling may theoretically occur, but it does not seem to be important at first sight.

According to the results of the 7-year investigations extending over the most significant tectonic zone of the area (and applying *in situ* methods), no evidence could be found that would disqualify this potential host formation. Based on the results of the final report, we recommended constructing an underground research laboratory (URL) for further research and to assess the suitability of the rock.

In 1999, however, a decision was made on final closure of the uranium mine that led to termination of the program. With the rejection by the government (in 1999) of PURAM's plan for an underground research laboratory at the Boda claystone site, as a step towards the development of a deep repository, Hungary was left with no practical plan for disposal of high level and other long-lived waste. In this situation, learning the lessons from the former procedure, a new project was launched in 2000 aiming at developing a new policy.

Hungary has to decide how to manage this waste in the long term. Implementing that decision will take decades. With this in mind, PURAM has contracted ENRESA of Spain for consulting services in the elaboration of a strategy for disposal of high level and/or long-lived radioactive wastes and the management of SNF. By the end of October 2001, the document was completed. This paper explains what the problems are and what decisions have to be taken describes several options, and sets out proposals for reaching decisions.

Based on this study, elaboration of the long-term strategy is to be started soon. A national strategy will be identified upon comparative assessment of several options. The nondelayed open cycle counting on a national repository facility is perhaps the most important among the potential scenarios assessed.

According to the presently preferred scenario, to ensure the disposal of HLW, the construction of a repository in a deep geological formation within Hungary shall be undertaken. In accordance with current international opinion, such a repository could also be used for direct disposal of spent nuclear fuels, and even more, for the accommodation of wastes from reprocessing of SNF assemblies. The reference scenario envisages a back-end fuel cycle completed with the direct disposal of SNF.

11.3.1. Restart of HLW Siting

In Hungary, the efforts made for disposal of HLW/SNF have been revived during the past few years. The geological explorations that were restarted in the West-Mecsek region were aimed at the identification of a site for the construction of a URL and, at the same time, the thinking over the elaboration of a strategy has strengthened again. The revival of the national programs has a close relationship with the accession of Hungary to the European Union (EU), since, as a member state, Hungary is also subject to the regulations and directives of the EU.

In 2000, based on desk studies, a nationwide screening (Figure 11.6) was carried out by evaluating the potential rock formations in detail. Thirty-two lithological formations potentially suitable for a deep geological repository, within the territory of Hungary, were identified. This comprehensive investigation confirmed that the BSF has the leading position among the potentially suitable sites for an HLW repository.

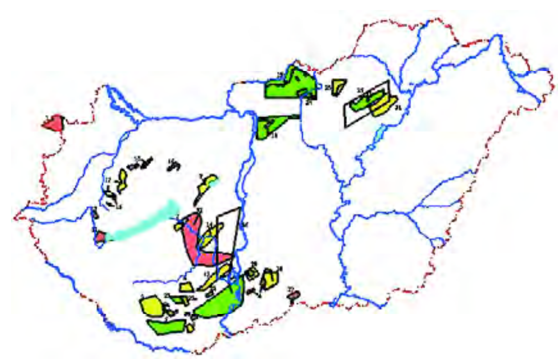

Figure 11.6. The results of nationwide screening

In 2004, in parallel with policy development, the exploration program for a HLW repository restarted at Boda (Kovács, ed., 2004). Because the access to the host rock from the uranium mine is no longer available, the investigations should be carried out from the surface. The aim is to designate, up to 2008, a location for an underground research laboratory where exploration of rock could be accomplished.

The development and approval of a strategy is not expected before 2008. If geologic disposal were selected as a preferred option, then international experience indicates the execution of the research preparatory activities will require some 20–25 years of work. An additional 10 –15 years for licensing and construction of the disposal facility can be envisaged.

The time schedule prepared for the disposal of HLW and the management of SF is presented in Table 11.1 below:

In September 2003, TS Enercon Kft prepared a site-independent conceptual design entitled "Preparation of Conceptual Design and Cost Estimation for the Construction of a Deep Geological Repository for the Final Disposal of Long-lived High Level Radioactive Wastes and Spent Nuclear Fuels of Hungary." This Conceptual Design proposes the construction of a facility within the Boda Claystone Formation, a potential host formation, in the West-Mecsek region, at an appropriate depth and location to be specified by subsequent exploratory phases (TS Enercon Kft, 2003). The study was based on the information provided by SKB regarding the Swedish geological repository for high-level radioactive waste (SKB, 2004).
The Conceptual Design describes the scope of engineering work to be performed in relation to the open fuel cycle activities. This work includes the building of a repository facility; consideration of waste management, transport, and storage activities; and a cost estimate for these activities. The Conceptual Design has a twofold objective:

1. To present the engineering work to be performed for the open cycle without delay—as a strategy—assuming a national repository facility
2. To perform a cost exercise for the implementation of the strategy, including the itemized comparison of each disposal scenario

In accordance with the Conceptual Design, the deep geological repository consists of a drift system constructed at an appropriate depth, along with associated surface facilities (see Figure 11.7). The surface and underground construction are interconnected by two central and one diagonally positioned vertical shafts, which provide the construction, exploration, and operating conditions in accordance with safety requirements.

Table 11.1. Time schedule prepared for the disposal of HLW and management of SF

2005–2008	Start of R&D work. • Surface exploration of the BCF region for the construction of a URL. • Preparation of a Preliminary Environmental Impact Study Report. • Finalization and approval of the HLW management strategy.
2009-2012	• Start of construction of the URL. • Elaboration of a research/exploration program.
2013-2032	• Construction of the URL. • Implementation of the research/exploration program. • Completion of safety assessments.
2033-2046	Construction of the repository.
2047-2069	• First phase operation of the HLW repository. • Transfer of spent fuel assemblies from the ISFS to the repository.
2070-2094	Operation of the repository
2093-2094	Extension of the capacity of the repository to accommodate the decommissioned HLW.
2095-2104	• Second phase operation of the high level radioactive waste repository. • Transfer and loading of the decommissioned waste (HLW) from Paks NPP.
2015-2108	Sealing of the repository.

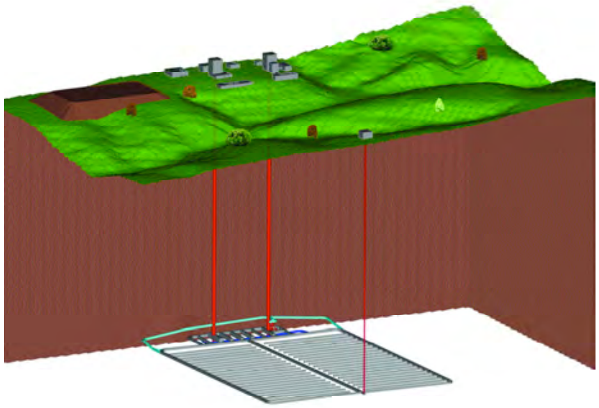

Figure 11.7. Axonometric diagram of the reference deep geological repository

The most important design assumptions are as follows:

- The development of the underground space is performed by conventional drilling and cautious blasting methods.
- Similar to the Swedish method, the canisters containing the SNF assemblies are stored one by one in large-diameter boreholes drilled into the base of the disposal drift (Figure 11.8).
- The engineering barrier (buffer) around the canisters is pure precast bentonite compacted at high pressure, the benefit of which is that it is a natural mineral material, capable of swelling upon absorption of water, has very low permeability, and the pre-cast elements are easy to handle.
- Other HLW is packaged into sealed stainless-steel containers and disposed of through various arrangements in the disposal drifts.
- The backfilling of the disposal drifts is performed after the completion of the loading operations in each drift, using an appropriate mixture of bentonite and excavated rock granulated to the appropriate grain size.

The SNF is loaded into the disposal drifts, into the shafts drilled in the base of the drifts. It is planned that these drifts will form a system developed in the form of a regular net within the area between the downcast airshafts and the ventilation, with upcast airshafts removing the exhaust air. Because of the separate air supply for transport and ventilation drifts, two independent ventilation systems are provided by the drift network. The driving of some four drifts in one wing of the facility will be enough to start the loading operations. Two drifts are planned to be loaded with SNF in a year, during which time another two drifts can be driven into the other wing of the disposal facility. Thus, the loading operations can be performed independently of the drift driving work. In this manner, the development of drifts can proceed at the required rate, up to a total number of ~42.

The facilities for the deep geological repository include a central service area, a system of abutments and drifts for the disposal operations, drifts for transport, man haulage, and ventilation, and a main upcast airshaft. The service area consists of units for various functions: a loading drift for man haulage and a transport shaft to receive smaller-sized equipment, accommodations for the control staff, and demonstration buildings. In addition, large-section drifts within the service area will be constructed for drainage functions, electric power supplies, machine-servicing and maintenance work, storekeeping functions, and vehicle parking. Also, there will be a reloading drift where larger-sized items of equipment, machines, and canisters containing the SNF are received and reloaded under safe conditions into radiation shielding tubes suitable for underground transport. Finally, bunkers will be built to collect rock debris produced by the space development work—bunkers interconnected with the drift beneath, where the rock is fed into the skip transport in the vertical shaft that is also used for man haulage. In this way, the excavated rock is transported to the surface.

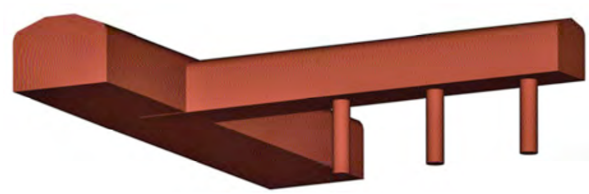

Figure 11.8. Axonometric diagram of the disposal drift arrangements

The backfilling material, used for sealing the disposal vaults, is carried down in the same way, by skip transport into another underground bunker, which is connected to a lower drift, where the material can be loaded onto transport cars for transfer to disposal drifts. Assuming undisturbed geological conditions, a drift profile designed for machinery based on the Swedish concept, and a single story arrangement within 700 x 700 m^2 overall dimensions, the disposal area will be capable of accommodating a total of 960 canisters containing SF. The actual number of drifts may vary depending on the geometry of each disposal area, if the length of the drifts or the dimensions of the abutments between them is restricted by unfavourable geological conditions. The distance between the disposal drifts, along with the pitch of the disposal shafts, is determined by the thermal parameters of the rock and the waste packages. A preliminary assessment of the thermal parameters indicates that an abutment size of 25 m and a shaft pitch of 10 m will be adequate for the Conceptual Design. The disposal shafts are connected to each other and to the service area by ventilation ducts and utility piping.

The canisters containing the SF are loaded into large-diameter boreholes driven into the base of the disposal drifts. Borehole dimensions are determined by the length and diameter of the canisters, and the design dimensions of the engineered barrier. The depth and the diameter of each borehole is ~ 7 m and 1.75 m, respectively. Bentonite slabs are prefabricated in the surface facility and taken down to their place of use.

It is assumed, based on the results of the cost estimate, that the total disposal area would consist of ~42 disposal drifts, each containing 24 SNF canisters. It is also assumed that for the large-section underground drifts, rock bolts and sprayed fibrous concrete lining with an average thickness of 10 cm would be required; for the small-section drifts (including the disposal drifts) rock bolts and a sprayed concrete lining 5 cm thick would be required.

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Chapter 12

The Japanese High-Level Radioactive Waste Disposal Program

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12.1. INTRODUCTION

The Japan Atomic Energy Commission (AEC) decided, in 1976, that management of high-level radioactive waste (HLW) in Japan should be based on the concept of geological disposal. Thus, a research and development (R&D) program for geological disposal should be started, and comprehensive R&D studies have been carried out for over 20 years in advance of formulating an institutional framework (including regulations) and selecting sites. In this comprehensive phase, the Power Reactor and Nuclear Fuel Development Corporation (PNC, now the Japan Nuclear Cycle Development Institute, JNC) has been assigned the role as leading organization in the R&D program.

PNC submitted the first progress report on research and development for geological disposal of HLW in Japan, referred to as H3 (PNC, 1992), in September 1992. H3 summarized the results of R&D activities up to March 1992 and identified priority issues for further study. JNC completed the second progress report, referred to as H12 (JNC, 2000) and submitted it to the AEC in November 1999. H12 demonstrated the technical feasibility and reliability of the specified disposal concept more rigorously and transparently, and provided input for future siting and regulatory processes.

Taking into account the technical achievements of H12, the "Specified Radioactive Waste Final Disposal Act" (hereafter "the Act") was legislated in June 2000, and the Nuclear Waste Management Organization of Japan (NUMO) was established in October 2000 as the implementing organization for vitrified HLW disposal. The assigned activities of NUMO include repository site selection, preparing relevant license applications, and construction, operation, and closure of the repository. The Radioactive Waste Management Funding and Research Center (RWMC, the former Radioactive Waste Management Center to promote R&Ds for radioactive waste management) was designated as the fund management organization in November 2000. More details on the evolution of the Japanese HLW program have been described in the *Third Worldwide Review* (Masuda and Kawata, 2001).

In this report, developments in the Japanese HLW program since the previous review are described, focusing on NUMO's implementation and the R&D activities of NUMO, JNC, and RWMC.

12.2. OVERVIEW OF THE FRAMEWORK FOR IMPLEMENTATION

The Act provides that the siting process shall consist of the following stages, as shown in Figure 12.1:

1. In the first stage, preliminary investigation areas (PIAs) for potential candidate sites are nominated, based on area-specific literature surveys and focus-

ing on the long-term stability of the geological environment.

2. In the second stage, detailed investigation areas (DIAs) for candidate sites are then selected from the PIAs by surface-based investigations, including boreholes drilled to evaluate the characteristics of the geological environment.
3. In the third stage, detailed site characterization, including underground experimental facilities, will lead to selection of the repository construction site.

The Japanese government (i.e., METI: Ministry of Economy, Trade and Industry) supervises the entire process as carried out by NUMO. For this purpose, according to the Act, METI (which succeeded MITI, Ministry of International Trade and Industry) is responsible for establishing a Basic Policy (MITI, 2000a) and a Final Disposal Plan (MITI, 2000b). According to the Act, the Final Disposal Plan shall be revised every five years. The Basic Policy requires that at every stage of the site selection process, NUMO must solicit the opinions of local residents, and METI must solicit the opinions of governors and mayors, and that these opinions be respected, as required by the Act. The Final Disposal Plan includes a schedule for the disposal program, which currently specifies that repository operation could start in the mid-2030s. In addition, as producers of HLW, the owners of the nuclear power plants are responsible for bearing the costs of the repository development program. The fund is managed by the RWMC.

The Act specifies that safety regulations will be established separately. Discussions on establishing safety regulations have been initiated by the Nuclear Safety Commission of Japan (NSC) and the Nuclear and Industrial Safety Agency (NISA). The NSC is responsible for providing guidelines for safety regulations. The NSC published the “First Report on the Basis for Safety Standards for HLW Disposal” (NSC, 2000) in November 2000, followed by a report entitled “Requirements on the Geological Environment for Selecting Preliminary Investigation Areas for HLW Disposal” (NSC, 2002) on September 30, 2002. These requirements should be reflected in PIA selection. NISA has identified the key issues in the R&D for formulating safety regulations (NISA, 2001).

The framework of R&D activities for geological disposal has been restructured in accordance with the Basic Policy Plan (MITI, 2000a), the Long-Term Program

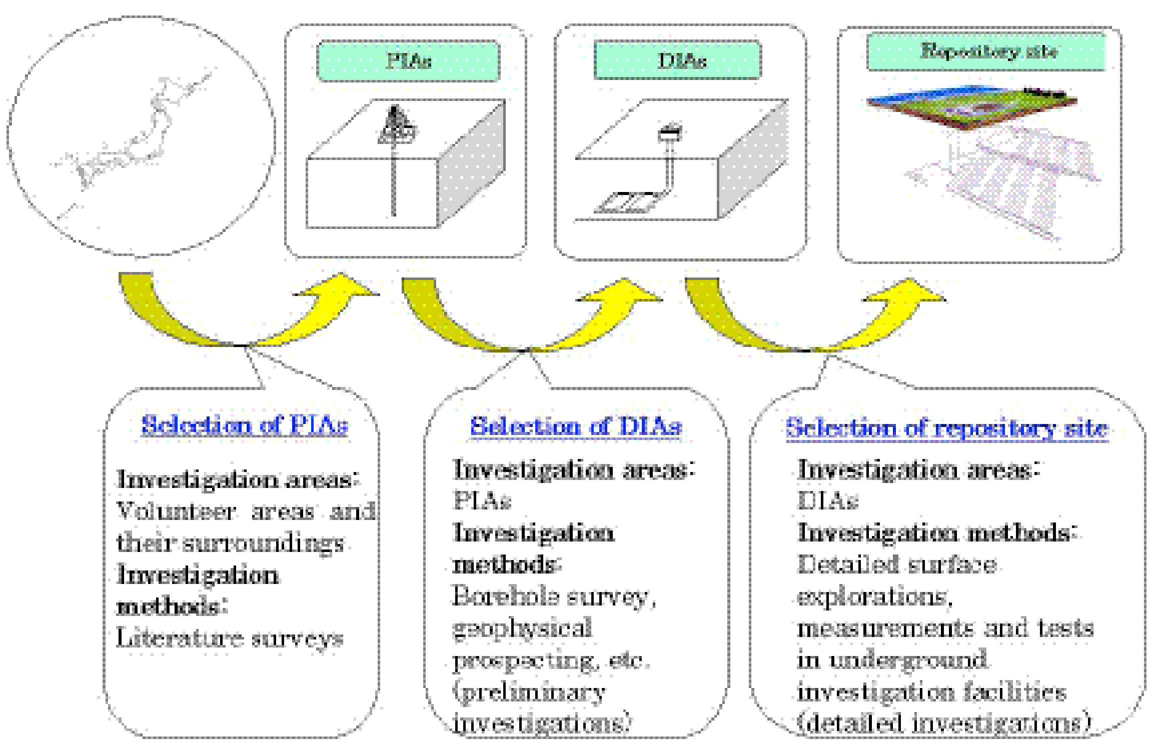

Figure 12.1. The three stages of the site selection process

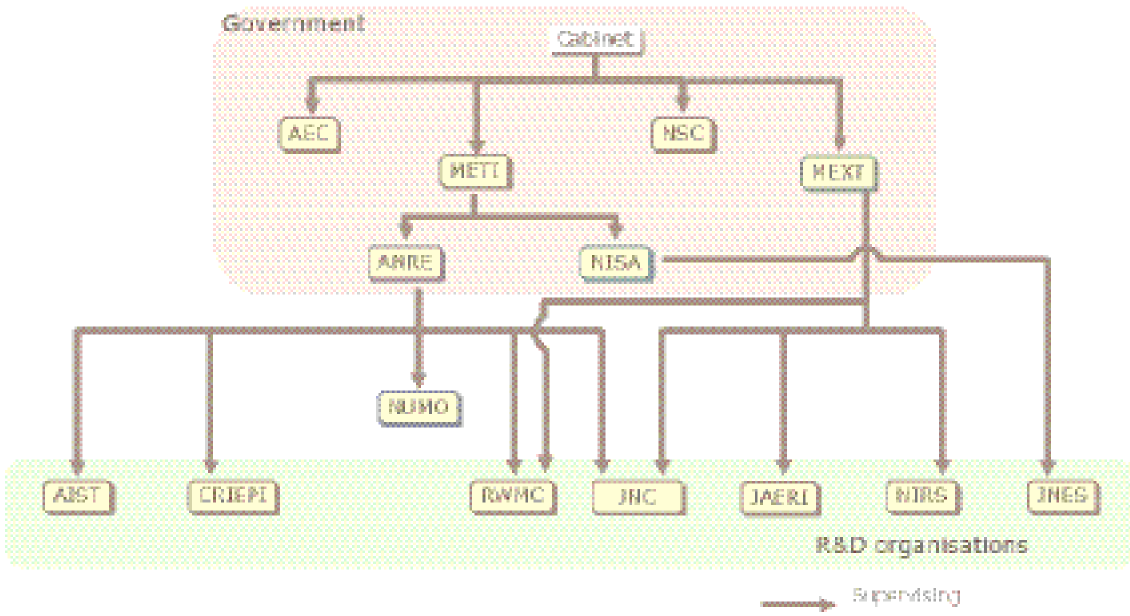

Figure 12.2. Relevant R&D organizations in the context of the Japanese HLW disposal program (modified from NUMO, 2004c, 2004d)

(AEC, 2000), and the Report of the Nuclear Sub-Committee (METI, 2001). NUMO is responsible for the R&D necessary for implementation of final disposal, with emphasis on safety, practical efficiency, and cost efficiency. The Government and relevant organizations carry out R&D to develop a technical basis for implementation of disposal and formulating regulations within the network indicated in Figure 12.2. One specific requirement for JNC is to further ensure the reliability of disposal technology and safety assessment (SA) methodologies, based particularly on studies at the Mizunami and Horonobe URLs and at ENTRY and QUALITY facilities. RWMC's various roles include supporting the national policymaking on geological disposal by pursuing R&D activities on sociological issues and advanced technological options, as well as gathering, analyzing, and producing information as appropriate.

12.3. STATUS OF NUMO ACTIVITIES

12.3.1. Activities Relevant to the Siting Process

The activities carried out for siting procedures by NUMO are supervised by METI. As specified in the Act, NUMO is required to submit a report describing the results of the investigations at the end of each stage and before proceeding to the next stage. The report will be published to local residents, and the document will be open for comments by them. METI must solicit opinions from the governors and mayors of concerned communities prior to finalizing decisions made during the site selection process. These opinions must be respected in terms of the decision-making process specified in the Final Disposal Plan.

Based on its fundamental policies, NUMO promotes public involvement in decision-making regarding the site selection procedure, which includes "adopting a stepwise approach," "respecting the voluntarism of municipalities," and "ensuring transparency." On this basis, an "open solicitation" approach for finding candidate sites has been chosen, in the belief that the support of local communities is essential to the success of this highly public, long-term project extending over more than a century. In light of this, NUMO has invited municipalities throughout the country to consider volunteering as candidate areas where the feasibility of constructing an HLW repository would be explored.

As the first milestone in the siting process, NUMO announced the start of open solicitation, to the public, of volunteer municipalities for preliminary investigation

Figure 12.3. Information package and associated technical documents

areas, with four documents published together as an information package on December 19, 2002. The information package, which provides basic information for supporting and promoting discussions by municipalities as to whether the repository plans could be accepted, has been sent to all (3,239) municipalities in Japan.

The information package shown in Figure 12.3 contains a brochure on open solicitation entitled "Open Solicitation for Candidate Sites for Safe Disposal of High-Level Radioactive Waste" and four separate documents, entitled "Instructions for Application" (NUMO, 2002a), "Repository Concepts" (NUMO, 2002b), "Siting Factors for the Selection of Preliminary Investigation Areas" (NUMO, 2002c) and "Outreach Scheme" (NUMO, 2002d). Informal translations of these Japanese documents into English can be found (and downloaded) on the NUMO website – www.numo.or.jp.

(1) Instructions for Application. This document includes an application form and general information for applicants, including the approximate extent of the volunteered area, procedures for prior confirmation of the geological conditions of the volunteered area, and the response after receipt of the application.

(2) Repository Concepts. This document is aimed at providing information on what the planned repository might look like and how it will be developed for siting environments at potential candidate sites selected, taking into account the siting factors. An overview of the performance of different repository concepts is also provided in the document. One of the repository concepts is shown in Figure 12.4.

(3) Siting Factors for the Selection of Preliminary Investigation Areas. This document provides guidance on area-specific surveys for PIAs. It includes factors specified in the Act, the first NSC report (NSC, 2000), and the NSC environmental requirements (NSC, 2002), such as no record of significant tectonic movement, no evidence of unconsolidated sediments, and no mineral resources. Siting factors are summarized in Table 12.1.

(4) Outreach Scheme. This document is aimed at outlining the benefits to volunteer municipalities, not only from a financial perspective, but also with respect to other positive social aspects. NUMO will conduct consultations with local residents regarding measures, that are appropriate to the conditions in an area and will make a serious effort to implement these measures.

12.3.2. Technical Documents

NUMO has been promoting R&D activities aimed at

Table 12.1. Siting factors

Evaluation Factors for Qualification (areas excluded as PIAs)	
- Clearly identified active faults	
- Within a 15 km radius of the center of Quaternary volcanoes	
- Uplift of more than 300 m during the last 100,000 years	
- Unconsolidated Quaternary deposits	
- Economically valuable mineral resources	
Favorable Factors (categories)	
- Geological formations	- Risk of natural disasters
- Hydraulic properties	- Procurement of land
- Geological environment	- Transportation

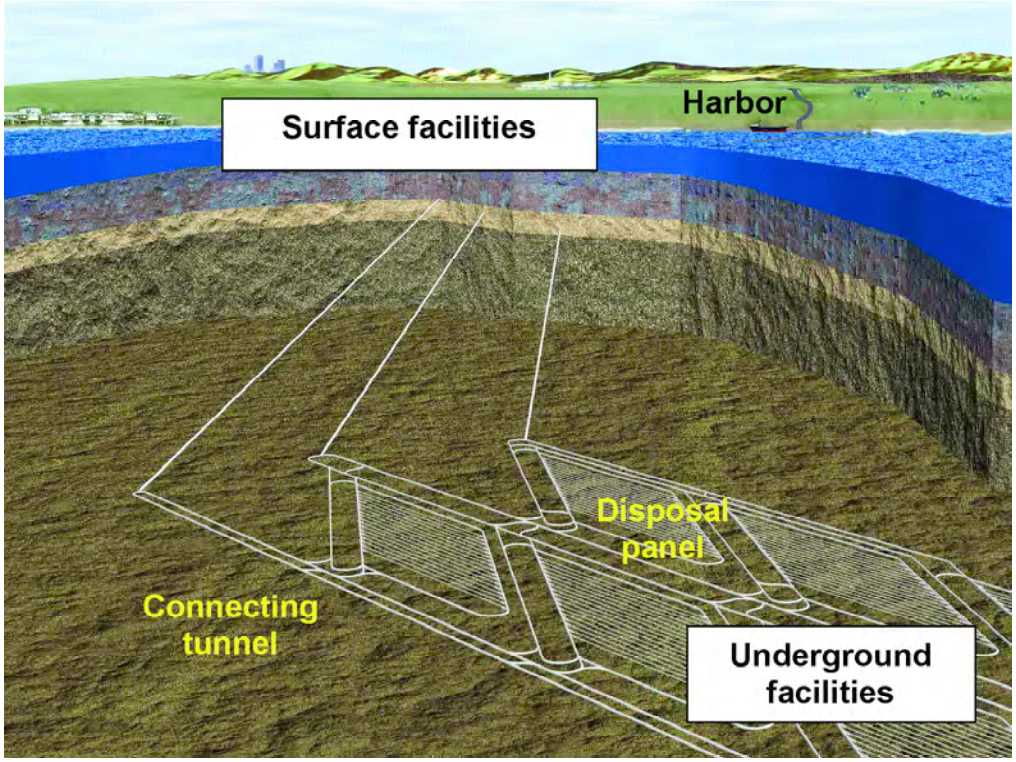

Figure 12.4. One of the repository concepts (coastal/sedimentary rock)

providing the scientific background of siting factors for selection of PIA, as well as approaches and methods for developing repository concepts tailored to volunteer sites. These two major issues have been summarized as associated technical documents, "Development of Repository Concepts for Volunteer Siting Environments" (NUMO, 2004a) and "Scientific Background and Practical Application of NUMO's Repository Siting Factors" (NUMO, 2004b) shown in Figure 12.3. The documents are compiled by following an open and transparent mode of operation, which allows all key stakeholders to become actively involved in considering the geological stability of a volunteer site—an important factor in a country like Japan, which lies in a tectonically active region.

The readers of these technical documents are expected to be stakeholders with more specialist knowledge and interest, who are looking to understand the scientific/technical (and economic) basis for the assertions made in the information package. The technical documents include technical achievements by NUMO so far, as presented at international conferences, workshops, and in journals. English versions of these technical documents, aimed at an international audience, have also been compiled (NUMO, 2004c, 2004d).

The concepts behind the technical documents are:

- To provide the scientific and technical basis to support the messages in the information package
- To provide convincing arguments to experts
- To describe all the details related to siting factors
- To clarify the future direction for developing repository concepts tailored to potential disposal sites

(1) Development of Repository Concepts for Volunteer Siting Environments

This document includes the basic principles involved in developing the repository concept, under the assumption that suitable volunteers will come forward, and introduces the concept of the "design factors" filter. It also focuses on the process of tailoring repository concepts to the environments found at a particular volunteer site and, in particular, on how to involve all relevant stakeholders in this process. The critical aspect of safety is discussed, with special focus on the time scales of most concern to the volunteer community. From the basis thus provided, it outlines the planned supporting program of R&D. Finally, it outlines how NUMO expects the program to develop as volunteers are characterized in increasing detail by desk and field studies, and the site-specific concepts are iteratively optimized for the particular environments involved.

In drawing up a strategy for iterative development of the basic repository concept as the process of site selection and characterization progresses, NUMO has used a structured approach. This includes both a bottom-up procedure for analyzing the applicability of particular repository subcomponents within the range of expected site characteristics, and a top-down approach for comparing alternative repository concepts or siting options. This structured approach extends step by step during the siting process, construction/operation and, finally, closure of repository. Although long-term safety is an essential requirement of all designs, NUMO also takes explicit account of safety principles. To assemble repository design options from the various subcomponents, NUMO examines a set of "design factors," each of which addresses an issue with direct bearing on the provisionally chosen design:

- Long-term safety
- Operational safety
- Engineering feasibility/quality assurance
- Engineering reliability
- Site characterization/monitoring
- Retrievability
- Environmental impact
- Socio-economic aspects

The design factors require the consideration of long-term safety, based on the starting-point provided by the reference H12 study (JNC, 2000), in addition to assessing the technological feasibility of constructing the repository. This consistent design policy optimizes long-term flexibility and allows reasonable designs to be obtained.

(2) Evaluating Site Suitability for a HLW Repository

This document introduces the siting factors that NUMO is using to select PIAs. It also provides geological and tectonic background information on Japan and describes how NUMO addresses the key issue of geological stability, which arises early in the siting program and which will have to be considered at the PIA selection stage. The document also describes the practical implications of using the siting factors and how they will be handled, as volunteers come forward, under actual site-specific geological and geographical constraints. It also looks at the tectonic setting and geology of Japan and the scientific basis for the application of the siting factors in a staged manner, as required in the various laws governing the current phase of the HLW disposal program. We chose to focus on this issue for two reasons: first, the siting factors that cover tectonic stability are applied from the very beginning of the siting program; second, given Japan's tectonic situation, they have more prominence than in many other national waste management programs.

The Japanese Islands lie at the junction of four major tectonic plates, and interactions of these plates—by subduction, collision, lateral movements, and accretionary processes—drive the seismic, thermal, and volcanic activity that characterizes the region. Such mechanisms are not unique to Japan: several other regions of the world are in island arc environments, and there are many examples of island arc tectonics in geological records, stretching back over hundreds of millions of years. Over the last decades, earth scientists have built up a broad understanding of the geometry and driving mechanisms of such tectonic systems by combining geological, geophysical, and geochemical observations. The latest satellite geodetic measurement techniques allow monitoring of crustal movements to test and calibrate models of surface deformation in response to tectonic strain. Compelling models exist for the controls on the distribution of earthquakes and volcanoes and for controls on crustal deformation. These models have been developed and tested in a large, integrated database. The main conclusions that have been drawn about stability are:

- The fundamental structure of the present plate system around the Japanese Islands was established approximately 15 Ma ago, when the spreading of the Japan Sea back-arc basin ceased.
- The directions of plate movement have not changed for the Pacific Plate over the last 2.5 Ma, and have not changed for the Philippine Sea Plate over the last 1.5 Ma.
- The movement of the plate system around Japan has been in a steady state since 15 Ma ago.
- It takes more than 1 million years for a significant change to take place within the plate system, and it is unlikely that any rapid change would occur with-

in the relatively short time period of 0.1 million years.

- It is therefore reasonable to extrapolate the knowledge about the main tectonic controls and patterns over at least the last 500,000 years to predict the long-term stability of the geological environment for at least the next 100,000 years.

12.3.3. Public Relations and Involvement

To promote public understanding, NUMO has carried out various publicity activities. NUMO considers that the initial and critical milestone in the process is the first applicant for a PIA. To encourage application, it is essential to initiate and develop nationwide discussion on HLW issues, which requires sufficient understanding of the characteristics of HLW and disposal options. Once these are understood, subsequent discussions should become smoother and more constructive.

NUMO organized public meetings at 31 different locations out of the 47 prefectures December 2001–November 2002, with a total of approximately 5,000 participants from the public. Local media at each location jointly hosted the meetings and reported the events as feature articles. These meetings provided NUMO with a better understanding of what the general public thinks and feels about HLW issues.

Since June 2003, NUMO and local newspapers have jointly hosted round-table talks with local opinion leaders at 32 locations up to now. The objective of these talks was to inform the public on HLW issues and to maintain dialogue with the public. Local newspapers reported the results of these activities as feature articles.

To promote public understanding, NUMO has been conducting information campaigns in leading newspapers, on TV, in magazines, etc. The TV campaign has been broadcast since October 2002, and NUMO's program has been advertised in magazines and major newspapers, including 49 local newspapers. An example of a newspaper campaign is shown in Figure 12.5.

A poster campaign was also conducted at major train stations between October 2002 and May 2003. Additionally, NUMO is planning to develop an interactive website for public dialogue. The information package, technical documents, booklets, videos and pamphlets, some of which have been produced in English, are available from the NUMO website.


いま電気を使っている
私たちの世代の問題です

Figure 12.5. Example of a newspaper campaign (discussion between NUMO President Fushimi and Prof. Kitano of Shukutoku University, an environmental expert)

12.4. JNC'S R&D PROGRAM IN THE IMPLEMENTATION PHASE

In accordance with the new framework (see Section 12.2), JNC continues to be responsible for R&D activities aimed at enhancing the reliability of disposal technologies and safety assessment methodologies, as well as the associated databases. JNC has thus actively promoted technical R&D with a view to contributing to both implementation and formulation of the safety regulations (see Figure 12.6).

12.4.1. URL Projects

One of JNC's key roles is to establish and demonstrate site characterization methodologies, based on investigations in two generic URL (underground research laboratory) projects, one at Mizunami (MIU) in crystalline rock and the other at Horonobe in sedimentary rock.

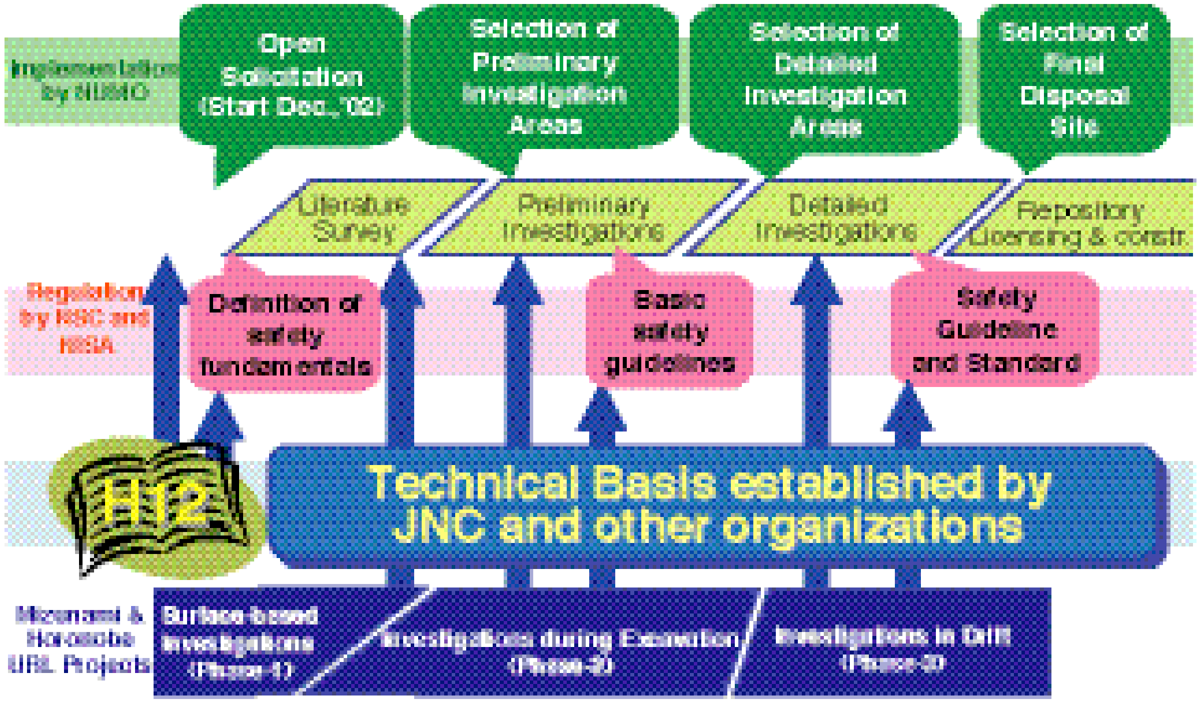

Figure 12.6. Contribution of R&D to implementation and safety regulations

These URLs are research facilities, distinct from the site-specific underground test facilities planned to be constructed at potential waste disposal sites by NUMO.

The URLs provide a wide range of possibilities for underground research in collaboration with universities or other research institutes, as well as serving as a tool for enhancing public understanding of R&D activities related to geological disposal. Data from the URLs will be widely published and will be expected to make a timely contribution to the disposal program and to the establishment of safety regulations, as follows:

1. Techniques will be developed for characterizing the geological environment from the surface to deep underground, based on investigations from the surface (Phase 1). This will take into account requirements relating to the design of the disposal system and safety assessment.
2. Data obtained from investigations during the excavation phase (Phase 2) will serve to verify the results from the surface-based investigation phase and characterize the evolution of the geological environment during drift excavation. Models will also be refined based on the geological data.
3. Detailed investigations in the underground facility (Phase 3) will contribute to refining geological investigation techniques that take into account the requirements of disposal system design and safety assessment. Data will also be compiled on the geological conditions to verify the reliability of models.

The MIU project began in 1996. The surface-based investigations at the MIU construction site, located on land owned by Mizunami City, began in March 2002 (JNC, 2002). Construction of the Mizunami URL (Phase 2) began in July 2002. According to the current design, the Mizunami URL will consist of two 1,000 m deep shafts. Shaft excavation began in July 2003 and, as of October 2004, entrance structures to the shafts had been constructed and shafts had been excavated to a depth of 50 m. Construction of the Mizunami URL is expected to be completed around 2010.

At the Horonobe URL, surface-based investigations (Phase 1) have been ongoing since 2001 (Goto and Hama, 2003). Following regional investigations, an area for intensive site investigations (URL area) was identified. The land for construction of the facility (URL site) was then acquired within the URL area, and site preparation began in July 2003. According to the present conceptual design, the Horonobe URL will consist of two 500 m deep access shafts. Shaft excavation (Phase 2) will start in 2005. Construction of the Horonobe URL is expected to be completed around 2010.

12.4.2. Developments in Repository Engineering and Performance Assessment

R&D on repository concepts and performance assessment has been focused on key issues and uncertainties identified in H12, which will be important for developing a convincing safety case. Basic studies and experiments in two surface-based laboratories, ENTRY for mainly engineering-scale experiments and the QUALITY facility for experiments using radionuclides, have been carried out to improve understanding of the long-term behavior of the geological disposal system, as well as to complement results obtained from the two URLs.

To enhance confidence in the H12 design concept, the robustness of key safety functions of the engineered barrier system (EBS), such as physical containment by the overpack and retention of radionuclides in the bentonite buffer, have been studied for saline groundwater expected at coastal sites and for the high-pH plume which might be caused by the cementitious material used in the repository (e.g., Taniguchi, 2003). Work has also been initiated to improve understanding of coupled THMC processes in the near field, to better evaluate system evolution (Neyama et al., 2003). This will provide a technical basis for possible preclosure monitoring, aimed at confirming the initial conditions for the post-closure safety assessment.

Assessment of the potential impact of natural perturbation phenomena (volcanic activity, fault movement, uplift/erosion, climate/sea-level change) is essential for increasing confidence in the safety of the disposal system. In this regard, the simplified and stylized approach in H12 to assessing such scenarios has been improved, by developing more realistic models that can simulate the safety-relevant impact of these perturbations. This approach can also provide useful guidance to help focus site investigation programs.

Studies on model uncertainty are focused on processes related to key safety functions, such as:

- Long-term glass dissolution
- Diffusivities and porosities of gouge within fractures
- Colloid-facilitated radionuclide transport in fractured rock
- Complexation of key radionuclides with natural organic substances

Development and updating of key databases include determination of thermodynamic and sorption data for safety-relevant elements under relevant conditions, including hyperalkaline and saline systems. To ensure quality, standardized data acquisition methods for these databases are under development.

12.4.3. Integration of Technical Achievements into a Knowledge Base

Through the surface-based investigations in the Mizunami and Horonobe projects (Phase 1), integration of work from different disciplines into a "geosynthesis" has been illustrated and is to be developed further in the underground facilities at these sites (Phases 2 and 3). These projects also serve for developing and testing the tools and methodologies required for site characterization. Further know-how will be gained through participation in foreign underground laboratory projects, transfer of experience from these projects to Japan, and tailoring this experience to Japanese conditions and requirements. This experience represents an important knowledge base, which is obviously important for the implementer, but is also needed by the regulator to assess how key site characteristics are derived and what uncertainties are associated with this process.

Such site investigations and subsequent repository design studies provide input for associated safety assessment. In turn, safety assessment has an important role to play in guiding site investigations and repository design activities, by quantifying key sensitivities and uncertainties. An integrated program in which these topics are developed iteratively is thus needed. Such integration requires experienced multidisciplinary project teams that should also have a wide perspective, provided by participation in multinational projects and familiarity with other national HLW programs.

JNC has developed the JNC Geological Disposal Technical Information Integration System (JGIS), to facilitate program integration and sharing of technical information between the site investigation, repository design, and safety assessment teams (Uchida et al., 2003). JGIS is an archive system in which a relational database stores technical information in the form of a structured flowchart that systematically represents the structure of research activities. In addition to the development of such a structured knowledge base, JNC will publish a state-of-the-art report, "H17," in autumn 2005, which will document R&D results based on the Phase 1 activities in the two URLs and studies in ENTRY and QUALITY, as well as future development of a knowledge management system for geological disposal.

12.5. RWMC'S R&D ACTIVITIES

In accordance with the new framework (see Section 12.2), RWMC has been pursuing R&D activities related to sociological issues and advanced technological options, as well as gathering, analyzing, and providing information as appropriate, to support METI policy-making for geological disposal.

12.5.1. R&D Activities on Sociological Issues

Sociological considerations are applied to postclosure monitoring and long-term record preservation, to enhance confidence-building. Much effort has been taken to structure the related information in data flow diagrams, technological menus, and visual aids, all of which must be transparent, traceable, and accountable not only to experts, but to all stakeholders (Ohuchi, et al, 2003, 2004).

12.5.2. R&D Activities on Advanced Technological Options

To expand the technological options and enhance traceability and usability in implementing the repository, RWMC has been investigating state-of-the-art technologies and some complementary technologies, such as site investigation, remote operation of the repository, and engineering barriers.

Site Investigation Technology

Because the site investigation might need scientific judgment based on highly specialized knowledge of the process, Site Investigation Flow Diagrams (SIFD) have been developed to make a site investigation plan, supervise the investigation procedure, and evaluate the investigated results—in such a way as to ensure transparency, traceability, and accountability for all stakeholders. Based on evaluations of possible site investigation technologies, R&D studies to improve electromagnetic surveys and seismic tomography to characterize concealed faults and low water permeabilities (including seashore) have been pursued at Monterey, California, and at the Grimsel Test Site in Sweden. To confirm potential repository performance within the Japanese geological environment, field tests at Horonobe and Mizunami URLs have been started in cooperation with JNC, based on the bilateral agreement (Yoshimura et al., 2003; 2004a; 2004b).

Remote Welding, Inspection, and Emplacement Technology

Based on the disposal concept in the H12 report, RWMC has been developing key technologies for remote operation of the repository. The integrity of the overpack lid closure is very important in guaranteeing the confinement of HLW for more than 1,000 years. The applicability of possible welding technologies such as arc and electron beam, as well as inspection technologies such as ultrasonic testing and alternating current field magnetic method, have been investigated, along with carbon-steel work-pieces with maximum thickness of 190 mm. The applicability of possible handling and emplacement technologies—such as vacuum suction cup to handle the compacted bentonite blocks, *in situ* compaction of bentonite powder, *in situ* filling of bentonite pellets, and pre-assembled packaging with air bearing units—have been demonstrated, in combination with vertical and horizontal emplacement concepts (Asano et al., 2004, 2005a, 2005b).

12.5.3. Information-Related Activities

To pursue the mission of providing appropriate information in support of national policy, RWMC gathers information related to geological disposal of HLW, analyzes that information, and decides whether the information is appropriate to policy and safety regulations. The collected information is stored in a database, with recent developments of HLW programs in various countries provided to the public through the RWMC website (http://www.rwmc.or.jp).

12.6. CONCLUSIONS

NUMO has chosen a volunteer approach to site selection in the belief that the support of local communities is essential to the success of this highly public, long-term project extending over more than a century. The siting process needs to more vigorously incorporate discussion with, and decisions by, municipalities. It is therefore particularly important to promote public understanding of geological disposal and to obtain and maintain public trust. To ensure that the decision-making process is transparent, NUMO makes available diverse information relevant to its siting activities through the publication of documents, websites, etc., and will provide opportunities for residents in the vicinity of PIAs to voice their opinions. To promote this communication, NUMO has been carrying out a range of public relations activities.

This unique approach presents particular challenges for the repository concept development program. Thus, NUMO intends to provide regular updates on how to pursue and apply repository concept development and site evaluation techniques. For this purpose, NUMO is using a structured approach, which is applicable for all stages of HLW disposal implementation. Variants of the "requirements management system," which could be integrated with the development of "knowledge management" and "quality management" systems, are being investigated.

The Japanese Government and relevant organizations (such as JNC and RWMC) have been promoting R&D aimed at increasing confidence in the technical basis provided in H12 by making maximum use of its infrastructure. The experience gained in the JNC URL projects at Mizunami and Horonobe will contribute to the NUMO site characterization program by providing tools, experience, and manpower, as and when required. Through development of a quality-assured knowledge base, JNC can also act as a valuable resource for both the implementer and the regulator. JNC will be integrated with the Japan Atomic Energy Research Institute to form a new organization in October 2005. A structured approach to managing technical knowledge on geological disposal is also critical in this regard, allowing the new organization to preserve a valuable legacy of intellectual property and to continue to play a central R&D role in the HLW disposal program. This knowledge management system, associated with the knowledge base, can be complementary with NUMO's requirement management system.

Finally, it should be noted that international collaboration has been playing a key role in the Japanese HLW Disposal program. NUMO, JNC and RWMC have concluded international bilateral collaboration agreements with several organizations such as Nagra, SKB, and ANDRA. NUMO has also joined EDRAM (International Association for Environmentally Safe Disposal of Radioactive Materials) and organized the International Advisory Committee, made up of experts who have specific knowledge and expertise in subject areas relevant to NUMO's project activities. The results of these collaborations have contributed both to preparation of JNC's progress reports (such as H12) and to RWMC's activities. International collaboration will be continued in the Japanese program, as an opportunity for sharing experience and applying that experience towards the final goal.

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Chapter 13

Radioactive Waste Disposal of LILW and HLW in Korea

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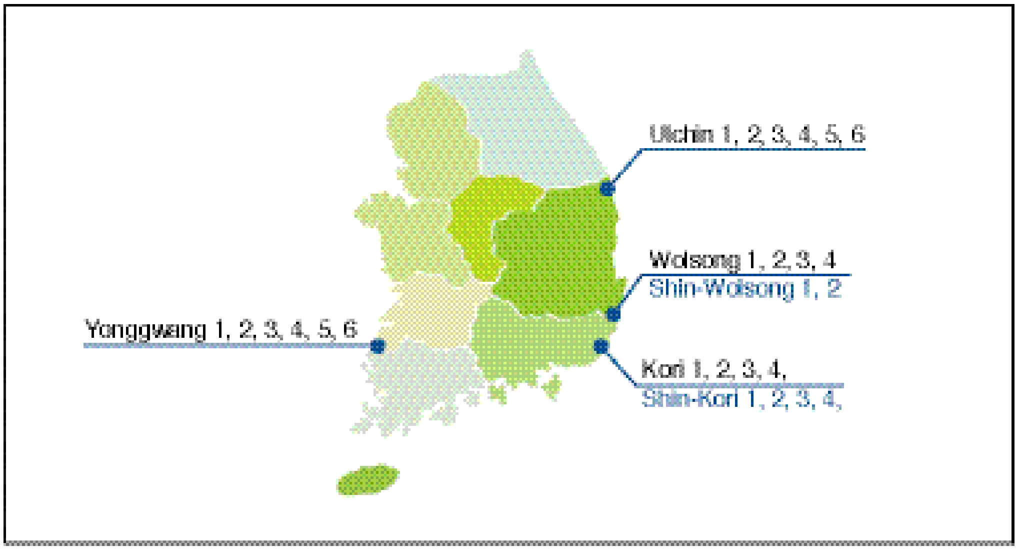
ABSTRACT. The Korean nuclear industry continues to show growth. Twenty nuclear power plants are in operation, and the accumulated amount of low- and intermediate-level radioactive waste (LILW) is approaching the saturation capacity of on-site storage. The national plan is to have an operational LILW disposal facility by 2008 with a capacity of 800,000 drums. Since 1986, there has been an ongoing effort to secure a site for LILW disposal and spent nuclear fuel (SNF) interim storage, but without success. We changed the site securing approach in 2005 to separate sites for an LILW disposal facility and an SNF interim storage facility. Based on a new site selection process in 2005, Kyungju was selected as the repository site for LILW. Research and development (R&D) programs for the safe disposal of LILW have been conducted actively and will contribute to the national radioactive waste disposal project. For safe disposal of high-level radioactive waste (HLW), there has been a three-step, 10-year program of R&D to establish a Korean reference disposal system (KRS) by 2006. Within this HLW disposal program, we have carried out repository system development studies, performance/safety assessment, geoscientific environmental research, and system performance validation.

13.1. LOW AND INTERMEDIATE-LEVEL RADIOACTIVE WASTE (LILW) DISPOSAL

13.1.1. Introduction

In Korea, the safe management of radioactive waste is a national effort, one that is required for sustainable generation of nuclear power and for energy self-reliance. Nuclear power generation started in Korea in 1978. Since then, a rapid growth in nuclear power supply has been achieved; Korea has 20 nuclear power plants (NPPs) with a total installed capacity of 17.7 GWe (see Figure 13.1). Moreover, the future NPP construction program is also ambitious: six NPPs will be constructed by 2010. Such a large nuclear power generation program has produced a significant amount of radioactive waste, both low- and intermediate-level waste (LILW) and spent nuclear fuel (SNF).

In the early stages of nuclear power development in Korea, the philosophy of the radioactive waste management was to reduce the waste generation volume by conditioning and storing the waste within the nuclear power plant sites. However, with the LILW at these plants ever increasing, the accumulated amount of LILW is approaching the anticipated saturation capacity. (Table 13.1 shows the accumulated amount of LILW as of November 2004.) Therefore, the necessity to find a final disposal facility of LILW becomes imminent. Moreover, because of the extensive application of radioisotopes (RI) in Korea, the volume of RI waste generation from industries, hospitals, and research organizations has also been increasing.



Nuclear Power Plants in Operation

Name	Unit No	Capacity (MW)	Commercial Operation	Reactor Type
Kori	#1	587	Apr. 29, 1978	PWR
	#2	650	Jul. 25, 1983	
	#3	950	Sep. 30, 1985	
	#4	950	Apr. 29, 1986	
Wolsong	#1	679	Apr. 22, 1983	PHWR
	#2	700	Jul. 01, 1997	
	#3	700	Jul. 01, 1998	
	#4	700	Oct. 01, 1999	
Yonggwang	#1	950	Aug. 25, 1986	PWR
	#2	950	Jun. 10, 1987	
	#3	1,000	Mar. 31, 1995	
	#4	1,000	Jan. 01, 1996	
	#5	1,000	May 21, 2002	
	#6	1,000	Dec. 24, 2002	
Ulchin	#1	950	Sep. 10, 1988	PWR
	#2	950	Sep. 30, 1989	
	#3	1,000	Aug. 11, 1998	
	#4	1,000	Dec. 31, 1999	
	#5	1,000	Jul. 29, 2004	
	#6	1,000	Apr. 22, 2005	

Nuclear Power Plants Under Construction

Name	Unit No	Capacity (MW)	*Commercial Operation	Reactor Type
Shin-Wolsong	#1	1,000	(2011)	PWR
	#2	1,000	(2012)	
Shin-Kori	#1	1,000	(2010)	PWR
	#2	1,000	(2011)	
	#3	1,400	(2012)	
	#4	1,400	(2013)	

* Planned for commercial operation

Figure 13.1. Nuclear power plants in Korea (as of April 2005)

Table 13.1. Accumulated amounts of LILW storage (as of Nov. 2004)

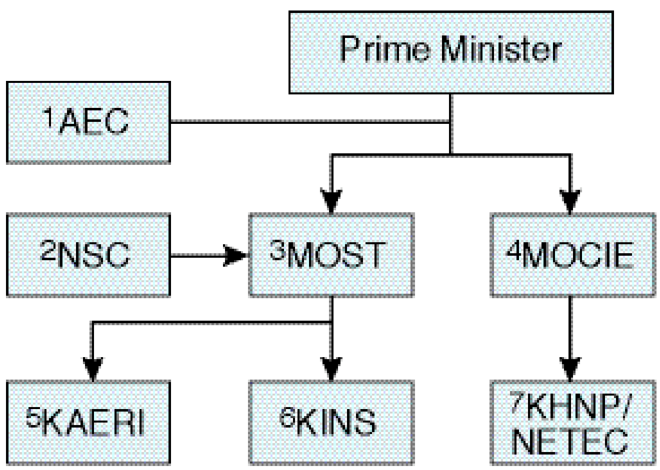
LILW	Storage Capacity (drums)	Cumulative Amount (drums)	Year of Saturation
Kori	50,200	32,925	2014
Younggwang	23,300	12,963	2012
Ulchin	17,400	13,388	2008
Wolsong	9,000	4,635	2009
Total	**99,900**	**63,911**	**-**

13.1.2. National Policy and Organizational Structure

In 1997, Korea Hydro & Nuclear Power Co., LTD. (KHNP) conducted a study of the national radioactive waste management policy, in consultation with experts from various fields, and a final report based on this study was presented to the Korean government. Based on this report, on Sep. 30, 1998, a new national program for radioactive waste management policy was approved by the Korean Atomic Energy Commission (AEC), which had full authority to make policy decisions related to atomic energy in Korea. The fundamental principles of this national radioactive waste management policy include (1) direct control by the government, (2) top priority on safety, (3) minimization of waste generation, (4) a “polluter pays” principle, and (5) transparency of the site selection process.

The policy making and regulation for licensing of all the nuclear activities in Korea are based on the provisions of the Atomic Energy Act, the Enforcement Decree and Enforcement Regulation of the Act, and the Notice of the Ministry of Science and Technology (MOST). Specifically, practical regulation and technical standard details are addressed in provisions of the MOST. A license applicant had to be approved by MOST, and facilities or activities had to be in accordance with designated regulations within the legal framework. Figure 13.2 shows the major legislative framework related to the nuclear industry in Korea.

The AEC is the highest policy-making body on nuclear matters. The Deputy Prime Minister is the chairperson of the AEC. The MOCIE (Ministry of Commerce, Industry and Energy) is the government authority in charge of national energy supply and supervises radioactive waste management, including the LILW/SNF site-securing project by KHNP. MOST, as the government regulatory authority, is responsible for establishing and implementing nuclear regulatory policies for the control of nuclear activities related to power and research reactors, radiation applications, etc. MOST is also responsible for R&D in the peaceful use of nuclear energy. The principal function of the Nuclear Safety Commission (NSC) is decision making on major



1. AEC (Atomic Energy Committee),
2. NSC (Nuclear Safety Commission),
3. MOST (Ministry of Science and Technology),
4. MOCIE (Ministry of Commerce, Industry and Energy),
5. KAERI (Korea Atomic Energy Research Institute),
6. KINS (Korea Institute of Nuclear Safety),
7. KHNP/NETEC (Korea Hydro & Nuclear Power Co., Ltd./Nuclear Environment Technology Institute)

Figure 13.2. Major legislative framework of nuclear industry in Korea

nuclear safety and regulatory policies and licensing issues. The Korea Institute of Nuclear Safety (KINS) was established to support MOST with its technical expertise and is entrusted with the duty of safety regulations. KINS performs safety review and inspection, and develops safety standards. KHNP is the sole nuclear power licensee and is in charge of radioactive waste management nationwide through its affiliated institute, Nuclear Environment Technology Institute (NETEC). Korea Atomic Energy Research Institute (KAERI), which is funded primarily by MOST, is in charge of research into peaceful application of nuclear energy, including production of radioisotopes for domestic consumption.

13.1.3. History of Site Selection for LILW Disposal Facility

Site selection activity for LILW disposal facility began in 1986. There were several attempts for siting along the Korean eastern seashore between 1988 and 1989, and there was also site-securing activity into the early 1990s. In 1990, Anmyon Island was proposed as a candidate disposal site, and in 1995, Gulup Island was also proposed as a site, but then cancelled, as a result of strong objection by antinuclear groups and local residents at that time.

In 1997, the responsibility for securing the disposal site was assumed by the KHNP. From that time on, the strategy of KHNP for securing the disposal site was based on the voluntary application of local governments and negotiation with local governments after the selection of the candidate site. During the one-year site subscription period from June 2000 to June 2001, nine coastal communities were organized for voluntary site subscription. Among the nine communities, seven submitted petitions to their local governments. However, all petitions were rejected by the local governments.

On February 4, 2003, after an intensive screening process that considered the social/natural environment and the feasibility of waste transportation, MOCIE and KHNP announced the candidate sites: two in the eastern coastal area and two in the western coastal area. Voluntary subscription from Korean local governments, including these four candidate sites, was expected, and finally, on July 14, 2003, Buan County, located on the western coast seashore, applied for the subscription. A preliminary site-evaluation committee was organized to review the natural/social environment of the applied site. Later that month, on July 24, 2003, Wido Island was announced as a candidate site. However, there was strong opposition to consideration of this site from antinuclear groups and some nearby local residents in Buan County.

Continuing effort to secure the LILW disposal and SNF interim storage sites resulted in a new announcement for a public solicitation process on February 4, 2004. After this announcement, 10 local communities announced their intention to petition for the site. Public hearings were conducted on Sept. 15, 2004, and preliminary applications were submitted by the local governors. However, as of November 30, 2004, there had been no final application for the site.

As a result of the AEC meeting on December 17, 2004, the policy on low- and intermediate-level radioactive waste management has been changed. The new AEC policy is basically to separate the sites—one site each for an LILW disposal facility and an SNF interim-storage facility, rather than constructing both on one site. To accommodate this new policy, special laws were promulgated to support the local community hosting an LILW disposal site. This law promises the residents of the host community that the LILW disposal site will not be combined with SNF interim storage. The government announced new application procedures for LILW candidate disposal site in early 2005.

After the announcement of the new procedure, four local governments petitioned to become a host community of the repository. Among them, one candidate site was located along the western seashore (Kunsan) and the other three sites were located along the eastern seaside (Youngduk, Kyungju, and Pohang). Based on the new site selection procedure, a local referendum was conducted simultaneously at all four local governments in November 2005. Based on the local referendum results, the city of Kyungju had the highest approval rate for the candidate repository site to be located near Wolsong NPP site and became the host community for the national LILW repository. Currently, a detailed site investigation as well as an environmental impact assessment is being conducted at the Wolsong site for licensing application.

13.1.4. Research and Development for LILW Disposal

KHNP/NETEC is conducting extensive R&D related to radioactive waste management, focusing on (1) safety

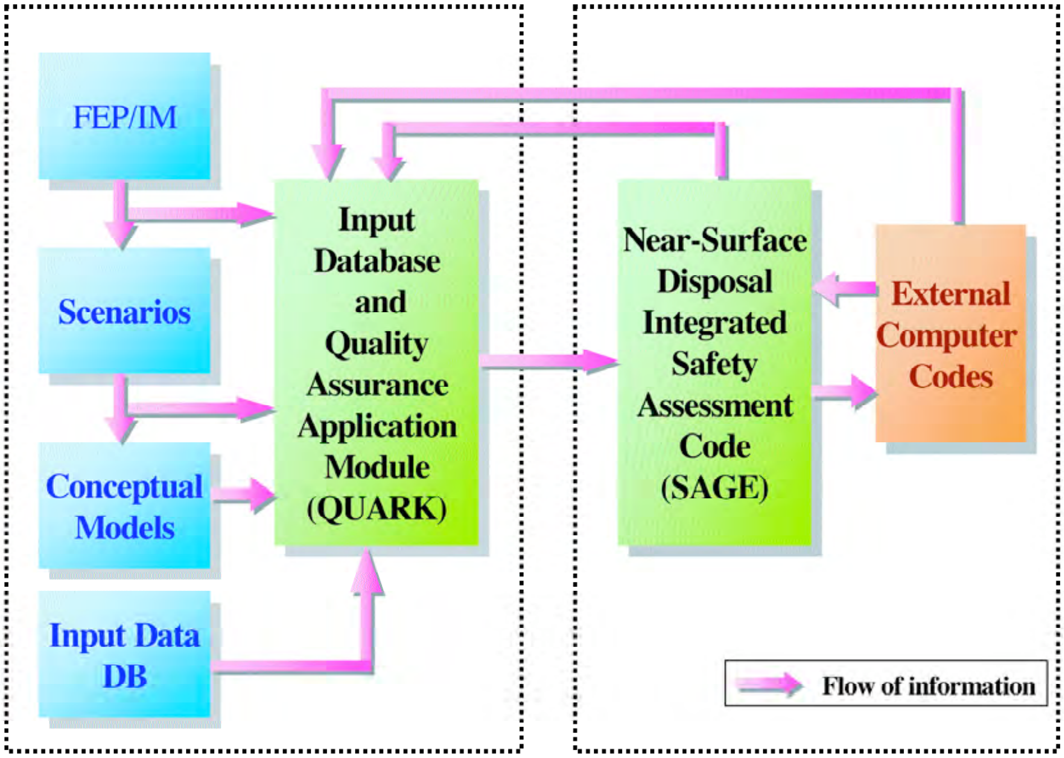


Figure 13.3. Interaction and relationship between QUARK and SAGE in an integrated safety assessment system

assessment technology for an LILW disposal system, (2) experimental performance testing of a near-surface barrier cover system, (3) site and environmental data management systems for site characterization, and (4) volume reduction technology by a vitrification process.

National R&D programs for LILW disposal system are implemented to establish a comprehensive safety assessment system framework, including safety assessment methodology/procedures, a conceptual design for both near-surface and rock-cavern-type disposal systems, development of assessment code systems, and validation of each code.

An integrated safety assessment system, to be used for evaluation of the near-surface disposal concept, has been developed as part of the safety assessment methodology for LILW disposal. This system is designed to provide disposal safety in a clear, comprehensive, and well-documented manner, and to integrate the results into a defensible package, showing reasonable assurance of compliance with the regulatory requirements for licensing application.

This system is made up of two key components that interact with each other: a system-level safety assessment code, called SAGE (Safety Assessment Groundwater Evaluation), and an input database/quality assurance module for safety assessment, named QUARK (QUality Assurance and database for Radioactive waste management in Korea). SAGE performs the safety assessment calculations, either deterministically or probabilistically, considering long-term radionuclide releases to the groundwater system in the engineered-vault-type LILW disposal concept. QUARK manages both information analysis and input parameter database for SAGE under quality assurance guidelines. SAGE is composed of three categories: near-field, far-field, and biosphere models.

From a technical perspective, the code has been tested extensively against existing analyses of LILW disposal facilities and other computer codes (J.-W. Park et al., 2003; J.B. Park et al., 2003a; J.B. Park et al., 2005). Excellent agreement has been achieved for a number of situations. These benchmark analyses provide a high degree of confidence that the code has been implemented correctly, and that the results generated using the code have a high degree of reliability. The data in the QUARK database is defined by parameters required by

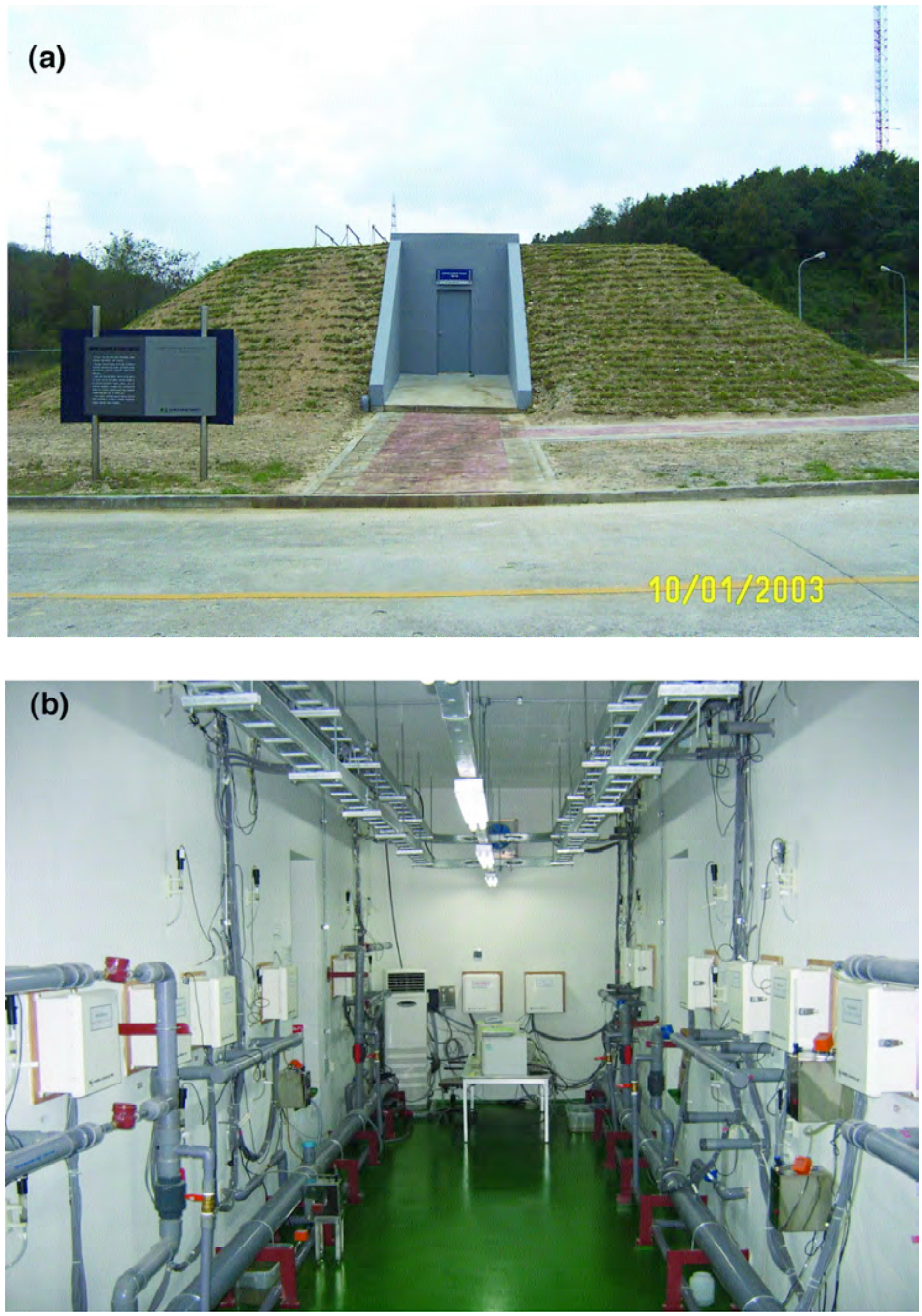


Figure 13.4. EBS test facility for near-surface disposal at KHNP/NETEC
(a) Front view of engineered barrier test facility;
(b) Experimental laboratory for near-surface cover performance

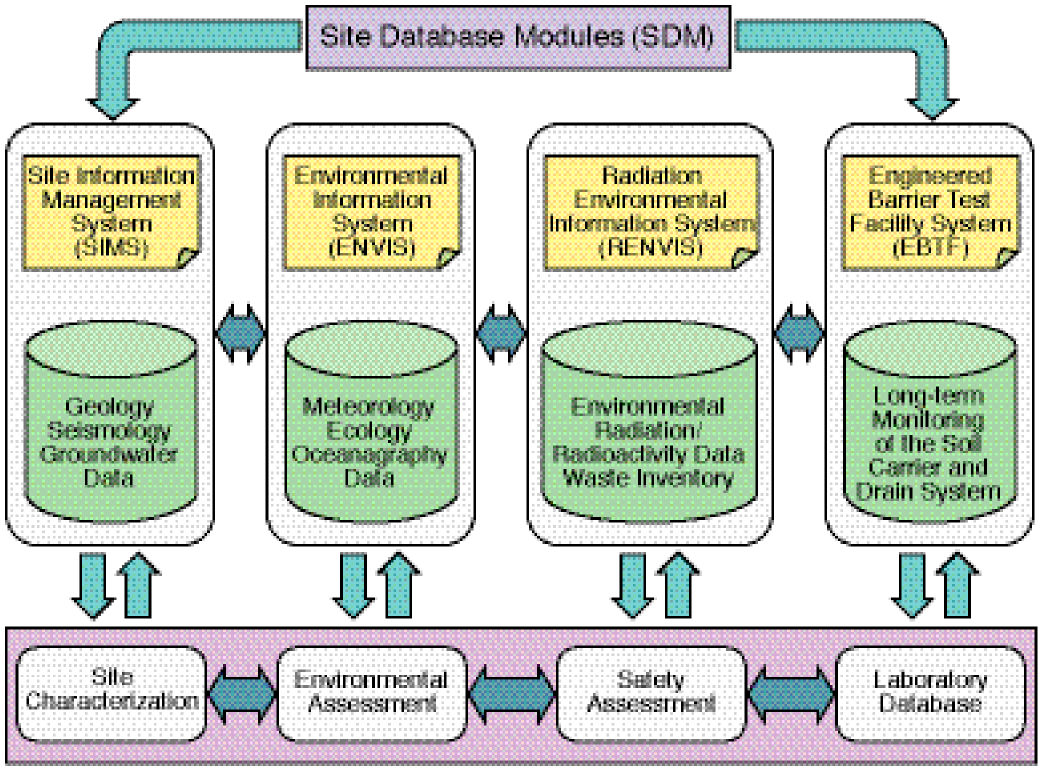


Figure 13.5. Site database management (SDM) system in SITES Program

the SAGE assessment code. QUARK stores and manages the safety assessment information under QA guidelines consistent with international standards. As can be seen in Figure 13.3, the information is organized and linked to maximize the integrity and traceability of information, through the coupling between SAGE and QUARK.

The pilot test facility has been constructed, and a performance demonstration test has been conducted to verify the performance of engineered barriers at the near-surface disposal facility. Figure 13.4 shows the front view of the test facility (Figure 13.4a) as well as a view of the experimental laboratory for near-surface cover performance (Figure 13.4b) (J.B. Park et al., 2003b).

KHNP/NETEC is developing a site and environmental data management program called SITES (Site Information and Total Environmental Data Management System). This program, funded as a National Research Laboratory (NRL) research program by MOST, facilitates site and environmental data management and the evaluation of nuclear facilities, including an LILW disposal facility. The purpose of the SITE program is to collect and manage site characterization data produced from radioactive waste disposal sites and to manage the safety assessment data required for a license. The SITES program contains a site database management (SDM) module and monitoring/assessment (M&A) module. Figure 13.5 illustrates the structure of the site database module. The M&A module is used for integrated safety assessment and disposal site environmental monitoring (S.M. Park et al., 2003; Kim et al., 2003; Ko et al., 2004; S.M. Park et al., 2004).

13.2. HIGH-LEVEL RADIOACTIVE WASTE (HLW) DISPOSAL

13.2.1. Introduction

The Korea Atomic Energy Research Institute (KAERI) launched a three-step 10-year R&D program in 1997 to develop a reference geologic repository system for

HLW by 2006. The program has now entered into its third year of the last step (2002–2006), in preparation of the Korean Reference (Disposal) System (KRS).

The R&D program has been carried out in four different areas: repository system development, performance/ safety assessment, geoscientific environmental research, and performance validation of the proposed HLW disposal system.

13.2.2. Repository System Development

Packaged SNF, encapsulated in a corrosion-resistant container, is to be disposed of in a mined underground facility, located at about a 500 m depth in a crystalline rock mass. No site for the underground repository has been specified in Korea, but a generic site with granitic rock has been considered. The waste packages that contain SNF are to be placed in boreholes drilled into the floor of disposal drifts. Alternatives to the baseline concept concerning the emplacement patterns of the container, waste packaging methods as well as the distance between deposition holes and tunnels, are also considered, together with the reference concept. The engineered barrier system is to be composed of the SNF itself, the waste package, clay-based buffer, backfill material, and other man-made items for plugging and sealing purposes in the underground facility.

The reference PWR SNF has an average burn-up of 45,000 MWd/tHM (initial enrichment of 4.0 wt%) and the fuel dimensions are 21.4 cm x 21.4 cm x 453 cm (length). The reference CANDU fuel has the average burn-up of 7,500 MWd/tHM, and the fuel dimensions are 10 cm (diameter) x 49.5 cm (length). Because of the significantly different properties of both fuel types, the reference container separately encapsulates the spent PWR and CANDU fuels. However, the overall sizes and component materials of the containers for both SNF types are proposed to be identical, to simplify the encapsulation and handling processes in the repository. The dimensions of the container were determined from the mechanical structural analysis under the expected mechanical loads in underground repository conditions.

The underground repository system includes two service mains that run the length of the repository, con-

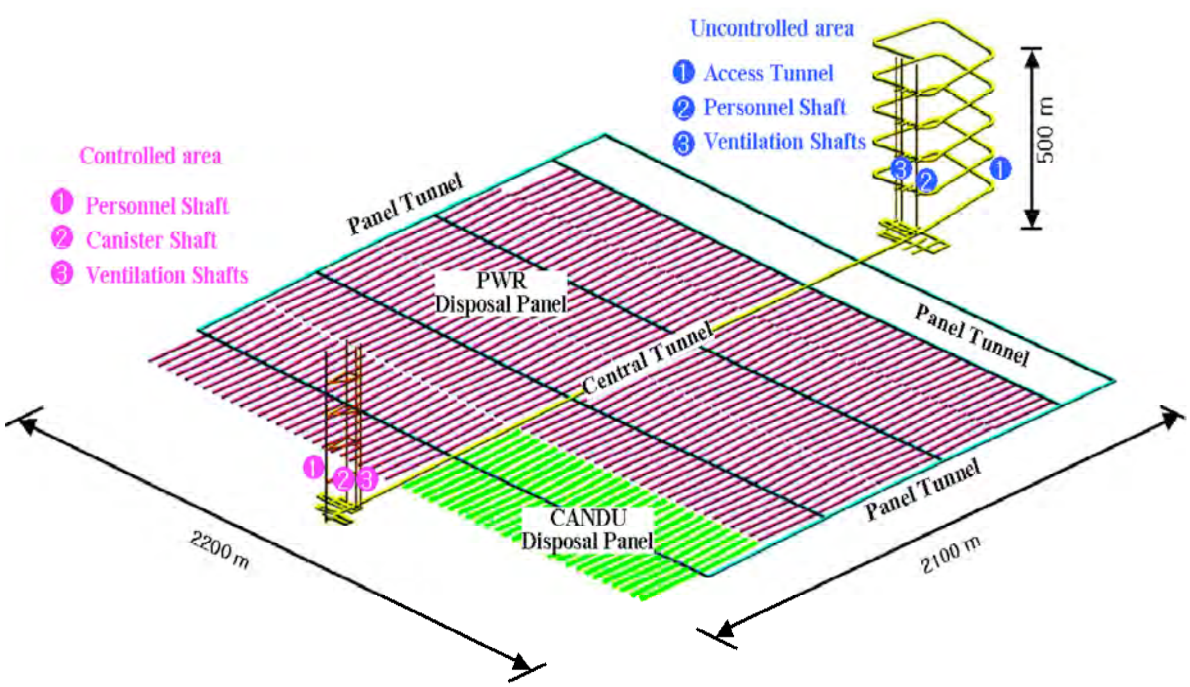


Figure 13.6. Schematic diagram of the preliminary layout for KRS-1

necting each shaft complex. The entire facility is bounded by a panel and a central tunnel, which functions for both ventilation and access/operations. The disposal area consists of eight disposal panels capable of future expansion. Based on a 40 m emplacement tunnel spacing, disposal tunnels for PWR and CANDU fuels are 323 and 54 tunnels, respectively. Each emplacement tunnel is 254 m long, which includes about 10 m end-standoff and about 17 m standoff at the entrance for emplacement works. The entire underground facility requires an area of about 4.6 km^2 (Figure 13.6).

Alternative disposal concepts being considered in the study include multiple-level waste emplacement with shaft access. These, in turn, have a potential impact on disposal room layout, ventilation, construction sequence, etc. Each concept has construction-related features that have advantages and disadvantages.

13.2.3. Performance/Safety Assessment

Since 1997, KAERI has worked on concept development for permanent disposal of HLW and its total system performance assessment. The FEP encyclopedia is actively developed to include all potential FEPs suitable for Korean geological and sociological conditions. The FEPs are prioritized and then categorized to the intermediate level FEP groups. For each specific scenario, the assessment contexts and associated assessment method flow charts are developed. All information on these studies is recorded into the web-based program, FEAS (FEP to Assessment through Scenarios.)

KAERI applies three basic programs for the postclosure radionuclide transport calculations: MASCOT-K, AMBER, and the new MDPSA being developed. During 2002–2004, a probabilistic safety assessment for reference scenario, radionuclide release through groundwater flow, and a few alternative scenarios (such as the effect of EDZ, initially failed containers, and potential climate change) has been performed. In 2004, the AMBER, based on a compartment theory, was applied to reassess the reference scenario. The new MDPSA code, which uses a control volume method, is under development for the probabilistic assessment of radionuclides in multidimensional regions, both of a fracture network and of a porous medium (such as top soil and buffer and backfill layers).

Preliminary assessment results show that under the given safety conditions, the KRS satisfies the guidelines given by KINS, the sole regulators in Korea. Results clearly indicate that future R&D in data acquisition should be focused on understanding the major water conducting features (MWCF).

13.2.4. Geoscientific Environmental Research

Korea is located on a stable platform subjected to major periods of significant tectonic movement between about 180 and 100 million years ago. The subsequent tectonic activities have diminished and have been limited to particular, localized areas (KAERI, 2002).

The lithology of the Korean peninsula has a complex structure, with about 29 rock types from Archean to Quaternary. In the southern part of the peninsula, Mesozoic plutonic rock is distributed over nearly one-third of the area. The wide distribution of plutonic rock is an important consideration, because the abundance of this rock provides flexibility in the siting process. Jurassic and Cretaceous granites will be preferable as competent host rock for the development of an HLW repository in Korea.

The geological environment of Korea has been summarized in this document, with emphasis on the plutonic rock mass, as part of the research program for HLW disposal. In a deep repository for radioactive waste, the geological formation plays an important role as a barrier system The long-term objective of the geological research work as part of an HLW disposal program in Korea is to develop the evaluation technology for a deep geologic environment. Common geological parameters suitable for the repository would be assessed using a multistaged and scale-dependent approach. The regional geologic conditions of the nation have been reviewed with emphasis on the fracture system. In addition, the hydrogeologic and geochemical characteristics, as well as mechanical properties, have also been evaluated from the results of a deep drilling program conducted in a Jurassic plutonic rock mass on the Korean peninsula.

The main task at this stage of the evaluation is to characterize the representative near-field conditions of the deep geological environment for the Korean Reference HLW Disposal System. It is also planned to define and characterize the geologic conditions, including rock-fracture systems and the mechanical properties of plutonic rocks, as well as the hydrogeological and hydrogeochemical characteristics of groundwater at a

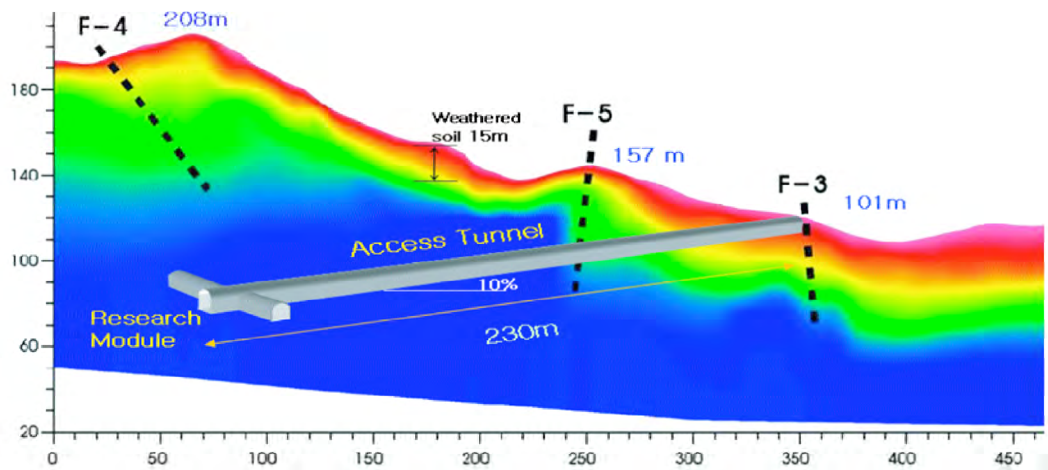


Figure 13.7. Schematic diagram of the first stage of Korean URT

prospective site. At some Jurassic plutonic rock mass sites, we have carried out a more detailed study on rock blocks, along with a deep drilling program.

13.2.5. Performance Validation for the HLW Disposal System

The major research topics of the validation study being carried out by KAERI are the construction of small-scale underground research tunnels, the behavior of the engineered barrier system (EBS), and radionuclide migration in fractured granitic rock. A small-scale Underground Research Tunnel (URT) is planned at the KAERI site (Figure 13.7). To investigate the geological and topological characteristics of the site, surface survey and borehole drilling had been carried out (Cho et al., 2004a). The first stage of the URT consists of an access tunnel and two research modules (Cho et al., 2004b). The access tunnel is planned to be linear, with a slope of -10% and length of 230 m. Research modules will be located on each side of and at the end of the access tunnel, with lengths and slopes of 25 m and +2%, respectively. The operation of the URT will be started in 2006.

For investigating the thermal-hydrological-mechanical behavior of the engineered EBS, a semi-engineered scale experimental facility is under installation. The facility will consist of four major components: a confining cylinder with hydration tank, a bentonite block, a heating system, and sensors/instruments. Results from preliminary calculations, using ABAQUS Version 5.8, showed that the temperature and maximum principal stress in the bentonite blocks reach near steady state after more than 20 days, although a longer time is required with increasing distance from the heater (Lee et al., 2004).

Experiments on contaminant migration through a natural fracture in granite were carried out to better understand contaminant transport in the deep underground environment. The scale of rock fracture was 100(L) x 60(W) x 25(H) (cm). To observe the migration characteristics, we used four kinds of tracers: (1) tritiated water (THO), (2) high-molecular organic dyes (Eosine and NaLS), (3) anions (bromide and chloride), and (4) sorbing tracers (strontium, cobalt, and copper). A particle-tracking method was applied to simulate contaminant transport through the heterogeneous flow field in the rock fracture (C.K. Park et al., 2002). By simulating realistic *in situ* geochemical conditions, migration experiments were also carried out in the glove box under reducing conditions to validate migration processes of actinides and to verify the migration models for chemical species in granitic rock fractures.

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Chapter 14

The Latvian Approach to Final Disposal of Long-Lived Radioactive Waste

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14.1. INTRODUCTION

Safe, final disposal of long-lived radioactive waste (RW) is one of the main priorities of RW management policy in Latvia. The RW management concept, accepted by the Cabinet of Latvia in 2003, identifies the key problems of RW safe management. In addressing solutions to these problems, it is emphasized that safe isolation of long-lived RW can be achieved only by disposal in a geological repository, in compliance with international recommendations and documents (Amended Proposal for Council Directive, 2004), which state that:

1. On the basis of present knowledge, geological disposal represents the most appropriate solution for long-term management of long-lived RW.
2. Member States shall study the possibility of obtaining a solution for deep geological disposal, taking due account of the specific circumstances.

As a supplementary measure to be discussed below, there is also the possibility of participating in regional waste disposal, in accordance with recent international activities and recommendations on regional disposal of long-lived and high-level RW.

The need to address the safe disposal problem in Latvia has arisen from the specific inventory present in an existing near-surface RW repository, as well as from recommendations in a recent safety analysis to remove used sealed sources from the operating vault before closure, and ensure their safe storage while remaining accessible for the final disposal program.

14.2. RW MANAGEMENT IN LATVIA

14.2.1. The Main Features of RW Management:

(a) Latvia has a single national RW near-surface repository of the RADON-type, developed for low and intermediate level short-lived waste (LILW-SL). In compliance with international recommendations, for the operational vault of a repository, currently only long-term storage of conditioned radioactive waste are licensed.

(b) A license for disposal will be issued after additional safety assessments and some safety enhancements (based on recommendations derived from a long-term safety analysis performed by an international consortium (Cassiopee, 2001)).

(c) Preparatory work is in progress for the Salaspils Research Reactor (SRR), in operation until 1998, to be dismantled and decommissioned. SRR systems will be switched off and dismantled, and equipment no longer being needed for maintenance of the SRR will also be switched off and dismantled.

(d) The State Agency "BAPA" (Hazardous Waste Management Agency) is the sole organization in Latvia for dealing with RW management, ensuring a unified administration and maintenance of the SRR shutdown, which involves spent nuclear fuel (SNF) storage. The whole cycle of RW management (collection at the site of origin, processing procedures, and storage/disposal), is fully financed by the State, which receives supplementary funding from an import tax on radioactive substances.

14.2.2. RW Management Concept

According to the principle cited in the Joint Convention on the Safety of Spent Fuel Management and on the Safety of RW Management of 2003, "The timely creation of an effective national legal and associated organizational structure provides the basis for appropriate management of RW." Latvia has accepted the RW Management Concept based on the recently established national, legal, and organizational policy and infrastructure of RW management.

The main problems to be solved with this RW Management Concept (Dreimanis and Shatrovska, 2004) are as follows:

1. The increase in the amount of RW and lack of sufficient storage space. In addition to the traditional producers of RW, such as medicine, industry, and science, there is a specific major operation in Latvia that has significantly increased its annual amount of RW in recent years and will continue to do so in the future. This situation results from the planned disposal of most of the SRR dismantling and decommissioning waste into the existing near-surface LILW-SL RW repository (in Baldone), which needs to be enlarged to accommodate this additional waste.

2. The safety of the existing repository. To ensure protection of the public and the environment from exposure and radioactive contamination, proper measures are to be taken. The safety of the RW repository should be upgraded by building a long-term cover, in accordance with the Cassiopee recommendations.

3. Construction of long-term storage in a geological disposal site. Safe isolation of long-lived RW can only be achieved by storage in a geological disposal site, thereby:
   - Supporting the interests of future generations by using standards that ensure minimal risk in the far future, and simultaneously minimizing the transferred responsibility for safe management of RW
   - Ensuring satisfaction for the current public interest by implementing the latest developments in repository construction
   - Not prescribing the necessity to allow removal of waste packages from the repository

4. Strategic aims of foreseen solutions.
   - To build a long-term storage facility for used sealed sources that can't be stored in near-surface repositories
   - To investigate the possibilities for constructing a geological repository

5. Risks that RW management poses to the public. The need to find a compromise between different parts of society may make it necessary to implement a compensation mechanism for hypothetical risks and the mitigation of negative perceptions by the public concerning activities with RW—perceptions that seriously hinder projections for new storage space. The introduction of an efficient compensation mechanism could influence opinion polls to be conducted before planning new facilities.

6. The competence level of the employees involved in RW management. To promote education and scientific development in the area of RW management, one should actively support various international projects and disseminate the results of such projects, as well as support the participation of employees in international forums.

14.2.3. Production of RW in Latvia

The use of radioactive and nuclear materials in Latvia is relatively minor. In the last decade, approximately 3–5 m^3 of unconditioned institutional RW was generated for long-term storage or disposal, along with 500–5,000 different sealed sources of ionizing radiation. In the process of conditioning and preparing this material for disposal, the gross volume of RW has increased by a factor of two. The total activity of such annually produced RW now reaches ~1-10 TBq.

In addition to the traditional institutional RW producers in medicine, industry, research and other areas, the D&D process at SRR has significantly increased the annual amount of RW to be disposed of in this repository. The projected annual volume of radioactive waste arising from the D&D of SRR is ~1,200 m^3, the vast majority of which is to be disposed of as LILW-SL. Only a relatively small part of this waste is of intermediate activity, and this waste will be handled through geological disposal. The SNF is projected to be moved out of Latvia, in the framework of the USA-IAEA-Russia cooperation project and the proposed Russia-Latvia governmental agreement on SNF management.

14.2.4. The Main Features of the Existing Near-Surface RW Repository

The single national radioactive waste near-surface repository is located near Baldone, about 27 km southeast of Riga, and is projected to receive LILW-SL waste. The total capacity of the repository is 1,880 m^3; currently, the total amount of RW in storage is slightly below 1,000 m^3.

Although current Latvian regulations encourage re-exportation of used sealed sources (USS), a considerable amount of USS with very different half-lives has already been disposed of in the existing vaults. The Baldone site, a permanent disposal site managed by BAPA, has received USS from approximately 300 Latvian institutions, as well as from the Kaliningrad region. On the whole, USS accounts for 80% of the radioactivity in the repository. At Baldone, six of the vaults used during the period 1962–92 have been closed, but the last vault (No. 7) is operating as a long-term storage site. The waste deemed unsuitable for near-surface disposal is to be stored at No. 7 until a deep geological disposal option is available. This could be either within the country, or outside, if a regional approach should be approved.

Geologically, the Baldone site is characterized by quaternary deposits of interbedded lacustrine, fluvioglacial and glacial sediments of sands and clay, and overlying dolomite and gypsum layers. No surface geological processes, such as erosion, landslides, or weathering, are known to have occurred at the disposal site or in the near vicinity. However, during its operation, there was some evidence of karst conditions that resulted in collapses at the land surface.

14.3. GEOLOGICAL AND NORMATIVE BASIS FOR POSSIBLE DEEP UNDERGROUND DISPOSAL OF RW IN LATVIA

14.3.1. Provision for Geological Disposal of Long-Lived RW—Strategic Aim of National RW Management Concept

In compliance with the International Atomic Energy Agency (IAEA) generic safety principles for RW management, the following are of particular importance:

Principle 4: "Protection of future generation"
Principle 5: "Burdens on future generation" ("Radioactive waste should be managed in such a way that will not impose undue burdens on future generations")
Principle 9: "Safety of facilities"

Latvia is planning to minimize the impact and burden on future generations, via:

1. Institutional measures in the post-closure period of the repository
2. Considering storage of long-lived RW at a geological site as the sole option that would minimize risk in the distant future, as well as decrease the transferred responsibility for the safe management of RW

Safe isolation of long-lived RW from the biosphere can be achieved by storage in the geological disposal site only, because the near-surface repository has a greater probability of unintentional destruction, which may result in harm to the environment and the public (IAEA, 2003a). Besides, geological repositories have the greatest potential for ensuring the highest level of waste isolation and are applicable to the disposal of the most demanding RW categories: high-level waste, SNF and other long-lived RW (IAEA, 2003b). The near-surface repository requires state supervision during the entire period of danger from RW, which for long-lived isotopes constitutes a much longer time than the possible existence of a particular state.

The disposal of long-lived RW in a geological repository would:

(a) Support the interests of present and future generations by using those standards that guarantee the minimum of risk at present as well as in the distant future, and simultaneously minimize the transferred responsibility for the safe management of RW.
(b) Ensure fair satisfaction for the current interests of the public, by implementing the newest scientific and technical methods in repository construction and by consultation with all interested parties throughout all phases of the process.

To achieve the main strategic aims while developing solutions to these problems, Latvia is planning:

(a) To construct long-term storage for used sealed sources and long-lived waste, which must be stored pending the availability of deep geological disposal. A newly recommended strategy has been accepted to build a dedicated long-term storage facility for used (spent) sealed sources. This involves the European Union (EU) PHARE ("Poland and Hungary Assistance for Reconstruction of Economics") project, Design of Additional Waste Disposal Vault and Integral Storage Facility for Long-Lived Waste (LE 01.09.01). Long term storage will be used for the RW types that are not suitable for near-surface disposal.
(b) To investigate the potential possibilities for constructing a geological repository in Latvia and, in the framework of a dual-track option, to develop international activities to evaluate the possibilities of regional disposal options.

14.3.2. Planning of Feasibility Studies for Geological Disposal in Latvia

Taking into account Japan's experience with these issues (NUMO, 2002), a national program of feasibility studies on geological disposal would require Latvia to:

(a) Formulate basic criteria and requirements for the probable arrangement of a geological facility.
(b) Select a group of possible candidate sites comply ing with these requirements.
(c) Make an Environmental Impact Assessment (EIA) for each site.
(d) For each site with a favorable EIA, make detailed feasibility studies and site investigations for deep geological disposal.
(e) Draw conclusions about how well the physical properties of the geologic layers at each site correspond to the recommended international standards for barrier layers at a geological repository.

In the process of conducting an EIA at each of the proposed sites, it will be necessary to follow European Union guidelines (European Union, 1999), including the siting guidelines, which would take account of:

1. Geological factors (including initial factors to determine whether volunteer sites, where applicable, should be included in the site characterization process)
2. Environmental protection factors (with requirements relating to population density and requirements relating to land use and biological diversity)
3. Areas of specific scientific and cultural interest
4. Utilization of natural resources, including water and soil
5. Requirements relating to costs
6. Transport considerations
7. Guidelines for social acceptability

In addition, potential mining sites (in particular, anticipated areas of oil output), underground gas-storage sites, and other economically important areas should also be excluded from consideration as eventual disposal sites (with suitable geological structure). The Latvian hydrocarbon prospects are associated with Western Latvia and the adjacent part of the Baltic Sea shelf, i.e., the northern part of the Baltic Syneclise.

14.3.3. Geological Conditions of Latvia Territory

The geological structure of Latvia is mainly formed by the old East-European Platform, taking into account that Latvia is a part of the so-called Baltic Syneclise and the Latvian Saddle, located on the northwestern edge of the East European Platform. The Baltic Syneclise is a sedimentary basin with a thick succession of sediments and is regarded as highly prospective, being part of a proven Baltic petroliferous province. Approximately 35 petroleum deposits have been discovered there, and most of them are oil accumulations.

The geological section of Latvia contains two components characteristic of palaeoplatforms: the crystalline basement and a sedimentary cover. The most important features within the territory of Latvia are: (1) subsidence at the top of the crystalline basement (mainly Archaic and Proterozoic crystalline slates, gneisses, and granites), which slopes from the northeast, at a relatively shallow depth of ~400 m, towards the south and southwest at a depth of ~1,800 m; and (2) the sedimentary cover, which unconformably overlies the heavily eroded surface of the crystalline basement, and consists of terrigenous, carbonate, and sulphate rocks of Proterozoic and Phanerozoic-Cambrian through Jurassic ages.

In 2004, geological mapping of the entire territory of Latvia, at a scale 1:200,000, was accomplished by the State Geological Service of Latvia. These maps contain information about the rock, the crystalline basin with its sedimentary cover, the tectonics, and the relief and modern geological processes that will be used in selecting areas for preliminary investigation and subsequent detailed investigation.

Although Latvia is located within the seismically low-active East-European Platform, over the last few centuries, seismic events reaching 5-6 Richter force have been recorded. On the basis of regular seismic monitoring and processing of domestic and international (NORSAR) data, some zones have been identified as epicenters of concentrated seismic events.

14.3.4. Waste Acceptance Criteria (WAC) And Safety Assessment For Geological Disposal

In parallel with the decision to carry out feasibility studies for deep geological disposal in Latvia, the Latvian Cabinet has also specified in detail the RW groups to be disposed of in a geological repository, as well as the main requirements and parameters for long-term safety assessment (SA) of a geological repository. These requirements and parameters are included in Cabinet Regulation No. 129 (2002), "Requirements for the Practices with Radioactive Waste and Related Materials." This regulation states that the wastes to be disposed of in a geological repository are:

1. RW containing radionuclides with half-life >30 years
2. High-activity RW
3. RW with activity values not complying with WAC for packages and vaults originally planned for near-surface disposal
4. LILW-SL RW with radionuclides whose activity, after the state supervision period, will exceed the limits prescribed by "Cabinet Regulations on Licensing" for determining the necessity to receive a license for work with such radionuclides

Regarding the requirements for the SA of a geological repository, Regulation No.129 states that, in particular:

- The time period of the SA analysis is at least 10,000 years
- The seismic parameters to be analyzed:
  - (a) the maximum magnitude of earthquake—5.4 Richter force
  - (b) the earthquake probability—once in 400 years
  - (c) horizontal acceleration—4 m/s^2
  - (d) vertical acceleration—1.2 m/s^2
  - (e) vertical movement—2.0 cm.

14.4. INTERNATIONAL ACTIVITIES IN PROVIDING SAFE FINAL DISPOSAL OF RADIOACTIVE WASTE

In the last decade, the importance of international disposal options has been developed because of the dual-track enhancement of global security and environmental safety (McCombie et al., 2001). In view of this dual-track option, Latvia, in parallel with its feasibility studies on geological disposal, is considering the option of regional disposal.

14.4.1. Recent Development of Regional Disposal Concept—Basic Organizational Features

At the present time, there are three basic scenarios for the possible preparation and implementation of a shared disposal facility (Štefula, 2004; IAEA, 2004):

1. The Add-on scenario, in which the host country offers to supplement its internal inventory of wastes for disposal with wastes imported from other countries.
2. The Cooperation scenario, characterized by the participation of other (partner) countries in developing a repository program jointly with a potential hosting country or countries. In this case, one or more

other countries interested in disposing their waste(s) in the potential hosting country or countries will be involved directly, at an early stage of the repository development and implementation.
3. The International scenario, in which a higher level of control and supervision is implemented by an international body.

The European regional project SAPIERR (Support Action: Pilot Initiative for European Regional Repositories) was started in 2004, with the aim of helping the European Commission to begin to establish the boundaries of the European regional repositories issue, to collate and integrate information to allow relevant concepts for potential regional options to be identified, and to gauge any new research and technical development (RTD) needs (Štefula, 2004).

The primary objective of SAPIERR is to bring together Member States of the current and extended European Union who wish to explore the feasibility of regional European solutions for deep geological disposal. Specific proposals for regional facilities, including potential siting, were deliberately left out of the scope of this initial pilot study.

The SAPIERR project, included in the EU FP6 Programme, has the following main stages and objectives:

- National data gathering analysis—to establish the RW inventory for the regional disposal option and to document the current legal framework for such an option
- Elaboration, by the working group, of scenarios and RTD requirements—to define the possible options open to interested countries for regional storage and disposal solutions, as well as scenarios under which such options might be considered; to identify the transnational RTD requirements that must be carried out in the future
- Information dissemination—to disseminate, widely, the objectives and the findings of the study, in a freely accessible form- to determine further management tasks

14.4.2. Latvian Activities Towards International Cooperation in Regional Disposal Options

With the objective of evaluating the possibilities for using the regional disposal option, Latvia has developed the following activities:

1. Implemented in its legislation (The Regulations of the Cabinet of Ministers No.129 (19.03.2002) "Requirements for the Practices with Radioactive Waste and Related Materials") the Council Directive 92/3/Euratom of 3 February 1992 on the supervision and control of shipments of radioactive waste between member states and into and out of the community.
2. Adopted regulations of the Cabinet of Ministers No.157 of April 16, 2002 "The Principles of Determination of the Equivalence of Various Radioactive Wastes," which are intended to manage the possible exchange of RW between relevant states, should the regional disposal option be implemented.
3. Submitted data on RW inventory to the SAPIERR project manager, specifying an approximate volume of 200 m^3 to be disposed of as long-lived RW, as well as the existing legislative basis for RW management, and regional/international RW disposal issues.
4. Joined ARIUS (Association for Regional and International Underground Storage) in 2004, to promote the most relevant solutions for national as well as regional/international geological disposal options for long-lived RW. This is an international association whose objective is "to promote concepts for socially acceptable international and regional solutions for environmentally safe, secure, and economical storage and disposal of long-lived radioactive wastes."

14.5. CONCLUSIONS

Latvia recognizes the ultimate importance of providing safe disposal for long-lived radioactive waste. Given this awareness of the ethical obligation to ensure protection for future generations without imposing an undue burden on them, Latvia has commenced dual-track activities in seeking possibilities for geological disposal of relevant waste categories.

The importance of studying deep-geological-disposal options also follows from the problem of public response (and acceptance of) basic RW management tasks in Latvia. To obtain public acceptance of any plans for enlarging the existing near-surface repository, the public shall get guaranties that long-lived RW will be stored there only for the shortest possible period, and thereafter moved to a deep geological repository/regional repository.

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Chapter 15

Lithuania's Approach to Disposal of Radioactive Waste and Spent Nuclear Fuel

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15.1. INTRODUCTION

There is only one nuclear power plant (NPP) in Lithuania—the Ignalina NPP. It is situated in northeast Lithuania, near the borders of Latvia and Belarus, on the bank of the largest Lithuanian body of water, Druksiai Lake. The Ignalina NPP possesses two similar RBMK-1500 reactor units. RBMK-1500 is the last and most advanced version of the RBMK-type reactor design series (actually only two units were constructed). The Ignalina NPP reactors were commissioned in December 1983 and August 1987, respectively. Original design lifetime for these reactors was projected to be 2010–2015.

The Ignalina NPP is a vital component in Lithuania's energy balance because it is producing more than 70% of the total electricity in Lithuania. There are a variety of reasons for this high percentage, but the main one is that nuclear power has a significantly lower production cost than other forms of power under the present economic and technical circumstances in Lithuania. (Note that after the nuclear accident in Chernobyl, safety systems at the Ignalina NPP were re-evaluated, and it was decided to decrease the maximum thermal power of the units from 4,800 to 4,200 MW. This limits the maximum electric power to about 1,250 MW per unit.) On October 5, 1999, Seimas (Parliament of Lithuania) approved the National Energy Strategy, in which it was stated that the first unit of the Ignalina NPP would be shut down before the year 2005, taking into consideration substantial long-term financial assistance from the European Union (EU), G7, and other states, as well as international institutions. On October 10, 2002, Seimas approved an updated National Energy Strategy, in which it was stated that the first unit would be shut down before the year 2005 and second unit by 2009, if funding for decommissioning were available from EU and other donors. Following this, on December 31, 2005, Unit 1 at Ignalina was finally shut down. There are no other nuclear facilities in Lithuania.

The RBMK-1500 reactors were designed for nuclear fuel with 2% enrichment of ^{235}U. Use of the new uranium-erbium fuel at Ignalina NPP started in 1995, and at the end of 2001, the first batch of nuclear fuel with 2.6% enrichment was loaded into the reactor. Introduction of fuel with 2.8% enrichment is under consideration. It is estimated that about 22,000 spent nuclear fuel (SNF) assemblies (about 2,436 tons of uranium) will have accumulated at the Ignalina NPP by 2010.

15.2. LOW-AND INTERMEDIATE-LEVEL WASTE

15.2.1. Current Situation and Plans for Conditioning and Interim Storage

The Ignalina NPP produces about 2,000 m^3 of radioactive waste (except SNF) annually: 1,100 m^3 of solid waste, 700 m^3 of bitumenized waste, and 200 m^3 of spent resins (Poskas and Adomaitis, 1997; 2001). Treatment methods applied at the plant are as follows: compaction (baling) of the solid low-level combustible waste with a low-pressure compactor, evaporation of liquid waste, and bituminization of evaporator bottoms. Implementation of the project for cementation of ion-exchange resins and perlite mixtures was started in 2001 and will be finalized in 2005. A project for modernizing the entire management system for solid short-lived low/intermediate level waste (LILW-SL) and long-lived intermediate level waste (ILW-LL) is in the final stage of the tendering process, and implementation will be started in 2005.

The complex for solid radioactive waste storage at the Ignalina NPP consists of four buildings (155, 155/1, 157, and 157/1) with auxiliary systems and equipment for the operations in each building. Buildings 155 and 155/1 are intended for storage of LILW-SL, and they are assembled monolithic buildings. Buildings 157 and 157/1 are intended for storage of LILW, and they are ground facilities of concrete with the surface reserved for expansion. Buildings 157 and 157/1 are separated into sections by solid partitions, with each section storing a specified group of wastes. A safety analysis report on these facilities was prepared during 1999–2000, and the regulatory authorities subsequently issued a license for its operation. Wastes from research, medicine, and industry are transported to the NPP and are also stored in these buildings. Liquid waste is collected in large concrete tanks, for temporary storage, before being evaporated and bituminized. Ion-exchange resins and perlite mixtures are also stored in the tanks, awaiting commissioning of the cementation facility.

15.2.2. Activities Related to the Disposal of LILW-SL

There are two old radioactive waste disposal facilities in Lithuania. Of Russian design, both of them could be classified as near-surface disposal facilities.

15.2.2.1. Maisiagala "Radon" Type Disposal Facility for Institutional Waste

This repository is designed for institutional waste and is a typical "Radon" type facility constructed in the early 1960s (1964 in Lithuania) and used throughout the former Soviet Union. It was closed in 1989. Waste at this site was disposed of in a reinforced concrete vault with internal dimensions 14.75 x 4.75 x 3 m (200 m^3 in volume). The vault was only partially filled with waste (about 60%) during its operation. When loaded, the waste was interlayered with concrete. In addition, sealed sources were disposed of in two stainless steel containers, each with a volume of 10 liters. Medical sources were disposed of with a biological shielding. At the end of the disposal period, the residual volume was filled with concrete and sand.

The disposal site is constructed of glacial sediments on the Baltic highlands, on a hill of medium relief in the catchment's area of the Neris River. A thick layer of sediments, up to 1,000 m, is prevalent in the repository area. The upper 100 m consists of sandy loams and clay loams from the Quaternary period. The disposal site is on the levelled top of a hill consisting of sand and gravel. Height above sea level is between 140 and 150 m, and the height of the surrounding hills is around 150 m. Most hills are elongated, with steep slopes and flat tops, and consist of moraine subclay and subsand or sand/gravel.

A preliminary safety assessment of this facility was performed by SKB (Sweden) with participation of the Lithuanian Energy Institute (Poskas et al., 2000). The Radwaste Management Agency (RATA) has applied for a PHARE project (2004–2006), which is aimed at safety assessment and upgrading of the Maisiagala repository. We plan to upgrade the physical protection system and collect the data needed for safety analysis (summarize the waste inventory, make necessary ground drillings for sampling, etc.). The project will also include the preparation of a Safety Analysis Report, a conceptual and detailed design of the facility upgrading, and documentation for work, supply tenders, and repository licensing.

15.2.2.2. Storage/Disposal Facility for Bitumenized Waste at Ignalina NPP site

A storage/disposal facility exists at Ignalina NPP for bituminized evaporator bottoms of liquid radioactive waste generated there. This is an aboveground two-story assembled concrete monolith (Building 158) consisting of 12 steel-lined vaults for loading the bitumen compound. At present, five vaults are filled with this compound.

This facility is located at the Ignalina NPP site, near the East-European platform, at the juncture of two large structural elements (the Baltic syncline and the Mazur-Belarus anticline). The granitic bedrock is separated from the sediment by a series of tectonic breaks, the dimensions of which are approximately 2 x 2 km. In the area of the Ignalina NPP, the surface sediments are quite heterogeneous. They were formed during the retreat of the last glacial period as a result of different glacial and water-glacial processes. Later, alluvial, marsh, and lake sediments were formed. The lithological structure, filtration, and geological engineering properties of the separate genetic types of surface sediments are not equal. Most prevalent are the permeable water-glacial sediments located in direct proximity to Lake Druksiai and the Ignalina NPP. All the surface sediments contain subsoil water. Supporting weight is provided by the marsh, lake-marsh, lake-glacial, and water-glacial sediments, all located near the surface. Lithologically, these sediments are composed of peat, sand, gravel, sandy soil, sandy loam, and clay. The depth of the aeration zone is from 1–2 m to 5–8 m, with the subsoil composed of fine sands and sandy loam.

A preliminary long-term safety assessment of this facility was also performed by SKB (Sweden) with the participation of the Lithuanian Energy Institute (Poskas et al., 2000). This assessment led to the conclusion that this radioactive waste facility could be converted into a disposal facility if a multilayer earth cover were used, but a more detailed analysis is necessary. At present, this facility is licensed as a storage facility, based on the Safety Analysis Report prepared in 1999–2000.

15.2.2.3. New Near-Surface Disposal Facilities

Activities related to the implementation of a landfill repository for very-low-level waste at the Ignalina NPP site were started in 2003. A reference design, recommendations on site selection, and preliminary waste-acceptance criteria were prepared in 2003. The siting and an Environmental Impact Assessment (EIA) are planned for 2005.

The project dealing with a reference design for a near-surface repository for low- and intermediate-level short-lived waste in Lithuania was performed in 2000–2002. A consortium consisting of SKB-SWECO International and Westinghouse Atom (Sweden), with participation of the Lithuanian Energy institute and some other Swedish organizations, suggested a robust and simple design for this near-surface facility in Lithuania. The design is modular and highly flexible with regard to its barrier design and overall geometry. It can easily be adapted to various site conditions, and the cell groups can easily be redesigned for different waste packaging. The reference environment and the assumed site conditions are such that the siting of the repository is flexible. It is possible to establish, operate, and close the repository with acceptable long-term safety characteristics at a variety of sites available in Lithuania.

Siting of a near-surface repository was started in 2003. The Lithuanian Energy Institute prepared the siting criteria that were used by the Lithuanian Geological Survey and the Institute of Geology and Geography in selecting two sites for the repository close to the Ignalina NPP. In 2004, the Environmental Impact Assessment Program was prepared and approved by the Lithuanian government. An Environmental Impact Assessment Report was also prepared by the consortium lead of the Lithuanian Energy Institute. Relevant authorities approved the report, and at the moment, it is under review by the Regulator (Ministry of Environment).

15.3. SPENT NUCLEAR FUEL AND LONG-LIVED WASTE

15.3.1. Interim Storage of SNF and Long-Lived Waste

According to Lithuania's Law on Nuclear Energy, SNF is a radioactive waste, so it will not be reprocessed. In 1992, the decision was made to build an interim dry SNF storage facility at the Ignalina NPP site with a lifetime of about 50 years. After the tendering process, GNB (Gesellschaft für Nuclear Behalter) casks were chosen. Twenty of these casks are ductile cast iron CASTOR RBMK-1500 casks; the remaining ones are metal-concrete CONSTOR RBMK-1500 casks. Both types of casks are designed to load 102 half-assemblies, which are arranged in the basket in a special configuration. In March 2000, INPP received its operational license, and regular loading of the CASTOR RBMK-1500 casks was started. The last CASTOR cask was loaded and transported to the storage site on September 20, 2000. At the same time, the licensing procedure for CONSTOR RB MK-1500 casks was under way, and regular loading of these CONSTOR casks started at the end of June 2001. The capacity of the existing SNF dry-storage facility at Ignalina NPP is for 80 casks.

It is estimated that after the final shutdown of Unit 2 in 2009, the total amount of SNF will be approximately 22,000 fuel assemblies, with 4,080 of them accommodated for within the existing facility, and the new interim storage facility accommodating the remaining more-than-18,000 SNF assemblies. After the tendering process in January 2005, Consortia GNB mbH (GNS), RWE NUKEM GmbH, Germany was awarded the contract. The Lithuanian Energy Institute, as local subcontractor, is supporting the Consortia in an Environmental Impact Assessment Program and Report and also Preliminary and Final Safety Analysis Reports. The SNF will be loaded in the higher-capacity (182 fuel half-assemblies) CONSTOR RBMK-1500 casks.

As indicated above, the long-lived intermediate level waste is also stored in big vaults within concrete structures (Building 157) at the Ignalina site. It is planned, within the solid radwaste management modernization project, that this waste will also be retrieved, loaded into containers and, after proper characterization, transferred into the new facility for an interim storage of at least 50 years.

15.3.2. Activities Related to the Disposal of Spent Nuclear Fuel and Long-Lived Waste

15.3.2.1. Strategy for disposal

The Strategy on Management of Radioactive Waste in Lithuania, approved by the Lithuanian government in 2002, also defines a number of activities related to SNF disposal:

- Draft and implement the long-term research program, "Possibilities to dispose of spent nuclear fuel and long-lived radioactive waste in Lithuania."
- Analyze the possibilities of having a deep geological repository in Lithuania for SNF and long-lived radioactive waste.
- Analyze the possibilities of creating a regional repository from the joint efforts of a few countries.
- Analyze the possibilities of disposing of SNF in other countries, and determining the cost and justification for such disposal.
- Analyze the possibilities of prolonging the storage period for interim storage facilities for up to 100 or more years.

The Program for Assessment of Possibilities for Disposal of Spent Nuclear Fuel and Long-Lived Radioactive Waste for the Years 2003–2007 was prepared and approved in 2003. In parallel, the Swedish Ministry of Foreign Affairs allocated special funding to support activities in Lithuania related to the closure of the Ignalina NPP. One of the subject areas identified was the development of national competence on issues related to disposal of SNF.

15.3.2.2. Geological and hydrogeological investigations

On the basis of some earlier investigations of the Lithuania's geological structure, several geological formations were identified as potentially suitable for a deep repository for SNF (Kanopiene and Marcinkevicius, 2000): rocks of the crystalline basement, Lower Cambrian clay, Permian sulfate deposits (anhydrite), Permian rock-salt, and Lower Triassic clay.

During 2001-2004, more detailed analyses of these geological formations have been performed by the Geological Survey of Lithuania and the Institute of Geology and Geography with support of Swedish experts (Suitability of Geological Environment, Investigations of Possibilities, Vol. 1, 2005). This work mostly involved desktop studies for the first stage (prospective or conceptual and planning stage investigations) of geological investigations related to the construction of a deep geological repository for SNF. The main part of this effort was concerned with the selection of criteria and evaluation of rock formations on the basis of archival data and published reports. The general overview of the geological structure and the composition of the sedimentary cover and crystalline basement were carried out in 2001–2002. This started the process of evaluating geological media to assess the territory of Lithuania in terms of its suitability for radioactive waste repositories. Several rock types—clayey formations, crystalline basement rocks, rock salt, and anhydrite formations—were selected for the studies. As a result of these investigations, four prospective clayey formations were selected in the sedimentary cover of Lithuania, from the Lower Cambrian, Lower Silurian, Middle Devonian, and Lower Triassic sequences. In addition, the Upper Permian anhydrite and rock salt were considered as prospective candidates for disposal of SNF.

A screening of the territory of Lithuania was performed based on the most important geological parameters describing the suitability of formations, such as: simple tectonic structure, absence of intraformation aquifers, low neotectonic and seismic activity, good isolation, and favorable mechanical properties of the rock. Then, the aerial distribution of these favorable formations was defined (Figure 16.1). The screening was also based on an evaluation of the general geological parameters—such as lithological homogeneity of the prospective layer, thickness, depth and lateral extent, and tectonic structure—of the candidate formation. It was concluded after these investigations that rock salt could not be regarded as having a high potential for disposal of SNF. However, the crystalline basement rocks were considered very good candidates for a geological repository. Among those, the best prospects appeared to be located in the southeastern part of Lithuania (Figure 16.1) where the rock is overlain by only 200-300 m of a thick sedimentary cover. The prospects for argillaceous clayey formations were reduced to only the Lower Cambrian Baltija Formation and the Lower Triassic, since they best fulfill the requirements with respect to depth, thickness, lithological composition, and homogeneity of sediments. Thus, the extended studies on media selection in 2001–2002 led to the conclusion that crystalline rock and argillaceous rocks are the primary candidates for a geological repository in Lithuania.

In 2003, activities were focused on more detailed studies of these geological media, because desktop studies are not sufficient for making a prioritization of these media. The direct observation and characterization of cores from reference wells were carried out for detailed lithofacies evaluation of candidate formations, analysis of similarities, and differences between the cores of the different lithofacies zones at different depths. At the same time, a sampling for a preliminary analysis of mechanical properties was carried out. After such investigations, the Lower Triassic formation was selected as a first priority, and the Cambrian Baltija Formation as a second priority—for investigations of the potential clayey formations. With regard to crystalline rock, it was determined that an area of 100 km^2 could be found between major fracture zones at an acceptable depth, and with a normal fracture content that fulfills the desired requirements both with respect

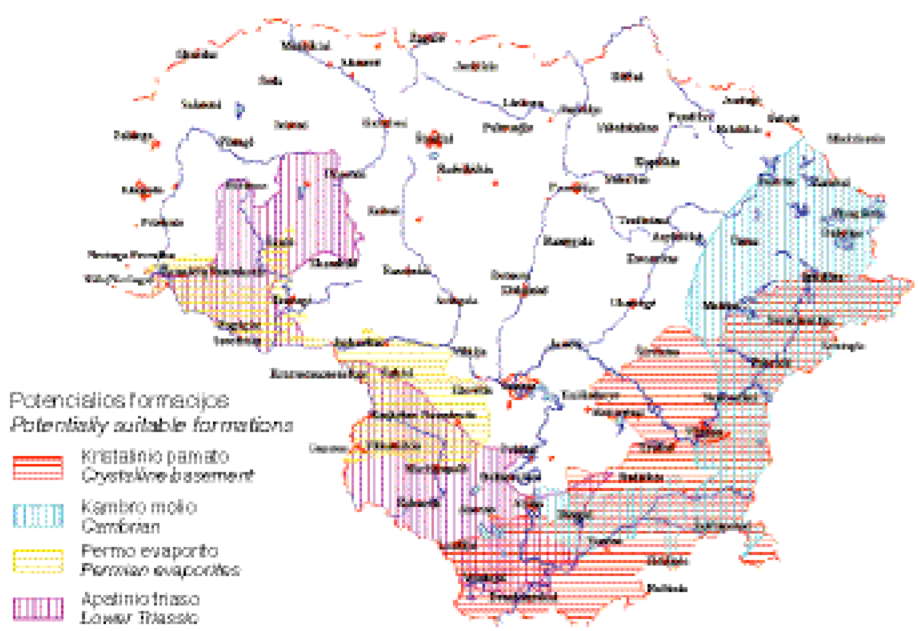

Figure 15.1. Occurrence of geological formations potentially suitable for disposal of spent nuclear fuel (Suitability of geological environment, Vol.1, 2005)

to tightness and stability. Furthermore, there is broad experience and competence in the field of investigations concerning SNF disposal in crystalline rocks in Sweden. Thus, crystalline rocks were selected for further characterization in 2004.

Laboratory investigations were performed on crystalline rocks to gather general information on the strength parameters of these rocks as a function of different petrological composition. Rock samples were taken from some old cores collected in boreholes drilled in southern Lithuania during deep geological mapping at a scale 1:200,000. Depths of the samples varied from 427 to 606 m and were selected as the optimal values for deep repository construction. The main characteristics of these crystalline rocks, such as density, porosity, unconfined compression strength, rock temperature, were collected from different literature sources. All the data were filed systematically, and a simple statistical analysis was performed. Tectonics and hydrology of the crystalline basement were analyzed using existing archival information.

Based on a four-year period of investigations, the following is a summary of the conclusions that have been reached:

- After compiling available geological data and inspecting selected drill cores, only the Lower Cambrian Baltija Group and the Lower Triassic clayey formations emerged as potential alternatives to crystalline rocks. They fulfill the requirements with respect to the depth, thickness, distribution, and lithological composition. The Lower Triassic has a higher priority than the Lower Cambrian
- The crystalline basement rock can be considered one of the best prospects for a geological repository. The best crystalline-basement prospects appear to be located in southeastern Lithuania, where these rocks are overlain by only 200–300 m of a sedimentary cover. Extensive geological information is available for southeastern Lithuania, and this makes it very attractive for future studies.
- A rather dense network of faults is located in southern Lithuania, yet large enough blocks, suitable for a geological repository, are defined. Nontectonic blocks of order 10 × 10 km can be found. The best rock-type prospects are represented by cratonic (anorogenic) granitoid intrusions that, in some places, have formed rather large massifs. These rocks are the least damaged by tectonic activity. Other rock types (gneisses, mafic intrusions, migmatites) have (in various places) formed only a weakly fractured block, which may also be a prospective repository site. Because of the very low seismic activity in this part of Lithuania, the tectonic stress is very low and should not affect tunnel construction. However, no instrumental observations of the stress field are available.
- Hydrogeological well tests indicate that the tectonic zones are water saturated, whereas the homogeneous blocks are waterproof. The salinity of the formation water does not exceed 30 mg/L (except in some rare anomalies), which is favorable for engineered barriers. The water flow field of the basement is not yet well understood.

15.3.2.3. Development of the disposal concept and safety assessment

During 2001–2004, the Lithuanian Energy Institute, with support of Swedish experts, was working on development of the repository concept (Concept of Repository in Crystalline Rocks, in Investigations of Possibilities, Vol. 2, 2005) and the generic safety assessment (Generic Safety Assessment, in Investigations of Possibilities, Vol. 3, 2005) of a repository in crystalline rock in Lithuania.

A detailed repository design is highly specific to waste type and its geological environment. But regardless of waste type, construction of the access and emplacement shafts and tunnels will involve the excavation of a substantial underground facility involving the removal of some hundreds of thousands of cubic meters of rock, to millions of cubic meters for larger waste disposal programs. Geological repositories presently being considered have underground dimensions varying from a few square kilometers to about twenty square kilometers, depending on the inventory of waste, its thermal output, and the repository design.

The proposed repository concept for Lithuania is based on the KBS-3 concept developed by SKB for disposal of SNF in Sweden. Specifically, the KBS-3H design, with horizontal canister emplacement, is proposed as the reference design for Lithuania. The scheme of the repository is shown in Figure 15.2.

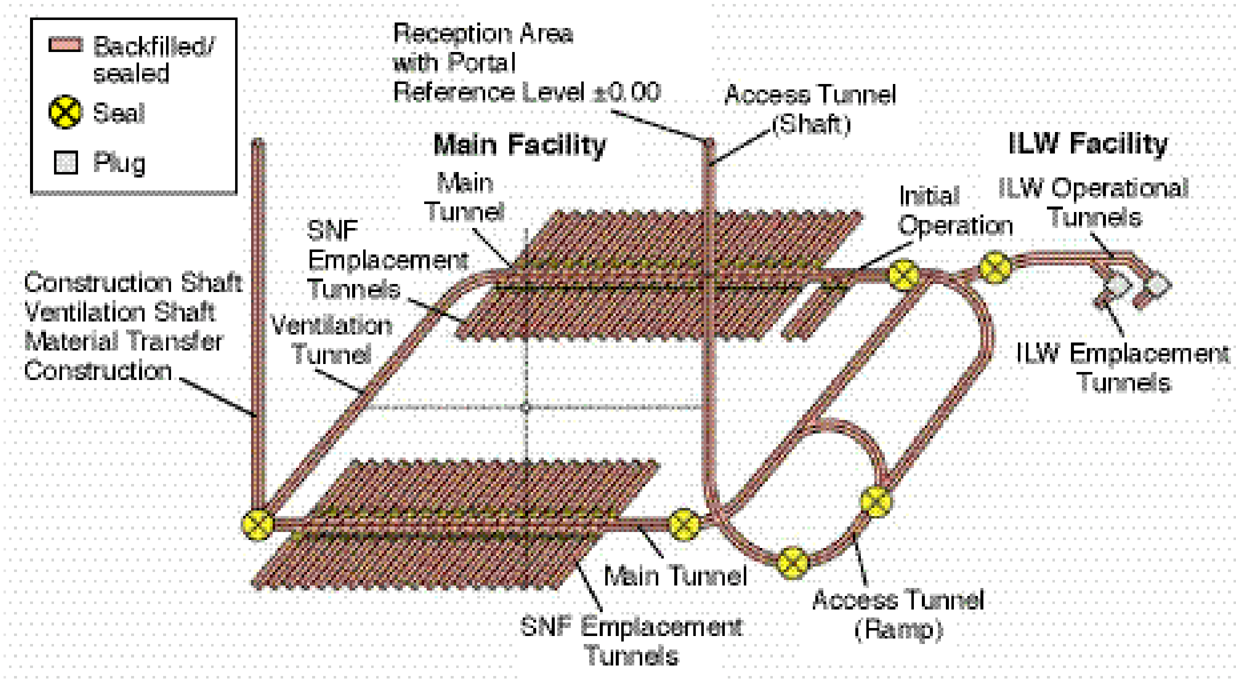

Figure 15.2. The scheme of the repository after final sealing and closure of the facility (Concept of Repository in Crystalline Rocks, in Investigations of Possibilities, Vol. 2, 2005)

The generic repository concept for SNF disposal in crystalline rock in Lithuania envisions the waste disposal in horizontal emplacement tunnels with diameters of 1.85 m and lengths of 250 m. The repository would be constructed in the crystalline basement at a depth of 300–500 m. Disposal canisters with RBMK-1500 SNF would be emplaced with a distance between canisters of 1.2 m. A copper canister is assumed for the disposal of SNF in the basement rock. The proposed canister for SNF is designed with two components: an outer corrosion protection of copper and a cast iron insert with channels for the RBMK-1500 SNF half-assemblies. This is to improve the mechanical strength of the canister, as is being done in Sweden as well as in Finland. The copper canister has a wall thickness of 50 mm and should be made of oxygen-free copper with low phosphorus content. The canister insert will be made of cast iron with a minimum wall thickness of 50 mm. Preliminary data for the RBMK-1500 SNF reference canister suggest a 1,050 mm diameter and a 4,070 mm length. One disposal canister can hold 32 RBMK-1500 fuel half-assemblies, and for Lithuanian SNF disposal, about 1,400 canisters would be necessary. The required area for the repository construction would be about 0.4 km^2.

The repository concept is described at the level of detail needed to perform a generic safety assessment and cost analysis. Determination of thermal evolution, criticality, and other important disposal characteristics for RBMK-1500 SNF emplaced in a copper canister supports the development of this concept.

Cost estimates for the disposal of SNF and ILW-LL within the crystalline basement in Lithuania is presented in Volume 2 of Investigations of Possibilities (2005). This preliminary assessment is based on the experience accumulated during development of the Swedish KBS-3 concept, as applied to the Lithuanian case. The method of analysis used is based on the application of a concept known as the "successive principle," which has been used especially as a tool for managing uncertainties that may develop due to unforeseen events in the future. The input data for the calculations are obtained from the "most likely" costs, or the so-called reference costs, by means of conventional (deterministic) calculations. These calculations are based on a functional description of each facility, resulting in layout drawings, equipment lists, personnel forecasts, etc., under established fixed conditions, but without allowances for variations and uncertainties. To provide guarantees to cover losses resulting from future unforeseen events, reasonable additional costs (cost variations) are included in the calculations. The influence of a specific variation in costs is evaluated, and the result provides a mean value of the future costs and the standard deviation of the cost for a desired degree of confidence.

A generic safety assessment of the Lithuanian repository is presented in Volume 3 of Investigations of Possibilities (2005). Because of the assumed similarities in the repository environment and repository concept, the selection of scenarios is based on experience from the safety assessment performed in Sweden. For this stage of generic safety assessment, only two scenarios were chosen: base scenario and canister defect scenario. In performing this safety assessment, Lithuanian parameter values were used as much as possible.

To analyze system evolution under the above-mentioned scenarios, thermal evolution, criticality, and dose assessment of the copper canister loaded with RBMK-1500 SNF were performed. Analysis of thermal-evolution effects at the top of the crystalline rock has shown that, over a period of one million years after deposition of the SNF, the heat released from the canister with RMBK-1500 SNF will have only a marginal impact on thermal conditions at the top of crystalline rock.

An essential component of the safety assessment was to calculate radionuclide release and doses to critical group members. The computer codes COMPULINK7, CHAN3D, provided by SKB, were used to calculate radionuclide release through the near-field and far-field regions from the canister with an initial defect (canister defect scenario). For dose assessment, the computer code AMBER 4.4 (UK) was also used. The results from calculations show that most of the analyzed radionuclides identified as safety relevant will be effectively retarded in the near field (by the bentonite buffer). Release rates of Cs-135, I-129, Tc-99, and Ra-226 are the most significant. The release rates from the near field are dominated by Cs-135 and I-129. At longer times, the release rate is dominated by Ra-226, which is formed by an in-growth from the chain decay of U-238. Modeling of the transport of radionuclides through the far field was performed for the most dominating radionuclides in the near field. The values of parameters that have an influence on radionuclide transport were selected from the study area in Lithuania, and if they were not yet available, values from the Beberg site (Sweden) were used. Results of total dose behavior demonstrate that the dose constraint of 0.2 mSv/y will not be exceeded in a period of a million years, and in fact will be lower, by two orders of magnitude, than this constraint. The total dose rate for the initial period is dominated by the nonsorbing radionuclide I-129; at later times, it is dominated by Ra-226.

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Chapter 16

Present Issues in the National Waste Management Program for Mexico

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16.1. HISTORICAL OVERVIEW OF MEXICAN NUCLEAR PROGRAM

The use of radioactive materials in Mexico started formally in the middle 1950s when, by means of a presidential decree, the National Commission of Nuclear Energy (CNEN) was created. As a pioneer organization, the CNEN carried out some programs that involved the use of radioactive sources, such as a radioisotope program consisting of applying radioactive tracers in physiological studies, evaluating blood volume, locating brain tumors, and studying protein metabolism, among others.

In the mid-1960s, the Nuclear Center of Mexico was inaugurated, located at a site 36 km from Mexico City. Several CNEN facilities were housed there, such as a Van der Graaf accelerator, a Triga Mark III reactor, and a number of well-equipped laboratories. In 1979, the National Institute of Nuclear Energy (INEN), which had replaced the former CNEN, was divided into three other nuclear organizations with well-defined responsibilities: the National Institute of Nuclear Research (ININ), devoted to conducting research and development (R&D) programs in nuclear and related matters; Mexican Uranium (URAMEX), a government enterprise devoted to the prospecting, exploring, and exploiting of uranium; and the National Commission of Nuclear Safety and Safeguards (CNSNS), the regulatory authority.

In the mid-1970s, the construction of two nuclear power reactors was initiated at the Laguna Verde site, on the southeastern coast of the state of Veracruz. In 1985, because of a union-workers strike (URAMEX), the government decided to close this enterprise and to cancel every activity related to exploitation of uranium. In 1988, the first Rules and Regulations Code on Radiation Protection and Safety was promulgated, and several subsidiary standards have been put into force since then. In 1990, the Laguna Verde Unit 1 reactor started its commercial operation, followed by the start of Unit 2 in 1995.

16.2. LEGAL FRAMEWORK

Article 27 of the National Constitution (Constitución Política, 1975) states that the Mexican government has the exclusive power to exploit radioactive raw materials in Mexico, and specifically to use nuclear fuel for the purpose of electricity generation and other activities.

The regulatory Law of Constitutional Article 27 (Ley Reglamentaria, 1985) in the field of nuclear matters, states that the beneficiation of radioactive ores, the fuel cycle, fuel reprocessing, and the temporary or definitive storage and transport of irradiated fuel or wastes produced in its reprocessing—all must be considered strategic activities. It also states that the generation of electricity by nuclear means is a right reserved by the Mexican Federal Commission of Electricity (CFE). Concerning radioactive waste management, the same law states that the government, through the Energy Ministry, is in charge of the storing, transporting, and depositing of spent nuclear fuel (SNF) and radioactive waste, whatever its origin, and that the same ministry has the exclusive authority to authorize other governmental entities for temporary storage of SNF (and radioactive waste from nuclear power utilization). However, the same law also confers to the CNSNS (currently part of the Energy Ministry) the responsibility to evaluate and authorize the siting, design, construction, operation, modification, ceasing of operations, definitive closure, and decommissioning of any nuclear waste storage facilities, as well as the processing, conditioning, releasing, and storing of radioactive wastes and any disposal of them.

The Rules and Regulations on Radiation Safety and Protection Code (Reglamento General, 1988) contains several stipulations concerning the management of radioactive wastes in the facilities where they were produced. In addition, ten official Mexican standards (NOM), mandatory in nature, have been issued, which cover different aspects relevant to radioactive waste (RW) management and disposal, such as classification, determination of activity, and concentration in RW packages. Also covered are the requirements for low-level RW packaging intended for final near-surface disposal; requirements for RW incineration facilities; lixiviation tests for solidified RW samplings; requirements for near-surface disposal facilities (siting, design, construction, operation, closure, postclosure and institutional control); management of RW in radioactive facilities using open sources; and limits for considering a solid remainder as RW.

16.3. ORGANIZATIONS INVOLVED IN RW MANAGEMENT

There are several organizations involved in radioactive waste management, as mentioned above. At the top, the Energy Ministry is the executive office responsible for nuclear and radioactive waste management, but with the capacity to delegate operative functions. The CNSNS, as the regulatory body, is responsible for the technical evaluation and licensing of everything concerning waste management and disposal. As owner of the nuclear power plant, CFE is the current administrator of SNF and low- and medium-level RW generated in plants. ININ is the current operator of an RW treatment plant (for low- and medium-level RW, in addition to that generated at the Laguna Verde Nuclear Power Plant [NPP] located inside the Nuclear Center of Mexico, and is also the operator of CADER, a facility located 60 km northeast of Mexico City, for temporary storage on the surface.

16.4. CURRENT SCENARIO

The Laguna Verde NPP has two BWR units, each with 2,021 MW of thermal power, which have been in commercial operation since 1990—Unit-1 since 1990 and Unit-2 since 1995. The initial fueling per unit was composed of 444 assemblies of 1.87% enriched U-235. The estimated refueling per unit is about 96 assemblies. The retired assemblies are stored in the SNF pool of each unit, operation of which is managed by the utility itself.

Conservative estimates lead to the conclusion that enough waste-storage space is available for the entire plant lifetime, even considering a possible license renewal for an extended period. This is a temporary measure, while a definitive decision is contemplated. This final decision will be made on the basis of a cost-benefit analysis considering all the relevant factors, and considering all the existing options (including geological repository, exporting for reprocessing, dry storage, and other options that may be identified in the future). However, at the present stage, efforts are devoted to establishing a near-surface disposal facility for low- and medium-level RW originating in the NPP as well as in radioactive facilities.

The low- and medium-level RW originating in the NPP are also managed by the utility. The total annual average waste produced is about 220 m^3, which is being conditioned at the plant and stored in two temporary storage buildings. These facilities are located near the plant, outside the exclusion area but within the premises of CFE.

On the other hand, a conservative annual average RW volume of 30 m^3, including spent radioactive sources, is produced in the radioactive facilities within the country. These include medical, industrial, and research applications of open and sealed radioactive sources. The management of this kind of RW is carried out by the ININ as the de facto responsible organization, which includes collection, treatment, and storage. ININ operates a small RW treatment plant where conditioning and solidification is performed. ININ also operates one interim storage facility on the surface, which is the above-described CADER. The CADER facility is located in an area where land use and demography trends have changed strongly with time, and because of this change, the regulatory body has recommended the facility's closure in the near future.

The rate of generation for low- and medium-level RW, both in the NPP and in the nuclear facilities within Mexico, creates the necessity for an adequate disposal facility in the near term (i.e., within the next 10 years). In this context, the region near the Laguna Verde site has been the subject of a complete site study for construction of a near-surface facility intended for long-term storage of the low- and medium-level RW. Although the coastal characteristics of the site are seen as a disadvantage (that should be compensated for with strongly engineered barriers), no serious problems are expected regarding public acceptance. If this site is the final option chosen for the low- and medium-level RW generated at the plant, it could also be the solution for RW generated in the rest of Mexico, because the volume of the latter is relatively smaller than that of the plant.

It is very well known that in the state of Chihuahua, there are regions with good characteristics for housing underground RW disposal facilities. However, there exists the political problem that the state and local governments have shown a strong opposition to such an idea. For example, the Nopal I uranium deposit in the Sierra Peña Blanca has been selected for investigation by U.S. researchers, since it has a remarkable number of characteristics in common with Yucca Mountain, Nevada, a site within the United States approved for the purpose of geological disposal of SNF produced in that country. The data compiled could be used to support a site study addressing the construction of an underground facility for RW disposal, in the same geological province, but taking care not to disturb any areas with the potential for uranium exploitation.

Among the similarities shared with Yucca Mountain (Murphy, 1992), the Sierra Peña Blanca is located in the Basin and Range province of western North America, which is the same location, geologically speaking, as Yucca Mountain, and is also a large rotational fault block composed of silicic volcanic rock of Tertiary Age, with similar pyroclastic textures and rock chemical composition. The preserved total thickness of the volcanic units varies over the Sierra Peña Blanca area from 106 m to 538 m, and rock ages range from 35 to 44 million years. The groundwater table is hundreds of meters below the elevated ground surface. The Nopal I uranium deposit, which is the specific location in the Sierra Peña Blanca where more data have been collected, is in the unsaturated zone, well above the water table. Although the rock in this unsaturated zone is not completely dry (because capillary and sorptive forces retain water on surfaces and in small pores), this factor is considered neither an impediment nor unsolvable, using a good design for the engineered barriers. The climate in Peña Blanca is classified as arid to semiarid, with an annual rainfall of about 24 cm per year. The groundwater recharge and flow through the unsaturated zone are probably low, owing to evaporation and transpiration by plants.

Because of the several stages of hydrothermal activity and near-surface weathering at Peña Blanca, numerous rich deposits of uranium have been produced there. As a matter of fact, the Peña Blanca district is one of the most important uranium ore reserves in Mexico. One advantage derived from this uranium deposit is that the natural uraninite there is similar to SNF in composition and structure. Thus, the processes affecting the uraninite ore body at Nopal I should resemble (to some extent) the long-term processes affecting SNF alteration and contaminant transport, which are determining factors in the performance of a geological repository.

16.5. CONCLUSIONS

In Mexico, the necessity exists for people within the nuclear industry to provide definitive solutions in the short term (i.e., within the next 10 years) concerning the definitive disposal of low- and medium-level RW generated at Laguna Verde and also at other facilities. Such a solution includes the nomination (or possibly even the creation) of an entity devoted to managing such a program, with enough political, administrative, and financial support to handle the problem. This organization would be responsible for the site selection process, as well as near-surface facility construction and operation. The now-existing legislation necessary for carrying out these tasks is considered appropriate and complete. On the other hand, people within the industry have the perception that enough time exists to analyze and make the proper decision about long-term management of Laguna Verde SNF, and that none of the presently available or future options should be discarded. Consideration of a possible geological repository, constructed at a site that is thoroughly studied and found to have suitable, demonstrable characteristics for that purpose, should be considered a valuable advance in this context. Among the pending tasks, it is necessary: to develop a better relationship and stronger influence with the politicians who have responsibility in these issues, to solve some political and public-acceptance challenges, and to reverse the low-profile policies sustained so far. In this way, such a project would become more acceptable, enabling it to meet the increasing demands for RW management in the country. Besides the considerations based only on the present scenario, any potential future reactivation of the Mexican nuclear program for near-term production of electricity will increase the necessity for viable, effective answers to questions about RW management—always one of the traditional issues in satisfying public opinion about nuclear energy.

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Chapter 17

Disposal of Radioactive Waste in the Russian Federation

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17.1. INTRODUCTION

In the second half of the twentieth century, implementation of defense programs to produce nuclear materials resulted in the formation of large quantities of radioactive waste. Handling this waste in a manner similar to that of industrial waste, by dumping it into water and soil of the vadose zone and into trenches and collector pits, led to the invasion of radioactive substances into the environment and to a radioactive waste impact on human beings, animal life, and vegetation. Radioactive waste is a byproduct of electric power generation from nuclear power, the production of nuclear fuel, and the processing of spent nuclear fuel (SNF). Radioactive emissions are thus an issue in various areas of human activity.

Governments in countries where radioactive waste originates are required to implement full-scale programs aimed at preventing the harmful impact of such waste. They have to provide for corresponding expense items in the state budgets, which diverts significant funds from other important social issues, such as eradication of poverty, health care, and education.

This issue is most acute in the Russian Federation and the new states of the Former Soviet Union (FSU), some of which appear to be in possession of accumulated radioactive waste. In the process of changing the forms of ownership, new enterprises for this waste have evolved out of the former defense plants. Other defense enterprises are out of business, and the waste left behind is either managed by the state, or belongs to nobody.

Complications in settling the waste handling issues are associated with (1) difficulties in the economic development of these new states, (2) the ongoing creation of waste handling legislation, and (3) the broad involvement of the public in the discussions.

The disposal of radioactive waste in geological formations is an important option for solving waste handling issues, one that has been utilized within Russia starting as early as the middle 1950s. The prevailing concerns then were with the geologic structure and properties of the rock formations when used as waste repositories. Although this approach initially appeared quite adequate, there have been further developments as a result of directed research programs. To date, our knowledge with regard to geological disposal of radioactive waste could be deemed to be almost completely developed.

This review is focused on the issues of deep geological radioactive waste disposal in the Russian Federation. We examine the history of this disposal in geological formations, the current conditions and further development of relevant knowledge, and basic directions for future practical research.

17.2. DEVELOPMENT OF RADIOACTIVE WASTE DISPOSAL IN RUSSIA

The idea to use geological formations for radioactive waste disposal was made public in the Soviet Union in the middle 1950s, as part of a proposed program on defense waste management, primarily involving waste handling. Liquid radioactive waste, generated in large quantities during the production of nuclear defense materials, appeared to be the most hazardous kind. The basic direction adopted for handling this type of waste was its conversion into solid waste forms by incorporating or embedding them into bitumen, cement, glass, and other materials, followed by storage and disposal.

Handling small quantities of medical and research radioactive wastes did not present any substantial difficulties. In the 1950s–1960s, several enterprises were established for the collection and treatment of solid and liquid radioactive wastes, generated by the above sources, as well as for the disposal of spent ionizing radiation sources used in industry. These enterprises performed liquid waste solidification, solid waste treatment and compacting, followed by disposal of this conditioned waste into shallow storage sites—repositories. These enterprises made up the "Radon" system; its 16 storage sites are currently in operation and are subordinate to the control of the Russian Federation and local municipal groups.

The situation with waste from the atomic industry in the production of nuclear materials was more complicated. Large volumes of liquid radioactive waste, ranging from hundreds to thousands of cubic meters per day, were produced; and certain categories had a high level of radioactivity. This waste was stored in open surface basins, often in former natural reservoirs. This created serious concern because the waste represented a potential source of environmental contamination in the event of inclement weather conditions (tornado, typhoon) or military operations with nuclear weapons.

The development of a technology for processing and solidifying large amounts of high-level radioactive waste, as well as the design and construction of appropriate facilities, was faced with substantial technical difficulties and required a considerable period of time. A heat-explosion incident in 1957, with high-level waste in a tank in the Southern Urals (the "Mayak" project), forced quicker decision making with regard to waste management.

Consequently, an interim alternative was proposed to either remove the liquid waste immediately after it was produced or pump it from open surface-waste storage sites and then inject it through wells into permeable geological formations (reservoirs). This proposal was based on experience gained in the oil industry, where produced water, after being separated from the oil, is handled one of two ways: (1) flooding the oil deposits to increase the oil yield or (2) pumping the produced water back into unneeded wells and thereby dispose of it. Concurrently, activities aimed at the development of a radioactive-waste-solidification technology were also under way. However, disposing of liquid wastes by injection in wells was the first technology with practical results.

Results of the preliminary experiments and an expert assessment, made by leading specialists and scientists in geology and oil-field exploration, were used to formulate the basic provisions for the practice of deep underground liquid radioactive waste disposal, as follows:

- Liquid radioactive waste may be disposed of solely if a selected horizon is capable of retaining the waste planned for disposal, and if the geologic structure and rock properties at the injection site meet the requirements of the horizon being isolated, both from the surface and from underground aquifers used for water supply.
- The injected waste must be localized within the preset boundaries of the geologic medium, i.e. within the limits of the mining lease provided for the disposal purposes, similar to the arrangements made to establish a mining lease for mining minerals.
- The waste injection must be preceded by exploration of the geologic medium to acquire information necessary for justifying the feasibility, safety, and consequences of the disposal.
- The injected waste should be compatible with the geologic medium, i.e., the injection must not be accompanied by processes that either make impossible further injection of the required waste volume or deteriorate the waste isolation conditions.
- The consequences of disposal should be predictable, and the injection process should be controlled at all times.

These provisions were endorsed by the government, used to define the scope and sequence of operations in implementing liquid waste injection, and afterwards used as a basis for the concepts of solid and solidified radioactive waste disposal in low-permeability formations.

Studies to implement liquid radioactive waste injection included the following:

- An analysis of the fundamental potential for waste injection in the vicinity of atomic industry enterprises
- A survey to specify the geological structure of areas that are promising for injection
- A study of waste and its interactions with rocks
- Options for the design and technology of well construction and surface equipment
- A forecast of disposal consequences.

A geologic survey and studies were conducted in the vicinity of enterprises where the available data suggested the presence of porous horizons suitable for the disposal of liquid waste. The goal of the completed work was to establish a basis for the creation of a repository and to acquire the data necessary for its justification and design. The work involved geophysical studies, well drilling, and an examination of well profiles and filtration tests, as well as an investigation of underground water samples, waste samples, and the interaction of waste with the geological medium. It was found that favorable geological conditions were available at three atomic industry enterprises—namely, (1) the Siberian Chemical Combine of Seversk, Tomskaya Oblast; (2) the Mining and Chemical Combine of Zheleznogorsk, Krasnoyarsk Territory; and (3) the Scientific and Research Nuclear Reactor Institute of Dimitrovgrad, Ulianovskaya Oblast.

As a result of the geological survey, permeable horizons possessing reservoir properties and confining low-permeability clay beds separating the reservoirs from one another have been selected. A correlation of well profiles has also been performed, the horizon sequence in the geological medium has been determined, and geologic sections have been plotted. According to data from filtration tests, the filtration factors for potential reservoir beds have been determined, and the degree of disconnection from overlying horizons and the surface has been confirmed. Special studies have been performed in areas of suspected and existing tectonic fractures to determine the degree of interconnection between horizons. None has been detected. At the same time, studies were conducted on the waste and its compatibility with rocks of the reservoir beds. The requirements imposed on the waste intended for injection have been set, and the methodology to control wells with a steady rate of injection has been developed.

Based on the results of this survey and study of the deep repository, designs were developed, an expert review was made, and the construction of the desired system was completed. Three deep liquid radioactive waste repositories, for the enterprises named above, commenced operations in the period 1963–1969. Waste injection was accompanied by monitoring the waste distribution and the condition of the geologic medium; and the results were used to determine efficient injection conditions. As the requirements imposed on waste handling safety and environmental protection were increased and new laws were adopted, regulatory conditions were developed to set up the desired level of control for these operations. Additional studies and justification have also been conducted, and additional wells have been bored for the injection of waste.

These deep repositories were scheduled to be closed down by the end of the 1990s. Favorable injection results and special studies on injection safety were used to justify the prolongation of these repositories. Designs for repository reconstruction have been developed and reviewed. Based upon the above documentation, the necessary licenses have been issued for the enterprises to continue their waste injection activities. The scheduled life of the deep repository at Siberian Chemical Combine was lengthened to the year 2016, at Mining and Chemical Combine to the year 2011, and at the Scientific and Research Nuclear Reactor Institute to the year 2020.

This accumulated experience on liquid radioactive waste injection has been used to create deep nonradioactive waste repositories for other atomic industry enterprises. Although the waste in question did contain radioactive nuclides, the quantities involved were insufficient for the waste to be categorized as radioactive. Repositories of this type have been created at the Tchepetsk Mechanical Plant (Republic of Udmurtia) and at the Kirovo-Tchepetsk Chemical Combine (Kirovskaya Oblast). Currently, the latter is being used to dispose of waste from fertilizer production.

A deep repository has been constructed at the Kalininskaya Nuclear Power Plant in Tverskaya Oblast to dispose of water-treatment waste and other effluents containing tritium. The tritium solutions could be referred to as liquid radioactive waste, but in combination with the water treatment waste, the tritium content is reduced below the limit for the waste mixture to be

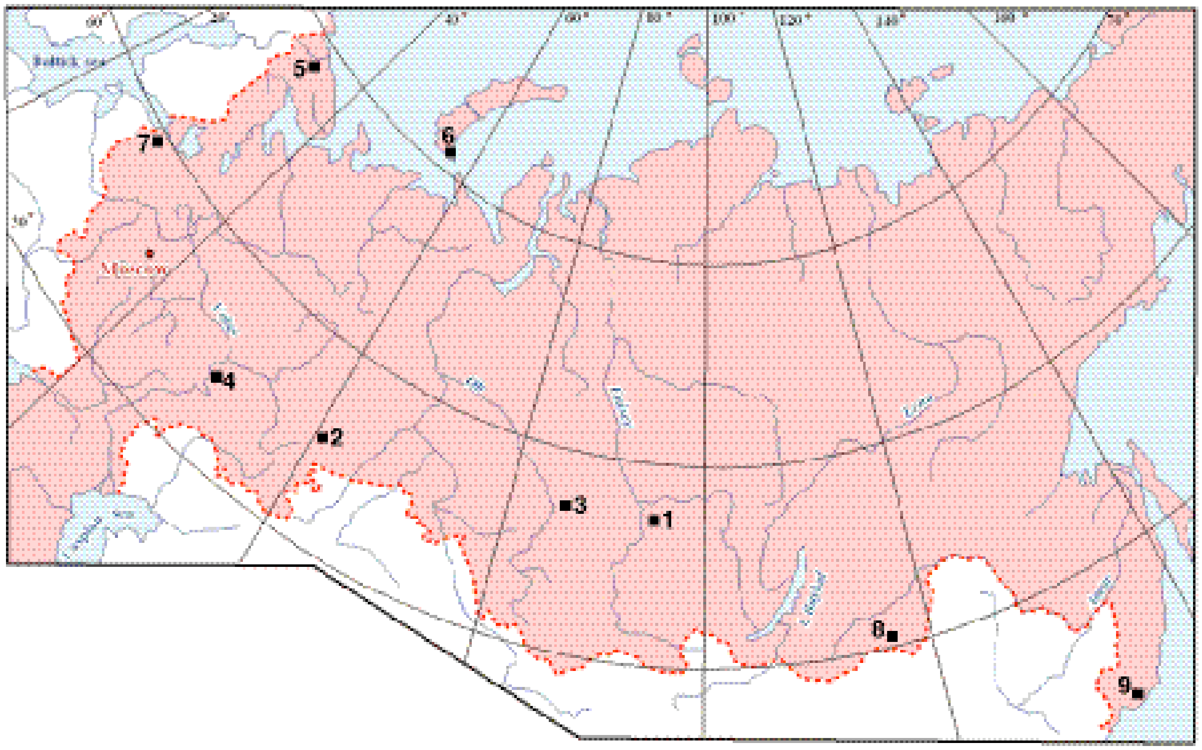

Figure 17.1. Locations of existing deep radioactive liquid-waste repositories and sites for deep solid and solidified radioactive waste repositories: (1) Zeleznogorsk (Krasnoyarsk territory), (2) Oziorsk (Mayak, Southern Ural), (3) Seversk (Siberian Chemical Combine), (4) Dimitrovgrad (Ulianovskya region), (5) Kola Peninsula, (6) Novya Zemlya Archipelago, (7) Leningradskaya region, (8) Krasnokamensk, Chitinskya region, and (9) Far East.

referred to as radioactive.

Concurrently with this ongoing work on liquid waste injection, a search has been under way for alternative techniques of waste disposal in the solid and solidified state, and the technology for solidifying liquid radioactive waste has been developed. Different options for the geological disposal of this waste form have been considered. including the following:

- Openings in depleted mines (shafts, adits)
- Mine openings constructed for radioactive waste disposal
- Natural cavities in geological media (caves)
- Cavities created by nuclear explosions in different rock types including rock salt
- Deep disposal in permafrost

Some unconventional techniques were also considered, specifically disposal in mine openings followed by the submersion of the waste into the rock melt obtained as a result of the heat released by the radioactive decay, and disposal in the immediate vicinity of volcanic vents.

Disposal of solid and solidified waste in dedicated mine openings has been recognized as a viable disposal technique. The requirements imposed on the formations suitable for the creation of a repository have been formulated, and the possible disposal schemes and sequence of operations to create a repository have been determined. An analysis of solutions found for the problems of the atomic and atomic-power industries, and the experience in relations with the public, have shown that it is advisable to locate the repositories in the immediate vicinity of the waste-generating enterprises, and if possible, directly on-site or in the sanitary buffer areas. This avoids the difficulties associated with waste transportation to the disposal site, public protests, etc. Figure 17.1

is a schematic map of the Russian Federation showing locations of existing liquid radioactive waste repositories and the sites being developed for solid and solidified radioactive waste repositories.

Investigations regarding the development of geological repositories for solidified waste were started in the Southern Urals in 1975, within the territory of the "Mayak" Combine, Tchelyabinskaya Oblast, where both an SNF processing plant and a waste evaporation/vitrification facility had begun operation. These studies were carried out for over 20 years. At the Mining and Chemical Enterprise (Krasnoyarsk Territory), studies to create a solidified waste repository were started in 1990, at a location in the immediate vicinity of the enterprise.

Disposal of waste in permafrost is considered one of the geological disposal options, with the operation conducted at a depth of over several dozen meters. The design for a repository located on the Novaya Zemlya archipelago was developed in 2002; however, its implementation has been postponed for a number of subjective reasons.

Since the beginning of the 1990s, a study has been under way to explore the possibility of developing a solid/solidified waste and SNF repository in granites of the Kola Peninsula, in the northern part of the Russian Federation. A similar project has also been started to justify using the openings in a nearly exhausted uranium mine for a waste repository in the eastern part of the Russian Federation in Tchitinskaya Oblast.

At the end of the 1990s, justification and a preliminary design had been developed for a solid and solidified waste disposal project, using a clay formation near St. Petersburg. At the municipal "Radon" project for radioactive waste disposal in the Moskovskaya Oblast, another new concept was undertaken in 1988 to create a demonstration facility for solid waste disposal, using large-diameter wells ranging in depth from 40 to 100 m.

17.3. GEOLOGICAL LIQUID RADIOACTIVE WASTE REPOSITORIES CURRENTLY IN OPERATION

At the "Siberian Chemical Combine" and "Mining and Chemical Combine" projects, the sand-clay horizons used for the reservoirs contain fresh water, and at the Scientific and Research Institute for Nuclear Reactors, the carbonate rocks (limestones) contain brines. The geological structure of the deep repositories, hydrogeological conditions, and rock properties provide the basis for waste localization and isolation in the various formations. The confining low-permeability clay beds overlie the reservoirs and prohibit vertical waste migration. Normal travel velocities of the underground water in the reservoirs are sufficiently low enough to provide waste localization in the areas of disposal for a reasonably long period of time. Table 17.1 provides some data for the deep liquid radioactive waste repositories currently in operation.

There are two repositories at the Siberian Chemical Combine project. Site 18 is used for low-level radioactive waste injection into Horizon III (depth, 270–320 m)

Table 17.1. Description of deep well injection storage sites for liquid radioactive waste

Storage Site	Year Began	Type of Rock	Depth (m)	Volume (Mln m^3)
Siberian Chemical Combine				
Site 18, Horizon II	1967	Sand	349-386	20,2
Site 18, Horizon III	1967	Sand	270-320	20,0
Site 18a, Horizon II	1963	Sand	314-341	6,1
Mining and Chemical Combine				
Site "Severny," Horizon I	1967	Sandstones	355-500	2,5
Site "Severny," Horizon II	1968	Sandstones	180-280	3,7
Scientific Research Institute of Nuclear Reactors				
Permeable Zone III	1966	Sandstones	1440-1550	0,6
Permeable Zone IV	1977	Limestone	1130-1410	2,2

and Horizon II (depth, 349–386 m); Site 18a is used for medium- and high-level radioactive waste injection into Horizon II (depth, 314–341 m). The characteristic water velocities in the reservoirs are in the range of 3–6 m/year. Because of physicochemical interactions, the radionuclide components in the waste are retained by the rocks, thus effectively restricting radionuclide migration.

The Mining and Chemical Combine enterprise has one repository, the North Test Site (Table 17.1) in the Krasnoyarsk Territory. Two horizons are in use: Horizon I (depth of 355–500 m) is used for medium- and high-level radioactive waste injection, and Horizon II (depth of 180–280 m) is used for low-level waste. The characteristic underground water velocities are 5–6 m/year and 10–15 m/year for Horizons I and II, respectively. The reservoir rocks also retain the radionuclide components of waste. A hydrodynamic barrier, formed by a tectonic fault plane, confines the reservoir beds from the west. The barrier effectively prohibits waste movement towards the Yenisei River. Figure 17.2 shows the layout and a geological section of the liquid radioactive waste repository for this enterprise. The red color in the sections shows the extent of waste migration.

The Scientific and Research Institute for Nuclear Reactors enterprise has one repository, the Experimental and Industrial Test Site (Table 17.1). In the initial phase of repository operations, they used permeable zone III (depth, 1,440–1,550 m) for the medium- and high-level radioactive waste injection, and permeable zone IV (depth, 1,130–1,410 m) afterwards. The horizons used are located in a stagnant flow zone, and hence the characteristic underground water-velocity values are less than 1 m/year.

In areas where a deep repository is located, a system is installed to monitor the geological formation conditions, waste distribution, and any processes that have occurred. The monitoring points are located in observation wells, where hydrodynamic and geophysical observations can be made, and samples of the underground waters can be collected, to identify waste components. According to data obtained, the radioactive components remain within predicted limits and within the mining lease. Based upon results from modeling and migration forecast calculations, the waste components are going to stay within the limits of the mining lease for about 1,000 years or more. Observations on reservoir bed warm-up,

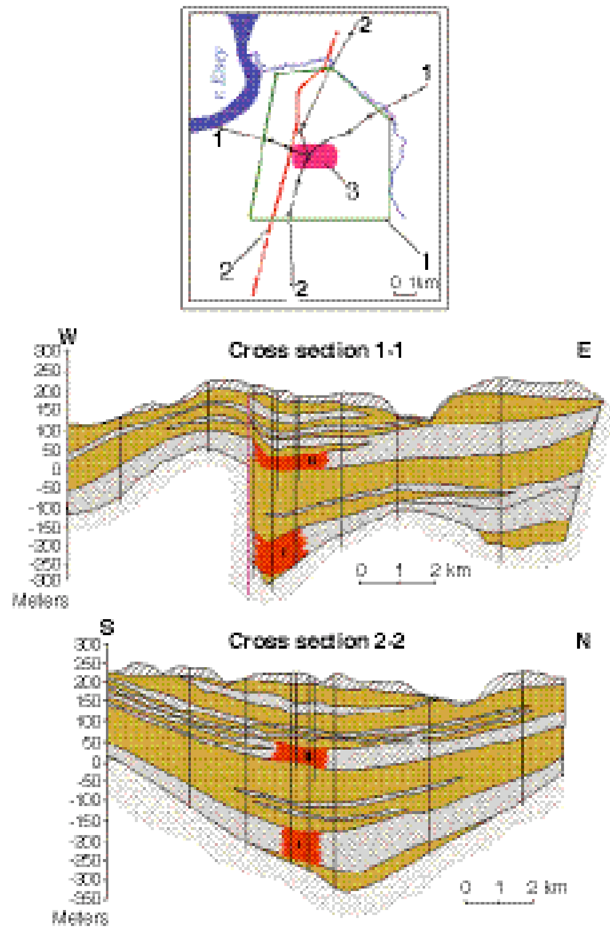

Figure 17.2. Deep injection disposal well site for liquid radioactive waste at the Krasnoyarsk-26 mining and chemical combine

in the high-level waste (HLW) disposal locations, show that temperature is not going to exceed the vaporization point under reservoir conditions.

In the period 1995–1999, several international projects evaluated the possible impact of deep liquid radioactive waste injection in Russia (Compton et al., 2000; European Union Report, 997; Vieveg et al., 1999; Wickham et al., 2003). According to the data obtained, the conservative approach showed a characteristic value for the possible accumulated dose load on the population (induced by injection) to be as high as several hundredths mZv. This level could be realized within 1,000 to 10,000 years.

After waste injection is completed, the waste repositories should be closed down to avoid any adverse effects on human beings and the environment, and to save future generations a lot of trouble about the disposed waste. A repository closedown technology is now under development.

17.4. INVESTIGATIONS ON SOLIDIFIED RADIOACTIVE WASTE DISPOSAL IN LOW-PERMEABILITY FORMATIONS AND PERMAFROST

17.4.1. Geological Repository in Southern Urals

The region planned for the repository is located in the "Mayak" project, the City of Oziorsk, Tchelyabinskaya Oblast. The project was the first in the USSR created for the production of nuclear materials. HLW in this project is vitrified, followed by interim surface storage of the highly radioactive glass and the disposal of it later on. To select a location for the deep repository, we have analyzed complexes of metamorphic and volcanogenic rocks and intrusive bodies within the site borders and in adjacent areas. A detailed examination has been made of the structural, mineral, and chemical composition, as well as of the tectonic faults within the geological medium. Volcanogenic rocks appeared to be the most suitable for the purpose; they were represented by tuffs and porphyritic lava breccias with an insignificant water permeability and a high mechanical stability and heat conductivity. Several fracture zones were located in the rock massif, the characteristics of which varied with depth. An upper heavily fractured zone was traced down to a depth of 35-40 m. A poorly fractured zone from 40 through 100 m was characterized by a non-uniform distribution of an open fracture network. Below 200 m, monolithic rocks are located with isolated fractures, and down to a depth of 2.5 m, there are more fractured areas with a filtration factor of 10^{-3}–10^{-4} m/day (Velichkin et al., 1997). As a result of the multi-year study, the highest priority areas have been selected; they are shown in Figure 17.3. It is planned to conduct a detailed exploration of two alternative areas and to create an under-

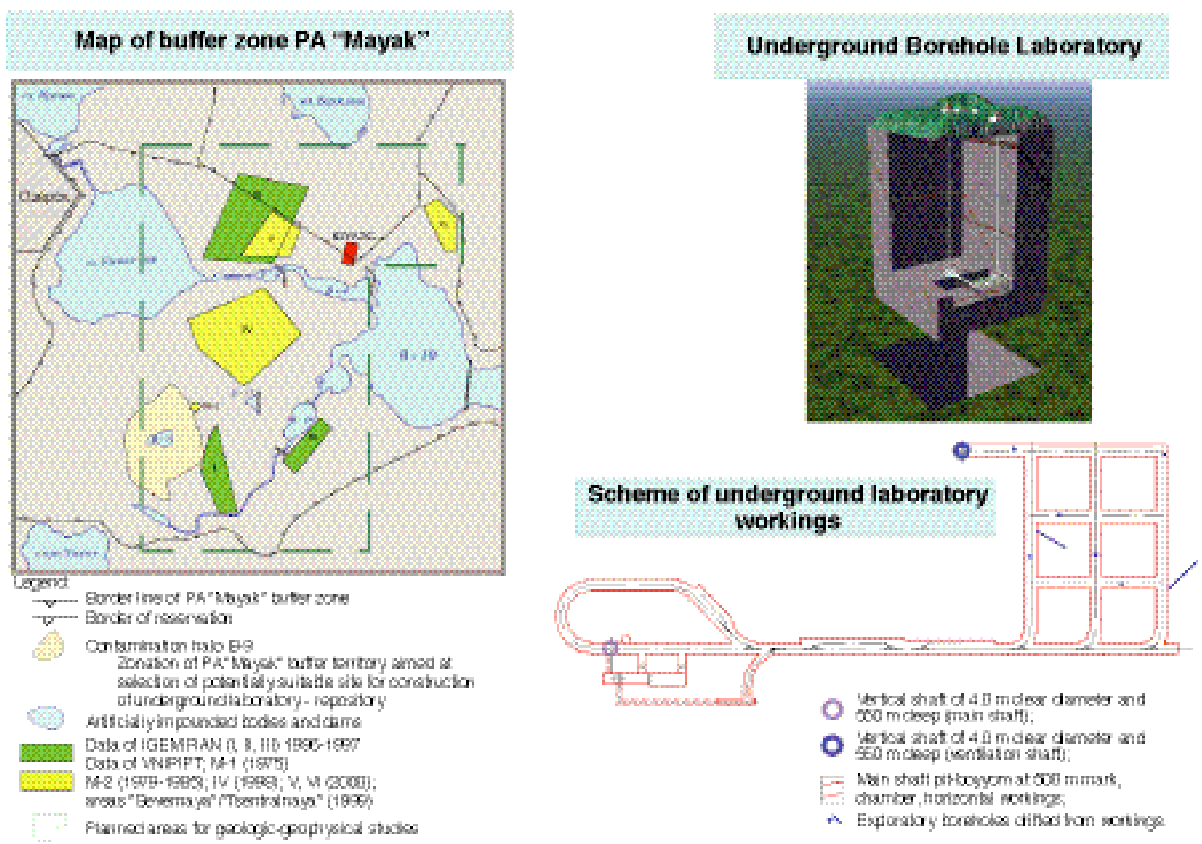

Figure 17.3. Underground laboratory in buffer zone of PA "Mayak"

ground laboratory to demonstrate disposal safety and the licensing activities required to develop a repository for geological isolation of radioactive waste.

17.4.2. Geological Repositories in Krasnoyarsk Territory

Prospective areas investigated for a solid and solidified radioactive waste repository near the Mining and Chemical Combine (The City of Zheleznogorsk, Krasnoyarsk Territory) are located within the boundaries of the Nizhnekanskii granitoid massif. The massif is one of the largest in middle Siberia, with an area that exceeds 1,500 km^2.

Using a combination of geological feasibility and other criteria, eight areas were selected from 20; and later, two areas were selected from these eight. The two most promising areas selected for further development are the “Verkhneitatskii” and “Yeniseiskii” sites (see Figure 17.4).

An extensive complex of investigations has been performed on these sites, including geological, geophysical, hydrogeological, and other studies. A combination of geological engineering and hydrological models, and forecast calculations, have been used to determine the velocity and time of filtration through the fractured zones from the depth intended for the waste isolation to the vadose zone. The drilling of test wells and several deep wells has also been carried out (Gupalo et al., 2004).

In 2001, proposals for the construction of an underground laboratory were drafted and approved. In 2002, studies on the Yeniseiskii site were started to assess the geologic structure, degree of tectonic faulting, and hydrogeological conditions. The purpose was to select blocks of homogeneous bedrock, with a low degree of disturbance, that would be suitable for underground laboratory construction and subsequent development of the repository. The geological engineering studies included neotectonic mapping, hydrogeological and meteorological observations, route-mapping studies, a helium survey, and field chemical and analytical studies.

During the geophysical work, the site area was explored along five main lines, 10 km each, and three profiles, 3 km each, using geomagnetic techniques, geoelectrical prospecting, gravitational prospecting, seismic studies, laboratory studies of rock samples, and thin sections. Three wells were drilled to a depth of 100 m each.

As a result of these geological and geophysical studies, a promising 5 x 5 km area has been selected, and the locations for deep-well drilling have been chosen. Geomagnetic profiling, seismic prospecting, and rock-fissuring studies have been performed. Deep-well drilling was started with complete core recovery, accompanied by a full complex of geophysical studies and filtration test work. Special hydrogeochemical studies have also been performed as well, to evaluate water exchange conditions on the Yeniseiskii site.

Among the promising areas of the Nizhnekanskii granitoid massif, the Yeniseiskii site, with an area of 70 km^2, is located most closely to the Mining and Chemical Combine, i.e., to the source of waste to be isolated. The geologic structure of the site is similar to that of the metamorphic rock massif containing underground production facilities of the Mining and Chemical Combine, which has already been studied in some detail.

In the course of the multi-year operations in the facilities of the Mining and Chemical Combine, the bedrock conditions were monitored to record any changes induced by natural factors (rock pressure, humidity, geodynamics) and man-made factors (temperature fields, etc.). The results of full-scale *in situ* parameter measurements of the physical processes (geomechanical, hydrogeological, geochemical) that have occurred in the bedrock under a 40-year thermal field impact have been obtained. The interconnections revealed among bedrock parameters could be used as an initial data set in designing for underground isolation objectives, both in the Nizhnekansky granitoid massif and for similar objectives in any other bedrock massif.

Basic Mining and Chemical Combine facilities are located in mine openings driven 50 years ago. This unique underground complex provides the possibility of investigating the technology for long-term storage and ultimate disposal of radioactive waste, and of investigating geophysical and geochemical processes in a bedrock massif. Experimental work is currently being conducted to obtain values for the hydrodynamic and geomigration parameters. These studies were started in tectonic fault zones (fragmentation and schistosity zones). Separate experiments are being carried out in water-saturated and drained-but-water-

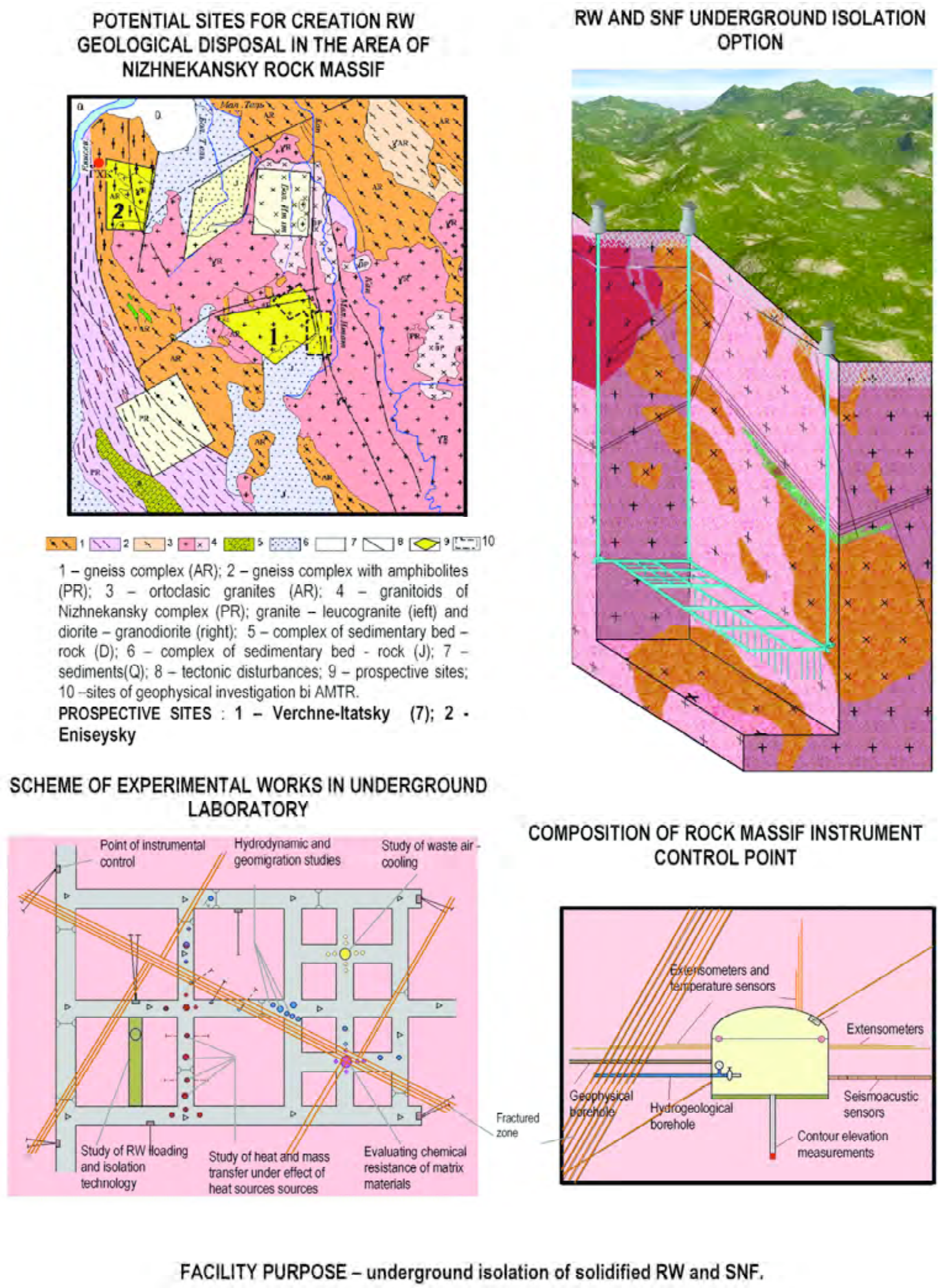

Figure 17.4. Nizhnekansky granitoid massif

permeable fracture zones.

A bench-scale facility for studying the migration of radioactive solutions through core samples, located underground in the Central Plant Laboratory, is being used for experiments with rock, vitrified waste, and actual radioactive solutions at temperatures up to 300°C and under pressures of as much as 30 MPa. This facility can be used to simulate the behavior of a multi-barrier isolation system under deep disposal conditions.

Filtration tests were performed in 20 research wells, within which hydrodynamic and geomigration studies were performed under the natural conditions of the fractured rock massif. Using complex radioactive solutions, diffusion coefficients, as well as sorption and desorption rates, were obtained *in situ* in man-made fracture zones.

An analysis of the multi-year *in situ* studies of physical processes controlling underground isolation safety led to a new approach for the justification of geological sites for radioactive waste and SNF storage and disposal. This new approach is based on a "risk-cost" complex of criteria.

17.4.3. Geological Repositories in the North of Russia

The geological repository on the Kola Peninsula is intended for storage and disposal of solid and solidified waste that originated in the northern regions of the Russian Federation. The sources are: the Kolskaya Nuclear Power Plant waste, waste from marine nuclear power plants, and the decommissioned marine nuclear power plant equipment coming from commercial shipping companies and from the Navy. Based on available data for the geological structure of the Kola Peninsula and adjacent areas of Arkhangelskaya Oblast, 22 promising repository sites have been selected. Special studies led to a further selection of 2 sites in the northern and 1 site in the southern part of Kola Peninsula, as well as 1 site in Arkhangelskaya Oblast.

In view of the favorable geological and hydrogeological characteristics of these sites, a concept for waste disposal in engineered facilities at a depth of 100–150 m has been chosen. The repository will consist of a complex of developed openings, transportation galleries, and modules for the waste packages.

The long-term safety of these sites was investigated under an international project with SCK·CEN of Belgium. Results from this investigation led to the conclusion that all sites are safe for use as waste repositories (Preliminary Estimate of Safety, 2002).

The conceptual design for a SNF repository was developed in 1999–2000. The design provides for dry storage and a multi-barrier system of package isolation, either into metal-concrete containers or into a built-in metal-concrete structure.

Special studies and geological exploration, along with well drilling, were carried out in the Novaya Zemlya archipelago. A design for a solid radioactive waste repository in permafrost has also been developed. The disposal of solid and solidified waste into wells drilled to a depth of over several dozen meters has been considered as an option for geological disposal. Foreign experts participated in a safety evaluation of such a repository, in which the hypothesis of 4°C increase in the global temperature per 100 years was taken into consideration. Despite the positive results and a favorable review, a decision on repository construction was not made, and the project was not undertaken.

Investigations are under way on the justification for disposing of solid and solidified waste from the Leningradskaya Nuclear Power Plant (NPP) in the workings at the "Radon" Combine (Leningradskaya Oblast). Waste is to be transported through an adit and disposed of in drifts and shafts driven in the Cambrian blue clay stratum (Rumynin, 2003).

17.4.4. Geological Repositories in the East of Russia

Proposals are being considered for the development of a deep, solidified radioactive-waste repository near a nearly exhausted uranium mine in Transbaikalia, at the Priargun Mineral Processing Combine of Krasnokamensk, Tchitinskaya Oblast. After 2020, the mining operations will be finished, and the available infrastructure could be used for a repository. Existing openings would be used along with the construction of new disposal sites. Mined openings used for the solidified waste disposal could be of different types—namely, chambers accessible through a shaft or an adit, and wells of large diameter constructed from the surface or from an opening. Multi-barrier protection of disposed waste has been integrated into the design and involves waste forms (glass or mineral-like composites), waste packaging in containers, and sealing of packages by cementing materials.

In the Far East, the rapid accumulation of decommissioned nuclear-powered submarines is under way, and repositories are designed in bedrocks into which submarine parts can be moved by pontoons. An adit-type repository is also planned for medium-level solidified waste disposal. The general characteristics of solid and solidified waste repositories that have been planned to date are given in Table 17.2.

Table 17.2. Potential deep repositories for solid and solidified radioactive waste					
Repository	**Year Began**	**Type of Geological Formation**	**Depth Intervals, meters**	**State of Work**	**Projected date for Putting the Repository into Operation**
Combine "Mayak", Chelyabinskaya obl.	1975	Hard rock, porphirites	100—500	Geological exploration. Underground laboratory being designed	2025
Mining and Chemical Conbine, Krasnoyarsk	1990	Hard rock, granites	100—1000	Detailed geological exploration	2030
Kola Peninsula	1990	Hard rock, granites	100—400	Demonstration designing	2030
Priargunskii mine	1998	Hard rock, diabase	100—800	Preliminary discussion	2030
Repository in "blue clay," Leningrad region	2000	clay	100—600	Investigation	2020
Archipelago Novaya Zemlya	1995	Hard-rock, permafrost	30—100	Delayed	Not determined

17.5. PLACE AND TASKS OF SCIENTIFIC RESEARCH ON THE RADIOACTIVE WASTE DISPOSAL PROBLEM

Optimal decisions on geological radioactive waste disposal can be made based on: knowledge about the structure and properties of the geological medium, the processes that have occurred therein, the waste transformations after disposal, and the waste interactions with the medium. That knowledge should be organized into a system of interrelated ideas (models) quantitatively filled with parameters and characteristics of the problem, and possessing appropriate mathematical tools to obtain quantitative results as an end product.

The scientific activities involved in solving this waste disposal problem have a number of specific features. The geologic medium is a natural formation not readily accessible for direct exploration and possessing a heterogeneity inherent to natural objects. Properties of the medium are known at several testing points (probe wells, mine openings), which require averaging the explored medium properties by an extensive use of interpolation and extrapolation. Therefore, estimates of the explored medium properties are probabilistic, and models based thereon are conceptually probabilistic.

During the safety assessment of a waste disposal project (of risks caused by disposal), it is necessary to obtain forecasts over long periods of time, in the thousands to hundreds of thousands of years. To do that, we must study processes that, at the present time, occur rather slowly, even though in the future the consequences could become rather significant. Among these are tectonic processes. Tectonic-associated transformations in the stress fields of bedrock containing waste could result in: (1) the development of fractured zones, (2) reviving old tectonic stresses and developing new ones, (3) altering the underground flow of water, and (4) the destruction of localized zones of waste. To forecast these processes, at the repository justification and selection phase, is a very complex task.

During these investigations, the following results should be obtained:

- An experimental section organized into a database containing all information on the determinations and studies performed, classified by types and objectives of the studies
- A methodological section containing a set of models for the geological medium and processes, in the form of equations and solutions thereof, with software for exploring models, model verification, and data approval
- Preliminary assessment results of the disposal consequences, risks, and scientific recommendations

An integral part of the scientific approach to the assessment of disposal safety is the consideration of the repository and its environment as a natural and technical system. The system is interpreted as a man-made object including (as an integral part) a closed section of the environment, with this piece of earth serving as the site of waste localization and isolation.

The exploration of a repository as a natural and technical system, taking into account various processes occurring both within the repository volume (internal effects) and outside of it (external effects) on different time scales, shows the system's high degree of stability—much higher than for waste storage at the surface. The loss of system equilibrium in geological formations (when the bifurcation point is reached) could not, from a practical standpoint, occur in the foreseeable future. Various risks associated with this approach, such as the impact risk of the disposed waste and the risk of environmental impact on the disposed waste, are considerably less than those for the waste remaining on the land surface. The above is also true for the anti-terrorist stability of the geological repository as well.

Results of this investigation can be used to justify the fundamental possibility of safe disposal, to forecast consequences of an environmental impact assessment, to model processes that occur in the repository and in its environment, and to justify technical decisions related to repository design and the feasibility study.

17.6. REQUIREMENTS AND CRITERIA FOR DISPOSAL SAFETY

Requirements and safety criteria for radioactive waste disposal are based on the corresponding provisions of the Russian Federation laws pertaining to the use of atomic energy, on protection of humans from radioactive emissions and environmental impact, and on depths, as well as on recommendations of the International Atomic Energy Agency and other international organizations.

Application of the dose criterion allows one to prove whether the basic requirements restricting the irradiation of humans are met. The dose criterion is established in the Radiation Safety Standards (NRB-99) as the limit of the effective equivalent dose of 1 mZv/year per 1 individual. According to Basic Sanitary Rules to Ensure Radiation Safety (OSPORB-99) and Sanitary Rules of Radioactive Waste Handling (SPORO-2002), the effective irradiation dose for the population induced by radioactive waste (including the storage and disposal phases) must not exceed 10 μZv/year. For low-probability (destructive) scenarios of radionuclide escape from a repository, the individual risk limit equivalent must not exceed an annual dose value of 0.1 mZv/year (i.e., a probability of 5.0×10^{-6} $year^{-1}$) for each person of the population. The probability of an individual receiving a dose above the threshold of the determined post-irradiation effects (as a result of an unlikely destructive event occurring in the repository) must be restricted to the value of 2.5×10^{-6} $year^{-1}$.

The application of depths for the disposal of toxic waste, inclusive of radioactive waste, is regulated by the Russian Federation (RF) "On Depths" Law, Article 101, Paragraph 7 (2000). The primary requirement is the localization of radioactive waste and toxic substances in deep horizons. Accordingly, the localization criterion is based on the distribution of radioactive waste, and components thereof, in a geological medium. Radioactive waste and its components must be confined within a known, predefined volume (i.e., within the limits of the mining lease provided for waste disposal purposes). This criterion does not contradict the dose criterion, as long as the constraints imposed on waste distribution and the implementation of management requirements promote a reduction of the possible irradiation dose. The criterion constraining migration within the limits of the mining lease is supplemented by a time criterion, the actual length of the waste localization period (the guaranteed isolation time). The guaranteed isolation time depends on disposal conditions.

Other requirements and corresponding criteria for the waste disposal safety (the low-level criteria) are also applied. The latter are derived from the requirements on restricting the irradiation of humans, dose criterion, and localization criterion. They are interdependent with the main criteria and are usually contained in normal documentation such as rules, norms, and technological regulations. Among these are: the compliance of the real waste-component distribution with design or forecast values, and the scheduled levels of various parameters (e.g., the amount of waste components in the waste and underground water, the reservoir bed warm-up temperature).

The "Technical Regulation" law, as a primary safety criterion for different forms of activity, sets the risk interpreted as the probability of inflicting harm to the life or health of civilians, the property of juridical and normal persons, the State and municipal property, the environment, and the life and health of animals and vegetation—taking into account the severity of the harm done. This formulation assumes the application of risk cost estimates as well. The procedure for evaluating disposal risks in terms of cost is still to be developed.

17.7. CONCLUSIONS

Experience gained in Russia in handling radioactive waste, including liquid radioactive waste, has clearly shown that the disposal of such waste in geological formations (that meet preset standards) is the safest and most suitable method of disposal, from the standpoint of environmental protection and the protection of the general public from the impact of radioactivity.

Nevertheless, the classic scheme for geological disposal (i.e., the disposal of solid and solidified waste into mine openings driven in bedrock massifs) requires significant funding and a long time to implement. Decisions related to developing a repository are governed, in many respects, by the attitude of the general public, public organizations, and the local administration toward the project. With this in mind, the most advisable disposal solution is to locate a repository on site with, or in the immediate vicinity of, waste-generating enterprises—i.e., the existing operations of the atomic and mining industries. This consideration of proximity could be more critical than any geological factors.

With regard for the prospects of further developments in geological disposal in Russia, liquid radioactive waste injection will likely be finished by 2010–2020, followed by the closedown of the repositories. During the period 2025–2030, there will be no geological disposal. However, solid and solidified waste will be accumulated on the surface in interim storage sites operated in accordance with appropriate safety requirements. Near-surface solid and solidified waste disposal will continue in parallel, along with the construction of new geological repositories that are expected to commence operation after the year 2020.

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Chapter 18

Status of the Deep Geological Disposal Program in the Slovak Republic

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18.1. INTRODUCTION

During their operation, the Slovak nuclear power plants (NPPs) will produce approximately 2,300 metric tons of spent nuclear fuel (SNF), expressed as heavy metal (18,654 SNF assemblies), and about 5,000 metric tons of radioactive waste considered unfit for the near-surface repository at Mochovce. These wastes originate mainly from the operation and decommissioning of nuclear facilities, and medical, industrial, and research applications. The volumes reflect assumptions for two reactors operating at the Mochovce site and a preterm shutdown of NPP V1 in Jaslovske Bohunice in 2006 and 2008.

In the former Czechoslovakia, the back end of the fuel cycle was based on close cooperation with the former Soviet Union. Accordingly, the Soviets delivered fresh fuel and were expected to accept spent fuel for reprocessing without the return of any radioactive waste. From 1983 to 1986, a small part of NPP-V1 SNF was shipped to the Soviet Union, and the rest was stored in at-reactor pools and later, in a wet interim storage facility. The interim storage facility (capacity of 5,040 SNF assemblies) was commissioned in 1987 to provide storage for 10 years before shipping to the Soviet Union.

After the political changes in the late 1980s and early 1990s, the contractual arrangements between the Slovak Republic and Russian Federation were modified, and this led to a new SNF management strategy. It was decided to extend the capacity and lifetime of the interim storage facility in Jaslovske Bohunice. This led to a higher storage capacity (14,112 fuel assemblies), improved seismic resistance and safety, and an extended operating lifetime up to 50 years. Increasing the storage capacity was achieved by re-racking into a higher capacity (originally 30, now 48 fuel assemblies). This capacity was considered to be sufficient for the expected lifetime of all reactors installed in Jaslovske Bohunice. After the expected preterm shut down of NPP V1, there should be free space for about 1,500 SNF assemblies (Matejovic, 2004).

The Slovak management policy for SNF and radioactive wastes not acceptable in a near-surface repository follows Governmental Decisions Nos. 930/1992, 684/1997, 5/2000, and 5/2001. Three possible alternatives for the back end of the fuel cycle were taken into consideration:

1. After 40–50 years of interim storage, direct disposal for SNF and high-level radioactive waste (HLW) in a deep geological repository (DGR) constructed in Slovak territory.
2. Shipping and final disposal of SNF externally
3. Reprocessing and storage of HLW abroad, followed by disposal in Slovak territory.

From the economic point of view, the first alternative, direct disposal after 40 to 50 years of interim storage, seems to be the most advantageous. The second and third alternatives are not considered at present, but may be included in future considerations.

In the 1980s, within the former Czechoslovakia, the Nuclear Research Institute, Plc., studied the possibility of using a DGR. At that time, it was highly probable that such a project would be located in Czech territory, because of the much more suitable geological conditions there. After the separation of the Czech and Slovak Republics in 1993, research and development for a DGR in the Slovak Republic began in 1996, along the lines of previous activities. As a first step, Decom Slovakia, Ltd., together with contractors, has prepared (and adapted to specific conditions of the Slovak Republic) a revision of the federal document entitled "Project of Deep Disposal Development," prepared by the Nuclear Research Institute in Rez.

The Program of Deep Geological Disposal of Spent Fuel and High Level Waste in the Slovak Republic is funded from the State Fund for Nuclear Facilities Decommissioning and Spent Fuel and Radioactive Waste Management and, to a lesser extent, by the Ministry of the Environment of the Slovak Republic. Decom Slovakia, Ltd., coordinated the program of activities from 1996 to 2001 (Matejovic et al., 2001a). The last phase of the project consisted of the following key areas (main contractors are given in parentheses):

- Design and implementation (EGP Invest, Ltd., Uhersky Brod, Czech Republic and Energoprojekty, Plc., Bratislava)
- Source term (Nuclear Research Institute, Plc., Prague, Czech Republic)
- Near field (Nuclear Research Institute, Plc., Prague, Czech Republic)
- Far field (Geological Survey of Slovak Republic, Bratislava)
- Siting (Geological Survey of Slovak Republic, Bratislava)
- Safety analyses (Nuclear Power Plants Research Institute, Plc., Trnava, Slovak Republic)
- Public involvement (AEA Technology, Harwell, U.K. and Decom Slovakia, Ltd., Trnava, Slovak Republic)
- Legislation (Decom Slovakia, Ltd., Trnava, Slovak Republic)
- Quality assurance (Decom Slovakia, Ltd., Trnava, Slovak Republic)
- Coordination (Decom Slovakia, Ltd., Trnava, Slovak Republic).

During this stage of DGR development, both crystalline and sedimentary host environments were investigated. A decision on selection of the host environment will not be made before 2005. Selection of candidate localities is expected around 2010, and commissioning of a DGR is expected by 2037.

18.2. PROJECT STATUS

18.2.1. Design and Implementation

EGP Invest, Ltd., together with Energoprojekty, Plc., have developed a preliminary technical design for a DGR at a hypothetical site with two alternative geological host environments (sedimentary and crystalline). Conceptual ideas for the DGR operational phase were focused on various aspects of surface and underground activities, approached from the perspectives of technology, economics, and feasibility (Certik et al., 1997; Kanocz, 1997; Mandik et al., 1997). These activities included transportation and reception of SNF and HLW, encapsulation, conditioning, manipulation, access to an underground shaft or tunnel, emplacement of containers (disposal borehole or tunnel), and auxiliary and control systems. Preliminary safety, technical, and economic requirements for implementation, as well as the time scale and priorities for preparation of the DGR construction phase, were defined and evaluated.

Recent activities have been concerned with analyzing the need for an underground laboratory (generic research or confirmation laboratory) for the disposal program. This has involved consideration of possible alternatives, the proposal of technical research and economic issues, and the role of public involvement. A confirmation laboratory at a potential repository site was found to be the most appropriate solution at this time. EGP Invest, Ltd., has also prepared a preliminary feasibility study, based on knowledge and experience acquired in the Slovak disposal program, current status of mining technologies, and worldwide experience.

18.2.2. Source Term

Nuclear Research Institute, Plc., has described the physical and chemical properties (radionuclide species and their quantities) of the WWER-440 (Russian type pressure water reactor) S N F assemblies after interim storage

(Vokal et al., 1997). Activity of selected isotopes, total SNF assembly activity, fuel-assembly thermal power, and the contribution of selected isotopes were calculated (using the ORIGEN 2.1 code with appropriate data) over various time periods after refueling. Also determined was the neutron source from actinides and their daughter products in an SNF assembly, in part resulting from (alpha, n) reaction and spontaneous fission. The effect of cladding and WWER-440 fuel-assembly construction parts on the total inventory of radionuclides and heat production were calculated considering the following enrichments: 1.6%, 2.4%, 3.6%, and a planned 3.82%, at the burn-up from 12,000 MWd/tU up to 46,000 MWd/tU (Burian et al., 1998). These data will be used as input for the repository concept, disposal container design, and the SNF disposal safety assessment.

Possible mechanisms of radionuclide leaching from SNF and vitrified or cemented forms of HLW were also reviewed. The basic relations for calculating the source term were given (i.e., the quantity and rate of radionuclides that can be released from waste after its disposal in a DGR). Bibliographic data regarding radionuclide leaching were gathered, with the source term estimated (using the computer code PAGODA) for selected radionuclides and a hypothetical repository. The results related to the expected behavior of SNF cladding in a DGR environment have also been analyzed. It was found that zirconium-niob cladding could be considered an effective barrier against radionuclide release.

Current activities are focused on investigating mechanisms of radionuclide release from S N F cladding and HLW glass and cement matrices in a repository environment, as well as identifying a critical group of radionuclides and their characteristics.

18.2.3. Near Field

The first step in the near-field study was a critical review of the current status of this kind of assessment, engineered-barrier modeling, available information on the materials suitable for engineered barriers, and the disposal container (Laciok, 1998). Physical and chemical properties of the materials suitable for sealing and closure of disposal boreholes, tunnels and galleries, and shaft and vaults were described. Attention was given to processes important for radionuclide migration and retardation in engineered barriers and factors influencing these processes, as well as to transport characteristics of the critical group of radionuclides.

Nuclear Research Institute, Plc., and Skoda, Plc., prepared the first proposal of a disposal container with seven WWER-440 SNF assemblies (Vokal et al., 1999). Carbon-steel (80 mm) coated with a nickel layer (3 mm) is proposed for the outer wall, with an inner wall made of stainless steel (5 mm). In this analysis, the inner cask would be made of an aluminium alloy, facilitating handling with fuel assemblies and improved heat removal. Such a container ensures subcriticality, effective heat removal, and pressure resistance up to 20 MPa. Total container weight with encapsulated SNF would be 7.7 metric tons. Container handling will require additional shielding to ensure a surface effective dose of 0.1 mS/h.

18.2.4. Far Field

Worldwide experience in modeling geological barriers and groundwater flow (dominant factors in radioactive transport) were reviewed by the Geological Survey of Slovak Republic and Nuclear Research Institute, Plc. (Laciok, 1998; Laciok et al., 1998). Special attention was paid to analysis of groundwater flow mechanisms (in saturated and unsaturated environments) and transport of dissolved substances. Detailed surveys of hydrogeological and transport models were carried out, focusing on present-day model verification and validation procedures. Basic information about coupled processes and the advantages and disadvantages of deterministic and stochastic models were reviewed.

For sites identified as potentially suitable for a repository, three-dimensional models of the geological barrier were prepared (Paudits et al., 2000). These models reflect the current status of knowledge about geology, petrography, seismicity, neotectonic, hydrogeology, and the geochemistry of given sites. Also, we analyzed the interactions between host environment (granites or clays) and engineered barriers, as well as the possible alteration of the host-rock environment and engineered-barrier materials (induced by expected hydrogeochemical processes).

18.3. SITING

18.3.1. Brief Outline of Geological Conditions

The territory of Slovakia has an area of 49,016 km^2, and is located in a mountain chain of the Western Carpathians. From a geological point of view, the area has been subdivided into several tectonic units, as shown in Figure 18.1 (Biely et al., 1996). Only a few of

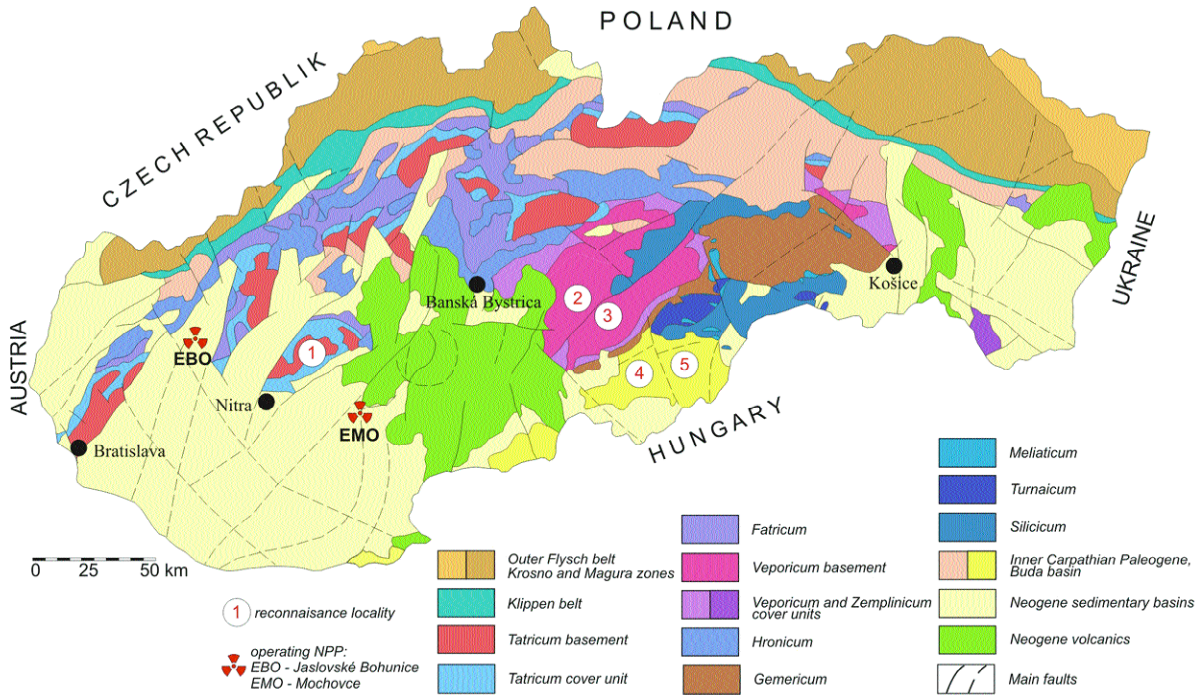

Figure 18.1. Simplified tectonic sketch of Slovak part of Western Carpathian. Status of deep geological repository siting in 2004: (1) Tribec Mountains, (2) Veporske vrchy Mountains, (3) Stolicke vrchy Mountains, (4) Rimavska kotlina Basin, and (5) Cerova vrchovina Upland

these units contain rock environments potentially suitable for an HLW/SNF repository site.

The Tatricum and the Veporicum large-scale tectonic units of the Central Western Carpathians include the Hercynian (Variscan) crystalline basement, with autochthonous Late Palaeozoic-Mesozoic sedimentary cover (Figure 18.2). The Tatricum Unit contains isolated cores of the crystalline basement (granitoids and metamorphosed rocks) partly covered by Late Palaeozoic and Mesozoic sediments. The Veporicum Unit is the largest granitoid pluton in the Western Carpathians, with a length of about 60 km. The rock is mostly granitoids and potentially can provide suitable sites for a repository.

A characteristic feature of the Western Carpathians is the basins that are infilled with Miocene sediments (predominantly clays, claystones, sands, and sandstones). The

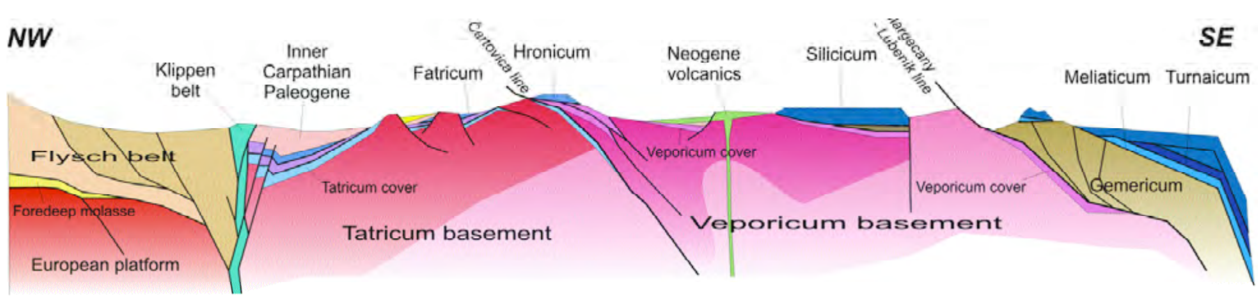

Figure 18.2. Simplified tectonic cross section of the Western Carpathians (not to scale)

overall thickness of these sediments is several thousand meters, and they are also potentially suitable for siting a repository in Slovakia.

Tectonics
Tectonic processes in the Western Carpathians have resulted in discontinuities (faults and fissures affecting the bedrock. Down to a depth of ~100 m, fissures provide good conditions for groundwater flow. At greater depths, they are usually closed, but a repository must avoid these important discontinuities.

Geomorphologic Evolution of the Relief
Long-term geodetic measurements indicate that, within the block structure of the Western Carpathians, the average rate of sinking blocks is approximately 0.5–1.5 mm/year. To assess the effects on relief, we are developing geomorphologic models to predict the future relief for a period with potential negative effects on the repository (from 104 up to 105 years) (Joo, 1992; Hok et al., 2001).

Seismic Conditions
In Slovakia, the generally accepted limit of seismic intensity is 60 MSK-64, and the majority of Slovakian territory roughly averages this value. Prospective sites for the repository are located away from recorded earthquake epicenters (see Figure 18.3) (Labak et al., 1997).

Climatic and Hydrogeological Conditions
The annual average precipitation in Slovakia varies between 400 and 1,000 mm, reaching 2,000 mm/year in high mountainous regions. Hydrogeological properties of the Western Carpathian crystalline complexes are not well known at depth. In terms of hosting a DGR, the main deficiency of these complexes is their petrophysi-

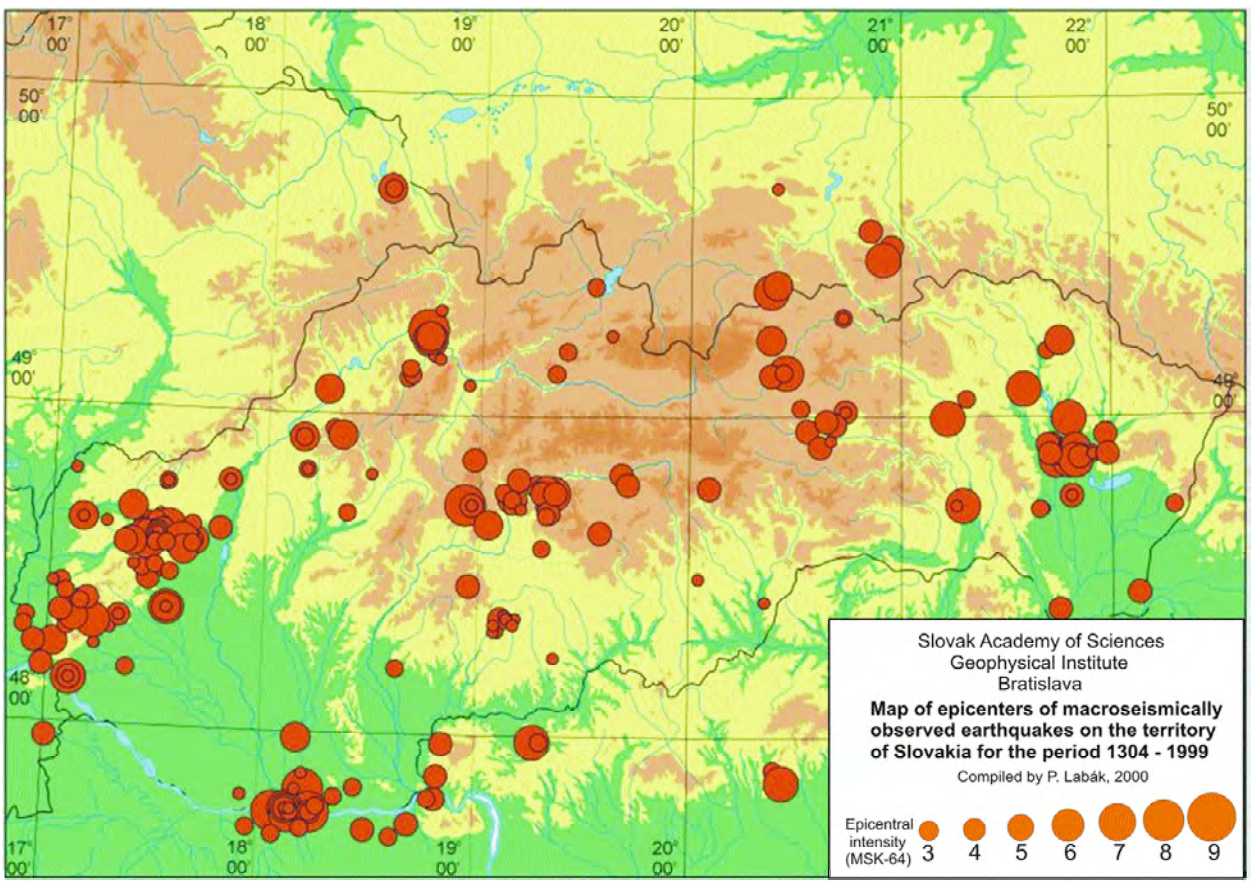

Figure 18.3. Map of epicentres of macroseismically observed earthquakes on the territory of Slovakia for the period 1304–1999

cal heterogeneity, resulting from tectonic effects. An unfavorable result of such effects is the frequent occurrence of ductile zones, which may indicate increased permeabilities even at greater depths (Jetel, 1996).

18.3.2. Siting Criteria

The site for a DGR, along with repository design and the engineered barrier system, must ensure long-term safety of the total system. Disposal in a deep repository serves two basic purposes:

- Long-term isolation of HLW and SNF from the environment, without relying on future generations to maintain the integrity of the system
- Long-term radiological safety for humans and the environment in compliance with existing regulations and internationally accepted recommendations.

Worldwide experience indicates that no unambiguous requirements exist in selecting a suitable geological environment, just as no rock type can be declared to be the most suitable. Various geological environments provide (by their character and structure) more or less favorable conditions for the location of a repository. If these natural settings are suitably complemented by properly engineered barriers and flexibility in repository construction and technology, various environments could guarantee similar levels of long-term safety.

A preliminary set of site-selection criteria for a DGR has been proposed (Lukaj and Jelinek, 1997; Lukaj et al., 1998), based on worldwide experience and consistent with International Atomic Energy Agency (IAEA) recommendations (Safety Series No. 111-G-4.1, 1994). Three groups of criteria were proposed for site selection:

1. Geologic and tectonic stability of prospective site (seismic activity, faulting, folding, uplift of the territory, etc.)
2. Characteristics of host rock (lithological homogeneity, hydrogeology, low hydraulic conductivity, absence of groundwater resources, favorable geotechnical conditions, rock stress, thermophysical and geological characteristics, absence of mineral resources)
3. Conflict of interests (natural resources, natural and cultural heritage, protected resources for wells or thermal waters)

From progress in program activities and a growing knowledge from exploration at different localities, these criteria were supplemented and adopted in 2001 (Matejovic et al., 2001b). This innovative set of criteria is based on program experience, as well as IAEA recommendations and Scandinavian experience. A qualitative evaluation of the suitability of a host rock includes specified rules and requirements in order of: preferences–requirements–criteria:

- Preferences. Worldwide accepted principles and conditions of host-rock or site suitability—advisable but not prescriptive
- Requirements. Specified principles and conditions of host-rock or site suitability—obligatory for host-rock or site selection
- Criteria. Defined qualitative and quantitative suitability measures, of limiting value for host rock or site selection

Evaluation of the geological environment is to be focused on all components: rock, water, morphology, and the geodynamic phenomena involved in predictions over more than 100,000 years. The conditions to be evaluated can be divided into the following groups:

- Geological
- Hydrogeological and hydrogeochemical
- Engineering geology
- Geomorphologic (surface area stability)

The basic social and economic requirements are:

- Areas with a higher degree of legal protection, mineral and underground water resources should be avoided.
- Areas with a lower population density and more favorable demography should be preferred.
- Beneficial effects of the repository on the site should be enhanced, whereas negative effects should be minimized.

18.3.3. Site-Selection Process

In 1996, this program started with a critical review of information (no field investigations) and included a survey of published and archival data on: regional geology, hydrogeology, engineering and geophysics. The results identified 15 areas potentially suitable for a DGR, which were spread among granitic (7), clayey (4), metamorphic (3), and flyschoid (1) formations (Kovacik et al., 1996, 1997). The accuracy of the assessment was on a map scale of 1:200,000. The next four years focused on screening through limited field verification and some technical measures (geophysical profiles and shallow drillholes). This produced a series of maps for each area, on a scale of 1:50,000, that included important geological and hydrogeological factors and data on mineral resources (Sestak, 1998; Kovacik et al., 2001). The result of this effort was the ranking of the potential areas into three groups:

- Areas recommended for further investigation. These areas are not expected to contain excluding factors.
- Abandoned, but not excluded areas. These areas may be considered as backup, if other sites prove unsuitable or, for any reason, unacceptable. Lithologic structure, tectonic classification, and presence of ore indicators at these sites probably rule out selection (though not conclusively) of a sufficiently large and homogeneous rock block.
- Excluded from further investigation. These areas contain utilizable geothermal energy below the prospective host rock; thin, hydraulically homogenous rock formations; and/or lithologic inhomogeneities.

Based on this analysis, three areas (five localities) were determined as prospective sites for construction of a DGR. Their total extent is 320 km^2. The next step was to perform a more detailed characterization of the sites, possibly reducing their number and extent. New field investigations (geophysical measurements—electric, gravimetric, magnetic, seismic—and shallow drilling down to 250 m, including hydrogeological and geophysical logging, were performed. More detailed maps will be developed for each site (scale 1:25,000). Three localities are situated in granitic rocks of Hercynian (Variscan) Early to Upper Carboniferous age and two, in argillaceous Neogene complexes. Further reduction in the number of sites is expected, but two alternative host environments should be considered, at least up to 2005. A decision on site selection is expected around 2010 (Kovacik et al., 2001; Matejovic et al., 2001a).

18.3.4. Prospective Sites for a Deep Geological Repository in Slovakia

Prospective sites in granitic formations:

- Central part of Tribec Mountains (46 km^2)
- Southern part of Veporske vrchy Mountains (78 km^2)
- Southwestern part of Stolicke vrchy Mountains (24 km^2)

Prospective sites in argillaceous and pelitic formations:

- Eastern part of Cerova vrchovina Upland (87 km^2)
- Western part of Rimavska kotlina Basin (85 km^2)

Central Tribec Mountains (Tatricum Unit)

This prospective site is an area of granitic rocks in the southern Tribec-Zobor block in the Tribec Mountains. The Zobor Massif, one of the largest crystalline complexes in the Western Carpathians, varies in tonalities from leucocratic granites, to fine- to medium-grained granites and granodiorites, to massive medium-grained granodiorites (Figure 18.4). Tectonic deterioration of the site is generally low (Figure 18.5), and thus hydrogeological conditions for a repository seem favorable. There are no known limiting structural-tectonic features present in the relief of the site. An ongoing drilling survey (started in 2001) has revealed an increase in rock quality with depth (homogeneity, low tectonic deterioration, etc.). Rock quality designation (RQD) at depths of 150–250 m is about 90–95%. No indications of ore concentrations or geothermal potential have been discovered in this region. Hydrogeological conditions in the deeper horizons are little known at present (Ivanicka et al., 1998; Hok et al., 2001; Kovacik et al., 2001; Madaras et al., 2004).

Southern Veporske vrchy Mountains and Southwestern Stolicke vrchy Mountains (Veporicum Unit)

These two sites are adjacent to each other, although they belong in two different geomorphologic units. A Muran-Divin tectonic line divides them into separate locations. The Vepor granitic pluton is the largest in the Western Carpathians (~60 km in length). Although it is a complex pluton, consisting of several granitic rocks, Alpine deformation (recrystalization) has a regional character here. Because of the pluton's size, it has been recommended for further investigation (Bezak et al., 1999). There are no indications of economically important mineralization.

Eastern Cerova Vrchovina Upland and Western Rimavska Kotlina Basin (Sedimentary Basins)

As previously mentioned, these sites belong to different geomorphologic units: the Cerova vrchovina Upland

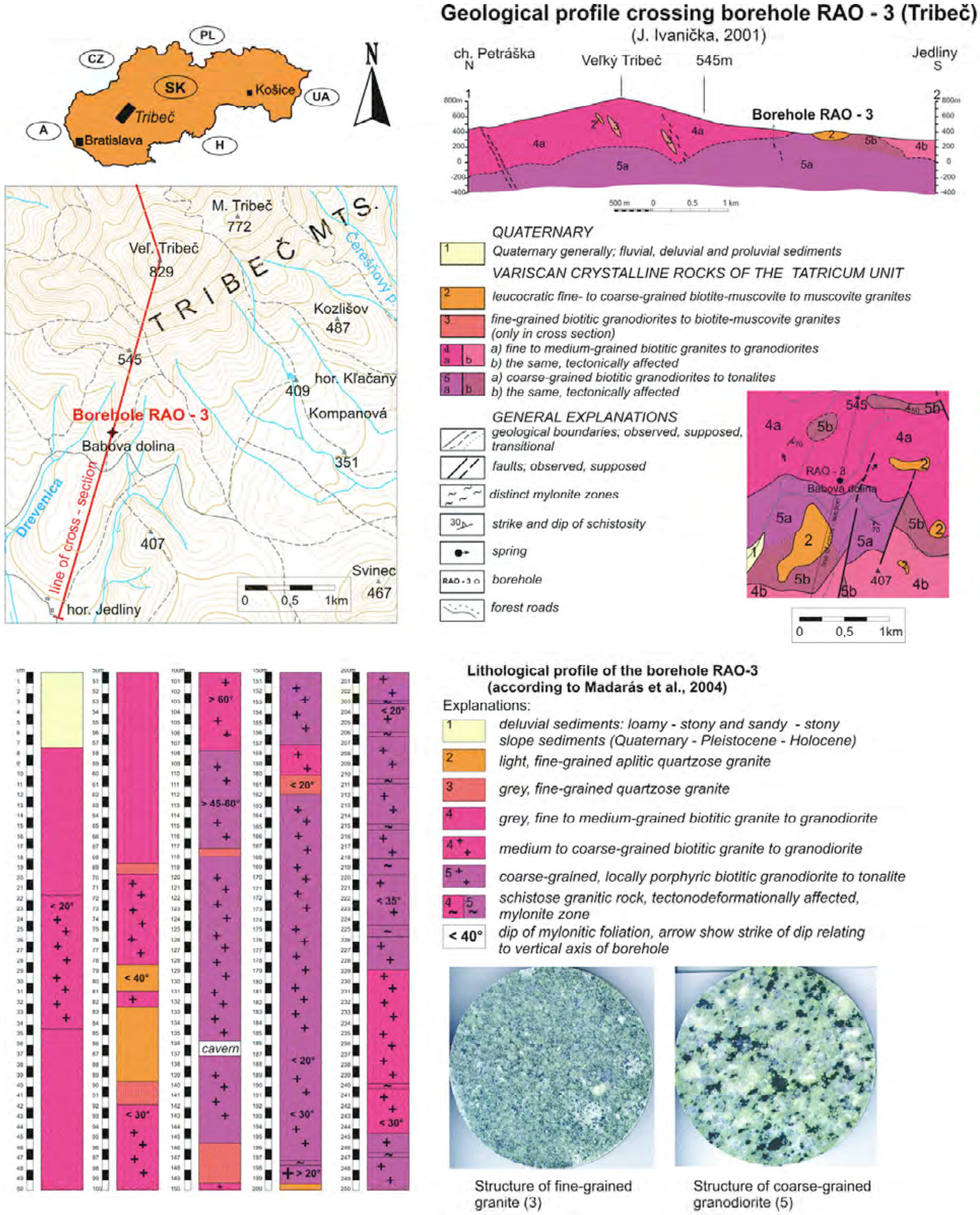

Figure 18.4. Example of geological research activities in the crystalline rocks complex of Tribec Mts. Borehole RAO-3.

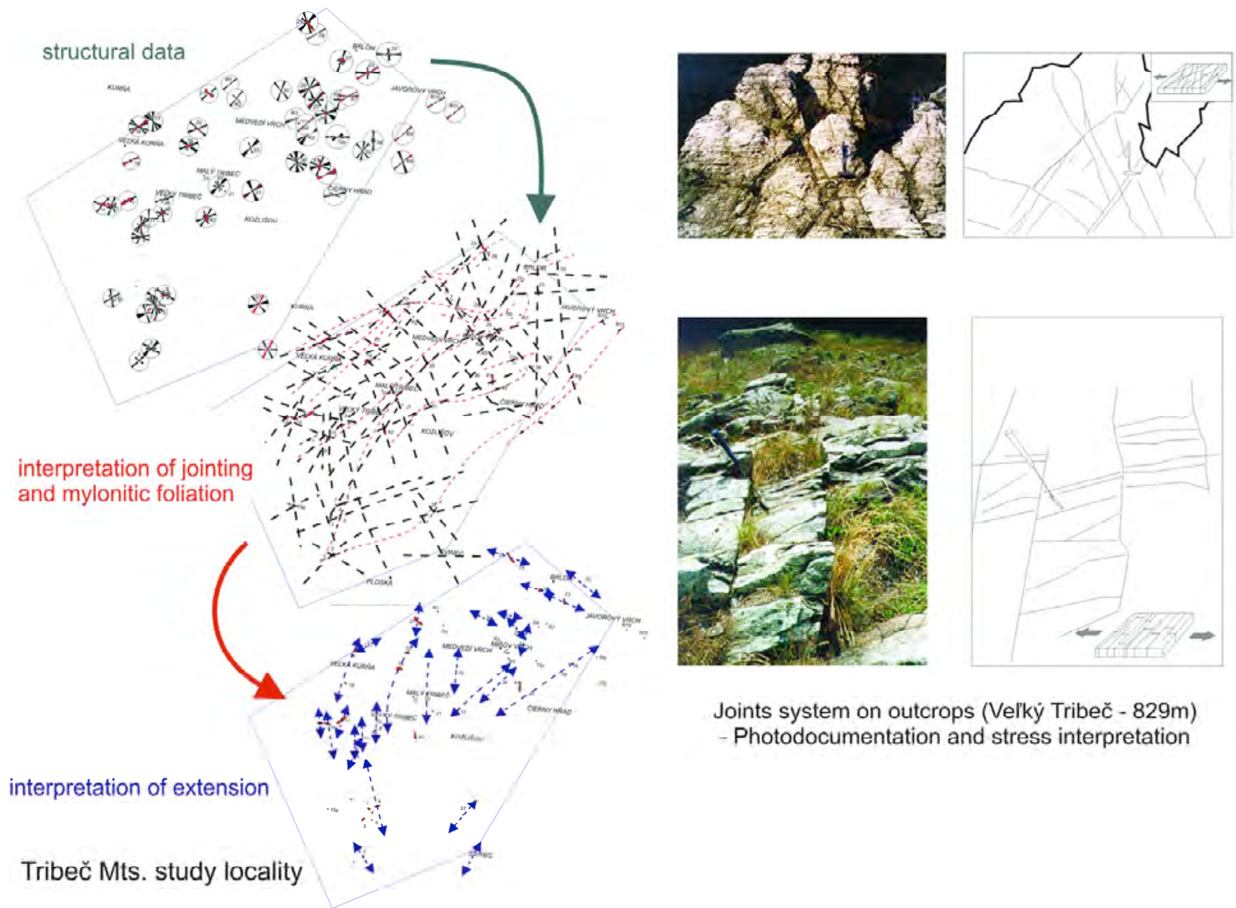

Figure 18.5. Interpretation of the neotectonic joints in granitoid rocks. The Tribec locality case of study.

and Rimavska kotlina Basin. From a lithological, structural, and spatial perspective, the most prospective host rocks appear to be two lithostratigraphic units: the Szecseny schlier of the Lucenec Formation (Egerian) and Lenartovce beds of the Ciz Formation (Kiscellian). These units form the principal mass of the basin filling. The predominant lithology in both formations is a mixture of siltstones and claystones. Maximum thickness of the Ciz Formation in the territory of Slovakia is 400–500 m, while the maximum thickness of the Lucenec Formation in Cerova vrchovina Upland is 1,300 m (1,100 m in the Rimavska Kotlina basin). The thickness of each formation increases from the northern margin toward the south, and the cumulative thickness varies between 1,400 and 1,600 m (Elecko et al., 1985; Vass et al., 1988; Kovacik et al., 2001; Nagy et al., 2004). Faults in the area form two orthogonal systems, the main system is oriented SW/NE and is related to the Carpathian uplift. The secondary system is oriented NW/SE. The rivers are believed to follow the fault system (Figure 18.6).

18.3.5. Geological Characteristics of Prospective Neogene Sedimentary Formations

18.3.5.1. Ciz Formation

The formation consists of five members that originated in a marine environment. It conformably overlies the Skalnik Member (pretransgressive river deposits), or unconformably the pre-Cenozoic crystalline basement. It is disconformably overlain by the Lucenec Formation. The Ciz Formation extends southward in Slovakian

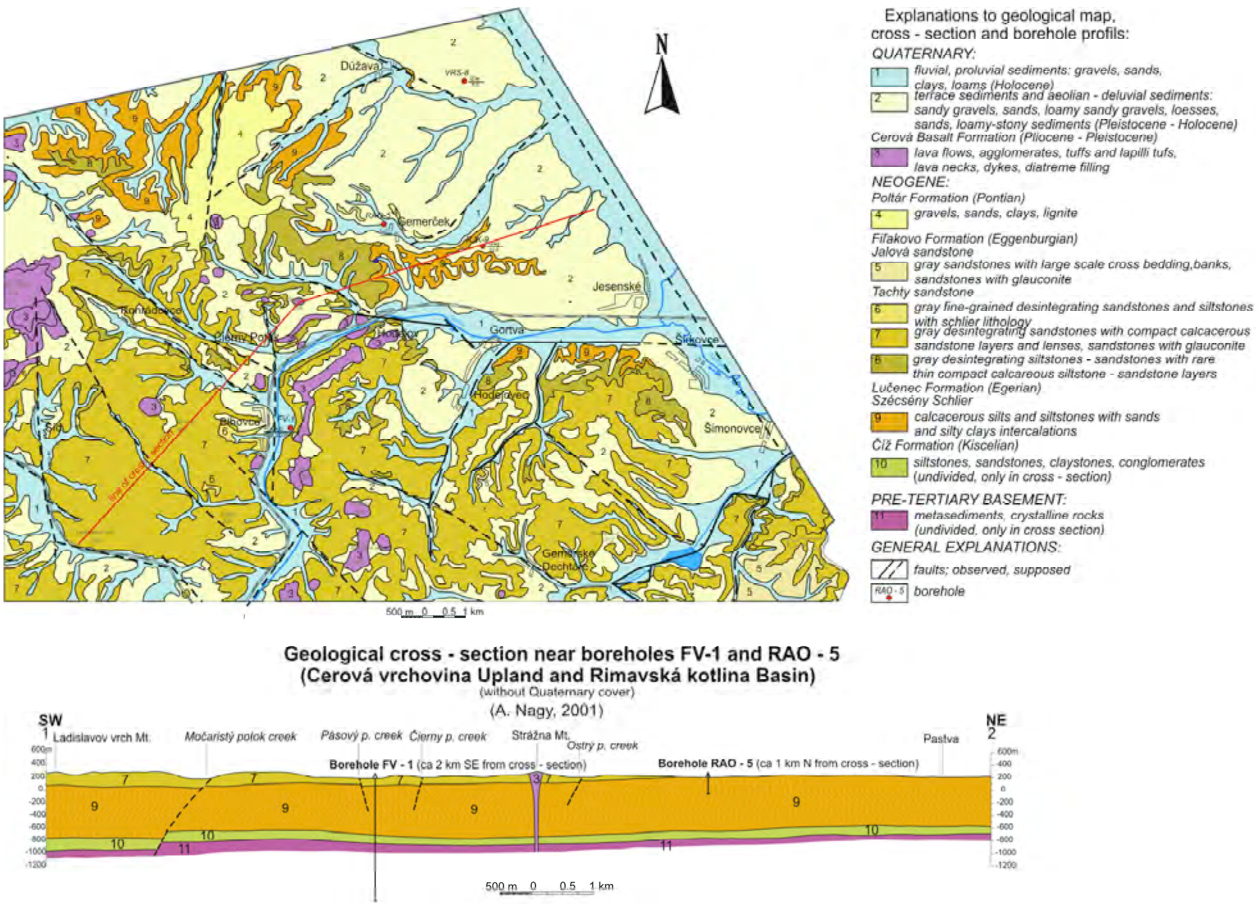

Figure 18.6. Example of geological research activities in the sedimentary rocks formations of the Cerova vrchovina Upland and Rimavska kotlina Basin—geological map and cross section of reconnaissance localities

Cenozoic depressions in the vicinity of the town of Sturovo. The formation continues into the territory of Hungary.

Lenartovce Member.
The predominant lithotype of the member is grey, green-grey, friable calcareous claystone and siltstone of slaty, or conchoidal jointing. In the lower part of the member, sandy laminations or lenticular bedding occur. The Lenartovce Member is 300–350 m thick, conformably overlies basal members of the Ciz Formation, and is overlain by the Lucenec Formation. Deposits of this member extend into the Rimava, Lucenec, and Ipel depressions, in the vicinity of the towns of Sturovo and Esztergom (Vass, 2002).

18.3.5.1. Lucenec Formation

The formation consists of six members that also originated in a marine environment. It disconformably overlies the Ciz Formation, or unconformably, the pre-Cenozoic basement. In the Cerova vrchovina Upland, the formation is overlain by the Filakovo Formation, and in the Lucenec and Rimava depressions, by the Poltar Formation. In both depressions, as well as the Ipel depressions, the formation crops out on the surface or is covered by Quaternary deposits. The formation occurs in all depressions of Southern Slovakia and in the vicinity of the town of Sturovo.

Szecseny Schlier is the dominant member of the Lucenec Formation. It is a friable calcareous siltstone of grey, green-grey color with slaty jointing. Layers of hard siltstone with intercalations of silty clay and fine-grained sandstone also occur. Thicker layers of sandstone occur within the lower part of the member. The Szecseny Schlier paleoenviroment, estimated on the basis of mollusks, was the shelf in an open sea of normal, or slightly hypersaline, salinity. The Schlier overlies the basal members of the Lucenec Formation and is overlain by the Filakovo Formation. Maximum thickness, verified in boreholes, is about 700 m (see borehole FV-1, Blhovce—Figure 18.7) (Vass et al., 1988, 2002), but one can estimate a greater thickness in the area of the Slovakian–Hungarian border (~1,300 m). The Szecseny Schlier extends southward in Slovakian depressions in the vicinity of the towns of Sturovo and Esztergom and continues to the border with Hungary.

The Lucenec Formation mineralogy is a complex mixture of clay minerals (smectite, illite, and kaolinite) with pyrite and calcite as other important minerals. A large fraction of quartz is also present. The dry bulk density is very high (2.2 g/cm^3), with only a small increase in density with depth, which is probably an indication of a high overburden during geologic time. A high dry density means a low porosity (14 to 20%), and as a result, a low hydraulic conductivity and low diffusion coefficients.

The possible permeability of the Szecseny schlier siltstones, as well as the permeability along main regional faults (with sporadic occurrence of acidic waters in springs), must also be taken into account and carefully investigated. Preliminary measurements of hydraulic conductivity performed on remolded samples (i.e., samples made by recomposing and compacting samples to their *in situ* density) yield values as low as 10^{-10} m/s and even 10^{-11} m/s. Taking into account the high density and low porosity of the Szecseny Schlier, these are more reliable values.

According to these preliminary data, one can assume that diffusion is the dominant transport mechanism. This result, caused by the well-preserved chemical character of pore water developed by sea sedimentation, is extremely important for performance assessment of the site and is a very critical property. This sea-sediment origin has led to high salinity, with a chloride content of about 12,500 ppm (see borehole FV-1—Figure 18.7).

The Szecseny Schlier geological formation has favorable properties as a potential host rock for geological disposal of radioactive waste, especially given its homogeneity, its high density and low porosity, good geomechanical properties, low hydraulic conductivity, and high thermal conductivity. However, all these parameters are currently based on a relatively small number of observations or derived from secondary information. Therefore, they must be confirmed by further detailed site investigations.

18.3.6. Safety Analysis

The starting point of a safety analysis was the critical review of concepts for a national deep repository and model approaches prepared by the Nuclear Power Plant Research Institute Trnava, Plc. (VUJE) (Slavik et al., 1998). These were examined for various safety issues, with the aim of identifying and assessing their applicability for conditions in the Slovak Republic.

Primarily, scenario-development methodologies were described (Mrskova et al., 2000). Emphasis was placed on points of international consensus and global issues related to deep disposal. It was agreed that an internationally accepted scenario-development methodology, supporting tools, and an international features, events, and processes (FEP) database should be implemented. VUJE selected appropriate conceptual and mathematical models for safety assessment of individual repository subsystems, as well as an assessment of how extensive modifications and adaptations must be carried out.

The importance of natural analogs for deep geological disposal has been briefly described, especially in selecting engineered-barrier materials and in stressing the importance of safety in SNF and HLW disposal. Results from this work have provided preliminary information about worldwide accepted experience and knowledge in safety analysis, both in establishing conceptual models and in developing methodology.

18.3.7. Public Involvement

Slow progress in a number of other countries in obtaining public acceptance of plans for various types of nuclear activity (including site selection for waste repositories) has led to a substantial reassessment of approaches to issues related to public awareness and involvement (Hudson, 1999). In general, there has been a move away from the "decide–announce–defend" method towards a more consensual approach to decision making. Although progress in many countries is still

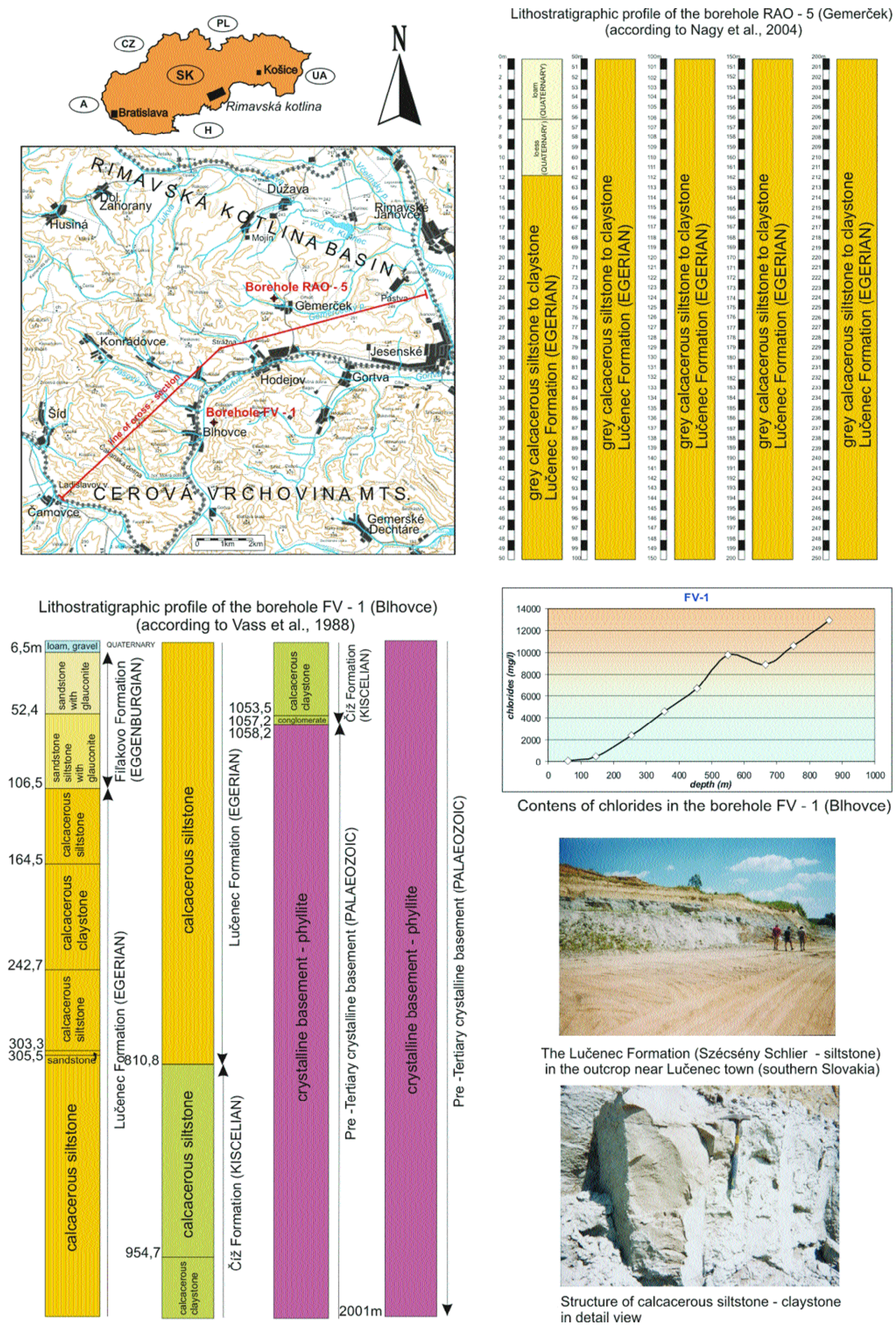

Figure 18.7. Example of geological research activities in the sedimentary formations of the Cerova vrchovina Upland and Rimavska kotlina Basin—Boreholes RAO-5 and FV-1

slow, there is evidence that prospects are enhanced by closer and earlier involvement with host communities. We identified key features of that involvement as:

- Better-focused and more open information
- Closer contact between the public and the nuclear community
- Real public influence and control

Five key aspects of public involvement need to be addressed in any public awareness program: information, communication, participation, acceptance, and compensation. Current activities were focused on:

- Informing the public about radioactive waste management (presentation of nuclear energy and radioactive waste management in the media, changing negative attitudes and responding to arguments against nuclear energy, and identifying constructive ways to respond to opposition).
- Investigating the socio-economic effect of deep-geological-repository development and its potential impact on the public (how to mitigate or eliminate any negative impact and maximize the benefit for the host community).
- Developing a program for public involvement in decision-making and information processes during repository development.
- Publishing a brochure explaining the alternatives related to managing the back end of the fuel cycle, possible solutions for the Slovak Republic, basic principles of deep geological disposal, and the national program for the general public.

18.3.8. Legislation

Decom Slovakia, Ltd. reviewed worldwide legal documents relevant to deep geological disposal issues with the aim of identifying the appropriate legal environment for repository development in the Slovak Republic. Existing relevant Slovak legal and regulatory documents should be taken into account during repository development.

Present activities are focused on evaluating the existing radioactive waste management infrastructure in the Slovak Republic, to ensure that necessary activities, task definitions, and responsibilities (as prerequisites of quality assurance for the repository program) are known.

18.3.9. Coordination

One of the basic activities of a coordination center is transferral of information among contractors, subcontractors, managers, administrative and regulatory bodies, and the public. An archival system for storage and maintenance of documents and data for the DGR program in the Slovak Republic has been proposed.

Yearly published annual reports summarize results achieved and the updated status of the program of repository development. Reports are intended for managing and administrative bodies and the public. Decom Slovakia, Ltd. has prepared detailed plans for future periods of the repository program (Matejovic et al., 1998).

We also have prepared a preliminary study devoted to selected issues important for the preparation of intent to construct a repository in the Slovak Republic (and an associated environmental-impact assessment) according to requirements of Act No 127/1994 (Matejovic et al., 2000). As part of this effort, we compared various management options and discussed selection of the preferred disposal method and alternatives, the consequences of zero alternatives, and other issues in preparing to initiate the environmental impact assessment process (EIA). Attention was given to European Commission (EC) Directive No 85/337/EEC and amending Directive 97/11/EC for the EIA process related to radioactive-waste repositories and storage facilities, their interpretation as well as internationally accepted principles, recommendations, and ethical aspects (IAEA and Nuclear Energy Agency [NEA] documents).

Brief information for the general public about deep geological disposal of spent fuel and HLW was prepared and issued as a booklet. A second booklet, devoted to the status of a DGR development program in the Slovak Republic, was also prepared.

18.4. INTERNATIONAL COOPERATION

In addition to Slovak institutions, relevant groups in the Czech Republic (source term, near field, and container design) and the United Kingdom (public involvement) have participated in the Slovak disposal program. Nagra (Switzerland) has reviewed plans for further development of a deep repository. Seminars and meetings dedicated to information exchange between our Czech colleagues involved in the Czech disposal program, and our group. The most important is Regional Seminar on

Radioactive Waste Issues, that is organized yearly. Slovak Electricity, Plc., and Belgatom have signed an agreement for cooperation on deep geological disposal plans. A number of meetings of experts and exchanges of knowledge between counterparts (Geological Survey of Slovak Republic, SCKCEN Belgium, and Belgatom Belgium) were carried out during these activities. The effort was focused mainly on geological investigations concerning DGR siting, site characterization, especially in the study of clay-environment geochemistry, scenario development, and hydrogeological modelling.

Slovakia has taken a very active role in the field of international cooperation concerning geological disposal. Under the 6th Framework Programme of the European Commission, a consortium of Decom Slovakia, Ltd. and ARIUS (Association for Regional and International Underground Storage—Baden, Switzerland) act as leaders in the project entitled Support Action—Pilot Initiative for European Regional Repositories (SAPIERR). This is a pilot initiative to help the EC begin to establish the boundaries of the issues, by collating and integrating information in sufficient depth to allow concepts for potential regional options to be identified and the needs of a new research and technology development to be scoped. SAPIERR brings together Member States and Candidate Countries, who want to explore the feasibility of regional European solutions. The work is aimed especially at establishing the boundary conditions for such collaboration and the implications for an enlarged European community. A rapid geopolitical development in Europe, as well as the socio-political reservations concerning multinational repositories expressed by some countries, may well have been overcome by the time of actual construction.

In addition, the environmental and economic advantages of these solutions may also prevail over the political problems. The project involves a working group of representatives from interested organizations to conduct data gathering and analysis. Scenarios and possible concepts for European storage and disposal are to be identified, as well as related needs to propose mechanisms for developing strategy options and future European Commission (EC) research programs. Results will also be reported to the EC at an international seminar at the conclusion of the project. A commitment to participate in the working group and to supply relevant national data has already been obtained from 21 organizations from 14 countries. Regional repositories outside of Europe are also of interest, but have not been studied. SAPIERR will put the EU in a leading position to provide advice and, possibly, services to other countries. An important overarching objective is to ensure that consideration of regional repository concepts in the EU proceeds in harmony with national programs. It is important that these two strategies are understood as complementary approaches, intended to ensure that safe and secure geological disposal is ultimately available to all members of the enlarged European Union. Already, the working group includes members from all EU member states and candidates, as well as associated countries that have a direct interest in exploring regional solutions. A number of documents have been prepared for this project that are published on the web, and more information can be found at: www.sapierr.net.

In 2003–2004, VUJE, Plc. and Decom Slovakia, Ltd. participated in project COMPAS (Comparison of Waste Management Alternative Strategies for Long-lived Radioactive Wastes). The project is part of the Fifth Framework Program of the European Commission. COMPAS was an excellent opportunity for national nuclear waste management experts to collaborate in order to evaluate and compare the different approaches and strategies used in the management of radioactive wastes that have been developed in EU Member States, Switzerland, and the Applicant Countries (Dutton et al., 2004). The output of this project provided information for policy makers, regulators, those responsible for implementing policy, and the public on issues associated with long-lived radioactive wastes and the different strategies that are being adopted to manage the wastes. Nearly all the countries that have participated in this project have adopted deep geological disposal as the preferred long-term management option for SNF (that will not be considered for reprocessing), HLW, and other long-lived waste. This option has been selected on the basis that safety of the stored material cannot be established for a definite period. However, there are different perceptions of the urgency for implementing deep geological disposal that depend on national policies and social, logistical, and economic issues. The incorporation of the requirement for disposal in national legislation has been an important factor in its early implementation in some countries.

The Harmonization of European Regulations and Legislation on Disposal (HERALD) project, which has been in progress since January 2005, has been organized to support the issue of internationalizing geologic disposal from the legislative point of view. Project activities are performed by an international consortium led by Decom Slovakia, Ltd., under contract with the European Commission. The main objectives of the project are: (1) an analysis of the national legislation needed in the underground disposal of radioactive waste and SNF in 25 countries within the European Union (plus Romania and Bulgaria), and (2) the basic problem of developing an effective approach to the harmonization of regulations that will be required.

The Slovak Republic has also been involved in technical cooperation (TC) projects headed by IAEA. These international projects were focused mainly on geological topics of siting and site characterization of a potential DGR. During the past two years, the Geological Survey of Slovak Republic participated in two IAEA TC projects:

SLR/4/009: Site Selection Process for a Deep Geological Repository in Sedimentary Rock Formations, Phase I. The overall objectives can be summarized as follows: identification of key geological problems in the study area, definition of scientific techniques for an optimal solution to these problems, identification of a pilot area in the South Slovakian Neogene basin, and scientific solution of problems in the pilot area.

INT/9/173: IAEA Network of Centers of Excellence—Training in and Demonstration of Waste Disposal Technologies in Underground Research Facilities. The main objective is the transfer of knowledge between counterparts with different developed levels in a DGR program, and experience with experts training courses using existing underground research facilities.

18.5. FUTURE ACTIVITIES

After a recent period of limited project activities, a restarting of research and development work is expected in the near future. Siting, safety analyses, and coordinating and supporting investigations will be the primary areas of activity. This should lead to a candidate site that is acceptable to the public and a feasibility demonstration of the proposed construction, operation, and closure of a DGR.

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Chapter 19

Radioactive Waste and Spent Fuel Disposal in Slovenia—Status and Trends

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19.1. INTRODUCTION

Over the last ten years, radioactive waste management in Slovenia has been primarily focused on low- and intermediate-level waste (LILW), which, because of the limited storage capacity at the Krško Nuclear Power Plant (NPP) (the only nuclear power plant and the main waste producer in Slovenia), requires an expedient solution. After an unsuccessful attempt to locate a repository for LILW in the early nineties a new siting project was initiated in 1996, based on a mixed-mode site selection procedure and step-by-step approach.

Simultaneuosly with the siting project, the disposal concept and the disposal technology for LILW has been well elaborated, safety aspects have been analyzed, and a preliminary cost estimate has been made. Implementation of the siting process for an LILW repository is now in full swing. According to the agreed-upon time schedule, the site is expected to be confirmed by 2008 and the repository put into operation by 2013.

On the other hand, the disposal of spent nuclear fuel (SNF) has been put aside. Long-term SNF management followed the recommendations of the Spent Fuel and High Level Waste Management Strategy (adopted by the Slovenian Government in 1996) postponing the decision on a SNF disposal solution until the year 2020. Based on this strategy, no significant action had been taken with respect to SNF and high-level waste (HLW) disposal, until last year.

A major step forward, regarding the long-term management of SNF was made only after the dual ownership of the Krško NPP had finally been clarified and agreed upon between Slovenia and Croatia. According to the requirements of this agreement, in effect since March 2003, future liabilities of the NPP, such as radioactive waste disposal and decommissioning of the facility, need to be known well in advance, for adequate time to prepare the tools and provisions for their future handling. For this purpose, the disposal strategy was re-investigated in 2004, and the disposal options for SNF were more seriously addressed.

19.2. PROGRESS IN LILW REPOSITORY SITE SELECTION

An amendment to the new nuclear act, adopted in March 2003, requires that the site for a repository of LILW be known by 2008 and the repository in operation by 2013. To achieve this goal, the suitable site(s) should be identified in 2005, and the site characterization should be completed in 2007.

The site selection procedure was already established in 1996. It is based on a mixed-mode site selection approach that is a combination of technical screening and volunteer siting. Full recognition of public participation and local-community involvement in the decision-making process is the essential component of this procedure (Mele and Železnik, 1998). Its inclusion into the site selection process is presented in Figure 19.1. The

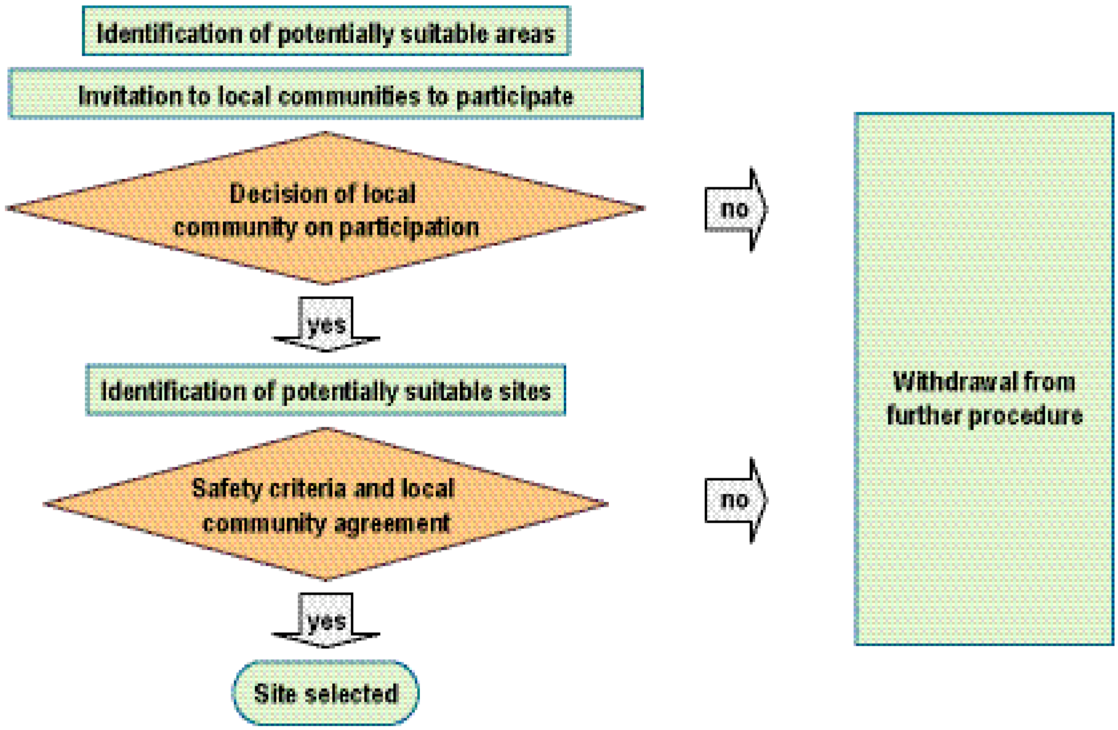

Figure 19.1 Participation of local communities in the site selection process for an LILW repository

participation of local communities in the process is based on their free decision—participation can be stimulated but not forced. A volunteer approach is respected throughout the entire procedure.

In 2001, the area survey stage was completed. The results, already reported in Tomše et al. (2001), showed that about 15% of Slovenian territory is potentially suitable for underground disposal, and almost 45% for surface disposal. The most suitable natural predisposition for LILW disposal was identified as surface disposal in rock with low permeability, or as underground disposal in plastic rock with low permeability.

After the area survey stage, the site selection process continued towards the identification of potentially suitable sites. This is the most difficult step, requiring extensive communication and negotiations with the local communities in the area of interest. The Slovenian Agency for Radwaste Management (ARAO) has decided to facilitate communication and discussions with the local communities through an independent mediator. The mediator for the site selection process was introduced in February 2002. During the initial period, when the boundary conditions and the benefits of local-community participation had not yet been fully clarified, the mediator's activities were directed mainly towards ensuring sufficient public and local community involvement in further site selection phases.

However, a breakthrough was made at the end of 2003, when the legal basis for financial compensations to the hosting local community was finally adopted (Železnik et al., 2004a). The Governmental decree offers the hosting local community a substantial yearly financial compensation for the full period of LILW repository operation. A tenth of the yearly financial compensation is also available to local communities during field investigations at the site and throughout the repository construction period.

With this mechanism for stimulating the participation of local communities in the process, the site selection project received new impetus. In November 2004, as part of the site selection process, the Ministry of Environment and Spatial Planning officially started the spatial planning procedure for the LILW repository. The entire spatial planning procedure, aiming at the preparation of a detailed site development plan for a repository, is tailored to the mixed-mode site selection process, respecting the volunteer participation of local communities. The first step in this administrative procedure was the offical invitation to local communities to participate, with clear instructions and rules for participation and publicly announced benefits to the participating parties. The call for tenders has no closing date. It remains open until the bidding process ends successfully. The first results of this invitation will be known only after first evaluation of bids in the spring of 2005, but so far the response among local communities is encouraging. However, the critical point will come after bids are opened and the registered communities are announced. If a local community enters the process without sufficient support of the local population, further participation may be jeopardized or even cancelled.

Assuming a successful bidding process, two approaches have been developed for the continuation of the process:

1. If there are more than three bids, a pre-study, comparing the candidate sites on the basis of safety, technical, and economic criteria, will be prepared, and three sites will be selected for further investigation and evaluation.
2. If there are three or fewer bids, all sites will be investigated and evaluated.

After detailed field investigations and favorable safety analysis, the site will be confirmed—only if the local community accepts hosting of the repository. If the site selection is completed successfully, the LILW repository will be in operation by 2013. Since this repository is intended for operational LILW as well as for LILW waste from the decommissioning of the nuclear power plant and other nuclear facilities, the repository represents the final solution for all short-lived LILW in Slovenia. Small quantities of long-lived LILW will await the disposal solution for SNF and high level waste.

19.3. WASTE CONSIDERED FOR GEOLOGICAL DISPOSAL

In Slovenia, only two nuclear facilities are generating SNF: the NPP at Krško and the research reactor TRIGA. While the research reactor is already nearing the end of its life expectancy and is expected to end its operation in the next decade, the NPP has reached only its mid-lifetime, and an extension of lifetime is a possible option. Long-term management of SNF from these two facilities follows two different approaches. While all SNF from the TRIGA research reactor was successfully shipped back to the USA in 1999, within the U.S. fuel take-back program—and the remaining fuel used for present operation of the research reactor is expected to be returned to the USA before the newly announced expiration date for spent fuel acceptance—the SNF from the Krško NPP remains the concern of the two owners, Slovenia and Croatia.

Quantities of SNF are small. At the end of 2004, the total amount reached about 310 tons of heavy metal. It has been estimated that by the end of its scheduled lifetime, the Krško NPP will have produced 1,500–1,600 SNF assemblies, or 615–660 tU. At present, the SNF is stored in a pool at the Krško NPP. The pool has recently been successfully re-racked to provide sufficient capacity for plant lifetime and even for possible lifetime extension.

19.4. SNF MANAGEMENT IN SLOVENIA

19.4.1. Long-Term SNF Strategy

Long-term management of SNF follows the recommendations of the Spent Fuel and High Level Waste Management Strategy (Strategija ravnanja, 1996), prepared in 1996 and adopted by the Slovenian government. The development of this report took place simultaneously with the first decommissioning plan of the Krško NPP (NIS Ingenieurgesellschaft mbH, 1996). Both documents were established at the time when Slovenia accepted a long-term strategy of withdrawal from further peaceful use of nuclear energy after the year 2023—the projected lifetime of the NPP—and upon unsuccessful attempts at siting the LILW repository. The strategy was also strongly influenced by the small quantities of SNF generated in Slovenia, anticipated high disposal costs, and the unresolved question of ownership of the NPP at Krško between the two investors, Slovenia and Croatia. This issue was the subject of negotiations between the two governments for more than a decade, and sharing the SNF and other radioactive waste remained an open possibility.

In this strategy, various long-term management and final-solution options for SNF were investigated. The reprocessing of SNF was not found to be a recommendable option; the strategy proposes direct disposal of SNF assemblies. Based on the anticipated lifetime of the plant, the strategy recommends that the final solution be postponed until the end of plant operation. A strategic final decision should be reached by the year 2020. In accordance with the requirements of the NPP Krško Decommissioning Plan from 1996, the SNF repository should be provided by the year 2050.

19.4.2. Joint Decommissioning and Waste Disposal Program

According to the agreement between Slovenia and Croatia on the ownership and exploitation of the Krško NPP, the decommissioning and disposal of SNF and LILW from the NPP is the responsibility of both parties. Regardless of shared ownership of waste, the agreement puts forward a joint single disposal solution for LILW as well as for SNF, but the details are left open. Clearer elaboration of these responsibilities is given in the program for the decommissioning and disposal of radioactive waste from the NPP, jointly prepared by the Slovenian and Croatian waste management organizations in 2004 (Lokner et al., 2004).

The Joint Programme was primarily aimed at providing a good estimation of future liabilities of the NPP (Železnik et al., 2004b). Cost estimates for Krško NPP decommissioning, disposal of LILW, and long-term management of SNF are required as a necessary input to the two national funds which, according to the agreement, take the responsibility of collecting the funds for implementing the program. For this purpose, the long-term SNF management—as the most important future liability—had to be addressed seriously, and the disposal options investigated in more detail.

19.5. DEVELOPMENT OF A REFERENCE SCENARIO FOR SNF DISPOSAL

Developing a disposal concept for small quantities of SNF, and for limited financial resources, requires a very rational approach. It is not expected that the small nuclear programme will carry on extensive research programs in the field, or make large investments in technology development. It is more likely that it will follow the best practice available and try to optimize it for specific needs. However, direct transplantation of best practice without adequate modifications should also be avoided.

In our case, instead of developing a disposal concept from scratch, the Swedish KBS-3 concept of disposal in hard rock and its cost analysis method, developed by Swedish Spent Fuel Management Agency, SKB, has been taken as a model concept. The decision is based on the well-developed Swedish waste disposal concept and their highly advanced cost-assessment methodology, with well-defined cost elements. However, the applied model disposal concept was developed for much larger SNF quantities and a much longer operating period than in our case. Many adjustments were thus required before it fitted our needs, and some limitations were also introduced. Wherever possible, further optimization was made in developing our concept.

19.5.1. Requirements and Limitations

The reference scenario covers only SNF and HLW management at the Krško NPP. It assumes that the operation of the Krško NPP will end in 2023. It also assumes that all SNF will be disposed of in a single deep geological repository. Regarding the timing, two options are analyzed: the repository being available shortly after the plant shutdown, and the repository being available a few decades after plant shutdown. In the first case, no interim storage of SNF is needed. After a few years of cooling, the SNF is relocated directly into the repository. In the second case, a longer storage period (dry or wet) is foreseen before the SNF is finally disposed of. Alternative solutions to the repository development (multinational repository, export of SNF) are being investigated as well, and a comparison is made between the different options.

Furthermore, in developing the disposal concept, additional assumptions were taken into account:

- Only direct disposal of SNF is considered (no reprocessing).
- The repository is developed for a hard rock environment at a depth of 500 m.
- The capacity of the repository is 620 tons of heavy metal, estimated to be generated over the plant's lifetime, as well as for a small quantity of HLW generated during Krško NPP decommissioning (~16 m^3).

In accordance with the operating lifetime of the NPP, the reference scenario considers two alternatives:

- Alternative 1: Start of repository operation in 2030. Until then, the SNF is stored in the SNF pool on the premises of the NPP.
- Alternative 2: Start of repository operation in 2050. Until then, interim dry storage of SNF is applied.

Table 19.1. Properties of host-rock formation	
Rock Property	**Estimated Value**
Rock	Diorite and tonalite, dacite, tuff, schist and phillyte, mica schist, gneiss, amphybolite, serpentinite
Thermal conductivity	2.6–3.4 W/m/K
Temperature gradient	20–35 K/km
Initial temperature at ref. depth	22–28°C
Rock heat capacity	700–1,000 J/kgK
Modulus of elasticity	5–50 GPa
Modulus of deformation	0.8–5 GPa
Unconfined compressive strength	10–100 MPa
Poisson coefficient	0.2–0.3
Water permeability	10^{-8}–10^{-12} m/s; within fault zones: 10^{-5} m/s

19.5.2. Host Rock

No site investigations for a deep geological repository have been carried out in Slovenia, and no specific data for geological disposal are available. The reference scenario is made for a generic location in hard rock media. Where needed, a set of most probable host rock properties, listed in Table 19.1 and derived from area-specific igneous and metamorphic rock properties from the northeast part of Slovenia, is applied (IBE, Consulting Engineers, 2004). The data originate from past investigations for other purposes.

Hydrogeological Properties

Most igneous and metamorphic rocks are overlain by a relatively thick weathered layer, characterized by higher permeability. The weathered layer can be up to 50 m thick, especially on exposed ridges, while it usually is about 10 m thick in valley bottoms. The permeability coefficient of the weathered layer is usually between 10-6 and 10^{-8} m/s. The permeability coefficient of deeper, unweathered parts of the rock is usually around 10^{-8} m/s, and occasionally the values range between 10^{-7} and 10^{-9} m/s. It is estimated that the permeability coefficient of sound magmatic and metamorphic rocks ranges from 10^{-8} m/s to approximately 10^{-12} m/s. Fractured and weathered rock zones have higher permeability and should be avoided in HLW disposal construction.

Geotechnical Rock Quality

The geotechnical rock quality is estimated to be good or even very good. It is expected that no support will be required for a span of 10 m in the underground structure. In some parts of the repository zone, rock of fair quality will require some support, but generally, no invert construction is expected to be necessary. Within the primary rock stress field, some anisotropy can be expected, owing to the nearby alpine orogenesis.

Thermal Properties of the Rock

At the depth of a geological repository, rock temperatures are expected to be higher than near the surface. Rock temperatures at a depth of 500 m, estimated from the available data on geothermal gradients in Slovenia, are presented in Figure 19.2. Depth temperatures in areas with igneous and metamorphic rocks range between 22°C and 23°C.

Heat conductivity for individual types of magmatic and metamorphic rock in Slovenia, important for heat transfer from canisters into the surrounding geological environment, are derived from available measurements. The values obtained are as follows:

Keratophyre	2.6 W/mK
Amphibolite	2.6 W/mK
Tonalite	2.7 W/mK
Gneiss	3.4 W/mK

19.6. DISPOSAL CONCEPT

The reference scenario is limited only to those elements that are directly connected to disposal activities. It addresses only packaging and disposal of SNF. It does

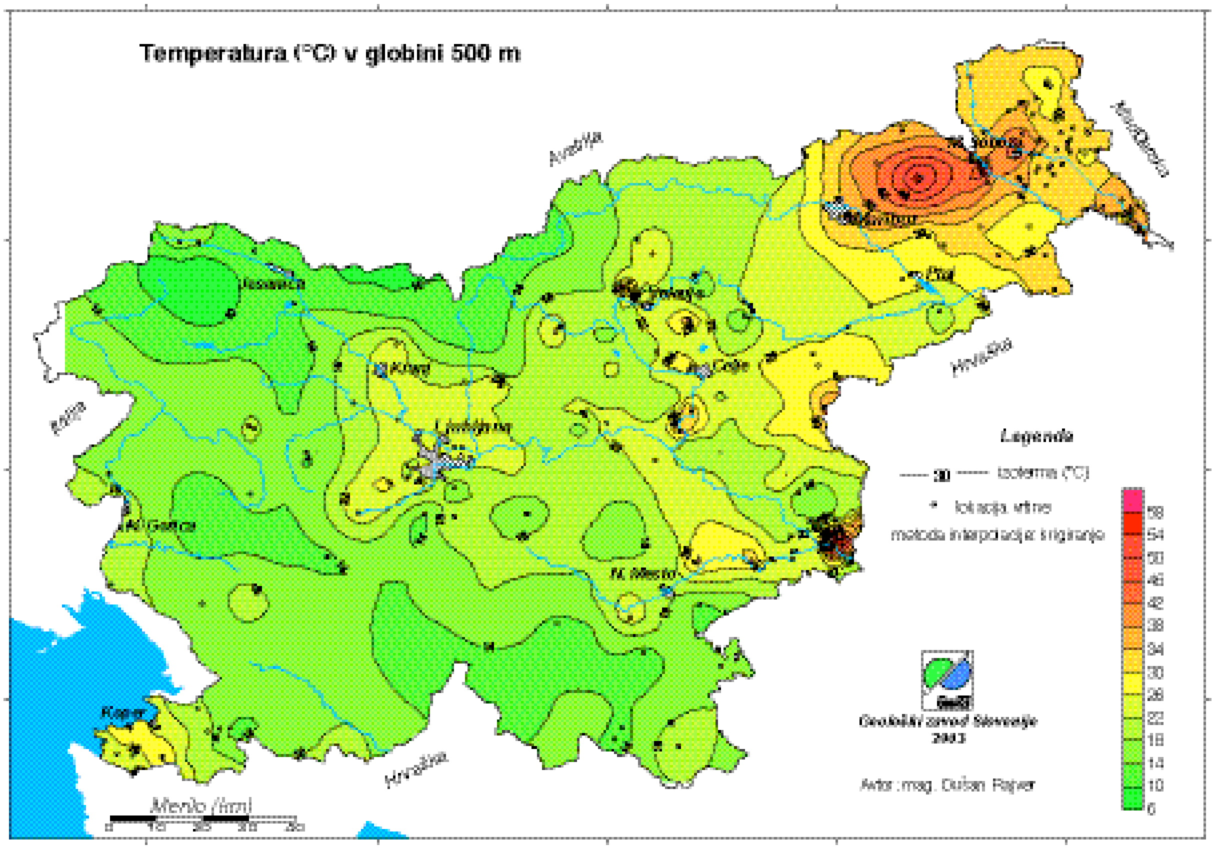

Figure 19.2. Map of rock temperatures (in °C) at a depth of 500 m (Source: Dušan Rajver, Geological Survey of Slovenia, 2003)

not address either predisposal of SNF management or off-site infrastructure.

Encapsulation of SNF is an important and quite costly element of the whole disposal concept. For a small quantity of waste, the construction of an encapsulation plant is economically not justified. However, in the present situation, very little room for optimization was found in this area. Since no packaging services are available, in addition to the repository, the disposal concept also includes an encapulation plant for SNF. Only the manufacturing of SNF canisters is excluded. It is assumed that the market for canister supply will exist at the time of SNF disposal.

19.6.1. Encapsulation of SNF

It is assumed that SNF will be encapsulated according to the Swedish concept. Fuel assemblies will be inserted and sealed into massive copper canisters. The canister is a 1 m diameter, 5 m high cylinder with a 5 cm thick anti-corrosion overpack of copper. From the inside, it is reinforced by a cast iron insert which can accept four PWR fuel assemblies. The weight of a canister filled with SNF is about 25 tons. Another type of canister, a steel canister, was also taken into consideration, but a rough cost comparison of copper and steel canisters shows no significant difference. To remain consistent with the model disposal concept, the copper canisters were adopted (Figure 19.3).

Based on provisions in the reference scenario, canisters are procured by an outside supplier rather than being manufactured locally. Their quantity requirements are based upon residual heat data of the SNF assemblies from previous cycles and upon estimated residual heat data of SNF assemblies from future cycles (IBE, 2004). Since the heat release in the year 2030 will still be quite high, most of the canisters can be filled with only two or three fuel assemblies instead of four. Also keeping in mind that a few dozen canisters have to be purchased for

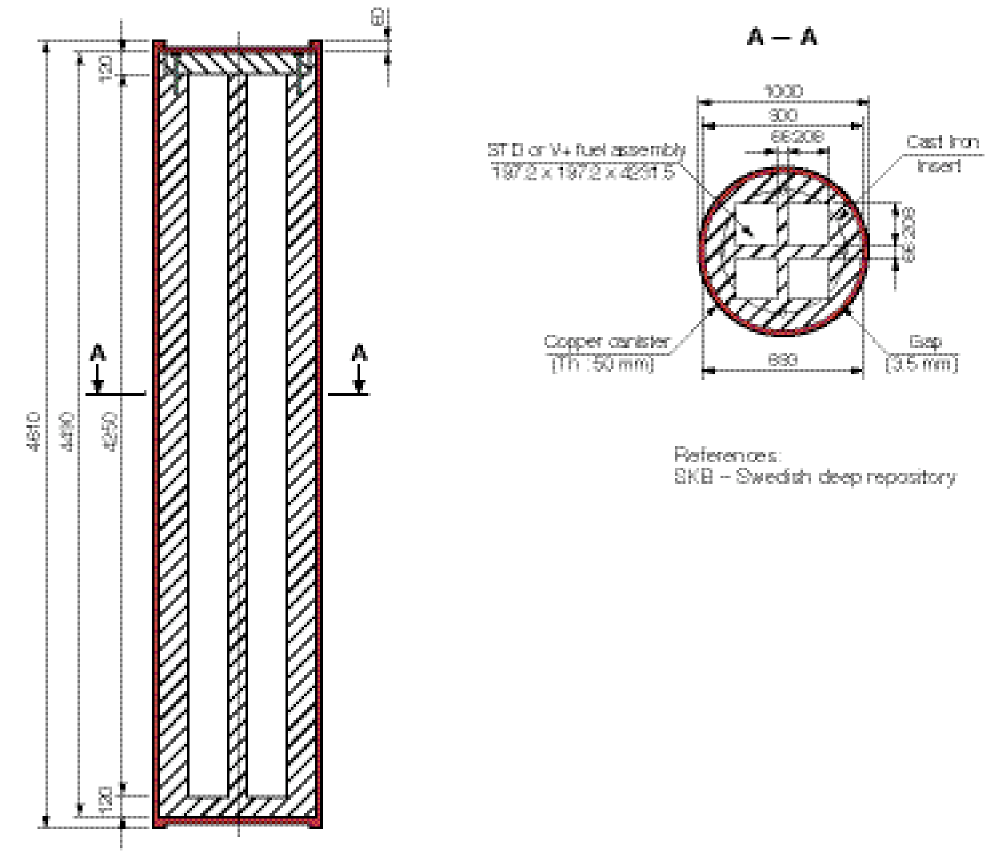

Figure 19.3. Copper canister with four positions for fuel elements

training and testing purposes (welding technology), for the first alternative repository in 2030, 750 canisters are needed, and for the second alternative repository in 2050, 450 canisters. Procurement is foreseen as a step process, the first canisters to be dispatched two years before start of the repository operation and then yearly during the operating period.

Encapsulation is planned to be performed in the encapsulation plant located at the repository site. The plant will consist of units for acceptance of transport containers, for encapsulation, and for dispatching and transportation of canisters to underground disposal facilities, office building, and auxiliary facilities and systems.

The encapsulation plant has a capacity of 200 canisters per year. Its operation should start one year prior to start of the deep repository operation. The overall operating period of an encapsulation plant depends upon the number of canisters to be filled up and welded. After a two-year test operation, personnel training, and test welding of a few dozens of canisters, encapsulation of all spent fuel assemblies would be completed within a six-year period in Alternative 1, and within a four-year period in Alternative 2.

At the end of operations, the encapsulation plant will be dismantled and its contaminated parts processed as radioactive waste. No substantial quantities of radioactive waste are expected. The dismantling should be finished within three years.

19.6.2. Repository

The deep geological repository consists of underground facilities and aboveground facilities, which are indispensable for normal underground repository operation. Since the repository capacity is small and the operation period very short, the aboveground buildings (e.g., offices, information center, garage) are reduced and simplified as much as possible. The surface facilities are

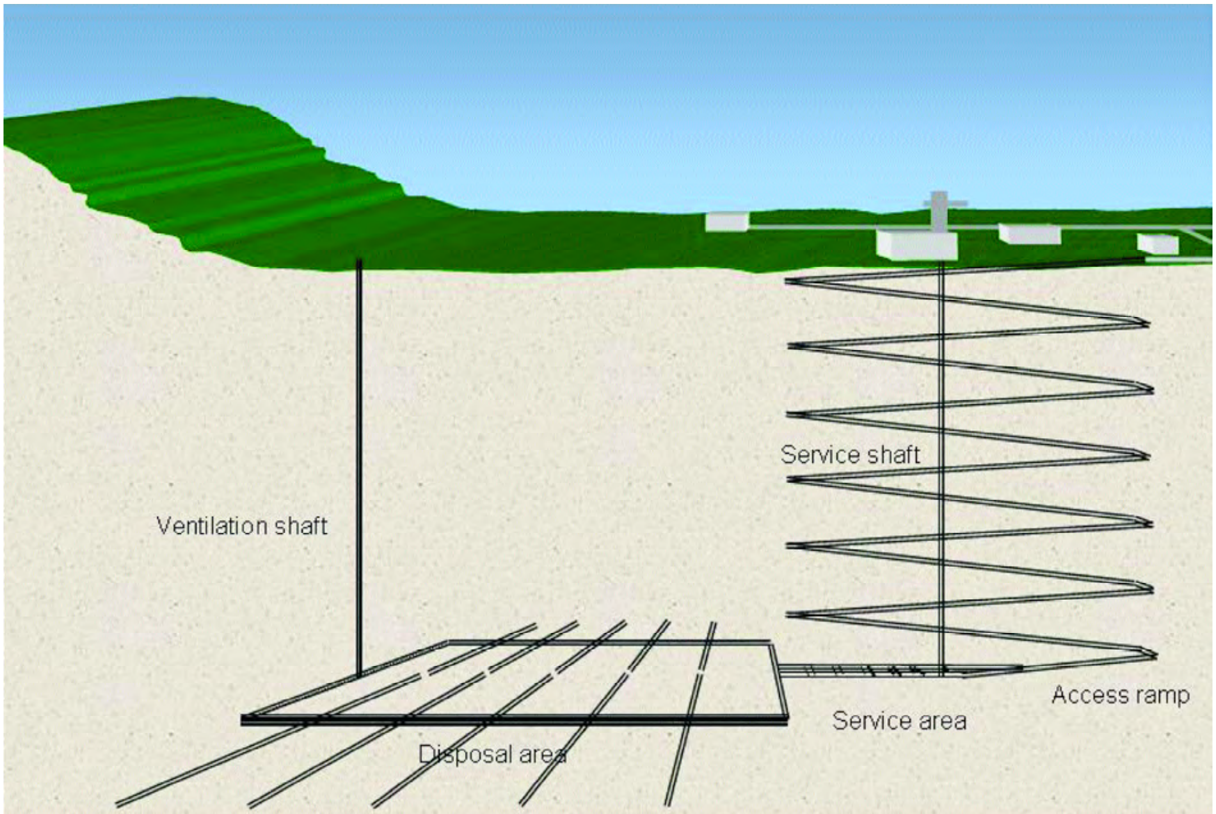

Figure 19.4. Deep geological repository (Alternative 2)

connected with the underground system through access shafts and a waste transportation ramp, as presented in Figure 19.4.

The surface facilities of the repository consist of the following buildings:

- Operations building
- Storage building
- Garage building
- Maintenance and service building
- Ventilation building
- Production building for high-pressure compacting of bentonite and preparation of backfill materials
- Office and workshop building
- Information center
- Rock stockpiles

The underground part of the repository is situated at a depth of 500 m below the ground surface. It consists of two areas: a central service area and a disposal area. The underground level can be reached in several ways: for personnel, through the service shaft; for waste and other cargo, through the spiral ramp (with at least a 15 m curve radius and 10% slope). The ramp is 5 km long, 10 m wide, and 7 m high. The service shaft has a 5 m diameter. It contains two elevators—a large one for personnel and small size loads, and a small one to serve as an emergency exit. The service shaft is also used as part of the ventilation system (air intake). The repository is supplied with a 3 m wide ventilation shaft which can also serve as an emergency exit. Access shafts lead towards a central service area with dimensions 190 m x 100 m, located directly below the operations area on the surface. Beside the shaft stations, there are four underground halls, 60 m long, 12 m wide, and 10 m high.

The central service area contains equipment for unloading of canisters and their transportation to disposal tunnels; it is equipped for acceptance, storage, and transportation of bentonite blocks and other items. There is also a maintenance vault intended for maintenance and

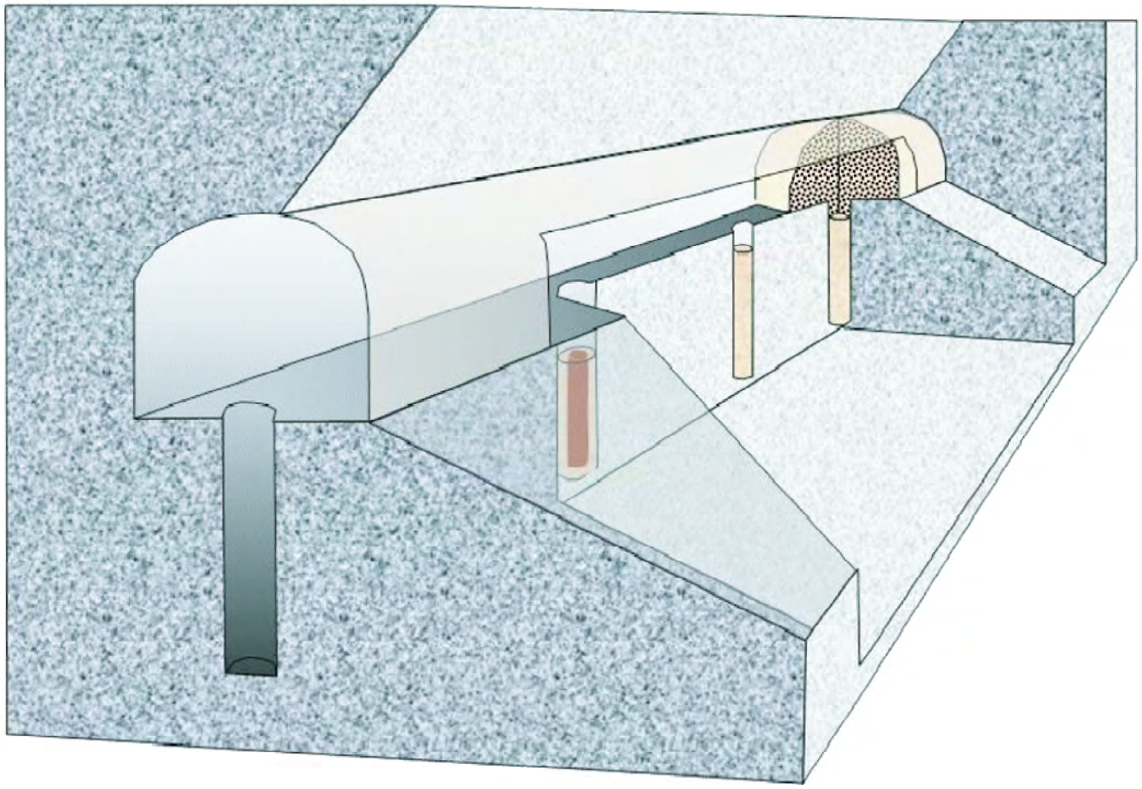

Figure 19.5. Canister disposal in tunnel boreholes

repairing of underground facilities and equipment, as well as a vehicle vault, storage vault, utility vault, and drainage vault.

The central service area is connected with the disposal area, which consists of two sections of parallel disposal tunnels. The tunnel spacing is 40 m and the tunnel length is 207 m. Canisters are deposited into 1.8 m diameter disposal boreholes at the bottom of each tunnel and surrounded by a 35 cm thick layer of compacted bentonite, as shown in Figure 19.5. Each borehole contains one canister; each disposal tunnel has 21 boreholes with a 9 m interval between them. The spacing was defined on the basis of thermal analysis made for Aberg (Ageskog and Jansson, 1999; Ikonen, 2003; RAWRA, 2000), which has a similar thermal conductivity and a boundary condition of 90°C on the canister surface. The number of disposal tunnels depends on the number of canisters. In Alternative 1, there are 34 disposal tunnels (total capacity of 714 canisters), and 20 disposal tunnels (total capacity of 420 canisters) in Alternative 2. In the first case, the disposal area occupies 455 m x 882 m, and in the second case, 290 m x 882 m.

The period of repository operation is 7 years for Alternative 1 (starting in 2030), and 5 years for Alternative 2 (starting in 2050). Disposal of SNF will start during the second year of repository operation. A total quantity of 710 and 420 canisters will be deposited within six and four years, respectively. The disposal schedule will be coordinated with the encapsulation schedule. As soon as the disposal tunnel is filled, it will be backfilled by a 15/85 compound of bentonite and sand. Backfilling of the entire disposal area will be completed within 1 year after the operation period has terminated.

19.6.2. Comparison of Alternatives

Comparing the two variants of repository development, there are strong technological and economical preferences for Alternative 2. In this alternative, the repository operation is planned to begin in 2050, almost 30 years

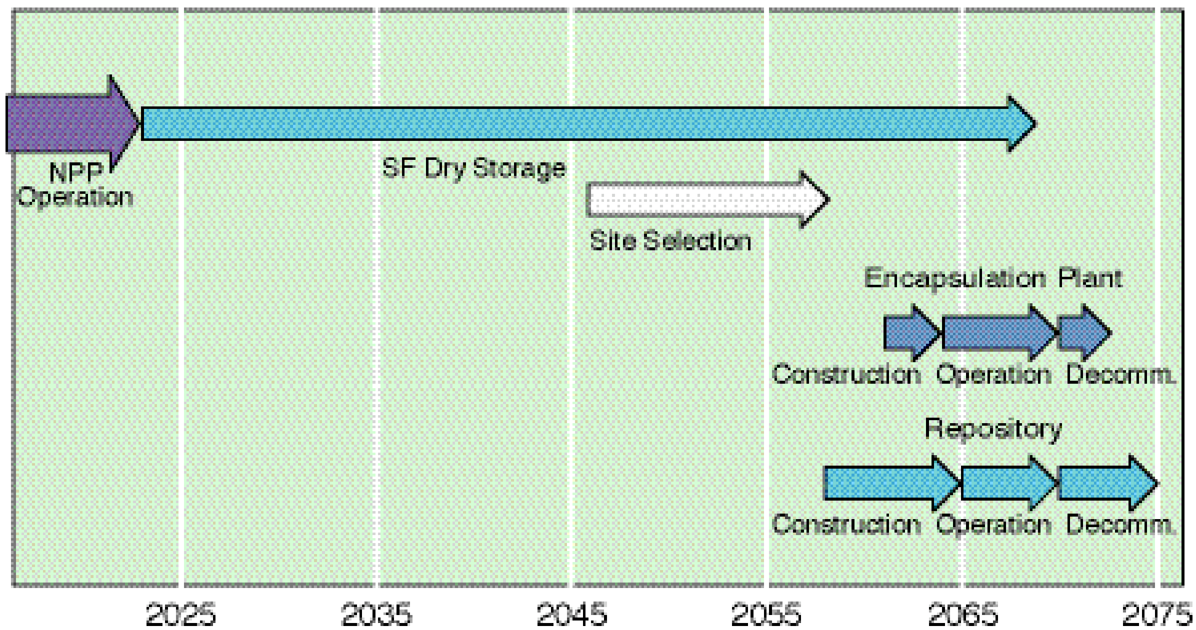

Figure 19.6. Approximate time schedule for the planned SNF disposal

after plant shutdown, giving sufficient time for the site selection process, which is not the case for Alternative 1. Heat release from SNF is sufficiently low so that the canisters can be optimally filled, and their required number is almost halved. The required SNF packaging is reduced, and consequently the operation of the encapsulation plant and repository is shortened.

A cost comparison between the two alternatives shows differences in fixed costs: Alternative 1 is about 20% more costly than Alternative 2. However, by taking into account the time value of money, the advantage of Alternative 2 becomes significant. For this reason, Alternative 1 has been dropped from further consideration, and Alternative 2 has been further analyzed for possible optimizations. These optimizations are studied in correlation with predisposal SNF management and planned decommissioning activities of the Krško NPP. From a financial perspective, the most favorable option shifts the repository operation 15 years ahead to take full advantage of SNF dry storage in casks. According to this scenario, after plant shut-down, SNF would be relocated to the repository only after a 45-year storage period. Repository operation would therefore start in 2065, as shown in Figure 19.6.

19.7. OTHER OPTIONS

Because of the high costs associated with development of an SNF repository, other options remain open. Disposal activities planned for the rather distant future give room for investigating other possibilities. "Multinational," shared repositories represent an option that may be interesting for countries with small quantities of SNF. However, in spite of the numerous advantages that a shared SNF and high-level waste repository might have, no initiative has been successful so far.

At present, the only alternative to national repository development is the export of SNF to Russia. This possibility has been open since 2001, when the Russian Federation (RF) adopted a package of three acts, upon which the import of irradiated fuel assemblies from other countries' nuclear reactors (to the Russian Federation for storage and/or reprocessing) has been made possible. An export of SNF without re-import of the reprocessed residue might be an applicable option in the future. There is, however, the question of price and economic justification that still remains open, and some other unresolved issues that need to be clarified before a decision is made.

All these options will be closely followed in the future. In the case of a positive development, the reference disposal scenario will be reviewed and adapted to the new situation.

19.8. FUTURE PLANS

The agreement between Slovenia and Croatia on the ownership and exploitation of the Krško NPP requires revision and updating of the joint decommissioning and the SNF and LILW disposal program every three to five years. The long-term SNF management and SNF disposal concept represent the vital part of this program, both from the financial as well as from the technological aspect. It is planned, therefore, that the reference scenario for SNF disposal will be periodically investigated and updated. Revisions will be focused on possible optimizations of the disposal system (e.g., the application of different types of canisters, a shared encapsulation plant). Disposal concepts in other geological environments will also be studied, in particular the disposal concept in a clay environment, a potential host environment in Slovenia.

The rather distant planned implementation of SNF disposal does not require any urgent activities related to the site selection for a repository. According to the rough time schedule, the site selection process is planned to be initiated only after 2030. However, for further development of the reference disposal scenario and for an investigation of new disposal concepts, more reliable data on a potential host rock for geological disposal will be required in the near future. For this purpose, systematic geological investigations (cabinet and field) are foreseen in the future.

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Chapter 20

Gearing for Geological Disposal of High-Level Radioactive Waste in South Africa: Current Status and Research Trends

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20.1. INTRODUCTION

South Africa is currently in the process of developing a national policy and strategy for radioactive waste management. A draft policy was formulated in 2003 and is currently under review. According to this draft policy, the South African Government shall establish a National Executive Committee on Radioactive Waste Management that will oversee the implementation of this policy and strategy. In view of these developments, a national strategy for spent nuclear fuel (SNF) management and other long-lived high-level radioactive waste (HLW) should become a reality in the near future.

At present, radioactive waste (SNF and "hot cell" waste) produced by the South African Nuclear Energy Corporation (Necsa) is stored at the Thabana Pipe Store, an interim storage facility at the Pelindaba site near Pretoria. The Vaalputs National Radioactive Waste Disposal Facility in the Northern Cape Province (see Figures 20.1 and 20.2) is licensed for the disposal of low- and intermediate-level radioactive waste (LILW) originating from the Koeberg nuclear power station (KNPS) near Cape Town. To date, only limited and low-key investigations have been carried out at Vaalputs with regard to its suitability as a deep geological disposal site.

20.2. SOURCES OF RADIOACTIVE WASTE

The main generators of SNF and other long-lived waste destined for a potential geological disposal facility in South Africa are the South African Nuclear Energy Corporation (Necsa), with its SAFARI-1 reactor for research and isotopes production, and the KNPS (Eskom).

Current projections shows that the SAFARI-1 Material Testing (MT) Reactor will produce about 5 m^3 of SNF and KNPS, and about 3,000 SNF assemblies (~1500 t U) during their lifetimes, respectively. About 10,000 m^3 of long-lived bulk waste and an unknown quantity of industrial and medical sources could potentially also be earmarked for geological disposal. These quantities may be subject to a radical change should the South African Government approve the implementation of the Pebble Bed Modular Reactor program. A strategic decision on this issue is currently being weighed against the rapidly growing energy needs of South Africa.

20.3. STORAGE AT PELINDABA

SNF from the SAFARI-1 reactor is currently stored in the authorized Thabana dry-storage facility at Necsa's Pelindaba site. A licensing application has been lodged with the National Nuclear Regulator, and an Environmental Impact Assessment is currently under way to extend the facility to double its size, to accommodate the future SNF storage needs of the reactor. To support future developments at Pelindaba, there is an ongoing, but limited research program into the regional

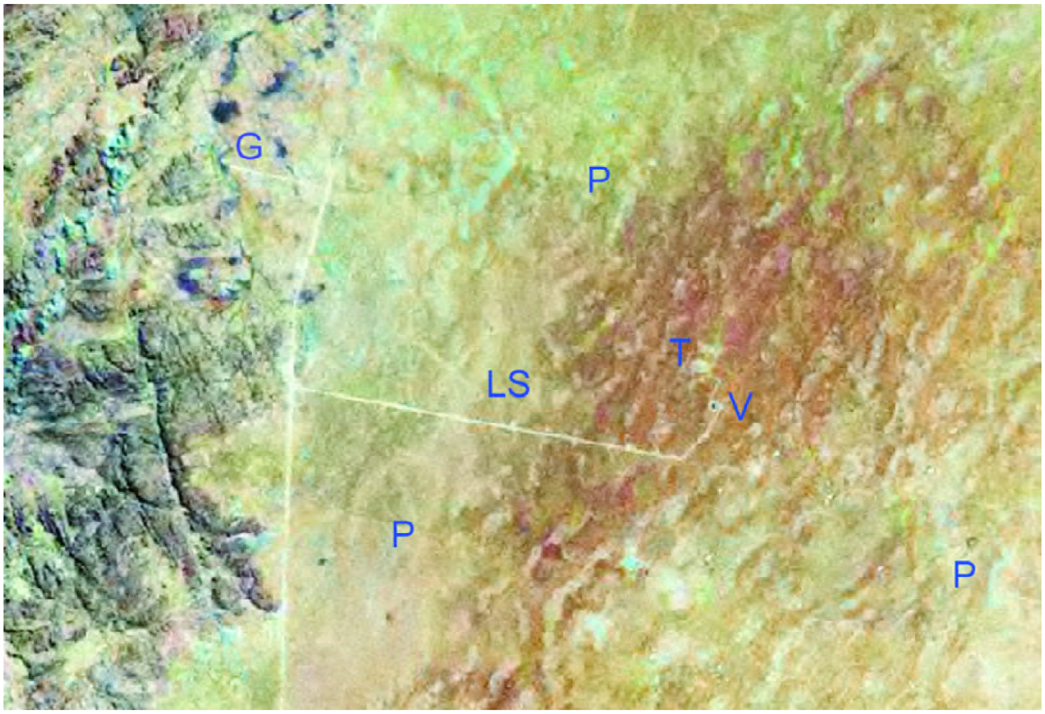

Figure 20.1. Landsat scene of the national radioactive waste disposal facility at Vaalputs. Clearly visible are the darker, better vegetated area of the Necsa property (P), the drums reception and offices building (V), the diposal trenches (T), the landing strip (LS) and, at Garing, the accomodation facilities for visitors (G). The N-S road runs along the watershed: to the west are outcrops of basement granite gneiss exposed by the escarpment, and to the east is the plateau with its cover of red sand and NNE-trending palaeo-dunes.

geology around the site, in collaboration with the University of the Witwatersrand and other institutions.

20.4. EARLY INVESTIGATIONS FOR GEOLOGICAL DISPOSAL AT VAALPUTS

A site selection program to locate a suitable site for the storage/disposal of radioactive waste in South Africa was initiated in 1978. The (then) Atomic Energy Board was tasked with this responsibility and had the following mandate:

- The construction of a low- and intermediate-level waste facility must be completed by 1986.
- The area to be chosen should have characteristics that are generally favorable should the site be considered for a deep geological facility.

The selection criteria (Corner and Scott, 1980) that guided the choice of Vaalputs in 1986 therefore had the additional purpose of not only LILW disposal, but also the potential for HLW disposal. The main studies centered on the following criteria:

- Rainfall, groundwater and surface hydrology
- Seismic hazard
- Mineral potential
- Agricultural production and growth potential
- Population density
- Ecologically sensitive areas
- Political and other issues

A working group on the disposal of HLW in South Africa, consisting of members of various government departments, Eskom, and the Atomic Energy

The final site selection was made according to geological criteria, which targeted a basin of Cenozoic continental sediments that were mostly derived from the underlying granite gneisses of the Namaqualand metamorphic Complex. The sediments are largely classified as a clayish feldspathic greywacke and represent a geological barrier for the shallow land repository at Vaalputs.

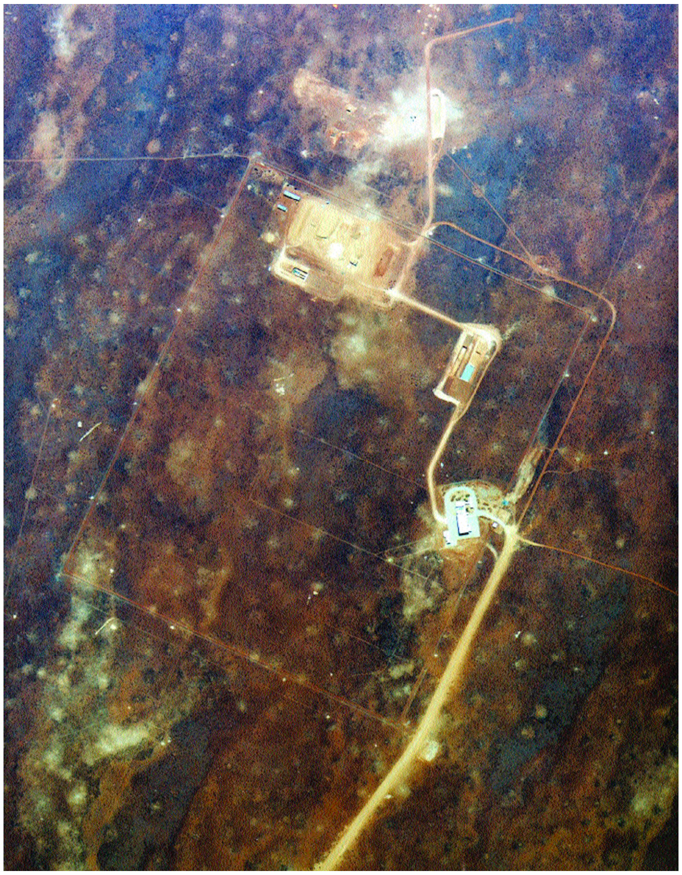

Figure 20.2. Aerial photograph of the Vaalputs Low to Intermediate level radioactive waste disposal site, showing the building hosting the waste-receiving facility at the end of the main road to the south, with two operational trenches and associated activities to the north.

Corporation (AEC, now Necsa), was established in 1987. This working group made the following recommendations (Toens, 1988) concerning SNF storage and geological disposal:

- Additional deep drilling at Vaalputs to investigate its suitability
- Detailed seismic monitoring at Vaalputs
- Development of criteria for site selection
- Investigation of SNF storage options at Vaalputs

NECSA decided in the 1990s to commence with "low-key" investigations on Vaalputs to establish its potential for geological disposal. The principle was that Vaalputs should be investigated first before other potential sites were considered. High-resolution aeromagnetic and ground-magnetic surveys formed the backbone of a structural geological program to identify suitable areas for deep diamond-drill boreholes. Four boreholes (between 550 m and 1,000 m deep) were drilled in selected sites to identify suitable host rock in the granite-gneiss formations. The first two boreholes intersected various fracture zones, but the last borehole (drilled in a different part of Vaalputs and completed in 1996) intersected megacrystic granite gneiss with excellent geotechnical qualities between depths of 200 m and 1,000 m. This granulite facies granite gneiss, dated at ~1060 Ma, has virtually no weathered zones or faults and very few joints.

Because of the lack of cohesion between the major role players and the lack of a national strategy for radioactive waste management at the time, the AEC halted any further investigations at the end of 1996.

20.5. CURRENT RESEARCH AT VAALPUTS

20.5.1. General

The halting of the investigations specifically targeted at the selection of candidate areas for a HLW repository did not mean an end to scientific research at Vaalputs. The rapid expansion of the geosciences in the past decade meant that the original studies produced by the AEC in the early 1980s were becoming increasingly outdated. In view of this, researchers at Necsa have embarked on a new, broad program of investigations into the geology, tectonics, and environmental aspects of the Vaalputs site in its regional context, highlights of which are presented below. Given the limited funds and manpower available to Necsa, it was no longer possible to conduct the said investigations in-house, as previously done. Instead, Necsa has pursued the objective through a policy of offering Vaalputs, with its unique infrastructure and knowledge base, as an *open field laboratory* available to all local and foreign scientists who may be interested in pursuing the research opportunities that Namaqualand, with its unique geology, could offer. Thanks to this policy, Vaalputs is now firmly embedded in the technical literature through an increasing number of specialist articles. Necsa and Vaalputs are also participating in the *Heart of Africa* initiative, a recently established collaborative program between the South African National Research Foundation (NRF), various local universities and research institutions, and a number of prestigious German research and development organizations.

20.5.2. Precambrian Geology

The Namaqualand metamorphic complex that underlies the Vaalputs site is largely made up of granulite facies, granitic orthogneisses, charnockites, and supracrustal granulites intruded by anorthosite, norite, diorite, and associated tonalite. The study of the basement rocks exposed at surface on the western sector of the Vaalputs site or intersected by the deep boreholes mentioned above (see Figure 20.1) revealed unusual mineralogical (Mouri et al., 2003) and geochemical characteristics. The latter include evidence for regional-scale, high temperature (~800–1,000°C) radioactive elements influx/mobility, whereby granite gneisses are converted to Th and U-enriched charnockites and charnockitic granulites (Andreoli et al., in press).

20.5.3. Cainozoic Geology

The Cainozoic sediments that blanket the Namaquan basement at Vaalputs were deposited in a tectonic basin established on the shoulder of a rifted continental margin. Given the fact that sedimentary basins of this nature are rare and only poorly known, the unusual sediments (including a feldspathic greywacke) found at Vaalputs are being studied (Brandt et al., 2003; 2005) because they represent the ultimate geological barrier between any buried radioactive waste and the environment. Aspects that are receiving attention include the tectonic setting of the basin, the palaeo-climate record, and the history of the denudation associated with continental margin uplift and the eastward movement of the Namaqualand escarpment (cf. Figure 20.1). This latter aspect is currently being monitored by measuring fission track cooling ages of apatite and the rate of accu-

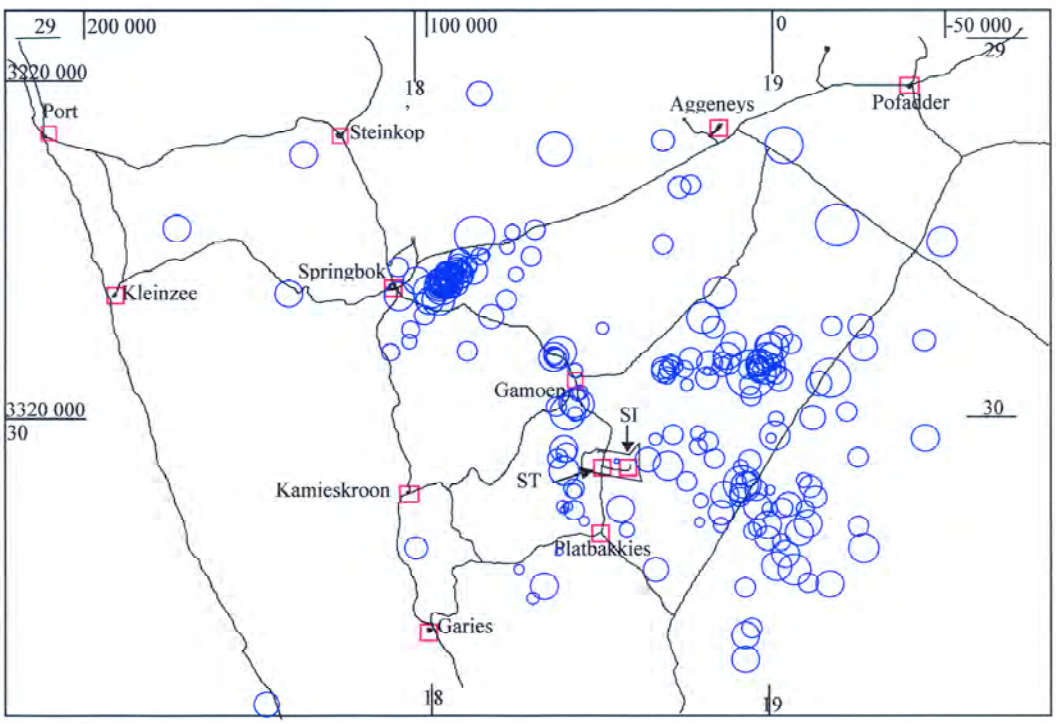

Figure 20.3. Distribution of seismic epicentres (Magnitude range: 1.6–5.1) in Namaqualand as recorded in the period 1989–2003 by the two seismic stations at Vaalputs (localities marked SI and ST). Depth restrictions to 42 km have been applied to events futher than 95 km away from Vaalputs, generally resulting in small offsets from depth unconstrained (Z>42 km) epicentres (cf. Scheepers and Andreoli, 2004).

mulation for cosmogenic Ne-isotopes in surface granites.

20.5.4. Neotectonics

In common with many other intraplate regions around the world, Southern Africa experiences sporadic, low grade (magnitude max.: 6.3) seismic activity as a result of the stress interplay originating at plate boundaries. At a more local scale, the distribution of the epicenters is heterogeneous, with Namaqualand occasionally experiencing a certain slightly above-average frequency of events that are recorded by two 3-component seismic recording instruments (Figure 20.3).

The Gutenberg-Richter equation applied to events with M_{min} 2.3 (the minimum magnitude that can be consistently recorded within the ~150 km radius) yields a recurrence rate of 15 events per year, an 85% probability of occurrence for a M ~3 seismic event within 6 months, and a M_{max} of ~5.8. Although these parameters are well within safety specifications (Scheepers and Andreoli, 2004), the causes of this seismic activity are under scrutiny through an array of techniques that include GPS-based geodesy, in conjunction with a regional structural analysis of Cenozoic age faults and *in situ* stress measurements (Viola et al., 2005).

20.5.5. Natural Analogues

Scattered within the Namaqualand metamorphic complex, there are occasional occurrences of Th, U-bearing monazite, a REE-rich phosphate, the most notable deposit having been mined some 100 km south of Vaalputs at Steenkampskraal (Andreoli et al., 1994). In light of its overall environmental and geological similarities with Vaalputs, this deposit was investigated by Reid et al (2002) to test the long-term degradation behavior of monazite. This was done because artificial phosphate-based matrices similar in composition to monazite may prove useful as waste forms for high-level radioactive waste. As a follow-up to this work, which proved that monazite experienced weathering under past climatic conditions, research is in progress to interpret the behavior of U, Th and their decay products (mainly ^{226}Ra and ^{234}U) in the groundwater system of Vaalputs under current climatic conditions. The role of colloids in actinide migration was instead investigated in samples of groundwater from Steenkampskraal by Ross et al. (1996). Their results proved that less than 1% of uranium, thorium and less than 3% of rare earth elements were associated with colloids in groundwater at Steenkampskraal.

20.5.6. Performance Assessments and K_d Values

The method of disposal at Vaalputs can be described as a multiple-barrier concept, because the metal drums for LLW and concrete drums for ILW that are used for containing waste can act as primary barriers. These containers are placed in a trench, which is then filled with backfill material (Figure 20.4).

Literature-derived distribution-coefficient values may not be relevant to Vaalputs backfill, since high chlorine and sulphate concentrations present in the buffer material may influence the values, thus encouraging or retarding the migration of radionuclides.

The aim of this project is therefore to determine the K_d (thermodynamic distribution coefficient) values of the different complexes of uranium and other short-lived radionuclides onto the Vaalputs (and Thabana) soils. It also aims at quantifying the influence of humic acid, sulphate-reducing bacteria, and degradation products of cellulose on complex formation. These results will then be compared with available literature data, to establish new *in situ* K_d values that could be used in the modeling for the safety assessments of Vaalputs as a nuclear waste disposal site.

Further research objectives involve the testing of engineered barriers to prevent corrosion, as well as modification of backfill material to minimize corrosion properties that may be applicable to metal as well as concrete drums. With respect to the latter, current research aims at extending the integrity of cement as a primary barrier. Neutron radiography techniques were found to be particularly valuable in evaluating how different cement compositions react to the anomalous concentrations of chloride and sulphate anions that distinguish the chemically reactive, semi-arid Namaqualand plateau.

20.6. MODELING

The complex geosphere and low infiltration rates make it difficult to apply generic models when simulating the performance of waste packages and radionuclide migration on the site. Consequently, the development of site-specific numerical models is becoming increasingly important for the comprehensive simulation of degrada-

Figure 20.4. Waste disposal practices at Vaalputs: the intermediate (top) and low-level waste (bottom) trenches at Vaalputs before their filling and planned closure in 2005

tion processes that impair the performance of engineered barriers. Site-specific models developed during the selection phase of the Vaalputs disposal facility (in the early 1980s) are now either technologically obsolete or no longer applicable, in light of subsequent studies such as those showing that the disposal medium is chemically reactive, owing to its high chloride and sulphate content.

As a result, current research focuses on the performance of containers such as those used for the disposal of LILW in the unsaturated zone. In particular, our situation analysis predicts that the conditions in the saturated zone should be more corrosive than those in the unsaturated zone. Thus, performance characterization of waste packages in the saturated zone will constitute the next phase of the research and will be more directly relevant to the disposal of HLW.

The main objective of conducting site-specific experiments on waste package performance is to formulate and then calibrate mathematical models that may predict the conditions of packages into the future. This modeling, which should account for all the drivers of corrosion, represents a major and critical input to safety assessment and to compiling waste acceptance criteria for waste generators.

Another objective for site-specific models is to account for the extreme heterogeneity of the disposal system, since the complexity of the latter tends to increase with time following the discovery of new geological features. A typical example is the recent discovery of neotectonic, potentially active structures such as faults and joints across western southern Africa in general and Vaalputs in particular (Viola et al., 2005). These features make it incorrect to assume homogeneity when modeling, so that site-specific models tailored for site conditions are becoming more relevant.

The more appropriate approach for handling the migration of radionuclides in the Vaalputs geosphere is to apply models capable of simulating dual porous media. In addition, HLW disposal is expected to benefit from a new generation of hybrid, physical-chemical (i.e. hydrological-thermal-chemical-mechanical) models in the study of contaminant transport in the near field and geosphere. The low infiltration rate and fairly slow mobility of groundwater will also expand the focus of simple groundwater transportation models to include colloids as possible vehicles for radionuclides. Despite evidence from Steenkampskraal that colloids do not seem to be effective in the transport of radioactive nuclides (Ross et al., 1996), global research in this field has identified other sources of colloids, such as degraded waste packages, which may be capable of mobilizing radionuclides, but which were not considered in our initial investigations. Finally, future research will also target stochastic modeling and computational fluid dynamics, to account for the uncertainties caused by the long half-lives of the radionuclides present in HLW.

20.7. POSSIBLE FUTURE SCENARIOS

The Draft National Policy recognizes that deep geological disposal is currently the most internationally acceptable solution for dealing with SNF and HLW, and as such will require very careful consideration. More specifically, if a formal site selection program for geological disposal should become a reality, it is NECSA's view that a transparent, countrywide study would be required.

Such a study would ideally identify a number of potential sites, from which one would be selected for detailed site confirmation. Vaalputs would naturally be considered as one of these candidate sites because of its highly favorable characteristics. If South Africa should embark on a selection exercise for geological disposal, there would be certain prerequisites:

- All role players and stakeholders must be involved.
- International cooperation and experience will be essential.
- International guidelines and criteria will be followed.
- The prevailing economic and political climate will be a major factor, and may permit the consideration of a regional repository involving various other countries.

20.8. ACKNOWLEDGMENTS

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Chapter 21

Geological Disposal of High-Level Radioactive Waste in Spain

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21.1. INTRODUCTION

The Empresa Nacional de Residuos Radiactivos, S.A. (ENRESA), a public company founded in 1984, manages all radioactive waste produced in Spain. All the activities of ENRESA are approved by the Spanish Government through the General Radioactive Waste Plan (GRWP). The fifth GRWP was approved by the Government in July 1999 and remains in force. This GRWP established that no decisions will be taken concerning the final management of high-level radioactive waste until the year 2010 (Ministry of Industry and Energy, 1999). Prior to that date, ENRESA will have to develop a greater understanding of all the techniques and methodologies required for deep geological disposal, with consideration given in parallel to R&D activities relating to separation and transmutation techniques that might reduce waste inventories.

The change in government that occurred in 2004 has led to the modification of the management strategy, which will be reflected in a new GRWP, the sixth, presumably to be approved by the government in 2005. This GRWP will establish as the top and most urgent priority the availability of a Centralized Temporary Storage (CTS) facility, to be operable by 2010. Consequently, the definitive disposal of spent nuclear fuel (SNF) will occupy a secondary plane since, with a CTS facility in operation, there will be no haste in this respect. Among other things, this new plan will imply the slowing down of all activities relating to deep geological disposal.

However, during the period 2001–2005, the period to which this revision corresponds, ENRESA had established a strategy aimed at adapting the performance of activities associated with deep geological disposal to a much longer time frame than initially foreseen. The information presented here corresponds exclusively to this period.

21.2. FUTURE ACTIVITIES AND STRATEGY

The activities performed in the area of deep geological disposal during the period 2000–2005 have been conditioned by the following:

- The remoteness in time of decision making regarding the definitive management of high-level wastes (2010)
- The need to provide adequate scientific and technological support for this decision making

Given these premises, the activities were oriented towards the following:

- The improvement of understanding and generic technologies for disposal in granites and clays through the R&D program, to be carried out fundamentally by way of projects within the European Framework Programs and others deriving from international agreements and focused on URL's, completed with specific activities performed at the national level
- The revision of existing generic designs for disposal in granites and clays
- Completion of the performance of two new generic

safety assessment exercises, ENRESA 2000 and ENRESA 2003, for granite and clay, respectively, incorporating the know-how and capacities developed to date
- Integration and summary of all the information related to deep geological disposal (DGD) obtained to date in two generic documents: Basic Granite DGD and Basic Clay DGD, to serve as key documents supporting future decisions and to be updated in the future as knowledge of the technologies and methodologies relating to deep geological disposal progresses

The initiation of many of these activities was dealt with in the previous revision, as a result of which this document will provide a brief summary of the most relevant conclusions.

Regarding future activities associated with deep geological disposal, it should be emphasized that these activities will be conditioned by the reorientation of ENRESA's activities towards the CTS facility, as the fundamental priority. This reorientation will imply a lower profile for their performance, with a view to adapting investments in this area over a longer time frame than initially foreseen.

Nevertheless, the future courses of action in relation to deep geological disposal will center on the following:

- Maintaining and updating of the capacities and technologies developed for characterizing the behavior of high-level radioactive wastes and the isotopes they contain, in relation to the separation and transmutation of high-level wastes as an issue supporting management
- Maintaining and updating the capabilities developed for the DGD facility through highly selective participation in international R&D programs, mainly those of the European Union

21.3. SITE SELECTION PLAN

Site selection activities were completed in 1996, and no additional study has been considered or performed. The only actions taken in this field have been limited to conservation of the information geo-referenced in a geographical information system (GIS), and to the connection of this information with the corresponding technical information in electronic format, such that in the future it may be consulted quickly and accurately.

21.4. REPOSITORY DESIGN

During this period and on the basis of progress in R&D, the generic designs for granite and clay have been revised, although the final results are still in the analysis phase.

21.5. REPOSITORY TECHNOLOGY COMPONENTS

The activities performed correspond to developments within the framework of the ENRESA R&D Plan for 1999-2003, in close connection with the 5th Framework Program of the European Union. The new ENRESA R&D Plan, covering the period 2004–2008, began in 2004, and the present report will not include any results for this plan or for participation in the 6th Framework Program.

The main international projects relating to the lines of R&D associated with the components of the repository, and that have constituted ENRESA's main activities in these fields, are indicated in Table 21.1 and described in the corresponding sections.

21.5.1. Physico-Chemistry of Actinides and Spent Nuclear Fuel Technology

ENRESA has set up a specific group, with research workers from ENVIROS Spain, CIEMAT, and UPC-DIT, to address activities relating to the physico-chemistry of actinides and the behavior of fuel. The main activities have been as follows:

- International collaboration in the development of thermodynamics databases: Participation in the Organisation for Economic Cooperation and Development-Nuclear Energy Agency (OECD-NEA) project. Termodinal Data Bases III and NEA-SORB.
- Understanding and development of models for radionuclide retention and release mechanisms: Study of retention processes for ^{138}Cs, ^{90}Sr, ^{60}Co, ^{152}Eu, ^{75}Se, ^{233}U, ^{99}Tc, and ^{36}Cl in bentonites, granites and clays, and specific experiments complemented with the results from the area of natural analogues (OKIO, Palmottu, and Mina Fe projects).
- Analysis of interaction between Th (IV) and Pu (IV) and iron oxy-hydroxides (magnetite and goethite) as cask corrosion products, and bentonite, including the development of surface complexing

Table 21.1. Participation of ENRESA and Spanish organizations in 5th Framework Program projects associated with DGD

Line of Action	Projects		Spanish Organizations Participating, in Addition to ENRESA
Fuel Technologies and Basic Technologies	SFS	Spent Fuel Stability under Repository conditions.	Quantisci, UPC-DIQ, CIEMAT
	ACTAF	Aquatic Chemistry and Thermodynamics of Actinides and Fission Products Relevant to Nuclear Waste Disposal.	CIEMAT, UPC-DIQ
	BORIS	Building confidence in deep disposal: the borehole injections sites at tomsk-7 and Krasnoyarsk-26	UPC-DIQ
Casks, Clay Engineered Barriers and Geological Barriers and Radionuclide Migration	Contamination Corrosion		INASMET
	FEBEX-II	Full-scale Engineered Barriers EXperiment in crystalline host rock	AITEMIN, CIEMAT, UPC-DIT, UPC-DIT, CSIC, UDC, UPM
	HE	Heater Experiment: rock and bentonite thermo-hydro-mechanical (THM) processes in the near field	AITEMIN, UPC-DIT
	EB	Engineered Barrier emplacement experiment in Opalinus Clay for the disposal of radioactive waste in underground repositories	AITEMIN, UPC-DIT
	VE	Ventilation Experiment in Opalinus Clay	AITEMIN, DM-Iberia UPC-DIT, UPV
	CROP	Cluster Repository Project	ENRESA
	ECOCLAY	Effects of cement on clay barrier performance	UAM, CSIC
	GASNET	A thematic network on gas issues in safety assessment of deep repositories for radioactive waste	ENRESA
	PROTO TYPE REPOSIT ORY	Prototype Repository	UPC-DIT, AITEMIN
	BENCH PAR	Benchmark tests and guidance on coupled processes for performance assessment of nuclear waste repositories	UPV
	RETROCK	Treatment of geosphere retention phenomena in safety assessments	CIEMAT, UPC-DIQ
	NANET	Network to review natural analogue studios	ENRESA, UPC-DIQ
	MODEX -REP	Elaboration of hydromechanical coupled models by interpretation of the disturbances observed during the sinking of the main shaft of an underground laboratory in Eastern France	UPC-DIT

Table 21.1. Participation of ENRESA and Spanish organizations in 5th Framework Program projects associated with DGD (continued)

Line of Action	Projects		Spanish Organizations Participating, in Addition to ENRESA
Biosphere	BIOCLIM	Modeling sequential biosphere systems under climate change for radioactive waste disposal	CIEMAT, UPM
	BIOMOSA	Biosphere Models for Safety Assessment of radioactive waste disposal based on the application of the Reference Biosphere Methodology	CIEMAT
	PADAMOT	Palaeohydrogeological data analysis and model testing	CIEMAT, UPM
Safety Assessment	BENIPA	Bentonite barriers in integrated performance assessment	ENRESA
	SPIN	Testing of safety and performance indicators	ENRESA
	COMPAS	Comparison of alternative waste management strategies for long-lived radioactive wastes	ENRESA
	TN-MONITORING	Nuclear Science and Technology, Thematic Network on the Role of Monitoring in a Phased Approach to Geological Disposal of Radioactive Waste	ENRESA
Abbreviations:	UPC-DIQ: Polytechnic University of Catalonia. Dept. of Chemical Engineering CIEMAT: Centre for Energy-Related, Environmental and Technological Research UPC-DIT: Polytechnic University of Catalonia. Dept. of Land Engineering CSIC: Scientific Research Council UPM: Polytechnic University of Madrid UPV: Polytechnic University of Valencia UAM: Autonomous University of Madrid UDC: University of La Coruña		

models. The results have underlined the different evolution of sorption mechanisms in granite depending on the presence or absence of bentonite.

- Characterization and behavior of irradiated fuel, with activities focusing on the effect of α, ß radiation and the presence of iron oxy-hydroxides on release mechanisms and models, on the mobility of the secondary phases formed, and on improvement of the models with consideration given to the generation of radiolytic products followed by oxidization-dissolution and release-precipitation (Figure 21.1).

These activities have been carried out in collaboration with the Karlsruhe Joint Research Centre (ITU).

21.5.2. Metallic Containers

These activities constitute the systematic course of action that ENRESA has been performing, mainly in collaboration with INASMET and to a lesser extent with CIEMAT and CENIM (National Centre for Metallurgical Research—Scientific Research Council), since the early 1990s.

During the period 2000–2004, activities in this field have centered on the study of corrosion of metals and alloys under the conditions of salinity existing in granite and (especially) clays. The studies for granite concluded with satisfactory results for carbon steel, stainless steel and alloys of titanium Gr7. Electron-beam welding provides the best results for resistance to corrosion. Technologies and methodologies exist for the study and analysis of the behavior of any material. At present, passivation studies are being undertaken, and the optimization of manufacturing processes is being analyzed. These activities are the basis for the performance of manufacturing tests on various prototypes for

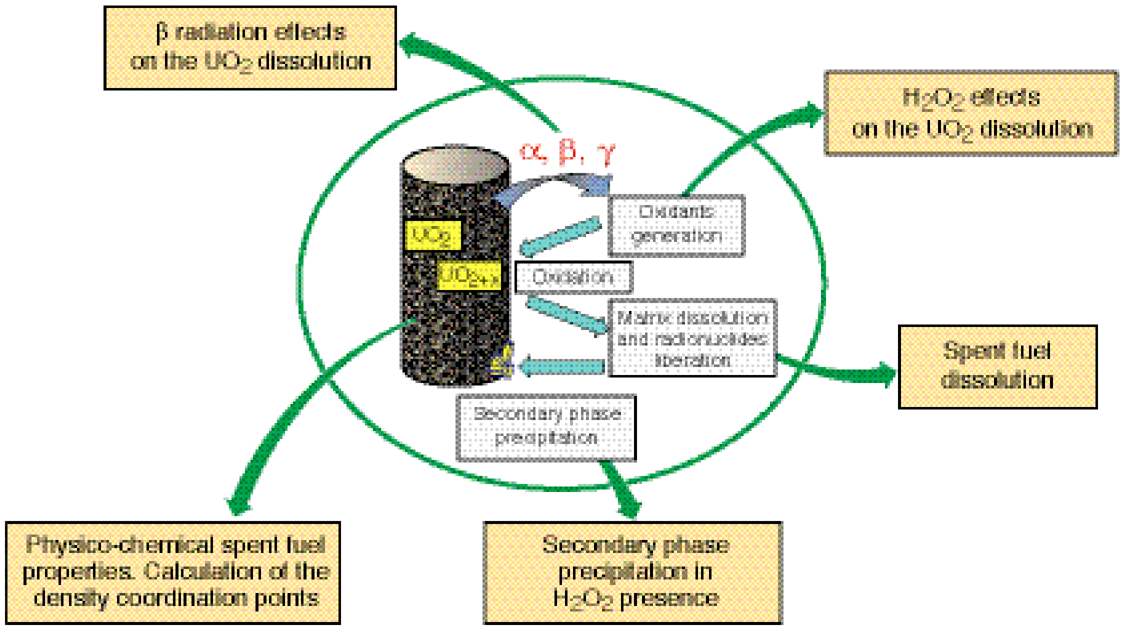


Figure 21.1. Behavior of spent nuclear fuel: oxidation model

the future CTS facility.

In parallel to the corrosion studies performed in the laboratory, the metallic materials natural analogues project ARCHEO has been undertaken (as a study of corrosion processes in archaeological metallic materials), the results obtained corroborating the corrosion rates obtained in the laboratory.

21.5.3. Engineered Clay Barriers

The activities in this area have been addressed with the Full-scale Engineered Barriers EXperiment (FEBEX) project as the central element (Figure 21.2). The experience obtained has allowed the experiments to be designed and implemented at the Mont Terri underground laboratory (Switzerland), co-financed by the European Union (EU), and has also led to a high level of participation in the prototype repository, Backfilling and Plug Test and TBT projects in the underground laboratory at Äspö (Sweden), as described in the previous revision.

Thanks to the performance of these projects, a powerful multidisciplinary team of researchers (including UPC, AITEMIN, CIEMAT, UDC, UPM-CSIC, and UPV) has been set up in Spain, ensuring the availability of capacities in the fields of design, characterization, monitoring, and modeling of the THMC (thermo-hydro-mechanical and chemical) behavior of compacted clay barriers for both granite and clay media.

In the year 2002, the initial phase of dismantling began within the FEBEX-II Project, with the extraction of the first experimental heater, accompanied by a systematic and exhaustive sampling and *in situ* analysis of the surrounding bentonite (Figures 21.3 and 21.4). Subsequently, a concrete plug was installed, and new instrumentation was introduced via boreholes for the monitoring of THM properties, along with a specific system allowing samples of bentonite saturation water to be obtained. As a result, the experiment has been rejuvenated and has become a part of the NF-PRO project, with connotations different from those considered in FEBEX. The experiment and its corresponding mock-up continue to generate data.

The conclusions of FEBEX II, completed in December 2004, are as follows:

In Situ Test

The activities performed with regard to the *in situ* test

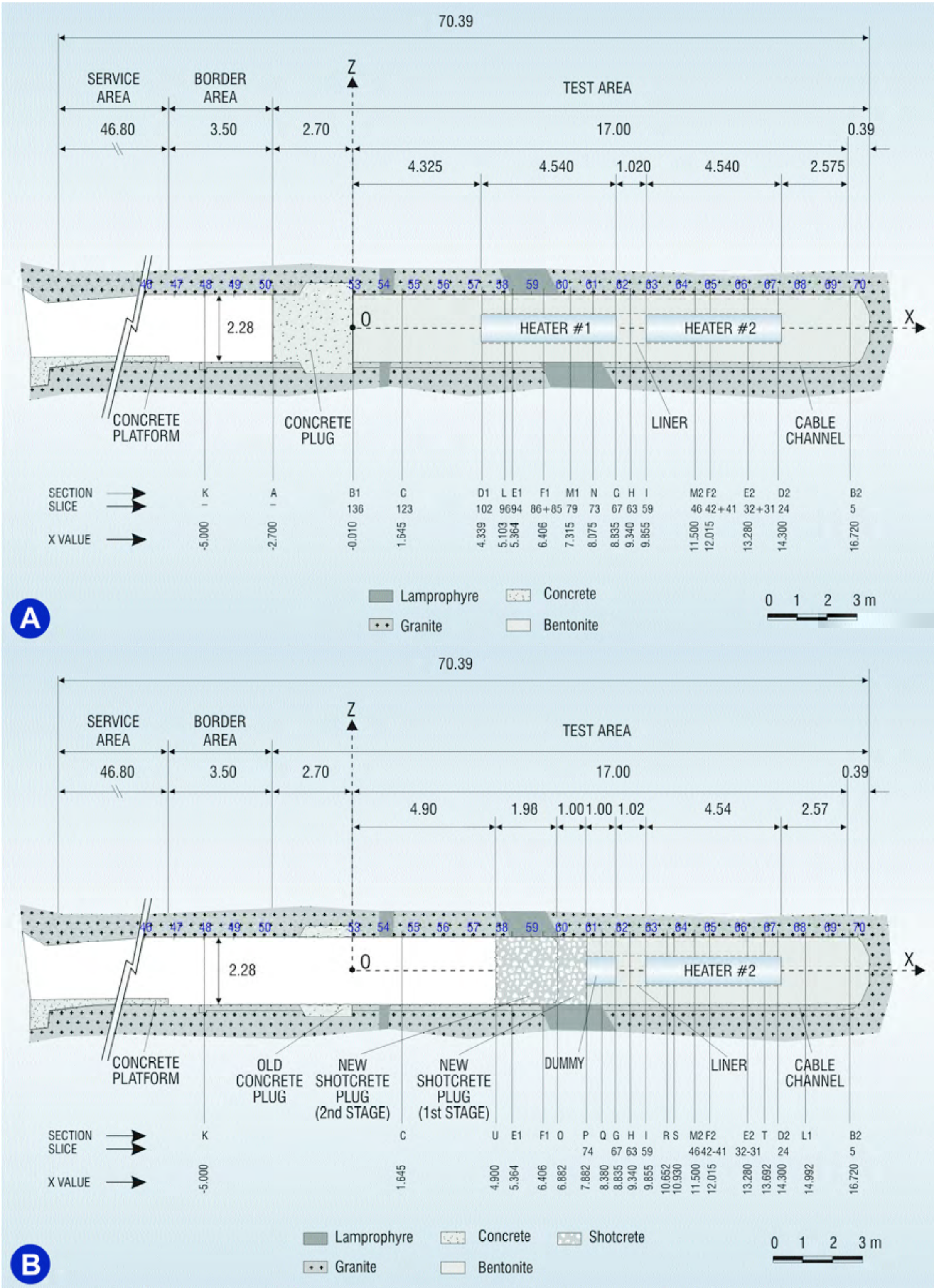


Figure 21.2. Layout for FEBEX (a) and FEBEX II (b)

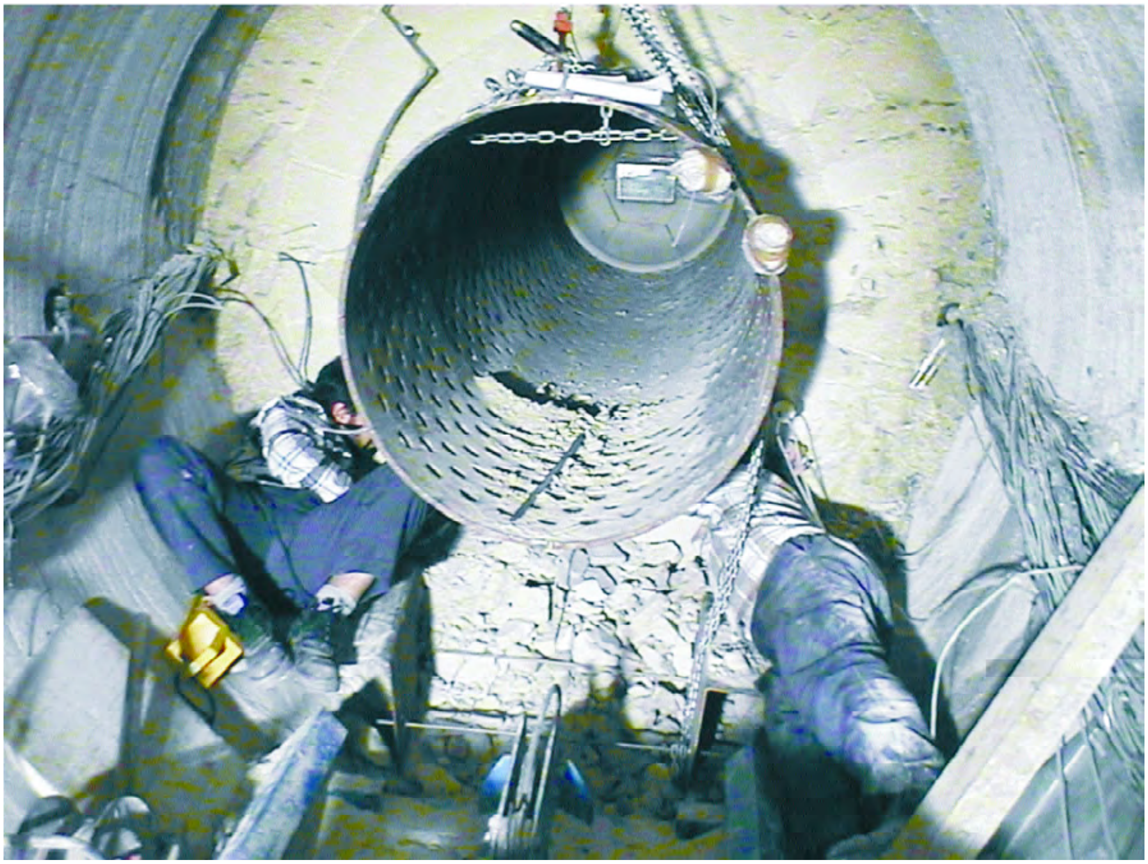


Figure 21.3. FEBEX dismantling

have met the objectives defined in the work plan:

- The FEBEX demonstration objective has been completed, partly through direct observation and analysis of the state of the bentonite barrier—which closely agree with the forecasts regarding the closure of all the voids and the homogenization of the barrier with hydration—and partly by checking the status of the measuring instruments.
- Apart from contributing to verification of the forecast behavior of the barrier, the data provided by the post-mortem laboratory tests on the bentonite have served to check the reliability of the monitoring data and, consequently, increase confidence in comparison to the results of modeling. They have also served to check the predictions of the geochemical model.
- The nonexistence of an excavation-disturbed zone (EDZ) in the FEBEX gallery is of value only in relation to the Grimsel granite massif and, perhaps, that of the FEBEX gallery. However, the study methods used may be generally applied to any massif. The same may be said about the lack of variation in the hydraulic properties of the rock massif near field.
- The shotcrete technology tested for the closure plug of the nondismantled part of the test may be of use in a real repository.

Mock-Up Test

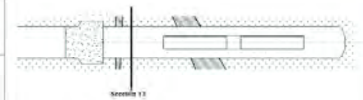
The only objective established for the "mock-up" was the supervision, control, and management of the monitoring data, to continue checking the model predictions. This has been accomplished according to plan.

Laboratory Tests

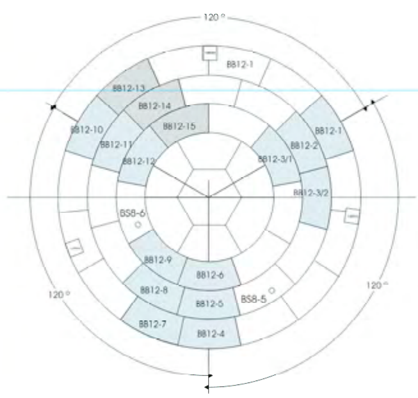
The main objectives proposed for the THM and THG investigations during FEBEX II have been fulfilled.

SAMPLING SECTION 12	LOCATION		
	"Actual" x coordinate: 1.981 m	Instrumented section: NA	Bentonite slice: 120

SAMPLES			
Code	No. of samples	Type	Destination org.
BB12-1 to BB12-6, BB12-10 to BB12-12	9	Bentonite block	CIEMAT
BB12-7 to BB12-9	3	Bentonite block	CTU
BB12-13 to BB12-15	3	Bentonite block	SPARE



DISMANTLING FIGURE



DISMANTLING PHOTO

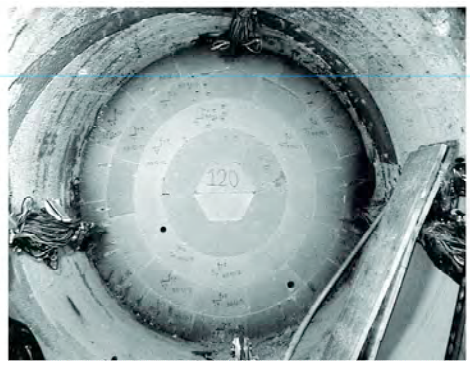


Figure 21.4. Bentonite sampling: example of the identification and location of the samples

With respect to the THM studies, information on those issues affecting the retention capacity of the bentonite has greatly increased, as has knowledge of the effects of temperature and salinity on several hydro-mechanical properties. The water infiltration mechanisms are also better understood, which has allowed for improvement of the models. Several techniques for characterization of the bentonite microstructure have been tested. As a result, a deeper insight into the behavior of compacted bentonite, and the basic mechanisms controlling it, has been achieved by taking into account the interplay between the microstructural and macrostructural levels. However, further complementary studies may be necessary to provide more information on the mechanisms influencing pore-size distribution changes in highly active clays. More research effort must be dedicated to the study of certain specific aspects, such as the existence of a threshold hydraulic gradient, water flow under low hydraulic gradients, and the effect of osmotic and temperature gradients on this flow. More generally, current knowledge of the physio-chemical aspects of clay microstructure should be related to the macroscopic behavior of compacted bentonite (Figure 21.3).

With regard to THG activities, considerable progress has been made in the acquisition and modeling of bentonite pore-water chemistry. Different types of water have been distinguished and characterized. The effect of pore-water composition on anionic exclusion is an interesting issue to be studied in the future. The influence of the exchangeable cations of bentonite on water adsorption capacity has been analyzed. Furthermore, the factors affecting the dissolution of bentonite have been identified. The sorption experiments have allowed us to determine the quasi-thermodynamic parameters for Cs and U. Exponential relations have been found between the diffusion coefficients of HTO and chloride and the dry density of the FEBEX bentonite, allowing these parameters to be easily determined at any dry density. Further studies should be devoted to analysis of the influence of water salinity on the anion exclusion processes that take place in bentonite and, consequently, on the diffusion coefficients.

The results obtained during FEBEX I on gas generation in bentonite have been confirmed.

Modeling

FEBEX II resulted from the two following ideas: (1) the good performance of the heating system and of the instruments in both large-scale tests, which authorized continuation of the test, and (2) the good expectations regarding the prediction capacity of the THM and THG models. At the same time, the need for certain theoretical improvements and, especially, improvement in determining certain parameters and constitutive laws, also pointed to the need for FEBEX II.

Noteworthy improvements have been achieved with respect to acquisition of parameters and constitutive laws, as pointed out in evaluating the laboratory tests. Progress has also been made in developing the theoretical models.

The THM modeling reflects the processes fairly well, both overall and quantitatively, especially in the short term. The THG modeling captures the concentration tendencies of almost all the chemical species. In addition, in the attempts to resolve the causes of the differences between the "mock-up" thermo-hydro-dynamic data and the model predictions, more detailed insight has been gained into the behavior of the clay barrier—it has even been possible to improve the adjustment of certain aspects. However, there are still gaps in knowledge of the processes, as is demonstrated by the clearly different tendencies shown by relative humidity between the monitoring data and the results of modeling, especially in the more hydrated zone of the barrier. Consequently, there is a need for research to continue on this divergence, which might have consequences for other hydration-dependent variables, unless this was demonstrated to be irrelevant from the point of view of repository safety assessment.

The project has also generated a particularly complete THM database, into which are incorporated the results currently being generated by both the *in situ* and mock-up experiments, which are further incorporated into the projects of the 6th Framework Program NF-PRO. This database includes the results of more than 10 years of experimentation and is one of the most complete in existence for the study of the THMC behavior of bentonites.

Regarding the corrosion studies, special mention might be made of the results obtained from analysis of the corrosion of metallic materials introduced in the FEBEX project bentonite, following their extraction during the partial dismantling of this experiment. It may be observed that the corrosion of materials such as stainless steel, carbon steel, and titanium alloys is lower than the results obtained in the laboratory, owing fundamentally to hydration of the bentonite not exceeding 17%, where the samples of these metals were placed. However, maximum corrosion has been observed in sensors located in areas of the bentonite where high levels of hydration have been reached. Analysis of the corrosion products indicates that the corrosion is caused by sulphate-reducing bacteria in the bentonite. This is another issue to be taken into account.

The objective of the engineered barrier experiment, the most important of those performed in clay media, was to analyze the feasibility of jointly using bentonite blocks at the base of the container and granular bentonite (highly compacted pellets) to backfill the rest of the gallery. The experiment was instrumented with artificial hydration systems and sensors to monitor the hydromechanical evolution of the barrier, the geological medium, and the behavior of the EDZ, and to verify the existing models (Figure 21.5). The 1st phase of the project has now been completed, and monitoring will continue up to dismantling in 2007.

In the completed HE project, the results are in the analysis phase, while the VE project is still in the phase of data generation, since its scope was increased through incorporation into the integral project of the 6th Framework NF-PRO. The results will be submitted in the next revision.

21.6. GEOLOGICAL BARRIER

Activities related to the geological barrier have focused on progressing with techniques for characterizing the performance of compacted clay media and, to a lesser extent, of granitic media.

Granitic Media

Great progress has been made, in relation to granitic media, in the verification of geophysical techniques (high-resolution seismics) to characterize hydraulically conductive discontinuities in this type of medium. These technologies are being applied successfully in supporting underground civil works associated with the optimization of tunnels.

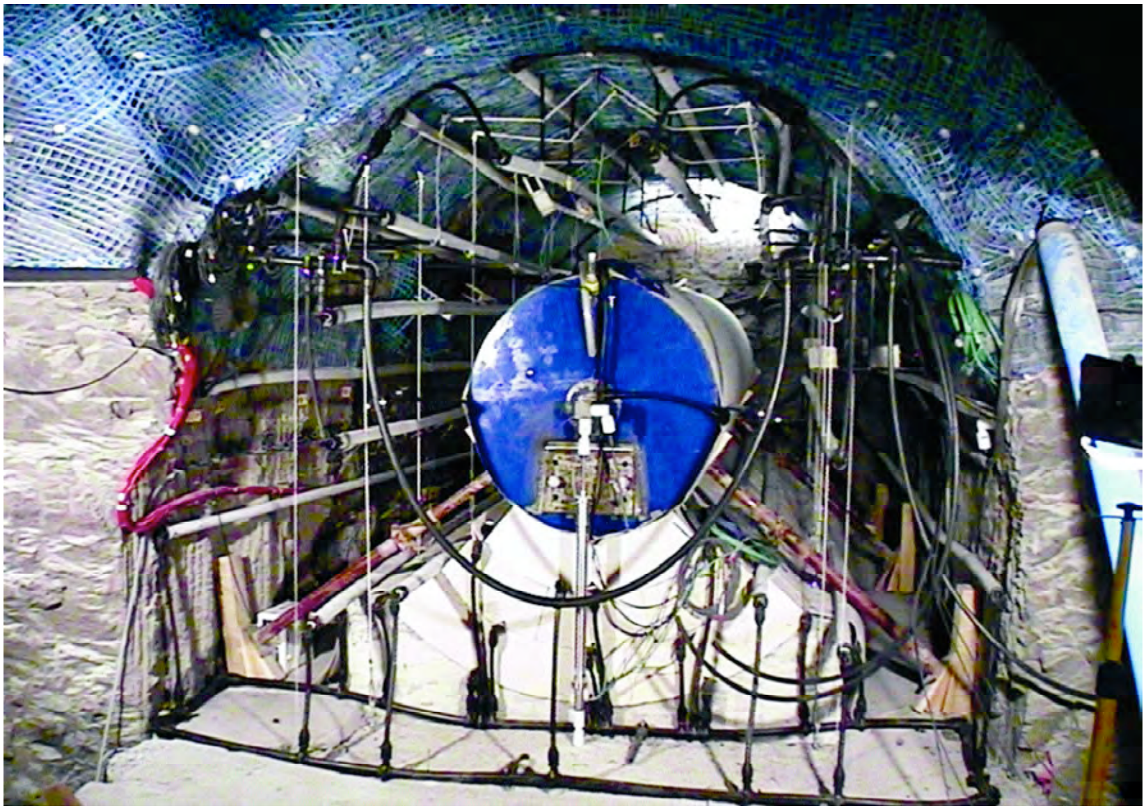


Figure 21.5. Construction of the EB experiment. Mt. Terri URL

Aspects of hydrogeological and geochemical characterization and modeling have been developed in association with the FEBEX project. The objective was to analyze the evolution and behavior of the granite surrounding the experiment during the heating and saturation of clay. The models belonging to the TRANSIN, RETRASO, CORE and INVERTO series have been notably improved. The colloid-induced transport of radionuclides has been studied at Grimsel within the framework of the CRR project, coordinated by NAGRA. Table 21.2 indicates the main activities that have been carried out in the investigations on granitic media.

Clay Media

The projects carried out at Mont Terri (DI-B, DI- PC, etc.) have served to develop and verify technologies for the extraction and *in situ* characterization of pore water for the performance of diffusion tests and improvement of the technique of clay water geochemical modeling. Figure 21.6 shows the methodology developed by diffusion testing. The INVERTO, TRANSIN, CORE and CODE BRIGHT codes have been adapted for their application to this type of material. Table 21.3 indicates the main demonstration activities in relation to clays.

Radionuclide Migration

The focus has been on radionuclide migration processes in the FEBEX, TRUE BLOCK SCALE, RATONES, CRR, DI-B and MATRIX projects, completing the studies performed in the field of natural analogues. The main migration experiments are indicated in Table 21.4. Technologies have been fine tuned for diffusion testing, the analysis of reaction surfaces, the acquisition of realistic distribution coefficients, modeling, etc.

Natural Analogues

The natural analogues studies were related to bentonite and spent nuclear fuel behavior (Barra and matrix natural analogues). In parallel, a database incorporating the most relevant published data was developed in collaboration with the Spanish Nuclear Safety Council (CSN).

Table 21.2. Activities for the demonstration of methodologies and technologies in granitic media

Projects	Activities
Berrocal Project: (Disused uranium mine, Spain)	Demonstration of migration testing technologies and verification of flow and radionuclide transport models in fractured media.
Ratones Project (Disused uranium mine, Spain)	Demonstration of new geophysical, hydrogeological, geochemical, and mechanical technologies. This has been the best ENRESA field of demonstration of the performance of crystalline media.
FEBEX Project (Grimsel-Switzerland)	Hydrogeological characterization of low-permeability media, techniques for the characterization of physical matrix properties, colloid transport and THM behavior of the geological barrier and EDZ in granites.
GAM Project (Grimsel-Switzerland)	Development and verification of tracer test technologies and transport models.
True Block Project (Äspö-Sweden)	Coupling of stochastic and deterministic models for the interpretation of flow and transport.

Table 21.3. Demonstration activities in clay media

Projects	Activities
HE (Mt. Terri)	Propagation of heat and performance of the compacted clay barriers (Mont Terri) in clay media (hydration, deformation, chemical evolution, etc.).
EB (Mt. Terri)	Positioning and hydration of clay barriers combining blocks and pellets. Readjustment of the EDZ following hydration of the clay blocks and barrier. Constitutive models and numerical models.
ED-B (Mt. Terri)	Response of the formation to different excavation techniques. Analysis of EDZ development and evolution EDZ.
DI-B (Mt. Terri)	Radionuclide migration under real conditions.
DI-A (Mt. Terri)	Extraction of pore water from the clay and diffusion testing.
VE (Mt. Terri)	Large-scale hydromechanical performance tests through the saturation/ desaturation of the geological barrier by the ventilation system. Evolution of hydromechanical properties in the repository depending on the time elapsing between excavation and emplacement of the wastes, in keeping with specific ventilation conditions. Evolution of the EDZ during the construction phase.

A natural analogue to the bentonite engineered barrier was found in the Miocene volcanic rock of SE Spain. The main processes studied were the high salinity fronts (seawater-bentonite interactions as an analogy of saline water intrusion in the near field of the repository) and thermal effects (temperature gradients induced by subvolcanic events as an analogy of thermal gradients induced by high-level radioactive waste (Figure 21.7). The participants were a group of Spanish institutions coordinated by ENRESA (CIEMAT, CSIC, ENVIROS, Granada University) and institutions from other countries: SKB, IRSN, and Prague Charles University. Some important features, events, and processes (FEPs) under consideration are the cementation and accessory minerals under thermal and salinity gradients, the variation of swelling pressure caused by accessory mineral behavior, and the effect of cementation on mechanical properties.

The work performed has covered geological, geophysical, geochemical, and mineralogical studies. The highest salinity in pore waters has been locally measured to be 10 times that of seawater. Swelling properties do not present significant modifications deriving from interactions with these very high salinities.

With respect to thermal effects, stable isotopic studies indicate a temperature range for carbonates interstratified with volcanic tuffs of between 28 and 90ºC. The

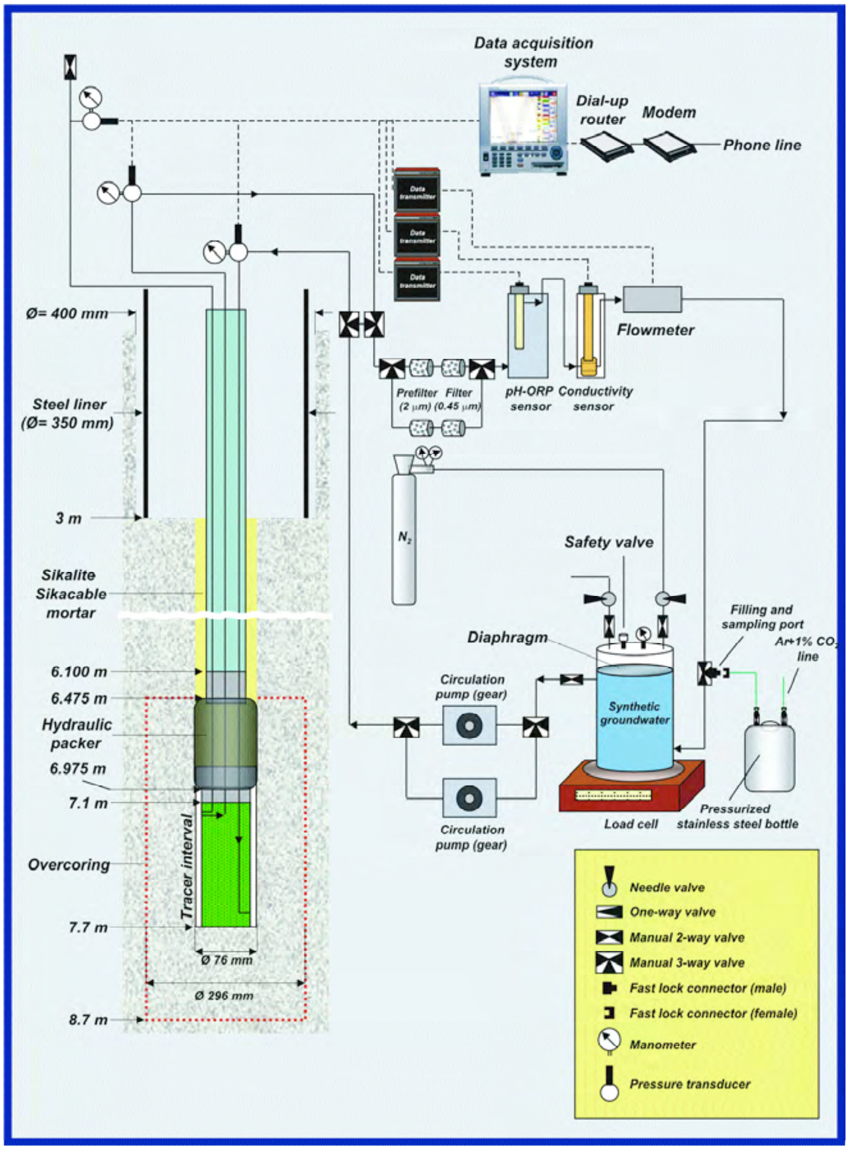


Figure 21.6. Diffusion experiment Mt. Terri URL

temperature range is pertinent for most of the deep geological concepts. No relevant modifications in mineralogical characteristics and physical properties have been identified.

Matrix (Analogue of Spent Nuclear Fuel Oxidation)

In western Spain, within a closed uranium mine (Mina Fe), a series of analogue features were identified:

- Large U concentrations (UO_2+x)
- Interaction with groundwater
- Weathering products similar to those from experimental dissolution of UO_2
- Presence of Fe(III)-oxyhydroxides similar to canister corrosion products
- Presence of clay materials

The objectives of the study were the *in situ* investigation of the oxidative dissolution of pitchblende, the potential retention of U(VI) in secondary phases, and the retention properties of RN in Fe(III) oxyhydroxides and clays (Figure 21.8). The main findings were the oxidation of UO_2 and pyrite in the oxidized zone and migration of

Table 21.4. Radionuclide migration studies (1999–2003)

Parameter/ Process	Type of Experiment	Physiochemical Conditions	Element/ Radionuclide	Material	Project
Complexing constant. Coefficients of selectivity.	Sorption depending on: pH Ion force Concentration	reducing	caesium selenium uranium plutonium	BIOTITES (Ratones granite, standard)	FISQUIA
Complexing constants. Coefficients of selectivity. Surface charge. Density of coordination sites.	Sorption depending on: pH Ion force Concentration	reducing	selenium plutonium calcium strontium	homoionic bentonite (Na, Ca)	FISQUIA
Complexing constants. Coefficients of selectivity. Surface charge. Density of coordination sites.	Sorption depending on: pH Ion force Concentration	reducing and oxidizing	uranium plutonium	geothite magnetite	ACTAF
Delay factor	Fractured column test	oxidizing	HTO uranium calcium sodium chlorine selenium	Ratones granite	FISQUIA
Delay factor	Fractured column test	reducing	HTO uranium chlorine	Grimsel granite	CRR
Coefficients of diffusion. Accessible porosity	Through-diffusion In-diffusion	oxidizing	HTO uranium chlorine sulfur europium neodymium	FEBEX bentonite	FEBEX
Coefficients of diffusion.	RBS	oxidizing	uranium technetium selenium europium	granite	FISQUIA
K_d	Batch	reducing	uranium technetium selenium caesium	granite fissue backfill bentonite colloids	CRR
Sorption isotherms Irreversibility	Batch	reducing	caesium uranium strontium	bentonite colloids	CRR
Complexing constant. Coefficients of selectivity. Surface charge. Density of coordination sites.	Sorption depending on: pH Ion force Concentration	reducing	caesium uranium	FEBEX bentonite	FEBEX

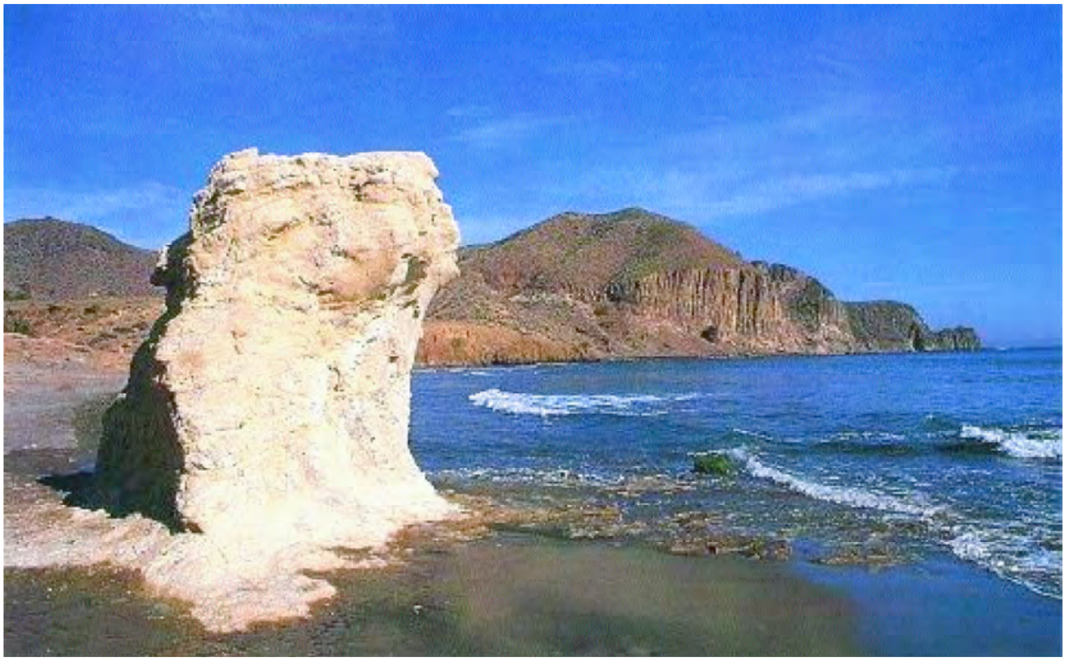


Figure 21.7. Barra natural analogue: saline effect on bentonite behavior

UO_2. In the transition zone, U(VI) co-precipitates with Fe (III)-oxyhydroxides, and pitchblende is transformed into coffinite. The redox of the system appears to be controlled by the oxidation of organic matter. Traces of Ni associated with Fe(II) and Mn (IV) oxides were found, as well as the secondary formation of Se(s) or SeO_2 in the transition zone, this being of relevance for solubility calculations.

Biosphere

The activities performed in relation to the biosphere and associated with DGD have been oriented towards the following:

- Improved understanding of radionuclide behavior in the different compartments of the biosphere, their interaction mechanisms, mobilization, transport, etc., along with the associated numerical models.
- Palaeostudies: The aim here is to establish the way in which the environment has evolved and what parameters control this evolution, as a basis for reliable estimation of future evolution. This requires techniques for the identification of environmental changes relating to the evolution of the biosphere and the establishment of paleo-environmental sequences (temperature, rainfall, vegetation, infiltration, etc.)
- Establishment, based on the above, of the most probable future evolution (future biosphere) and calculation (under these conditions) of the effects or impact of the presence of radionuclides (bio-prospecting).

With a view to the above, the BIORAD program has been developed, with CIEMAT, UPM-ETSIMM, ENVIROS, and UDC being the fundamental organizations supporting these activities and with a high degree of participation in the following EU programs:

- EQUIP
- BIOCLIM
- PADAMOT
- BIOSMOSA
- BIOPROTA

The results of the biosphere program have been applied systematically to the safety assessment exercises undertaken within the ENRESA 2000 (granite) and ENRESA 2003 (clay) projects.

Safety Assessment

The generic ENRESA 2002 and ENRESA 2003 exercises have been completed, and the knowledge acquired to date has been incorporated within the R&D program.

Figure 21.8. Matrix natural analogue (oxydative dissolution of pitchblende. ox-red-zone). Radionuclide migration under variable redox conditions (ox-red-zones)

The methodology developed by ENRESA considers the following steps:

- Description of the disposal system: initial state.
- Analysis of possible future evolution of the system (scenarios): The integration of the most significant factors gives rise to the "reference system," the normal evolution of which constitutes the "reference scenario."
- Analysis of barrier performance.
- Analysis of radiological consequences (reference scenario and other selected scenarios constituting the reference scenario).
- Analysis of results: sensitivity, uncertainty and comparison of safety criteria.

Deterministic and probabilistic calculations have been used for the analysis of consequences. The analyses and calculations cover a period of up to one million years.

The sources of uncertainty considered have been as follows:

- Uncertainty linked to forecasting the production of barriers and biosphere
- Uncertainties associated with selection of the conceptual model
- Uncertainties associated with limited knowledge of processes, data, and model input parameters

The dose acceptance criterion is 0.1 mSv/year, which is a minor fraction of the average dose for the Spanish population as a result of radiation of natural origin, which is 2.4 mSv/year.

In both ENRESA 2000 and ENRESA 2003, the annual and average doses obtained for the reference scenario are below the permissible maximum value.

Basic DGD

With a view to providing support for future decision making, ENRESA has included the available information on geological disposal in granites and clays in the corresponding strategic documents. These documents (which are not published) consider the available information in reference to the following:

- Reference concept: characteristics of components, basic design, and cost evaluation.
- DGD components: research and development activities relating to waste, engineered barrier, geological barrier, and biosphere
- Safety assessment
- Conclusions and recommendations: design and R&D

These documents will be updated on the basis of the progress made in relation to both R&D and safety assessment or design. They are living documents, the objectives of which are to facilitate decision making regarding the definitive management of fuel. They also constitute a good tool for forecasting in relation to activities linked to deep geological disposal.

21.7. CONCLUSIONS

Activities relating to deep geological disposal in Spain have been aimed at developing the technologies required for its implementation, improving key areas of knowledge required for safety assessment (with such knowledge being incorporated in generic safety assessment exercises for granite and clays), and integrating this knowledge within basic reference documents supporting future decision making. All the activities have been performed within a framework of close and efficient international collaboration. The current result of the above is the availability of equipment and capacities that will allow for implementation of solutions when they are considered appropriate.
The transition from generic activities to specific activities is an issue that has yet to be decided and is not a short-term priority in the new ENRESA strategy.

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Chapter 22

The Swedish Program for Spent-Fuel Management

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ABSTRACT. The Swedish principal alternative for a final geological repository for spent nuclear fuel is called the KBS-3 method. It involves encapsulating the fuel in copper canisters with cast iron inserts and embedding each canister (surrounded by bentonite clay) at a depth of about 500 m into the bedrock. An overall goal for the Swedish Nuclear Fuel and Waste Management Company (SKB) is that the first stage of the repository should be ready for initial operation in 2017. Regular operation should then be able to start around 2023.

The siting process for the final repository has now reached the site investigation phase. Two municipalities have been selected from the original eight. These are Oskarshamn in southeast Sweden and Östhammar in the region of Northern Uppland. SKB is now compiling the material needed to choose the most suitable location, as well as to apply for a permit for the encapsulation plant in 2006 and in 2008, and for the final repository in 2008. This work includes safety analyses based on site-specific repository designs and environmental impact statements.

22.1. BACKGROUND

Sweden has been generating electricity using nuclear power for more than 40 years. The first commercial reactor was put into operation in 1972, and the latest in 1985. A referendum in 1980 limited the nuclear program to 12 reactors. Two of these—Barsebäck 1 and Barsebäck 2—were closed down in 2005 and 1999, respectively, due to political reasons. The 10 remaining reactors are operated by three utilities and supply almost half of the nation's electricity. Their total capacity amounts to 9,200 MW. Seven of the reactors are boiling-water reactors (BWRs) and three are pressurized-water reactors (PWRs).

Back in the 1970s, legislation charged the nuclear power industry with the responsibility for managing and disposing of all the radioactive waste from its installations in a safe manner. In response, the owners of the nuclear power plants formed SKB (Swedish Nuclear Fuel and Waste Management Company) for this purpose. SKB's task is to plan, construct, own, and operate the systems and facilities necessary for transportation, interim storage, and final disposal. A fund to finance the activities was set up a few years later. Today, a final repository, a canister factory, and an encapsulation plant remain for the system to be complete.

Up to now, approximately 4,200 tonnes of spent nuclear fuel (SNF) have been used in power production and are at present stored in Clab—the central interim storage facility—outside Oskarshamn. As mentioned above, the Swedish government has decided to start phasing out reactors. Because of this, it is difficult to estimate the total amount of SNF to be processed in the future. A 40-year operating period will (for example) result in a total of 4,500 canisters, equivalent to 9,300 tonnes of uranium. Deposition in the final repository will be concluded in ~2050. The different repository units can then be sealed and the buildings decommissioned prior to 2060. However, both smaller and larger fuel quantities must be handled. The variables affected are mainly the operating time of the system and the space requirements for the final repository.

Over the past decades, SKB has built up a system to manage various kinds of radioactive waste. The structure of the Swedish system is shown in Figure 22.1. There is a specially built ship to transport the waste, a final repository for various types of radioactive waste, and an interim storage facility (Clab) for SNF.

SKB's principal alternative for SNF disposal is called the KBS-3 method. The disposal concept involves encapsulating the SNF in copper canisters and embedding each canister in bentonite clay at a depth of about 500 m in the bedrock, as shown in Figure 22.2. The final repository consists of an access tunnel and a system of deposition tunnels. Each deposition tunnel contains a number of vertical holes in which the canisters with SNF will be deposited. After the canisters have been placed in the holes and surrounded by tightly compacted bentonite, the tunnel is filled with clay and crushed rock.

22.2. ACTION PLAN

The current situation in the nuclear fuel program can be summarized as follows:

- The development of the KBS-3 method is in a phase featuring pilot- and full-scale tests of parts of the system. The activities are largely being pursued at the Canister Laboratory and the Äspö Hard Rock Laboratory (HRL), both situated in Oskarshamn.
- Design of an encapsulation plant is under way, at the same time that development of the encapsulation technology is proceeding. The encapsulation plant should preferably be situated adjacent to Clab. A siting at a possible final repository in Forsmark will be studied as an alternative for comparision. SKB will apply for a permit under Nuclear Activities to build an encapsulation plant adjacent to Clab in 2006. In 2008, a corresponding application under the Environmental Code will be submitted.
- There are two candidate sites at which a deep repository could be sited: Forsmark, in the municipality of Östhammar, and Laxemar, in the municipality of Oskarshamn.
- Investigations and design of a final repository are being pursued at both these sites. SKB will apply for a permit to build and operate a repository on one of the sites in 2008.
- Both the encapsulation plant and the deep repository require permits under the Environmental Code and the Nuclear Activities Act. Statutory consultation procedures for this have begun.

An overall goal for SKB is that the first stage of the repository for SNF should be ready for initial operation in 2017. Regular operation should then be able to start in ~2023. According to SKB's action plan (SKB, 2004a) and as shown in Figure 22.3, it will take at least 40–50 years to carry out all measures needed to dispose of all SNF in a safe manner. It is, therefore, appropriate to proceed in steps and keep the door open for technological development, changes, and possibilities for retrieving canisters that have already been disposed. This will ensure freedom of choice for the future, while at the same time demonstrating final-disposal-method performance on a full scale and under actual conditions.

Approximately 5–10% of the total quantity of SNF will be deposited in the repository upon its opening—equivalent to about 200–400 canisters. The numbers are based on the 40-year operation scenario previously described. If the evaluation of the first phase shows that the method has deficiencies or that better methods exist, the canisters will be retrieved. We must therefore show that it is technically feasible to retrieve the canisters before starting deposition. If the evaluation gives a positive result, we will apply for an operating license to begin regular operation, and the remaining SNF will then also be disposed. The regular operation phase is projected to last for between 20 and 30 years, and the deposition rate amounts to 160 canisters per year on average. Decisions regarding siting, construction, and operation of an encapsulation plant and a final repository will also be taken in steps, based on progressively more detailed information.

22.3. SITE INVESTIGATIONS

Site investigations, started in 2002, are divided into three main phases, as shown in Figure 22.4. Initial site investigations are performed to identify the site within a specified area deemed to be most suitable for a final repository, and to determine whether the feasibility study's judgment of the area's suitability holds up in light of in-depth data. If the assessment still stands up, complete site investigations will follow. The purpose of the complete site investigations is to gather the material required to select one of the sites as the main alternative, carry out consultations, and compile an application to build a final repository. The application will eventually be reviewed by the appropriate authorities.

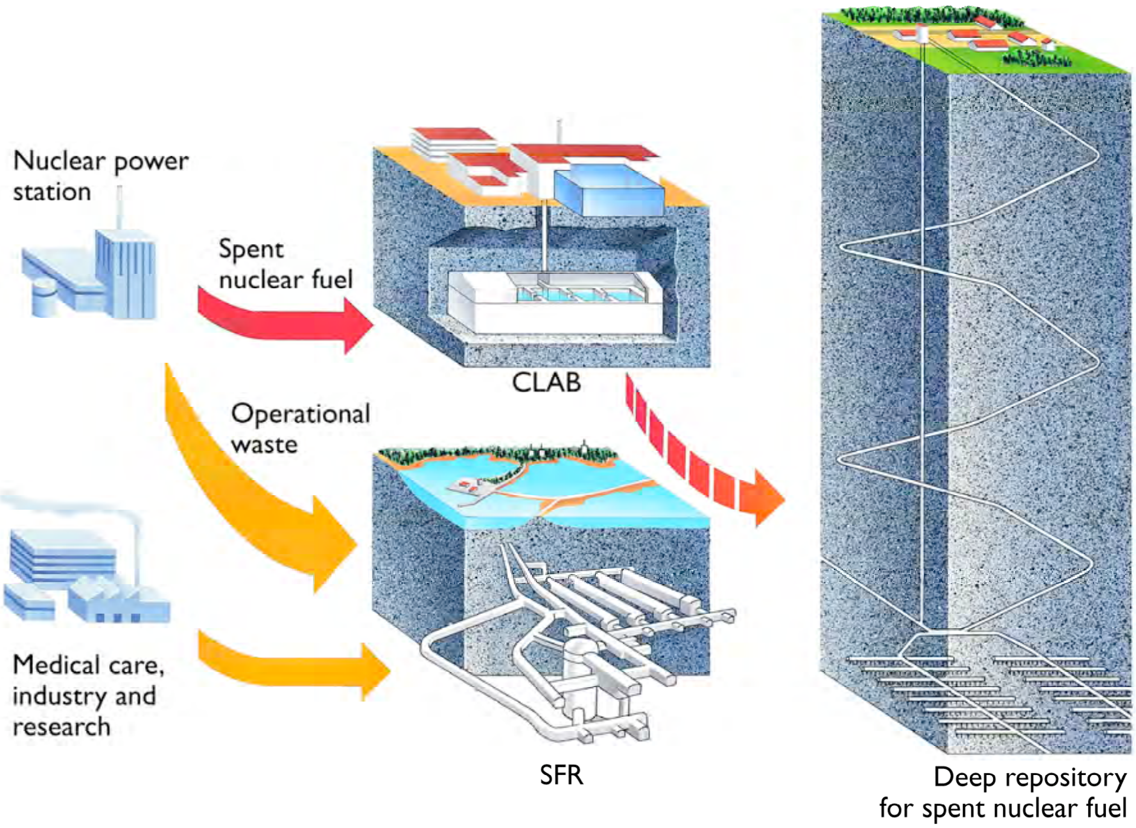

Figure 22.1. The Swedish system for handling radioactive waste

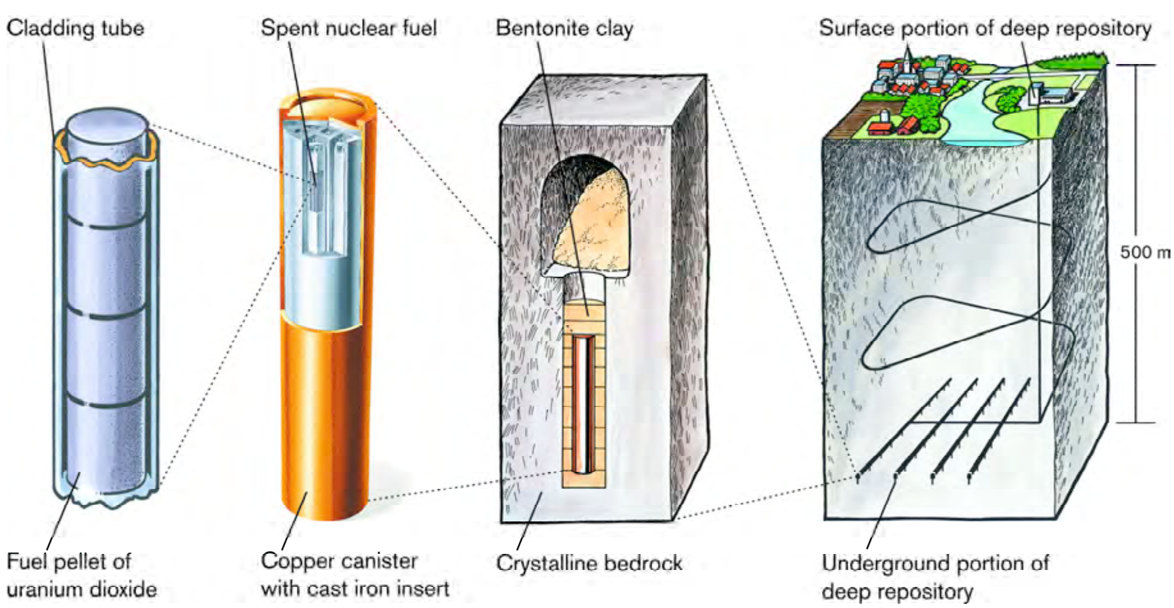

Figure 22.2. The SNF will be encapsulated in copper, embedded in bentonite clay, and emplaced at a depth of 500 m in the rock.

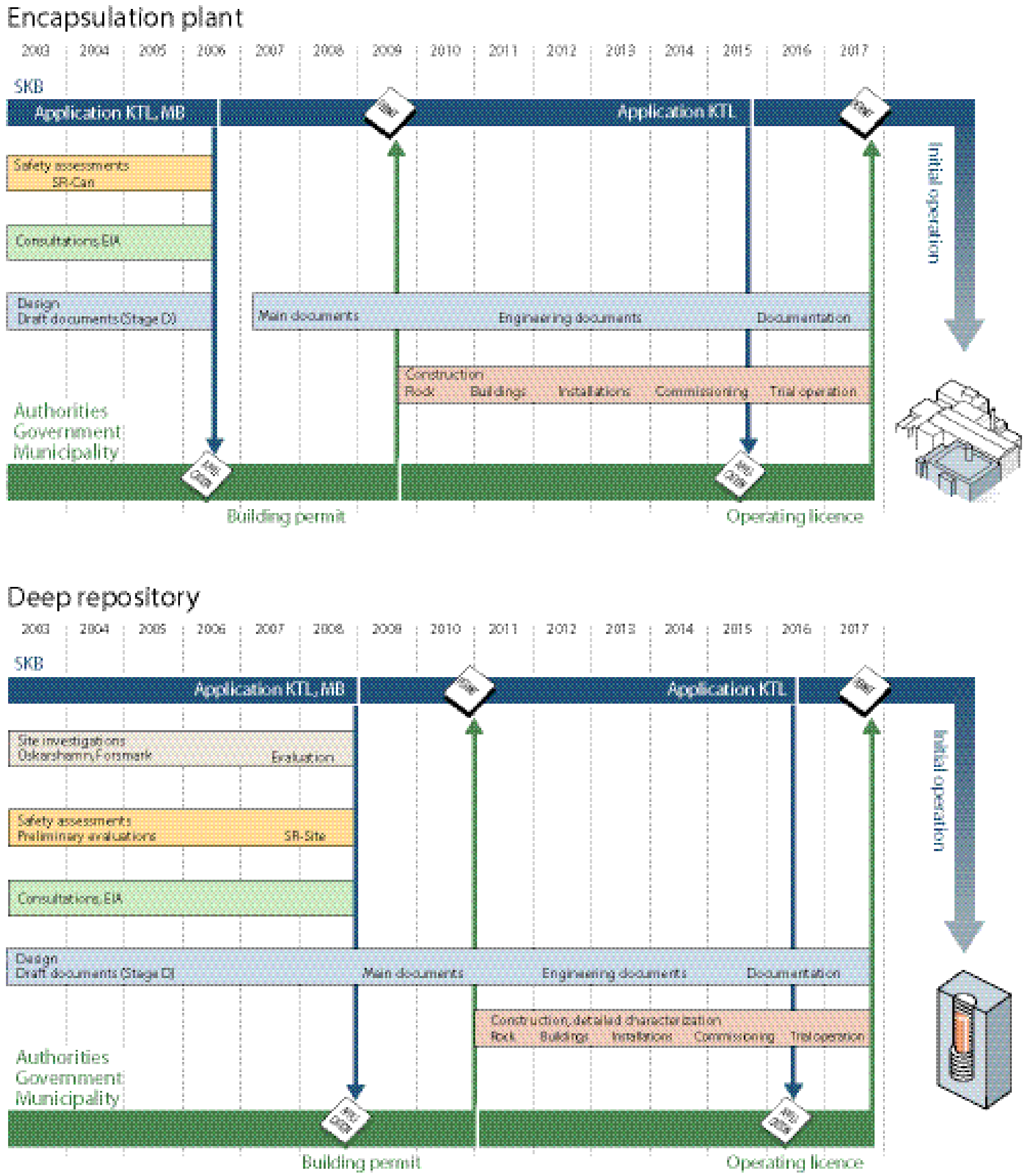

Figure 22.3. Reference time schedule for final repository and encapsulation plant

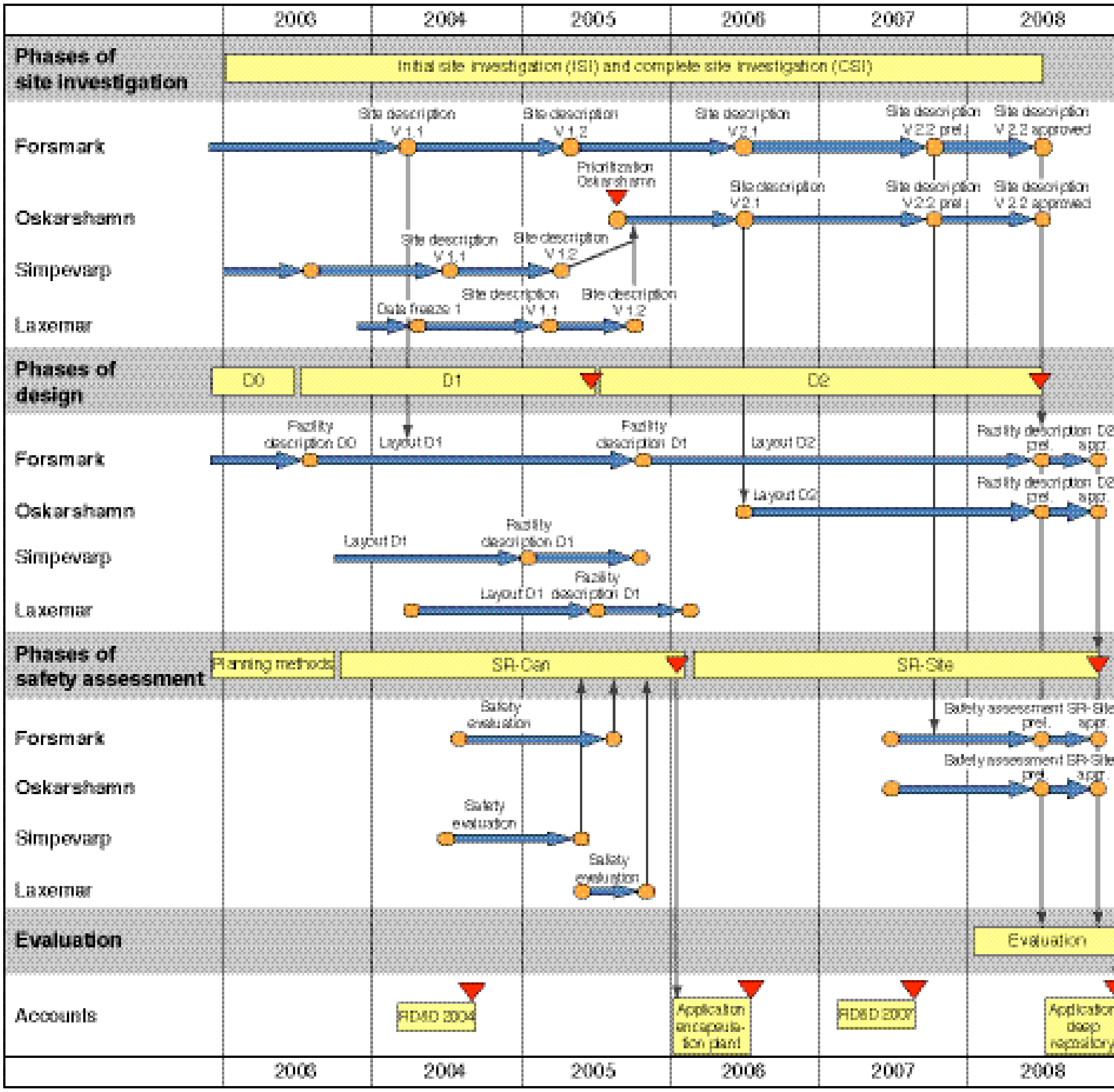

Figure 22.4. Structure of the site investigation program

At present, the initial investigation phase is ongoing. The investigations are performed according to site-specific execution programs (SKB, 2002a; SKB, 2002b). The programs were designed with respect to the state of knowledge and specific questions for each site. They also needed to take into account the viewpoints of the municipality, landowners, and nearby residents, as well as nature conservation and other interests. For each of the investigated sites, the initial site investigation will result in:

- A preliminary site description
- A preliminary facility description
- A preliminary safety evaluation

22.3.1. Site Descriptions

An important component of the characterization work is the development of a site conceptual model that constitutes an integrated description of the site and its regional setting. This model covers the current state of the geosphere and the biosphere, as well as those ongoing natural processes that affect their long-term evolution. The site description includes two main components:

- Synthesis of the site characterization summarizing the current state of knowledge, as well as describing ongoing natural processes that affect their long-term evolution
- One or several site-descriptive models, in which the collected information is interpreted and presented in a form that can be used in numerical models for rock engineering, environmental impact, and long-term safety assessment

Conceptual models are developed on a regional scale (hundreds of square kilometers) and on a local scale (tens of square kilometers). The regional-scale model serves to provide boundary conditions for the local-scale models. Descriptive model versions are produced at specific times (e.g., repository design, safety assessment) that are adapted to the primary users, These specified moments define different "data freezes," which single out the database informing the model in question. The results of descriptive modeling also serve to produce feedback to, and set the priorities for, the ongoing site characterization. The data flow from site investigations to site description is shown in Figure 22.5. Preliminary site descriptions (version 1.2) have been published for both Forsmark (SKB, 2005a) and Simpevarp (SKB, 2005b). The preliminary site description for Laxemar is due in early spring 2006.

Investigations in Forsmark

The area SKB is investigating lies southeast of the Forsmark Nuclear Power Plant, and the candidate area where the deep repository could be located is a coastal area of about 10 km^2, (see Figure 22.6). The vegetation is dominated by forest, with pine as the dominant forest type, and wetlands are common. Parts of the area are classified as being of national interest for nature conservation. SKB is therefore performing ecological investigations, including extensive inventories of the flora and fauna. The results will be input to the environmental

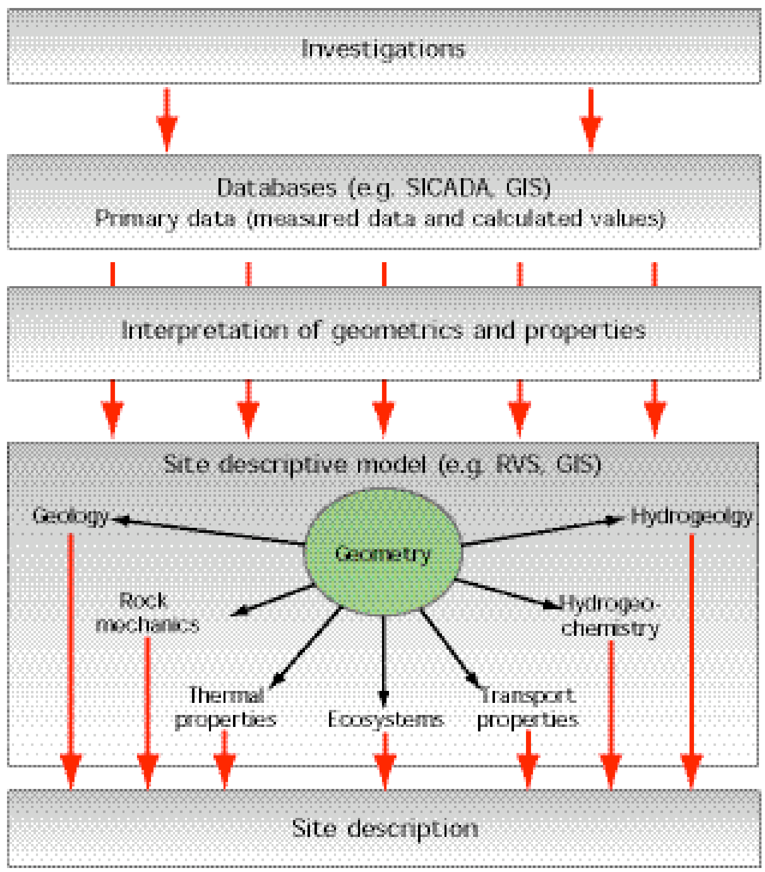

Figure 22.5. Primary data from site investigations are collected in databases. Data are interpreted and presented in a site descriptive model, which consists of a description of the geometry of different units in the model and the corresponding properties of the site.

impact statement.

The region is part of the sub-Cambrian peneplain belonging to the Fennoscandian shield and is relatively flat with a gentle slope to the northeast. The candidate area lies at low altitude. Quaternary deposits are more dominant than exposed bedrock or bedrock with only a thin Quaternary cover. The predominant rock type in the region is grey-to-red equigranular metagranitoids.

The major part of the bedrock was formed 1,900 million years ago, and it has been affected by both ductile and brittle deformation. The ductile deformation has resulted in large-scale ductile high-strain zones, and the brittle deformation has given rise to large-scale faults and fracture zones. Tectonic lenses, within which the bedrock is much less affected by ductile deformation, are enclosed between the ductile high-strain zones. The candidate area is located in one of these tectonic lenses.

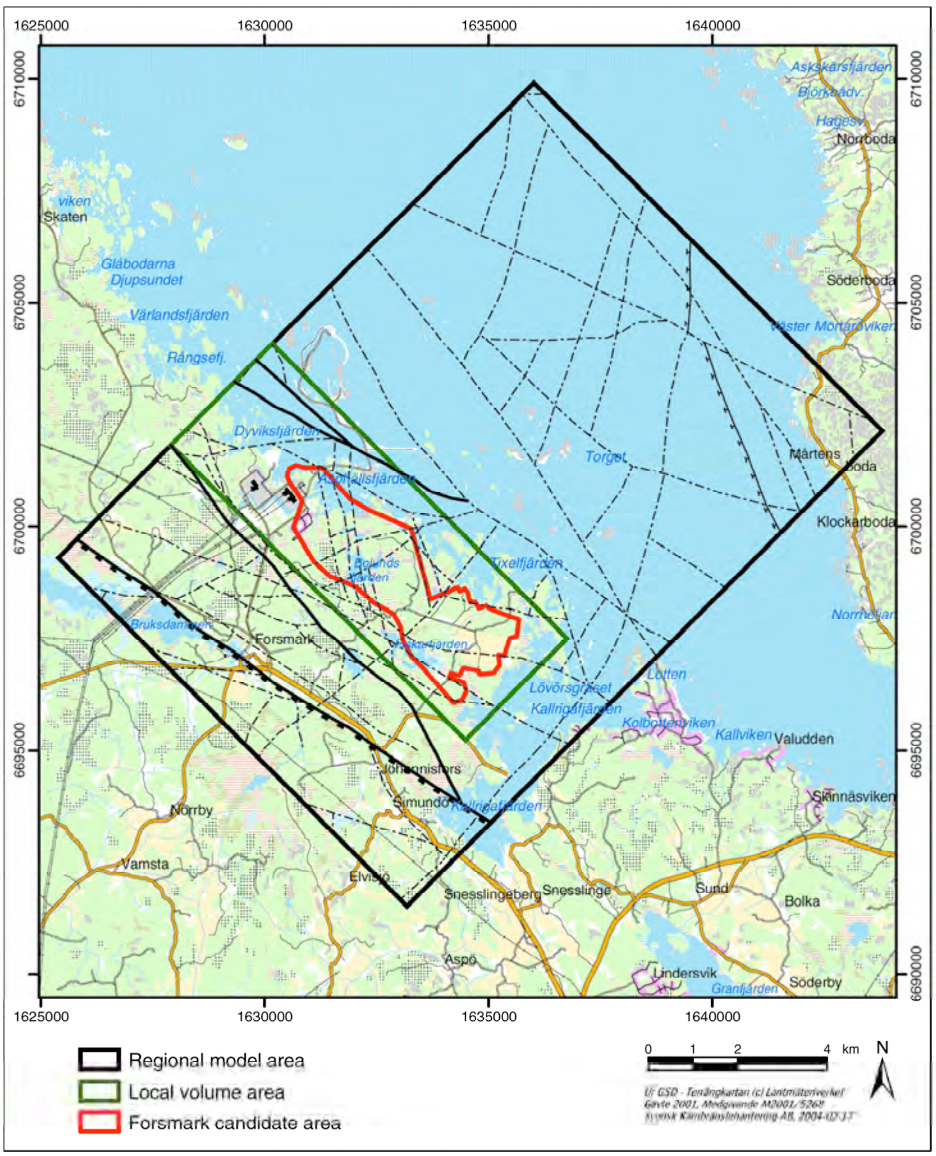

Figure 22.6. The Forsmark candidate area (red) and the regional model area (black) in the preliminary site descriptive model

Table 22.1. Current status of site-specific key issues in Forsmark

Key Issue	Status
The vertical extent and shape of the tectonic lens	Several boreholes have verified a depth in excess of 1,000 m. The presumed steep dip at the margins of the lens has been verified for certain sections. For other sections, further investigations remain to be done before the issue can be settled.
Possible ore potential at depth	The issues have been studied particularly closely in conjunction with the holes that have been drilled. No traces of mineralizations and/or bedrock with ore potential have been detected within the lens. The issue should therefore be able to be dismissed.
Presence of gently dipping fracture zones	Highly conductive, gently dipping fractures and fracture zones are common down to a depth of about 200 m. Only a few gently dipping fracture zones have been found at greater depth.
Possible occurrence of high rock stresses	The measurements made so far have shown relatively high, but not extreme, rock stresses. Additional measurements are planned in order to obtain a better overall picture.

The site investigation was initiated at the beginning of 2002. During the first year, work organization and activities were developed and reached full scope. The current status of the investigation can be summarized as follows:

- Characterization of the area's surface geological and ecological conditions have been completed.
- Ten deep cored boreholes (1,000 m) and a number of shallower percussion boreholes (maximum depth about 200 m) have been completed for investigations of the bedrock.
- A comprehensive preliminary site description (version 1.2) has been presented.
- The consultation process for a possible final repository in Forsmark is ongoing.
- An active information and communication program has been established for ongoing dialogue with nearby residents, the public, the municipality, and other local stakeholders.

The drilling results show that the rock is relatively dry and fracture-poor at depth, but more hydraulically conductive near the surface. This is particularly clear in the deep-cored borehole nearest the Forsmark plant. The nonconductive rock starts in this hole at a depth of only about 200 m. There are indications that the rock in Forsmark may have high rock stresses. Such indications will probably not prevent the construction of a deep repository, but the stresses are important for the layout of the tunnel and the repository. As far as hydrochemical and thermal conditions are concerned, the investigations have not revealed any surprises, which suggests favorable conditions at repository depth. During the complete site investigation, efforts will be concentrated in the northwest part of the tectonic lens.

Before the site investigation was initiated, a number of site-specific geoscientific issues were identified as being particularly important to clarify in order to judge the suitability of the site for a final repository. Table 22.1 summarizes the current status of these key issues in terms of how well they are understood.

Investigations in Simpevarp and Laxemar

Site investigations in Oskarshamn have been conducted at two adjoining localities, the Simpevarp and Laxemar subareas. The size of the Simpevarp subarea is approximately 6.6 km^2, whereas the Laxemar subarea covers some 12.5 km^2. Both subareas are located close to the shoreline of the Baltic Sea (see Figure 22.7).

The main argument for a site investigation in the Simpevarp/Laxemar area was the combination of bedrock judged to be favorable and the advantages offered by Simpevarp for industrial establishment in the form of available industrial land and infrastructure, the Clab facility, and the planned co-siting of the encapsulation plant.

The two areas have different geological conditions, and the investigations are based on somewhat different system solutions for a repository. A siting on the Simpevarp

Figure 22.7. The Simpevarp and Laxemar candidate areas (red) in the preliminary site descriptive model

Peninsula would mean that the surface facilities would also be located there. A siting in the Laxemar area could entail either that certain aboveground facility parts are located on Simpevarp (connected by tunnel to the repository), or that the entire facility is placed in Laxemar.

The predominant lithologies in the Oskarshamn area are rocks from the Transscandinavian Igneous Belt (TIB), which is a mix of granites, syenitoids, dioritoids, and gabbroids emplaced approximately 1,800 Ma during the end stages of the Svecokarelian orogeny. Younger rocks, including coarse- to fine-grained granitic plutons (1,450 Ma) and dolerite dikes (1,000–900 Ma), are also encountered.

Questions existed regarding the bedrock on the Simpevarp Peninsula, because of the relatively high frequency of fractures and thus uncertainty regarding the space available for a deep repository. However, given the advantages offered by the site in other respects, an investigation was judged to be warranted.

Boreholes drilled in the Simpevarp Peninsula, in combination with lineament interpretations, show that the Simpevarp subarea is probably surrounded by deformation zones of such importance that they constitute boundaries for a possible repository area. Within the area are lineaments that can be interpreted as zones of lesser importance. Most of the fractured formation in the boreholes exhibits relatively low hydraulic conductivity.

SKB has decided not to conduct a complete site investigation at this site. The complete site investigation is instead going to be conducted in the southern parts of the Laxemar area. The predominating rock in this area is quartz monzodiorite. The rock normally has a high fracture frequency, about one fracture per meter. It is intersected by fracture zones of varying size with a higher fracture frequency, which are more conductive than the rock between the zones. As a whole, the hydraulic conductivity of the Laxemar rock can be described as fairly typical for granitic rock.

At the level where the deep repository may be built, the hydraulic conductivity of the rock is usually low. Detailed measurements have shown that there are plenty of zones with dry rock. But at repository depth, there are more conductive fracture zones or individual fractures between the dry zones.

Long-term safety requires that the groundwater be free of dissolved oxygen and that its salinity be less than ten percent. Both of these requirements are met in Laxemar, and the results indicate that the repository may be located in fresh (nonsaline) groundwater. Measurements in the boreholes show that the water is fresh down to a depth of between 400 and 600 m.

22.4. FACILITY DESCRIPTIONS

The final repository will be designed and engineered prior to the permit application. Technology will be developed for mining of tunnels and sealing the rock around them, fabricating and emplacing the buffer, handling the canisters, and finally backfilling and plugging the deposition tunnels. Other areas being studied are sealing of boreholes and technology for retrieving canisters after initial operation, if this should prove necessary. Design of the final repository is based on the results of technology development and site investigations, and is coordinated with the environmental impact statement, as well as a safety assessment and a system analysis.

During the site investigation phase, the design process is divided into a number of design steps, which are linked to the stages of the site investigation and provide supporting material for various versions of facility descriptions. The different design steps are called D0, D1, and D2. The design work in each new step is based on the products of preceding design steps and the updated site description. The result of design step D0, Layout D0, has been completed. Step D1 will be carried out based on the underground design premises (SKB, 2004b) as well as the results of the initial site investigations. Layout D2 is the result of the complete site investigation.

22.5. SAFETY ANALYSES

SKB's work with safety assessment for a final repository, and the method and model development for it, is being conducted during the site investigation phase. This work is almost exclusively within the framework of the projects aimed at producing safety reports for permit applications to build an encapsulation plant and a final repository. The safety assessment projects are called SR-Can and SR-Site. An interim report for SR-Can, published in 2004 (SKB, 2004c), presented the methodology for safety assessment prior to its use in the planned permit applications. The report demonstrates not only completed methodology development with examples, but also plans for how different problems are intended to be solved in the final report for SR-Can, due in October 2006. The final safety report for the final-repository application, SR-Site, is due in 2008.

22.6. CONCLUSIONS

Two site investigations are being conducted according to plan. Crucial prerequisites to realizing the goal of submitting permit applications for the encapsulation plant and the deep repository are:

- Application for a permit to build an encapsulation plant assumes co-siting at Clab. If the Clab siting is rejected as the main alternative, the timetable must be revised. The second-choice alternative is a siting at Forsmark, which assumes that the final repository is also located on this site.
- The initiated site investigation for the final repository can largely be carried out within the planned time frame, and with results favorable enough to warrant an application for a permit to build and operate the final repository on one of these sites. An outcome in which additional candidate sites for the final repository must instead be added to the siting program would require revision of the timetable.
- Development and optimization of the technology for encapsulation and final disposal of SNF can

continue at roughly the planned pace and with the anticipated results. Unexpected problems at some critical point may lead to delays and revision.

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Chapter 23

Swiss Geological Studies to Support Implementation of Repository Project: Status 2005 and Outlook

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23.1. BACKGROUND

Switzerland is a small country, with limited natural resources (other than hydropower), and must import about 80% of its primary energy needs (predominantly petroleum products). Electricity supplies about 20% of the energy demand, with about 40% of the electrical energy supplied from nuclear plants and almost all of the rest from hydropower.

Nuclear power production is the main source of Swiss radioactive waste, although waste also results from medical, industrial, and research applications. Switzerland currently has five nuclear power plants (NPPs) (pressurized-water reactors [PWR] and boiling-water reactors [BWR]) with a total capacity of 3.2 GW(e). Spent nuclear fuel (SNF), which contains most of the radionuclides, may be prepared for direct disposal or reprocessed to recover usable uranium and plutonium, with the resulting high-level waste being immobilized in glass blocks. Initially, Swiss disposal planning focused on waste returned from foreign reprocessing plants, but currently, the preferred strategy of the utilities is to keep both options open (reprocessing or direct disposal of SNF), while a recent revision of the nuclear legislation restricts future reprocessing, prescribing a 10-year moratorium as of July 1, 2006.

The producers of nuclear waste are responsible for waste management (for all waste categories). Hence, the electricity-supply utilities involved in nuclear power generation and the Swiss Confederation (which is directly responsible for the waste from medicine, industry, and research) joined together in 1972 to form the "National Cooperative for the Disposal of Radioactive Waste" (Nagra). Nagra is responsible for disposal and for providing advice on conditioning of all types of waste. The responsibility for SNF reprocessing and transport, waste conditioning at the power plants, and interim storage remains directly with the utilities.

The legal framework regarding nuclear energy was extensively revised in recent years, and a new law (KEG, 2003), passed in March 2003, came into force on February 1, 2005, together with a supplementary ordinance (KEV, 2004). The new Nuclear Energy Law keeps the nuclear option open and addresses a number of key issues related to radioactive waste, generally requiring a stronger commitment from the Swiss Federal Government for the implementation of repositories for all types of waste. Disposal within Switzerland remains the preferred option. However, disposal abroad might be envisaged as an exception under certain conditions. The new law prescribes disposal of all radioactive waste in geological repositories. Repository closure shall be preceded by an extended monitoring period, during which the waste should be retrievable with reasonable effort. The law requires the waste producers (Nagra on their behalf) to compile a "waste management program" outlining the steps for the planning and operation of disposal facilities in Switzerland. This program will be reviewed by the supervisory authorities and approved by the Swiss Federal Government. New nuclear facilities (e.g., NPPs, disposal facilities) must, as in the old law, apply to the Federal Government for a General License. The new legislation includes extensive provisions for stakeholder involvement, such as consultation of siting cantons and neighboring countries, as well as a facultative nationwide popular referendum. Many of the requirements previously set forth in various ordinances (e.g., regarding financing of radioactive waste management) and in guidelines of the Swiss regulatory body (HSK) have now been integrated into the new legislation or incorporated into a set of new ordinances. The full set of HSK guidelines will be adapted to the new legislation within the next years.

In response to the discussions about site selection, the Swiss Government announced, on December 3, 2004, that a site selection process, including technical criteria as well as socioeconomic aspects, would be defined within the framework of the existing Land Use Planning Legislation in the form of a specific Sectoral Plan for Geological Repositories (Sachplan Geologische Tiefenlagerung as part of the Raumplanungsgesetz). The "Strategy" part of the Sectoral Plan currently being drafted will include extensive provision regarding the involvement of interested parties (cantons, neighboring countries, etc.) and the public in the site selection process. It will also list a number of criteria related to safety. A Government decision on process and criteria is expected in the second half of 2006.

23.2. CHARACTERISTICS AND HISTORICAL EVOLUTION OF THE SWISS NUCLEAR WASTE DISPOSAL PROGRAM

Since the founding of Nagra in 1972, work has been carried out on the development of disposal concepts and identification of potential sites for such facilities. Working on the multibarrier principle, the requirements for packaging, engineered structures, and geological isolation were derived for different types of waste. Two separate geological repositories are planned; one for low- and intermediate-level waste (LILW) and another for SNF, high-level waste (HLW), and long-lived intermediate-level waste (LL-ILW). The option of co-locating these facilities at a single site but not necessarily in the same rock formation is, however, not excluded.

Highest priority was allocated to the LILW repository intended to be implemented in horizontally accessed rock caverns with a few hundred meters of overburden. An extensive site selection procedure resulted (in 1993) in the nomination of Wellenberg in Central Switzerland as the preferred repository location. The principle of developing Wellenberg as a repository site was accepted in public referenda in the local community, but blocked by a narrow margin at the cantonal level in 1995. The concept was revised, but in a cantonal vote in 2002, the plans for the proposed underground investigations were again rejected, and thus the site had to be abandoned for political reasons. The definition of a site selection procedure, including site selection criteria, is now under development by the Federal Office of Energy (see previous section). Nagra is currently compiling technical-scientific material (mainly geological information) to support the site selection process.

For the present limited nuclear program in Switzerland, operation of all plants for 60 years will result in around 4,500 tons of SNF. Existing reprocessing contracts cover less than 25% of this inventory, and current planning assumes that all remaining SNF will be directly disposed of. Vitrified waste and LILW from reprocessing abroad will be returned to Switzerland for disposal. It is planned to store the SNF for approximately 40 years to reduce the thermal loading of the repository, so that ample time is available for project development. A centralized facility for dry cask storage of SNF and vitrified HLW and for other reprocessing wastes was constructed by the ZWILAG (Zwischenlager Würenlingen AG) organization, a daughter company of the utilities, and started operation in 2001.

HLW cooling (especially high-burnup UO_2 and MOX SNF if directly disposed) suggests an operational date for a SNF/HLW repository around 2050. Nevertheless, there is a requirement to demonstrate the fundamental feasibility of siting a repository in Switzerland and develop a sound scientific basis for future implementation. For that purpose, the "Project Opalinus Clay" was submitted to the Federal Government in late 2002 (see Section 23.3).

Site selection for SNF/HLW is very much constrained by the small size of Switzerland and by its geological setting. The current geological consensus is that uplift of the Swiss Alps may still continue (~1–2 mm/year). Excluding alpine areas and other complex geological structures associated with the Jura Mountains and the Rhine Graben leaves only limited areas in Central and Northern Switzerland that would be potentially suitable. Within this area, three host-rock variants are considered —either the crystalline basement or one of the two overlying, low-permeability sediment layers.

The current conceptual repository design (see Figure 23.1) has the following features:

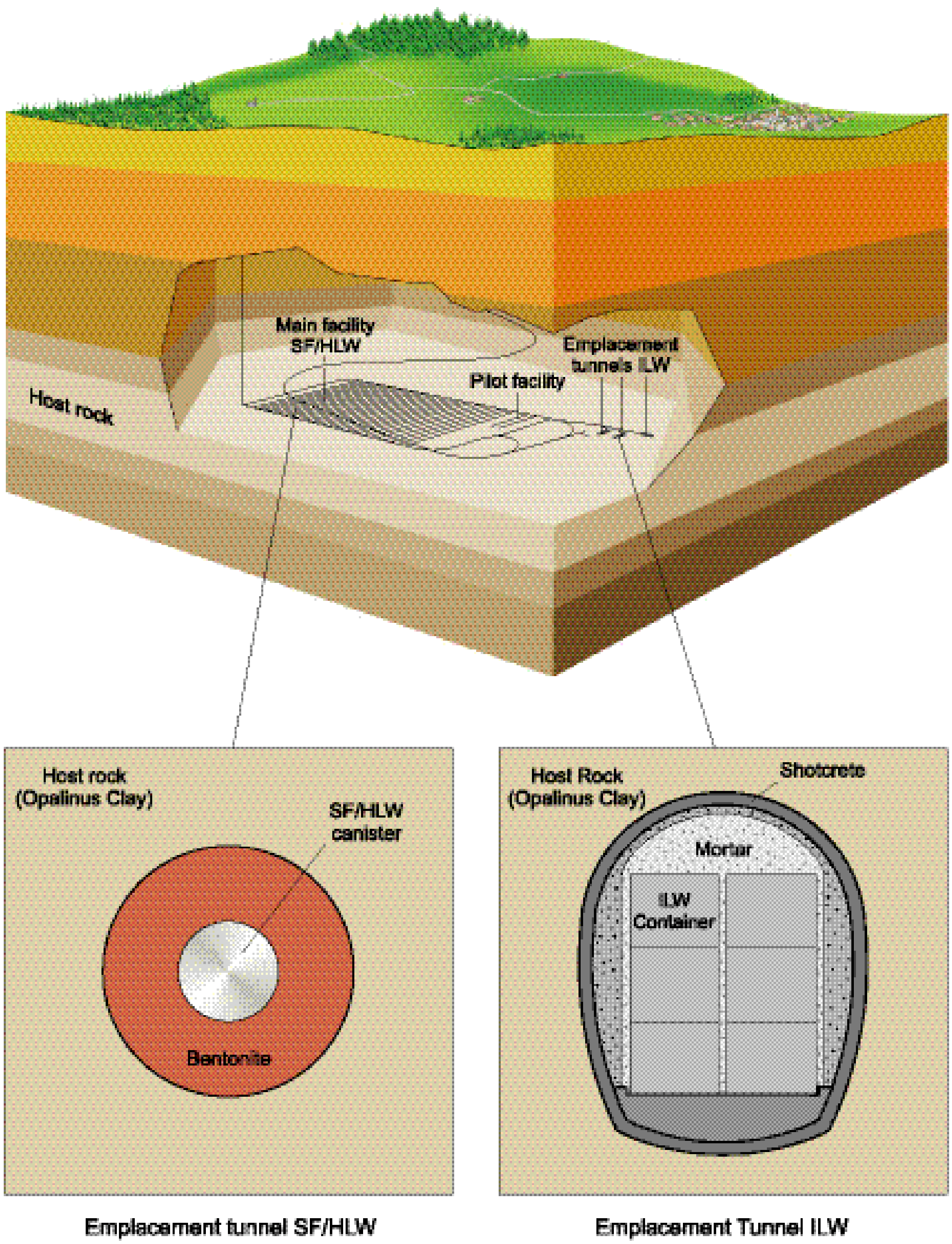

Figure 23.1 Possible layout for a deep geological repository for SNF/HLW/ILW in Opalinus Clay

1. Deep disposal (about 500 m to 1 km below surface) in a specially constructed facility.
2. In-tunnel emplacement of SNF/HLW waste packages in a geologic medium (sediment or crystalline basement), which physically protects the engineered barrier system (EBS), has low water flows and favorable groundwater chemistry, and acts as an efficient radionuclide transport barrier.
3. Engineered barriers: In addition to the vitrified waste in its canister or spent UO_2/MOX fuel within its cladding, a thick overpack is envisaged, surrounded by compacted bentonite clay.
4. Co-disposal of LL-ILW in tunnels located in a separate part of the repository.

Analysis of this basic concept in the Project Gewähr 1985 study (Nagra, 1985) showed that, for all realistic scenarios analyzed, the performance guideline was met with large margins of safety. In their review of this project, the government concluded that this concept would provide sufficient safety in a crystalline basement rock with the properties of that postulated by Nagra. However, only limited data from isolated boreholes were available in 1985, and the Swiss government authorities requested more evidence that suitable rock formations of an appropriate extent could be identified in Switzerland. The government also required that the option of disposal in sedimentary formations be considered in more depth.

Since 1985, the regional investigation of the crystalline basement has been completed and documented (Nagra, 1994a; 1994b). Geological studies have clearly shown that the extent of accessible crystalline basement is much less than originally thought because of the presence of a previously unknown, extensive Permocarboniferous trough that cuts through the region. Only two restricted areas remain for selection of a possible site, each covering about 50 km^2.

23.3. PROJECT OPALINUS CLAY

In addition to the studies of the crystalline basement, investigations of the sedimentary options have proceeded from a desk study, to selection of potential host formations, to identification of specific potential siting areas. The two sedimentary host rocks investigated in detail were Opalinus Clay, which exists in a laterally extensive layer in Northern Switzerland, and Lower Freshwater Molasse, which is very extensive but rather heterogeneous (Nagra, 1994c). For Opalinus Clay, which was identified as the priority option by Nagra in consensus with the regulatory authorities and their experts, a field program in the potential siting area of the "Zürcher Weinland" has been completed.

The regional characterization program included regional hydrogeological measurements (isotope hydrology, head measurements), microseismics, neotectonics (geomorphology, precision-leveling geographic positioning systems, etc.). The characterization also included the successful performance of a novel—in the area of waste disposal—technique, namely close-mesh 3-D seismics. The technique was chosen because, based on existing seismic data and supported by previous deep boreholes, this host rock was expected to form a fairly simple, almost flat layer in the Zürcher Weinland exploration area, having good velocity contrast with neighboring formations (Birkhäuser et al., 2001).

In addition, the Benken deep borehole provided key structural, hydrogeological, rock-mechanical, lithological, and geochemical properties of the Opalinus Clay and surrounding formations. The borehole database was complemented by off-site studies at the Mont Terri test site (see below) and observations in relevant boreholes and tunnels intersecting these formations. This raw database was integrated and analyzed within a "geosynthesis" to provide the engineering and performance datasets required for the Entsorgungsnachweis project (Nagra, 2002a).

The particular characteristics of this host rock that influenced the resultant repository design and layout (Figure 23.1) are:

- Planar geometry of the ~100 m thick Opalinus Clay: this feature, together with the regional stress field, dictated the layout of the waste emplacement tunnels.
- Very low hydraulic conductivity of this formation.
- Increasing strength of the rock as it dries due to ventilation, which means that ~2.5 m diameter tunnels drilled at a depth of 600–800 m below the surface are self-supporting. No lining is required: steel mesh and rock bolts are sufficient to ensure operational safety.

The very low hydraulic conductivity of the Opalinus Clay—and also of surrounding sediments, which together with the Opalinus Clay form a nearly 300 m thick low-permeability layer—plays a major role in the safety concept developed (Figure 23.2; see also Nagra, 2002b). Despite the head differences between underlying and overlying aquifers (giving gradients ~1), there is good geochemical and isotopic evidence that solute transport within the Opalinus Clay occurs predominantly by diffusion (Figure 23.3). This means that the Opalinus Clay is an extremely powerful geological barrier. The latter is also illustrated by the fact that calculated dose maxima for expected repository evolution scenarios are several orders of magnitude below the regulatory guideline of 0.1 mSv a^{-1} and typically occur after about 1 million years. An analysis of the results

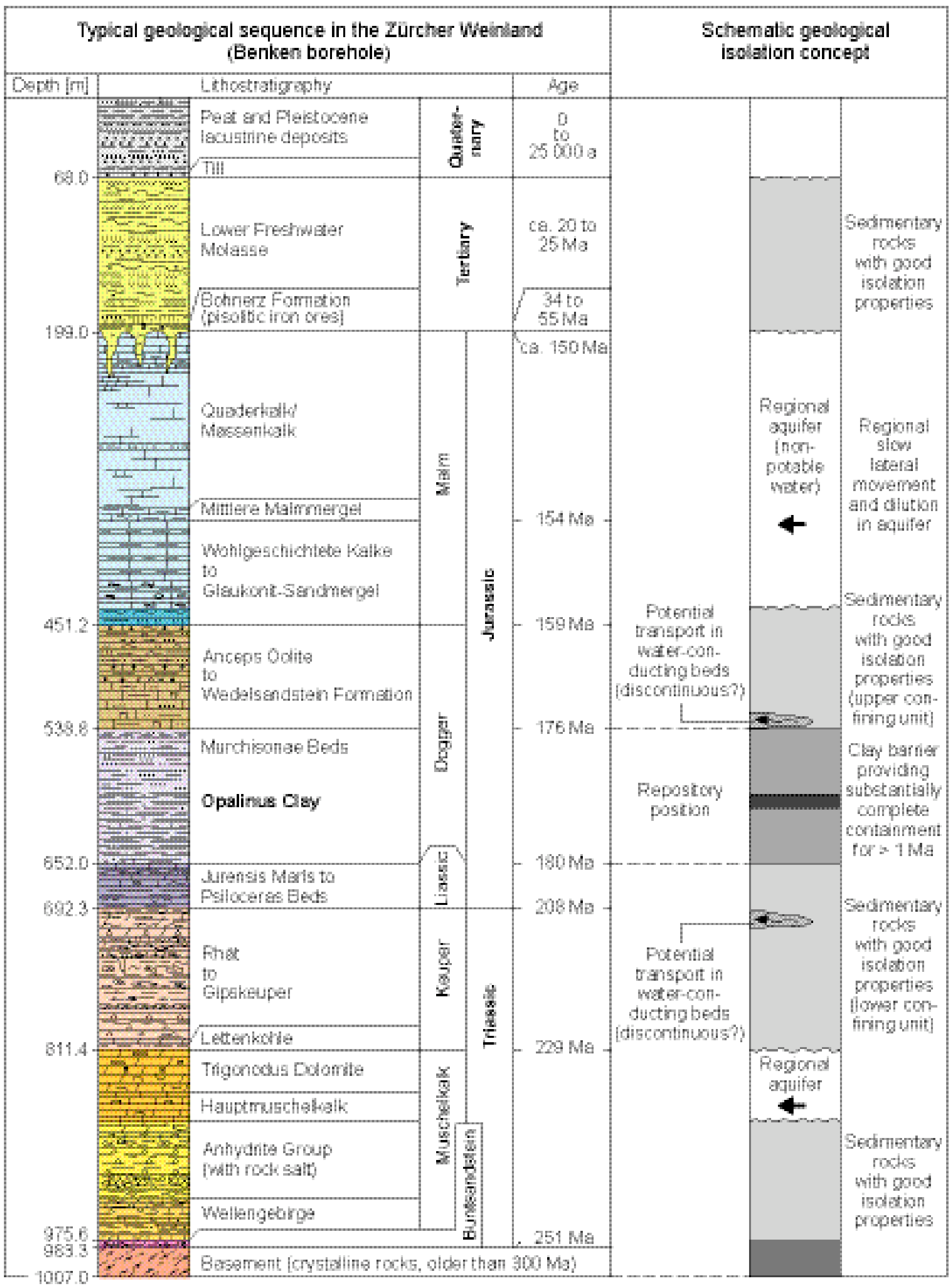

Figure 23.2 The geological sequence in the Benken borehole (left) and the simplified features illustrating the isolation concept (right)

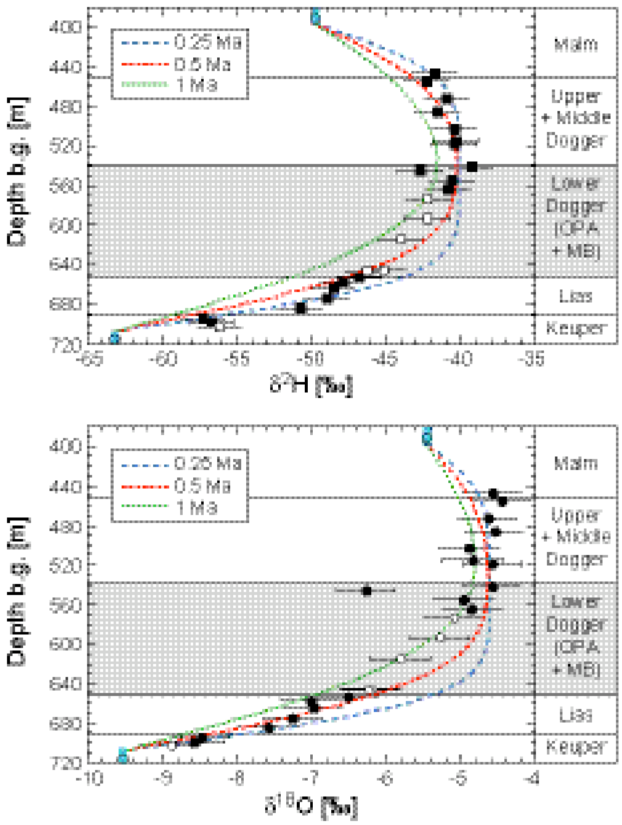

Figure 23.3 Isotope concentration profiles in pore water across the Opalinus Clay and adjacent rock strata due to diffusion that occurred at 0.25, 0.5, and 1 Ma

shows that the calculated doses are caused by a limited number of radionuclides that are both very long-lived and highly mobile (e.g., ^{129}I).

The extremely effective barrier does, however, give rise to situations that need special attention. One of the most relevant of these involves the buildup of gas pressure within the repository near field, caused predominantly by H_2 produced by anaerobic corrosion of steel. Such pressures could potentially drive contaminated water flows (e.g., along the excavation disturbed zone [EDZ] of repository tunnels). Gas flows containing gaseous radionuclide (e.g., ^{14}C methane) have also been analyzed. All these aspects required a focused analysis of gas production and release in the complex, hydromechanical environment of the Opalinus Clay (Figure 23.4). Nevertheless, a strong case can be made that even conservative gas scenarios do not compromise the safety of the repository.

The review process for the feasibility study Entsorgungsnachweis by the authorities was completed in September 2005, and resulted in a positive judgment and recommendation to the government for acceptance of the demonstration project. The review process culminated with the official expertise by the Swiss Federal Nuclear Safety Inspectorate (HSK, 2005), the evaluations carried out by the Commission for the Safety of Nuclear Installations (KSA, 2005) and the Commission on Nuclear Waste Management (KNE, 2005), and an international review by an NEA expert group commissioned by the Federal Office of Energy (OECD/NEA, 2004).

During the review process in 2004, the Minister and Head of the Federal Department of Environment, Energy, Transport, and Communications requested an updated review of options (rock formations and areas) that might be considered as alternatives to the Zürcher Weinland for a geological repository for SNF/HLW/ILW. The report submitted by Nagra in September 2005 (Nagra, 2005) summarizes the current state of general academic and applied geoscientific research, as well as the specific knowledge base developed by Nagra over the past 30 years. This assessment, carried out from the point of view of long-term safety, led to the identification of a preferred geologic-tectonic region roughly located in the north–northwest of Switzerland. Within this region, different host rocks and areas could, in principle, ensure the safety of a deep geological repository, provided engineered barriers were adapted to the geological environment. However, the report pointed out that, from a geological standpoint, Opalinus Clay had distinct advantages over other potential host rocks (e.g., crystalline basement, claystones of the Lower Freshwater Molasse). Besides the Zürcher Weinland, three areas with Opalinus Clay formations are considered to be suitable for the siting of a geological repository for SNF/HLW/ILW. The site selection process will follow the procedure defined in the recently drafted Sectoral Plan for Geological Repositories (see Section 1), which will take into consideration not only geological criteria but also socioeconomic and spatial planning issues.

The 3-month open public consultation process for the Entsorgungsnachweis evaluation was initiated in September 2005. It is expected that the Federal Government will reach a conclusion, additionally considering input from the consultation process, in the second half of 2006.

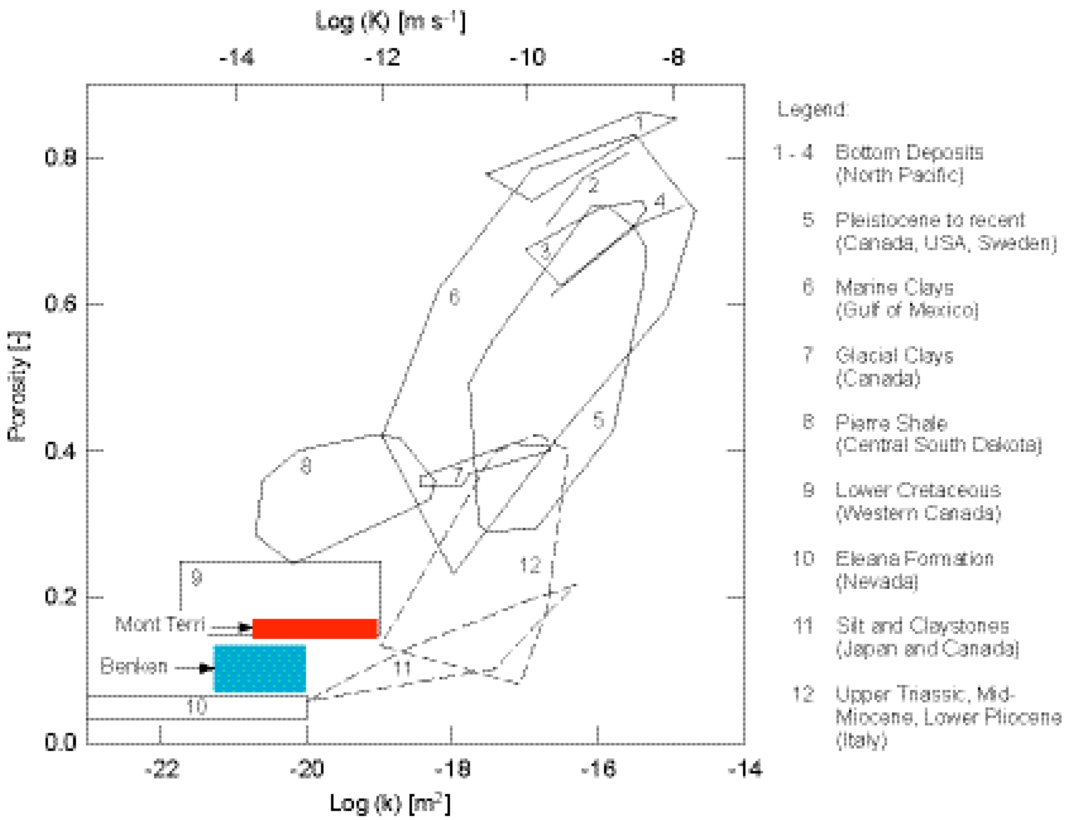

Figure 23.4. Phenomenological description of gas transport processes in argillaceous rock, along with the relationship of transport mechanisms to geomechanics and the barrier function of the rock

23.4. GEOLOGICAL STUDIES AT GRIMSEL AND MONT TERRI

Despite having a relatively small program, Switzerland is in the fortunate situation of having two major underground test sites—Grimsel in crystalline rock and Mont Terri in Opalinus Clay.

Nagra's Grimsel Test Site (GTS) is situated below ~500 m of overburden in granite/granodiorite of the Swiss Alps. Over a period of 20 years, this site has become an international center for *in situ* research supporting nuclear waste management. Earlier phases of work concentrated more on development and testing of methodology to characterize the subterranean environment. More recently, emphasis has shifted towards large-scale demonstration of technology for waste emplacement, testing models of evolution for the engineered barrier system and the immediately surrounding rock, development and testing of monitoring technology, and long-term studies of processes influencing radionuclide migration in natural or perturbed (high pH plume, colloids) fracture flow systems. Figure 23.5 shows preliminary results from the HPF (High-pH Flow) experiment (Maeder et al., 2005), where fluid with a pH=13 has been circulating in a natural fracture through a borehole-dipole arrangement for a period of almost 3 years. Tests of radionuclide migration in this fracture were completed in 2005 and are currently being analyzed.

A novel aspect of the current "Phase VI" of the Grimsel Test Site (GTS) program is that it has a nominal duration of 10 years, but experiments can be planned in a modular form considering time scales that are compatible milestones of repository programs, i.e., when particular input is required. This is especially important for the long-term (multidecade) databases, which may be needed to validate key models for performance assessment or, more qualitatively, provide arguments to support the safety case.

Also, in work directly associated with repository implementation, studies are planned of remote (tele) handling

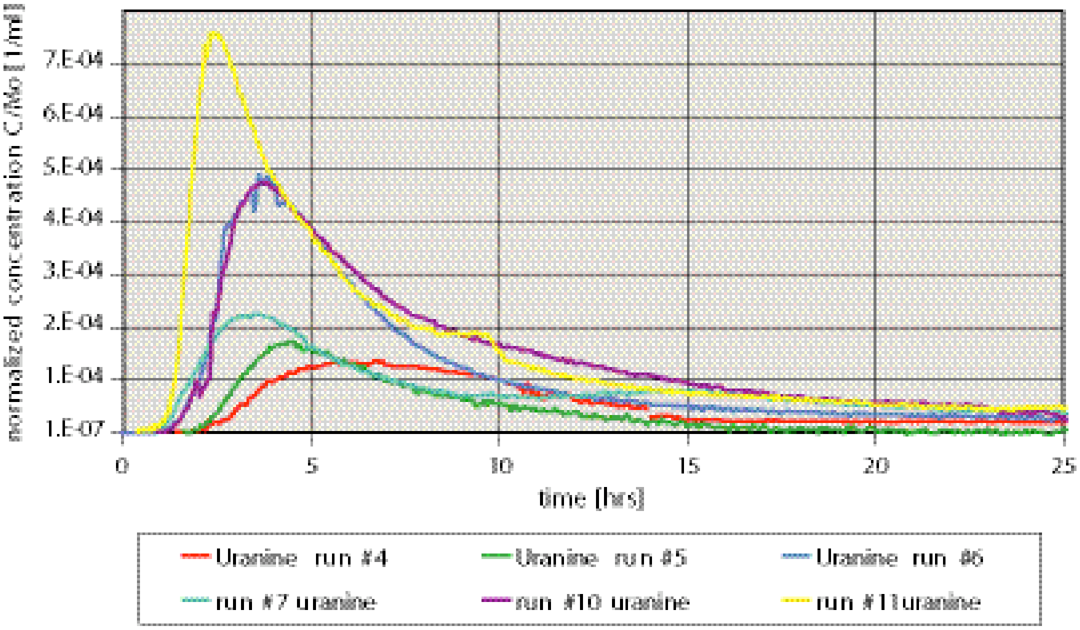

Figure 23.5. HPF Experiment at Grimsel. Top: breakthrough curves for Na-florescein (uranine) measured in six dipole tracer tests spaced 3-4 months apart spanning 21 months of hydraulic evolution of a shear zone at GTS (Maeder et al, 2005). The flow field becomes progressively more focused (Run #4 to Run #11) as the high-pH fluid circulating in the fracture interacts with the fracture infill and the matrix. Bottom: excavation of the resin-impregnated shear zone to study the alterations caused by the high pH circulation and their effects on radionuclide migration. The core diameter is 280 mm (Photo Comet).

of transportation and emplacement of waste packages and associated engineered barriers. Opportunities exist for other interested organizations to join GTS projects; further information is available at www.grimsel.com.

The second test site, at Mont Terri in the Jura Mountains, also provides horizontal access to the Opalinus Clay under an overburden of about 300 m. The Mont Terri project, initiated in 1995, is managed by the Swiss Federal Office of Water and Geology and (in addition to Nagra) involves numerous partner organizations from six countries. At this site, continuing projects characterize the thermal, hydraulic, mechanical, and geochemical properties of this rock. (Owing to the rock's high content of swelling clay minerals, such properties tend to be inherently coupled.) Work has, however, also commenced on studying solute migration in this rock (constrained by negligible advective flow in undisturbed rock) and engineered barrier system emplacement/evolution (e.g., Mayor et al, 2006, Figure 23.6). Again, other organizations could join future phases of work at this site (see www.mont-terri.ch).

23.5. THE SWISS PROGRAM IN AN INTERNATIONAL CONTEXT

The Swiss waste management program, while relatively small in terms of budget and manpower, is wide in scope—in terms of both waste type and host rock considered. This program is only feasible if maximum advantage is taken of work performed elsewhere. Therefore, extensive use is made of international collaboration and information-exchange agreements with other national programs, to allow effort to focus on specific key areas.

Apart from active participation in the International Atomic Energy Agency (IAEA) and the Nuclear Energy Agency (NEA), Nagra has formal agreements with the European Economic Community (EEC) and numerous organizations in various countries (Belgium, Canada, the Czech Republic, Finland, France, Germany, Japan, South Korea, Spain, Sweden, Taiwan, the USA, and the UK). Informal collaborations extend the list further.

Since 1997, Nagra has also expanded the provision of

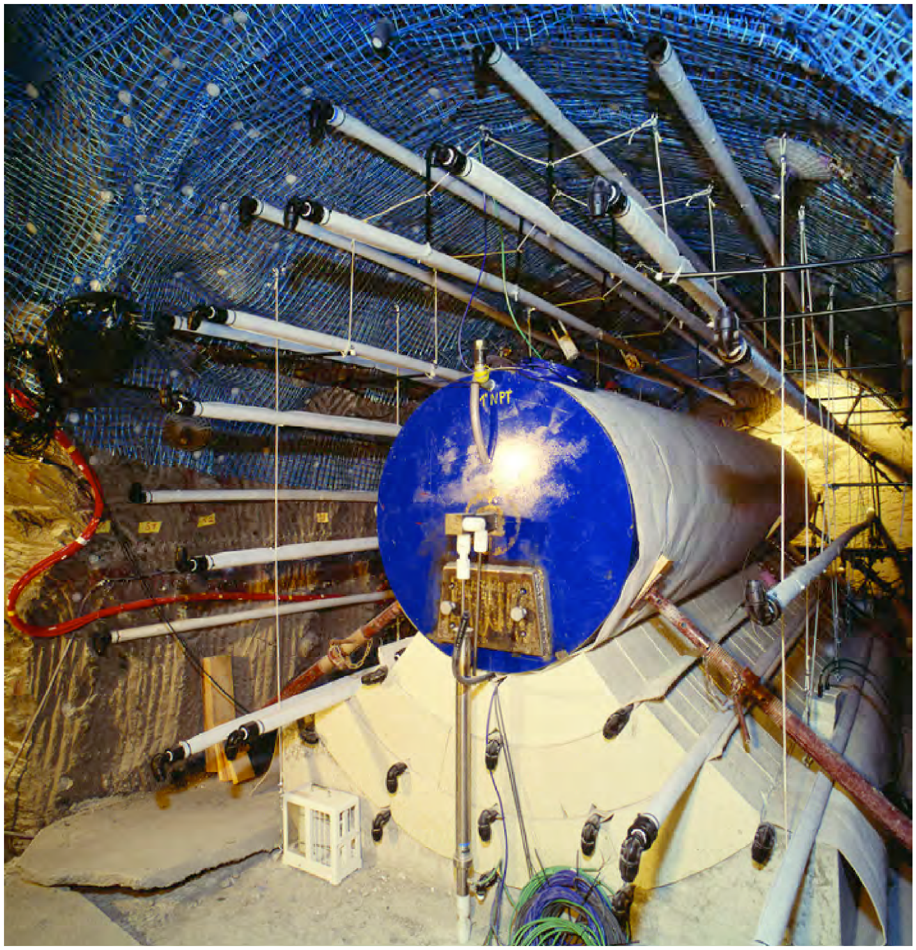

Figure 23.6. Photo from the engineered barrier experiment at Mont Terri. Note the prefebricated bentonite blocks that form the canister pedestal, the mock canister, and the saturation system for the buffer (bentonite pellets) to be emplaced (Photo Comet).

technical support services to other countries (organizations) and to applications outside the nuclear waste management field. This development has clear advantages for both Switzerland and those other countries—the experience accumulated at considerable investment within the Swiss national program can be made available for other purposes, while Nagra staff have the opportunity to further widen their experience, which provides a perspective essential for such a small program.

23.6. CONCLUSIONS

The Swiss program is, after 30 years, relatively mature. It is probably fair to say that most of the fundamental technical "geological problems" associated with nuclear waste disposal in Switzerland have now been solved. Key milestones in the near future will, however, provide challenges, particularly in communication with the general public. As we move towards implementation, demonstration and validation play key roles in gaining and maintaining acceptance; here, our underground test sites will play a key role. The major next step in Nagra's program, namely site selection, is expected to commence at the end of 2006/beginning of 2007, after the legislative and political framework for radioactive waste disposal currently under discussion is finalized.

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Chapter 24

The Long-Term Management of the United Kingdom's Radioactive Waste: Status in 2005

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24.1. INTRODUCTION

The United Kingdom (UK) has considerable holdings of long-lived radioactive waste that will remain potentially hazardous for many thousands of years. Previous attempts to develop a disposal route for some of these wastes have ended in failure. The most recent attempt ended with a refusal from the Secretary of State for the Environment to allow the construction of an underground Rock Characterization Facility (RCF) at a location close to the BNFL Sellafield Works in West Cumbria (northwest England).

The waste still exists of course. And if we are to avoid passing the legacy of waste onto future generations, the need for a long-term solution remains. In 2001 the Government initiated a six-year long consultation on long-term radioactive waste management. This is intended to lead to a new policy that will win widespread public support. The consultation encompasses a very wide range of options, so that it is no longer a foregone conclusion that the UK solution to this problem will involve geological disposal.

During this consultation period, there have already been some far-reaching changes to the structure of the UK nuclear industry and to Nirex's role. This report provides a summary of these developments; it very much presents the Nirex perspective on the issues.

24.2. BACKGROUND

For the past quarter of a century, attempts to implement Government policy on the disposal of long-lived radioactive waste have ended in repeated failure. The key events may be summarized as follows:

- 1976—The Royal Commission on Environmental Pollution (Flower's Report) recommended the creation of a National Waste Disposal Corporation.
- 1979—Start of program of geological investigations for HLW disposal.
- 1981—Termination of the geological investigations and suspension of a decision on high-level waste disposal for 50 years.
- 1982—Nuclear Industry Radioactive Waste Executive (NIREX) created to implement Government policy on intermediate-level waste (ILW) and low-level waste (LLW).
- 1983—Sea disposal of ILW "suspended" through a moratorium agreed at the London Dumping Convention; UK Government formally renounces sea dumping in 1988.
- 1983—A near-surface site for LLW and short-lived ILW, and a deep site for long-lived ILW, were announced; subsequently,
  - the deep site was abandoned by the end of the year; and
  - three additional near-surface sites were to be sought.
- 1985—NIREX is reconstituted as United Kingdom Nirex Limited (known as Nirex).
- 1986—The three additional near-surface sites are announced—near-surface disposal subsequently limited to LLW only.
- 1987—Abandonment of the near-surface program

and adoption of new policy that all ILW and LLW should go deep (subsequently decided that LLW allocated for deep disposal would be limited to LLW that was unsuitable for disposal at Drigg-BNFL's near-surface site for LLW in Cumbria); new deep site selection process started.
- 1989—Following the site selection process, Nirex decides Sellafield and Dounreay would be investigated first.
- 1991—Nirex decides to focus investigations on Sellafield in Cumbria.
- 1992—Nirex announces plans for a Rock Characterisation Facility (RCF) at Sellafield; the plans were eventually considered at a public inquiry which ended in 1996.
- 1997—Decision by Government not to allow Nirex to proceed with the RCF, thus terminating the UK's siting program.

This timeline should be viewed against Government policy decisions on radioactive waste management that started as early as 1956 and resulted in several reports by various Government Committees and associated White Papers. A summary can be found in POST (1997).

Following the 1997 decision, Nirex reviewed what went wrong, a process that culminated in a shift from its being an entirely scientific organization to one that also encompasses social considerations. At the same time, changes have been made to make it more accountable and transparent (Wild and Mathieson, 2004; Dalton and McCall, 2003). Other organizations have also examined the implications of the 1997 decision. These investigations include:

- An inquiry and subsequent report by the House of Lords Select Committee on Science and Technology (House of Lords, 1999)
- A consensus conference held by the UK Centre for Economic and Environmental Development (UKCEED) involving a citizens' panel (UKCEED, 1999)

The Government response was to set up the *Managing Radioactive Waste Safely* consultation under the Department of the Environment, Food and Rural Affairs (Defra) and the Devolved Administrations of Scotland, Wales and Northern Ireland (Defra, 2001). Subsequently, a new independent committee was established in November 2003—the Committee on Radioactive Waste Management (CoRWM)—to oversee the second phase of the consultation exercise (see later section). CoRWM is expected to recommend to Government its preferred long-term waste management option or options by July 2006 (CoRWM, 2005).

24.3. RECENT DEVELOPMENTS

24.3.1. Nirex and Its Independence

The Independence Debate

Nirex was originally set up by the nuclear industry with the support of Government to find a disposal solution to the UK's LLW and ILW. Following its inception in 1982, it was owned and financed by the nuclear industry: at that time British Nuclear Fuels Limited (BNFL), Central Electricity Generating Board (CEGB) and United Kingdom Atomic Energy Authority (UKAEA). These organizations (or their successors) and the Ministry of Defence have mostly financed Nirex's activities. The Government, through the Department of Trade and Industry (DTI), owned a special share designed to guarantee that Nirex remained free of commercial pressure from the industry; this intention did not translate into the view held by the public, however. To them, Nirex was compromised by short-term commercial considerations and was perceived as acting more on behalf of the waste producers than on behalf of society as a whole and the environment.

Analysis of the events of 1997 led certain nongovernmental organizations (NGOs), the "greens" and Nirex itself, to argue that Nirex should be made independent of the nuclear industry, including the newly formed Nuclear Decommissioning Authority (NDA—see later section). The argument was that an independent Nirex would be seen as more legitimate by the general public and other stakeholders and, further, that it would be more capable of making legitimate contributions to policy development and, ultimately, implementation. The Government supported this view and announced in July 2003 that Nirex would become "independent of industry" (Defra, 2003). Following consultation with shareholders, the Government further announced in July 2004 that its shares would be transferred to a Company Limited by Guarantee (CLG). This was to be established by the DTI and Defra and would, in turn, own the shares in United Kingdom Nirex Limited. A CLG is a type of incorporation used for not-for-profit companies. Financing would, for the most part, be through a contract with the NDA, but Nirex would remain separate from, and independent of, the NDA itself.

These arrangements came into operation on April 1, 2005. Nirex believes they are a necessary and important step towards greater transparency and accountability. Part of the remit of the board of the CLG is to ensure there is no overlap or conflict of interest between Nirex's work and that of the NDA or CoRWM. These arrangements will hold until CoRWM makes its recommendations to Government in July 2006, when future radioactive waste management policy will be decided, together with the future of Nirex.

A New Mission

As a result of these changes, the Nirex mission has been modified to read:

"In support of Government Policy, develop and advise on safe, environmentally sound, and publicly acceptable options for the long-term management of radioactive materials in the UK."

The statement refers to "options" because no choice of strategy has yet been made, and it refers to "radioactive materials" rather than "radioactive waste." This latter point refers to the fact that the UK has radioactive materials that in the future may be declared as waste. These include spent nuclear fuel (SNF), which is currently regarded as a resource in view of the nation's reprocessing policy, and stockpiles of separated plutonium and uranium associated with the strategic deterrent and the now-abandoned fast reactor program. It also brings consideration of the long-term management of vitrified high-level waste, produced from the reprocessing of SNF, into Nirex's remit.

Thus, Nirex is now engaged in examining options for more than just its historical remit of intermediate- and some low-level waste. To this end, Nirex is actively undertaking research to establish long-term management concepts for these additional materials, in collaboration with other national waste management organizations.

Nirex continues to be responsible (together with Defra) for the UK inventory of radioactive waste. The following table (Nirex, 2003a) summarizes the latest waste inventory data and the radioactive materials that may be declared as waste (as of April 1, 2001).

Transparency

One of the criticisms of Nirex before 1997 was that it lacked transparency, and the post-1997 review suggest-

Table 24.1. Reference repository concept volumes (packaged). The table excludes LLW, very-LLW and contaminated land, which are the responsibility of the NDA.

Waste category/source	Volume m^3
Other materials	
Uranium	75,000
Plutonium	4,000
Advanced Gas Cooled Reactor (AGR) fuel components	1,000
Highly Enriched Uranium	1,000
Submarine Spent Fuel (estimate)	100
Recovered Sealed Sources	1
HLW/ Spent Fuel	
AGR Spent Fuel	4,500
Vitrified HLW	1,600
PWR Spent Fuel	1,300
ILW/LLW (including Stage 3 Reactor Decommissioning Wastes)	
ILW	350,000
LLW unsuitable for Drigg	35,000

ed this was one of the reasons for the failure of the RCF proposals. To address this, in 1998 Nirex established a comprehensive Transparency Policy, overseen by an independent Transparency Panel (Nirex, 2005a). The panel's membership of five is external to Nirex. It is chaired by Andrew Puddephatt, Executive Director of Article 19, an international human rights group promoting freedom of expression and access to information. Until recently, the panel was confined to considering Nirex's response to requests for information, but its role is now extended to cover all aspects of the Transparency Policy, including how decisions are made and how Nirex operates internally.

Nirex has been continually challenged over the lack of transparency in not releasing the names of sites considered in its 1987–1991 site selection process for an ILW/LLW repository, other than Dounreay and Sellafield. These had not been published, in line with Government policy to keep the information confidential to prevent blight affecting any of the areas that had been considered as having possible sites. There have been several requests for the lists of the sites over the years that have been refused in line with Government policy. On January 1, 2005, a Freedom of Information Act came into force: along with other legislation (the Environmental Information Regulations 2004); it puts greater emphasis on openness, transparency, and the publication of information. In light of these developments, the Government changed its previous policy and, in conjunction with Nirex, decided that the process of site selection and the names of all of the sites considered should be published. This was done on 10 June 2005 (Nirex, 2005).

24.3.2. Managing Radioactive Waste Safely (MRWS) Consultation

The Government consultation, "Managing Radioactive Waste Safely: Proposals for Developing a Policy for Managing Solid Radioactive Waste in the UK" (MRWS), started in September 2001 (Defra, 2001) and continues. The consultation document anticipates five stages, expected to run, as a "rough guide" from 2001 to 2007 and ending with "legislation, if needed." It was stressed, however, that it was more important to "get the decisions right, and ensure that the strategy wins public confidence" than to adhere to any predetermined timetable. The first stage, now complete, was initiated by the publication of MRWS and included consideration of the responses to MRWS and planning the next stage.

Committee on Radioactive Waste Management (CoRWM)

CoRWM was established as an independent body to oversee the second stage of the Government's MRWS consultation. Its remit was to consider a wide range of options for long-term radioactive waste management and, after consulting widely, to make recommendations to Government by July 2006.

The committee currently has eleven members drawn from a range of disciplines and interests; it is chaired by Professor Gordon MacKerron (CoRWM, 2005). As a result of the creation of CoRWM, the Radioactive Waste Management Advisory Committee (RWMAC) has been put into abeyance.

CoRWM's main focus is the UK's HLW and ILW. It began by drawing up a "long list" of options. Through a process of consultation, the Committee has produced a provisional short list of options, that when finalized will be subject to more detailed evaluation, comprising:

- Long-term interim storage
- Near-surface disposal of short-lived wastes
- Deep geological disposal
- Phased deep geological disposal

In seeking to refine the list of options, the criteria used by CoRWM to screen out options were:

1. There is no proof of concept in the form of
   - Actual implementation of the options in the UK or elsewhere.
   - Sufficient research and development of the international criteria to demonstrate that the option can be implemented.
2. It causes us to breach our duty of care to the environment outside national boundaries.
3. It causes harm to areas of particular environmental sensitivity.
4. It places an unacceptable burden (in terms of cost, effort, or environmental damage) on future generations.
5. It involves a risk to future generations greater than that to the present generation which has enjoyed the benefits.
6. It results in unacceptable risk to the security of nuclear materials.
7. It poses unacceptable risk to human health.

8. Cost is disproportionate to the benefits achieved.
9. It breaches internationally recognized treaties or laws, and there is no foreseeable likelihood of change in the future.

Of most interest to Nirex is Nirex's own preferred option of a phased geological repository concept, covered by phased deep disposal in the provisional shortlist. In the post-1997 review, it became clear from discussions with stakeholders that, in developing its deep disposal concept, Nirex had paid insufficient attention to monitoring and retrievability. This signalled the need for changes to the concept. As part of the work, three workshops were held at which stakeholders were able to contribute to the redesign process. The Nirex Phased Geological Repository Concept (Nirex, 2003b) is the result. This is based on the previous Nirex concept, but has the capability of not backfilling the waste-filled caverns until a future society might decide to do so. This period of "storage with a view to disposal" could be as long as 300 years, during which waste could be monitored and, if required, retrieved.

Beyond Stage 2

Following the short-listing phase, and before making any recommendations to Government, CoRWM will aim to refine its list through further stakeholder engagement. The submission of the CoRWM recommendations will mark the end of the second stage of the MRWS process and the start of the third phase in which Government will consult on the implementation of the recommendations. This is scheduled to last for about a year prior to the actual implementation itself sometime in 2008.

House of Lords Review

Following on from its earlier work (House of Lords, 1999), the House of Lords' Select Committee on Science and Technology again considered the question of radioactive waste management and, specifically, the work of CoRWM in 2004. The committee took evidence from, among others, the Government Minister responsible and Professor MacKerron. In its report (House of Lords, 2004) the Select Committee was critical of the Government, accusing it of procrastination on policy development, especially between 1997 and the establishment of CoRWM in 2003. CoRWM was commended for its openness and transparency, but also criticized on a number of points, including spending too much time talking about decision-making methodologies at the expense of identifying the right solution.

24.3.3. Nuclear Decommissioning Authority

The Nuclear Decommissioning Authority (NDA) was formally established on April 1, 2005. It has "a specific remit to ensure the [civil] nuclear legacy is cleaned up safely, securely, cost effectively and in ways which protect the environment for the benefit of current and future generations" (DTI, 2002). The NDA is a nondepartmental public body, which gives it the "management freedom and flexibilities it needs to deliver results whilst ensuring that there is a clear line of public accountability and direct Ministerial oversight" (DTI, 2002).

With the creation of the NDA, existing nuclear site licensees such as BNFL and UKAEA have become contractors to the NDA. To make the best possible use of the available skills, the management of cleanup is to be opened up to competition with a greater emphasis on competitive procurement of decommissioning and support services.

Openness and transparency are seen as fundamental to the successful operation of the NDA and its contractors. This applies particularly to the interests of local stakeholders in decisions bearing on the cleanup of individual sites, so that major decisions will be taken only in the light of full consultation with stakeholders.

The NDA program has an estimated cost of £48 billion and is expected to take about 100 years, stretching 150 years in some cases. Nirex's work, in contrast, addresses much longer periods of time—hundreds of thousands of years. Nirex will continue to support the UK's decommissioning and nuclear cleanup through its active program of research into long-term waste management concepts, and its Letter of Compliance (LOC) system that provides industry with guidance on its ILW waste packaging proposals (Barlow and McCall, 2005).

24.3.4. Organizational Interactions

The Industry

The structural changes outlined above effectively divide the UK civil nuclear industry into three sectors:

1. Commercial—fuel fabrication, nuclear power production, plant construction and operation, and contracting to the NDA
2. Decommissioning and cleanup—the responsibility of the NDA, focused on a 100 yr timescale
3. Long-term management—currently the responsibility of Nirex and dependent on whatever long-term solution is chosen by Government

Clear separation between these areas allows potential conflicts between long- and short-term issues to be visible to stakeholders. Nirex believes this is important for achieving legitimacy in policy development. In particular, it will provide transparency of process for any trade-offs that have to be made between the needs of long- and short-term waste management.

The Regulators

The NDA now has overall responsibility for the conditioning, packaging, and storage of most of the UK's legacy waste, while the implementation of the work is done by its contractors at the various sites. The nuclear regulator, the Nuclear Installations Inspectorate (NII), requires waste packagers to submit new packaging proposals to the Nirex "Letter of Compliance" (LoC) process. Through this process, Nirex has the responsibility for ensuring that the packaged wastes meet the standards and specifications that would allow them to be emplaced in a deep Nirex repository for ILW, i.e., that the packaged wastes are consistent with the Phased Geological Repository Concept. This requires Nirex to undertake a detailed independent assessment of the proposal in relation to the suitability of the waste package for transport to a repository, operations at the repository, and postclosure safety (Barlow and McCall, 2005). Without a LoC, the NII would be unlikely to give its permission for the proposal to go ahead.

Since January 2004, this process has been formalized under the conditions attached to the nuclear site license. The arrangements also require Nirex's procedures to be scrutinized, and this is done by the Environment Agency and the Scottish Environment Protection Agency on behalf of NII through their Memorandum of Understanding.

24.4. CONCLUSIONS

Over more than a quarter of a century, successive policies for the long-term management of long-lived radioactive waste in the UK have suffered repeated failures. Government initiated a new consultation process in 2001. Seeking to learn from past mistakes, the consultation aims to lead to a new policy for long-term radioactive waste management that will meet the appropriate safety standards and win public confidence.

While there have been certain criticisms aimed at the slow progress of policy development and implementation, the UK is engaged in a process which seems to have resolved issues surrounding the structure of the nuclear industry and Nirex's role. Policy development is on target to be concluded in 2007, with implementation of that policy in 2008. Nirex believes that that policy should include the implementation of its own scheme for the long-term management of ILW: the phased geological repository concept.

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Chapter 25

Preparing to Submit a License Application for Yucca Mountain in the United States

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25.1. INTRODUCTION

In 1982, the U.S. Congress passed the Nuclear Waste Policy Act, a Federal law that established U.S. policy for the permanent disposal of spent nuclear fuel (SNF) and high-level radioactive waste (HLW). Congress amended the Act in 1987, directing the Department of Energy (DOE) to study only Yucca Mountain, Nevada, as the site for a permanent geologic repository. As the law mandated, the DOE evaluated Yucca Mountain to determine its suitability as the site for a permanent geologic repository. Analyses of decades of scientific studies demonstrated that Yucca Mountain would protect workers, the public, and the environment during the time that a repository would be operating and for tens of thousands of years after closure of the repository. A repository at this remote site would also: preserve the quality of the environment; allow the environmental cleanup of Cold War weapons facilities; provide the nation with additional protection from acts of terrorism; and support a sound energy policy. Throughout the scientific evaluation of Yucca Mountain, there has been no evidence to disqualify Yucca Mountain as a suitable site for the permanent disposal of SNF and HLW.

Upon completion of site characterization, the Secretary of Energy considered the results and concluded that a repository at Yucca Mountain would perform in a manner that protects public health and safety. The Secretary recommended the site to the President in February 2002; the President agreed and recommended to Congress that the site be approved. The Governor of Nevada submitted a notice of disapproval, and both houses of Congress acted to override the disapproval. In July 2002, the President's approval allowed the DOE to begin the process of submitting a license application for Yucca Mountain as the site for the nation's first repository for SNF and HLW.

Yucca Mountain is located on federal land in Nye County in southern Nevada, in the southwestern United States, approximately 100 miles (160 kilometers [km]) northwest of Las Vegas (Figure 25.1). The location is remote from population centers, and there are no permanent residents within approximately 14 miles (23 km) of the site. Overall, Nye County has a population density of about two persons per square mile (two persons per 2.5 square km); in the vicinity of Yucca Mountain, it is significantly less.

Yucca Mountain is a series of north–south-trending ridges extending approximately 25 miles (40 km) and consists of successive layers of volcanic tuffs, millions of years old, underlain by older carbonate rocks. The alternating layers of welded and nonwelded volcanic tuffs have differing hydrologic properties that significantly influence the manner in which water moves through the mountain. The proposed repository horizon will be in welded tuff located in the unsaturated zone, more than 1,000 ft (300 m) above the water table in the present-day climate, and is expected to remain well above the water table during wetter future climate con-

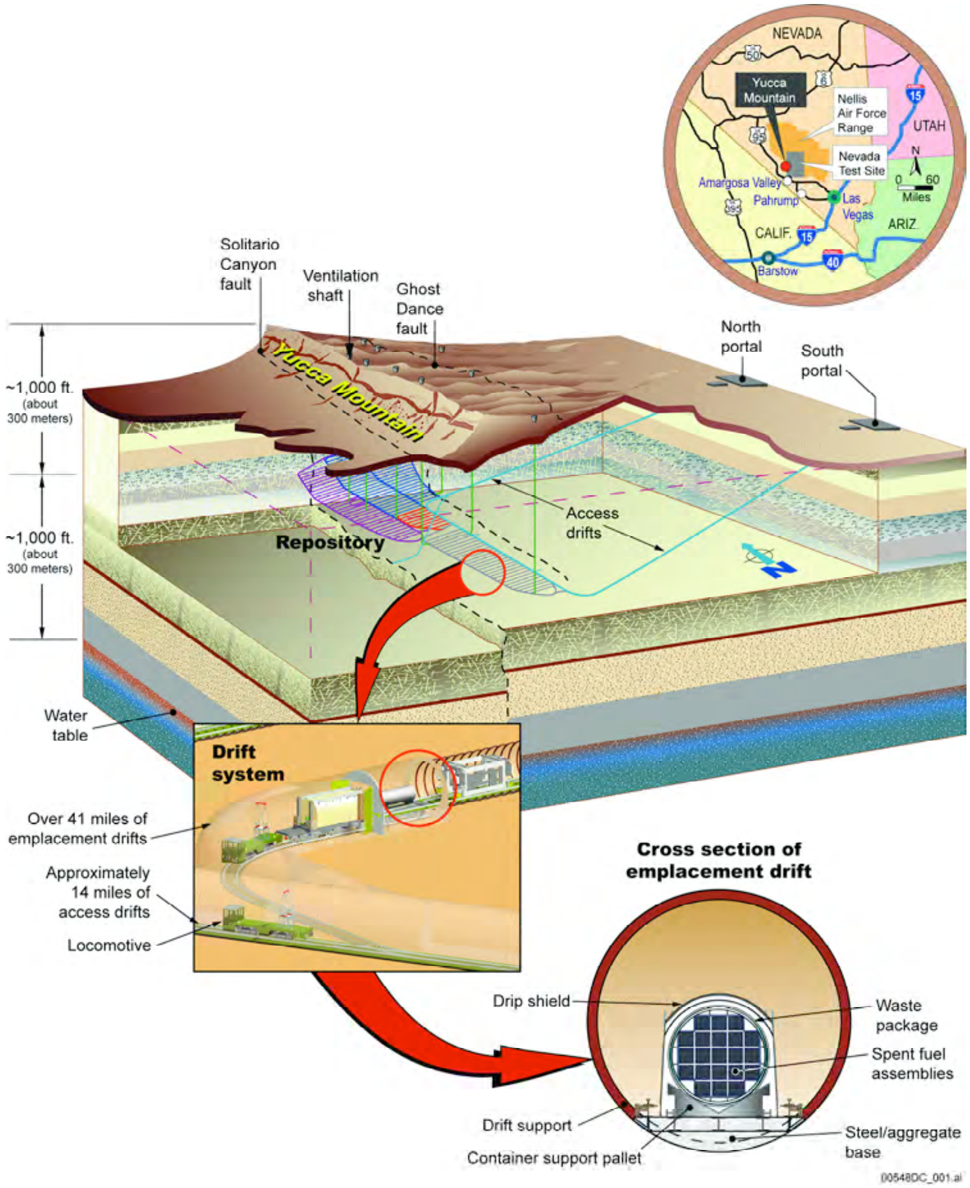

Figure 25.1. Location of Yucca Mountain and illustration of access to underground development, with details of emplacement drifts

ditions. Future meteorology and climatology at Yucca Mountain are important elements in understanding the potential amount of water available to interact with the waste.

25.2. SURFACE FACILITIES

The surface facilities support the operations necessary to receive, stage, age, package, and support emplacement of waste. The DOE has considered the use of on-site Waste Handling Facilities, namely, a Fuel Handling Facility, a Canister Handling Facility, and Dry Transfer Facilities, as well as process support facilities. The DOE is also considering the canisterization of fuel at reactor sites to simplify operations at the repository.

Repository packaging operations could be simplified significantly if, at the individual reactor sites, the fuel assemblies were placed in canisters that would permit transportation, aging, and disposal with the addition of suitable external components. This would be a departure from conceptual designs examined to date, wherein packaging operations primarily would be carried out at the repository site.

Operations would be performed remotely from shielded galleries within the Waste Handling Facilities. Operators would use closed-circuit television or direct observation through shielded windows to control manual and power manipulators, overhead bridge cranes, SNF transfer or handling machines, and other remote equipment to perform SNF and HLW handling functions. Thick, reinforced concrete walls would shield workers from exposure to radiation and radioactive material.

Sealed canisters containing SNF or HLW would be transferred from transportation casks to waste packages in a Canister Handling Facility for emplacement underground or to site-specific casks for aging or staging, if needed. The facility would be designed to handle only waste contained in sealed canisters, typically HLW canisters, DOE SNF canisters, naval SNF canisters, and canisters containing commercial SNF.

If at-reactor canisterization operations were not selected as the preferred operation mode, a Fuel Handling Facility likely would be the initial repository facility in operation to receive waste. SNF and HLW contained within the transportation casks would be transferred into waste packages for subsurface emplacement or transferred into site-specific casks for movement to an aging pad. The Fuel Handling Facility would be designed to receive a variety of waste forms, including uncanistered commercial SNF; canistered commercial SNF; canistered naval SNF; canistered HLW; and canistered DOE SNF.

The transfer of SNF and HLW from transportation casks to waste packages or to site-specific aging casks could also be accomplished at the repository in Dry Transfer Facilities. The Dry Transfer Facilities could also handle off-normal, damaged, and nonstandard fuel, damaged transportation casks, and damaged waste packages. Dry Transfer Facilities would provide space, layout, structures, and systems that support waste handling and remediation operations.

SNF will be aged if emplacement must be delayed because of thermal management or operational considerations, including blending to meet waste package thermal limits or other considerations. Loaded casks would be moved to concrete aging pads and later retrieved so that their contents could be transferred into waste packages. Alternatively, if the fuel assemblies were placed in containers at reactor sites, it could readily be aged at the repository site in these containers with appropriate external components added as needed. Space will be provided for initial and future aging capability.

25.3. SUBSURFACE FACILITIES

The repository subsurface components include the facilities necessary to transport and emplace waste packages. These subsurface facilities include excavated drifts, rail lines, emplacement pallets, engineered inverts, and support systems. The emplacement drifts will be large circular tunnels, nominally 18 ft (5 m) in diameter, used to provide emplacement for about 11,000 waste packages. The total subsurface emplacement area required to accommodate the waste packages containing 70,000 metric tons[1] will be about 1,250 acres (500 hectares). This area will include more than 40 miles (26 km) of emplacement drifts excavated by tunnel boring machines. Ground support will be installed behind the tunnel boring machines to provide structural support and worker protection. The emplacement drift inverts will consist of engineered steel structures and granular fill materials at the base of the emplacement drifts. The inverts will support the pallets, waste packages, drift rail system, and drip shields. The granular material will slow the movement of radionuclides into the host rock in the event that a waste package is breached.

The current design for the waste package consists of two concentric cylinders into which the waste forms will be placed. The inner cylinder will be made of stainless steel. The outer cylinder will be made of a corrosion-resistant nickel-based alloy. The closure end of the waste package has three lids that provide a leak-tight

[1]The Nuclear Waste Policy Act prohibits the emplacement of more than 70,000 metric tons until a second repository is licensed. Congress stopped work on the second repository program when it amended the Act in 1987, and directed the Secretary to report to the President and Congress on the need for a second repository.

closure. The basic waste package design is the same for all waste forms; however, the sizes and internal waste package configurations will vary to accommodate the different waste forms. Before the inner vessel is sealed, air and vapor will be evacuated, and helium added as a fill gas. If the fuel assemblies were placed in canisters at reactor sites, these canisters could likewise be placed into waste packages at the repository for disposal. A pallet in the emplacement drift will structurally support the waste package. Emplacement pallets will be fabricated from the same materials as the waste packages and will themselves be supported by the emplacement drift invert. After a decision to close the repository is made and approved by the Nuclear Regulatory Commission (NRC), titanium drip shields will be installed to protect waste packages from potentially dripping water and rockfall.

25.4. UNDERGROUND DEVELOPMENT

Facilities that have already been constructed during the site characterization phase are planned to be incorporated in the subsurface facilities. These existing facilities will be upgraded to support the construction of the repository or to become integral parts of the repository. The majority of the subsurface facilities will be excavated using tunnel-boring machines. Repository emplacement and development activities will occur concurrently; the separate areas will have separate ventilation systems. A pressure differential will ensure that any releases of airborne radioactivity will be confined to the emplacement side of the repository. The underground layout avoids areas of more fractured rock in the south and restricts emplacement to the north to avoid an area where groundwater levels rise at a steeper gradient.

The layout for the emplacement drifts is based on a phased approach to development that supports the start of waste emplacement after a period of initial construction. Concurrent construction and emplacement will continue for a period of about 24 years. The subsurface layout is designed to be constructed in a four-panel sequence (Figure 25.2) The initial emplacement panel will be located within the central section of the overall layout and uses the existing Exploratory Studies Facility and the north ramp for access to the repository horizon. The size of the panel is small in comparison to the other panels, and will be developed and commissioned as soon as possible after construction authorization. The general excavation process of a panel will begin with the excavation of the access and exhaust mains, followed by the excavation of the ventilation shafts, and finally the turnouts and emplacement drifts. As the emplacement drifts within a panel become available to accept waste, the emplacement drifts will be isolated from the ongoing development construction by installing isolation barriers.

25.5. OPERATIONS PERIOD SAFETY ANALYSIS

The operational period is anticipated to last for approximately 24 to 50 years, during which wastes will be received at the site and transferred into waste packages that are transported underground and emplaced. The NRC requires that the waste be retrievable from the repository beginning at any time up to 50 years after emplacement begins.

Assessment of preclosure safety involves identification of essential safety functions needed to assure the health and safety of workers and the public; identification of structures, systems, and components that will be used to implement these essential safety functions; and design criteria employed to achieve high reliability and defense in depth. Safety will be assured if the repository systems prevent the dispersal of unacceptable quantities of radioactive source materials and prevent excessive radiation doses to workers and the general public. A combination of essential safety functions, including the integrity of radioactive sources, containment and confinement, and filtration, will be used to assure safety.

Shielding integrity will be provided by assuring that shield walls and other components remain intact during external initiating events, and that moveable openings and doorways are interlocked to prevent opening when significant source terms are present on the other side of the moveable shield. Radioactive source-term integrity will be provided by assuring that event sequences, which could breach canister or SNF cladding barriers, have been considered and applicable design criteria applied. Confinement integrity requires a combination of assuring, via design and procedural controls, implemented in accordance with applicable design criteria, that the integrity of the preferential airflow paths to the high efficiency particulate air (HEPA) filters is maintained. This involves assuring structural integrity against external event sequence impacts, and that all vital support systems required to keep airflow through the HEPA filters are provided. External and internal haz-

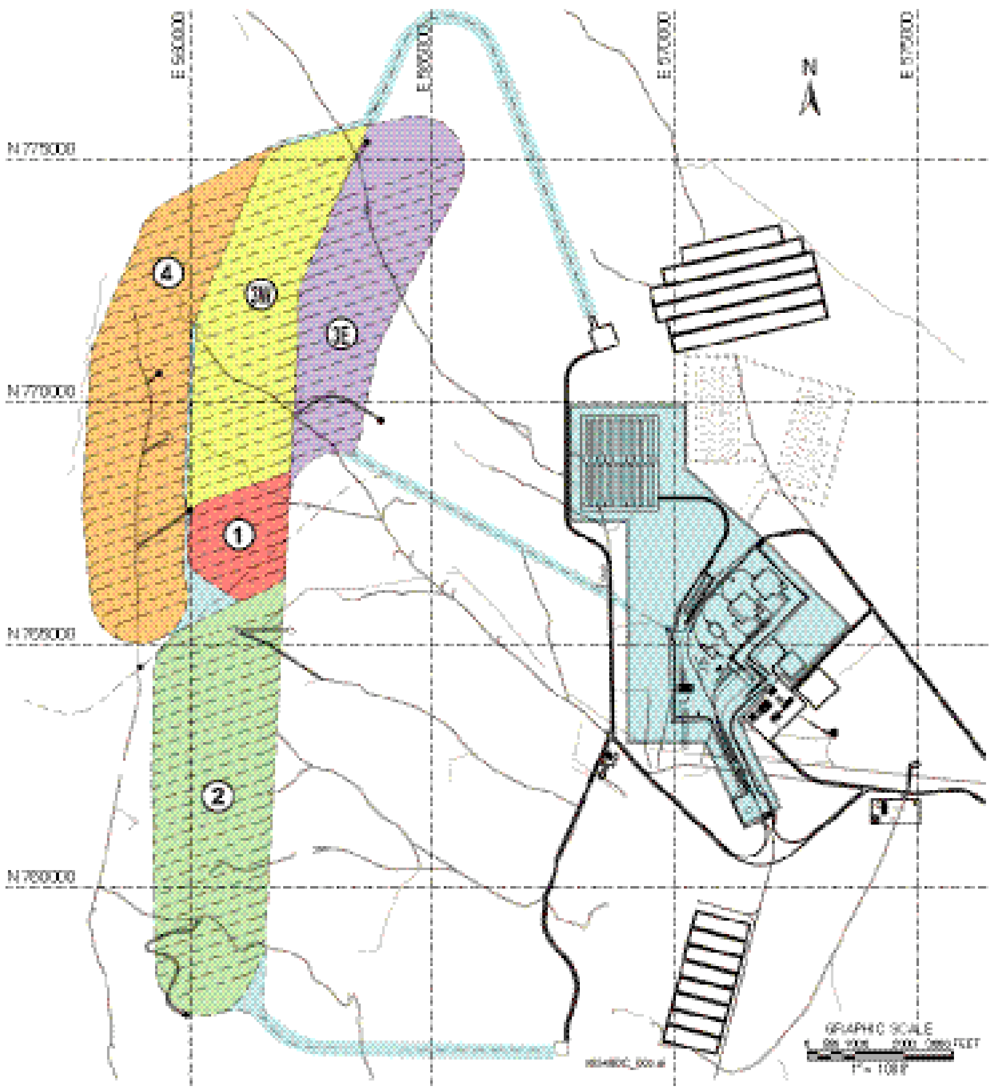

Figure 25.2. Illustration of underground facility, with emplacement sequencing and general location of surface facilities indicated

ard analyses show that considering all relevant event sequences and their consequences, the Yucca Mountain repository operations facilities can be designed, constructed, and operated to ensure that the safety and health of the public and repository workers will be protected.

25.6. THE LONG-TERM SAFETY OF A REPOSITORY AT YUCCA MOUNTAIN

The safety of a repository at Yucca Mountain will be assured by the performance of the natural and engineered features of the site, acting in concert, to prevent or delay the transport of radioactive materials to where the public could eventually be exposed to them. The Yucca Mountain repository system has three barriers that are relied upon to isolate radioactive wastes: the Upper Natural Barrier, the Engineered Barrier System, and the Lower Natural Barrier. Figure 25.3 is a schematic representation of the repository system highlighting the three barriers. The Yucca Mountain site's geologic and hydrologic characteristics form effective natural

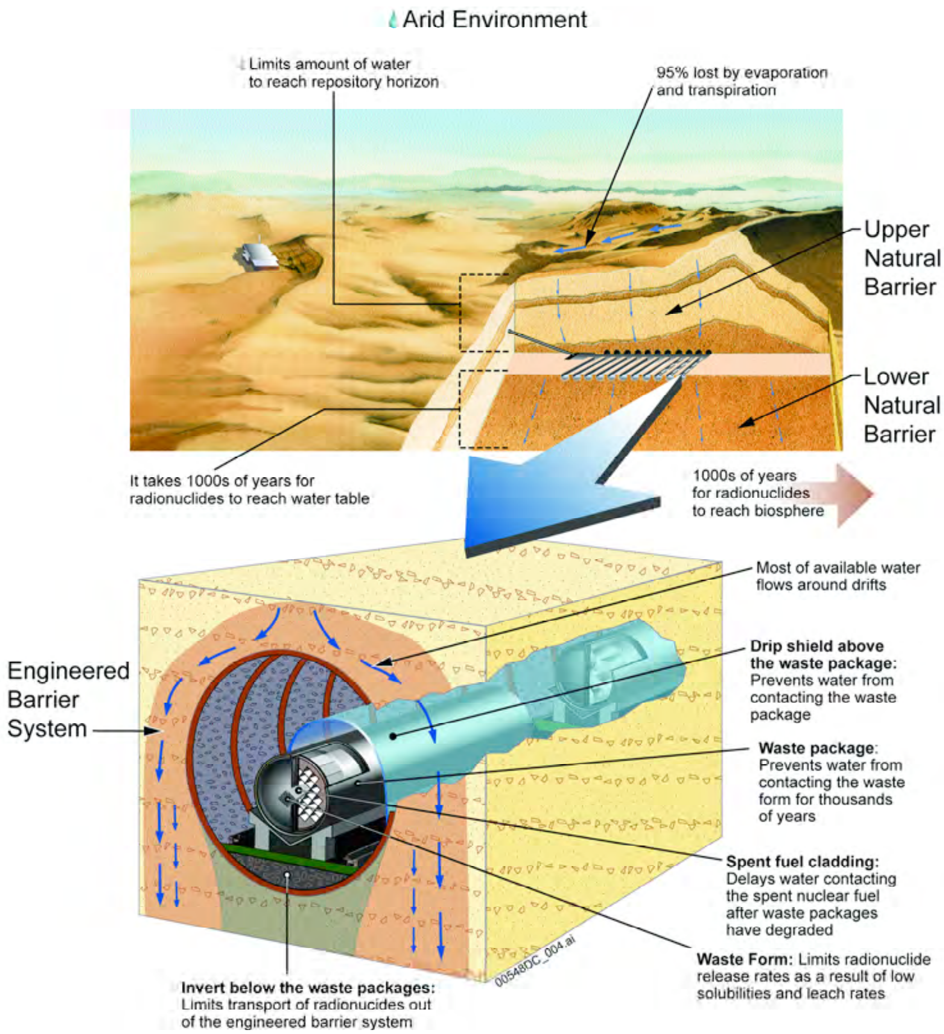

Figure 25.3. Schematic representation of the repository system highlighting the three barriers that are relied upon to isolate radioactive wastes: the Upper Natural Barrier, the Engineered Barrier System, and the Lower Natural Barrier

barriers to the flow of water and to the potential movement of radionuclides. The underground environment within the natural setting is conducive to the design and construction of components that prevent or reduce the movement of water or the potential release and transport of radionuclides. This waste isolation capability of the natural setting is a function of the favorable characteristics of the geologic units. The capability of the Engineered Barrier System will be achieved by designing components to function in the natural setting of Yucca Mountain, particularly its unsaturated rock units.

Materials in the Engineered Barrier System have been chosen so that the components perform their intended functions for many thousands of years. Each of the barriers in the repository system works individually, and together, to limit the movement of water and the release and movement of radionuclides.

25.6.1. Upper Natural Barrier

Yucca Mountain is in a semiarid region where precipitation and humidity are low, thus promoting high evaporation rates. The area near Yucca Mountain also is characterized by great depth to the water table. In the vicinity of the repository, the depth of the water table is approximately 2,000 ft (600 m) below the ground surface, allowing the repository itself to be located approximately 1,000 ft (300 m) above the water table. The topography and surficial soils of Yucca Mountain provide the initial barrier feature that limits the movement of water into the mountain. Runoff, evaporation, and plant transpiration combine to divert water and permit only a small fraction of the already-low expected precipitation at the site to infiltrate into the mountain. The climate and infiltration analyses for Yucca Mountain demonstrate that limited infiltration of water into Yucca Mountain is expected for present and future climates. Precipitation falling on Yucca Mountain is expected to remain low, even for the glacial-transition climates that are forecast for most of the next 10,000 years.

Volcanic tuffs in the unsaturated zone above the repository horizon function to prevent or reduce the movement of water through the unsaturated zone and into the emplacement drifts of the repository. These tuffs divert percolating water, damp pulses of infiltration, and have capillary forces that limit seepage into the emplacement drift. Water flow within the nonwelded tuff unit is predominantly in the matrix, and the rock matrix has a large storage capacity. Only a small percentage of the water is expected to pass through fractures, because they are not well connected. The nonwelded tuff unit thus attenuates flow from the overlying welded unit. The welded tuff units that comprise the repository host rock have lower matrix porosity and higher fracture frequency than the overlying nonwelded tuff. Unsaturated flow in the welded units is expected to be primarily through the fractures.

The rate and distribution of seepage into waste emplacement drifts control the amount of water available to contact the Engineered Barrier System. In the unsaturated zone, seepage into the emplacement drifts will be limited because of capillary forces that limit the movement of water into the drift openings. Water is likely to be retained in the small pores and tight fractures of the low-porosity welded tuff, and a substantial fraction of the flow is expected to move around the drift opening and drain through the rock pillars between the drifts. For a period of time, the decay heat of the emplaced waste will be great enough to heat the rock near the emplacement drifts, and the water will be driven away from the emplacement drift wall surfaces.

25.6.2. Engineered Barrier System

The Engineered Barrier System prevents or substantially reduces the release of radionuclides from the waste. Should the waste packages remain intact for tens of thousands of years, as expected, none of the waste forms will be exposed to water and no release of radionuclides will occur.

The drip shield will be designed to divert any seepage away from the waste package. It will prevent water from contacting the waste package as long as it remains intact. Similarly, as long as the waste packages are intact, water cannot contact the waste forms. The Zircaloy cladding that will encase much of the SNF also will prevent the contact of seepage water with the waste form as long as it remains intact. Corrosion rates for Titanium Grade 7 are expected to be sufficiently low such that none of the drip shields is expected to breach by these mechanisms before 10,000 years. Stress corrosion cracking may occur as a result of rockfall onto the drip shields. The general corrosion rates for Alloy 22 are sufficiently low so that none of the waste packages is expected to breach before 10,000 years. Stress corrosion cracking may also occur in the weld regions of some of the waste packages. Mitigation techniques (e.g., laser peening) will be employed to reduce residual stresses below the stress corrosion cracking threshold, but there remains the potential for breaches of some waste packages before 10,000 years. Early failure of some waste packages potentially may occur due to flaws that are undetected during fabrication or as a result of damage during handling. The probability for early failure in this case is likely to be small because of the quality control measures employed.

The rate of release for any radionuclides that escape from a breached waste package will be limited by the characteristics of the Engineered Barrier System. The release of radionuclides will be first impeded by the slow rate of waste form degradation. Waste form degradation cannot begin until the waste package is breached, which allows air and moisture to enter. Because of the unsaturated environment, the elevated temperatures within waste packages, and the presence of drip shields and waste packages, the amount of water in contact with the waste form is expected to be limited to a thin film that could possibly adsorb on the waste form surface.

Release can only occur if radionuclides are dissolved in water or attached to colloids, and if there are continuous liquid pathways in the waste package, including thin films of water. Advective transport of radionuclides out of the barrier can occur only if there is a liquid flux of water along these pathways and if breaches are sufficiently open to permit flow. The movement of many dissolved radionuclides, including those that are the greatest contributors to the total inventory, will be retarded by sorption on iron corrosion products within the waste package. Retardation depends on the volume of these corrosion products and on the distribution coefficients associated with them. Sorption onto the corrosion products also reduces movement of those radionuclides reversibly attached to colloids in the water. The radionuclide inventory released from the Engineered Barrier System after the waste package is breached will be limited, so long as the drip shield remains intact, precluding advective flow out of the waste package and through the invert; in this case, the release will be purely diffusive. The release that could occur in the unlikely event the drip shield is breached, and does not divert seepage away from the waste package, will be greater than the purely diffusive case and depends on the failure characteristics of the waste package. Both diffusive and advective releases have been considered in the evaluation of radionuclide releases from the Engineered Barrier System.

25.6.3. Lower Natural Barrier

Radionuclides that migrate down through the unsaturated zone to the water table are transported through the saturated zone before they can reach the accessible environment. Groundwater below Yucca Mountain is part of the Alkali Flat–Furnace Creek groundwater sub-basin within the Death Valley groundwater system. Present-day flow is to the south and east near Yucca Mountain. This southeasterly flow from the site is incorporated into the stronger southward flow east of the Mountain. The saturated zone of the Lower Natural Barrier includes the fractured volcanic rocks from below the repository to approximately 7 to 9 miles (12 to 14 km) south of Yucca Mountain and the saturated alluvium to the accessible environment. The movement of radionuclides in the saturated zone will be slow because the velocity of water that can carry them is low, particularly in the alluvium. In addition, several processes cause the movement of radionuclides to be slower compared to the rate of movement of the water.

Flow in the volcanic aquifers will be predominantly in the fractures. Radionuclides can exchange between the fractures and matrix via matrix diffusion. This diffusive exchange results in a slower effective travel velocity for the bulk of the released radionuclides relative to water-flow velocities in the fractures, for two reasons. First, the velocity of water in the pores of the matrix will be slower than velocities in the fractures. Second, sorption onto mineral surfaces in the matrix pores will result in even slower movement for those sorbing radionuclides that diffuse into the matrix materials.

Because the alluvial materials are a porous medium, water flow and radionuclide transport occur in intergranular pores. The effective porosity of the alluvium is significantly greater than the fracture porosity of the tuffs. Consequently, pore velocities in the alluvium are expected to be smaller than those in the fractures of the volcanic aquifers. Although matrix diffusion is not considered important in the alluvium, radionuclide movement will be slow because of the low water velocity. In addition, sorption onto minerals in the alluvium results in retardation of the radionuclide movement relative to the water movement in these sediments.

25.7. THE ENVIRONMENTAL PROTECTION AGENCY STANDARD

Existing U.S. Environmental Protection Agency (EPA) standards for Yucca Mountain after permanent closure address all potential pathways of radiation exposure and limit an individual's annual radiation exposure from all pathways, consistent with what the EPA considers an acceptable risk to society for 10,000 years following closure. The EPA also sets a standard by which to protect the groundwater around Yucca Mountain. This standard sets specific limits for the concentration of different types of radioactive particles in the groundwater. Further, the DOE must assess an inadvertent human intrusion into the repository at some time in the future.

Three scenario classes are considered in the Total System Performance Assessment (TSPA) that is used to evaluate repository performance: (1) the nominal scenario class, which includes consideration of potential early waste package failures; (2) the igneous scenario class, which includes consideration of unlikely volcanic intrusion and eruption, both of which are low-probability events; and (3) the seismic scenario class, which includes consideration of waste-package mechanical damage associated with a low-probability seismic event and unlikely deliquescence-induced localized corrosion processes. The sum of the mean annual dose estimates for the individual scenario class modeling cases (i.e., the nominal and the disruptive scenario classes) provides an estimate of the mean annual dose for the total system.

Demonstration of repository performance includes an assessment of the repository's ability to limit radiological exposures to the reasonably maximally exposed individual for the period after permanent closure, in the event of human intrusion into the Engineered Barrier System. The earliest time after disposal that the drip shields and waste package would have degraded sufficiently that a human intrusion could occur without recognition by the drillers would be well after 10,000 years. Because such a human intrusion without recognition is not projected to occur before 10,000 years after disposal, the results of the human intrusion analysis and its bases were included in the Yucca Mountain Environmental Impact Statement, as required.

Demonstration of the performance of the repository also includes compliance with the separate groundwater protection standards. Radionuclide concentrations are calculated by summing the mass of radionuclides reaching the accessible environment in each year for all likely features, events, and processes, and dividing that sum by the representative volume of water to calculate the annual radionuclide concentrations.

Calculations performed to date indicate that a repository at the Yucca Mountain site will likely meet all of the Environmental Protection Agency requirements for 10,000 years following permanent closure. The standard also includes a requirement to assess the performance of the Yucca Mountain repository at time of peak dose. This evaluation has been done and the results have been reported in the Environmental Impact Statement. However, since that time, a Federal Court has vacated that standard to the extent that it only required compliance for 10,000 years. The EPA and the NRC are currently assessing paths forward; their potential actions will have a bearing on the license application process. Proposed revisions to the rules would require the DOE to demonstrate compliance with a health and safety standard to the time of peak dose.

25.8. THE LICENSE APPLICATION

Before a repository can be constructed and operations can begin, the DOE first must apply for and receive construction authorization from the NRC, an independent agency of the federal government. This construction authorization must then be followed by an application for, and receipt of, a license to receive and possess waste. An amendment to the license will be needed to close the repository. The DOE is in the process of preparing the license application.

Although the DOE and NRC are both federal entities, they operate as independent agencies, a relationship that is defined in the Energy Reorganization Act of 1974. This federal law assigns the Nuclear Regulatory Commission the responsibility of regulating the nation's civilian use of nuclear materials to ensure adequate protective measures are in place to protect the public and the environment. DOE was given the responsibility for promoting nuclear power and conducting other energy-related work, including the storage and disposal of SNF and HLW.

If the DOE is approved to construct and operate a repository at Yucca Mountain, it will become an NRC licensee and be subject to their regulations in all phases of operation, including receipt and possession of nuclear materials, and transportation of nuclear materials to the repository site.

Chapter 26

Progress with Multinational Repository Concepts

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ABSTRACT. Since the beginning of this decade, there has been a resurgence of interest in multinational disposal concepts, after early 1970s proposals for such facilities remained undeveloped for many years. Increased nuclear security and proliferation concerns have made obvious the need to keep radioactive and fissile material under close control at all times—including the period after these have been declared as waste to be disposed of in geological repositories.

The past years were marked by strong support from multinational organizations for shared repositories. In particular, the International Atomic Energy Agency (IAEA) published an overview on the topic, emphasized the potential advantages in several top-level speeches, and also initiated a high-level international Expert Group on Multilateral Nuclear Approaches (MNA) that considered initiatives for both the front- and back-end of the fuel cycle. The European Commission (EC) also included the topic of shared regional European repositories in the Nuclear Package of legislation that it is trying (as yet unsuccessfully) to put through parliament. More specifically, the EC provided direct support for the multinational project Support Action: Pilot Initiative for European Regional Repositories (SAPIERR), which involves a working group of representatives from 14 nations in a pilot project on regional repositories in Europe. Further international cooperative efforts have been organized by the not-for-profit association Arius, which currently includes organizational members from eight countries.

In addition to the above initiatives based on "partnering" concepts, one major nuclear nation, Russia, has expressed interest at the governmental level in possibly hosting an international repository. In 2005, two international meetings took place to discuss this initiative further.

26.1. INTRODUCTION

Virtually all countries in the world with nuclear power programs have concluded that geological disposal is a necessity, if we are to make the nuclear fuel cycle safe and environmentally acceptable without putting undue burdens on future generations (IAEA, 2002). There will be no spent nuclear fuel/high-level radioactive waste (SNF/HLW) repository in operation until the next decade, and many countries are looking towards the middle of the century.

For the larger, advanced nuclear programs, the problems are mainly societal issues associated with achieving sufficient public and political acceptance for specific sites for a national repository. For small countries, however, or countries with limited nuclear power programs, or countries with no nuclear power but long-lived wastes from other applications, a national deep geological repository may be ruled out on economic or practicality grounds. If SNF and HLW are not to remain dispersed for indefinite periods in dozens of surface stores around the world, these small countries need access to geological repositories.

This implies that multinational facilities for disposal of

SNF/HLW are a prerequisite for the sustainable, safe, and environmentally friendly use of nuclear power and other nuclear applications. Other activities in the nuclear fuel cycle—uranium supply, enrichment, fuel fabrication, reactor construction, reprocessing—are all provided as international services. The same status must be achieved for disposal.

26.2. THE INCREASING IMPORTANCE OF NONPROLIFERATION AND SECURITY

In addition to the economic, safety, and environmental benefits that multinational repositories can offer, the nonproliferation advantages have often been stressed (IAEA, 2004; Stoll and McCombie, 2001; IAEA, 2005a; Bunn et al., 2001). In recent years, in particular following the series of terrorist attacks from 2001 onwards, increasing attention has focused on both nonproliferation and security aspects (Alvarez et al., 2003; USNRC, 2003; USNRC, 2005). Repeated statements by the Director General of the IAEA have pointed out the need to control the most sensitive parts of the fuel cycle. In speeches to the 2003 General Conference of the IAEA (El Baradei, 2003a) and at the major Waste Management Conference in December 2003 in Stockholm, Director General Mohammed El Baradei pointed out the potential advantages of small countries sharing disposal solutions. Still wider attention to the issue was drawn by an invited article by El Baradei, published in the Economist in October 2003 (El Baradei, 2003b), in which he states:

> …(W)e should consider multinational approaches to the management and disposal of spent fuel and radioactive waste. More than 50 countries have spent fuel stored in temporary sites, awaiting reprocessing or disposal. Not all countries have the right geology to store waste underground and, for many countries with small nuclear programs for electricity generation or for research, the costs of such a facility are prohibitive. Considerable advantages—in cost, safety, security and non-proliferation—would be gained from international co-operation in these stages of the nuclear fuel cycle. These initiatives would not simply add more non-proliferation controls, to limit access to weapon-usable nuclear material; they would also provide access to the benefits of nuclear technology for more people in more countries.

It is important to note that nonproliferation efforts include controlling not only enrichment of fissile uranium and reprocessing to separate plutonium, but also long-term storage and disposal of SNF/HLW. This point is made clear in the February 2005 report published by the Expert Group on Multilateral Nuclear Approaches (MNA) that El Baradei set up in mid-2004 (IAEA, 2005b). The MNA report addresses the security and nonproliferation issues in a manner directly applicable to all aspects of the nuclear fuel cycle, and suggests five specific approaches for multinational initiatives. The implications of these proposals for storage and disposal concepts are discussed below.

26.3. ASSURANCE OF NONPROLIFERATION AND OF SUPPLY AND SERVICES

The MNA Group sets out, as deciding factors influencing the assessment of multilateral approaches, "assurance of non-proliferation and assurance of supply and services." The former objective is clearly easier to achieve if multinational storage and disposal facilities can be made available. There are currently 35 countries with nuclear power plants (with more than 500 plants operating, being constructed, or planned) and a total of 69 with research reactors. A total of 674 research reactors were operational, shut down, under construction, or planned in 1997, according to the most recent survey in the IAEA database (www.iaea.or.at/worldatom/rrdb/). Leaving SNF in all of these locations for many decades is obviously less proliferation-resistant than collecting the material into a smaller number of facilities with very strong safeguards controls. In practice, the existing strict controls of the IAEA and EURATOM might even be enhanced by a further level of direct international control over a storage or disposal facility for SNF.

For the short and intermediate time frames, shared storage facilities alone would suffice to contain the proliferation risk. Shipping SNF removed from reactors to one of a few centralized facilities as soon as it has cooled enough for transport would be sensible. Technically, with assured centralized interim storage, the question of implementing repositories could be postponed. There have been various proposals from potential hosts and user countries for shared storage facilities (see, for example, Bunn et al., 2001; Ansolabehere, 2003). However, in practice, as is strongly emphasized in the IAEA multinational storage report currently being drafted (IAEA, 2005b), it will be difficult to transfer SNF/HLW to another country for storage without some clarity on the end-point of the agreement. Returning cooled SNF to many countries after several decades would simply reinstate the current proliferation risks of dispersed storage. Returning HLW from reprocessed SNF reduces proliferation risks by retaining central storage of plutonium, but increases security concerns. Moreover, accepting returned HLW would compel small countries to seek national deep disposal solutions—in which case they may as well have retained the fuel for disposal.

In short, the assurance of nonproliferation sought by the MNA Group is best attained by early implementation of shared storage facilities, with the essential ingredient of an agreed-upon further step of disposal in multilateral repositories—either in the countries storing the waste or in a limited number of other volunteering host nations.

How could assurance of supply and services be guaranteed in a situation where many countries are relying on storage or disposal facilities being available in another country? One answer is to have more than one multinational facility and thereby avoid the danger of creating a monopoly. An alternative or complementary measure would be to have direct international guarantees that avoid monopolistic behavior. One way to achieve this is for the IAEA to guarantee continued provision of storage and disposal services. This could be done by establishment of specific internationally operated facilities, whereby agreements with the host country or countries would be required. An alternative is that the IAEA promotes binding arrangements between the service providers, ensuring that each will agree to taking over commitments of others, should these cease to provide promised services for storage or disposal.

The MNA Group recognizes in its report that there is currently no international market for storage or disposal. The MNA Group therefore recommends that the IAEA support the concept "by assuming political leadership to encourage such undertakings." Specific ways forward are possible based on both of the multinational repository scenarios defined by the IAEA—"partnering" and "add-on" (by a large nuclear nation), as documented in TECDOC 1314 (IAEA, 2004). These possibilities are discussed below.

26.4. SCENARIOS FOR MULTINATIONAL APPROACHES TO DISPOSAL

26.4.1. The "Add-On" Scenario

The "add-on" scenario is one in which a large nuclear program accepts wastes from smaller ones. There are several conditions that could enhance the probability of an add-on scenario being successfully implemented:

- The international community should recognize that any country offering storage or disposal services is potentially a contributor to global safety and security.
- A willing host country (or countries) must come forward and should be able to demonstrate to the international community that they have the necessary level of support for the project within the host country.
- Appropriate benefits for the host(s) must be agreed upon. These need not be purely financial; strategic and political issues may also be involved.
- The potential user countries of a multinational repository should develop mechanisms to assure that the safety standards in a multinational repository are not lower than those that each would accept for a national repository.
- International or supranational bodies (e.g., the IAEA or the EC) must be willing to play an active role in developing and controlling the multinational initiatives.
- Existing backlogs of stored SNF, HLW, and LL-ILW must also be transferred, since complete avoidance of the need for an expensive deep repository will be the driver.

In recent times, most discussion of the add-on option has revolved around concepts in which Russia acts as host country. Over the past few years, Russia has been seriously examining the issue of SNF import and is currently the only country supporting this at government level. However, at present, Russia is not legally able to provide disposal facilities as the essential adjunct to storage facilities—nor has the issue of U.S.-flagged SNF import and storage by Russia been resolved. This is an area where significant developments are to be expected in the next few years.

26.4.2. The "Partnering" Scenario

For the "partnering" scenario, in which a group of usually smaller countries cooperate to move towards shared disposal facilities, exploratory studies have been performed most recently by the Arius Association, which also co-manages the European Commission's Support Action: Pilot Initiative for European Regional Repositories (SAPIERR) project on regional repositories (Arius, 2005; SAPIER, 2005).

The following stages can be envisioned for a partnering scenario. It is interesting to observe that they do not differ greatly from steps taken within a federally organized state to seek a national disposal solution.

Pilot feasibility studies. A sufficient number of interested national organisations cooperate to organize and fund pilot studies.

Establishment of a formalized study consortium and dedicated Regional Repository Project Team. To progress to the detailed level of study needed, a structured project team must be created, staffed, and funded at the appropriate level.

Siting studies leading to candidate siting areas in different partner countries. The siting study is clearly the most sensitive work area. Optimally, it should involve working in parallel on a volunteering strategy and on a technical/societal study aimed at ranking options and keeping multiple options open.

Establishment of a Business Consortium or a Joint Venture. The purpose of this organization is to organize and fund the characterization of sites, to finalize agreements on the key issue of compensation for host communities and countries, to select a short list of preferred sites, and to interact with political and regulatory bodies in the candidate countries.

Establish a construction and operation company. This is specific to the hosting country or countries with respect to legal structures, shared liabilities, funding mechanisms, etc.

Repository operation. During the decades for which the repository will operate, the relationships between the partners can be of various types. Given the nature of the facility, international oversight by the IAEA will be a necessity (and the EC for a European repository).

Closure and postclosure. At some time in the far future, the multinational repository will be closed and possibly monitored for some long time. As with the shared benefits, agreements for sharing liabilities must be agreed upon long before this final stage is reached.

The scenario sketched above is one of many possible variants. At the heart of a successful project lies the siting issue. This is indeed a difficult problem even in national programs—but this has not prevented local communities in some countries from agreeing to host repositories. The MNA group of the IAEA also recommends an initial cooperation phase, with participating countries working on a "Siteless Pilot Project"—which is the precise course taken by the European SAPIERR project described in the next paragraph below.

26.5. SAPIERR PROJECT: PILOT INITIATIVE FOR EUROPEAN REGIONAL REPOSITORIES

SAPIERR is a project under the 6th Framework Programme of the European Commission. It is carried out by a consortium of DECOM Slovakia and ARIUS. SAPIERR was launched on December 1, 2003, and its overall duration is 2 years. This project aims to bring together countries in Europe with an interest in investigating the possibilities for shared repositories for SNF/HLW, and in particular those countries with small nuclear power programs that do not have the resources or the full range of expertise to build their own repositories.

A significant achievement of this project is that 21 organizations from 14 countries (Austria, Belgium, Bulgaria, Croatia, Czech Republic, Hungary, Italy, Latvia, Lithuania, The Netherlands, Romania, Slovakia, Slovenia, and Switzerland) have agreed to take part in the SAPIERR working group. Using the inputs of these working group members, the consortium has produced a series of technical reports. Two data reports, one on inventories of radioactive wastes in the SAPIERR countries and one on legal aspects of the regional repository, have been used as the basis for producing a report on options and scenarios for European regional disposal and on recommendations for future research and development in the EU. The findings and other information are available on the project website.

The first two technical reports describe in detail the inventories of SNF, HLW, and long-lived intermediate-level waste, radioactive waste management policies, storage facilities, national programs for repository development and their cost aspects, as well as the legislative framework in the individual countries represented in the SAPIERR working group. The inventory report also includes cumulative inventories for all the SAPIERR countries and their accumulation in time. The third report examines the way to move forward with the concept of regional repositories in Europe. The steps proposed are in line with those outlined in the previous section. One of the key observations is the huge economic savings that the European Union countries could attain if the SAPIERR project members alone were to share a disposal solution—around 8 billion EUR.

A follow-on project to SAPIERR is being suggested, with the objective of responding in more depth to some of the legal and technical issues raised; but more importantly, of establishing, by 2008, a legal entity that could formally take on the role of pursuing the European regional solution.

Regional repositories are also of interest outside Europe, but have been little studied. SAPIERR will hopefully put the European Union in a leading position to provide advice and, possibly, services to other countries.

26.6. RECENT FURTHER PROGRESS WITH MULTINATIONAL INITIATIVES

In the past few years, there have been significant developments towards multinational repositories in several respects. The key points are listed briefly below.

26.6.1. IAEA Support

- A series of public statements by the Director General emphasizing the need for multinational approaches
- Publication of a technical document on multinational disposal and one on regional storage
- Establishment of the Multinational Approaches Expert Group mentioned above

26.6.2. European Commission Support

- Inclusion of regional repository concepts in the draft EC Waste Directive
- Support of the SAPIERR project mentioned above

26.6.3. Further International Developments

- Support by U.S. workers at MIT working on a project on "The Future of Nuclear Power" (Ansolabehere et al., 2003).
- Financing by the independent Russell Foundation of a U.S. Academy of Sciences-Russian Academy of Science meeting on international repositories in Vienna.
- Recent discussions among representatives of Latin American countries on the possibilities for regional disposal facilities.
- The topic of multinational disposal is integrated into numerous international conferences on waste management at the technical and also the legal level.

26.6.4. The Arius Association

The Arius (Association for Regional and International Underground Storage) is a small group of organizations, currently from eight countries, cooperating in an association to support the concept of sharing facilities for storage and disposal of all types of long-lived radioactive wastes. Arius is an organization without commercial goals. The mission of the association is to promote concepts for socially acceptable international and regional solutions for environmentally safe, secure, and economical storage and disposal of long-lived radioactive wastes. A key objective is to explore ways of making provision for shared storage and disposal facilities for smaller users, who may not wish to—or may not have the resources to—develop facilities of their own.

26.6.5. Russian Developments

- Government efforts to establish the legal basis for import

- Joint Russian Academy of Science—U.S. National Academies Workshop held in Vienna in June 2005, as a follow-on from the Moscow 2003 meeting
- Dedicated conference sponsored by the Russian Ministry of Atomic Energy and the IAEA in Moscow in July 2005.

26.7. POLITICAL/PUBLIC ATTITUDES

There are no legal obstacles to countries deciding that they will implement a common repository in a willing host country. If this course is chosen, then lawyers from the partner countries involved can develop a legal framework for the cooperation. The feasibility of realizing a regional repository is thus not strongly influenced by legal constraints.

The feasibility is, however, strongly dependent upon the political and public attitudes in both host and user countries, and the issue remains a sensitive political topic in various countries—particularly in those that fear that the prospect of a regional solution could disrupt national programs. This could, some believe, happen in either of two ways: (1) concern that a national repository compelled to accept foreign wastes might make acceptance of a site by a local community more problematic; or (2) the prospect of being able to export wastes to a regional repository might lead national politicians or waste owners to reduce the priority on (and the funding for) a national disposal program. The former concern should be allayed by the firm assertions from the IAEA, the EC, and from some national governments that waste import cannot be forced upon any country. The latter concern has not prevented various countries from pursuing a "dual track" option, keeping both national and regional alternatives open. This strategy is not a difficult path to follow, since implementation of either option lies relatively far into the future, and similar national expertise must be built up and maintained for evaluating either option.

Distinct from, but related to, the political attitudes in EU countries are the views of the public in each country on the desirability of a multinational, regional disposal option. This question has been put to the public in the scope of the EUROBAROMETER polling done for the EC (EC, 2002). The polling work done in 1998 and 2001 showed that, while the majority still favor national disposal solutions, increasing numbers recognize the advantages of shared solutions. In some individual EU countries there have also been dedicated polls on the issue. Interesting results have, for instance, been published from Germany, a country whose political leadership is strongly opposed to multinational repositories.

Opinion surveys on waste disposal in Germany carried out by the Institute for Technology Assessment and Systems Analysis in Karlsruhe (Hocke-Bergler et al., 2003) included questions on the topic of international disposal. Only 31.5% of those questioned favored a national solution, with 55.6% preferring an international option. The supporters of an international solution were 70% in favor of an EU solution as opposed to a repository outside the EU. Questioned about whether this multinational EU repository could be in Germany, 40% agreed, 40% disagreed, and the rest were undecided. Significantly, however, 80% were against the repository being sited in their own region of Germany—whether the facility be national or international.

The results of all polling exercises indicate clearly that achieving local acceptance for a repository remains a very challenging task, even for a national facility. This is borne out by actual experience. Only in Finland and Sweden have local communities democratically agreed to host a geological repository for HLW/SNF, provided that it could be shown to be safe. In both cases, the local communities already host nuclear power plants and have a long history of interacting with the repository implementing body. In France, a local region at Bure, with no prior nuclear experience, has agreed to host an underground laboratory that may later be developed into a repository. In various countries (e.g., France, Sweden, Switzerland, UK), local communities have also rejected specific proposed facilities.

For a multinational, shared repository, it can only be expected that the challenge of interacting constructively with the public is still greater than in the national setting. The fear that opposition would increase massively has certainly led to advanced programs explicitly excluding import of foreign waste as an option. The polling results quoted, however, indicate that there is a growing minority of the public that already recognizes the potential advantages of shared regional repositories.

26.8. CONCLUSIONS

There is clear recognition internationally that multinational approaches in the overall nuclear fuel cycle can enhance security and can hinder proliferation. Despite earlier controversies, the potential advantages are also recognized for multinational storage and disposal facilities. Concrete steps can be taken now to move beyond empty expressions of support towards specific practical initiatives.

Specific repository projects involving technical and societal efforts towards siting and constructing a shared repository will need closer coordination, direct involvement of the interested countries and the international agencies, and significantly increased resources. Most of the small countries that could benefit most directly from shared repositories have not yet accumulated sufficient funds to implement a national repository. However, there are certainly sufficient resources available in these countries, if pooled, to support a serious joint waste-disposal program. Initially, this would be aimed at clarifying the options for a shared regional facility. However, more support for back-end studies on storage and disposal is needed. The relatively large funding proposed for tackling security issues at the front end should be complemented by increased—although still comparatively modest—financial support for advancing shared repository projects for commercial reactor fuels. The "partnering" scenario outlined earlier in this paper exemplifies one possible practical approach. A wider and more intensive follow-up to the current SAPIERR project could greatly help further progress.

However, the biggest, potentially fully international storage/disposal initiative that could be grasped and developed immediately is that proposed by Russia. A combination of fuel leasing, allowing take-back of Russian origin fuels, and acceptance of foreign fuels requiring USA consent under existing fuel-flagging rules, would be a first step. In our view, however, the Russian storage initiative will only be acceptable if the endpoint of disposal is also available for those client countries that wish to use it—this means actually available, or specifically planned and financed, rather than held out as a vague future prospect. Of course, real interest in sending SNF to Russia (or to any other country with an international repository) will be shown by small countries only if existing backlogs of stored SNF can also be transferred, since complete avoidance of the need for an expensive deep repository will be the driver. A key driver behind providing storage is the prospect that spent light water reactor (LWR) fuel can be regenerated for use in a new generation of fast reactors. Both the waste-type and the waste take-back implications of this are much different from those of reprocessing. Whatever new technology eventually has to offer in this direction, it is nevertheless likely that Russia would need to offer a menu of storage and disposal services if it were to develop widely used multinational facilities. Currently, some movement towards openly exploring these issues is taking place, as evidenced by the conferences this year in Vienna and Moscow.

If the international community wants to make a really useful contribution to global security and safety, then this is where it could direct its resources. Specifically, we propose that the IAEA offer to assist Russia to move forward, by assembling both the funding and the enormous expertise that exists internationally to develop, in a timely fashion, a state-of-the-art international geological repository for long-lived wastes. In return for this offer, Russia should agree to a new level of transparency and international oversight in the development work. Only in this way can the trust of the international community be enhanced to a level needed for small democratic countries to enter into long-term commitments to transfer fuel to the Russian Federation. This would be a truly worthy project with truly global benefits—it is surely to promote solutions such as this that the IAEA was founded and exists today.

Of course, a single supplier of disposal services could present strategic and economic risks for potential customer countries. Global waste inventories, however, easily justify multiple international repositories, and commercial competition could conceivably encourage this. If the international community acknowledges the global value of having international repositories available and is prepared to support their development, then it is not unlikely that other candidates could also appear. These might be other large countries or they might be smaller countries willing to consider hosting a facility implemented with partners.

We need supranational solutions if we are to achieve any of the goals discussed in this article. These solutions need not only the strongest of support from the United Nations and its Member States, but they also need to be championed by the major countries, working together.

In the ways suggested above, real progress could be made over the next few years, with projects based both on partnering and add-on scenarios. We need bold initiatives for global solutions if we are to achieve the global improvements in safety, security, and economics that multinational repositories can bring.

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Chapter 27

International Training in Geological Disposal

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27.1. INTRODUCTION

The international community has been studying the geological disposal concept for almost 50 years. For the last 30 years, there has been an intensive effort to answer outstanding questions, develop the technology to a practically implementable level, and to bring the concept to fruition. Yet even a brief survey of national programs, such as those presented in this volume, reveals repository development programs spanning a wide range of maturity. Some programs have followed a slow but always forward progression from generic concept to site selection and are on the threshold of repository construction. Others have almost reached this point, then suffered total setbacks that have taken them back to the options-comparison stage of the 1970s. In doing so, not only have they lost all momentum, but they have lost most of the skills and experience built up over decades.

Around the world, the community of scientists, technologists, and program managers with truly in-depth understanding of the broad issues behind a radioactive waste management and disposal project is numbered in the hundreds. While it is easy to find specialists in many of the engineering sectors—rock mechanics, underground construction, seismic surveys, package handling technology and many others are widely used outside the nuclear sector—people skilled in identifying data requirements and putting repository projects together are scarce.

New programs want to follow the paths of the most successful national projects. Those countries just embarking on repository projects can see the value in underground test facilities, can see the problems in site selection methodologies, want to apply state-of-the-art safety assessment tools, and so on—but they do not have the decades of trial-and-error experience that helps them discriminate between what is useful and what is not.

The other aspect to this problem is that of time scales. Most programs have taken decades to reach the implementation stage (even those that have been skilled in negotiating the political, economic, and societal issues that dominate decision making) and now plan to manage repository projects that will take decades more to complete. Those who make designs and construct imaginatively staged programs today will not likely be involved in any of the key decisions in the future (on monitoring, refurbishment, backfilling, sealing, retrieval, closure licensing, etc.).

Ensuring that we have sufficient skills internationally over many decades into the future is thus regarded as an important challenge to current generations of managers. It is particularly relevant today, when many of those involved in developing the geological disposal concept are on the verge of retirement. The experience of what works and what does not, both technically and in terms of stakeholder involvement, are vital knowledge if national programs wish to maximize their efficiency and avoid some of the pitfalls of the past.

With this background in mind, there have been several initiatives taken in the last two to three years to ensure proper provision of education and training in radioactive waste management—in particular geological disposal. The background to these initiatives was recently reviewed by Chapman and Sakuma (2005). Notable among these initiatives are:

- The establishment of the International Training Center (ITC) School of Underground Waste Storage and Disposal in Switzerland
- The International Atomic Energy Agency (IAEA) Network of Centres of Excellence *Training in and Demonstration of Waste Disposal Technologies in Underground Research Facilities*
- The CETRAD project in the European Union
- The establishment of the World Nuclear University by the WNA

These initiatives are briefly reviewed in the following sections.

27.2. THE ITC SCHOOL

More than 40 organizations around the world got together in 2003 to address the training situation, founding the ITC School of Underground Waste Storage and Disposal. The ITC School is a nonprofit association, based in Switzerland. The first courses were held at the end of that year, and the School, which now has 50 members from 14 countries, has an active program of courses and workshops, with plans stretching into 2007.

ITC membership is open to any organization supporting the broad concept of providing training and education in waste disposal. Initially established to deal with the issues of radioactive waste disposal, the School is also concerned with the parallel problems of underground disposal of other types of toxic, hazardous waste, where the technical and societal issues are very similar.

The current membership of ITC is very broadly representative of all sectors, fulfilling an original requirement that training not be dominated by any specific sector. The spread of membership between universities, regulatory authorities, research institutes, industry, and nuclear waste management agencies is shown in Figure 27.1 below. In 2004, its first full year of operation, ITC ran five courses and workshops.

The style of these activities varied considerably. The two November courses were large-scale training courses, organized in collaboration with the IAEA, within the framework of its Network of Centres of Excellence *Training in and Demonstration of Waste Disposal Technologies in Underground Research Facilities* (see

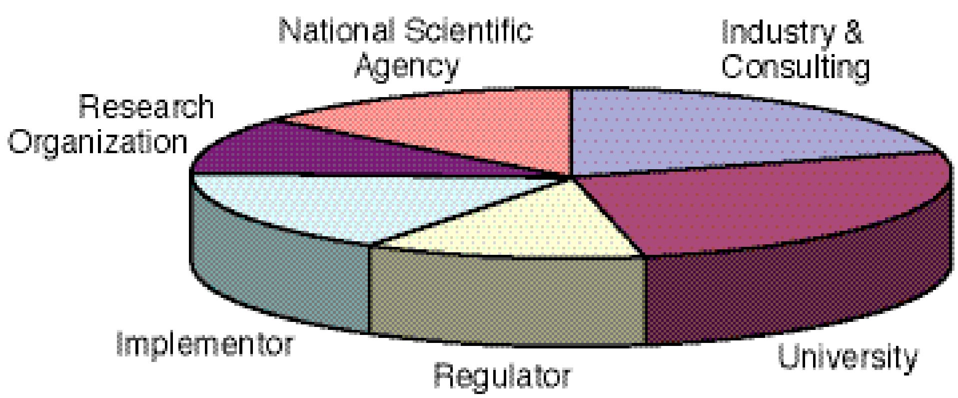

Figure 27.1. Percentage breakdown of ITC membership

Table 27.1. ITC courses and workshops in 2004			
Month	**Course**	**Location**	**Participants**
March	Geochemical Modeling of Natural and Contaminated Groundwaters	Bern, Switzerland	14
May	Workshop on Case Studies of Subsurface Radionuclide Migration	Meiringen, Switzerland	39
November	Siting of Deep Geological Repositories	Rez, Czech Republic	22
November	Fundamentals of Geological Disposal	Meiringen, Switzerland	24
December	The Role of the Safety Case in Planning and the Implementation of Repository Program	Tokyo, Japan	21

next section), and with two Member organizations, RAWRA (Czech Republic) and Nagra (Switzerland).

The contaminated land workshop in May was not a training activity as such, but a scientific symposium that attracted 39 participants from 15 countries and led to a high-profile publication in EOS (Chapman et al., 2004). The workshop was organized in collaboration with EAWAG, Switzerland.

In March, an intensive specialist-training course on geochemical modeling was held in Bern, Switzerland, organized for a small number of scientists in collaboration with the University of Bern. There will be a follow-on specialist course in late 2005 on reactive transport modeling, building upon the material presented in 2004.

December 2004 saw ITC's first course in Japan, where many of its members are based. This was a short and intensive course, held partly over a weekend, for senior managers (generally scientific directors) of waste management organizations and government and regulatory agencies, aimed at defining how safety cases are defined, developed, and used.

Attendance at all the courses was high and at, or close to, the maximum that could be accommodated. Altogether, ITC was able to welcome over 100 participants to its first year of activities. Feedback from participants was generally excellent and showed that all the courses were well received. In particular, the quality of the course documentation has been widely acknowledged.

27.3. THE IAEA NETWORK OF CENTRES OF EXCELLENCE

Several years ago, partly in response to calls from its Member States, the IAEA recognized the need for propagation of experience and dissemination of knowledge to newer programs worldwide. In particular, it saw the value of using existing underground laboratories to help provide practical, hands-on training to those scientists and technologists just entering the field. Consequently, in 2001 it established a Network of Centres of Excellence entitled "Training in and Demonstration of Waste Disposal Technologies in Underground Research Facilities." The stated objective of the Network is to assist in the transfer of knowledge and technology from member States with advanced research and development programs in underground research facilities, to Member States with less well-developed repository implementation programs and/or having no direct access to underground research laboratories.

The project is led from the IAEA's Department of Energy, Division of Waste Technology (NEFW). Primarily, the project is funded and managed through Agency Project L.402/11 (Building Confidence in Geological Disposal of Radioactive Waste). Significant support to the project is provided directly by the Department of Technical Co-operation and indirectly through extra-budgetary contributions from some Member States that own underground research facilities.

Through various pathways such as classroom and underground instruction, scientific visits, fellowships, coordinated research programs, and on-site group train-

Table 27.2. IAEA Network recipient countries

Aremenia	Hungary	Russian Federation
Bulgaria	India	Slovenia
China	Korea (Republic)	Slovakia
Croatia	Lithuania	South Africa
Czech Republic	Romania	Ukraine

ing, the Network provides opportunities to increase levels of competence in nuclear waste management among countries having spent nuclear fuel and highly radioactive waste to be disposed of.

The core members of the Network are radioactive waste management organizations and universities from Belgium, Canada, Sweden, Switzerland, the UK, and the USA. Member countries participating in the program are as follows:

The first Coordinated Research Project (CRP) on the topic of swelling clays for use in engineered barrier systems was agreed upon in 2002, and activities started in 2003. The CRP's are intended to develop the research capabilities of Member States through international cooperation on repository technologies of interest to at least several of them. A second CRP on numerical modeling was approved for implementation in 2005. This CRP provides an excellent mechanism to ensure that modeling procedures used in Member States are suited for reliable long-term prediction, and that modelers are familiar with methods to assess the reliability of models.

Classroom and underground instruction began in 2003, with courses offered in Europe through the ITC (as discussed above) and in North America through a cooperative effort between the USA's Department of Energy and Lawrence Berkeley National Laboratory, and Canada's Atomic Energy Canada Limited (AECL). The North American course covers key aspects and topical issues concerned with managing a geological disposal program. Discussions focus first on the concepts of geologic disposal, development of regulatory bases, interactions with the public, and selection of potential disposal sites, and then a more in-depth discussion is undertaken regarding the processes of characterizing a site, developing testing programs, and refining modeling techniques. Participants also tour the underground facilities at the Yucca Mountain site to view surface and subsurface testing facilities and discuss the integration of the underground research laboratory into a permanent disposal facility.

27.4. THE EUROPEAN COMMISSION CETRAD PROJECT

The European Union anticipates a growing need for education and training in the nuclear sector and has already established broad projects such as NEPTUNO (Nuclear European Platform of Training and University Organisations) and ENEN (European Nuclear Engineering Network). In 2004, it launched a project specifically on geological disposal of radioactive wastes to evaluate needs and capabilities in existing Members States and those European countries joining the European Union (EU) in coming years.

CETRAD was a study over a 15-month period to identify training needs, capabilities, and activities in the EU and accession States, and to propose a means of integrating these. It began in January 2004 and was completed in March 2005. The ITC School acted on the project secretariat with the University of Cardiff, UK, the project co-ordinator. The main survey work was completed in 2004, and the analysis was in draft form at the end of the year. The final presentation of the results was made at a workshop in Prague in March 2005.

Some of the key findings were that there is a definite reduction in numbers of experts at the MSc or higher level of training (due to staff retirement), creating a generation gap. Future specialist staff needs are related to the phases of national program implementation. Most countries have no (or only an ill-defined) national long-term strategy for education and training in radioactive waste management. Those European organizations involved in waste management (both implementers and regulators) expect to recruit a minimum of about 200 specialists during the next five years, with a sharp upturn expected when milestones such as development of URLs and repository construction come nearer. Unlike the field of radiation protection, there are no legal requirements for specific levels of training in radioactive waste management.

Among the recommendations of the project were the introduction of a mechanism to coordinate education needs and provisions at a European level, and the introduction of mechanisms to allow recognition and accreditation of training. A new project, aimed at implementing recommendations on training integration, is being prepared, and the ITC is involved in this activity. Further information on the results of CETRAD can be found at: http://www.grc.cf.ac.uk/cetrad/index.php

27.5. THE WORLD NUCLEAR UNIVERSITY

The World Nuclear University (WNU) originated from collaboration between the IAEA and the World Nuclear Association (WNA). The WNU begins its activities in July 2005, with a six-week summer school located in Idaho, USA, aimed at more than 70 participants and covering the whole spectrum of nuclear energy. Waste management is only a very small (one day) component of the WNU summer school, which so far does not have a working group focused on radioactive waste management. However, the WNU represents a significant initiative, with the same drivers and aims as the other activities reported here, but a much wider range of ambition—to ensure skills propagation across the entire nuclear sector.

27.6. CONCLUSIONS

Over a relatively short period of three years, a perceived need for propagation of knowledge and experience in geological disposal has been addressed by a range of successful initiatives. Where almost no training courses existed, there are now active organizations and programs providing training to many dozens of users.

One of the tasks for those involved is to make the clear message that providing education and training is not a peripheral activity that is added on at the end of the budget line. High-quality training is expensive to organize and provide, and all concerned organizations need to ensure that it is given high priority and is properly funded. This will be essential if we are to take full advantage of the success of the last three years and not lose the enthusiasm and momentum with which the international community has started.

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Chapter 28

Why Every National Deep-Geological-Isolation Program Needs a Long-Term Science & Technology Component

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28.1. INTRODUCTION: RATIONALE FOR A LONG-TERM SCIENCE & TECHNOLOGY PROGRAM

The objective of this chapter is to set down the rationale for a separate Science and Technology (S & T) Program within every national deep-geological-isolation program.

The fundamental rationale for such a program is to provide a dedicated focus for longer-term science and technology activities that ultimately will benefit the whole repository mission. Such a program, separately funded and with a dedicated staff (separate from the "mainline" activities to develop the repository, the surface facilities, and the transportation system), can devote itself exclusively to the development and management of a long-term science and technology program. Broad experience in governments worldwide has demonstrated that line offices are unlikely to be able to develop and sustain both the appropriate longer-term philosophy and the specialized skills associated with managing longer-term science and technology projects. Accomplishing both of these requires a separate dedicated program office with its own staff.

As discussed in more detail below, the philosophy of an S & T Program should be to undertake projects that are longer-term in character. The S & T Program should not be explicitly charged with developing information or tools to support the regulatory needs of the repository development office, nor should its projects be intended to be in the critical path vis-à-vis any such line-office responsibility. Each repository-development line office has the responsibility to maintain its own technological capabilities to support its own mission. The S & T Program's jumping-off point must be to assume that the main repository-development activities are all proceeding on their own schedule, with more than adequate technical capabilities, and with more than adequate safety margins when compared to all applicable regulations. Nevertheless—that is, even if the mainline repository-development work is proceeding successfully, including getting to the point that the repository begins waste-emplacement operations—it is clear that the most that can be expected of the repository's design approaches and analytical methods is that they capture the best current practice at the time. However, such best practices do not remain static over time. Hence, given the multi-decade period over which a repository program will be operating both the underground repository and its supporting surface and transportation facilities, it is incumbent upon every repository program to work diligently to introduce both advanced technologies and advanced analytical methods, so as to improve on "current practice." Ultimately, it is expected that many of the new approaches will become "current practice."

If any repository program were to remain technologically static over a several-decade-long time period, it would be a serious disservice! More importantly, because of the unprecedented requirement to assure safe radioactive-waste isolation for many thousands of years, the country's regulations, as well as the broad society speaking through the spirit of those regulations, should insist that throughout the operating phase of the repository, there must be efforts to continually challenge and probe the technical basis for the repository's ability to isolate the radioactive wastes adequately. Only by so doing can society achieve increasing confidence that the ultimate disposal of the waste deep underground is adequately safe. Achieving this increasing confidence requires a continually improving science-and-technology knowledge base—hence the need for an ongoing Science & Technology Program.

The S & T Program must be an institutionalized part of the overall repository program, but it must be explicitly separate from the mainline activities—connected to them enough to assure that the advanced technologies and methods being pursued are useful and appropriate, but separately funded and managed. This is because if the S & T Program is managerially linked directly to the mainline repository-development work, the shorter-term perspective of the latter will end up compromising the longer-term perspective of the former. This inevitability is well established by experience in other government programs.

28.2. S & T PROGRAM OBJECTIVES

The objectives of an S & T Program should be:

(1) To improve existing and develop new technologies to achieve savings in the waste management system schedule and life-cycle costs, as well as efficiencies in the waste management system (transportation, waste handling, disposal)
(2) To increase understanding of repository and surface-facility performance.

These objectives are complementary, in the sense that some of the projects to be undertaken will address parts of both of these objectives, and also in the sense that increasing our understanding of system performance often will enable the introduction of new technologies that can achieve savings or efficiencies.

28.3. S & T PROGRAM PHILOSOPHY

The S & T Program philosophy should embody a few major principles. These are discussed below:

28.3.1. The Scope, Nature, and Character of the S & T Program Activities

- The S & T Program should be charged with taking the longer view, supporting efforts that sometimes might take 3 or 5 or even 10 years to mature.
- The S & T Program should be explicitly distinct from the mainline repository-development effort, which will always comprise the vast majority of the repository program's effort and resources. In practice, this means that no S & T activities should be on the critical path of the repository program's work, nor should S & T activities have as their principal goal to provide direct short-term support for that work. This is because the nature of the S & T work is that it is research that is expected to be successful, but not guaranteed to be successful. Indeed, it is desirable for the S & T Program to support some speculative technical projects that have only a modest likelihood of success but would have large benefits if successful. It is clear that, given the desirable speculative character of the overall S & T Program portfolio, nothing within that portfolio can be on the critical path for the overall repository program.
- The S & T Program should encourage innovative ideas that are new, creative, or conceive of a different approach to an existing program activity or analysis. To accomplish this, the S & T Program staff is charged with keeping abreast of technological advances in all of the fields relevant to the mainline program.
- The S & T Program scope should be very broad—the intent is that the S & T Program should support all repository program activities, including not only the underground repository but the surface facilities, the transportation activities, and systems integration work.

Thus, the S & T Program should be devoted to the pursuit of long-term technical opportunities, including enhanced understanding, that have the potential for implementation within the mainline program. This makes the S & T Program an applied program in practice, insofar as results are always intended for future application to the mainline geological-repository system. Hence, technical projects within the S & T portfolio can represent a mix of scoping studies, exploratory investigations, applied engineering projects to develop a promising technology, and more fundamental scientific studies.

28.3.2. How the S & T Program Should Operate

- The S & T Program should carry out its projects using the best available resources at a variety of institutions. It should support projects at government laboratories as well as at outside institutions (for example, private firms, universities, institutes, and foreign entities).
- The S & T Program should encourage multi-institution collaborations wherever feasible. The rationale is that historically such collaborations have often been more effective than a single institution at developing new ideas, novel tools, and improved methods of analysis.
- The S & T Program requires periodic independent peer review. One effective way is to appoint a peer-review committee that meets periodically, with membership drawn from the appropriate technical disciplines, but other approaches are also workable.
- The S & T Program should welcome periodic communications with outside groups. This includes presentations to prestigious review boards both within the country and internationally. The benefit to the Program from diverse perspectives is always a valuable source of input.
- How a Science & Technology project's results are ultimately implemented by the mainline repository program is important. When an S & T project has carried its work sufficiently far that it is ready for implementation, then the S & T Program's work has been completed, and a "handoff" to the mainline program should occur. At which stage this handoff will occur can only be determined on a case-by-case basis. However, the governing philosophy is that the S & T Program's responsibility extends to the point of proof-of-concept, or sometimes further to a demonstration or a trial. After this point, the work will approach the implementation phase, which should be supported and managed by the mainline repository program.
- It is important for the S & T Program to reach out to the broader technical community by publicizing the existence of the program, and by informing itself about potential advances in each of the several technical areas within the Program's scope.

28.3.3. Products of the S & T Program

The S & T Program should produce several different types of "products," depending on the project. Typical products could include:

- Periodic and final reports for each project
- Technical data or design information
- Hardware or software prototypes
- Analysis tools, either developed, demonstrated, or both
- Improved understanding, aimed at making the waste management system more efficient or effective.