Geological Challenges in Radioactive Waste Isolation

Third Worldwide Review

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The cover illustration from the United States Department of Energy shows a schematic drawing of the Yucca Mountain Project in the State of Nevada. It is used with the permission of the U.S. Department of Energy.

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Chapter 1

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

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1.1. INTRODUCTION

The first worldwide review of geological problems in radioactive waste isolation was published by the Ernest Orlando Lawrence Berkeley National Laboratory (Berkeley Lab) in 1991 (Witherspoon, 1991). This review was a compilation of reports that had been submitted to a workshop held in conjunction with the 28th International Geological Congress that took place July 9–19, 1989, in Washington, D.C. Reports from 15 countries were presented at the workshop, and four countries provided reports after the workshop, so that material from 19 different countries was included in the first review.

It was apparent from the widespread interest in the first review that the challenges of providing a permanent and reliable method of isolating radioactive waste from the biosphere for long periods of time were quite formidable. This is especially the case in connection with high-level waste (HLW), after its removal from 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 and do not pose the same level of difficulty as HLW, they constitute another problem for the nuclear industry.

After the 1991 review, new technology was being developed to solve these waste isolation problems, and there was a need to publish the results for the benefit of the nuclear industry. Thus, it was decided to gather material on the latest developments, and Berkeley Lab published the *Second Worldwide Review* in September 1996 (Witherspoon, 1996). This second review contains reports from 26 countries.

By 1996, no repository for HLW had been put in operation, but some important progress had been made on the fundamental problems involved in characterizing a site for a disposal project. To decide where to locate a repository for HLW requires a lengthy and detailed process of characterizing the rock mass in which the waste will be placed. Some countries have been working on this process for a number of years, and the 1996 review describes a wide variety of technologies developed by the various countries. As more experience has been gained, the use of underground test facilities in countries such as Canada, Germany, Sweden, Switzerland, and United States has been found to provide a means of carrying out large-scale investigations on rock masses, as an important part of the characterization process. The term "underground research laboratory" (URL) is now used to describe such facilities.

Five years have passed since the 1996 review was published, and there are now over 30 countries where a great variety of field investigations are being carried out to locate suitable disposal sites for both HLW and LILW. Many countries have made significant progress since 1996, and therefore it was decided to gather material on the latest developments on radioactive waste isolation and publish another review.

1.2. SOME HIGHLIGHTS FROM THE *THIRD WORLDWIDE REVIEW*

1.2.1. *Third Worldwide Review Workshop*

We initiated the collection of material for this *Third Worldwide Review* on Geological Challenges in Radioactive Waste Isolation by organizing a workshop held on April 27–28, 2001, in Berkeley, California, sponsored by the U.S. Department of Energy. This event was scheduled immediately before the "International

High-Level Nuclear Waste Conference" that was held in Las Vegas, Nevada, from April 29 through May 3, 2001, with a special field trip to the Yucca Mountain Project (see Dyer and Voegele, 2001) on the last day of the conference. This enabled workshop attendees to obtain a comprehensive briefing on the current status of waste isolation activities worldwide before attending the international conference and joining in the field trip to Yucca Mountain.

There were 60 participants at the workshop, which included representatives from 20 of the 32 countries contributing reports to the *Third Worldwide Review*. A list of all attendees is given in Appendix A, and a group photograph is reproduced in Figure 1.1.

At the workshop, short reports on the status of waste isolation activities in the 32 countries were presented and discussed, and all of this material is available, in much more detail, in Chapters 2 through 33 of this volume. In addition to these reports, there are four more presentations/chapters on other topics of interest. We have added Chapter 34 because we felt it would be of interest to bring the reader up to date on the Waste Isolation Pilot Plant in New Mexico, the first radioactive waste repository in operation in the United States. In view of a growing interest in regional and international repositories, we have added Chapters 35, 36, and 37, which provide current thinking with regard to the applicability of this concept from three different viewpoints. Below, we briefly discuss some of the important highlights from the *Third Worldwide Review*; Table 1.1. summarizes major aspects of the progress in each of the 32 countries.

1.2.2. Increasing Use of Underground Research Laboratories

One of the most important developments in the *Third Worldwide Review* is the recognition of the value of an underground research laboratory (URL) in providing direct access to a rock mass at depth that needs to be investigated and characterized for the purposes of radioactive waste isolation. In the 1996 review, there were only five countries with well-developed URLs, which they had been using for a considerable number of years. As can be seen on Table 1.1, at the end of this chapter, there are now 13 countries that are using URLs or are in various stages of planning and developing such facilities. Two countries (Japan and Switzerland) have each developed two URLs, which enable them to carry out research investigations in two different rock systems at the same time. France is currently constructing their first URLin argillite and planning a program for a second URL in granite.

This increasing use of URLs has led to another important development, in which two or more countries can contribute to joint projects in underground research on problems of mutual interest. At present, clay and granite are the dominant rock types that are being investigated in the URLs of Europe, and a number of joint projects have been set up to investigate the characteristics of these two different rock types. For example, in Switzerland, Nagra and 17 other organizations from nine countries have been involved in granite research at the Grimsel Test Site (GTS) for some time and are now investigating options for a continuation of joint projects when present work in "Phase V" ends in 2002/2003 (McKinley et al., 2001). In effect, the GTS has become an international center for

Figure 1.1. Photograph of participants at the *Third Worldwide Review* Workshop at LBNL on Geological Challenges in Radioactive Waste Isolation taken April 28, 2001, Berkeley, California.

research on radioactive waste isolation in granite. In addition, Nagra is involved in a second test site, at Mont Terri in the Jura Mountains, where Nagra and partner organizations are now entering a seventh one-year phase of research in the Opalinus Clay.

Another example is the Empresa Nacional de Residuos Radiactivos (ENRESA) in Spain, which because of regulatory issues is engaged in joint research projects on many fronts. ENRESA is now participating in three different projects at a URLin Sweden (Äspö) and a number of projects in two URLs in Switzerland (Grimsel and Mt. Terri). ENRESA is also involved in projects on the Boom Clay at Mol in Belgium and is taking part in a project on the Callovo-Oxford argillite in the Meuse/Haute-Marne URL at Bure in France, (Astudillo, 2000).

1.2.3. Public Acceptance of the Management of Radioactive Waste Isolation Projects

Problems with public acceptance of the management of radioactive-waste isolation projects are reported in this third review. As the first investigations on waste isolation were being developed over 20 years ago, it soon became clear that HLW isolation was a very complicated problem. Many scientific and technical difficulties had to be overcome in solving the basic problem of characterizing underground rock systems in which a repository could be designed, constructed, and expected to isolate radioactive waste from the biosphere for thousands of years.

Understandably, the efforts of investigators were focused on these technical difficulties, but as the results of their investigations were published, the magnitude of this unusually complicated situation began to catch the attention of the general public. The reaction of the public, especially where deficiencies in the technical and social aspects were revealed, has led to setbacks in the waste management projects of a few countries. An example of this result occurred in the United Kingdom, when Nirex applied for permission to construct a Rock Characterization Facility (RCF) at their Sellafield project. The application was rejected by a local county council, and when the rejection was upheld in a public inquiry, Nirex terminated the Sellafield project. Further hearings and a Parliamentary review did not change the situation. Nirex has since recognized that in the events leading up to the RCF inquiry, "their decisions were not transparent, and there was a lack of stakeholder involvement" (Hooper, 2001).

In approaching this problem of public acceptance, the Japanese have taken an unusual step to promote public understanding of the nature of an underground repository. JNC (Japan Nuclear Cycle Development Institute) has developed a special demonstration tool named "Geofuture21" that is in operation in their JNC Tokai center. In Geofuture21, people can enter a virtual repository placed 1,000 meters underground by a combination of scientific simulation, 3-D visualization, and a motion system. In this way, the public can experience a simulated earthquake deep underground, observe the behavior of an engineered barrier system, and witness how radionuclides move through bentonite. According to Masuda and Kawata (2001), about 90% of over 12,000 visitors so far have responded that they could understand geological disposal quite well.

Discussions concerning public acceptance indicate that practically all the countries in this review are aware of this problem, as well as the importance of stakeholder involvement in waste-isolation projects. The need to keep the local public informed of developments in field investigations is one of the key features being stressed. Assuring transparency and traceability is another key feature.

1.2.4. Approval of Decision-in-Principle for a Final Disposal Facility in Finland

In May 1999, Posiva submitted an application to the Finnish government for the so-called Decision-in-Principle (DiP) on the final disposal facility for HLW to be built at Olkiluoto in the municipality of Eurajoki. 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 and operate this facility (Ryhänen, 1991).

Finnish law requires the passage of the DiP before any significant commitment to a nuclear facility is made. In March 2001, the Parliament Committee of Environment, having confirmed that certain requirements (which include municipal approval) had been met, passed a statement in support of the decision, and May 18, 2001, the Parliament ratified the decision (Vira, 2001).

1.2.5. Developments in Site Selection and Specialized Repository Equipment in Sweden

Having worked in the development and utilization of several URLs in Sweden, starting with the Stripa project in 1977

(Witherspoon and Degerman, 1978), the Swedish Nuclear Fuel and Waste Management Company (SKB) has carried out a wide variety of research investigations in this type of underground facility. After evaluating various alternatives, they have now reached the end of the feasibility study phase and have selected three sites in the municipalities of Oskarshamm in southeast Sweden, and Tierp and Östhammar in Northern Uppland. It is now up to the municipalities concerned to decide if they want to participate in the next step of the siting process. If the decision is positive, site investigations will start in 2002 (Lundquist, 2001).

For many years, SKB has also been considering what a repository will look like and what materials, special equipment, and technology will be used in its development. In investigating this concept, SKB has been working on the design and fabrication of a copper canister that holds either 4 PWR or 12 BWR assemblies and weighs about 25 tons. The large size of this canister has 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 holes, and creating a prototype repository to simulate the integrated function of the repository components and provide a full-scale reference for comparison with models.

1.2.6. Yucca Mountain Project in the United States

In 1997, the U.S. Congress required the U.S. Department of Energy (DOE) to prepare a Viability Assessment of the Yucca Mountain site in Nevada. The Viability Assessment (U.S. DOE, 1998) described the following: (1) a preliminary design concept for the critical elements of a repository and waste package; (2) a total system performance assessment that describes the probable behavior of a repository at Yucca Mountain; (3) a plan and cost estimate for the remaining work required to complete and submit a license application to the U.S. Nuclear Regulatory Commission; and (4) an estimate of the costs to construct and operate a repository in accordance with the design concept. It was concluded that the Yucca Mountain site remains a promising site and that work should proceed to support a decision to recommend the site for development as a repository by the end of 2001 (Dyer and Voegele, 2001).

On January 10, 2002, just before this volume went to press, DOE Secretary Abraham notified Governor Guinn of Nevada of his intent to recommend the Yucca Mountain site to the President. The Secretary's recommendation and comprehensive basis for a site recommendation were submitted to President Bush and made available to the public on February 14, 2002. President Bush considered the Yucca Mountain site to be qualified for a license application to construct a repository and submitted his recommendation to Congress on February 15, 2002.

1.2.7. Waste Isolation Pilot Plant in the United States

The Waste Isolation Pilot Plant (WIPP), located in southeastern New Mexico, is the first radioactive waste repository in operation in the United States. The WIPP project received governmental approval on March 26, 1999, to begin deep geologic disposal of defense-related transuranic (TRU) waste in a bedded salt formation about 650 m below the surface.

WIPP is the world's first deep geologic disposal site designed specifically for TRU wastes, and it is one of a very small number of permanent repositories in salt beds for any type of waste. The capacity of WIPP is limited by the WIPP Land Withdrawal Act to 6.2 million cubic feet of waste. A discussion of its design, development, and the strict controls under which the project must operate are given by Patterson and Nelson (2001).

1.2.8. National and International Repositories

It is of considerable interest to note that the political and social climate for utilization of nuclear power plants in the United States appears to be improving. A recent example of this improving climate can be seen in the Fall 2001 issue of *The Bridge*, a quarterly journal of the National Academy of Engineering. Four articles in this issue present different viewpoints concerning: (1) how nuclear energy will be an important component in addressing climate change (Holdren, 2001); (2) the fact that we may be on the threshold of a "second frontier" in the development of nuclear energy (Marcus, 2001); (3) the role of industry in the development of safe, efficient nuclear technologies (McNeill, 2001); and (4) too few Americans understand or appreciate the enormous benefits of nuclear technologies (Domenici, 2001).

The revival of the industrial utilization of nuclear energy will mean the relicensing of existing power plants in the United States and the construction of new plants that can benefit from the significant improvements in design

Table 1.1. Developments in radioactive waste isolation from 2001 Third 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
Argentina	Comisión Nacional de Energía Atómica (CNEA)	TBD	Clay, evaporites, volcanoclastic, granite	Seven provinces chosen for site-selection purposes	TBD	Data being organized in GIS. Fracture mapping of satellite images is under way.
Armenia	Institute of Geological Sciences of the National Academy of Sciences	Semibiete-Karabakh, Central folded zones	Volcanic, gypsum, salt and clay, granite	Geologic and geophysical investigations	TBD	Complete geological, geophysical, volcanological, and seismotectonic investigations
Belarus	Institute of Geological Sciences	Polesie's area, Gomel region	Granite, salt, clays	Geologic and geophysical investigations of region, numerical models of nuclide migration developed	Originally near-surface repositories; trench-type burial in future	Choose burial grounds with natural barriers, install barriers where needed, provide monitor and testing equipment
Belgium	SCK·CEN	TBD—investigating Boom and Ieper Clay Formations	Clay	URL used 25 yrs for Boom Clay research, now studying effects of waste heat on safety and feasibility of clay	HLW in stainless steel over-pack with bentonite backfill and concrete liner	EURIDICE Consortium managing URL research including PRACLAY project on effects of waste heat on clay
Bulgaria	Geological Institute of Bulgarian Academy of Sciences	TBD—Loess near Kozloduy NPP (LILW), marl at Sumer and granite in Sakar plutons (HLW)	Loess for LILW, clayey marls and granite for HLW	Site-selection methodology developed, funding needed for specific investigations.	LILW pre-stressed concrete cubes at NPP; HLW in deep repository	Need waste-management regulatory body and funding for geological surveys and specific investigations
Canada	Ontario Power Generation (OPG)		Granitic	Since 1978, URL and other facilities used in site characterization studies and technology development.	2 designs: copper containers with waste in boreholes or drifts. Buffers and backfill seal off containers and rooms.	Develop waste-management organization to work with all stakeholders to develop approach that is socially, environmentally, and financially acceptable
China	Beijing Research Institute of Uranium Geology (BRIUG), China National Nuclear Corporation (CNNC).	Beijing Block in Beishan region of Gansu Province, NW China.	Mesozoic granite	Beishan region selected from geological and geo-physical investigations; 2 boreholes drilled and cored	TBD	Comprehensive laboratory studies on core samples, investigate two other blocks in the next five years, and plan to start URL construction by 2015.

Country	Lead Organization	Potential Sites	Prospective Rock Types	Status of Site Characterization	Prospective Design/Engin. Barrier	Near-Term Plans
Croatia	Hazardous Waste Management Agency (APO)	Trgovska gora site in Banovina region	Clayey schist, clay, sandstone	Analysis based on existing data and reports. Mines from war prevent field investigations.	Steel tube waste containers inside near-surface concrete vaults	Detailed geological mapping, geophysical surveys, and borehole drilling to provide data for site characterization.
Czech Republic	Radioactive Waste Repository Authority (RAWRA)	Dukovany for ILW and LLW. TBD for HLW	Granite for HLW	Five areas selected for site characterization. Methods & techniques to be developed at Melechov Massif.	Steel container, bentonite buffer and backfill	Two candidate sites selected by 2015 and final site by 2024. URL starts at site in 2025 and in operation by 2030.
Finland	Posiva Oy	Olkiluoto near NPP at Eurajoki, with existing repository for LLW and new repository site for HLW	Granite	Olkiluoto chosen from detailed site characterization, favorable EIA and small social impact. Eurajoki OK'd site and Parliament gave final approval 5/18/01.	Encapsulation in bedrock several hundred meters deep in metal canisters with buffer and backfill	Verify site suitability, define adequate repository space, characterize host rock for repository, safety assessment, and plan construction. Submit construction license application in 2010.
France	Agence Nationale pour la Gestion des Déchets Radioactifs (ANDRA)	Meuse/Haute-Marne and 15 potential sites in granite areas	Argillite at Meuse site and granite batholiths	Sinking shaft at Meuse site for URL in clay, and designing series of key experiments. Participating in 5 granite experiments in foreign URLs.	Investigating preliminary concepts for designs of disposal cells in clays using tunnels and caverns	Conduct research in URL at Meuse site to provide reliable data to answer key questions on performance assessment of repository in clay. Prepare feasibility report on granite site in 2006.
Germany	Federal Office for Radiation Protection (BfS)/Federal Institute for Geosciences and Natural Resources (BGR)	Gorleben salt dome, Konrad iron ore mine	Rock salt, hard rock sediment within clay barrier	Licensing procedure for Konrad finished in 1992 but not yet approved. Exploration at Gorleben stopped. New procedure for site selection expected in 2002.	Steel container, backfill material depending on waste type and host rock	Pursue concept of constructing one single repository for all types of radwastes. Participating in R&D programs in Belgium, France, Spain, Sweden, Switzerland, and US.

Country	Lead Organization	Potential Sites	Prospective Rock Types	Status of Site Characterization	Prospective Design/Engin. Barrier	Near-Term Plans
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	At Üveghuta, site characterization and social impact programs peer reviewed and approved by IAEA. HLW to be stored at NPP for up to 50yrs pending policy decisions.	At Üveghuta site, waste drums and disposal containers to be emplaced in tunnels with clay as backfill material.	Site characterization and repository design at Üveghuta to continue, public outreach program to continue to establish long-term relationship between local communities and project management.
India	Bhabha Atomic Research Center	Sankara plutons in northwest Rajasthan	Granite	Geophysical surveys, geological and geochemical studies, hydrogeological/rock-mechanical testing carried out.	TBD	Favorable results over an area of a few thousand km² indicate need for additional, more detailed investigations.
Italy	National Agency for New Technology, Energy, and Environment (formerly ENEA)	Near-surface repository for LLW. Long-term storage for HLW at the same site.	TBD	Nuclear energy phased out after 1987. General site-selection process using GIS methodology, in three steps, is ongoing over entire country.	Repository with modules of reinforced concrete containers and steel boxes for LLW. Long-term storage of HLW in cask-type casks.	Third step in GIS methodology currently ongoing for 200 suitable areas identified in second step. Repository scheduled to begin operation in 2009.
Japan	For implementation, the Nuclear Waste Management Organization of Japan (NUMO); for R&D, Japan Nuclear Cycle Development Institute (JNC)	TBD	Crystalline or sedimentary rocks, URLs at Mizunami and Horonobe	Second progress report (H12) completed to demonstrate feasibility, safety, and reliability of disposal concept and to provide input for future siting and regulatory processes. Needed methodology being developed at Mizunami and Horonobe.	Vitrified waste in steel over-pack embedded in bentonite and emplaced either in tunnels or in vertical holes drilled from bottom of tunnels.	Keep stakeholders informed on all developments via data-base on JNC website, provide public with virtual repository to visualize underground system. NUMO will keep public advised of potential sites, details of planned repository and basis for final site selection. Work on new methodology continues at both URLs.

Country	Lead Organization	Potential Sites	Prospective Rock Types	Status of Site Characterization	Prospective Design/Engin. Barrier	Near-Term Plans
Korea	Korea Hydro & Nuclear Power Co. (KHNP) for LILW, Korea Atomic Energy Research Institute (KAERI) for HLW	TBD	Andesite for LILW, Mesozoic plutonic rocks for HLW	Preliminary conceptual design and safety assessment for LILW repository in rock cavern and vault options completed. Plutonic rocks screened as primary host rock for HLW repository. Conducted radio-nuclide migration studies and developed performance assessment code.	Rock storage cavern and cement-grouted vault for LILW. Encapsulate HLW in corrosion-resistant containers in boreholes, with bentonite buffer, drilled in tunnels at depth of ~500 m.	Preliminary assessments of conceptual facilities for LILW provide firm foundation for site-specific assessment activities. Site-specific data required for next stage of LILW project. Next step for HLW repository is further development of repository concept using field data from in situ investigations at specific site(s).
Lithuania	Lithuanian Energy Institute	TBD for SNF and LILW. Existing solid LILW storage facilities at Ignalina NPP cannot be converted to repositories.	For SNF, clay, anhydrite, salt, and crystalline bedrock	Analysis is ongoing, based on existing data and reports	TBD for SNF. Reference design for near-surface LILW repository (concrete vaults)	For SNF, research to develop competence in performance assessment and to select disposal concept that can be adapted to different sites; development of site-selection methodology. For LILW, siting for near-surface repository
Netherlands	Ministry of Environment Steering Commission for Research (CORA)	Retrievable disposal site for HLW at Borssele NPP	Rock salt, clay	Analysis of long-term retrievable storage at surface and underground, in either salt or clay, for up to 300 yrs appears technically feasible.	HLW container in individual cells in tunnel wall. Crushed salt used for buffer in salt; clay-bentonite in clay.	CORA recommends continuation of research program to further improve technical solutions and involve stakeholders in ethical and social aspects.

Country	Lead Organization	Potential Sites	Prospective Rock Types	Status of Site Characterization	Prospective Design/Engin. Barrier	Near-Term Plans
Poland	National Atomic Energy Agency of Poland	Shale at Jarocin and salt domes at Damaslawek, Klodawa, and Lanieta	Triassic shale in SW Poland, salt domes in the Polish Lowland	First geological review led to selection of 44 sites of which four were chosen as promising: one in shale and three in salt domes.	TBD	Continue more detailed site-selection process, construct URL in one of the salt domes to investigate in situ argillaceous-salt conditions.
Romania	Institute of Nuclear Research	TBD	Salt	Long-term safety assessment for repository in hypothetical salt formation has been carried out.	TBD	As site characterization work is carried out and in situ data become available, more realistic safety assessment investigations will be made.
Russia	Ministry; All-Russia Designing and Research Institute of Production Engineering	Tomsk-7, Krasnoyarsk-26, Dimitrovgard, Mayak, Novaya Zemlya archipelago	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 now being 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 2025.
Slovak Republic	Decom Slovakia	Tribec, Zlar, Veporske vrchy, Stolicke vrchy in granitic rocks Cerova vrchovina, Rimavska kotlina in sedimentary rocks	Granite, siltstones and clays	A revised program in development activities has been used to select 4 prospective granitic sites and 2 argillaceous sites. Selection of a host rock will not be made before 2005. Selection of candidate sites is expected around 2010.	Prepared disposal container with 7 WWER-440 SF 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.	Activities should lead to proposal for a first-reference disposal concept, a public-involvement program, information database, investigation of prospective sites, revision of siting criteria, performance assessment based on available data, and selection of materials for engineered barrier.
Slovenia	Agency for Radioactive Waste (ARAO)	TBD	Unconsolidated sediments, hard clay, granite	Areas suitable for LLW repository selected; preliminary geological assessment done	Geological conditions suitable for disposal in surface and underground	Site suitability investigations to be carried out, subject to public response. Plan to select site by 2005.

Country	Lead Organization	Potential Sites	Prospective Rock Types	Status of Site Characterization	Prospective Design/Engin. Barrier	Near-Term Plans
South Africa	South Africa Nuclear Energy Corporation (NECSA)	Vaalputs National Radioactive Waste Disposal Facility	Clay, granite	Drilling in Vaalputs area (1986) found excellent granitic rock, but work was stopped—new national policy now being drafted.	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	Empresa Nacional de Residuos Radioactivos (ENRESA)	TBD	Clay, granite	Detailed analyses of several potential repository sites have been made. Large database from this work now being managed on GIS, which will be updated until 2010.	Carbon-steel canisters embedded horizontally in bentonite buffer spaced 2 m apart in drifts	Develop geologic disposal R&D program with models for site characterization, flow and transport scenarios and performance assessment. Develop generic design for repository in clay or granite; study natural analogs; establish safety criteria.
Sweden	Swedish Nuclear Fuel and Waste Management Co. (SKB)	Oskarshamn in SE Sweden, Tierp and Östhammar in northern Uppland	Granite URL in granite at Äspö	Feasibility studies led to 3 potentially suitable sites for deep repository, approval for site investigations needed from local municipalities. URL at Äspö conducting R&D on methodology needed for deep repository.	Waste in copper canister with cast inserts embedded in bentonite in vertical holes in tunnel filled with bentonite and crushed rock at depth of ~500 m.	With local approval, site investigations to start in 2002. Much work on canister fabrication underway at Canister Laboratory in Oskarshamn. Research on repository technology now concerned with a retrieval test, backfill and plug test, and prototyp e repository.
Switzerland	National Cooperative for the Disposal of Radioactive Waste (NAGRA); Genossenschaft für nukleare Entsorgung Wellenberg (GNW)	Wellenberg for LLW, HLW/SF/TRU siting studies in Northern Switzerland	Marl at Wellenberg; clay, granitic basement in N. Switz.; URLs at Mt. Terri in Opalinus Clay and at Grimsel in granite/granodiorite	Wellenberg L/ILW site accepted at local level but blocked at cantonal level by narrow margin. Opalinus Clay and granitic basement in Northern Switzerland extensively investigated, including 3D seismics and deep boreholes.	SF/HLW in steel canisters embedded horizontally in tunnels with bentonite backfill. L/ILW / TRU in concrete emplacement containers in caverns backfilled with cementitious grout	Second referendum at Wellenberg in 2002 may permit exploration tunnel to gather data to support application for construction license. Siting feasibility project for HLW/SF/TRU focused on the Opalinus Clay of the Zuercher Weinland to be produced in 2002

Country	Lead Organization	Potential Sites	Prospective Rock Types	Status of Site Characterization	Prospective Design/Engin. Barrier	Near-Term Plans
Taiwan	Fuel Cycle and Materials Administration (AEC)	Little Chiu Yu at Wu-Chiu Hsiang for LLRW. SF currently stored in on-site pools at NPPs.	Granite, shale, mudstone	EIS for Little Chiu Yu under review with Taiwan EPA. Approval of feasibility and safety analysis reports plus EIS needed for final approval of site.	TBD	SF disposal under study in project spanning 40 years (1991–2031). Expect SF disposal site identified by 2016 and repository commissioned by 2032. On-site dry storage to supplement on-site pool storage.
Ukraine	Institute of Geological Sciences	Korosten pluton and Malukhev block in Ukrainian shield and salt domes in Dnieper-Donets depression	Granite, salt domes	Site selection and characterization methodologies defined, funding problems with economic restrictions	TBD	Complete R&D on site characterization (1999-2005), characterize selected site, develop URL, demonstrate site safety, obtain license and decision on construction (2005-2020).
United Kingdom	United Kingdom Nirex, Ltd.	TBD	TBD	Request for permission to build a URL near Sellafield rejected by local council and decision supported by Secretary of State for the Environment. Work at Sellafield terminated	TBD	Parliamentary review (1999) points to need for public acceptance of policy on waste management before problem can be settled. Citizen's Panel issues number of suggestions. Government issues proposal (2001) to develop, and implement, a waste-management program that inspires public support and confidence.
United States	United States Department of Energy	Yucca Mountain, Nevada	Volcanic tuff	Site selection and site characterization methodologies have been developed and applied in evaluating Yucca Mountain	Waste within two concentric cylinders (stainless steel inside corrosion-resistant nickel alloy) covered with drip shield and placed horizontally in drifts.	Quantitative assessments of long-term performance of repository for various features, events, and processes is ongoing. Performance-confirmation program established to monitor and confirm repository is behaving as expected. These preclosure-period activities may last up to 300 years.

and operation made in the last 20 years. There will undoubtedly be an increased demand for nuclear power plants around the world as more countries strive to raise their standard of living.

Thus, the potential need for regional and international repositories, particularly for those countries with limited resources and/or unfavorable geological conditions, is a matter of concern. The last three chapters of the review present discussions on three different aspects of this problem. An analysis of international repositories as an essential complement to national facilities is given by McCombie et al. (2001) in Chapter 35, and Pentz and Stoll (2001) discuss the importance of the nonproliferation advantages of deep geologic repositories in Chapter 36. In Chapter 37, Burkart (2001) presents the position of the U.S. Department of State with regard to the development of international repositories that involve spent fuel from countries where the U.S. has consent rights over the retransfer of much or all of the spent fuel.

1.3. CONCLUSIONS

The broad range of activities on radioactive waste isolation that are summarized in Table 1.1 provides a comprehensive picture of the operations that must be carried out in working with this problem. A comparison of these activities with those published in the two previous reviews shows the important progress that is being made in developing and applying the various technologies that have evolved over the past 20 years.

There are two basic challenges in perfecting a system of radioactive waste isolation: choosing an appropriate geologic barrier and designing an effective engineered barrier. One of the most important developments that is evident in a large number of the reports in this review is the recognition that a URL provides an excellent facility for investigating and characterizing a rock mass. Moreover, a URL, once developed, provides a convenient facility for two or more countries to conduct joint investigations. This review describes a number of cooperative projects that have been organized in Europe to take advantage of this kind of a facility in conducting research underground.

Another critical development is the design of the waste canister (and its accessory equipment) for the engineered barrier. This design problem has been given considerable attention in a number of countries for several years, and some impressive results are described and illustrated in this review.

The role of the public as a stakeholder in radioactive waste isolation has not always been fully appreciated. Solutions to the technical problems in characterizing a specific site have generally been obtained without difficulty, but procedures in the past in some countries did not always keep the public and local officials informed of the results. It will be noted in the following chapters that this procedure has caused some problems, especially when approval for a major component in a project was needed. It has been learned that a better way to handle this problem is to keep all stakeholders fully informed of project plans and hold periodic meetings to brief the public, especially in the vicinity of the selected site. This procedure has now been widely adopted and represents one of the most important developments in the *Third Worldwide Review.*

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Chapter 2

Deep Geological Disposal Research in Argentina

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2.1. INTRODUCTION

Argentina requires a deep geological repository for the disposal of radioactive waste—mainly conditioned spent fuel and high-level waste—produced by two research reactors and two nuclear power plants. From 1980 to 1990, the first phase of a feasibility and engineering study was carried out. Geologically, this study was based on the search for granitic rocks. One site (Sierra del Medio, Chubut State) was selected for detailed geological, geophysical, and hydrogeological studies. (This phase of the project was presented in the first *World Wide Review* in 1991 [Palacios et al., 1991].) At the end of the 1980s, however, the project was rejected, and the Comisión Nacional de Energía Atómica (National Atomic Energy Commission [CNEA]) cancelled these studies shortly thereafter. This cancellation resulted directly from inadequate explanation and awareness of what impact a repository would have on the local community. Government decision-makers felt a great deal of pressure from local political action groups, who clamored for the interruption of the project because of lack of information and the subsequent fear this lack created.

In 1991, a new project, entitled "Study for Favorable Geological Formations Suitable for Siting of Repositories for High, Intermediate, and Low-Level Waste Disposal," was begun. Within this project, which continues into the present, research activities have been broadly focused on the search for sedimentary (clay and evaporites), volcaniclastic, and granitic formations suitable for deep geological disposal of radioactive waste.

In 1996, the search for a deep geological disposal facility began. After a detailed review of regional geological literature, seven provinces were selected for further studies. Furthermore, some site-selection factors were taken into consideration to make a preliminary decision about suitable and unsuitable areas (e.g., seismicity, neotectonism, volcanism, geothermal gradient, hydrogeology, climatic changes). Since this is a progress report, not all the information from these selecting factors is available. To compile and efficiently process all the information, we have organized different kinds of data into a Geographical Information System (GIS). As a next step, a program of digital processing and geological interpretation of satellite images is underway, involving the mapping of fractures and structural lineaments in some randomly selected areas, at a 1:500,000 scale.

From the institutional point of view, Law No. 25018, "Radioactive Waste Management Regime," passed by the Parliament in 1998, requires the formulation and deployment of a "Strategic Plan for the Management of Radioactive Waste." This plan includes both deep-geological-disposal research activities and a social communication program.

The lesson learned is that *social communication activities must be carefully undertaken as preparation for any scientific investigation into deep geological disposal of radioactive waste.*

2.2. GEOLOGICAL INVESTIGATIONS

Argentina generates spent fuel mainly from two nuclear power plants in commercial operation (Atucha I and Embalse) and one isotope production and research reactor (RA3). The Atucha I nuclear power plant (NPP) (365 MWe) is driven by a pressurized-vessel heavy-water reactor that is fully fueled (as of November 2001) with 0.85% slightly enriched uranium (SEU); Embalse NPP (600 MWe) is driven by a CANDU reactor fueled with natural uranium. The RA3 reactor is a 5 MW pool type research reactor, upgraded this year to 10 MW, fueled with low-level enriched uranium (LEU). At the Atucha I NPP, spent fuel containing about 1,300 metric tons of heavy metal (HM) are in wet interim storage in pools, whereas at the Embalse NPP, spent fuel containing about 1,400 tons HM is in interim storage—53% in wet (decay pool) and 47% in dry (concrete silo), respectively. From the RA3 reactor, 63 LEU spent-fuel elements containing about 120 kg HM are in wet interim storage in its decay pool at the reactor, as well as in a centralized storage facility away from the reactor. Additionally, at the training and research reactor RA6, a 0.5 MW pool-type research reactor fueled since 1982 with highly enriched uranium (HEU) fuel elements was used by the RA3 in the late 1960s and wet-stored for about 10 years. Nine HEU fuel elements containing about 13 kg HM are in wet interim storage in its at-reactor storage pool.

Within this context, CNEAstarted a program in 1980 to study granitic bodies all over Argentina, entitled “Feasibility Study and Engineering Project—Repository for High-Level Waste” (CNEA, 1990). This program initiated the search for a deep geological disposal facility (repository) site to dispose of high-level waste (HLW) and conditioned spent fuel (SF) from Argentinean NPPs and research reactors, respectively.

The characteristics of almost 200 granitic bodies (presented in Figure 2.1) were studied as a result of this work, carried out under a contract with San Juan National University (UNSJ). Subsequently, eight granitic bodies in the Patagonian Region were selected from almost 200 nationwide (UNSJ, 1980a,b). Finally, one of them, Sierra del Medio, about 40 km away from Gastre in the Provincia de Chubut (Chubut State), was chosen as the place to carry out more detailed studies. At

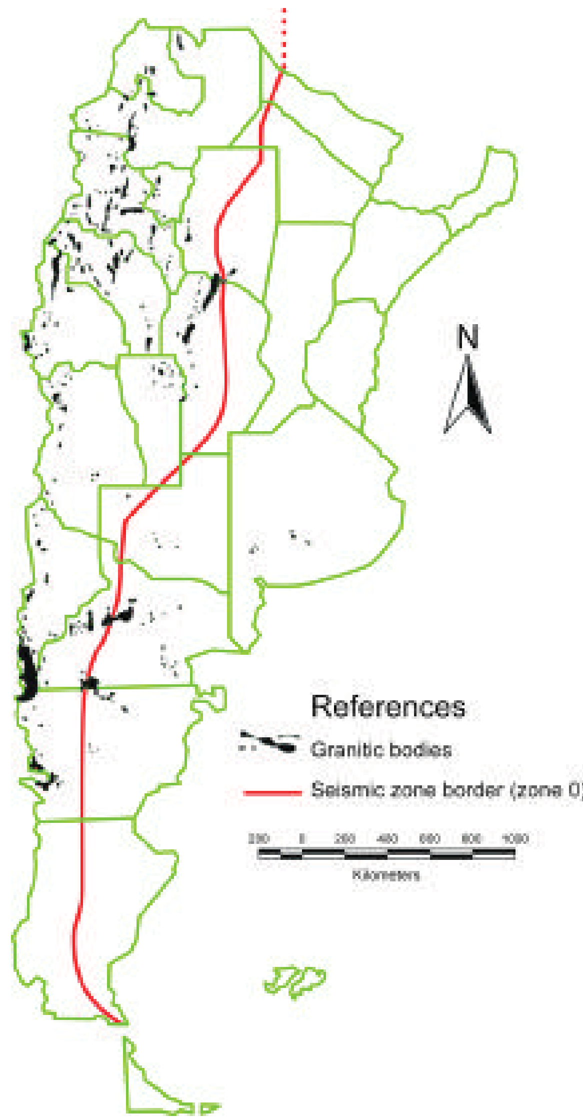

Figure 2.1. Granitic bodies in Argentina (modified from C.N.E.A. [1990])

Sierra del Medio, many geological, geomorphological, hydrogeological, geophysical and seismological studies (among others) were carried out. Furthermore, a drilling program was included for the investigation of the rocky massif (CNEA, 1990). However, in the late 1980s, this study was discontinued due to lack of public acceptance, and consequently CNEAofficially cancelled it in 1992.

In 1991, after the interruption of studies at Sierra del Medio, CNEA started a new research program entitled “Study for Favorable Geological Formations Suitable for the Siting of Repositories for the Disposal of High, Intermediate and Low Level Waste.” Specifically, in

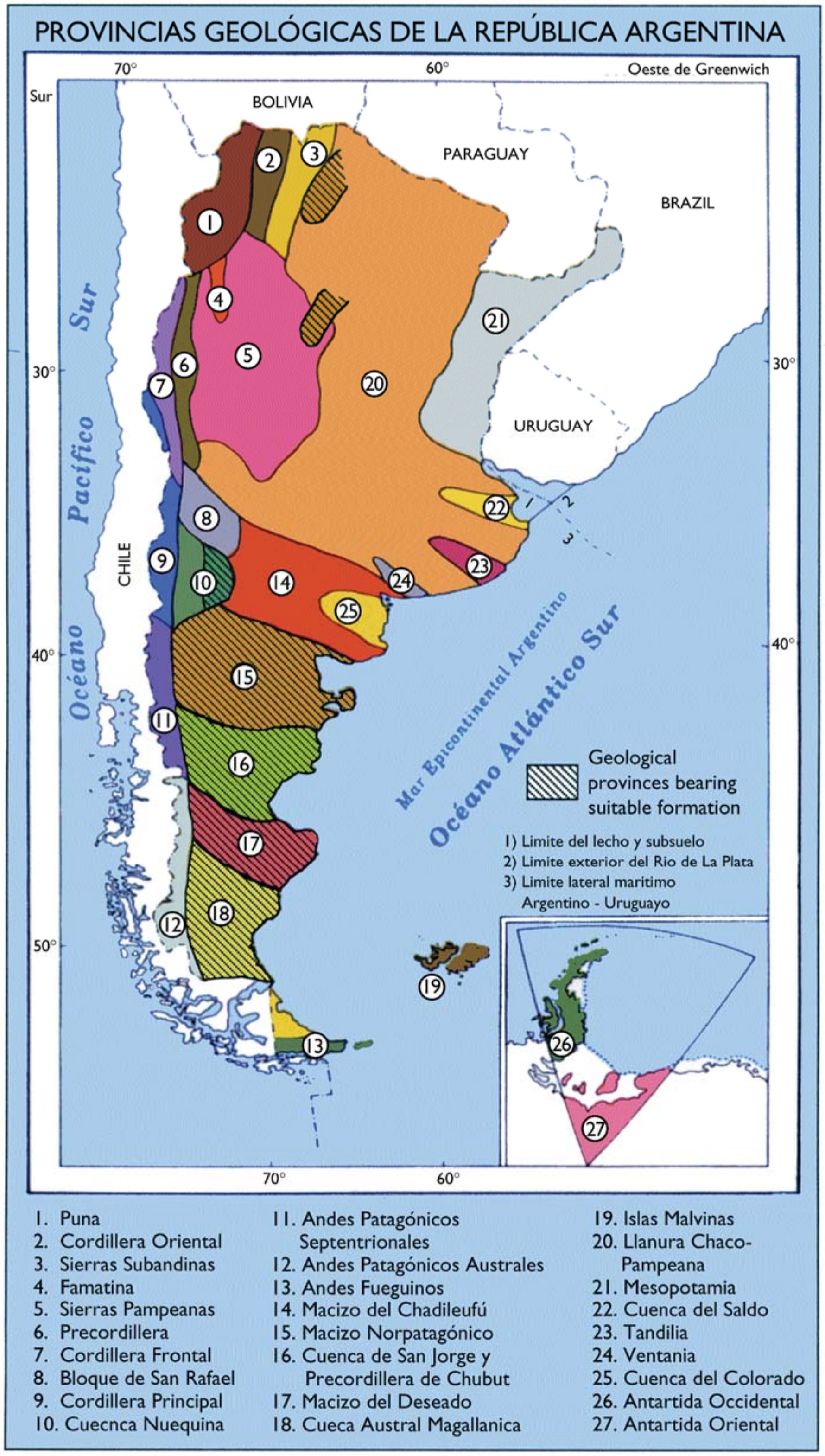

Figure 2.2. Geological provinces bearing potentially suitable sedimentary and volcaniclastic host rock formations

1996, CNEA´s Geology Department began to develop an inventory of prospective areas and to identify rocky formations favorable for deep geological disposal. Concurrently, the International Atomic Energy Agency (IAEA) and CNEAapproved the Technical Cooperation Project entitled "Geology of Repositories for High-Level Waste Disposal" (ARG/4/084). This project, recently completed, included scientific missions of foreign experts to Argentina, as well as the training of CNEA researchers in several countries performing similar studies on deep geological disposal (France, Spain, U.S.A., Belgium, and Canada).

2.2.1. Promising Geological Formations

As part of the new program, and on the basis of the established literature (Cadelli et al., 1988; Roxburgh, 1987; Savage, 1995) and IAEA recommendations (IAEA, 1982), the study of deep geological disposal started in 1996 with a search of promising clay, evaporite, and volcaniclastic formations. The aim was to reach the same level of information obtained in our previous study of granitic rocks.

Argentine territory is divided into several geological provinces, according to its lithostratigraphical characteristics and morphotectonic development in geological history. Many of these geological provinces were analyzed to determine whether such favorable rock formations were present. Several unpublished geological reports document this first approach to the study (Vullien, 1996a–e; 1997a–c; 1998; Maloberti, 1997; 1998; Elena, 1997; 1998; Elena et al., 1996; Ninci, 1998).

Starting from this preliminary evaluation, we considered seven different geological provinces (either completely or partially) containing potentially suitable sedimentary and volcaniclastic formations for further geological analysis. These provinces are shown in Figure 2.2; from north to south, they are:

1. Chaco-Salteña Basin: clay formations
2. Salinas Basin: clay formations
3. Neuquén Basin: evaporite and clay formations
4. Nord-Patagonian Massif: volcanoclastic formations
5. Golfo San Jorge Basin: volcanoclastic formations
6. Deseado Massif: volcanoclastic formations
7. Austral Basin: clay and volcanoclastic formations

2.2.2. Site-Selection Factors (Exclusion Criteria)

Taking into account the IAEArecommendations (IAEA, 1982; 1994), we started gathering basic information and generating maps that depicted the factors considered most important for site selection. The compilation of this information is still in progress. Site-selection factors, also known as exclusion criteria, are to be part of a first screening for suitable areas.

2.2.2.1. Seismicity

Seismic risk must be considered in siting studies. As part of our siting investigation, Argentina's territory has been divided into five seismic zones by using instrumental data available since 1920, as well as historical information (Zamarbide et al., 1978). This is the basic information available. Figure 2.3 shows the last version of this zoning produced by the Instituto Nacional de Prevención Sísmica (National Institute for Seismic Prevention), INPRES. Note, however, that this map was

Figure 2.3. Seismic zones in Argentina

drawn by taking into account the conditions and characteristics of repositories built on the surface. In comparison, a deep geological facility will be safer, given the same seismic intensity levels as determined from surface motions (Dowding et al., 1978).

2.2.2.2. Neotectonism

Seismic instrumental data and historical information are limited over time, and are inadequate for properly assessing the seismic potential of an area and the long-term stability required for a deep geological repository (Wallace, 1986; Vittori et al., 1991). Even if the historical record is accurate, these data and information have minimal value in determining the recurrence time between large seismic events (Crone, 1987). Therefore, neotectonic observations and an adequate paleoseismological record are needed. Currently, the Map and Database of Quaternary Faults and Folds, produced under the Project Major Active Faults of the World (for the International Lithosphere Program), is available (Costa et al., 2000). Furthermore, a map of Argentina showing both tectonically active and stable areas is being prepared as a guide for the deep geological repository siting. Information from this map will be combined with information about Quaternary structures and the regional geological history of the different provinces.

2.2.2.3. Volcanism

Active volcanic areas must be avoided in siting a deep geological repository. Although current volcanic activity is restricted to the Argentina-Chile border area, large Quaternary basaltic lava outcrops are disseminated in some western Argentine states. They will be regarded in a risk analysis. Presently, a map showing these volcanic zones is being prepared.

2.2.2.4. Geothermal Gradient

The only available information is a map constructed by measuring the temperatures from induction-resistivity profiles, registered for over two thousand oil exploitation drillings, disseminated throughout eleven sedimentary basins within Argentina. From these temperature data, isothermal lines indicating "ºC each 100 m" have been drawn (Robles, 1987).

2.2.2.5. Hydrogeology

Currently, the basic available information is the Hydrogeological Map of the Argentine Republic at 1:2,500,000 scale (INCYTH, 1991). The national territory has been divided into four main zones with regard to geomorphological and geological characteristics, porosity, permeability, water chemical properties of the aquifers, and the kind of water discharge. These main regions are:

1. Intermountaneous valleys
2. Chaco-pampeana plain
3. Misiones plateau
4. Patagonian plateaus.

Using this basic information source and shallow hydrological data, we are performing a study concerning the diverse favorability of these regions for the deep geological disposal. Moreover, hydrological information like the fluvial network, shown in Figure 2.4, as well as the precipitation zoning (Atlas de Suelos INTA-Aeroterra S.A., 1995) is also being used for this study.

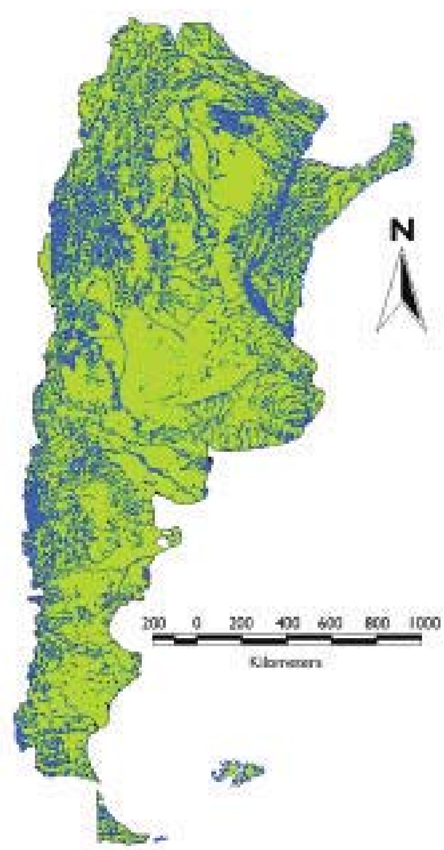

Figure 2.4. Hydrological map of Argentina (modified from Atlas de Suelos de la República Argentina)

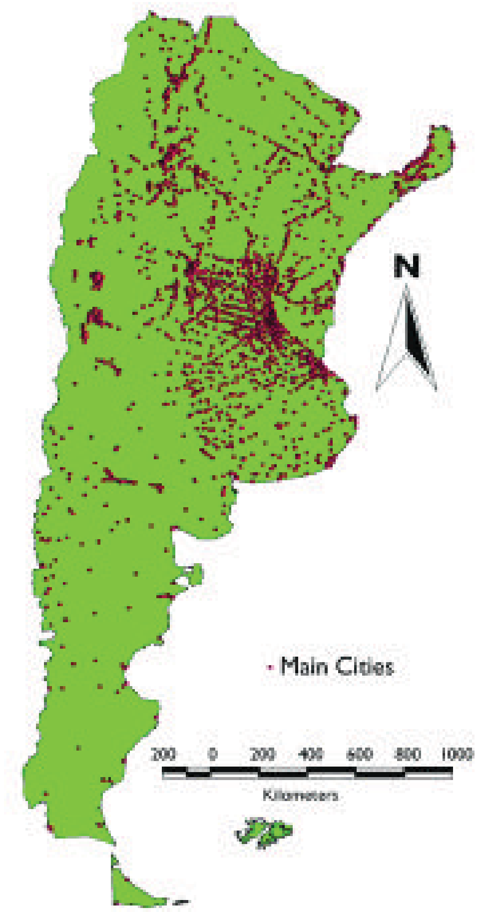

Figure 2.5. Main cities in Argentina (modified from Atlas de Suelos de la República Argentina)

2.2.2.6. Climatic Change

Beyond hydrogeologic considerations, it is deemed likely that in forthcoming centuries, moisture and precipitation would increase in some areas. Some thought has been given to the probability of climatic changes resulting from the greenhouse effect (produced by anthropogenic gases), and their potential influence on future changes of the hydrological regime near a deep geological disposal site (Ninci, 1995).

2.2.2.7. Current Status of the Site-Selection Factors (Exclusion Criteria) Study

Since this is a progress report on the ongoing research performed for deep-geological-disposal site selection, not all the information concerning these selection factors or exclusion criteria is available. To this end, assessment of the consequences or implications of any exclusion/inclusion and the characterization of areas remains to be done.

2.2.3. Development of a Geographical Information System

As a subsequent stage of research, a GIS is being developed to organize all the geoscientific information needed to define suitable areas for deep geological disposal. Because geological information on Argentine territory was not available in digital format, our own GIS had to be built (using the ESRI Arc View 3.2 software). Different kinds of information, on diverse layers, are now being stored in the GIS.

2.2.3.1. Geological Information

This information has been digitally stored using state geological maps already published by the Servicio Geológico y Minero Argentino (Argentine Geological and Mining Survey), SEGEMAR, at 1:500,000 and 1:750,000 scales. These maps have been scanned and logged as files into the GIS. Starting from this basis, the granite, volcaniclastic, and clay rock outcrops were digitized (López, 2000; López et al., 1999; 2001; Reyes et al., 2001a,b). From the available bibliographic information, a database containing specific sampling data (e.g., geochemistry and petrography) has been constructed and integrated into the GIS.

2.2.3.2. Geophysical Information

The seismic zones of Argentina (Zamarbide et al., 1978) supplied by INPRES have been digitally logged and stored as files in the GIS (López, 1999).

2.2.3.3. Geographical Information

The files for diverse layers of information about roads, railways, climate, hydrology, etc. (shown in Figure 2.4), as well as the main cities (shown in Figure 2.5), have been logged into the GIS (Atlas de Suelos INTA-Aeroterra S.A., 1995; López, 2000).

2.2.3.4. Current GIS Organization Stage

More information remains to be added as new layers to the GIS. This task will be completed as more data on different site-selection features are obtained. Furthermore, different GIS spatial analysis methodologies (such as fuzzy logic, weights of evidence, and neural networks) have been used by applying site-selection factors to specific territories selected as key areas (López, 2000). This kind of modeling provides a new tool for the evaluation and definition of areas and sites, and will be scaled to the entire national territory.

2.2.4. Future Geological Research Tasks

It will be necessary to increase knowledge of site-selection factors described above, so that they can be effectively used as area exclusion/inclusion or subsequent qualification tools. While the first research stage is at present being completed, further methodologies are being applied to obtaining useful geological data for key areas. A program for digital processing and geological interpretation of Landsat TM satellital images is underway, with the principal aim of mapping fractures and structural lineaments at a 1:500,000 scale, as well as detecting neotectonic features.

2.3. SOCIAL COMMUNICATION

2.3.1. The Lack of Public Acceptance

The cancellation in the early 1990s of the "Feasibility Study and Engineering Project—Repository for High-Level Waste" resulted directly from a lack of communication with the public.

Politicians and decision makers from diverse management levels of the national, state, and municipal government were subject to a great deal of pressure from different social groups to stop the ongoing studies at Sierra del Medio. These groups lacked information about the subject and reacted fearfully to it. This fear was further exploited by groups with interests opposed to nuclear energy. Together, they compelled decision makers to discontinue research on deep geological disposal and thus defeat the project.

Consequently, within the new program, entitled "Study for Favorable Geological Formations Suitable for the Siting of Repositories for the Disposal of High, Intermediate and Low Level Waste," attention will be paid to those factors that caused the cancellation of the former studies, and effective social communication activities will be deployed to allow the current program to progress. These communication activities will be emphasized in a social communication program that requires the involvement and training of a large group of scientists and technologists.

2.3.2. Rejection Cause Analysis

The acceptance of nuclear power by the general public is related to allaying fears about its potential hazards or dangers. People's perceptions cannot be ignored. A significant portion of the public lacks extensive knowledge about nuclear energy. Given this lack of knowledge, people react out of fear, having read about, heard about, or seen on television the disasters associated with nuclear accidents. These images work to make people reject anything that is associated with nuclear power, including socially beneficial projects that have to do (for example) with keeping nuclear waste isolated from the population. Because of a lack of information, the public can reject what could be in its best interest to endorse. Given that the public can greatly influence nuclear policy, the task is then to educate the public about nuclear power, so that it will make informed (rather than irrational) judgments about this issue.

As Figure 2.6 shows, the structure of society according to its perception of the nuclear issue is composed of four main groups:

A. The general public
B. The politicians (and decision makers)
C. The informed public
D. The nuclear technology area participants.

A great responsibility lies with Group D, since this group is the one that must effectively communicate with and educate the other groups regarding this issue.

2.3.2.1. Group A: The General Public

The general public can be defined as a group with negative perceptions of nuclear power, including the siting of a deep geological repository. Lacking in information about nuclear power, this group can be overly influenced by media sensationalism of the nuclear industry's past shortcomings.

2.3.2.2. Group B: The Politicians

Politicians and decision makers can be considered as generally an informed group, one which forged the essential legislation regarding spent fuel and radioactive waste management. This group also created an independent regulatory authority, the "Autoridad Regulatoria Nuclear" ("Nuclear Regulatory Authority") (ARN) through national laws and an international treaty (the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management) approved by a national law [Constitución Nacional (1994), Ley 24.084 (1997), Ley 25.018 (1998), Ley 25.279 (2000)]. Furthermore, this group

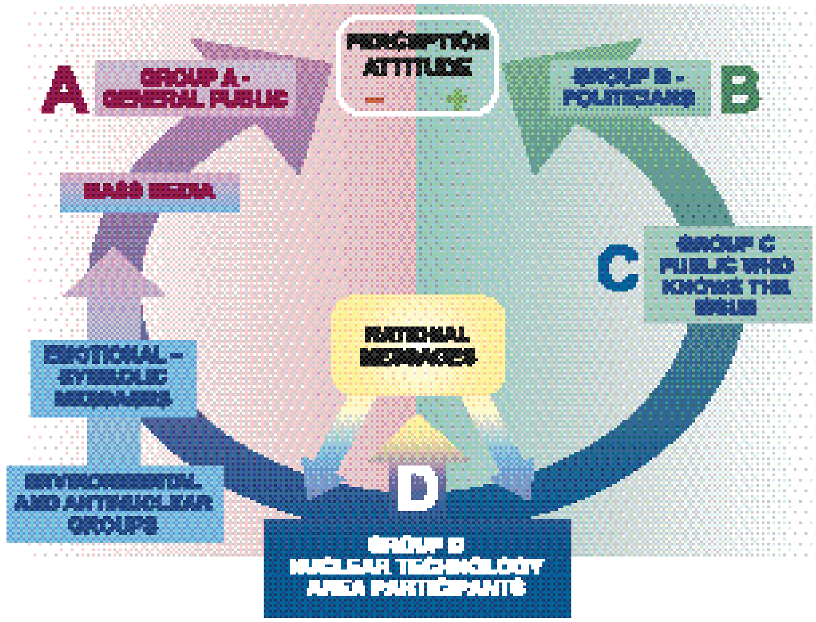

Figure 2.6. Structure of the society according to the perception of the nuclear issue

developed the legal framework reflected in the regulatory standards issued by ARN, particularly the standard entitled "Gestión de Residuos Radiactivos" ("Radioactive Waste Management") [Norma AR 10.12.1, 2000].

Despite all this constructive legislation, some members of this group, in response to public opinion, did introduce and support adverse legislation. In general, however, this group positively perceives the nuclear issue, although its actions are strongly conditioned by Group A due to political and electoral necessity.

2.3.2.3. Group C: The Informed Public

The informed public constitutes a group that generally perceives the nuclear issue in a positive way, with few exceptions, but it has little or no influence in modifying the climate of negative perception created by Group A.

2.3.2.4. Group D: The Nuclear Technology Participants

We should emphasize the responsibility of this group, the nuclear technology participants, with regard to the other groups. The responsibility belongs to Group D as a whole (not to a part of it) to educate and inform Group C first and then Group A about nuclear power.

2.3.2.5. Mass Media and Opinion Makers

The mass media can be matched to Group B (the politicians). However, there is an essential difference: their mission is not to make decisions for the welfare of the community but to inform. Furthermore, it is worth noting that science and technology news does not generally get significant coverage from the mass media.

2.3.3. The Way to the Solution

The solution to this situation will require mass media presentation of the nuclear issue, including the siting of a deep geological disposal facility, in a social communication program that succeeds in modifying the perception of Group A. This, in turn, will condition the actions of Group B.

From the legislative point of view, the "Radioactive Waste Management Regime" passed by the Argentine Parliament in 1998 (Ley 25.018, 1998), requires the formulation and deployment of the "Strategic Plan for the Management of Radioactive Waste." This plan includes both deep geological disposal research activities and a social communication program.

To fulfill the statement of the "Radioactive Waste Management Regime" (Ley 25.018, 1998), the staff of

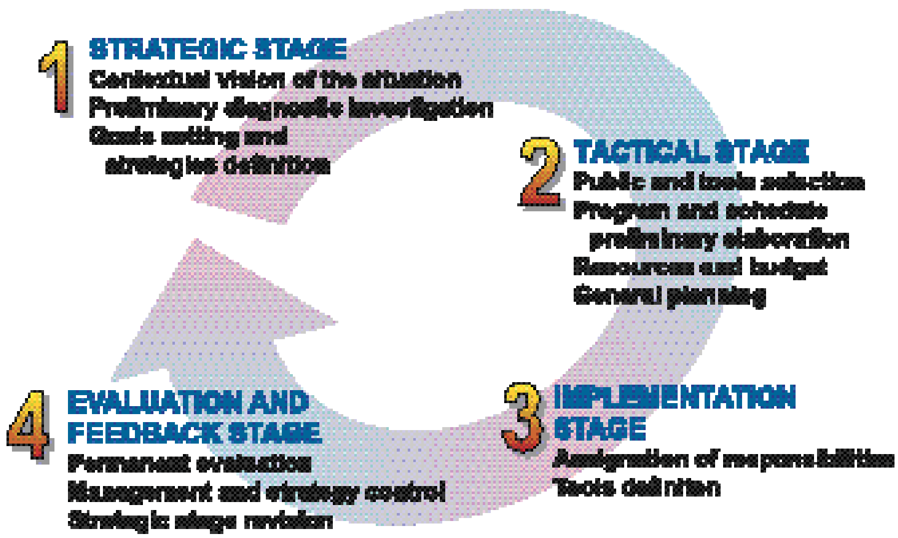

Figure 2.7. Basic structure of the Social Communication Program

CNEA has issued (and the President of CNEA has approved) the "Strategic Plan for the Management of Radioactive Waste" generated in Argentina (Plan Estratégico GRR, 2001). This strategic plan has been submitted to the President who, after consultation with the ARN, will forward it to the Parliament for debate to pass it as a national law. The strategic plan must be reviewed and updated every three years.

As long as social and political consensus is needed for selection and approval for siting of a deep geological disposal facility, the strategic plan will involve a social communication program to inform society about the scientific and technological features of spent fuel and radioactive waste management. The program must provide clear and objective information, allowing society to know the scope of established activities, as well as the direct and indirect social benefits resulting from a deep geological repository. Clearly, the way the mass media goes about informing society about nuclear power has huge importance, because of the influence the media exerts on public opinion and acceptance.

In addition, it is important to reach out to society through a permanent communication link with its national, state, and municipal representatives. Group D and other proponents of nuclear power also need to reach out to additional influential opinion makers among nongovernmental organizations (NGOs), the business community, professional associations, and communal and neighborhood organizations.

2.3.3.1. Basic Elaboration of the Social Communication Program

The social communication program will require defining the situation, adopting strategies, and then, during its implementation, constant refinement of these strategies (leading to corrective actions). Figure 2.7 shows the four stages (Strategy, Tactics, Implementation, and Evaluation and Feedback), with their corresponding phases, of the proposed social communication program (Maset et al., 2000; Jolivet, 2000).

2.4. CONCLUSION: GEOLOGICAL RESEARCH WITHOUT SOCIAL COMMUNICATION

The history of geological research into deep geological disposal in Argentina demonstrates the strong necessity for social communication. The cancellation by CNEAof the "Feasibility Study and Engineering Project—Repository for High-Level Waste" in the early 1990s is the most conclusive evidence of this necessity.

Because of the existing international consensus on the suitability of geological disposal, Argentina will proceed with geological research on the subject. However, as long as the social communication program to be deployed does not succeed in changing public opinion, the research on deep geological disposal (as part of the "Study for Favorable Geological Formations Suitable for the Siting of Repositories for the Disposal of High, Intermediate and Low Level Waste") that has been carried out since 1996 will not include field studies aimed at site selection (Bevilacqua, 2000; 2001). Under the current conditions, geological research will be limited to national-scale desk studies, assessing the existing information, and developing site-selection factors or exclusion criteria such as seismicity, neotectonism, volcanism, and hydrogeology. International cooperation, however, is being considered to accomplish field studies abroad for training in site-selection or exclusion-criteria development.

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Chapter 3

Some Geological Aspects of Underground Disposal of Radioactive Waste in the Republic of Armenia

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ABSTRACT . In this report, we state the problem of long-term and safe isolation of radioactive waste in geological media within the Republic of Armenia, and present some possibilities for its solution. Preliminary results from comprehensive analyses of geologic information for the region are presented. Geologic-tectonic, hydrologic-hydrogeologic, and other conditions related to the suitability of the media in the territory of the Republic are evaluated. Based on the analysis of the available databases on regional geology, several areas with relatively favorable conditions, as well as several potential sites having first priority for further investigation, are discussed. Also, multistep, complex research investigations needed for selecting potential sites and characterizing them for underground radioactive waste disposal are recommended.

3.1. INTRODUCTION

The Territory of the Republic of Armenia is located within the southern segment of the Caucasus and the northeastern region of the Armenian uplands. The area of the Republic is 29,800 km^2 and is significantly subdivided by the watersheds of the Kura and Araks rivers, which are the two main streams in the Lesser Caucasus. These rivers are in the basin of the Caspian Sea. Elevations range from 400 m to 4,090 m; about 90% of the territory of the Republic is located above an elevation of 1,000 m above sea level. The climate is dry and continental.

Before the construction of the Armenian Nuclear Power Plant (NPP), electric power was supplied from hydroelectric and thermal-electric power plants. The economic development of the country at the end of the 1960s led to a significant deficit in electric power, which motivated the need for the design and construction of the Armenian NPP, the only one in the region beyond the Transcaucasus. Along with the exploitation of the Armenian NPP and the accumulation of an increasing volume of radioactive waste, the problem of long-term and safe isolation of waste was created. In addition to the Armenian NPP, sources of radioactive waste in Armenia are medical, commercial, and some other enterprises that use radioactive materials. However, among all these sources, that portion of radioactive waste created by the operation of the Armenian NPP is predominant and continues to grow.

The Armenian NPP started in December 1976. After 12 years of continuous operation, it shut down in 1989 after the catastrophic Spitak's earthquake (7.1 on the Richter scale) on December 7, 1988. Based on the results of a wide range of additional investigations to evaluate the safety of geological conditions in the region and the internal systems within the NPP, its operations were resumed at the end of 1995. Currently, the second unit (one of two reactors) is operating. The NPPincludes two nuclear reactors of the WWER-400 type, as manufactured in the Russian Federation, each with a power output of 420 MW.

Before the operation of the Armenian NPPstopped and after operations resumed, the sources of electric power in Armenia (based on information from the Ministry of Energy of Armenia) are given in Table 3.1.

All radioactive wastes from the Armenian NPPand other sources are temporarily stored in special tanks located in isolated areas within and outside the NPP site. Before 1995, the spent fuel from reactors was sent to Russia for

Table 3.1. Sources of electric power in Armenia (in billion KWh)

	1988	1996	2000
Nuclear Power	4,810 (31.5%)	2,324 (39%)	2,034 (34%)
Thermal Electric	8,950 (58.5%)	2,318 (39%)	2,695 (45%)
Hydro Electric	1,534 (10.0%)	1,568 (22%)	1,262 (21%)
Totals	**15,294**	**6,210**	**5,991**

Table 3.2. Summary of types and quantities of radioactive wastes (RW)

Types of RW	Total Volume of Storage, m^3	Accumulated Waste m^3	metric tons	Specific Activity Bq/kg/l
SOLID WASTE*				
Low activity	17,301.6	4,288	9,164	7.4E4 – 3.7E6
Intermediate	1,374.0	1,450	1,885	3.7E6 – 3.7E9
High	147.0	39.21	50.97	>3.7E9
LIQUID WASTE				
Ion exchange tar		300.	390	2.6E5
Radioactive oil		50.	39	<3.7E4
Distilled residue, salt content 300g/l		2,211	2,874	Cs^{134} – 4.0E4 Cs^{137} – 1.9E6 Co^{60} – 1.1E5
High activity liquids		332	418.6	Cs^{134} – 3.8E4 Cs^{137} – 1.2E6 Co^{60} – 5.7E4
Residual salt after evaporation		878.7	1,142.3	Cs^{134} – 1.2E5 Cs^{137} – 3.3E6 Co^{60} – 1.6E5

** Isotopic composition of solid waste not determined.*

further handling; since 1995, the spent fuel has been stored in basins at the NPP. Since this can only be a temporary solution, finding a location within Armenia for safe underground disposal of radioactive waste and spent fuel in geologic media is very important.

3.2. GENERAL CHARACTERISTICS OF RADIO ACTIVE WASTES

As mentioned above, the main source of radioactive waste in Armenia is the Armenian Nuclear Power Plant (NPP), which has produced an ever-increasing amount of waste from different activities since 1976. Different kinds of radioactive waste are generated from medical, commercial, research, and other such enterprises. The characteristics given below on all radioactive wastes of Armenia and spent fuel from NPPare based on data provided by the Armenian Nuclear Regulatory Authority (ANRA) in 1998.

Collection, recycling, and storage of waste in Armenia are carried out by two organizations:

1. Special Enterprise (formerly Combine "Radon"). This enterprise deals with radioactive waste from medical facilities, commercial organizations, and other users of radioactive materials. Since 1982, for the period of its operation, it disposed and stored 11,058 units of radioactive sources and packages with radioactive waste of the following isotopic composition: ^{226}Ra, ^{132}Cs, ^{14}C, ^{90}Sr, ^{192}Ir, ^{170}Tu, ^{239}Pu, ^{147}Am, ^{131}I, ^{198}Au, ^{32}P, ^{75}Se, ^{3}H, ^{22}Na, ^{119}Pb, ^{99}Te, and ^{109}Ku. These elements are almost all intermediate- to low-level waste (LLW) (elements of high activity are less than 1%).
2. Armenian NPP. The NPPdeals with waste accumulated during its operation. Table 3.2 summarizes the quantity and composition accumulated at the NPP.

Storage facilities for intermediate- and high-level solid and liquid wastes (ILW and HLW) are located within the NPP, and storage for LLW is located outside the NPP. The total amount of nuclear fuel at the Armenian NPPis 1,073 units. Currently at the NPP, an aboveground module-type facility was constructed (based on a design by FRAMATOM) for dry storage of spent fuel. Fifty-six cassettes can be stored in each module.

3.3. STATEMENT OF PROBLEM AND APPROACHES TO ITS SOLUTION

Although urgently needed, the national program for long-term and safe isolation of radioactive waste in geologic media in Armenia has only recently begun to be developed. In 1985, the State Committee on Atomic Energy Utilization of the USSR developed a draft proposal entitled "Scientific Technical Justification of Need and Possibilities for Disposal of Wastes from the Armenian NPP in Underground Salt Structures." However, in this proposal, the standard methods and criteria for selecting and characterizing waste sites were not taken into consideration. For this reason, the proposal was rejected by the Academy of Sciences of Armenian SSR. This problem became the subject of detailed discussions after Armenia became a member of the International Atomic Energy Agency (IAEA) in Vienna.

ANRA is the only regulatory organization that initiates measures for the management of radioactive waste disposal. The Institute of Geological Sciences of the National Academy of Sciences of Armenia (NASA), in collaboration with ANRA, initiated an investigation of the geological aspects of radioactive waste disposal. In particular, the authors of this paper presented a report (Djrbashyan and Ghukasyan, 1998) at the International Seminar FTAM in Berkeley in December 1997. This report presented several possible preliminary approaches to the selection of potential sites, based on geologic-hydrogeologic criteria. A review of the material from this seminar (with corresponding conclusions) was published in the scientific journal *Izvestiya of the National Academy of Sciences of Armenia* in 1998 (Ghukasyan, 1998a). Also in 1998, a special presentation aimed at the necessity for the development of a national program for radioactive waste disposal, and the creation of organizations capable of carrying out work on the problem, was made at the Republican Conference (Ghukasyan, 1998b) held by the Ministry of Ecology and NASA.

3.4. METHODS OF SELECTING APPROPRIATE AREAS AND POTENTIAL SITES

In the *Second Worldwide Review* (Witherspoon, ed., 1996) different countries presented various methods for selecting sites providing long-term and safe isolation of radioactive waste. Some differences existed in the methods used for different waste-storage concepts, depending on the depth of geological formations within the various countries. Among the different methods, the most widely used were deep (geological) and near-surface types of storage. Selecting a specific kind of storage depends on a variety of factors, including the type, condition, composition, and category of radioactive waste. The geologic-geomorphologic and structural tectonic conditions of the region play an important role in this selection process as well.

In Armenia's case, storage-concept selection has not been carried out sufficiently. Among the different types of radioactive waste in Armenia, significant volumes of long-lived radionuclides have an intermediate level of radiation, which are recommended to be disposed of as HLW in a geologic repository (IAEA, 1981; IAEA, 1995) Taking into account the idea that spent fuel is considered by many researchers as HLW (Savage, ed., 1995; IAEA, 1995), spent fuel from reactors of the Armenian NPP should also be considered HLW and be disposed. It follows that the most appropriate option for the disposal of radioactive waste in Armenia is deep geologic burial (though other options cannot be excluded at this time). Since the approaches for selecting appropriate areas and sites for each type of repository are almost the same for any disposal option (at least at the current [preliminary] stage of investigations), an analysis of geologic conditions was carried out.

Deep repositories are planned to be located at depths from 50 m to several hundred meters (sometimes 1,000 m and deeper), while near-surface repositories are planned for depths down to 50 m from the surface (IAEA, 1994a; 1994b; 1995; Savage, ed., 1995).

3.4.1. Database Used in Conducting Research

A significant number of literature sources, cartographic sources, and other material (including complex information on geology, geography, and other natural sciences in the territory of the Republic and adjacent regions) was used in organizing databases. The most important graphical materials representing the whole territory of the Republic are:

Geologic map (scale 1: 600,000)
Lithologic map (1: 500,000)
Tectonic map (1: 600,000)
Surface relief map: (1: 600,000)
Seismotectonic map: (1: 500,000)
Seismic zonation map (1: 500,000)
Hydrogeologic map (1: 500,000)
Geographic atlas (1: 1,000,000 and 1,500,000)
Suite of air-space photographs and topographic maps of different scales.

In addition to these graphic materials, large-scale geologic maps, air-magnetic maps and other geophysical anomalies were used. Materials on geology, geomorphology, geophysics, volcanology, hydrogeology, and seismotectonics of the region were included in several monographs and fundamental publications, in particular the multivolume edition *Geology of Armenian SSR, Regional Geology of Armenia, and Seismotectonics of Armenian SSR.*

The results of a critical analysis as well as a number of other special investigations were used as a basis for this report. This material can also be used for carrying out complex investigations in the subsequent stages of investigation.

3.4.2. Main Criteria in the Preliminary Selection of Appropriate Areas and Potential Sites

In selecting appropriate areas and potential sites, we used recommendations and technical documentation from the IAEA (IAEA, 1981; 1994a; 1994b) and other materials, taking into account the amount of experience that different countries had with this problem (Witherspoon, ed., 1996; Savage, ed., 1995). In this work, we particularly took into account the geologic-tectonic characteristics of our region. Among many criteria recommended by IAEA, as a first order, we used geological and hydrogeological conditions; seismicity, seismotectonics and active fault tectonics; recent volcanic activity; composition, character, and type of formations; and rocks of possible repositories. Even more important criteria for this process are the following specifics for our region: high seismic activity (8-9+, MSK-64), high elevation and intensity of drainage subdivisions, blocks and mosaic-microblock structure of the earth's crust, active fault tectonics and geodynamics, recent (N^3_2-Q) volcanic activity, a wide variety of rocks based on their composition and genesis, etc. Important ecological and socio-economic factors should also be considered.

In evaluating the geological media of the region, the main criteria listed above have not always been used in an equivalent manner. Depending on the specific situation, a priority was given to one or another group of criteria. The factors listed above make the solution of the problem much more complex.

3.5. GEODYNAMIC CONDITIONS AND THE MAIN CHARACTERISTICS OF TECTONICS IN THE REGION

The territory of Armenia is located in the northeastern part of the Armenian Upland and includes the southern regions of the Caucasus segment of the Alp-Himalayan folded belt. The area is an intracontinental, geodynamically active, heterogeneous mountain structure, known as the Lesser Caucasus.

Armenia's geologic history reflects the main regional-formation stages under the conditions of interacting Euro-Asian and African-Arabian continents and contraction of the ocean Tethys with adsorption of the oceanic crust. As a result, the continental blocks of the basin foundation were welded, with the Transcaucasus intermediate massif of the north and the Armenian block of the south. Geologic conditions of the territory are characterized by significant stratigraphic differences and thicknesses of layers, as well as tectonic structures and broad development of deep magmatic and volcanic processes.

Different authors describe separate tectonic zones within the territory (Paffengolts, 1959; Gabrielyan et al., 1968; Aslanyan, ed., 1970; Gabrielyan et al., 1981). The northwestern part of Armenia, which is within the Somkheto-Karabakh zone, is made up of Jurrassic-Cretaceous, marine Carboniferous, and a volcanogenic-terrigeneous type of sedimentation, as well as a broad development of island-arc subduction volcanism of a basalt-rhyolite series. The Somkheto-Karabakh paleo-island arc has a fragmented structure, developed on a foundation of Hercynian consolidation (Melkonyan, 1989).

At the beginning of the Alpian stage (late Cretaceous), the convergence of the Euro-Asian and Iranian-Arabian plates led to a complete absorption of the oceanic crust of Tethys and to the formation of the heterogeneous blocks of the continental crust. This epoch marks the conclusion of the subduction and the beginning of complex "continent-continent"-type collisions.

The central and southern parts of the region (Armenian block) are located at the basement of the Iranian plate of the Baikalian stage. These locations are characterized by the development of local bending flexures with the accumulation of flysch, volcanogen-flysch, terrigenous, and molasse formations. Intrusive-subvolcanic and effusive-explosive facies of magmatic processes were intensively developed. Basalt-trachite, an andesite-rhyolite-phonolite volcanic series, and intrusive complexes of the gabbro-monzonite-granodiorite and gabbro-silenite series were developed (Meliksetyan, 1989; Djrbashyan, 1990).

The impacts of Neogenic and particularly Pliocene-Quaternary volcanism developed within this block were most significant in the southern Caucasus. Large polygenetic volcanoes of the central type (Aragats, Arailer, Ishkhanasar) and many other (~550) areal-monogenetic volcanoes produced flows that covered 60% of the Republic's territory.

The boundary between the two above-described large structural units (continental blocks) is the Sevan seam zone. This zone is evident from the number of exposures of ophyolite complexes. Paleogenic volcanogenic-flysch bending flexures and thick effusive-extrusive-intrusive complexes were formed. The far southwestern part of Armenia is called the Near-Araksian zone, featuring a well-developed sedimentary complex from Devonian to Neogene in age.

The thickness of the earth's crust for this region, based on geophysical investigations, varies in general from 48 to 55 km; the thickness of the sedimentary cover does not exceed 8 km (Mkrtchyan, ed., 1972).

3.6. MAPPING OF AREAS WITH APPROPRIATE CONDITIONS AND SELECTION OF POTENTIAL SITES

The selection of prospective regions and appropriate sites is based on a wide complex of geologic investigations. These are, indeed, preliminary and should not be considered a final solution to radioactive waste disposal.

3.6.1. Structural Tectonic Conditions as the Basis for Selecting Areas

The analysis of the structural tectonic conditions of Armenia is one of the most important factors in an investigation of sites suitable for underground isolation of radioactive waste. In this report, we generalize from numerous schematics and maps of the tectonics of Armenia developed by different authors (Gabrielyan et al., 1968; Aslanyan, ed., 1970).

Based on the age of the Pre-Alpian basement, the region is subdivided into two continental mega-blocks: the Transcaucasus intermediate massif in the north and the Armenian block in the south. Based on the complex historic-geologic and structural-formation indicators, four structural formational zones can be recognized in Armenia (Gabrielyan et al., 1968; Aslanyan, ed., 1970; Gabrielyan et al., 1981). These zones are graphically depicted on the generalized tectonic scheme (Figure 3.1). A brief description of these zones and a comparative analysis to be used in the following preliminary examination of the territory are given below.

1. The Somkheto-Karabakh zone is located in the southern part of the Transcaucasus intermediate massif, which is a Mesozoic island-arc block on the Hertzian crystalline basement.

   From the structural point of view, this zone is a large mega-anticlinorium consisting of a number of anticlinoria and sinclinoria attached to each other along faults. This zone is characterized by relatively weak current tectonic movements, medium seismicity (up to 8, MSK-64), almost no Quaternary volcanics (excluding the Lory subzone), an undisturbed magnetic field, and a low rate of heat transfer. From the point of view of the latest tectonics, this zone presents a one-directional hanging block that is gradually sinking toward the Kurinsian intermontaine depression.

   It is important to emphasize that this zone has been derived directly from large Mesozoic structures, which experienced a mild but stable elevation. From this point of view, this region can be considered a relatively more suitable area for underground isolation of radioactive waste.

2. The basement of the Central folded zone of Armenia (Armenian block) is located on the Baikal

Figure 3.1. Scheme of tectonic mapping for Armenia.
Legend: (1) Somkheto-Karabakh zone , (2) Sevan-seam zone , (3) Armenian central folded zone , (4) zone of Near-Araksian (depression of the Araks River)

folded foundation of the Iranian plate. The block includes the Bazum-Zangezur intensively folded zone of Early Alpian age and Near-Araksian slightly folded zone of Middle Alpian age. The boundaries of these zones are regional deep faults. In general, these zones are characterized by the high contrast of the latest tectonic movements, with their sharply diverging domal blocks, medium and high seismicity (8-9+, MSK-64), the (N32 –Q) volcanism, highly anomalous magnetic fields, and a relatively high rate of heat transfer.

Some areas located within this complex and diverse tectonic zone can be considered prospects for the isolation of radioactive waste. These sites are located in the western, southern, and far southeastern parts of Armenia, in areas having relatively suitable structural-geodynamic conditions.

3. The Sevan-seam zone is a boundary between the geotectonic zones described above (Somkheto-Karabakh and Armenian block). This zone is characterized by the presence of the ophyolite complex,

highly divergent new tectonic movements, a high level of seismicity (9+, MSK-64), a highly anomalous magnetic field, and a high rate of heat transfer. These characteristics indicate that research investigations should not be carried out in this zone.

4. The zone of Near-Araksian (depression of Araks River) is located on the southern edge of the Armenian block, at the edge of the Iranian Continental Plate. The zone is characterized by Late Alp folding with the Baikal platform type foundation in the basement. This zone is, in general, characterized by a mosaic-block type of structure, intensive current movements, high seismicity, and the presence of several underground artesian basins. Based on the structural tectonic and geodynamic characteristics, as well as inappropriate geologic-hydrogeologic conditions, this zone is not suitable for the problem at hand.

3.6.2. Selecting Areas with Appropriate Condition

Based on preliminary analysis, the Armenian territory has been subdivided into several areas of different categories. Figure 3.2 is a schematic map showing the distribution of selected areas. This map also shows the current zones of seismicity based on seismic mapping (Karapetyan et al., 1995), as well as the locations of the main active faults of the Pleistocene-Holocene age (Karakhanyan, 1993; Karakhanyan et al., 1996; Djrbashyan et al., 1998). There are three categories according to the degree of the appropriateness of isolation of radioactive waste: (1) inappropriate areas; (2) areas selected for additional investigations; and (3) areas with relatively appropriate conditions (requiring more detailed investigation).

1. Based on an overall evaluation of the principal criteria, the main part of the territory is not suitable, as is depicted in Figure 3.2. If we take into account some basic geological screening criteria (high seismicity, active tectonics, recent volcanism, hydrological and hydrogeological conditions) recommended by IAEA and used in many countries, we can determine that these areas are inappropriate for the final isolation of radioactive waste. This was determined using th following:

   - Unsuitable hydrological-hydrogeological conditions (inundated areas, natural and artificial reservoirs of potable and irrigation waters, flood plain zones of permanent and temporary creeks, artesian basins, shallow groundwater).
   - Unsuitable tectonic and neotectonic conditions (unstable structures, strong and differently oriented current movements).
   - Unsuitable geological conditions (unsuitable formations).
   - Seismicity (≥9, MSK-64, A > 0.34 g).
   - Volcanism (Holocene activity, long-term recent activity).
   - Active faults (current, zone < 6 km).
   - Geothermal regions (high values of thermal gradient).
   - Ore bodies (under exploitation or prospecting).
   - National parks (historical, architectural, and other monuments).
   - Urbanized regions (high population, cities with more than 5,000 inhabitants).
   - Socio-economic conditions (large energy sources, commercial enterprises, communication lines, military bases, etc.).

2. Areas selected for additional special investigation to evaluate the possibility of their use in constructing a potential repository occupy a significant part of the territory.

   These areas include geological media with young (N^3_2-Q) volcanic formations. These formations occupy middle- and high-mountain slopes of polygenic volcanoes and volcanic uplands. Within the distribution of these zones, there are many volcanic pipes and faults (characterized by subsequent activity with respect to those on the periphery plateaus), which are magma channels of volcanic structures. Based on geodynamic conditions and magmatic geology, these areas are not appropriate for the current task. However, to obtain reliable data and justified conclusions, we recommend carrying out special volcanological and geophysical investigations. These investigations should be directed mainly to the evaluation of the geothermal field, amount and intensity of the deep thermal resources, and the potential for new volcanic activity.

3. Areas with relatively suitable conditions, which need additional investigations are located in the northern, southern, and western parts of Armenia (Figure 3.2). Based on the main criteria, nine areas can be selected, five of which are located in

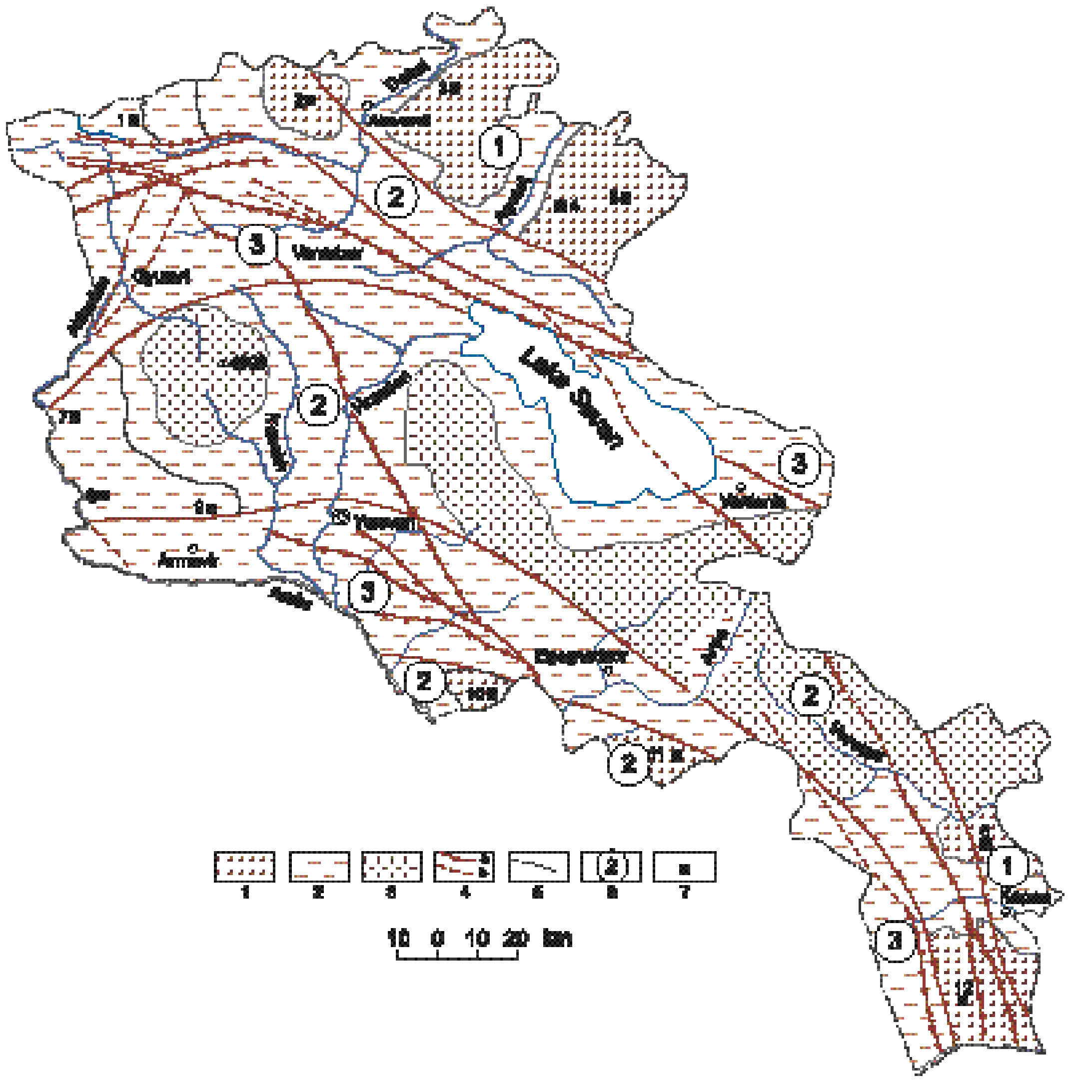

Figure 3.2. Scheme of preliminary selected prospective areas and potentials sites.
Legend:(1) Areas with appropriate conditions,(2) Inappropriate areas,(3) Areas selected for additional investigations,(4) Traces of main Pleistocene-Holocene active faults: (a) *indisputable,* (b) *assumed*, **(5) Boundaries of zones of different seismicity, (6) Seismicity zones (numbers circled in figure):** *1. zone of mild seismicity, Amax = 0.2 g, V = 16 cm/s, <8, MSK-64, 2. zone of average seismicity, Amax = 0.3 g, V = 24 cm/s, 8-9, MSK-64, 3. zone of high seismicity, Amax = 0.4 g, V = 32 cm/s, ≥9, MSK-64,* **(7) Potential sites (1-12).**

Somkheto-Kharabakh and four in the central-Armenian folded zones.

The westernmost area of the first tectonic zone is located in the Lori monoclinal subzone, within the Kechut volcanic-tectonic block (with seismicity 8-9, MSK-64). Three other northern areas are located in the Alaverdi-Shamshadin subzone within the same-named anticlinoriums (with an 8 and 8–9 seismic rating, MSK-64), with relatively mild seismotectonic conditions. The last prospective site in the southern area is located within the Kapan anticlinorium (with basically a seismicity of 8, MSK-64).

The relatively favorable areas selected within the Armenian block are located within the 8-9, MSK-64 seismic zone, with relatively mild geodynamics. The western area is located on the western periphery of the Aragats volcanic-tectonic block, within the Pliocene-Quaternary monogenic volcanoes.

Based on a preliminary assessment (Ghukasyan and Shirinyan, 1998), these volcanoes will not present hazardous volcanism in the near future. The area is characterized by arid, semi-desert climatic conditions. Two southern areas are located within the Yerevan-Ordubad structural subzone, located at the Urts-Aiotsdzor anticlinorium, which is characterized by arid, desert conditions and appropriate geological formations of PreAlp and MiddleAlp consolidation. The latter, the most southern area, is located in the Ankavan-Zangezur subzone within the isolated tectonic-magmatic block, as part of the Megrinian pluton.

3.6.3. Selection and Characteristics of Potential Sites

Within the selected favorable areas, we can preliminarily consider 12 potential sites with more suitable parameters. The most important parameters are seismicity, newest fault tectonics, structural types, geomorphological conditions, types of geologic formations, content and genesis of bedrock, and hydrologic-hydrogeologic conditions.

Figure 3.3 shows a generalized schematic geological map of Armenia, scale 1:1,000,000, prepared using regional geological maps at scales of 1:500,000 and 1:600,000, as well as more detailed maps of individual regions. This map shows approximate locations of the selected potential sites. As a result of the analysis of available databases, the locations of several sites were slightly relocated relative to those selected previously, based on hydrologic-hydrogeologic conditions (Djrbashyan and Ghukasyan, 1998). The geologic media in the selected sites are represented by volcanogenic, plutonic, volcanogenic-sedimentary, and sedimentary complexes of rocks of different ages.

The western group of sites (1, 7, 8, and 9) is located within the recent (N32-Q) volcanic formation on the periphery plateau of the Aragats and Kechut uplands. These plateaus are composed of lava-pyroclastic formations of a basalt-andesite series of volcanites of significant permeability. The thickness of the lava formations varies from 200 m (sites 1, 7, and 8) to 450 m (site 9). The effusive formations at northwestern site 1 are underlain by thick volcanogenetic (andesite, dacite, rhyolite) and volcanogenetic-sedimentary (tuff, tuff-sandstone, tuff-conglomerates) of Paleogene age, which are metamorphosed and well lithified. These formations are a regional aquitard. The base of the effusive formation of the peripheral plateau Aragats (near sites 7, 8, and 9) is composed of gypsum-salt and clay materials (thickness > 1,000 m), which in turn are underlain by a thick volcanogenic complex of Neogene age. Another group of sites (2, 4, 5, and 6) is composed of thick, well lithified, and dense volcanogenic complexes (different effusives and their pyroclasts) of Paleogenic (site 2) and Mesozoic (sites 4, 5, 6) ages, which are aquitards or formations of low permeability. The northernmost and southern sites (3 and 12) are composed of granitoid formations created in late Cretaceous-Eocene and Neogene stages of plutonic intrusions. The other two sites (10 and 11) are located within the region of Paleozoic and Paleogenic thick sedimentary clay-carbonate formations, which are unsaturated and practically impermeable.

3.7. STAGES IN THE PROCESS OF SELECTING SITES AND DIRECTIONS FOR FUTURE RESEARCH

The long-term experience gathered by other countries in the multistep process of selecting sites, as well as recommendations by IAEA, can be used as a basis for developing a site-selection program in Armenia. A literature review (particularly *Geological Problems in Radioactive Waste Isolation: Second Worldwide Review* [Witherspoon, ed., 1996)]) suggests that the examples most appropriate for Armenia were developed in eastern European countries (Czech Republic, Croatia, Bulgaria, Slovenia, etc.). Such a multistep site-selection process begins with a geologic investigation of the territory.

The initial stage, called the area survey (which has already begun), usually includes (IAEA, 1994a; 1994b):

a. Regional mapping, including the selection of areas with potentially appropriate sites
b. Preparation of selected sites for future research.

In this report, we present the main approaches for a preliminary regional model to select potential sites. These approaches require additional investigations for the correction and improvement of the main ideas and tasks. In the future, research should be carried out to confirm and localize the selected sites within an area of 20–50 km^2, using both detailed database analysis and the results of field geological-geomorphological, seismotectonic, and hydrologic-hydrogeologic investigations.

Figure 3.3. **Schematic geological map of Armenia (from geological maps 1:6,000,000,1968,1971).**
Legend: (1) Pleistocene: *alluvium-proluvium, talus, lake and other deposits (clay, sand, pebble, travertine),* **(2) Neogene-Quaternary:** *volcanic and volcanogenic formations (basalt, andesite, dacite, rhyolite, tuff, tuffbreccia, tuff-sandstone),* **(3) Neogene:** *sedimentary deposits (clay-diatomaceous formations, sandstone, rubble, clay, coquina),* **(4) Upper Paleogene-Miocene:** *sedimentary deposits (gypsum-salt clays, multicolored formations, sandstone, conglomerate),* **(5) Paleogene:** *volcanic and volcanogenic formations (basalt,andesite, dacite, rhyolite, tuff,tuff-breccia),volcano-sedimentary formations (tuffite, tuff-conglomerate, tuff-sandstone), and sedimentary deposits (limestone, marl, sandstone, shale),* **(6) Cretaceous-Paleocene:** *sedimenatry, volcano-sedimentary, and partially volcanic deposits (limestone, dolomite, clayey shale, marl, conglomerate),* **(7) Jurassic:** *volcanic and volcano-sedimentary formations (basalt, andesite, rhyolite, tuff, tuffite, tuff-breccia, tuff-conglomerate, tuff-sandstone) and sedimentary deposites (limestone, dolomite, clayey shale, marl, conglomerate),* **(8) Devonian-Triassic:** *sedimentary deposits (dolomite, limestone, marl, shale),* **(9) Pre-Cambrian-Lower Paleozoic:** *(a) metamorphic shales with lenses of limestone and marble, (b) metamorphic volcanites, quarzites, limestones.* **Intrusive rocks: (10) Mesozoic-Neogene:** *different granitoids, granodiorites, granosyenites, monzonites, etc.,* **(11) Mesozoic:** *gabbroids, pyroxenites, peridotites, dunites, serpentinites.* **(12) Upper Pliocene_Quaternary:** *polygenic volcanoes,* **(13) Upper Pliocene-Quaternary:** *monogenic volcanoes,* **(14) Traces of main faults, (15) Magnetic channels** *(volcanic pipes of monogenic and polygenic volcanoes),* **(16) Preliminary selected potential sites.**

The next stage of work should include a detailed characterization of the selected sites. This should include a complex of scientific research and multiscale investigations on specific areas (10–20 km^2) using different methods (field, laboratory, analytical, etc.).

The purpose of this stage of the research is to select the most appropriate one or two sites. Methods used for this selection should be based on the following main criteria: seismotectonic, hydrogeological, geotechnical, physical-mechanical, and thermal. In general, this stage of research should include the following main directions of geologic investigations: geology, hydrogeology, geochemistry, hydrochemistry, tectonics and seismics, geophysics, volcanology, surface processes (erosion, landslides, etc.), and geo-ecology.

The final stage of research should include special investigations on more limited sites (1-2 km^2), within one or two prioritized areas. Research should determine the physical-mechanical, hydrological, geochemical, thermal, and other media characteristics needed for investigation and construction of the repository. The research should include field, laboratory, and analytical investigations, as well as computer modeling.

3.8. CONCLUSIONS AND RECOMMENDATIONS

To solve the problem of underground radioactive waste disposal in a timely and successful manner, the following is recommended:

- Creation of a general Armenian program for long-term and safe isolation of radioactive waste in geologic media
- General scientific supervision of this program by The National Academy of Sciences of Armenia
- Creation of special programs and corresponding projects for multistep scientific research to select and characterize potential and prioritized sites
- Creation of special management committees, including groups of experts and scientific organizations, to conduct and control planned investigations
- Development of a permanent connection with specialized international organizations (IAEA, NEA, NRC, etc.) on all stages of the investigations
- Close collaboration with foreign research-scientific organizations and leading scientists with extensive experience in the field of nuclear and radiation safety.

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Chapter 4

Geological Problems in Radioactive Waste Isolation for Belarus

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ABSTRACT. At present, there are no nuclear power plants (NPP) in the Republic of Belarus. However, long-term planning for power-industry development makes the construction of similar plants quite possible. Given this possibility, we have carried out a geological investigation to justify isolation of radioactive waste, from which three groups have been distinguished: (1) low-level and conventionally radioactive waste, (2) high-and medium-level radioactive waste, and (3) an intermediate group of radioactive waste. The optimum and actual geological conditions for radioactive-waste isolation are specified and described, and the ecological consequences of the decisions arising from our investigation are analyzed.

4.1. INTRODUCTION

For the last 15 years, the Republic of Belarus has been living under the conditions of a national ecological calamity. The Chernobyl Nuclear Power Plant accident on April 26, 1986, has been recognized as the greatest disaster that has ever occurred on Earth in terms of its scope and the damage caused. Following the accident, the reactor, which contained 190.2 tons of nuclear fuel, released about 1.8×10^{18} Bq of radiation into the environment, approximately 70% of which fell within the territory of Belarus. The released material included large amounts of radionuclides, including those of iodine, caesium, cerium, barium, strontium, and plutonium. According to Israel et al. (1996), the total ^{137}Cs fallout in Europe after the Chernobyl accident was 8×10^{16} Bq, of which 19.6% fell in Western Europe, about 20% within the Ukraine, 24% in Russia—and 33.5% in Belarus. The total area contaminated by radionuclides in Belarus is 136,500 km^2, of which 2,200 km^2 had an initial ^{137}Cs activity of over 1,480 kBq/m^2, about 4,200 km^2 had 555–1,480 kBq/m^2, and 10,200 km^2 had 185–555 kBq/m^2. This territory was inhabited by 2.1 million people (over 20% of the total population). Radioactive contamination has affected over 1.8 million hectares of agricultural land (22% of the total agricultural land in Belarus), with 26,400 hectares of this land-having been removed entirely from agricultural use. The territory of the Polessie State Radioecological Reserve (131,400 hectares) has turned into a "plutonium reservation" in the center of Europe: the high concentration of radioactivity has removed it from use for many years to come.

The contaminated territories in Belarus and the Ukraine are drained (Figure 4.1) by the river Dnieper and its tributaries (Pripyat, Sozh, Iput, and Besed). It was shown (Kudalsky et al., 1998) that the total natural ^{137}Cs decontamination of the territory for 12 years (1987–1998) after the accident was 8.0332 PBq and the efficiency of ^{137}Cs removal from river runoff reached 0.079 PBq, (or 1.08% of the radionuclide decay within catchments).

Cleanup of contaminated territories (1986–1989 and after) resulted in collection and burial of about 381,000 m^3 of radioactive waste, with a total radioactivity of 270.35×10^{10} Bq.

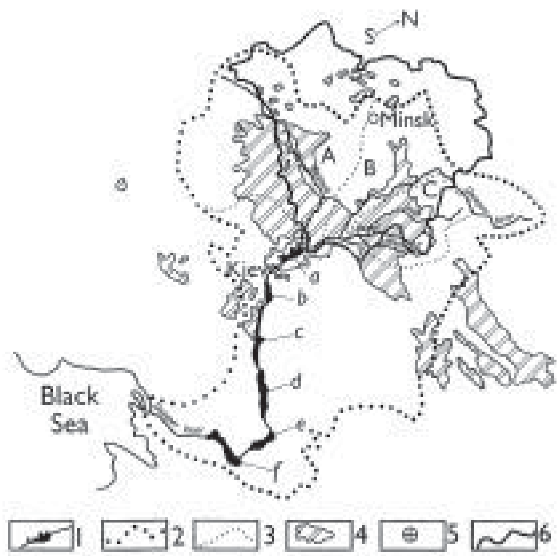

Figure 4.1. Scheme showing the radionuclide contamination of the hydrographic network in the drainage basin of the Dneiper and its tributaries.
Legend: (1) Rivers and reservoirs: a. Kiev, b. Kanev, c. Kremenchung, d. Dneprodzer-zhinsk, e. Zaporozhie, f. Kakhavka; (2) Dnieper superbasin limited; (3) Dnieper tributaries basin limits: A. Pripyat river, B. Belarussian-Russian part of the Dnieper river-head, C. Sozh river, with the tributaries Iput and Besed; (4) Territory contaminated by radionuclides; (5) Chernobyl NPP; (6) Frontiers of the Republic of Belarus.

4.2 LOW-LEVEL AND CONVENTIONAL RADIOACTIVE WASTE

4.2.1. Classification, Quantity, and Existing Burial Conditions

This first group, low-level and conventional radioactive waste (LLW), includes decontamination products removed as a result of cleanup of the territories contaminated from the Chernobyl NPP accident. This group of wastes may be complemented by mainly LLW materials resulting from dismantling of a decommissioned NPP (19,000–20,000 tons from one NPP).

At the first stage of the Chernobyl cleanup, the attempts to minimize the consequences of the accident by decontamination work in the territories of the Gomel and Mogilev regions resulted in collection of about 1 million m^3 of waste (with different levels of contamination). This waste was disposed of in 74 inadequately equipped and seven adequately equipped disposal facilities (about 30,000 m^3 of waste in each).

We expect that at this new, present stage of decontamination and rehabilitation work, about 5 million m^3 of waste would be collected. All this waste should be properly treated and disposed. The management problem related to this kind of waste has some unique features: this is not industrial waste (which has specific parameters that could be more or less controlled at the generation stage). This is waste resulting from uncontrolled transfer and fallout of radionuclides from the damaged Chernobyl reactor. This fallout caused a relatively low (in comparison with industrial criteria) contamination, spread over a vast amount of territory. The specific character of this contamination requires a different approach to decontamination and rehabilitation, and the appropriate technologies should be simple, effective, and inexpensive. Also, the approach to waste treatment and classification should be specific.

Existing classification systems are not applicable to such waste, because their concentration (minimization) would not be possible or even feasible. So we propose new, additional categories of waste: (1) conventional radioactive and (2) conventional clean waste. The necessity of introducing these two new categories of waste could be justified by the huge amounts of such material and its potential danger to the environment.

The following solid waste should be considered as conventionally radioactive:

- Radioactivity between 7.4×10^3 Bq/kg and 7.4×10^4 Bq/kg for β-active wastes
- Radioactivity between 7.4×10^2 and 7.4×10^3 Bq/kg for the α-active wastes
- Radioactivity between 10^{-8} and 10^{-7} gram-equivalent of Ra per kilogram for γ-active wastes
- When the surface contamination of items, considered as the waste, is between 20 to 50 β-particles per cm^2 in 1 min, or between 3 to 5α particles per cm^2 in 1 min.

Waste with radioactivity lower than that defined as “conventionally radioactive” could be considered “conventionally clean” waste.

Table 4.1. General description of the decontamination product (DP) storage sites

Storage sites	Number	DP volume $\times10^3$ m^3	Average specific activity of DP ^{137}Cs, kBq/kg	^{90}Sr, Bq/kg	$^{239,240}Pu$, Bq/kg	Total activity in DP, Bq ^{137}Cs $\times10^{12}$	^{90}Sr $\times10^{11}$	$^{239,240}Pu$ $\times10^9$
Equipped repositories	7	81	1.3-8.9	30-100	0.2-2.5	0.8	0.1	0.1
Unequipped repositories	74	300	0.1-33	40-900	0.2-20	1.9	1.4	3.4

Note:*Skurat et al.,2000

In general, the radioactivity of waste collected as a result of decontamination work is very low, and its isolation from the biosphere could be achieved through shallow land disposal. Disposal safety with this option could be realized by:

- Proper site selection
- Use of natural sorbents and isolation materials to prevent migration of radionuclides from the disposal site
- Proper operation, control, and sealing of disposal facilities.

For disposal of "conventional radioactive waste," a simplified near-surface repository could be used for which complicated engineered safety barriers would not be necessary. However, the following safety requirements should be met:

- The lower level of repository (bottom of the trench) should be 4 m higher than the groundwater level.
- The repository should have an upper isolation layer (clay, other isolation material), which could prevent filtration of surface water through the repository.
- Where necessary, the repository should have isolation and absorption screens made of natural sorbents and materials.
- Conventional clean waste could be disposed at municipal waste treatment/disposal facilities.

The existing status of scientific support programs for decontamination and disposal of radioactive waste from decontamination could not be considered satisfactory. Urgent decontamination and cleanup work, after the Chernobyl accident, was performed mainly by special military troops using their equipment and technologies. Waste was collected at so-called "makeshift storage sites." In most cases these sites should be considered final repositories, but some others need improvements and additional isolation work.

Effective decontamination work carried out in 1986–1989 outside the 30 km zone (the plutonium contamination zone) cleaned 12 × 10^6 m^2 of the exterior surface of buildings and constructions, removed and buried 13.3 × 10^3 m^3 of contaminated ground, and demolished (and buried) 7,570 old buildings. Waste accumulated from Chernobyl decontamination, or "decontamination products" (DP), as well as contaminated soils, building parts, and domestic wastes were stored (Figure 4.2) far from populated localities, mainly in random waste deposit areas (flat sites, gullies, topographic lows including bogs, mined-out rock quarries, etc.).

All DP and other contaminated waste are stored in zones with Cs-137 radioactivity of 555 kBq/m^2 and above; the gamma radiation exposure dose rates in storage sites (DP repositories) vary from 0.03 to 0.23 mR/h. The total area is 6.76 × 10^5 m^2, 2.55 × 10 m^2 of which is occupied by proper DP (total volume of 3.81 × 10^5 m^3). Total DP activity is 2.7 TBq for caesium-137, 0.15 TBq for strontium-90, and 3.5 GBq for plutonium-239, 240 (Table 4.1).

Twenty-four DPrepositories are surface and 50 are near-surface waste repositories. Seven DPrepositories (DPR) of those 50 are of the dump type, and five of them are located in gullies (and in these cases the danger of DP washing away into rivers exists). Eleven DPR are exposed to the hazard of a groundwater seasonal rise, and in eight DPR the decontamination products and accompanying materials are located beneath the groundwater level. Groundwater-contamination magnitude can be judged from observations carried out in the region of the DPR locations Babchin-1 and Babchin-3. The radioactivity of the ground there is 92.5 kBq/m^3 for ^{90}Sr

Figure 4.2. Map of DP repositories located within contaminated territories of Belarus

and 259 kBq/m^3 for ^{137}Cs, and the depth to groundwater varies between 2.5 and 6.2 m. The ^{90}Sr concentration in water samples ranged from 0.01 Bq/L in 1987 to 0.14 Bq/L in 1989 and to 0.15 Bq/Lin 1993–1996. The ^{137}Cs concentration in 1987 was 1.04 Bq/L, but in 1996 it was only 0.53 Bq/L (Skurat et al., 2000).

The radionuclide concentration found in the subsurface water of DPR areas was high: 23 Bq/Lfor ^{90}Sr and up to 500 Bq/L for ^{137}Cs (village of Kulazhin, Braguin district). Radiochemical studies of the soil have established that about 19% of the ^{90}Sr is water-soluble and as much as 93% of it is in exchangeable forms. For ^{137}Cs, nonexchangeable forms prevail, 36–70% of it firmly bound in the crystalline structure of soil minerals. Taking into account the imperfect engineering equipment in a DPR and its exposure to precipitation and groundwater effects, we can expect that as much as 50% of the total ^{90}Sr (and large amounts of ^{137}Cs and other radionuclides) will be picked up by groundwater. At the later stages of decontamination work (after 1993), 5 million m^3 of low-level radioactive waste were expected to be cleaned up, as well as considerable amounts of radioactive waste with specific radioactivity of at least 10^4 Bq/kg, common to areas with contamination density of at least 1.480 kBq/m^2. On an annual basis, the volume of domestic ash makes up more than 8,000 tons, with specific activity above 10^4 Bq/kg and 32,000 tons with a radioactivity over 10^3 Bq/kg. The amount of wastewater-settled solids with radioactivity over 10^3 Bq/kg exceeds 3,500 tons and, with radioactivity of 10^2 Bq/kg, runs up to 10,000 tons. It was established that the volume of waste wood during sanitation cutting of forests could be 230,000–250,000 tons, with radioactivity of 10^3 Bq/kg.

All in all, the total radioactivity of decontamination products could total 2.75×10^{13} Bq. All these wastes are subject to suitable processing for subsequent burial in technologically appropriate and ecologically safe repositories. Moreover, a part of the earlier-accumulated DP needs to be reburied under ecologically appropriate conditions.

4.2.2. Optimum Conditions for Subsequent Burial of Radio active Waste

Most of the Chernobyl radioactive fallout in Belarus fell within the territory of the Mogilev region and (particularly) within the Belarussian Polessie area (Gomel region). The Polessie is a vast, flat, gently rolling plain built by lake-alluvial, alluvial, and (to a lesser degree) fluvioglacial and morainic deposits of Quaternary age. The thickness of Quaternary deposits within the Polessie area ranges from 20 to 120 m, averaging 50–60 m. The territory is poorly drained, and hence the unsaturated zone is small in thickness, never increasing above a few meters except at the most elevated sites of the plain (which are usually formed by fluvioglacial, morainic, or aeolian sediments). The unsaturated zone is composed mainly of sandy formations.

Flat plain relief is most common, although occasionally aeolian landforms occur as dunes and other high elevations. A distinguishing feature of aeolian formations is their good morphological manifestation. All linearly oriented ridges and systems of ridges extend from northwest to southeast. The width of these ridges ranges from a few meters to 1.0 km, and the height ranges from 2–3 to 25–30 m, with slope steepness no more than 25°. Aeolian deposits are represented by dust and fine (to occasionally medium-sized) sand grains, and their thickness varies from a few meters to 10 m.

Considering the natural conditions of the radioactively contaminated territories of Belarus and a low DP activity in natural and engineered locations in the Chernobyl contamination zone, near-surface burial in suitably equipped trench-type repositories seems to be most effective. Waste storage safety in such repositories is provided by choosing geologically and hydrogeologically correct burial grounds; using natural sorbents and insulators (sand, clay, concrete blocks) as natural barriers; making available the necessary engineering protection (concrete, insulation coating, films); and providing monitoring and testing facilities to control repository tightness and radioisotope migration through protective barriers. In the natural and geological conditions of contaminated Belarus' territories, the low-radioactive DP seems to be contained, most effectively and ecologically, within landforms elevated above a low groundwater level (Kudelsky and Yasoveyev, 1991; Kudelsky et al., 2000.). These landforms could also accommodate reburial of DB now located at ecologically unacceptable repositories.

We have divided southeastern Belarus into zones according to DP burial conditions, to select those geological and geomorphological locations most suitable for appropriate repository construction. The following factors have been considered:

- Location of testing grounds in evacuation zones and obligatory evacuation of people from areas with an radioactivity over 1.480 kBq/m^2
- Geological and hydrogeological conditions meeting the requirements of safe radioactive waste storage
- Surface radiation control data for localities and roadways
- Hydrological structure of a territory, including information on flooding and seasonal water levels, and areas subjected to flooding..
- Atmospheric circulation and seasonal wind patterns in relatively uncontaminated territories and populated areas
- DP transportation routes, quality of major and bypassing roadway conditions (near settlements), and dust generation during transportation and loading.

Special attention was given to the geological and hydrogeological conditions in the evacuation zone near the Chernobyl NPP, where the contamination levels of the territory are rather high (and which therefore would in any case be unusable for a long time).

Based on this investigation, we recommend that disposal facilities for radioactive-waste decontamination and rehabilitation work should be constructed in well-aerated (dry) zones, preferably on large hills, more than 10 m high and not less than 1 km^2 in area (Figure 4.3). These hills should be formed by sand or clay and located close to good road systems.

These landforms show promise as locations for correctly prepared repositories for reburial of radioactive waste (from the earlier-built ecologically unacceptable makeshift storage facilities), as well as for repositories

Figure 4.3. Map showing positive landforms and clay rock occurrence within the southeastern part of Belarus (30 km zone).
Legend: (1) Small positive forms (sand dunes),(2) clay rock occurrence areas; (3) limits of objects potentially suitable for low- and conventionally radioactive waste repositories (an altitude of landforms in meters); (4) boundary of 30 km zone.

of medium-level waste and ionizing radiation sources. Some geological and geomorphological locations within these landforms suitable for recycling radioactive waste were selected. In addition, numerical models simulating radionuclide migration from a repository (built in recommended locations with given migration parameters) have been developed (Gvozdev et al., 1996; Surat et al., 1995).

In conclusion, we can suggest the urgency of the task ahead by estimating the ecological efficiency of the earlier-created DPR. Their total effective area is 0.25 km^2, while the volume of buried waste is 381,000 m^3. The total radioactivity of buried DP is about 270.35×10^{10} Bq. Having compared the activity of buried DPwith that of radionuclides from fallout within the territory of the Republic of Belarus (129.5×10^{18} Bq), we conclude that the radioactivity of buried DP is barely 2.08×10^{-6} %. This would suggest that the ecological efficiency of the Belarus decontamination and DP burial program has been marginal. However, it is difficult to overestimate the program's social importance for the recovery period after the Chernobyl accident.

4.3. HLW AND MEDIUM-LEVEL RADIO ACTIVE WASTE

The second group includes sealed waste sources of ionizing radiation (about 12 tons or 4-6 m^3/year, with a radioactivity of 7–500 MBq/kg. The total radioactivity is as high as 7 TBq/year when prospective NPP operation waste is included.

In the context of the governmental program for future nuclear power plant construction, the sites for their location have been selected, and a preliminary investigation of geological conditions has been carried out to locate potential sites for HLW and ILW repositories. Three rock types have been recognized as potentially suitable for radioactive waste disposal: crystalline basement, salt beds, and thick deposits of monomineral clays.

As shown in Figure 4.4 (and mentioned above), the territory of Belarus has been subdivided into zones showing the various conditions to be considered in selecting sites:

- Zone I areas are those where depth to the crystalline-rock basement ranges from 0.0 to 0.4 km

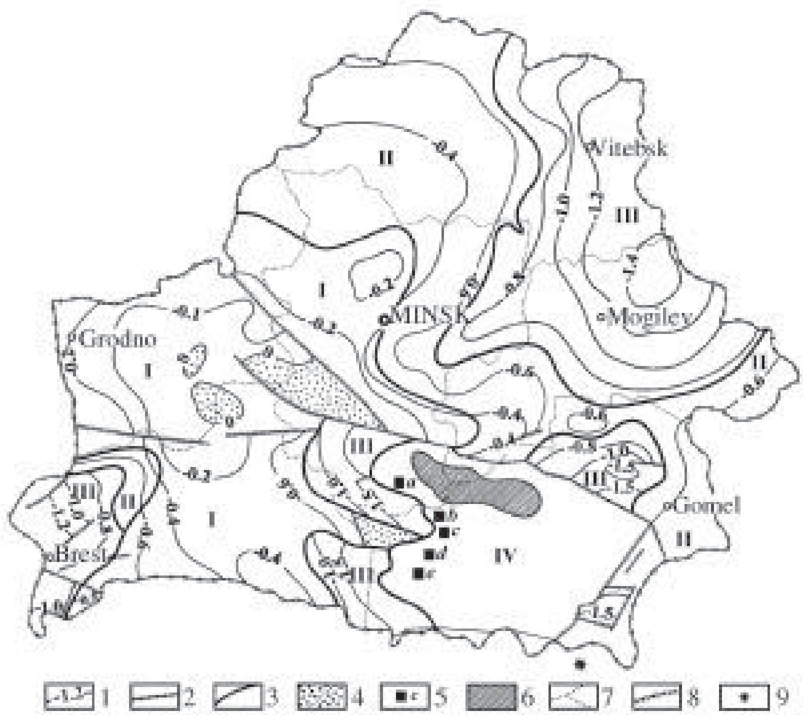

Figure 4.4. Geological division of Belarus into zones in accordance with conditions for medium- and high-level radioactive waste isolation.
Legend: 1-Isolines showing depth to crystalline basement (km); 2-tectonic faults penetrating through the sedimentary cover; 3-boundaries of zones with different conditions and potential for waste isolation. Symbols for regions with geological structures considered as primary candidates for repositories; 4-the crystalline basement of Bobovnya, Slonim and Mikashevichy uplifts; 5-salt dipir domes of: *a-Novaya Dubrova, b-Zarechie, c-Kopatkevichy, d-Konkovichy, e-Shestovichy;* **and 6-palygorskite clay beds of the Pripyat Graben. Other symbols: 7-boundaries of administrative regions; 8-frontier of Belarus; and 9-Chernobyl nuclear power plant.**

and conditions are considered to be very suitable for engineering and constructing a repository.

- Zone II areas are those where depth to the basement ranges from 0.4 to 0.6 km and conditions do not show much promise for constructing a repository.
- Zone III areas are those where depth to the basement ranges from 0.6 to 1.5 km and conditions show very little promise for constructing a repository.
- Zone IV is the Pripyat Graben, considered to have geological structures highly suitable for a waste repository. It contains stratified salt formations of the Middle and Upper Devonian, domed salt diapirs of the upper salt strata and palygorskite clay beds of Upper Devonian.

Crystalline rocks are exposed or overlain by Anthropogene deposits of limited thickness within the Central-Belarussian Massif, Ukrainian Shield, and Mikashevishy Uplift (Horst). The Central-Belarussian Massif with the Bobovnya Uplift of granitic rocks (100–170 m in thickness) and the Slonim Uplift of granulitic complexes with blastomylonite show the greatest promise for radioactive waste isolation.

Salt formations cover a vast area (23,000 km^2) within the Pripyat Graben and are represented by two Upper Devonian salt strata: Upper Frasnian (up to 1,100 m thick) and Upper Famennian (up to 3,000 m thick) separated by terrigenous-carbonate strata. Salt formations occur in a zone of diaparisim in the depth range from 300 to 400 m and are considered to be promising waste-disposal sites (Novaya Dubrova, Zarechye and Kopatkevich Uplifts in the northwest, and Konkovichy and Shestovichy Uplifts in the southwest of the Graben).

Large deposits of palygorskite clays (with thickness ranging from 140 to 150 m) are found in the Pripyat Graben at depths of 80 to 120 m. They are of particular interest as potential sites for radioactive waste repositories.

4.4. INTERMEDIATE RADIO ACTIVE WASTE

Aspecial category of radioactive waste is an intermediate group involving a small volume from the immediate use of radioisotopes (up to 10 m^3/year in plastic bags). This waste may be isolated along with decontamination products or along with sealed waste sources of ionizing radiation, taking into account the half-lives of some isotopes.

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Chapter 5

The Belgium RD&D Program on Long-Lived and High-Level Waste Disposal: Status and Trends

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ABSTRACT. For more than 25 years, characterization studies have been performed in the context of radioactive waste disposal in Belgium. For conditioned long-lived and high-level waste (HLW), no siting procedure has yet been launched, though studies on geological disposal continue, focusing on a potential host rock, the Tertiary Boom Clay Formation. In the last decade, however, some geological and hydrogeological studies have also been performed on an alternative host formation, the Ieper Clay.

The Belgian program for the management of HLW contributed to the acquisition of extensive expertise on disposal in clay. Special attention has been devoted to the characterization of various waste forms arising in Belgium, the behavior of these waste forms in a clay environment, the characterization of the geosphere, and the study of the different processes controlling the transport of radionuclides to the biosphere—as well as studies regarding the disposal concept and repository construction. Performance studies have helped in assessing the level of confidence that can be reached at the various stages of the characterization in both near and far fields and in determining research priorities.

In the 1980s, research was focused on characterization and feasibility studies performed in the context of constructing an underground research laboratory (HADES URL). The program was upgraded in the 90s, and R&D has evolved towards large-scale integrated demonstration *in situ* (RD&D) tests. Ongoing activities include studies on spent fuel that became, in parallel with studies on vitrified HLW, a key issue as a result of changing fuel cycle policies in Belgium.

During the last five years, synergies and resources have been consolidated between the waste management agency NIRAS/ONDRAF and the research center SCK•CEN, which resulted in the creation of the Economic Interest Grouping (EIG) EURIDICE. The mission of this consortium is to manage the present extension of the URL and the underground experimental program. The latter includes the PReliminAry demonstration test for CLAY disposal (PRACLAY) project aimed at demonstrating the overall safety and feasibility of deep geological disposal of high-level, heat-emitting waste in a clay formation.

This paper gives an overview of the more relevant results of the Belgian program. New research areas, like retrievability, also include different aspects related to human sciences.

5.1. THE BELGIAN FRAMEWORK

Belgium has a well-established nuclear industry, including fuel fabrication, low-level waste (LLW) treatment, reprocessing, and storage. A pilot plant has even been built to vitrify high-level liquid waste from the former Eurochemic facility, but reprocessing services were purchased abroad. In Belgium, reprocessing is not the reference policy any more. Since the decision on future reprocessing is pending, studies on the direct disposal of spent fuel in a clay formation have gained increased interest. The volume of high-level and long-lived waste to be produced by a fully installed nuclear power of 5.5 GWe over a 40-year period is an estimated 10,000 m^3. The main challenge of a radioactive waste management program for a densely populated country like Belgium is the disposal issue. It is assumed that in the present reference scenario, the operation of a repository for underground waste disposal could start around 2040/2050 and last until 2070/2080 (closure phase). This requires, in the case of vitrified waste, a temporary surface storage of about 60 years, allowing heat output to be significantly reduced. This storage facility is now operational.

For high-level and alpha-bearing waste, an inventory of the potential deep geological formations in the country was made in the 1970s by SCK•CEN, based on criteria established with the National Geological Survey and suggestions provided by the International Atomic Energy Agency (IAEA) and the Commission of the European Community (today, the European Union). In 1976, SCK•CEN started the characterization of the Boom Clay Formation situated in the Mol-Dessel area in the northeastern part of the country (Figure 5.1). In the 1980s, an underground laboratory was built in this potential host formation at a depth of about 224 m.

The research program and assessment studies were therefore developed according to a site—or at least a formation-specific approach—without anticipating the conclusions of a more detailed site-selection procedure. The underground laboratory, after evaluating the technical feasibility of such a construction, became an *in situ* facility for performing long-term integrated tests in close to real conditions. The European Commission (EC) and the radioactive waste management agency (NIRAS/ONDRAF) provided scientific and financial support. Although most of the research performed at SCK•CEN has become contractual, both the extent of international partnerships and the broadening of collaborations with universities allow SCK•CEN to complement the contractual work with exploratory and more fundamental scientific studies.

Currently, NIRAS/ONDRAF is finalizing the SAFIR 2 (Safety and Feasibility Interim Report) report that will be presented to the Belgian government in the year 2002. This report is expected to have an important impact on the Belgian geological-disposal program from 2003 onwards. As an alternative to the Boom Clay, preliminary investigations on the Ieper Clay from reconnaissance boreholes and core samples will be reported.

Besides the R&D program on waste disposal, which is orientated towards characterization and long-term safety, substantial effort still needs to be devoted to concept demonstration. Activities since 1995 have been focused on the PRACLAY project, which aims at demonstrating the concept for the disposal of high-level heat-emitting waste (HLW) in clay. In 1995, SCK•CEN and NIRAS/ONDRAF set up an Economic Interest Grouping (EIG PRACLAY). Since 2000, the EIG is responsible for management of all activities taking place in the integrated underground infrastructure. The name of the EIG was recently changed to EURIDICE (European Underground Research Infrastructure for the Disposal of waste in a Clay Environment).

SCK•CEN continues its partnership within the Mt. Terri project (Switzerland) with active contribution to the diffusion and gas migration experiments of the Opalinus Clay, a potential host formation in Switzerland.

5.2. THE REFERENCE CONCEPT IN PERFORMANCE-ASSESSMENT STUDIES

Performance-assessment studies evaluate the long-term safety of the disposal system by identifying possible scenarios that might lead to the exposure of humans to radioactivity or toxic substances, analyzing the consequences of the most relevant scenarios, and comparing the estimated concentrations, fluxes, doses and risk with appropriate safety criteria. Based on the multibarrier concept illustrated in Figure 5.2, key issues or components contributing to the isolation function of the system can be highlighted, namely the waste inventory, the short- and long-term behavior of waste forms and engineered barriers, and the barrier properties of the host formation.

For argillaceous formations like the Boom Clay, the host "rock" is the primary barrier against radionuclide migra-

Figure 5.1. Location of nuclear installations in Belgium

tion because it provides good sorption capacities, very low permeability and favorable geochemical conditions. Its sorption capacity is mainly related to its cation exchange capacity, linked due to the presence of smectite and smectite-illite interstratified clay, the strong reducing conditions due to finely dispersed pyrite and humic materials present in clay, and the slightly alkaline environment, caused by the presence of carbonates. In the Boom Clay, the radionuclide migration is dominated by diffusion. Advection plays a secondary role due to the very low hydraulic conductivity and the absence of preferential paths for water in this clay formation.

The waste packages would be emplaced in separate disposal galleries. Approximate dimensions (expressed as inner diameter) for the different components are 6 m for the shafts, 4 m for the primary galleries, and 2 m for the HLW disposal galleries (Figure 5.3). The current Belgian disposal concept for HLW consists of a concrete-lined gallery with a central tube containing the vitrified waste drums in their individual overpack. The integrity of this physical barrier needs to be ensured at least for the thermal phase. Bentonite-based backfill blocks fill the gap between the central 0.5 m diameter tube and the 2 m inner diameter lining. The design for HLW disposal is currently undergoing a systematic review. The disposal galleries could be as long as 200 m. Concrete would be used for the lining of shaft and galleries, ensuring the mechanical stability during the operational phase of the repository. For long-lived medium-level waste, the whole section of the galleries, 3.5 m in diameter, would be filled with canisters, with remaining voids possibly filled with concrete.

Preliminary safety assessments were carried out for many scenarios ranging from normal evolution to important disruptive events. For the reference inventory of reprocessing waste (HLW, MLW, hulls) resulting from a 40-year nuclear program in Belgium, the highest calculated dose to population, for a normal evolution scenario and water well pathway, is of the order of 3 10^{-7} Sv/a, which can be compared with a dose constraint of 3 10^{-4} Sv/a, as suggested by international regulatory bodies. These doses would occur after a period of about 60,000 years. The methodology for risk assessment was established and tested in the framework of international exercises in cooperation with the Nuclear Energy Agency of the Organization for Economic Cooperation and Development (OECD) and the EC ("PAGIS," "PACOMA," "EVEREST," and recently "SPA"). The two first exercises were updated in the early nineties and the results published in "Updating

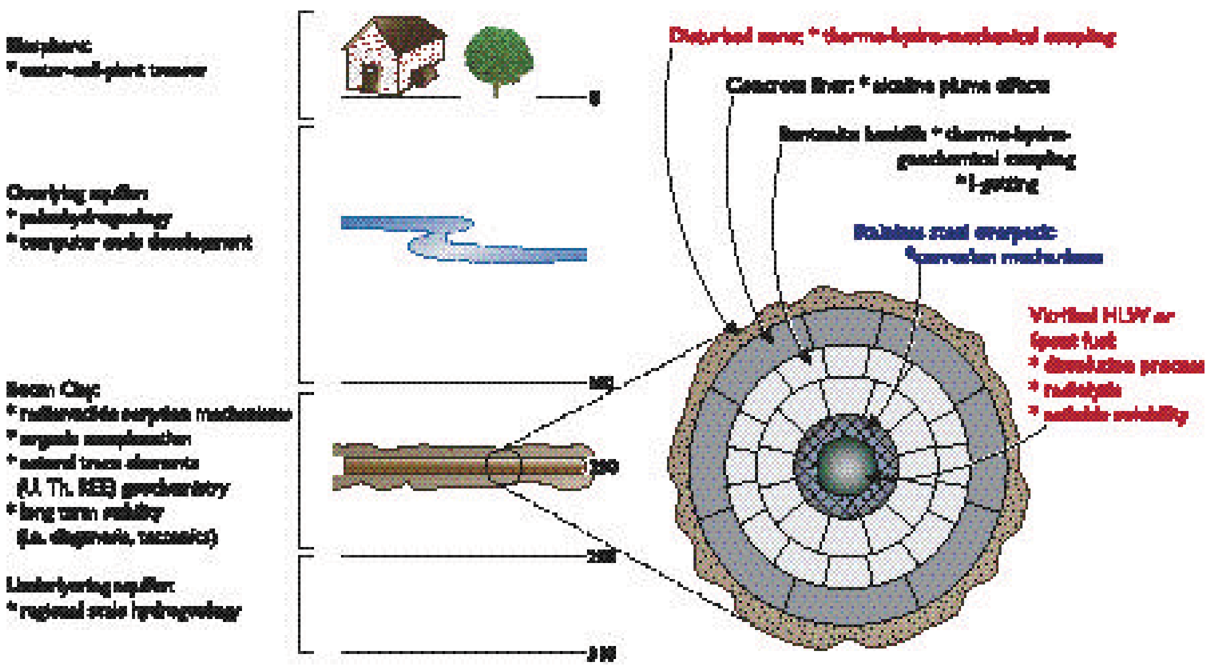

Figure 5.2. Main research items and processes associated with the present reference concept

1990”). The results of the last update form an important contribution to the SAFIR 2 report.

The role of the hydrogeological system in the overall methodology applicable in performance assessment studies for the geological disposal of radioactive waste requires consideration of its variability (groundwater flow patterns) over long time scales, e.g. as a consequence of the expected climate evolution. Due to reduced infiltration during glacial periods, the dilution of radionuclides released from the host clay layer into the surrounding aquifers can be lower, inducing higher radionuclide concentrations than estimated on the basis of present flow conditions.

5.3. THE BELGIAN PROGRAM FOR GEOLOGICAL DISPOSAL

The formation- and site-specific approach allowed SCK•CEN to collect the valuable results and input data required for modeling work. The continuity of the research program for more than 20 years now, and the availability of both underground and surface laboratories, ensures the integrated character of the Belgian geological disposal program.

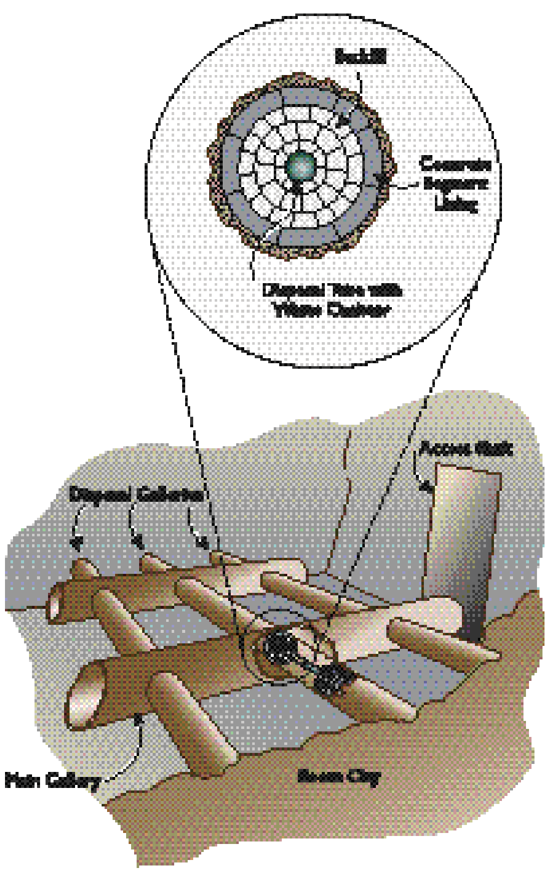

Figure 5.3. Multibarrier concept for HLW disposal

5.3.1 CHARACTERIZATION OF WASTE PACKAGES AND THEIR COMPATIBILITY WITH CLAY

For about 15 years, the performance of physical barriers and waste forms relevant to the Belgian program has been studied under realistic geological repository conditions. A large part of the research program is related to the compatibility of waste forms and their potential container or overpack material with the clay host formation.

With regard to the compatibility of waste forms in clay, the current main research activities deal with:

- The corrosion mechanisms of high-level waste glass in geological disposal media, with special emphasis on the leaching behavior of Np and Tc, and the modeling of these mechanisms
- The effect of radiolytic degradation of bituminized waste and the degradation products of alpha-contaminated cellulose waste on the solubility of Am and Pu in geological disposal situations
- The solubility of UO_2 in Boom Clay water, including the effect of humic acids and carbonates, and the study of the effect of α-radiolysis on the corrosion of UO_2
- The study, by electrochemical techniques, of how sensitive stainless-steel-container materials are to localized corrosion in geological disposal conditions.

Estimations of glass-canister lifetimes using analytical codes point out the need for further refinement. A conceptual model has been developed to describe the phenomena that occur during glass dissolution on a molecular level. Monte Carlo simulations were carried out to take the uncertainties in the model parameters into account.

So far, *in situ* tests on waste forms were restricted to small, inactive, or doped samples. A new experiment including glass samples doped with large amounts of alpha emitters has recently been initiated in the URL. This experiment will determine the dissolution of glass in simulated disposal conditions and assess the migration of the radionuclides through different clay-based buffer materials in a radiation field. This 10-year project, called CORALUS (CORrosion of Active gLass in Underground Storage conditions), consists of four modular test tubes containing inactive and alpha-active SON68 glass samples and several buffer materials. These test tubes are operated at different temperatures. In two of the test tubes, ^{60}Co sources are installed to simulate the gamma radiation field for a real HLW canister. A blank test (inactive tube) was performed for three months at 90°C to test and optimize the operation of the active tubes (sampling of interstitial solutions, measurement of *in situ* pH and redox potential, dismantling operations).

One of the potential problems related to geological disposal of bituminized waste is that radiolytically generated water-soluble complexants might increase the solubility and reduce the sorption of important radionuclides. Inactive Eurobitum, a reference Belgian bitumen-conditioned waste type (produced by Eurochemic) containing (on average) 35 wt% of reprocessed sludge, was irradiated in contact with representative solutions until a total adsorbed dose (corresponding to the alpha dose) accumulated within 10^5 years (about 5 MGy).

This study showed that the radiolytic degradation products have only a minor effect on the mobility of actinides (i.e., the solubility increased slightly, but the sorption on the Boom Clay increased as well).

Studies have been launched pertaining to the direct disposal of spent fuel. Preliminary performance assessments based on literature information indicate that this option could be acceptable. As the experimental basis for radionuclide release from spent fuel is very narrow, characterization activities have been started to identify the processes controlling the dissolution/alteration of spent fuel under realistic disposal conditions. The first studies showed that the uranium solubility is about 10^{-8} mol/l in Boom Clay water (reducing conditions, nominal concentration of humic acids and carbonates).

Regarding the waste container (overpack), the susceptibility to localized corrosion (pitting) of the candidate container materials is determined by cyclic potentiodynamic polarization (CPP) measurements performed in different clay water and slurries with regard to chlorine and thiosulfate content and for different temperatures and oxygen concentrations. This program includes immersion tests on the stainless steel AISI 316L hMo.

5.3.2 Far-Field Studies in Host Rock

The term *migration* refers to a large range of activities dealing with methodology, laboratory and field experiments, geochemical transport phenomena, and codes for flow and transport. The most critical radionuclides of interest in the Belgian program are ^{129}I, ^{79}Se, and some other fission products.

Previous programs led to the following important results:

- Laboratory experiments have to be performed under strictly anaerobic conditions. A short exposure to traces of oxygen alters the physico-chemical characteristics of Boom Clay.
- The aqueous speciation, the clay surface properties, and the concept of anion exclusion and diffusion-accessible porosity are fundamental for understanding radionuclide migration.
- The organic matter distributed between the solid phase and the interstitial solution seems to play a dual role in the sorption/complexation processes of the trivalent actinides and lanthanides.
- The 3D migration model using parameter values determined from experiments on small clay plugs (3 cm) for HTO (tritiated water) and iodine, carried out in the surface laboratory, have been validated at a metric scale by comparing the predictions with the experimental results of a large scale *in situ* injection test.

Electrokinetic methods are developed and used as a technique to reduce the very long time presently needed for the migration experiments, and to study the speciation of the different radionuclides in the reducing clay environment. After a detailed assessment and a successful application of the electromigration technique for directly determining the diffusion parameters of moderately and strongly retarded species, we apply it to the study on migration behavior of redox-sensitive species like uranium.

Naturally occurring radioactive isotopes of U, Th, and REE (rare earth elements) in the Boom Clay can be considered natural analogues of critical elements for the long-term safety of a radioactive waste repository. The study of the long-term behavior of these trace elements and radionuclides naturally occurring in the Boom Clay allows us to get information in realistic geological conditions and over geological time-periods relevant for the disposal safety assessment. This study is meant to increase confidence in the long-term model predictions of radionuclide migration. The mineralogical, geochemical, and radiochemical composition of the Boom Clay is studied over the entire depth of the clay deposit. Special attention is paid to the U-rich interval at the base of the Putte Member and the "Double Band," which is the siltiest layer of the Boom Clay (and therefore a potential zone of higher permeability and pore water mobility).

5.3.3. Near-Field Aspects

The basic issues to be dealt with in the near field are the possible disturbances caused by the construction of disposal galleries or waste packages and the performance of the engineering barrier system. The mechanical and chemical stability of the near field must provide a set of controlling factors for groundwater movement, radionuclide transport, and heat. We need to demonstrate that the excavation-disturbed zone in clay and the backfill/sealing material will not substantially affect the subsequent migration behavior of radionuclides. These aspects are quantified by modeling coupled processes of the thermal-hydrologic-mechanical (THM) and geochemical behavior of these materials.

The quantification of the disturbed zone around an HLW disposal gallery successively submitted to decompression and thermal loading is one of the main objectives of the PRACLAY demonstration test, simulating the thermal output of a 30 m long HLW disposal gallery at scale 1:1.

The PRACLAY test requires an extension of the HADES URL. After completion of the second shaft at the end of 1999, the excavation of the gallery connecting this shaft to the existing facility is scheduled to start at the end of 2001 (Figure 5.4). The disturbance observed around the shaft requires further study to understand the processes taking place and to follow (over time) the expected self-healing behavior of fissures. The construction of the connecting gallery provides a unique opportunity to monitor the evolution of these hydrologic-mechanical disturbances in the Boom Clay formation. A monitoring program has been set up (mine-by test), and current models will be improved and validated for describing the hydrologic-mechanical behavior of the Boom Clay.

The surface mock-up (OPHELIE) prepares and complements the underground work. It is 5 m long with a cross-section similar to that of the disposal gallery (central tube, backfill). A steel liner keeps the backfill under pressure when the latter starts to swell from water uptake. The experience gained during the installation and operation of this mock-up can now be used to improve the design of the *in situ* demonstration test, including feasibility aspects and instrumentation techniques. One bit of information provided by the mock-up is the existence of strong chemical gradients in the backfill during the test. The corrosion of the stainless-steel components and the mineralogical and chemical analyses of the backfill materials are considered as important items in the dismantling program, scheduled for October 2001.

The experience gained from the mock-up allows us to revise the objectives of the PRACLAY experiment. It will contribute to a better understanding of the disposal system, including interactions among its components. The THM characterization of the backfill material requires still a lot of work. Also, modeling efforts will be further developed to simulate the unsaturated hydrologic-mechanical behavior during the hydration process of the backfill. A fairly extensive database containing THM properties of unsaturated clays is a prerequisite to allow for physical model testing, model calibration, and validation.

When highly compacted bentonite blocks were used for the mock-up, a mixture of high-density pellets and clay powder was selected as backfill material to test the water- and gas-tightness of a seal plug in a shaft configuration (RESEAL project). This ongoing, large-scale *in situ* demonstration project is developing and assessing backfill and sealing materials (and methods) in close to real conditions. The second phase, now starting, will investigate the follow-up of the seal hydration and seal-testing

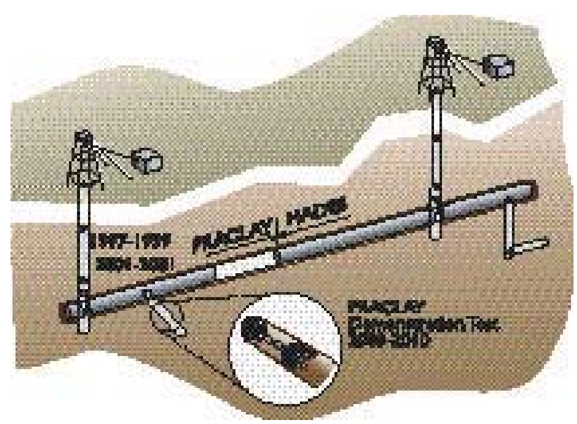

Figure 5.4. Extension of the HADES URL in the frame work of the PRACLAY project

phase. Water intake, relative humidity (suction), swelling, and pore-water pressure are monitored, as well as the effect of the bentonite swelling on the host clay. The validation of models describing water and gas flow through the seal and the near field will take advantage of this test.

5.4. TRENDS REGARDING THE FUTURE OF THE PROGRAM

The research priorities for the next ten years will be defined as a consequence of the discussion and review of the SAFIR 2 report to be presented in early 2002. Yet some trends for the future research are already clear:

Waste
Whereas most efforts were till now devoted to vitrified HLW, future research will be more devoted to the behavior and release of radionuclides from spent fuel. Regarding medium level waste (MLW), potential problems such as gas generation and swelling, which affect the acceptability of EUROBITUM waste for geological disposal in clay, will be studied in detail.

Near Field
Here the main focus will be on the effects of the repository and waste on radionuclide mobility in the host rock (i.e., sodium nitrate release from the EUROBITUM waste, alkaline plume from concrete liner and backfill, and fissuration caused by the excavation of galleries and their self-healing).

Far Field
With regard to migration of radionuclides in the Boom Clay, more attention will be paid to the study of migration mechanisms. The aim is to increase confidence in the migration models and parameters used in the long-term performance-assessment studies by increasing our understanding of the fundamental mechanisms.

While our geological, geochemical, and hydrogeological knowledge of the Boom Clay is rather broad, our knowledge of the alternative host rock, the Ieper Clay, is relatively small. A concerted effort will be needed over the next few years to assess the feasibility of a geological repository in the Ieper Clay.

Concept, Engineered Barriers, and Demonstration
The observations on the OPHELIE mock-up and the engineering studies preparing the PRACLAY demonstration test have revealed a number of technical problems with the HLW disposal concept and some gaps in our knowledge of the engineered barriers. These are mainly related to the thermal-hydrologic-mechanical and chemical behavior of the backfill, and its chemical and mechanical interaction with the central disposal tube. Therefore, in the future, a more systematic approach is required for concept development. It will include a clear definition of the safety functions of each engineered barrier and the experiments and models needed to demonstrate their performance. For the development of the new concept and the PRACLAY demonstration project, this systematic approach is applied.

In the development of the concept, more attention will be given to retrievability and monitoring, together with considerations of overall safety and concept robustness.

Performance Assessment, Confidence Building, and Social Aspects
The future research described above will be oriented toward providing the information needed to enhance the credibility of long-term performance-assessment studies. In these assessments, confidence building and the systematic argumentation of decisions concerning scenarios, models, codes, boundary conditions, and parameters will become more and more important. In this respect, the application of alternative safety indicators and the study of natural or man-made analogues will be further developed.

Because it is clear that technical knowledge will not be sufficient to create confidence for a wider public, societal and ethical studies will get more and more weight in the decision-making process. Since last year, SCK•CEN started to contribute in this field by studying the gap between our technical understanding of safety and the public perception of it.

Valorization of Expertise in an International Framework
Different forms of collaboration have already been experienced at SCK•CEN in the past, from assistance and training (methodologies, models, and databases) when performing safety studies for potential repositories (Russia, Slovenia, Hungary, Slovak Republic) to consulting activities for URL development or experiments in clay.

The HADES underground research laboratory is a unique deep-clay-layer infrastructure, offering important opportunities for co-operation. As such, its administrators will in the future be receptive to the participation of foreign organizations. New large-scale, integrated experiments, designed to demonstrate the performance of disposal systems or components in real conditions, will be developed in an international framework.

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Chapter 6

Current Status of the Site Selection for Radioactive Waste Disposal in Bulgaria

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ABSTRACT. This report describes the current status of investigations concerning site selection for a repository to hold radioactive waste from the Bulgarian Nuclear Power Plant (NPP) in Kozloduy. A centralized approach was applied to site selection for NPPwastes, and the potential site-screening phase of an area survey in Bulgaria has been completed. A methodology adapted to conditions in Bulgaria was used. Four potential sites for low- and intermediate-level waste disposal in the loess and clay terrain in the vicinity of the Kozloduy NPPhave been selected. In addition, two potential sites in marl terrain, a little further from the NPP, and two sites in the Sakar granite pluton, have been selected for high-level waste deep disposal. This report also describes the radon storage facility for waste from medical, agricultural, industrial, and research sources that was built in the early 1960s in Lozen Mountain, close to Sofia, the capital of Bulgaria. The complex geological setting of this site is also discussed.

6.1. INTRODUCTION

The methodology for, and main results from, the conceptual and regional-mapping phases of the area survey for deep geological disposal of high-level nuclear waste (HLW) (produced by nuclear power plants in Bulgaria) were described in the *Second Worldwide Review* (Kozhoukharov et al., 1996).

In this report, we discuss the further development of these investigations and the current status of low- and intermediate-level nuclear waste (LILW) repository-site-selection studies. We also present the geological setting of the only radioactive waste storage facility in the country designed for wastes from medical, agricultural, industrial, and research sources. The main data is taken from various projects completed during the period 1995–2000. The first of these projects was financially supported by the NATO Science Committee and was realized in the period 1995–1996 by a team from the Geological Institute of the Bulgarian Academy of Sciences (BAS), in collaboration with the British firm Quantisci Ltd. Our main goal in this project was to develop a site-selection methodology for deep HLW disposal in the marl terrain of Bulgaria.

The project, Radioactive Waste Management in Bulgaria (PHARE contract BG 9107-02-04-01), was developed during the period 1996–1997 with the collaboration of some western European companies, including CASSIOPEE, AEA Technology, SGN, and Ove Arup and Partners. Previous site-selection investigations have been analyzed, and a methodology for their continuation and expansion has been suggested. In the same period, a team of Bulgarian experts took part in two international projects (EC FI3PCT 930073 and COMETES ERBI C15 CT 98-0203) devoted to modeling radionuclide migration through groundwater (financially supported by the European Community). A project related to the site-selection investigations for LILW disposal (supported by the Bulgarian Committee for Use of Atomic Energy for Peaceful Purposes) was finalized in 1999. A site-selection methodology has been established in accordance with International Atomic Energy Agency (IAEA) recommendations, and some new potential sites have been identified in the Kozloduy NPP area. Selection of potential sites for deep geological disposal was continued through another project financially supported by the Bulgarian Ministry of Education and Science. Project activities were carried out in the period 1998-2000.

As a result of the last two projects, the area survey's site-screening phase has been completed, with four potential sites for LILW disposal and four for deep geological HLW disposal specified (Figure 6.1).

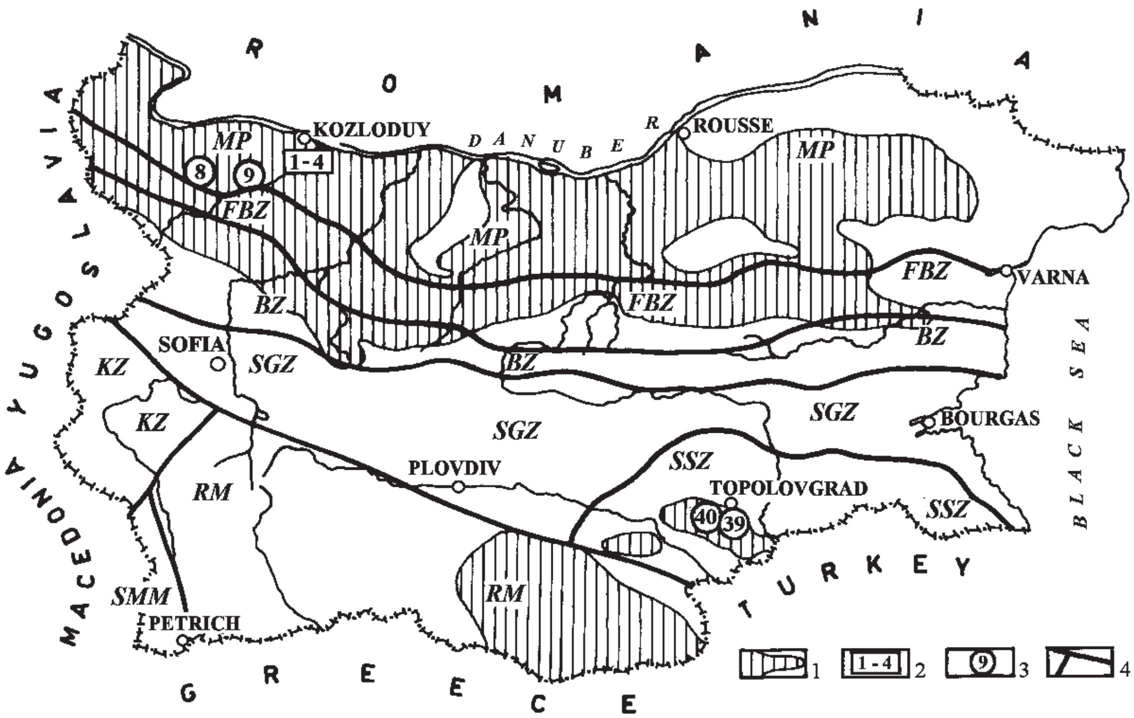

Figure 6.1. Prospective areas and sites for further investigation: (1) Prospective areas; (2) Potential sites in loess close to Kozloduy NPP;(3) Potential sites for geological disposal (Sites 8 and 9 in marl; Sites 39 and 40 in granite);(4) Boundaries between morphotectonic units according to E. Bonchev (1971).
Legend: MP-Moesin Plate: FBZ-Fore Balkan Zone:BZ-Balkan Zone; SGZ-Sredna Gora Zone; SSZ-Sakar -Strandzha Zone; KZ-Kraishte Zone; SMM-Serbian-Macedonian Massif; RM-Rhodope Massif

In 1999, the Bulgarian government approved a national strategy for safe management of spent nuclear fuel (SNF) and radioactive waste. This strategy includes deep geologic disposal of HLW and near-surface disposal of conditioned LILW. In the beginning of 2001, the Bulgarian government approved regulations establishing a fund for safe radioactive waste storage. In February 2001, the national LILW conditioning plant at Kozloduy NPP started operations.

6.2. SOURCES OF RADIO ACTIVE WASTE

The main source of radioactive waste (90% of the total volume) is from electric-power generation at the Kozloduy NPP. This site consists of six units: the first four units are WWER-440/230 reactors (each with a net capacity of 408 MW), while the last two, most recently built reactors are WWER-1000 types (each with a net capacity of 953 MW). Nuclear power is particularly important for the Bulgarian economy because the Kozloduy NPP produces about 45% of the entire nation's electricity. Construction of the new Belene NPP (with two WWER-1000/320 reactors) was suspended in 1990, principally because of the uncertainty surrounding the neotectonic setting of the site, but also because of the uncertainty over where and how to dispose of the radioactive waste. About 23,700 m^3 of conditioned waste will result from operation over the lifetime of the six reactors at Kozloduy (and one assumed new reactor at Belene). Decommissioning will lead to about 67,600 m^3 of conditioned waste, so the total volume of LILW is expected to be about 100,000 m^3. The conditioned LILW will be in the form of pre-stressed concrete cubes, known as BB-cubes, and they will be embedded in an interim storage facility on the site of waste production until a final disposal facility can be constructed.

The problems with spent nuclear fuel and its reprocessing are still unsolved. The volume of vitrified HLW for the whole lifetime of the NPP, according to some preliminary estimates, will reach about 310 m^3. An additional volume (about 1,800 m^3) of long-lived ILW, containing a significant amount of alpha emitters, will be generated during the reprocessing, and this ILWwill also have to be disposed of in a deep geological formation.

Another source of radioactive waste is the use of radioisotopes in industry: research, medicine, and ionizing fire detectors. (This waste constitutes about 10% of the total volume.) BAS has one research reactor (IRT-2000 with 2 MW net capacity). It was commissioned in 1961, but is now out of operation. This waste from this source (so far about 255 m^3) has been stored at the Novi Han facility.

Uranium mining waste is not considered in this report.

6.3. LILW REPOSITORY SITE SELECTION

6.3.1. A Brief History

The first investigations involving the selection of potentially suitable areas for LILW disposal were carried out at the end of the 1970s (Bonchev et al., 1980). This study included consideration of loess terrain and abandoned mines in Northwest Bulgaria, near the Kozloduy NPP. On the basis of the investigations, it was proposed to build an interim storage facility in the loess deposits close to the Kozloduy NPP. It was also decided to study the geological and hydrogeological conditions at the abandoned Smolyanovtzi uranium mine as a possible site for final disposal of conditioned radioactive waste. The possibilities for the LILW disposal in this loess area have been analyzed in more detail, resulting in the selection of two potential sites near the Kozloduy NPP (Stefanov et al., 1984). Several boreholes have been drilled on the potential loess sites, and some abandoned mines were investigated during the period 1986–1988.

The site-selection survey in Bulgaria was extended by the Concept for a National Radioactive Waste repository developed by BAS (Evstatiev and Kozhoukharov, 1995). Geological, seismotectonic, geomorphologic, hydrological, hydrogeological, geotechnical, and socioeconomic criteria (as well as additional criteria from abandoned mines) have been applied to the site-selection procedure. Forty sites and a significant number of abandoned mine shafts and galleries (about 280) were subjected to screening according to the defined criteria. The possible sites were then reduced to 18 and the number of abandoned mines to 2. Additional surveys were carried out, and more precise data were collected for all 20 potential sites. System analysis established that the investigated sites could be divided into three groups according to the degree of preference:

1. Group 1—sites consisting of Lower Cretaceous marls
2. Group 2—sites consisting of loess
3. Group 3—sites consisting of Neogene clays, the Smolyanovtzi uranium mine, and the Koshava gypsum mine.

Subsequent investigations (Evstatiev and Angelova, 1997) agreed that marl and clayey sites are the best prospects for near-surface LILW disposal, but that the loess deposits located near to the Kozloduy NPP offer great advantages if the problems with loess collapsibility are solved.

6.3.2. Potential LILW Disposal Sites near the Kozloduy NPP

6.3.2.1. Study Methodology and Geology of the Area

The possibilities for site selection near the NPP have been studied in all previous investigations with regard to public acceptance, costs, and risk of waste transportation. Expanded studies in the area of the Kozloduy NPP were carried out in 1998–1999. An assessment of the potential sites has been performed, using 22 criteria recombined into four groups:

1. Geological safety criteria
2. Engineering safety criteria
3. Environmental impact criteria
4. Socioeconomic acceptability criteria.

System analysis was used to set up a site-selection procedure and establish a rating of the potential sites.

The investigated area is located south of the Kozloduy NPP and covers about 150 km^2 (Antonov, 2000). This area is situated in the western part of the Danube plain. During the Pliocene period, the region was a fragment of the deep Lom Depression, which was gradually filled with lacustrine sediments. The recent relief has been shaped as a result of erosion, alluvial and aeolian sedimentation, and soil formation. These processes were manifested during the Quaternary. The main geomorphologic forms are Pliocene denudation surfaces (PDS), old erosion-accumulative levels (OEAL), Pleistocene, and Holocene (e.g., the Danube river and its tributary-

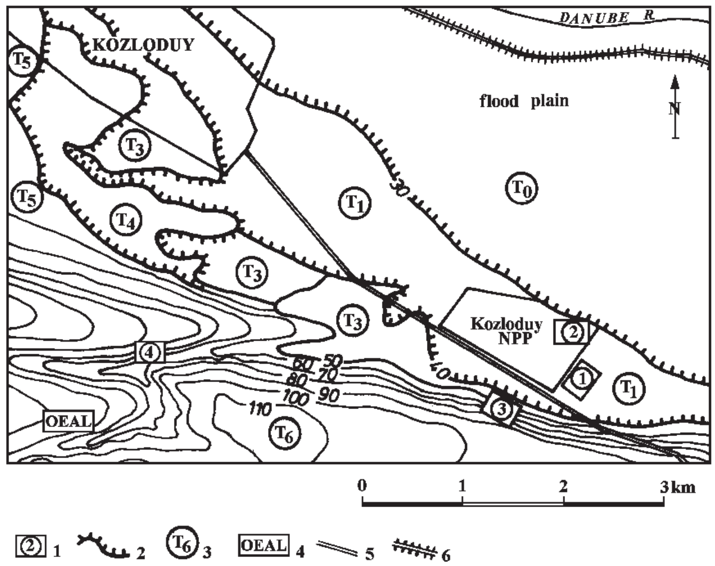

Figure 6.2. Geomorphology and potential sites of the Kozloduy NPP area: 1-Potential sites; 2-Boundary between river terraces; 3-Terrace number; 4-Old erosional-accumulative level; 5-Road;6-Dike (by Evstatiev, ed., 1999)

river terraces). All these relief forms are covered with thick aeolian loess deposits.

Five potentially suitable zones were identified during the regional-mapping phase. Ultimately, four sites were identified as preferable (after screening all sites using exclusionary criteria). Two of these sites are located next to the Kozloduy NPPon one loess-covered Danube terrace, the third site is located on a high Danube terrace south of the NPP, and the fourth site is located about 1 km south of the Kozloduy NPP, in the Marichin Valog tributary valley (Figure 6.2).

In the lithostratigraphic section of the Kozloduy NPP region, the two main rock complexes can be divided into (1) pre-Neogene lithostratigraphic-unit complexes and (2) Neogene and Quaternary lithostratigraphic-unit complexes. The pre-Neogene complex is situated at a depth of more than 600 m and will have negligible influence on the possible LILW near-surface disposal facility. The Neogene sediments are represented by six formations. The lower part of the Neogene consists of calcareous clays, limestones, and marls. The upper part consists of clays with sandy intercalation and lenses (Figure 6.3). The transitional zone between Pliocene and Quaternary sediments consists of clayey gravels and sandy gravels. Alluvial and aeolian deposits represent the Pleistocene. The complete aeolian loess complex consists of eight loess horizons divided by seven palaeosoils. The main conclusions from the analysis of the geodynamic evolution and tectonic setting of the investigated site area are as follows:

- The most important geodynamic event during the Neogene is the development of the Lom Depression (in the beginning of the Middle Sarmatian) that was developed from an older tectonic structure. The maximum depth of the Lom Depression reaches 900 m, and the greater part of the Depression extends into Romanian territory. It is filled mainly with lacustrine clay sediments intercalated by water-bearing sands.
- The Lom Depression ceased its development at the end of the Pliocene and the beginning of the Pleistocene. Since then, the investigated region has been characterized by stable tectonics. The seismic intensity of the region is degree VII, according to the MSK-64 scale.

High-magnitude earthquake focuses and active faults have not been established near the examined sites.

The absence of active faults can be verified by the uniform gravel elevations from the old erosion-accumulative

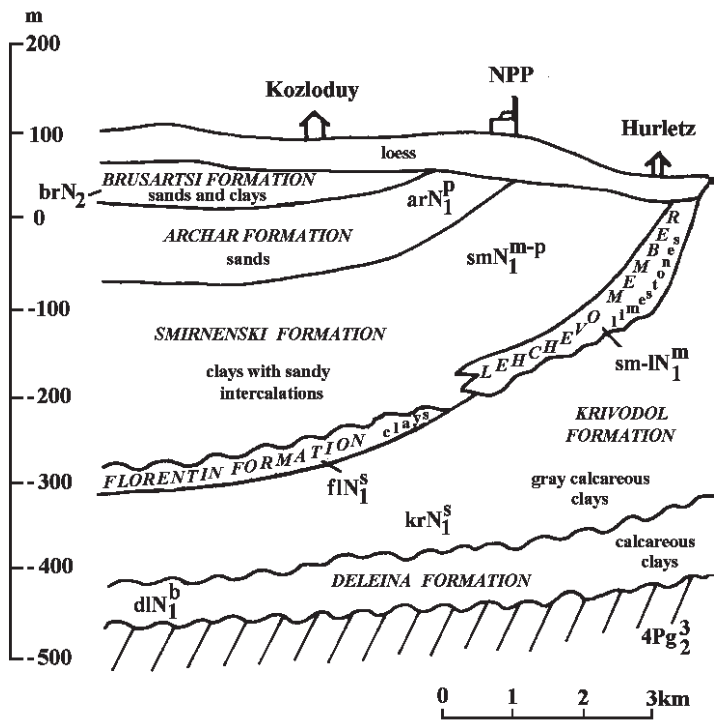

Figure 6.3. **Lithological and stratigraphical cross section of the Kozloduy NPP area (by Filipov et al., 1992)**

level (OEAL) (dated 2.59 Ma) in the vast area around the Kozloduy NPP. Displacement of the terrace base level on Bulgarian territory has not been determined.

6.3.2.2. Hydrogeology of the Area

The oldest Neogene aquifer in the Kozloduy NPPregion is located in the sediments of the Krivodol Formation, at a depth of about 400 m (Figure 6.3, Sarmatian). It is a confined aquifer with water discharge of up to 5.5 L/s. The next aquifer is located in the graded sands of the Archar Formation (Upper Pontian, Pliocene). It is isolated from the strata of the Krivodol Formation by clay deposits more than 200 m thick within the Smirnenski Formation (Meotian-Middle Pontian, Pliocene). Even if one postulates that a probable hydraulic connection exists between these formations, this connection could not be established within the borders of the sites under consideraton. In the Archar Formation, a confined aquifer has been recharged by some outcrops from the south and by the overlying Quaternary sediments. A spring located east of Kozloduy is characterized by water discharge of 18–20 L/s and a transmissibility of 100-150 m^2/d. The upper Neogene aquifer most important for the selection of a near-surface facility site was formed in a sandy intercalation of the Brusarci (Brusartsi) Formation (Dacian-Romanian, Pliocene). Its recharge is most likely realized through the overlying Quaternary sediments. The Brusarci aquifer has lower water productivity than the Archar aquifer and is characterized by a significant filtration anisotropy, with discharge values of 0.2–13 L/s and a transmissibility of 20–130 m^2/d.

The most important of the aquifers in the Quaternary sediments is an aquifer formed in the Holocene deposits of the Kozloduy lowland (where the NPP is located). Flood-plain-deposit thickness is about 20 m, and the Neogene horizons and Danube River supply their recharge. The transmissibility for the Holocene aquifer is considerable—1500–2000 m^2/d.

The hydrogeological setting of this investigated region is very complicated and needs additional surveys geared toward specific site-selection requirements.

6.3.2.3. Preliminary Characteristics of Potential Sites

Site N1 and Site N2

Both N1 and N2 are situated near the first Kozloduy NPP reactor. Their position is on the loess deposits of terrace T_1 at a surface elevation of 35 m, about 5–7 m above the mean water level of the Danube river.

The sites have a similar lithological structure. The soil base is aeolian loess, 10 m in thickness, with a collapsibility of 12% at a load of 0.3 MPa. Alluvial soils with a thickness of 6–8 m are deposited beneath the loess. They are represented by sandy-clayey sediments in the top of the layer and gravels in the bottom. Lacustrine Pliocene deposits of grayish-green and yellow clays (with thickness of 6–10 m) are beneath the alluvium. Pliocene sediments are composed of alternating clays and sands.

The lithological variations (and their geotechnical properties) of the Kozloduy NPPsoil base have been studied in detail. The alluvial deposits as well as Pliocene sands contain aquifers. There is a hydraulic connection between these aquifers and also with the Danube River. These aquifers are recharged when the river reaches a high water level. Predictions of radionuclide transport have been carried out (Galabov, 1992); the results concerning possible contamination of groundwater and the Danube River are very promising. Risk of flooding from the Danube River has been analyzed with reference to the safety assessment of the Kozloduy NPP. Different scenarios include the potential risk of flooding from existing and future water reservoirs. In the worst-case scenario, the groundwater level will reach to about 0.7–0.8 m beneath the present site surface. Raising this surface would protect the sites and the possible near-surface LILW repository against any hypothetical risk of flooding.

Site N3

Site N3 is situated about 500 m south of the Kozloduy NPP, on the loess deposits of terrace T_6 (Figure 6.2). At this site, the potential repository is projected to be a tunnel structure to be constructed in a bed of loess 40 m thick. Elevation of the slope bottom is about 35 m; top elevation is 100–110 m. The loess complex is composed of six horizons separated by five palaeosoils. Silt and sand prevail in the grain size distribution of the loess horizons. The palaeosoils consist of more clay-size particles. There are alluvial clayey gravels and sands with a total thickness of about 5 m beneath the loess. Lacustrine Pliocene sediments intercalated with sands are deposited under the alluvium.

Loess formations in Bulgaria are well understood from a geological and geotechnical point of view. They have also been previously investigated as a potential medium for a near-surface or tunnel-type LILW repository (Stefanov et al., 1984; Evstatiev and Angelova, 1997). Loess advantages and disadvantages in terms of radioactive waste disposal have also been analyzed (Evstatiev et al., 1998). From a hydrogeological point of view, loess is an unsaturated medium of low water content (about 8–12%). Only the top several meters of the loess cross section changes its water content under the influence of seasonal climatic variations. The lower part of the loess, with a thickness of about 30–33 m, is situated above the capillary zone in the alluvial gravels. Loess deposits with an unsaturated zone of considerable thickness (more than 90 m) are located about 30–40 km east of the Kozloduy NPP. They are also suitable for tunnel-type repository construction (Stefanov et al., 1984; Evstatiev and Angelova, 1997). Besides being a thick unsaturated medium, loess has other favorable properties: it has a homogeneous geological structure (that facilitates radionuclide migration modeling), it has good sorption properties, it can preserve a vertical slope and retain stable underground openings, and it is easily excavated.

The main difficulties with loess are its instability with changes in load, its susceptibility to settlement, and its water permeability (permeability in the studied area is $K_f = 0.5\text{-}1.5$ m/d). These disadvantages could be largely eliminated using ground improvement methods. Bulgaria has considerable experience in this field (Evstatiev, 1995): all six reactors of the Kozloduy NPP were constructed on an improved loess foundation.

Site N4

Site N4 is situated in the Marichin Valog tributary valley southwest of the Kozloduy NPP (Figure 6.2). The maximum depth of the valley is in its central part (about 30 m deep). The valley walls are relatively flat, with slope up to 10º. A rill that flows through the valley can grow into a stream after heavy rains or during snow melt. Agricultural land exists only at the upper and lower part of the valley.

The existing valley was derived from an old Pliocene valley covered with aeolian loess during the Pleistocene. Pliocene clays have been found at a depth of a few meters below the surface. Alluvial-colluvial sediments overlay them. The thickness of the loess cover gradually increases from the slope toe to the top. Generally, the loess cover is thinner and is characterized by lower collapsibility compared to the loess plateaus.

The valley (with its branches) represents a closed topographic basin, which provides the possibility for adequate radionuclide-transport modeling. A shallow, very productive aquifer is located in the alluvial-colluvial deposits.

The site is located at the lower, wider section of the valley where better conditions for repository construction exist. Part of the collapsible loess could be excavated, with the remaining loess soil subjected to engineered treatment. This site is more suitable than sites N1 and N2 because the improved loess and Pliocene clays are reliable barriers against radionuclide migration.

6.4. HLW REPOSITORY SITE SELECTION

6.4.1. Tasks and Methodology of the Investigation

Two main tasks have been defined in the investigations performed after 1998 (Kozhoukharov, ed., 2000). The first task was to differentiate potential sites in areas of Bulgaria where investigations had not been terminated in previous surveys. The second task was to compare the sites and to select the best potential sites for further investigation. Foreign achievements and experience in this field have been thoroughly analyzed during implementation. A methodology based on a set of site-selection criteria (as well as a system-analysis procedure for site classification) has been developed. This methodology conforms to the recommendations of the relevant IAEA documents from the Safety and Tecdoc Series, and has some general principles in common with the methodologies of other European countries (for example, Croatia). A methodology recommended by the European Community Commission was applied in the previous study.

The total number of investigated sites is 47: eight of them are in Miocene clays; five in Oligocene volcanites and tuffs; nine in Lower Cretaceous marls; six in Paleozoic granites; three in Pre-Cambrian gneissic-granites; five in Pre-Cambrian serpentinites; and 11 in Pre-Cambrian gneisses. Thirty sites have been selected using exclusionary (mostly geological) criteria. Additional investigations have been carried out, and sites have been evaluated and compared by means of a system-analysis procedure, based on the application of 28 criteria divided into five groups:

Group I.	Safety ensured by host geological formation
Group II.	Long-term stability of the geological medium
Group III.	Engineering safety of the repository
Group IV.	Environmental impact
Group V.	Socioeconomic acceptability of the site.

Final comparison of 30 potential sites led to choosing the following four sites: two sites composed of the Lower Cretaceous clayey marls in northwest Bulgaria and two sites in the Sakar granite pluton in the southeast part of the country. This conclusion does not differ in principle from the results obtained in previous investigations (Kozhoukharov et al., 1996; Evstatiev and Kozhoukharov, 1998). However, this recent study identifies specific sites and contains more detailed characterization, according to a defined set of 28 criteria.

6.4.2. Potential Sites in the Lower Cretaceous Marls

In the last few years, nine sites in the Lower Cretaceous marls of the Fore-Balkan have been subjected to additional surveys (Nikolov and Ruskova, 2000). The data from the surveys prove once more that the sites selected before, Sumer and Varbitsa, offer the most appropriate conditions for HLWdisposal. Both sites are about 50–55 km south of the Kozloduy NPP. They will be described together because they possess similar conditions.

Sumer and Varbitsa

The Lower Cretaceous marls of the Sumer Formation are a potential host medium for the proposed repository. Their thickness is about 750 m at one site and 1,000 m at the other. Amap of the formation thickness and geological profiles of the sites were already published in the *Second Worldwide Review* (Kozhoukharov et al., 1996). The formation mainly consists of clayey marls and rare thin-layered sandstones. The marls are composed of clay minerals, calcite, and fine-grained terrigenous components such as quartz, muscovite, and biotite. Silt (32–50%) and a clay fraction (26–43%) are the predominant components. This composition of marls translates to good sorption properties. The values of unconfined compressive strength vary between 11 MPa and 29 MPa, depending on sample depth and degree of weathering.

The total annual precipitation is 600–800 mm. An aquifer of low productivity has been formed in the superficial weathered marl layer. Generally, marls are virtually water impermeable, and in hydrogeological publications they are known as Lower Cretaceous water-impervious layers. The Upper Jurassic-Lower Cretaceous artesian aquifer is situated at a depth of 1,000–1,500 m.

Rivers with deep erosion (owing to climatic variations during the Quaternary) do not cross the area under investigation. Some old faults without any evidence of Young Alpine tectonic movements have been established at a distance of 7–10 km from the sites. During the Neogene and Quaternary, the investigated region was tectonically stable. No risk of flooding exists at these sites.

Hydraulic tunnels have been constructed without any difficulty in the marls near to these potential sites. However, we have no mining experience at great depths in marl deposits. The sites are located in a region with seismic intensity of degree VII on the MSK-64 scale. Seismicity is not connected with fault structures of the area. Elevations in this region are about 250–300 m, which offer good conditions for construction of an infrastructure network. Other surface processes besides moderate weathering and surface erosion have not been observed. Mining operations and other industrial activities have not affected the areas near the sites.

Mineral resources have not been established in the area around the sites. A potable reservoir is located about 12 km from the Sumer site, but topographic and hydrogeologic features preclude the potential risk of its contamination. Agricultural land does not exist near the sites. The transportation of the conditioned radioactive waste from the Kozloduy NPP will not provoke adverse environmental effects. Safety assessment of the sites has not been performed, but preliminary expert evaluations of the geological and hydrogeological setting indicate no risk of environmental pollution.

Population density is 20–40 people per 1 km^2 in the site area. Close to the site are several small villages with negative population increases. The region has a well-developed infrastructure in need of only minor restoration. It is not expected that repository construction will disturb other economic and industrial activities in the area. The local population has no experience with the nuclear industry.

In summary, both marl sites are suitable not only for a deep HLW disposal but also for a near-surface LILW repository. This means that the possibility exists of constructing both types of repositories in the same place.

6.4.3. Potential Sites in the Sakar Granites

Six sites in the Paleozoic medium-grained granites of Sakar pluton have been studied (Kozhoukharov, ed., 2000). The geological structure of the Sakar region (as well as its granitic characteristics) had been described during the last years (Kozhoukharov et al., 1996; Evstatiev and Kozhoukharov, 1998). As a result of the these investigations, several sites with similar conditions have been located (Kozhoukharov, ed., 2000). Two of these sites—Garvanski Kamak and Sakartzi—have some advantages over the others; their conditions will be further discussed together.

Garvanski Kamak and Sakartzi

Strong, dense, equigranular granites of the Sakar pluton make up the host rock at these sites. They are distinguished by massive fabric and hypidiomorphic granular texture. Grain size is between 0.6 mm and 4-5 mm. The mineral composition consists mainly of plagioclase (oligoclase), orthoclase (microcline) and quartz. Other components of the mineral composition include biotite, muscovite, and some accessory minerals such as apatite, sphene, zircon. The bulk density of granite is 2.62 g/cm^3, the density of solid particles 2.70 g/cm^3, the porosity about 3%, the absorbed water content 0.35%, and the unconfined compressive strength about 120–140 MPa.

The granite is fissured with three sets of joints:

- The first set has a strike 120°–130° and an angle of dip 45°–50° in a northeast direction. The joints are poorly defined and spatially discontinuous, with density of 1–2 fissures per 1 m.
- The second set has a strike 30°–40°, with vertical to subvertical fissures transversal to the first set. They are relatively persistent, with density of 1 fissure per 1–2 m.
- The third set has a strike 120°–130° and an angle of dip 40°–55° in a southwest direction. The joints are poorly defined and discontinuous, with a density of 2–3 fissures per 1 m.

Deep hydrogeological boreholes have not been drilled in the granite batholith. Data exist only for groundwater in the granite-weathered zone. Water is discharged by springs located in gullies outside the site area. Rate of discharge is from 0.02 L/s to 0.005 L/s. Groundwater is recharged only by precipitation (total annual rainfall is about 637 mm).

Our investigations indicate that the granite rock at both sites is a suitable host medium for deep radioactive waste disposal. The analysis of the topographic and tectonic features of the sites as well as the climatic evolution during the Quaternary period suggest that the isolation capability of the deep disposal system would not be disturbed by erosion processes in the next one million years. There is no danger of flooding at the sites.

Mining experience in this area indicates that the granites possess good conditions for underground construction. These sites are located in a region with seismic intensity of degree VIII on the MSK-64 scale. The elevation at the Garvanski Kamak site is about 500 m; the elevation at the Sakartzi site is 425 m. The site area is characterized by middle-height relief and the topography has not been dissected by erosion. Surface processes other than moderate weathering have not been observed. Mining installations do not exist under the sites, nor are these sites endangered by other human-induced events.

Significant mineral and water resources have not been encountered in the area around the sites. Low deciduous forests occupy the investigated region. Both sites are situated about 300 km from the Kozloduy NPP. The well-developed infrastructure of the country provides unimpeded access to the sites. National parks, historical monuments, and natural heritage sites do not exist near the site area. Repository construction and operation will not seriously damage the flora and fauna of the region, if appropriate measures are taken to protect some rare species (e.g., tortoises). Inadmissible radiation effects on the local population are not expected, given the existing geological and hydrogeological conditions.

The Sakar region has one of the lowest population densities in Bulgaria—less than 20 people per 1 km^2. There are some small villages about 10 km around the sites. The shortest distance between the Garvanski Kamak site and the largest village is 7 km; between the Sakartzi site and the same village, it is 1 km. The local population has some previous nuclear experience because of mining operations at a nearby uranium mine. Local infrastructure is not well developed; the road leading to the sites is of poor quality. Economy and industry are poorly developed in this region, and possible repository construction will only improve this situation. The distance between these sites and the Turkish border is about 25 km.

6.5. GEOLOGICAL SETTINGS OF THE NOVI HAN STORAGE FACILITY SITE

The only operating radioactive waste facility in Bulgaria is situated near the capital, Sofia, in Lozen Mountain. A shallow-type facility, it has been functioning since 1964 and is designed for radioactive wastes from medical, industrial, agricultural, and research sources. Evstatiev et al. (1994) and Evstatiev and Kozhoukharov (2000) have described its geological settings.

The facility is located close to the watershed of the Lozen Mountain at an elevation of 920 m. Surface slopes are inclined at 13–16% in the N-NE direction. The nearest inhabited locality is 4 km from the storage site. The geology of the area consists of Pre-Cambrian, Lower Paleozoic, Upper Paleozoic, Lower Triassic, Neogene and Quaternary deposits, as well as several magmatic bodies of granitoides and lamprophyre dikes. The storage site consists of Ordovician phyllite-schists and phyllites. As a result of weathering and tectonic processes, they have been transformed to clayey mylonites in some zones. The thickness of the weathered phyllites is about 5–7 m.

The area of the facility is situated in the Maritza fault zone. The most important fault is the Iazdirastovo fault passing north of the site. There is geological and geomorphologic evidence for the rise of the Lozen Mountain from the Miocene until now. Permanent earthquake epicenters in the Sofia seismic zone verify recent tectonic activity. The site is located in a very complex tectonic structure, with intensive tectonic activity after the Neogene. As a consequence, the rock mass has been greatly deformed and divided by faults, with rocks broken, fissured, and turned into mylonites in some zones.

The soil base of the facility is composed of a weathered and fissured phyllite layer 3.0–4.5 m in thickness. The bearing capacity of the weathered phyllite is calculated to be 0.3–0.4 MPa, so it may be assessed as a suitable foundation rock for existing bunkers and shafts. Fissured phyllites have been located at a depth of more than 5.5 m under the surface. The bearing capacity of the strongly fissured phyllites is 0.5 MPa and is 1.5 MPa in the less fissured ones. High seismicity is the main hazard at this facility. According to the seismic zoning of Bulgaria, over a period of 1,000 years, the site has been located close to the boundary between one region with a seismic intensity of degree VIII and another of degree IX on the MSK-64 scale. Groundwater in the Paleozoic rocks circulates in fissures and is recharged mainly by precipitation. The Ordovician phyllite schistosity determines the anisotropy of the water permeability. The groundwater table is at a different depth from the surface—from 6–7 m to 15–16 m. In a recent pump test, the permeability was found to be 0.06 m/d–0.7 m/d. Unfortunately, the main direction of the fissures coincides with that of the slope inclination.

The radioactivity log of the rocks, the gamma-ray logs of boreholes, and the laboratory radiochemical analyses of the water from springs located in the repository area have not detected any radioactive contamination (Evstatiev et al., 1994). Data from investigations carried out up to now are insufficient to make a safety assessment of the storage facility. Its location near Sofia requires the performance of additional studies, mainly in the fields of hydrogeology, seismotectonics, and neotectonics.

6.6. CONCLUSION

The site-selection procedure in Bulgaria is still at a preliminary stage, despite the progress achieved in site-selection work. Recent investigations corroborate the finding that the Lower Cretaceous marls are the best host rock for a deep HLW repository. Next best are the sites located in the Sakar granite pluton. The marls have the advantage of being suitable not only for deep HLW disposal, but also for a near-surface LILW repository. This means that both repository types could be constructed at the same place. The loess-clay terrain near the Kozloduy NPP is also desirable as a medium for the near-surface LILW repository.

Site-selection studies have been delayed in Bulgaria because of the economic situation of the country, the lack of a radioactive-waste-management regulatory body, and lack of funds for additional geological surveys and specific investigations.

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Chapter 7

Long-Term Management of Nuclear Fuel Waste in Canada: Technical Developments in the Concept for a Deep Geologic Repository

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ABSTRACT . Since 1978, Atomic Energy of Canada Limited (AECL) and more recently Ontario Power Generation (OPG) have been developing the concept for emplacement of nuclear fuel wastes in a deep geologic repository excavated in plutonic rock. In 1994, AECLsubmitted its Environmental Impact Statement on the concept for review by a federal Environmental Assessment Panel. This review included input from government agencies, non-government organizations, and the general public. 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 to meet current regulatory requirements. However, the review also concluded that there wasn't sufficient public support at this time to implement a repository siting program. Federal government policy expects the producers and owners of nuclear fuel waste to establish a waste management organization. OPG is planning to establish a waste management organization in conjunction with other waste owners in Canada. The waste management organization will study and compare options for the long-term management of radioactive waste and recommend its preferred alternative to the federal government. In the interim, OPG is continuing to maintain the deep-geologic-repository technology for used fuel and to address technology gaps identified in the federal review of the AECLconcept. Since 1996, the technical program has focused the research and development activities in geoscience, safety assessment, licensing and approvals, and repository engineering. Significant advances have been made in rock-mass characterization, understanding groundwater flow and solute transport, safety-assessment modeling, and advancing conceptual designs for an underground repository in plutonic rock.

7.1. CANADIAN NUCLEAR FUEL WASTE MANAGEMENT—HISTORICAL REVIEW

In Canada, used fuel is stored in water-filled pools and, more recently, in steel-lined concrete dry storage containers. Current storage practices, while safe, require continuous institutional controls such as security measures, monitoring, and maintenance. Thus, storage is an effective interim measure for protection of human health and the natural environment, but is not a permanent solution.

In 1978, the governments of Canada and Ontario established the Canadian Nuclear Fuel Waste Management Program as a step towards the safe and permanent management of nuclear fuel waste. Responsibility for research and development of the concept for used-fuel emplacement in a deep underground repository within plutonic rock was assigned to Atomic Energy of Canada Limited (AECL). Responsibility for studies and technology development for interim storage and transportation of used fuel, plus technical assistance to AECL in repository development, was assigned to Ontario Hydro. In 1981, the governments of Canada and Ontario announced that site selection for a repository would not be undertaken until after the concept had been accepted.

Over the last 20 years, Canada has developed a broad range of technology and expertise for used-fuel emplacement in a geologic repository, which includes the following:

- Researching used-fuel dissolution and radionuclide release mechanisms
- Evaluating material properties and failure mechanisms of the candidate container materials, titanium and copper
- Optimizing engineered-barrier performance of clay-based buffers and backfills
- Developing high-performance cement and clays for vault seals
- Preparing conceptual designs of underground repository and surface facilities
- Developing site-characterization methods and tools applicable to surface-based and underground investigations of plutonic rock
- Developing underground construction techniques and understanding rock mass properties and behavior in high-stress regimes in plutonic rock
- Advancing the understanding of groundwater flow and contaminant transport in fractured plutonic rock
- Conducting experiments and repository demonstrations in plutonic rock at the Underground Research Laboratory, located in southeastern Manitoba
- Understanding the behavior and transport of contaminants in the surface biosphere, and their potential impact on human and nonhuman biota
- Developing detailed and integrated performance assessment methods, models, and computer software to assess the safety of the deep-geologic-repository concept
- Developing alternative safety-assessment methods, models, and tools to complement the detailed safety assessment models.

AECL constructed the Underground Research Laboratory (URL) near Lac du Bonnet, Manitoba in a large, previously undisturbed granitic pluton. The site lease allows underground excavation, drilling, and testing for research purposes. The surface facilities were constructed in 1984, and the underground excavations to the 420 m level were completed by 1989. AECL's URL has provided the Canadian repository program with a unique research facility for developing site-characterization methods, developing *in situ* stress measurement techniques, demonstrating full-scale container emplacement, evaluating rock excavation methods and rock opening stability, modeling groundwater flow and contaminant transport, and conducting grouting and tunnel-sealing experiments. The URL has also played host to a number of co-operative international projects with Sweden, Japan, France, and the U.S.A.

In 1994, AECL submitted its Environmental Impact Statement (EIS) (AECL, 1994) on the concept for review by a federal Environmental Assessment Panel. This review included input from government agencies, nongovernment organizations, and the general public. Public hearings associated with the review took place over 1996 to 1997. The submission included two post-closure safety assessments associated with the concept. The first assessment, or "EIS case study," evaluated the performance of a vault design consisting of titanium containers placed in boreholes drilled in the floor of the deposition rooms at a depth of 500 m, in a hypothetical geosphere based on site-characterization data obtained during development of AECL's URL near Lac du Bonnet, Manitoba (Goodwin et al., 1994). The second assessment, commonly referred to as the "Second Case Study," evaluated the performance of copper containers placed in an in-room configuration at a similar depth, using a more permeable geosphere (Goodwin et al., 1996). These two repository concepts are illustrated in Figure 7.1.

The Panel's report, which contained a number of recommendations, was submitted to the Canadian federal government in March 1998 (CEAA, 1998). The federal government responded to the Panel report in December 1998 (NRCan, 1998). In its response, the federal government concluded that the AECL concept for a deep geologic repository of nuclear fuel waste was technically safe and meets current regulatory requirements, in that it can provide passive safety in the long term. However, the federal review also concluded that there wasn't sufficient public support at this time to implement a repository siting program.

In response to the Panel recommendations, the federal government report also announced the following responsibilities for the future management of used fuel (NRCan, 1998):

- The producers and owners of nuclear-fuel waste in Canada will establish a waste-management organization, incorporated as a separate entity, with a mandate to manage and coordinate the full range of activities relating to the long-term management, including disposal, of nuclear fuel waste.
- The waste-management organization will have a board of directors representative of nuclear-fuel-

waste producers and owners; have an advisory council; and allow for the participation of all nuclear-fuel-waste producers and owners.

- The producers and owners of nuclear-fuel waste in Canada will establish a fund to fully finance all activities and operations of the waste-management organization, including the costs of developing and comparing waste-management options, and for designing and siting the preferred approach for long-term management, including disposal.

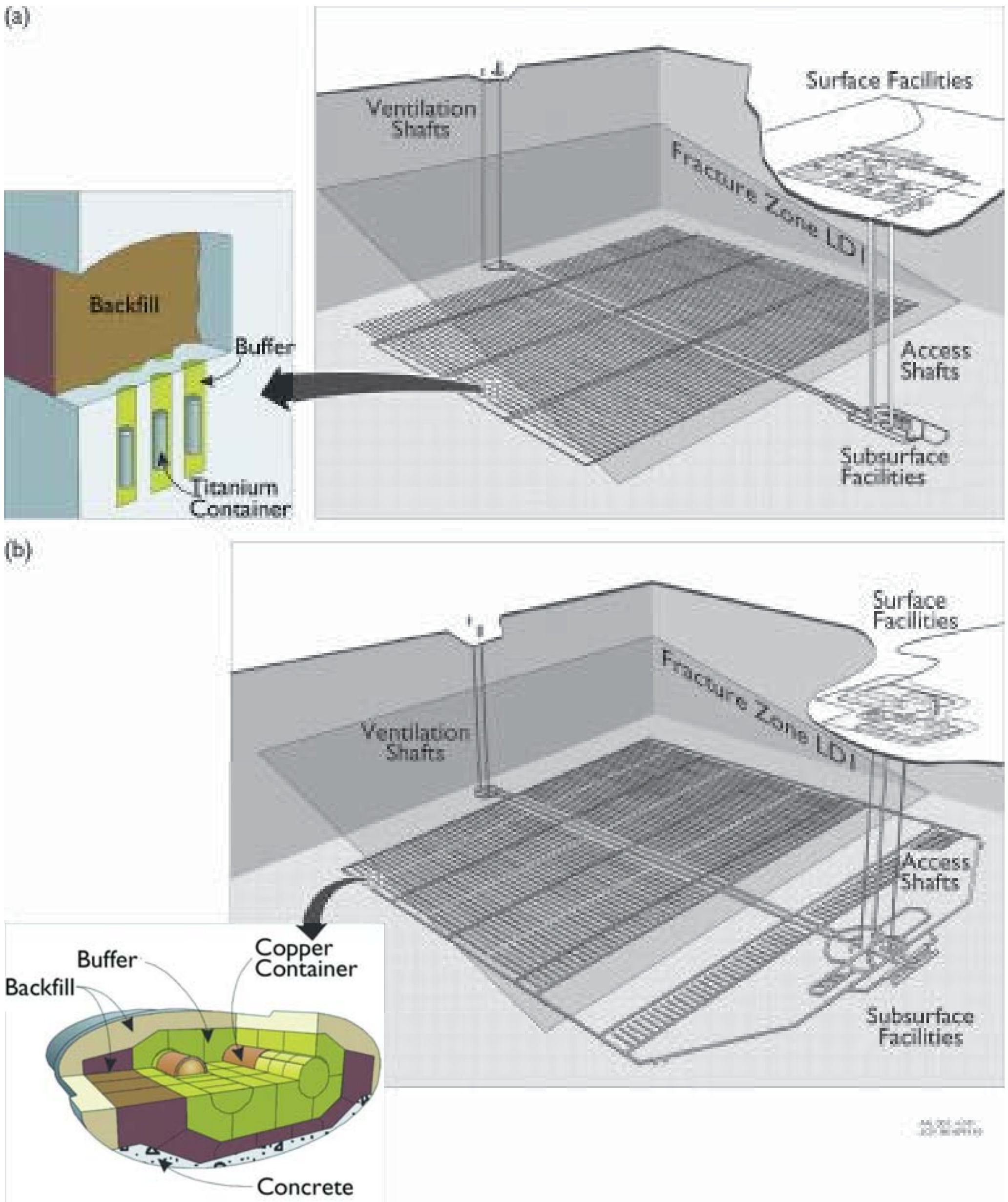

Figure 7.1. Illustration of two hypothetical used-fuel repositories: Part (a) depicts the repository that was evaluated in the EIS case study (borehole emplacement of titanium containers); Part (b) illustrates the repository that was evaluated in the Second Case Study (in-room emplacement of copper containers). The insets show details of the repository deposition rooms.

- The waste management organization will evaluate practical long-term waste-management options in Canada, which include the following:
  - a modified AECL concept for deep geological disposal;
  - storage at reactor sites;
  - centralized storage, either above or below ground.
- The waste-management organization will report to the Government of Canada, setting out its preferred approach for the long-term management, including disposal, of nuclear-fuel waste, with justification.

The government also adopted a Panel recommendation that the waste-management organization should review all of the social and technical issues associated with the AECL concept that were raised at the EIS hearings, and that this review should start as soon as possible. The government indicated that further development of the deep geologic repository concept should be undertaken at the URL.

In April 1996, Ontario Power Generation (OPG, formerly Ontario Hydro) assumed the lead national role in providing program direction and funding for the continued research and development of the repository concept. This is consistent with the 1996 federal policy framework for radioactive waste management (see Section 7.2) and also reflects the fact that OPG owns 90% of the used fuel produced in Canada.

OPG is committed to the long-term management of nuclear-fuel wastes in an environmentally, socially, and financially responsible way. By the end of 2000, the Canadian Nuclear Fuel Waste Management Program and OPG had spent over $800 million for research on long-term nuclear-fuel waste management. OPG itself has invested over $100 million in the deep-geologic-repository technology program to complete the federal hearing process, address outstanding technical issues, and maintain the technology and expertise required to initiate repository siting, pending the outcome of the long-term management options study and the federal government decision.

OPG is planning to establish a national waste-management organization in conjunction with the other nuclear-fuel waste owners (Hydro-Quebec, New Brunswick Power, and AECL). This waste-management organization will study and compare options for managing radioactive waste, including emplacement in a deep geologic repository, and recommend the preferred alternative to the federal government.

7.2. POLICY AND REGULATORY FRAMEWORK

The federal government announced a radioactive-waste-policy framework in July 1996, which specifies the roles of government and waste producers in the long-term management of radioactive waste in Canada. The major elements of the framework include the following:

- The federal government will ensure that radioactive-waste disposal is carried out in a safe, environmentally sound, comprehensive, cost-effective, and integrated manner.
- The federal government has the responsibility to develop policy, to regulate, and to oversee producers and owners to ensure that they comply with legal requirements and meet their funding and operational responsibilities in accordance with approved waste-disposal plans.
- The waste producers and owners are responsible, in accordance with the principle of "polluter pays," for the funding, organization, management, and operation of disposal and other facilities required for their wastes. This policy recognizes that arrangements may be different for nuclear-fuel waste, low-level radioactive waste, and uranium mine and mill tailings.

Under the Nuclear Safety and Control Act, the Canadian Nuclear Safety Commission (CNSC), formerly the Atomic Energy Control Board (AECB), regulates the emplacement of radioactive waste in a deep geologic repository. Under the regulations, licenses are required from CNSC for all phases of a project (i.e., site preparation, construction, operation, decommissioning, and abandonment). A provision in the Nuclear Safety and Control Act requires a financial guarantee from the licensee for all phases of a project when a license is issued.

Regulatory guidance on requirements for a used-fuel repository is provided in regulatory policy statement R-71 and R-104 (AECB, 1985; 1987). Presently, CNSC staff are reviewing their regulatory guidance in this area.

Application for a CNSC license generally triggers a review under the Canadian Environmental Assessment Act (CEAA). Under the CEAA, an environmental assessment is required to assess the environmental effects of most projects requiring federal action or decisions. Public input from comments before and after submission of environmental assessment documents, and from public hearings, will be required at an appropriate stage.

In addition to federal licensing and environmental assessment, provincial environmental assessment may be required. Harmonization agreements between the federal government and the provinces are in place or under negotiation to ensure that a joint panel review can be used to avoid duplication. Furthermore, radionuclides released from nuclear facilities are currently being assessed by Environment Canada under the Canadian Environmental Protection Act (CEPA). Demonstration that the environment is protected may be required under CEPA as well as under the CNSC regulations.

7.3. STATUS OF TECHNOLOGY FOR A DEEP GEOLOGIC REPOSITORY OF USED FUEL

The main objectives of OPG's Deep Geologic Repository Technology Program (DGRTP) are:

1. To further develop used-fuel repository technology by closing technology gaps that require long lead times, including those identified by reviewers of the Environmental Impact Statement, submitted in 1994, on the deep geologic repository concept developed by AECL
2. To maintain repository expertise supporting the review of options for long-term used-fuel management
3. To maintain sufficient expertise to initiate a repository-siting program, should the federal government endorse proceeding with this option following the review of long-term used-fuel management options.

OPG has developed a number of technical work packages to meet these objectives, pending the planned formation of a waste management organization. The technical program of the DGRTP conducts research and development activities in geoscience, safety assessment, licensing and approvals requirements, and engineering. Technical program activities are summarized in an annual progress report (Gierszewski et al., 2001). Staff at OPG, AECL, Kinectrics (formerly Ontario Power Technologies), and other external consultants performed these activities. A list of all previous technical reports is available in Garisto (2000). In summary:

- Geoscience work-program activities continue to develop and advance site-characterization and performance-assessment methodologies relevant to demonstrating long-term repository performance within crystalline rock. Geoscience activities examine and explore site screening and evaluation techniques, including geologic, geophysical, hydrogeochemical and hydrogeologic characterization of fractured crystalline flow systems. Experimental programs, both within the laboratory and underground, focus on advancing the understanding of groundwater flow and solute transport in fractured crystalline rock. For example, the geoscience program is conducting a number of *in situ* and laboratory experiments at the URL to derive a statistically valid database of pore-water diffusivities for low-permeability, sparsely fractured granitic rock.

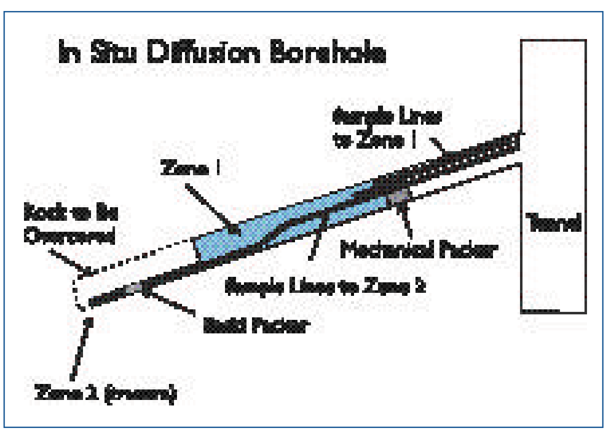

Figure 7.2. Schematic of *In Situ* Diffusion Experiment bore design and packer system configuration

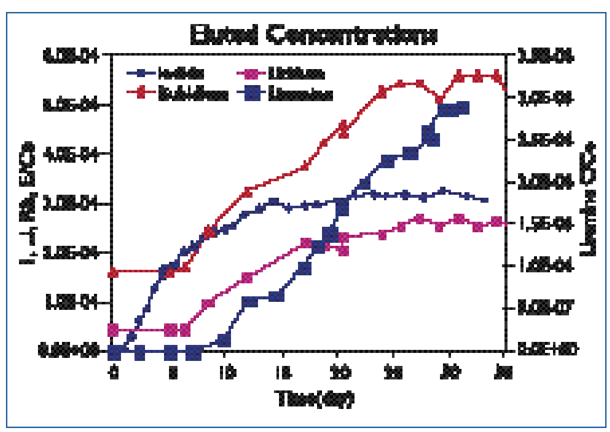

Figure 7.3. *In Situ* Diffusion Experiment Phase II—Steady-state diffusion cell breakthrough curves

The *In Situ* Diffusion Experiment is designed to explore issues of scale dependency and anisotropy, as well as the influence of sample volume, texture, structure, mineralogy, pore geometry, fracture infilling, and stress regime on estimated pore-water diffusivities (see Figures 7.2 and 7.3). Also, predictive mathematical models for simulation of mass transport are being evaluated for their applicability in geosphere performance assessment. Groundwater flow and transport data are being collected from the Moderately Fractured Rock Experiment at the URL.

- Safety-assessment work-program activities include experimental work to improve our understanding of spent-fuel dissolution rate, studies of nuclide transport in moderately fractured rock, and experiments concerning the long-term performance of buffer materials and seals. Work in safety analysis includes development of models in support of these experiments, as well as development of alternative safety-assessment codes for the deep-geologic-repository system. For example, the safety-assessment program has developed a simpler alternative model and code for the repository system. The Radionuclide Screening Model (RSM) is a sequential one-dimensional model that can simulate one stylized pathway for radionuclide release from the vault, transport through the geosphere, and discharge to a surface well in the biosphere (Goodwin et al., 2000; Garisto et al., 2000; 2001). RSM's main purpose is to provide an objective screening of all the radionuclides that could be released, to identify those radionuclides that need to be considered in more detailed safety-assessment models. It can also provide a simpler model for exploring key features of the repository concept. It runs under the SYVAC3 executive and is based on the system models used in SYVAC3-PR4. Figure 7.4 illustrates the agreement between RSM 1.0 and the reference SYVAC3-PR4 system model for the case of the three most important nuclides in the Second Case Study (SCS). For this comparison, water-ingestion-dose results are compared considering only the most important pathway from the vault to the biosphere (the full SYVAC3-PR4 model includes multiple pathways). In addition, safety-assessment work is continuing on quality assurance testing of existing models and codes, and providing

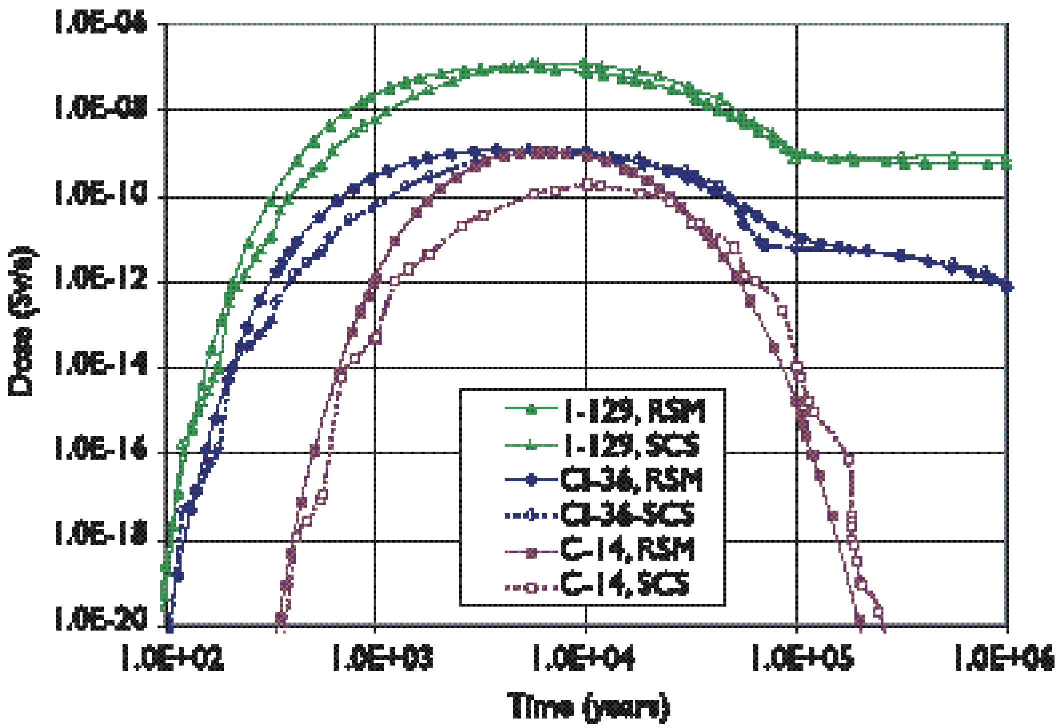

Figure 7.4. Comparison of water ingestion dose calculated by Radionuclide Screening Model (RSM) Version 1.0 with that from the Second Case Study (SCS) for an assumed failed canister in Vault Sector 11, for a hypothetical geosphere in crystalline rock.

safety-assessment support for geoscience and repository engineering.

Licensing and approvals work has included compiling a database of technical comments from the federal hearings, as recommended by the Panel and accepted in the government response. The 1,877 technical comments identified have been categorized in 291 categories to assist in reviewing the issues and identifying the work required to address them.

The repository-engineering work program continues to conduct research and development activities, which have focused on engineered barriers, the characteristics and performance of the rock mass, and underground facility engineering. Underground activities have focused on establishing methods and models for measuring the properties of rock mass. Most of this work has been done in the context of the physical and chemical environment of the sparsely fractured rock, in many cases using the particular characteristics of the Lac du Bonnet batholith, the site of AECL's URL. The repository engineering program recently recommended oxygen-free, phosphorous-doped copper as the reference corrosion-barrier material for used CANDU fuel containers (Maak, 1999). A screening study is underway to examine various container geometries, capacities, and emplacement methods. The results of the screening study will be used to support the development of a reference design for a deep geologic repository in crystalline rock. In addition, the repository engineering program is examining the sealing requirements and repository layout implications for the in-floor borehole and in-room emplacement configurations (Baumgartner, 2000). These configurations are illustrated in Figures 7.5 and 7.6, respectively. Further engineering studies and assessments are planned to evaluate the feasibility of these container emplacement configurations.

7.4. INTERNATIONAL ACTIVITIES

An important part of the OPG's repository development program is interaction with the corresponding research and development activities conducted in other countries with similar radioactive waste management programs.

OPG has formal agreements with Sweden (SKB) and Finland (Posiva) to exchange information arising from their respective programs on nuclear waste management. These countries are following used-fuel repository concepts that are very similar to the Canadian concept, and their programs are quite advanced. OPG holds bilateral co-operation and information exchange meetings with SKB and Posiva. OPG is also conducting material property tests on a prototype copper-canister shell material provided by SKB.

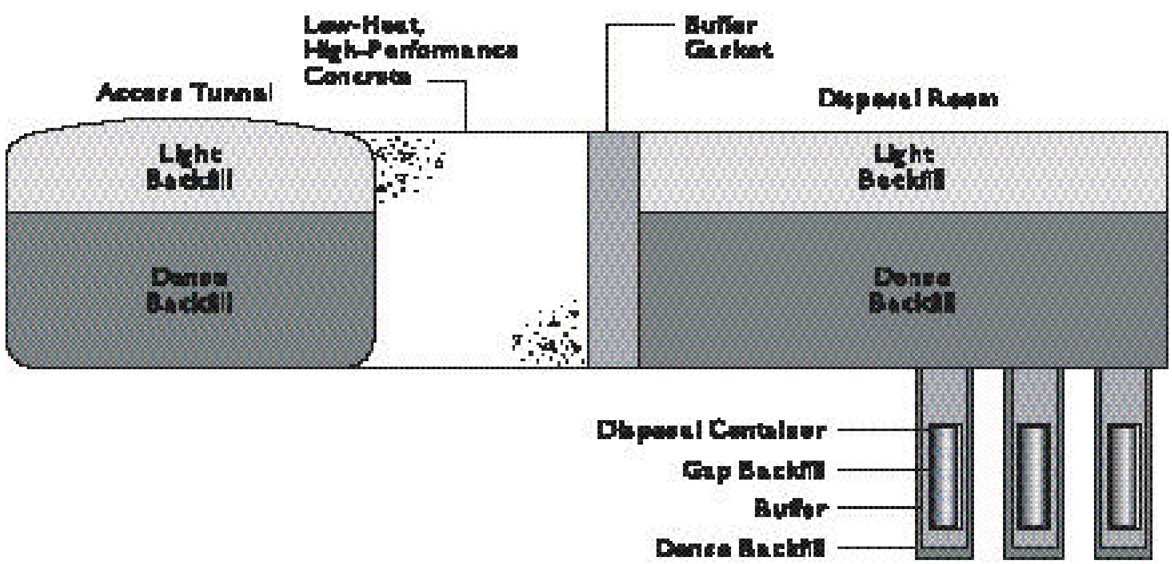

Figure 7.5. Potential arrangement of engineered barriers applied to the in-floor borehole emplacement method

OPG continues to participate in the international radioactive waste management program of the Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA), co-ordinated by the Integration Group for the Safety Case (IGSC). Both OPG and the CNSC have representatives on the IGSC. Participants in this group include all the major nuclear energy countries, both waste owners and regulators. Participation in the NEA working groups, and their special program groups such as the Expert Group on Integrated Performance Assessment (IPAG) and GEOTRAP, have been useful in providing direction to the Canadian technical program.

OPG is a funding member of the International DECOVALEX III Project and is actively supporting research on the application of coupled thermal-hydraulic-mechanical numerical models in the geosphere. OPG is also a funding member of the NEA FEPs Database group. This activity provides an international reference point for identifying factors that could affect a used-fuel repository, the first step in a safety analysis.

OPG also participates in international conferences and workshops. In recent years, OPG staff and contractors presented results and/or participated in the following meetings: the 1999 Spent Fuel Workshop, the 2000 NEA IPAG-3 Workshop on Approaches and Arguments Used to Evaluate Confidence in the Long-Term Safety and the Overall Results of IPAs, the Canadian Microbiology Conference, the POSIVA Salinity Effects Workshop, the Prototype Repository Kick-off Meeting, the Helsinki Copper Workshop, and the SKB International Seminar on the First Stage of the Tracer Retention Understanding Experiment (TRUE-1).

OPG is a self-funded participant with the Commission of European Communities' Cluster RepOsitory Project (CROP)—A Basis for Evaluation and Developing Concepts of Final Repositories for High Level Radioactive Waste, along with the radioactive waste-management organizations in Sweden, Belgium, Finland, Germany, Spain, France, Switzerland, and the United States. Starting in 2001, participants in the three-year-project will examine design, construction, and performance-assessment issues in underground facilities, and share lessons learned.

7.5. CONCLUDING REMARKS

The technology associated with developing the concept for emplacement of nuclear fuel wastes in a deep geologic repository excavated in plutonic rock has advanced considerably over the past several decades as a result of research and development efforts in Canada and internationally. Efforts are being directed to support the safety case for the geologic repository concept and to provide confidence to support the decision-making process (NEA, 1999). OPG is committed to the long-term

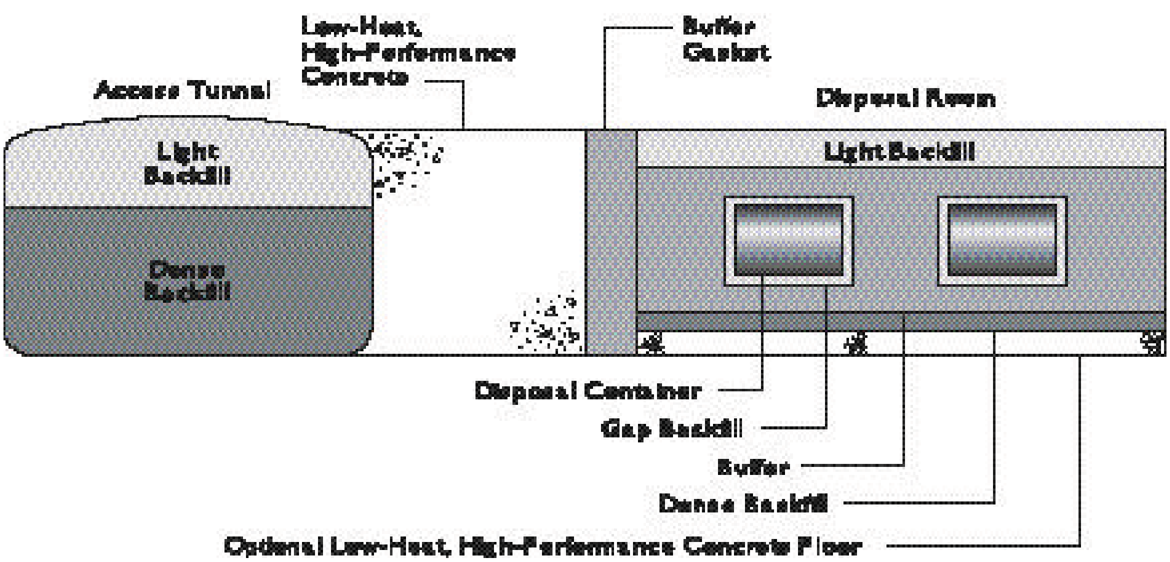

Figure 7.6. Potential arrangement of engineered barriers applied to the in-room emplacement method

management of nuclear wastes and will work with the people of Ontario, the provincial government, and the Canadian federal government and other nuclear waste owners and producers in identifying an approach that is socially, environmentally, and financially acceptable.

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Chapter 8

Preliminary Site Characterization at Beishan, Northwest China—A Potential Site for China's High-Level Radioactive Waste Repository

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8.1. INTRODUCTION

8.1.1. Nuclear Power Plants and Radioactive Waste

The final safe disposal of high-level radioactive waste (HLW) has been a challenging task for the sustained development of the nuclear industry in China. At present, there are two nuclear power plants (NPPs) in operation on the Chinese mainland: the Qinshan NPPin eastern China's Zhejiang province and the Daya Bay NPP in southern China's Guangdong province. Together with the four NPPs (8 units) under construction, Qinshan NPP(Phase 2 and 3), Lin'ao NPPand Tianwan NPP, the total electrical capacity produced by the NPPs will be 8.7 GW by 2005, with plans to reach 20 GW by 2010 and 40 GWby 2020.

In China, work related to radioactive waste disposal is managed by the China National Nuclear Corporation (CNNC), which is responsible for the transport of HLW including spent fuels, reprocessing of spent fuels, vitrification of liquid HLW, and final disposal. We estimate (from the Chinese nuclear power plan) that accumulated spent fuel will reach 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. Low- and intermediate-level waste (LILW) will be isolated by a near-surface disposal method or an underground disposal method, but spent fuel (as a portion of HLW) will be reprocessed first, followed by vitrification and final geological disposal. The repository will be a shaft-tunnel model, located in saturated granite.

In 1985, an Expert Group was organized to coordinate HLW geological disposal. This group was composed of experts from the Beijing Research Institute of Uranium Geology (BRIUG), Beijing Institute of Nuclear Engineering (BINE), China Institute of Atomic Energy (CIAE), and China Institute for Radiation Protection (CIRP). This group is responsible for R&D programs related to site characterization, repository design, safety analysis, and performance assessment. At present, the leading institute of the Expert Group, BRIUG, is conducting a preliminary site-characterization project at the potential site (Beishan) for an underground research laboratory (URL) and future high-level radioactive waste repository. Drilling for the first two boreholes (BS01 and BS02) was completed on October 20, 2000, and May 17, 2001, respectively, and favorable findings have been obtained.

8.1.2. The Deep Geological Disposal Program

In 1985, CNNC proposed an R&D program for the deep geological disposal (DGD) of HLW (Yang, 1992). The program is divided into four phases: (1) a technical preparation phase, (2) a geological study phase, (3) an *in situ* test phase, and (4) a repository construction phase. The objective of the program is to build a granitic national geological repository by 2040 that can dispose of vitrified waste, transuranic waste, and small amounts of CANDU spent fuel.

During the technical preparation phase (1986–1995), the main activities included planning, site screening, feasibility studies, and R&D. The goal of this phase was to select candidate areas and to prepare the technical basis for final disposal.

During the geological study phase (1996–2010), systematic studies of site screening, site characterization, performance assessment, methodology of environmental impact assessment, model development, and buffer/backfill materials are being carried out. The goal is to determine the final disposal area for high-level waste.

During the *in situ* Test Phase (2011–2025), an underground research laboratory will be built, and detailed site evaluation, *in situ* heater tests, tracer tests, and demonstration of disposal technology will be carried out. The experience of Western countries in dealing with radioactive waste will serve to enhance the Chinese deep-geological-disposal technology. At the end of this stage, performance assessment of the total disposal system and a feasibility report will be completed. Licensing application for the construction of an HLW repository will be submitted to the National Nuclear Safety Administration.

In the repository construction phase (2025–2040), the design and construction of the repository will be carried out. The goal of this stage is to have an operational repository by 2040.

The DGD program is a preliminary step: it will be revised as China's nuclear power program develops. We also believe that the financial investment in radioactive waste disposal will be increased to ensure the safe disposal of the waste.

8.1.3. Progress in Geological Disposal R&D

Since 1985, progress has been made in the following areas:

- Site screening and site characterization for an HLW repository: following a national screening, Beishan, located in northwest China's Gansu province, was selected as a potential site for the repository, and preliminary site characterization is under way.
- Technical strategy for development of an underground research laboratory (URL): a site-specific URL will be built at the potential repository site, most probably at Beishan.
- Laboratory experimentation on radionuclide migration: some sorption and diffusion parameters of Pu, Tc and other radionuclides on bentonite and granite samples have been obtained.
- Buffer/backfill material study: following nationwide investigation and screening of bentonite deposits in China, the Gaomiaozi bentonite deposits in Xinhe County, Inner Mongolia Autonomous Region, are considered to be the best candidate for providing sufficient high-quality bentonite for the potential repository. The Gaomiaozi deposit has a bentonite reserve of 127 million tons, and the montmorillonite content could reach as high as 64–81%. A systematic test on the bentonite is underway.
- Natural analog study: bronzeware dated back 3,000 years and a hydrothermal granite-type uranium deposit in southern China's Hunan province were used for natural analog studies.
- Methodological study of performance assessment.

8.2. BELSHIAN REGION (GANSU PROVINCE, NORTHWEST CHINA)—THE POTENTIAL AREA FOR CHINA'S GEOLOGIC REPOSITORY

8.2.1. Geology

The Beishan region (in Gansu province), the preselected area for China's high-level radioactive waste repository, is located in the Erdaojin-Hongqishan compound anticline of the Tianshan-Beishan folded belt in western China. The candidate host rock for the repository is granite. The regional brittle faults, including the Erduanjinnan fault, Zhongqiujin-Jinmiaogou fault, and Erdaojin-Hongqishaltl fault, are nearly EW-striking, shallow, nonactive faults. The crust in the area has a block structure, with a crust thickness of 47–50 km. The depth contour strikes nearly EW, with very little variation. The gravity anomaly is approximately l50 × 10^5–225 × 10^5 m/s^2. The gravity gradient is less than 0.6 mGal/km. On a gravity-anomaly map, the gravity-anomaly contour is distributed very sparsely, without

Table 8.1. The main hydrologic features of each groundwater unit

Groundwater units	Water outflow rates (1/s.m)	Permability (m^2)
Upland rocky fissure water unit	<.0115	shallow system 10^{-12}—10^{0} deep system <10^{-12}
Valley and depression pore-fissure water unit	0.050—0.150	shallow system 10^{-10}—5×10^{-8}, deep system <10^{-10}
Basin pore-fissure water unit	0.0115—1.150	shallow system 5×10^{-10}—10^{-8}, deep system 5×10^{-10}–5×10^{-8}

obvious step zones, indicating that there are no large faults extending to the depth of the crust. The seismic intensity of the region is less than 6, and no earthquakes with Ms>4.75 have occurred. The topography of the area is characterized by a flat Gobi and small hills, with elevations ranging between 1,000 m and 2,000 m. Variations in height are usually several tens of meters. Since Tertiary, the region is slowly uplifting without obvious differential movement.

Comprehensive analysis of the structural deformation of the Cenozoic faults and folds indicate that the area is undergoing horizontal compression at present, and the principal compression stress is between 30º and 60º. Data provided by an analysis of the earthquake source mechanism show that the direction of the principal compression stress is about 40º and the superimposed fault angles are in a stable area, suggesting that the main faults are stable and will not have a strike-slip displacement. Geological characteristics of the Beishan region show that the crust in the area is stable and that it has great potential for the construction of a high-level radioactive waste repository.

In this region, eight granite blocks have been selected as potential sites for the future URL and HLW repository:

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

Among these, three blocks (Jiujing, Xiangyangshan, and Yemaquan) have been chosen as the sites with the most potential, and detailed work is now concentrated on them.

8.2.2. Hydrogeology

The Beishan region is poor in groundwater resources. Pump tests carried out by local geological teams in the 1980s in the area have shown that, for most of the wells, the outflow rates are less than 50 m^3/d. However, wells penetrating fracture zones can have an outflow rate of about 1,000 m^3/d, showing that water-bearing conditions vary greatly in the area. Based on the topography, lithology, and geological structure, Beishan groundwater can be divided into three categories: (1) an upland rocky, fissured unit, (2) a valley and depression pore-fissure unit, and (3) a basin pore-fissure unit. The upland rocky, fissured unit is the most prevalent one in this area.

1. *Upland rocky fissured groundwater*
   This is the most important water type in the Beishan region, occurring in weathered and structural fractures. Groundwater recharge is primarily from precipitation infiltration, with discharge mostly through evaporation and lateral outflows into the fracture water-bearing zones, intermountain areas, and valley depressions. The present water table in the potential site area is about 28–46 m below the surface.

2. *Valley and depression pore-fissured groundwater*
   In the Beishan region, valley and depression topography is generally coincident with the fault zones. This water is commonly more abundant than in other areas. The water table is shallower, with

depths of 2–8 m below the surface. The water is mainly recharged by infiltration of rainfall and temporary seasonal floods, and the main discharge includes evaporation and runoff towards the basin and the Hexi Corridor.

3. *Basin pore-fissured groundwater*
   This water is mainly distributed among the basins in the north and northeast parts of the area, and also among the fault basins of the Hexi Corridor. The basins are mainly composed of Jurassic, Tertiary, and Quaternary formations. Groundwater is recharged from the lateral inflows. Well production varies within a wide range (from 10 m^3/d to 1,000 m^3/d), depending mainly on the conditions of the basin scale, lithology of the aquifer, and structure. In general, the water table is close to the surface. In some areas, the groundwater becomes artesian.

The well outflow rates and permeability for each groundwater unit are shown in Table 8.1.

8.3. SITE CHARACTERIZATION AT JIUJING SECTION, BEISHAN AREA

The Jiujing Section is one of the potential host blocks for a URL and HLW repository in the Beishan area. Based on the previous results, two boreholes were drilled in the period of 1999–2001 to obtain rock, water, and gas samples from deep boreholes, to understand the *in situ* stress field at depth, to obtain a preliminary understanding of the deep geological environment, and to evaluate the suitability of the site. The first borehole (Beishan 01 or BS01) was vertical, with a depth of 703.08 m. The second (Beishan 02 or BS02) was inclined, with a depth of 502.15 m. Before drilling, a surface geological survey, hydrogeological survey, and geophysical survey were carried out.

8.3.1. Geological Mapping

In the Jiujing Section, a 1:50,000-scale surface geological map was completed during April and July 2001, covering an area of 462 km^2. Based on detailed field investigations and laboratory work, a geological map (Figure 8.1) has been generated for the block.

In this area, four granite units are recognized: Jiujing, Bantan, Jiazijing, and Shimenkan. The Jiujing unit is composed of middle-Proterozoic tonalite with an area of 220 km^2. The rocks have undergone intensive ductile deformation, giving rise to the degradation of rock integrity. The Bantan unit is composed of late Proterozoic porphyritic-monzonitic granite with an area of 53 km^2. The granite in this unit is of good integrity, with less deformation and fractures. Thus, it was chosen as the candidate unit for drilling, and borehole BS01 was located in the northern part of this unit (Figure 8.1).

In the Jiujing Section, there are three ductile shear zones, namely: the Bantan'nan-Huayaoshan ductile shear zone (2 km wide, 27 km long), the Shimenkan ductile shear zone (2 km wide, 10 km long), and the Sheyaoqu ductile shear zone (2 km wide, 21 km long). Investigation has shown that these brittle faults developed in shear zones are stable.

Two fault groups have been recognized: an EW striking fault group and a NE striking fault group, including the Sheyaoqu fault, Bantan-Huayaoshan fault, Jiujing fault, and Shiyuejing fault. There are four fracture groups: the EW group, the 325º group, the 15º group, and the 35º group.

8.3.2. Borehole Drilling

BS01 and BS02 are the first two boreholes in the Beishan candidate site. BS01 is a vertical hole with a depth of 703.08 m, drilled to evaluate the potential Bantan Granite unit, while BS02 is inclined, with a depth of 502.15 m, to evaluate the characteristics of Shiyuejing fault, the key NE-striking fault. Diamond drilling was used, with pure water as the drilling fluid, and full core sampling was carried out in both boreholes.

Drilling for BS01 started on July 8, 2000, and was completed October 20, 2000, at the expected depth of 703.28 m. Borehole television, borehole geophysical surveys (gamma, resistivity, temperature, pressure, U, Th and K content, full waveform sonic, borehole televiewer), pump tests, injection tests, sample collection, and geostress measurements have been conducted.

Drilling for BS02 started on July 26, 2000, and was completed on May 17, 2001, at the expected depth of 502.15 m. It cross-cuts the NE-striking Shiyuejing fault from its hanging wall, through the fault zone, to its foot wall. Perfect core samples for the fault have been obtained, and the hydrogeological investigation has preliminarily shown that the fault is not a water-conducting fault.

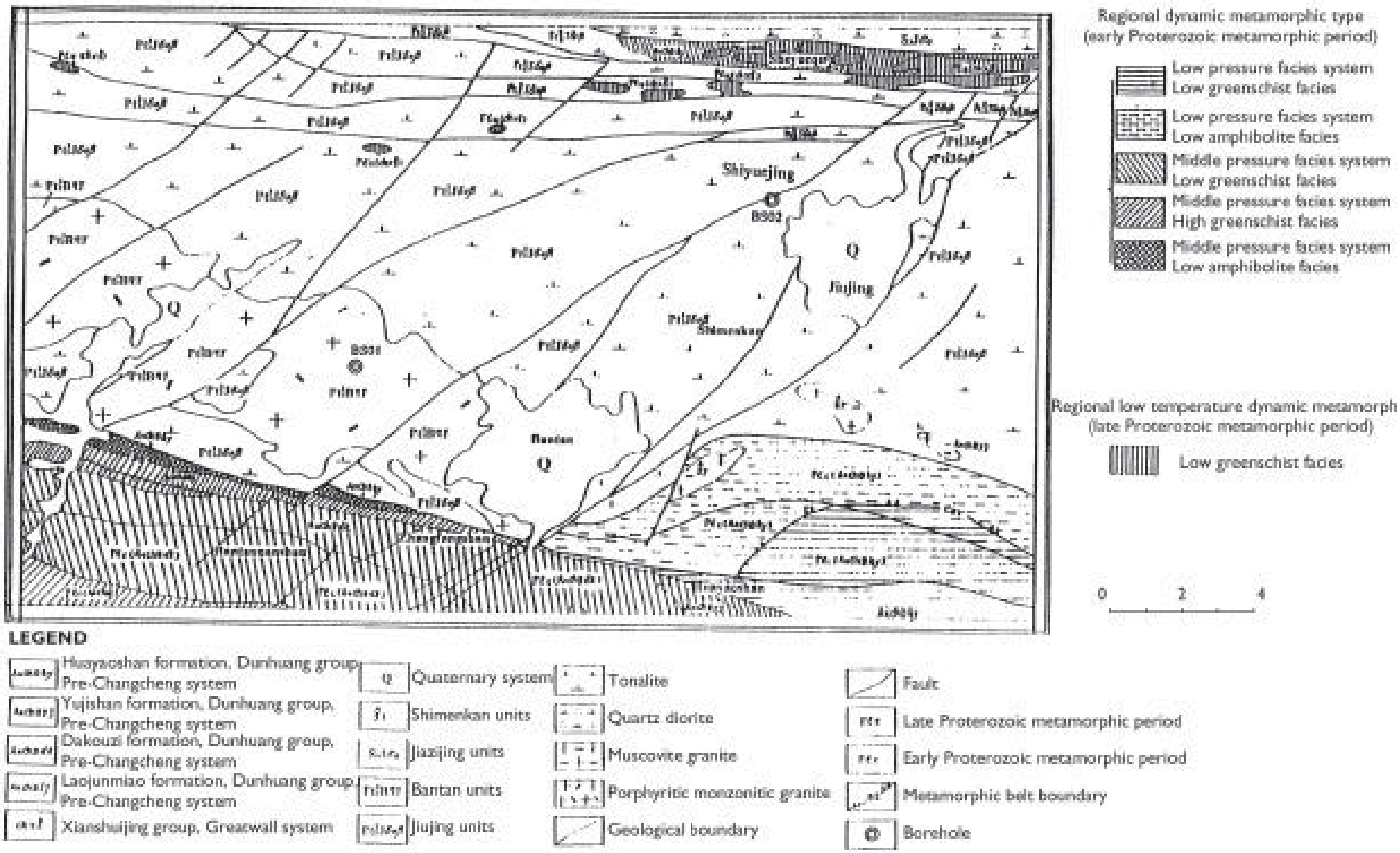

Figure 8.1 Map showing the geology of Jiujing section, Beishan area, Gansu province, NW China—the pre-selected site for a high-level waste repository

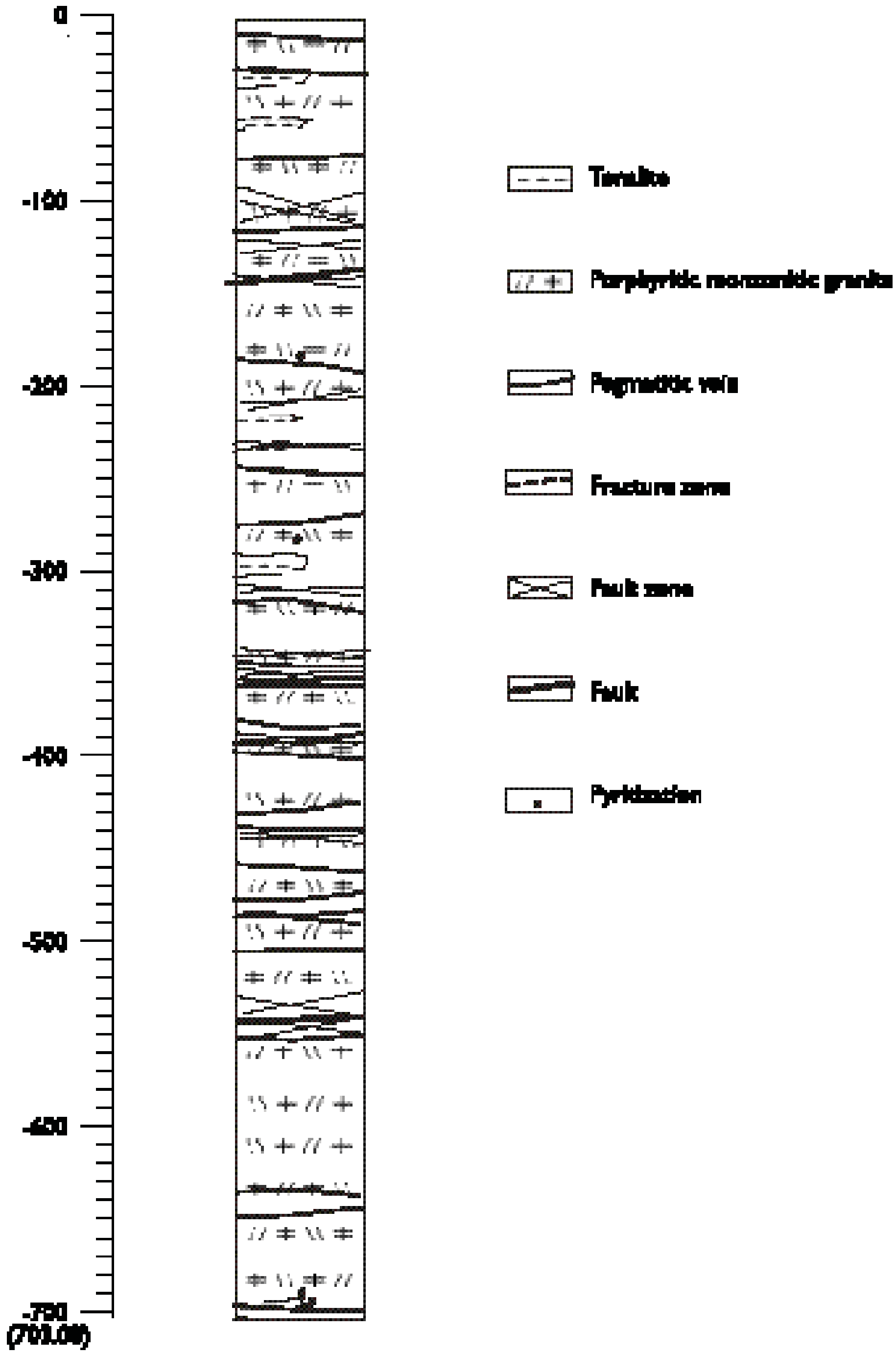

Figure 8.2. Simplified lithological column of the Beishan borehole BS01

8.3.3. Borehole Geology

The lithological column for BS01 is shown in Figure 8.2. The dominant lithology in BS01 is monzonitic granite, with coarse-grained pegmatite veins distributed locally and tonalite enclaves or veins. The monzonitic granite (Figure 8.3) is pinkish, with a porphyritic texture. The porphyritic crystals are composed of potassium feldspar, ranging from 8–20%. The mineral composition of the monzonitic granite is andesine (40–45%), albite (5%), microcline (25–35%), quartz (20–25%), and biotite (5%). The accessory minerals include apatite, zircon, and garnet. Core samples show the good integrity of the monzonitic granite. Figure 8.4 shows the longest core samples from BS01.

Most of the fractures in BS01 are vertical or have large dipping angles (approximately 65º–80º). Borehole acoustic televiewer surveys also reveal characteristics of the fractures. The statistics of the survey show the average fracture density is about 8–12 per 10 m, with most of the fractures striking NE and NW. The fracture fillings include sericite, calcite, pyrite, and chlorite.

8.3.4. Borehole Hydrogeological Tests

Borehole hydrogeological tests have been conducted at BS01 after drilling was completed. The tests show the water table at a depth of 46.6 m. Six intervals with local fractures were chosen to carry out injection tests in this borehole using a double packer system, and the results of injection tests are shown in Table 8.2.

Table 8.2. Injection tests for Borehole BS01.

No.	Test interval (depth,meters)	Permability (K,m/s)
1	120.5--128.5	$<8.0\times10^{-9}$
2	222.0--230.0	$<8.0\times10^{-9}$
3	315.0--323.0	1.74×10^{-8}
4	436.0--444.0	$<8.0\times10^{-9}$
5	476.0--484.0	$<8.0\times10^{-9}$
6	496.76--504.76	$<8.0\times10^{-9}$

8.3.5. Acoustic Borehole Televiewer Measurements

An acoustic borehole televiewer system (Fac-40 system) was used in BS01 to investigate the fractures and lithology of the borehole. The accuracy of the system is to 1%, while the resolution for fractures can be as small as 0.1 mm, which is very powerful for the study of fractures. The interval 60–550 m was investigated with this system, and the results clearly identify azimuth, tilt, and depth of fractures.

The statistics for fractures generated from the Fac-40 televiewer survey reveal that they can be divided into two groups, NE- and NW-striking. Most of the NE-striking fractures occur below a depth of 230 m, but with two opposite azimuths: NW and SE. Above 230 m, most of the fractures are NW-striking. The average fracture density is about 8–12 per 10 m.

Table 8.3. Results of hydrofracturing geostress measurement at BS01

No.	Depth (m)	Tensile strength Thf (Mpa)	Vertical stress δV (Mpa)	Minimum lateral stress δH (Mpa)	Maximum lateral stress δH (Mpa)	Direction of minimum lateral stress (δH)
1	161.5	5.29	4.36	4.56	7.72	
2	166.5	2.73	4.49	4.12	7.11	N25°E
3	250.5	7.74	6.76	5.06	8.18	
4	252.4	4.96	6.81	4.71	7.71	N30°E
5	332.4	1.33	8.97	6.10	8.99	
6	339.0	6.40	9.15	4.72	6.51	N38°E
7	413.3	5.94	11.15	8.37	12.65	N45°E
8	457.0	-	12.3	11.13	19.14	
9	493.2	-	13.31	14.85	25.66	

Hydrofracturing was used to measure the *in situ* stress in Borehole BS01. The major results are listed in Table 8.3, showing that the maximum lateral principal stress has a direction between N25ºE–N45°E.

8.3.6. Future Plans

The drilling of the first two boreholes in the Jiujing Block, Beishan region, will be completed late in 2001, and the comprehensive site-investigation reports for the Jiujing Block and the two boreholes will be completed at the same time. Based on the findings and the water/rock samples taken for investigation, much laboratory work will be conducted, 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.

In the coming five-year period (2001–2005), two other blocks, the Xiangyangshan and the Yemaquan, will also be investigated. This work will include surface geological mapping, geophysical surveys, and at least three boreholes. By about 2005, a candidate site for our site-specific URLwill be proposed, and further work will be carried out continuously.

8.4. CONCLUSIONS

High-level radioactive waste disposal is a challenge for sustained development of the nuclear industry in China. Necessary resources have been arranged for the final geological disposal of high-level waste. A deep geological disposal program has been proposed by the China National Nuclear Corporation, and the Beijing Research Institute of Uranium Geology is the current leading institute for the program. Since 1985, progress has been made in site selection and site characterization, backfill material studies, radionuclide migration studies, and natural analog studies. Beishan, located in NW China's Gansu province, has been selected as the best potential site for China's high-level radioactive waste repository. The first two boreholes were drilled in the Jiujing Block of the Beishan region, and the results have shown the advantages of the site. A continuous effort will be concentrated on the Beishan site, and other associated laboratory research will also be carried out in the coming years—all for the purpose of building China's high-level radioactive waste repository during the period 2030–2040.

Figure 8.3. Typical rock from BS01: Porpyritic monzonitic granite

Figure 8.4. Core samples from BS01

Chapter 9

Site Selection and Characterization for a Low- and Intermediate-Level Radioactive Waste Repository in the Republic of Croatia

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9.1. INTRODUCTION

The aims of this report are to give: (a) essential information on the current status of the low and intermediate level radioactive waste (LILW) repository site-selection process in Croatia, including a short description of activities performed since the previous report on Croatia appeared in the *Second Worldwide Review* (Schaller, 1996); (b) a description of known repository site characteristics; (c) a brief review of the conceptual model and technical design of the LILW repository; and (d) a schedule and description of LILW repository project activities in the future.

As has been previously reported, Croatia is (along with Slovenia) a co-owner of the Krsko Nuclear Power Plant (Krsko NPP) and is therefore obliged to dispose radioactive wastes generated at that plant. Total LILW volume from Krsko NPP (operational and decommissioning waste) is expected to be about 18,000 m^3 at the end of plant lifetime (in the year 2023). Although it is true that only half of this volume is generated by Croatia, it is possible that one country will eventually receive all LILW, while the other one will dispose spent fuel and other high-level waste. However, preparation of the Krsko NPP decommissioning strategy is still underway, and final options are not yet known in detail. (LILW, spent fuel, and dismantling of the nuclear facility itself, along with site decontamination, are considered in the strategy.) On the other hand, since there is no nuclear plant in Croatian territory, the radioactive wastes coming from Croatia do not presently exceed ~50 m^3 and are not expected to exceed 500 m^3 in the next 50 years. The waste generated in Croatia itself results from several nuclear applications, such as medicine, industry, agriculture, scientific investigations, as well as some specific applications such as radioactive lightning protectors and ionizing smoke detectors. Estimated LILW disposal activities in a Croatian repository that are expected to continue until 2055 (in both Slovenia and Croatia) are given in Table 9.1.

As is apparent from Table 9.1, the total activity of waste expected to be buried at the LILW repository will reach about 1.3×10^{15} Bq, and mean specific activity will be about 7.2×10^{10} Bq. According to legislation (Narodne novine, Official Gazette 27, 1999), this waste is categorized by its physical and chemical form as well as by radionuclides contained within the intermediate level radioactive waste.

9.2. FINAL STAGE OF LILW REPOSITORY SITE SELECTION

The *Second Worldwide Review* (Schaller, 1996) reported the selection of seven potential areas for a site: Petrova gora, Trgovska gora-Zrinska gora, Bilogora, Moslavacka gora, Psunj, Papuk-Krndija and Pozeska gora (Figure 9.1). Each of them covers 200–600 km^2 in area.

Almost all potential site areas are mountainous regions composed of solid rock (granite, gneiss), that represent remobilized and exhumed massifs of the old Pannonian mass, stretching generally beneath thick Tertiary and Quaternary sediments of the Pannonian basin in northern Croatia. From the geological standpoint, they are all typical horsts (except Bilogora). A re-examination and correlation of these areas by previously defined and evaluated weighting criteria (referring to specific characteristics such as waste transportation possibilities, meteorology, hydrology, seismicity, soil mechanics,

Table 9.1. Estimated activity of LILW generated up to year 2055 to be disposed of in the repository (Bq)

Radionuclide	Waste from Krsko NPP Operational	Waste from Krsko NPP Decommissioning	Other waste from Slovenia	Waste from Croatia	Total	Percent of Total
^{3}H	4.03×10^{13}	-	-	-	4.03×10^{13}	3.15
^{55}Fe	7.34×10^{13}	2.53×10^{13}	-	-	9.87×10^{13}	7.71
^{60}Co	1.39×10^{14}	5.70×10^{14}	-	-	7.09×10^{14}	55.39
^{63}Ni	2.91×10^{13}	2.53×10^{13}	-	-	5.44×10^{13}	4.25
^{137}Cs	3.35×10^{14}	-	-	-	3.35×10^{14}	26.16
Other	1.67×10^{13}	1.27×10^{13}	7.70×10^{12}	4.90×10^{12}	4.20×10^{13}	3.28
Total	**6.33×10^{14}**	**6.33×10^{14}**	**7.70×10^{12}**	**4.90×10^{12}**	**1.28×10^{15}**	**100.00**

hydrogeology, demography, land use, and environmental protection capability) led to the selection of four significantly smaller sites.

Thus, the sites of Trgovska gora (8 km^2), Moslavacka gora (20 km^2), Psunj (14 km^2) and Papuk (8 km^2) were proposed for further investigation in 1997. The survey undertaken in 1998 pointed to two of them (Moslavacka gora, Trgovska gora) as potentially suitable. Finally, Trgovska gora was selected as the most suitable site for inclusion into the Regional Plan of Croatia. After its selection in 1999, it was necessary to prepare a detailed program of on-site investigations.

It should be emphasized that this entire selection process was based exclusively on multicriteria analysis of data from investigations undertaken in the past for other purposes (e.g., geology, hydrology, mine exploration) and not specifically for the LILW disposal site selection. In other words, the complete site-selection procedure, as described above, was in fact nothing more than desk work, since the Croatian government has not allowed any on-site activity (except site prospection that excludes sampling or any other "invasive" on-site technique) before the site could be approved. This restriction was overcome somewhat by remote sensing techniques (aerial and satellite imagery) that provided some relevant data. However, access to the site for on-site investigation is still impossible because the newly adopted Regional Plan of Croatia needs to be validated by the site's local and county authorities and incorporated into their plans. For that reason, it is unrealistic to plan for site investigations before 2002 (or even 2003—whenever approval is granted).

Other obstacles to field investigations also exist. Mines from the war (1991–1995) stretching around the site still prevent access to the site. Moreover, the NIMBY ("not in my back yard!") reaction of local communities is strong (although the broader area of interest is underpopulated). Finally, the site itself is located only a few kilometers from the border of Bosnia-Herzegovina, which also makes project realization difficult.

On the other hand, it is also worth mentioning that project performance is periodically discussed with experts from the International Atomic Energy Agency (IAEA). They have given a positive evaluation to work already completed, provided valuable comments on achieved results, and made constructive suggestions and recommendations for follow-up activities.

9.3. RESULTS OF SITE-CHARACTERIZATION ACTIVITIES PERFORMED

During the last two years, we have prepared a set of studies for site characterization of Trgovska gora (some of which also apply to the Moslavacka gora site) and the possible safety and environmental impact of an LILW repository at that location. These are summarized below:

- *Preliminary geoecological analysis of preferred sites for LILW repository in Croatia (Volume 1: Trgovska gora site; Volume 2: Moslavacka gora site)*
  The study describes some relevant geo-ecological issues of the sites (such as lithostratigraphy, seismicity, seismotectonics, geomorphology, hydrogeology, hydrology, pedology, meteorology), but also raises demographic issues as well. Some interpretation has been made, based on maps at a scale of 1:25,000, leading to conclusions and recommenda-

tions for further investigations for each respective issue. On the basis of site characterization (i.e., evaluation of the sites considering each issue), microsites are proposed (about 1 km^2 in size) for a disposal facility.

- *Preliminary bioecological characterization of preferred sites for LILW Repository in Croatia (Volume 1: Trgovska gora site; Volume 2: Moslavacka gora site)*
  The main ecosystems are described in this study. Aquatic and terrestrial ecosystems are described separately. Aquatic ecosystems are analyzed with regard to: (a) physical and chemical properties of stream waters, (b) bacteriological characteristics of main streams, (c) algological, biocenological and saprobiological evaluation of aquatic microzoobentos, and (d) ichthyofauna (fish). Terrestrial organisms, comprising both flora (vegetation) and fauna (including game and birds) are also included in the study. Main food-chains at the sites and organisms relevant for radiological monitoring are identified.

Figure 9.1. Potential areas for LILW repository in Croatia

- *Geographic Information System application of relevant geotechnical data for LILW repository site at Trgovska gora*
  This document describes the preparation of Geographic Information System (GIS) (ArcView) tools for the PC, based on physical and thematic (lithological, hydrological, geomorphologic, pedological, and land use) maps of the Trgovska gora region. All maps are geocoded, serving as separate layers to be analyzed separately or superimposed. They are also given an additional layer containing data from LANDSAT 5 (Thematic Mapper) imagery. All the available layers are to be subsequently modified as new data is acquired. Thus, from this study, we have access to convenient tools for dynamic spatial analysis of the preferred site, as well as decision making based on a specific georeferenced database.

- *Analysis and interpretation of satellite digital imagery and geophysical data of the LILW repository site in Trgovska gora*
  Based on the LANDSAT 5 TM imaging of the potential-site area lithology, geomorphology, tectonics, seismicity, hydrology and hydrography, as well as thermal characteristics of the site, are evaluated. Acquired data are used as a "correcting factor" for previously obtained site information from the aforementioned geo-ecological study. At the same time, the study offers geophysical information on vertical changes in site lithology, at depths of 300 m, 1,000 m, 3,000 m and 5,000 m. Geoelectrical profiles and gravimetry residual maps are also enclosed and interpreted in the study.

- *Radio-ecological monitoring of radionuclide migration through food-chains at the Trgovska gora site*
  Both "zero-state" identification and regular monitoring programs are presented in this study. Two main food chains are considered, and four groups of samples—(a) stream water, detritus and ichthyofauna; (b) groundwater; (c) terrestrial flora and fauna; and (d) honey—are proposed for monitoring. Sites for monitoring, types of radiological analyses to be undertaken, and annual sampling frequencies for a "zero-state" and the regular monitoring are recommended. Also, tentative annual financial plans for a"zero-state" and a regular monitoring program are described in this study.

- *Preliminary program of detailed field investigations at the preferred site for LILW Repository*
  The preliminary program for detailed field investigations includes statements on general site-acceptance requests, elaboration of a site-characterization-plan strategy, description of preliminary characterization foreseen to be undertaken in Phase I, general review of activities to be done in Phase II, and finally, a detailed review of relevant geotechnical investigations necessary for a successful on-site characterization. The methods of survey (such as geodesy, mapping, remote sensing, surficial and bore-hole geophysics, and basic equipment required) are also described in this study.

9.4. BASIC GEO-ECOLOGICAL CHARACTERISTICS OF THE TRGOVSKA GORA SITE

The preferred site of Trgovska gora is located in the Banovina region of Central Croatia, 90 km southeast of Zagreb (population 780,000) and 50 km southwest of Sisak (population 60,000). The site is only 3 km from the Bosnia-Herzegovina border. The road distance between the Krsko NPP and the site is 170 km. (A road is the only communication to the site; the road distance to the nearest railway leading from Krsko NPP via Zagreb is 25 km from the site.) Some 62,000 inhabitants live within the 20 km radius around the site (includes both countries). Mean population density in this area is about 45 inhabitants per km^2—a level of population density that is significantly below values for Croatia (~76 inh./km^2) and Bosnia-Herzegovina (~78 inh./km^2) (Preliminary Geo-Ecological Analysis, 1999).

From the geotectonic viewpoint, Trgovska gora mountain is a faulted and folded remobilized ancient massif of Paleozoic-Tertiary orogeny. Its lithology features younger Paleozoic schists and clastites, bounded on the west by Triassic ("Werfen") layers. The area is part of a major geotectonic unit of the Inner Dinarides. From a morphostructural perspective, Trgovska gora is a horst that was subsequently (in the Tertiary and Quaternary) uplifted by fault tectonics. Young tectonic movements caused a remarkable dissection of relief in the area, so that a series of creek valleys, orientated SW-NE and draining toward the stream of Zirovac, have been formed. As a consequence, the mountain is not a homogeneous massif but is morphologically represented as separate blocks (Milinkovac, Kalabina kosa, Veliko brdo, Pavlovo brdo, Hleb, etc.), reaching elevations of 400–500 m and separated by creek valleys of Majdanski potok, Veliki potok, Kalabin jarak, Velebitski potok, etc. Although slope inclinations in the area are not negligible (12–32°) and relief energy is rather high (132–154 m/km^2), there are some flat portions (plateaus) at the top of the mountain blocks (Preliminary Geo-Ecological Analysis, 1999). These plateaus allow implementation of a vaulted shallow-land disposal facility without major technical interventions (although a tunnel-type repository is being considered as a contingency).

The site itself is about 8 km^2 in area and covers the upstream section of Majdanski potok creek (i.e., source streams of Kalabin jarak, Zirovacka vrlanda, and Majdanska vrlanda). All these creeks are lacking in water, and some are only intermittent streams. In general, the area is not abundant in water, and seasonal sources prevail. As mentioned above, surface hydrology is directed toward the northeast, i.e., toward the Zirovac stream, a tributary of the Una River flowing north to the Sava River. Hence, all surface waters are drained toward Croatian and not toward Bosnian territory (some studies have pointed out that groundwater circulation is probably orientated in the same direction). The lithology of the site area is predominantly clayey schist, while sandstones are present in small amounts. The seismic activity of the area is low compared to some other parts of the country: the maximum expected seismic activity does not exceed 8° on the Mercalli-Cancani-Sieberg (MCS) scale, and maximum ground acceleration is below 0.3 g (Preliminary Geo-Ecological Analysis, 1999).

The site evaluation has qualified three preferred microsites as prospects for further investigations. These investigations will also include a detailed field survey supported by an adequate safety assessment (i.e., including the calculation of all possible effects of the repository on the environment during the disposal facility lifetime of 250–300 years). Three proposed microsites (Milinkovac, Veliko brdo, and Pavlovo brdo), each comprising about 1 km^2 in area on the Trgovska gora site, are shown in the Figure 9.2.

Particular site characteristics (parameters) are listed in Table 9.2. Some of them are site specific, i.e., obtained by investigations done in the past (for other purposes), while others are quite generic.

9.5. LILW DISPOSAL FACILITY

We have agreed that two basic LILW disposal facility types should be considered for implementation in Croatia

Table 9.2. Some parameters regarding prefer red site for LILW repository in Trgovska gora mountain

PARAMETER	VALUE / DESCRIPTION	REMARK
1.GEOGRAPHY & GEOMOPHOLOGY		
geographical position of the site	Banovina region;90 km south of Zagreb	site specific
microsites	Milinkovac, Veliko brdo, Pavlovo brdo	site specific
accessibility	regional road Glina-Dvor + local road	site specific
altitude	320-508 m	site specific
slope inclination	12–32°	site specific
relief energy	132-154 m km^{-2}	site specific
dominant slope processes	regressive erosion,gulling,creeping (defluction)	site specific
2.LITHOSTRATIGRAPHY		
lithology	clayey schists (70 %) – sandstones (30%)	site specific
age of rocks	Paleozoic	site specific
inclination and stretching of layers	50-70° NW-SE	site specific
3.TECTONICS		
local structural position	limb of Trgovska gora reverse structure	site specific
minimum distance to a recently active fault	4 km	site specific
distance to major fault zone	> 10 km	site specific
4.SEISMICITY		
maximum expected seismic capability	7-8 MCS	site specific
earthquake intensity for return period of 100 years	6.4 MCS	site specific
earthquake intensity for return period of 1000 years	7.3 MCS	site specific
maximum ground acceleration	< 0.3 g	site specific
5.HYDROGEOLOGY		
type of aquifer	local	site specific
groundwater age (retention capability)	4-5 years	site specific
permeability	0.69 - 6.9 x 10^{-15} m s^{-1} (?)	site specific
porosity	2.9-10.2 %	site specific
thickness of non-saturated zone	5 m	site specific
density of non-saturated zone	1.9 g cm_{-3}	generic
hydraulic conductivity of non-saturated zone	5 x 10^{-6} m s^{-1}	generic
content of moisture in non-saturated zone	0.034	generic
effective porosity of saturated zone	0.05	site specific
hydraulic gradient of saturated zone	10 %	site specific
hydraulic conductivity of saturated zone	5 x 10^{-6} m s^{-1}	generic
groundwater circulation velocity	3.125 m $year^{-1}$	site specific
density of saturated zone	1.9 g cm^{-3}	generic
depth of water-table	10 m	site specific
longitudinal dispersivity of saturated zone	20 m	generic
transversal dispersivity of saturated zone	6.66 m	generic
distribution coefficient (K_d) for loams:		
cobalt (Co)	1,300	generic
cesium (Cs)	4,600	generic
strontium (Sr)	20	generic
uranium (U)	15	generic
thorium (Th)	3,300	generic
radium (Ra)	36,000	generic
thickness of unsaturated zone (depth of water table)	?	
hydrogeologic characteristics of rocks	fairly impermeable (fractured-intergranular porosity)	site specific

Table 9.2. Some parameters regarding preferred site for LILW repository in Trgovska gora mountain (continued)

PARAMETER	VALUE / DESCRIPTION	REMARK
6.GROUND WATER CHEMISTRY		
groundwater type	calcium-hydrocarbonatic	site specific
groundwater mineralization	low: 146-160 mg l^{-1}	site specific
pH of spring- and well-water	7.3	site specific
groundwater hardness	2.2 - 5.0 dH	site specific
7. HYDROLOGY		
catchment area (drainage basins)	Zirovac-Una-Sava-Danube	site specific
ultimate surface waters recipient	Black Sea	site specific
springs capacity	mostly < 0.1 l s^{-1}	site specific
Schumm elongation ratio (R_e)	0.58	site specific
Miller's circularity index	0.58	site specific
bifurcation index	5.13	site specific
total length of streams	86.16 km	site specific
all streams/only permanent stream network density	3.42 / 2.37 km km^{-2}	site specific
8.METEOROLOGY		
annual precipitation	1,000 mm	site specific
surface runoff rate	46-50 % (448-493 mm $year^{-1}$)	site specific
infiltration rate	5-10 % (50-100 mm $year^{-1}$)	site specific
evapotranspiration rate	45-53 % (436-529 mm $year^{-1}$)	site specific
extreme meteorological phenomena	none	-
mean annual temperature	9.8 °C	site specific
snow cover retention period (average)	months XII-III	site specific
snow cover duration	17.6 (I),10.8 (II,XII) days	site specific
maximum thickness of snow cover	78 cm (I)	site specific
dominant wind blowing direction	NNE (10 %)	site specific
calm	23 %	site specific
wind velocity (average/maximum)	3.7 (NNE) / 18.5 (N,NNE) m s^{-1}	site specific
9. ROCK MECHANICS		
hardness coefficient f_ξ (0-20):sandy clayey schists	category 5:rather firm	generic
10. PEDOLOGY		
soil type	acid brown soil (80%) -eutric/distric ranker (20%)	site specific
11. VEGETATION		
type and dominant species of vegetation	forest (beech,hornbeam)	site specific
12.NATURAL RADIOACTIVITY		
total natural specific activity of groundwater	132.5 Bq m^{-3}	site specific
^{238}U activity	17.5 Bq m^{-3}	site specific
^{232}Th activity in groundwater	44.0 Bq m^{-3}	site specific
^{40}K activity in groundwater	71.0 Bq m^{-3}	site specific
13.DEMOGRAPHY		
population density < 5 km around the site	6 inhabitants km^{-2}	site specific
population density < 20 km around the site	45 inhabitants km^{-2}	site specific
number of settlements < 5 km around the site	3	site specific
the largest settlement < 5 km around the site	Ljubina (352 inhabitants)	site specific
number of settlements < 20 km around the site 85	site specific	
the largest settlement < 20 km around the site	Bosanski Novi (12,186 inhabitants)	site specific
site downstream living population (up to r=20 km)	5,542	site specific
site downstream number of settlements (up to r=20km)	14	site specific

Table 9.2. Some parameters regarding preferred site for LILW repository in Trgovska gora mountain (continued)		
PARAMETER	VALUE / DESCRIPTION	REMARK
14.LAND USE		
dominant types of land use (land cover)	extensive forests (uncultivated)	site specific

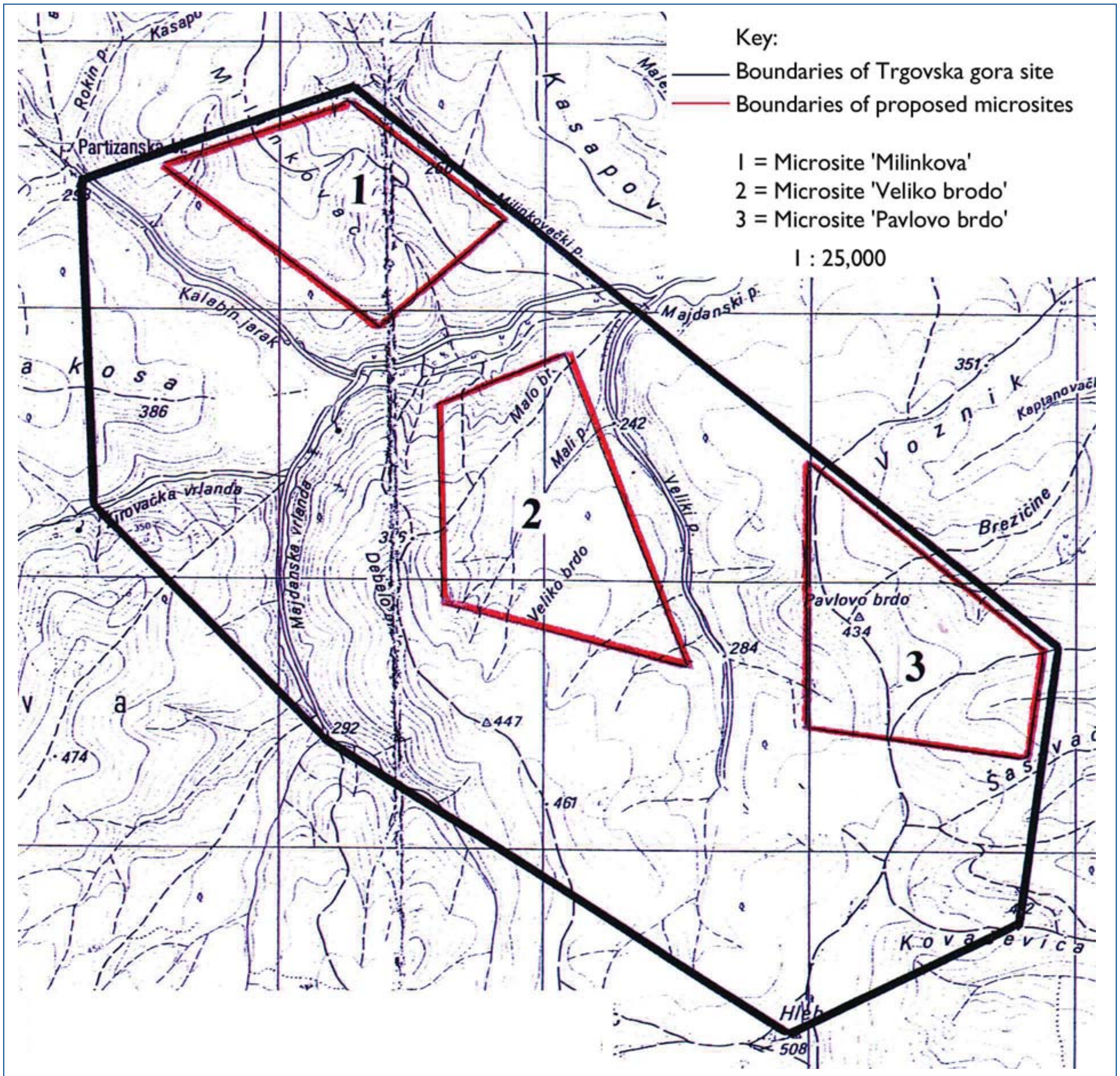

Figure 9.2. Proposed microsites identified in Trgovska gora site

(assuming that project realization is approved by the Croatian Parliament). Near-surface engineered burial (in vaults) seems to be more probable, although subsurface disposal (in tunnels) is still a contingency option.

9.5.1. Conceptual Model

We are developing a repository conceptual model that would allow analysis of radionuclide migration mechanisms and their impact on the most exposed individual, as well as a comprehensive explanation of the following processes:

- Precipitation infiltration and water percolation through cover layers into the disposal area
- Leaching of radionuclides at constant velocity and radionuclide transport by advection through porous geologic media surrounding the waste disposal facility
- Transport of radionuclides by advection through an unsaturated zone
- Transport of radionuclides through the saturated zone, i.e., aquifer (advection-dispersion transport model), and calculation of radionuclide concentrations in the well water
- Calculation of radioactivity in humans from long-term ingestion of contaminated well water
- Risk assessment of cancer incidence leading to death (Preliminary Performance Assessment, 1998).

The conceptual model under consideration is to be additionally characterized by involving the following assumptions:

- The drainage system completely fails at the repository closure (i.e., at the starting point of the analysis).
- The formation overlying the repository has a reduced isolation capability, so that at the start of analysis, it is about three times higher than the natural infiltration rate.
- Metallic containers with radioactive waste completely lose their isolation capability within 50 years after closure.
- Reinforced concrete structures in the waste disposal area completely lose their isolation capability within 50 years after closure.
- Host rock is expected to be stable long-term, so that all parameters considered in site characterization are not changed either during active or passive institutional control periods (no changes in geologic formation geometry are assumed).
- Mechanisms of water infiltration into the repository area, radionuclide leaching, and migration through both unsaturated and saturated zones are expected to be in a stationary regime during active and passive institutional-control periods.
- The waste disposal facility, as well as unsaturated and saturated zones, are considered as porous and isotropic media in all directions.
- Effects of dispersion are neglected in both the repository itself and in the nonsaturated medium.
- A linear radionuclide sorption model is assumed to be applicable in all sections of the repository system.
- There are no unknown sources or sinks at the waste disposal site.
- Effects of dispersion and dilution are taken into consideration in the saturated medium (Preliminary Performance Assessment, 1998).

According to this conceptual model, the repository-site system consists of three basic elements: the waste disposal facility, an unsaturated zone, and a saturated zone. The repository itself is conceived as a homogeneous mixture of concrete and waste during the first-50-year post-closure period. Calculations are based on the assumptions that the disposal facility is 35 m long and 10.5 m high, unsaturated and saturated zones are each 5 m thick, and the distance to the nearest well is 200 m.

9.5.2. Technical Design: Description of the Two Options

1. *Near-surface engineered burial in vaults.* A widely used option, should be the least expensive one, and might relatively easily be modified to allow for waste retrievability. The disposal facility will be designed to accept all LILW waste quantities from the Krsko NPP operation, as well as from the plant decommissioning and the nuclear applications in Croatia. The packages of highly compacted waste in steel-tube-type containers (TTC) and steel containers with biological shielding (MOSAIK) will be placed in reinforced concrete containers and then into concrete vaults. The vaults, set on or below ground level, consist of a concrete base and walls with an underlying drainage layer. The top is sealed with concrete and water-resistant isolation, and covered with soil layers of alternating hydraulic conductivity, topped with vegetation.
2. *Subsurface disposal in tunnels.* The disposal facility will be designed to accept all LILW waste quantities from the Krsko NPP operation, as well as

Table 9.3. Basic technical data on LILW repository (Preliminary Performance Assessment, 1998)

Parameter	Value	
Outer dimensions of the LILW disposal facility		
• length (parallel to groundwater circulation)	35	m
• width (perpendicular to groundwater circulation)	152	m
• height	10.5	m
Capacity of a unit for final disposal		
• total quantity of concrete containers by width	9	
• total quantity of concrete containers by length	16	
• total quantity of concrete containers by height	3	
• capacity of a unit for final disposal	432	concrete containers
	(1,728	metallic containers)
Total capacity of waste disposal facility		
• total quantity of concrete containers	4,320	
• total quantity of metallic containers	17,280	

from decommissioning and nuclear applications in Croatia. The packages of highly compacted waste in TTC and MOSAIK will be placed in reinforced concrete containers (typically smaller than those for the vault facility) and then placed in tunnels. Although the waste will be covered by host rock a few tens of meters thick, the public perception might strongly favor the safety of this option. It would probably not require an additional period of passive institutional control. The subsurface facility design consists of tunnels that would be drilled more or less horizontally into a hill slope.

Regarding the more probable option (near-surface engineered burial in vaults) it should be added that all LILW is assumed to be disposed of in tube-type containers, foreseen to be placed in reinforced concrete containers. The concrete containers are expected to be set in units for final disposal, with the final disposal area consisting of 10 units. Basic technical data on the repository are given in Table 9.3.

9.6. PROJECT ACTIVITIES TO BE UNDERTAKEN

At present, the LILW repository project consists of a series of specific activity groups necessary for a successful project realization. Besides the site selection and site characterization, the project also involves waste transportation analysis, project economic evaluation, repository safety analysis, repository design development, licensing, and public participation.

From the standpoint of site selection and characterization, we need to emphasize that the actual situation with only one site included in the Regional Plan of Croatia (and thus available for further investigation) is for various reasons still quite uncertain. First, the final political decision on construction of the LILW repository in Croatia has not yet been made. Second, the selection of Trgovska gora as the final site among four (and then between two) candidate sites was done to a large degree arbitrarily (i.e., through parliamentary debates and decisions), and that all four candidate sites (Psunj, Papuk, Moslavacka gora, and Trgovska gora) are still considered by certain experts as equal-standing candidates. This fact could be fully exploited by a local community that wanted to prevent a repository from being built nearby (and thereby push the site-selection process back to equal consideration of other candidate sites). Third, detailed field investigations at the chosen site will ultimately be needed to confirm or reject the site for the LILW repository.

In any case, it is first necessary to allow access to the site (after clean-up of the mines remaining from the war) and then to secure needed equipment and instrumentation for realization of the on-site survey. The planned field investigations should: (1) confirm the proposed site as acceptable for LILW repository siting, and (2) evaluate the three microsites and select the most suitable one. This requires equal-scale investigations at all three microsites proposed in the first stage. After one of them is assessed as the most suitable, detailed mapping, geophysics, and drilling should finally confirm this microsite as the best location for the repository (Preliminary Program of Detailed Investigations, 2000).

In accordance with these activities, the recently adopted site characterization plan strategy is expected to help us in achieving the following goals:

- Gather complete information on the geology of the site and its particular formations. (This information should also be decisive in the final LILW disposal facility design—either a near-surface engineered burial, in vaults, or subsurface disposal in tunnels.)
- Develop conceptual and mathematic models of radionuclide migration from the repository into the environment. (This is an essential first step in the safety assessment necessary to prove long-term site safety.)
- Collect additional site-related seismological, geomorphologic, and geotechnical data necessary for preparation of a feasibility study and disposal facility construction (Preliminary Program of Detailed Investigations, 2000).

To achieve the described goals, the characterization plan was split into two phases.

Phase I includes preliminary characterization of the selected final site (8 km^2 in area) at a scale of 1:5,000. Phase I, expected to be finished in 2 years, should result in:

- Final selection of one microsite (among the three possibilities), which would be evaluated in detail in Phase II.
- Sufficient background information needed for preliminary safety analysis, especially crucial for developing a regional groundwater circulation model. This regional model is necessary for understanding radionuclide migration and transport at the microsite, expected to be developed in Phase II.
- Accumulation of information required for a repository technical design and feasibility analysis (Preliminary Program of Detailed Investigations, 2000).

To achieve these goals, we need in Phase I to assess: (a) repository construction feasibility (seismological, geomorphologic, and preliminary geotechnical characterization of the site), and (b) safety analysis inputs (determination of groundwater circulation pathways and water-table depth and oscillations). Both of these activities are based on development of geological and hydrogeological models.

In Phase II, we will conduct a detailed site characterization of the microsite (about 1 km^2 in area), selected in Phase I. It is expected to be finished in three years, and should result in:

- Information to provide support for long-term repository safety, based on the development of a local transport model for radionuclides dissolved in groundwater
- A sufficient background of information for final repository design
- Validation of microsite geological, seismological, geomorphologic and geotechnical characteristics, of paramount importance to feasibility assessment and construction of the repository (Preliminary Program of Detailed Investigations, 2000).

The planned investigations are based on three major groups of methods: (1) geodetic methods (mapping, remote sensing), (2) geophysical survey (surface geophysics, geophysics in boreholes), and (3) drilling (shallow and deep boreholes).

Taking into account all these activities, the project time schedule—as far as it is presently predictable—is as follows:

- Completion and validation of preliminary site characterization 2002
- Allowance for access to site (planning documents+physical access) 2003
- Detailed on-site characterization: Phase I 2005
- Detailed on-site characterization: Phase II 2008
- Site approval 2009
- Disposal facility construction 2011
- Repository start-up 2012

9.7. CONCLUSIONS

The LILW repository project in Croatia started in 1988. Unfortunately, the project activities could not be completed in the predicted time because of circumstances beyond the regular project planning responsibilities. The most important among these delaying factors were: (1)

the end of the former Yugoslavia, (2) war in Croatia, (3) organization of the new state legislature in Croatia, (4) misunderstanding between Slovenia and Croatia over the ownership of the Krsko NPP, (5) hazardous access to some of the preferred sites resulting from mines left over from the war, and (6) lack of finances. Since the first three reasons are no longer issues, and the agreement between Slovenia and Croatia regarding the ownership of the Krsko NPP is expected to be made in the next few months, it is reasonable to assume that further project activities will proceed in a more predictable way. Meanwhile, some important preliminary site-characterization documents, as well as a preliminary disposal-facility design and safety analysis, have been prepared. Free access to the proposed site seems to be essential for the regular continuation of project activities, leading eventually to final site approval.

The main milestones in the project history are as follows:

- Adoption of site-selection methodology and development of multicriteria analysis (1988–1990)
- Site screening of the territory of Croatia (1991–1993)
- Selection of seven potential areas (1993)
- Identification of 34 potential sites within the seven potential areas (1994)
- Comparative analysis and evaluation of those 34 sites (1995–1996)
- Selection of four preferred sites (1997)
- Determination of two most suitable sites (1998)
- Selection of the final site and its inclusion into the Regional Plan of Croatia (1999)
- Incorporation of the site into regional and local physical plans (2000–2001).

Further activities, including detailed site characterization, completion of safety analysis, and development of the facility design, are planned to start in 2002. Final site approval would be granted in 2009, and construction of the repository would be completed in 2011. It seems that the repository start-up in 2012 would not be a serious obstacle to the further regular operation of the Krsko NPP and the accumulation of operational waste in the plant's temporary-storage facilities.

IAEAassistance to Croatian experts participating in the project was of great importance, and is expected to continue in project follow-up.

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Chapter 10

Progress Towards a Deep Geological Repository in the Czech Repubic

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10.1. INTRODUCTION

Radioactive waste management in the Czech Republic changed significantly after the Atomic Act, a law concerning the peaceful use of nuclear energy and radiation sources (Atomic Act, No. 18/1997 Coll.), came into force on July 1, 1997. To provide for activities associated with radioactive waste disposal, the Act established the Radioactive Waste Repository Authority (RAWRA) as a new state organization. RAWRA operates on the basis of a statute approved by the government. Among other responsibilities and actions, RAWRA has been engaged in the following activities:

- Preparation, construction, commissioning, operation, and closure of radioactive waste repositories and the monitoring of their impact on the environment
- Radioactive waste management
- Provision and coordination of research and development in radioactive waste management.

Critical RAWRA's priorities (since its establishment) have been to keep existing repositories in operation, to continue the development of a deep geological repository, and to formulate a proposal for a state Radioactive Waste Management Concept and other significant documents.

10.2. LEGAL FOUNDATION FOR WASTE MANAGEMENT

The Atomic Act (accompanied by a number of decrees and regulations) has created a firm and up-to-date legal foundation for waste management. The following provisions of the Act are particularly important:

- Under the terms of this Act, the Czech Republic guarantees the safe disposal of all radioactive waste, including the monitoring and supervision of repositories after their closure.
- To provide for activities associated with radioactive waste disposal, the Ministry of Industry and Trade of the Czech Republic establishes RAWRA as a state organization. RAWRAis required to carry out its activities based on a license issued under Article 9 of this Act. If RAWRA ceased to exist, its rights and obligations would be transferred to its establisher.
- The activities of RAWRA are financed from an interest-bearing account opened at the Czech National Bank (hereafter referred to as "the Nuclear Account"). The Ministry of Finance of the Czech Republic manages the Nuclear Account, which is included among state financial assets and liabilities, and the utilization of which is approved by the Czech government. The use of resources from the nuclear account is limited only to activities specified by this Act.
- RAWRA is engaged in the following activities:

  a) Preparation, construction, commissioning, operation, and closure of radioactive waste repositories and the monitoring of their environmental impact
  b) Management of radioactive waste
  c) Conditioning of spent or irradiated nuclear fuel into a form suitable for its disposal or further utilization

d) keeping of records on radioactive waste production and their generators
e) administration of levies paid to the Nuclear Account
f) drafting of proposals for the determination of levies to the Nuclear Account
g) providing for and coordinating research and development in the field of radioactive waste management
h) monitoring of the creation of reserves by licensees for the decommissioning of their nuclear installations
i) provision of services in the field of radioactive waste management
j) management of radioactive waste transported into the Czech Republic from abroad when it is not possible to return it
k) provision of a temporary administration in the case of radioactive waste that, under a specific Act, has become State property; if these are items that are found, left, or hidden, the RAWRA is also entitled to manage them.

- RAWRAoperates on the basis of a statute approved by the Czech government, a budget, and one-year, three-year, and long-term plans of its activities. RAWRA provides such activities mainly by contracting suppliers based on an assessment of nuclear safety, radiation protection, and economic benefit.
- The income to the Nuclear Account is specifically comprised of:

l) payments from radioactive waste generators
m)interest from the Nuclear Account
n) revenues from operations with Nuclear Account resources on the financial market
o) income received and payments claimed by RAWRA
p) subsidies, gifts, grants, and other income.

10.3. RADIO ACTIVE WASTE MANAGEMENT CONCEPT

According to the Atomic Act, RAWRA is responsible for the coordination of research in the area of radioactive waste management. In addition to the project for developing a deep geological repository, its attention is focused primarily on the preparation of the Radioactive Waste Management Concept (the Concept).

The Concept will serve as a basic document on which the long-term strategy of the government will be built with respect both to those organizations generating radioactive waste and those bodies and institutions otherwise involved in radioactive waste or spent-nuclear-fuel management. Hence, in the long-term, the Concept will also provide guidance for the activities of RAWRA.

The Concept's purpose is to come up with generally acceptable ideas concerning the management of radioactive waste and spent nuclear fuel in the Czech Republic, ideas that are strategically justified and scientifically, technologically, environmentally, financially, and socially acceptable. Hence, the document will provide a general system framework for decisions to be made by those bodies and organizations responsible for radioactive waste or spent-nuclear-fuel management.

Last but not least, the Concept will provide a source of clear information on the long-term strategy of radioactive waste and spent-nuclear-fuel management. The Concept is intended for all the institutions and individuals involved as well as for the general public. The design of the Concept relies on an analysis of the up-to-date development and professional estimation of future trends in the peaceful utilization of nuclear energy and ionizing radiation.

The Czech government is expected to approve the Concept after discussions on the environmental impact assessment portion of the Concept (SEA). The SEA study is expected to be completed during August of 2001. A statement from the Ministry of Environment of the Czech Republic on the Concept and the SEA is expected in September 2001.

10.4. REFERENCE DESIGN FOR THE DEEP GEOLOGICAL REPOSITOR Y

The reference design for a deep geological repository (Reference Design) was completed at the beginning of 1999. This document summarized the latest developments that might be employed in a potential solution for a deep geological repository system, covering both its underground and surface components. Without indicating any specific site, the location of the repository in granitoid rocks was considered. In the subsequent stages of development, this design will serve as a benchmark against which all the actual solutions will be compared. The document also includes the estimated requirements for funding and the time needed for the repository's construction and operation. Evidence of compliance with the requirements for nuclear safety and radiation protection is also included in the document. The potential environmental impact of repository construction and operation of the repository was also estimated. A proposal for research and development activities resulting from the Reference Design was set out, summarizing the designer's requirements for the preparation of background information and documents for activities within the repository.

10.5. PROGRESS IN SITE SELECTION

10.5.1. Evaluation of Pre-Existing Geological Information

The evaluation phase concerning pre-existing geological information, which began in 1995, was completed at the end of 1997. The Ministry of the Environment approved the report at the beginning of 1998. The development as well as the present situation of siting activities in the Czech Republic (and further activities included in the general project) are shown in the following chart (Figure 10.1). It should be noted that, pending the results of governmental discussions concerning the Czech waste management concept, all site-specific activities have been suspended.

During the critical collection of pre-existing geological information, 1,411 reports were evaluated containing descriptions of 1,999 boreholes from 13 selected areas. The majority of these reports are stored in the Geofond, the central archive of geological information for the Czech Republic, which is part of the Czech Geological Survey. In the Geofond, about 20,000 geological maps of different scales are stored, along with some 180,000 geological reports, which include descriptions of more than 550,000 boreholes. The evaluation considered a wide range of available geological information and its applicability to site characterization. We were also looking for information that allowed us to designate certain areas as unsuitable.

Both materials stored in the archives and all previously published information concerning the selected areas were evaluated from the following points of view:

- Geology, tectonics, mineral deposits exploration
- Borehole sinking and description
- Petrology, mineralogy, geochemistry
- Geophysics
- Hydrogeology
- Hydrology and climatological situation
- Engineering geology and geotechnics
- Demography and conflicts of interest.

Beside the reports associated with the 13 selected areas, some 825 reports containing information on seismicity and geodynamic movements were evaluated.

According to complex criteria, all the information obtained was divided into three groups:

- Information fully applicable
- Information with limited applicability
- Nonapplicable information.

Only high-quality information was considered applicable. Of course, some information would not be applicable for deep geological repository siting irrespective of its quality (e.g., information from shallow water wells situated in Quaternary sediments, engineering geology tests for small buildings, or geophysical measurements in small areas at shallow depths).

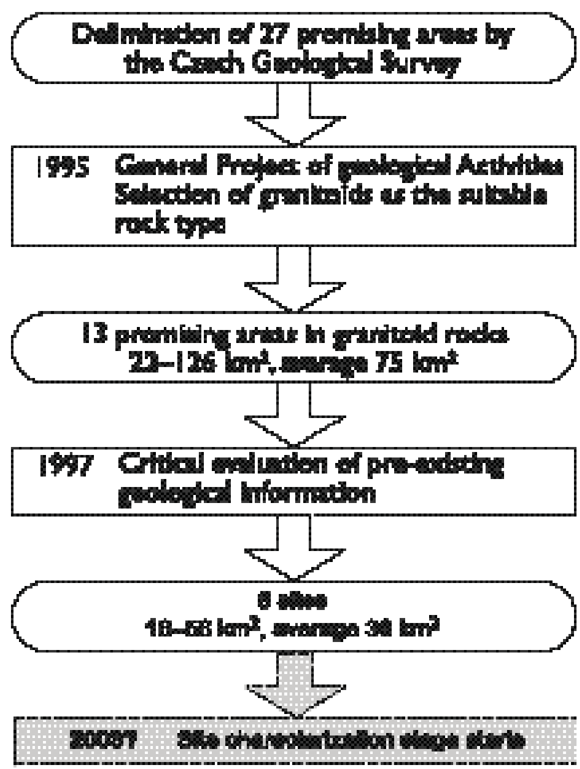

Figure 10.1. Development of siting activities in the Czech Republic

The largest number of reports were those concerning hydrogeology and engineering geology. For both these classes of reports, the applicability evaluation was practically identical. Less than 1% of all reports were fully applicable and more than 60% were nonapplicable. This is because of the limited extent and shallow depth of most of the information. Of the boreholes, some 60% were found to be nonapplicable, mostly because of shallow depths or low core recovery. However, some 34% of the information from geology, tectonics, and reserve-calculation reports were categorized as fully applicable and only 22% as nonapplicable.

An evaluation of the petrological, mineralogical, and geochemical information categorizes more than 40% as fully applicable and only 20% as nonapplicable. This positive evaluation reflects the fact that during the last decade, intensive studies of granite geochemistry have been carried out. Also categorized as fully applicable were the results of regional metalometry and heavy mineral prospecting, which covers a large part of the Czech Republic.

Only about 5% of the geophysical information was categorized as fully applicable. The criteria for full applicability require, for example, that the original data must be accessible for prospective reinterpretation. On the other hand, more than 50% of the geophysical information was categorized as having limited applicability.

10.5.2. Seismicity and Geodynamic Movement Monitoring

In this project, we proposed to locate individual stations for seismicity monitoring based on the positions of individual study sites and field checking. Seven stations are situated on each study site and their surroundings. Some stations are also located in the largest quarries near to individual study sites. The criteria for quarry selection have been (1) a distance of up to 50 km from the center of the study site and (2) the use of charges containing more than 200 kg of explosives. Also, certain individual points have been located for geodynamic-movement measurement, using a geographical positioning system. These locations were chosen on the basis of remote sensing and structural analyses of the sites and their surroundings.

The project proposes a 10-year monitoring period. Equipment to be used, the method of data transfer to the center, the methods of data processing, as well as the codes to be applied in processing and evaluating the results—are also described in the project. A progress report will be compiled after each year of monitoring that will contain the data obtained and require a preliminary evaluation of the data.

The project is based on the premise that work will start in the year 2003 with a re-evaluation of data from existing stations and with seismic noise measurements at proposed stations before their final location. The aim of these measurements is to obtain the optimum conditions for the subsequent monitoring. The project is accompanied by a time schedule and a calculation of expenses. Work on this project will start immediately after the Concept’s final approval from the Czech government.

10.5.3. Siting Activities and Their Legal Requirements

The main target of this document has been the selection of relevant requirements in legislation and the inclusion of steps leading to the fulfillment of these requirements with respect to geological activities.

Among numerous laws, the following contain the most essential requirements for geological activities on siting:

- Mining Act (No. 541/1991)
- Geological Act (No. 543/1991)
- Act on Environmental Impact Assessment (No. 244/1992)
- Act on Land Planning (No. 50/1976).

All the above-mentioned acts have been amended during the past few years and are accompanied by many special government decrees.

The laws on environmental impact assessment and land planning are clear as far as siting activities are concerned. These acts set out in detail which stage of the work needs to be reached and which particular data and information are necessary to obtain a decision or permission for further activities.

Mining and geological laws are not so clear concerning geological activities relevant to siting. This is primarily because both these laws are aimed predominantly at the exploration and mining of mineral deposits.

10.5.4. Overview of Existing Criteria

The State Office for Nuclear Safety issued Decree No.

215/1997 on the Criteria for Siting Nuclear Facilities and Very Important Sources of Radiation. This decree was issued in conjunction with the Atomic Act and respects the recommendations of the International Atomic Energy Agency and other international institutions. The decree, which does not focus directly on the siting of a deep geological repository, defines:

- Excluding criteria—17 regulations
- Conditional criteria—15 regulations.

The conditional criteria allow an area to be used only if there is the possibility of addressing any unfavorable conditions by technical means. The criteria defined in this decree have been divided, in the overview, into:

- Criteria adequate for the surface part of a deep geological repository
- Criteria adequate for the subsurface part of a deep geological repository
- Criteria inadequate for a deep geological repository.

Corresponding methodology has been defined to implement the conditions of each criterion. The overview is accompanied by a complete list of international recommendations. It is also concluded that certain criteria will be modified after obtaining relevant data from the first stage of geological work on site.

10.5.5. Results of Site-Selection Process

The number of reports and boreholes and the areal extent of individual selected areas evaluated in the course of the above-described investigations are given in Table 10.1.

The areas in bold type show sites recommended for the characterization stage. The following map (Figure 10.2) shows the location of selected areas and areas selected for the characterization stage.

10.6. SUPPORTING ACTIVITIES: PROGRAMS AND PROJECTS

The program for the development of a deep geological repository in the Czech Republic also includes relevant supporting activities, aimed at gathering information required for selection of a suitable site, design of engineering barriers, and safety evaluations. The main supporting requirements are as follows:

- A testing site
- An underground research laboratory
- Natural analogue studies.

10.6.1. Testing Site

Geological methods available in the Czech Republic have been reviewed and compared with those methods generally used worldwide. The result of this comparison shows clearly that a significant number of methods are not commonly used or are not developed to their maximum potential. It is necessary for us to learn these meth-

Table 10.1. Data evaluated for selected areas

Area code	Name of area	Area (sq.km)	Number of reports	Number of boreholes
4.	Vetrn´y Jenikov	65	87	278
5.	**West of Trest**	**106**	**104**	**137**
6.	**North of Nova Bystrice**	**85**	**89**	**228**
7.	**Klenov**	**52**	**39**	**14**
8.	Trebic	162	138	100
10.	Blatna	126	234	383
12.	Milevo – Brod	55	86	89
13.	Sedmihori	22	67	39
14.	**Tis**	**44**	**63**	**186**
15.	Cista	62	122	99
16.	Kdyne	37	99	67
30.	**Northeast of Milevsko**	**122**	**166**	**153**
31.	Southeast of Jihlava	44	117	226
	Total	982	1,411	1,999
	Average	75.5	108	154

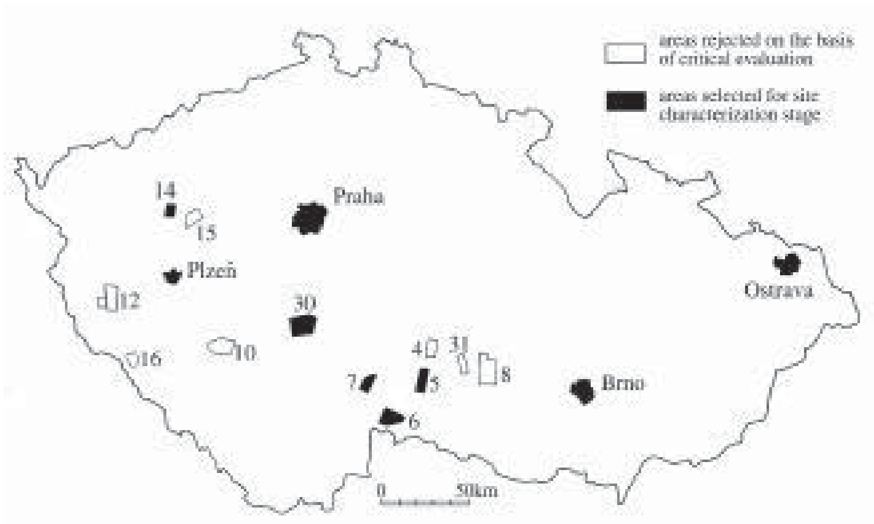

Figure 10.2. Map showing sites considered for characterization

ods, to evaluate their potential contribution, and to work out procedures for their use in the site-characterization stage. One of the important results will also be information concerning the economic ramifications of particular methods.

The Melechov Massif was selected as the testing site, which forms the northern-most part of the Central Pluton, from which it is separated (according to geological maps) only by a narrow strip of metamorphosed rocks. The geological position of the Melechov Massif is very close to the main area of study sites selected for site characterization. However, this massif is unacceptable as a study site because part of it is situated in a protected drinking water source area.

In the western part of the Melechov Massif (about 125 km^2 in area) the following work has been completed with the aim of obtaining detailed geological information:

- Geological mapping on the scale 1:10,000
- Hydrogeological investigation
- Airborne gamma-spectrometry and magnetic measurement
- Ground gravimetric survey
- Petrology of individual granite types
- Fracture measurement and characterization of individual fracture systems.

After the above-mentioned activities have been completed, it will be possible to select, within the area of the Melechov Massif, polygons with characteristic geological conditions for the testing of individual techniques and methods. We expect to complete this selection at the end of 2002.

During the year 2002, the selection of methods and techniques for testing and testing preparation will commence. The time schedule for individual methods testing will respect the time schedule of the site-characterization stage.

10.6.2. UNDERGROUND RESEARCH LABORA TORY

Based on international experience and progress, the Czech approach to underground research laboratory (URL) experiments was revised several times during the last few years. Some underground constructions, such as an underground gas storage project (1,000 m depth, with volume of 700,000 m^3) and abandoned mines, both built in granitic rock, have provided a good opportunity to verify methods involved in construction of a deep geological repository. Finally, with respect to economic aspects and the know-how available from foreign underground experimental facilities, it has been recommended to incorporate such a URL in the plan. This could provide evidence for approval of the final site for repository construction. This so-called "Confirmation URL" will be built after final selection of the repository site, which is scheduled to be completed in 2025. The laboratory itself shall start its operation in 2030, with duration of experiments exceeding 10 years. A detailed program has not been proposed; however, repository design demonstration and verification of results received during application of nonintrusive techniques are supposed to be the primary missions of this facility.

10.6.3. Natural Analogue Studies

The Nuclear Research Institute Rez (NRI), in close cooperation with Gesellschaft für Anlagen und Reaktorsicherheit (GRS) mbH Braunschweig (Germany), began a study of the natural analogue Ruprechtov in 1996. Ruprechtov is situated in the Sokolov Basin in the northwestern part of the Czech Republic. The sediments of interest belong to the Stare Sedlo and Nove Sedlo Formations of the Oligocene-Miocene age. The prevailing lithology is montmorillonitic clay (bentonite), intercalated with organic clay enriched in uranium.

Clay materials play an important role as engineered and geological barriers for repositories in deep geological formations. Models and computer codes for the simulation of all relevant processes are used for long-term performance assessment (PA) of repositories. The models and data are mostly derived from laboratory and *in situ* experiments. The crucial question is whether results from small-scale, short-term experiments can be extrapolated to real time and space conditions. The aim of the natural analogue study is to investigate the potential mobility of uranium in tertiary argillaceous sediments (under representative long-term conditions) and to contribute to data acquisition and model validation for PA.

The Ruprechtov site was chosen because:

- Its geological and geochemical conditions are similar to those of sediments that in many cases cover host rocks for underground waste repositories.
- Its argillaceous sediments might be an analogue for buffer/backfill materials.

As part of the investigation program, five boreholes have been drilled, and the geology of the site has been reviewed. It has been proved that uranium enrichments occur in Oligocene–Miocene sediments (clays and lignites) at a depth of 10–40 m. Geochemical, radiochemical, and mineralogical measurements are being carried out on selected samples to determine the natural distribution and mobility of uranium. Groundwater is monitored in two horizons where uranium has accumulated. In particular, we are investigating the long-term influence of such processes as diffusion/advection, dissolution/precipitation and sorption/desorption, as well as the impact of bacteria/organic matter on natural radionuclide and trace element migration. Analyses will be accompanied by geochemical modeling.

We have found that processes occurring at the Ruprechtov site are very similar to those expected in the far field of a repository. The following issues are especially important:

- The age of the processes studied covers a period of more than 10^6 years, which corresponds to the time frame considered in long-term safety assessment.
- Geochemical conditions (low Eh value, neutral pH value) are typical of those conditions expected in the surroundings of underground repositories.
- Tertiary argillaceous and sandy sediments (including lignite seams or lenses) are often found in the overburden of repositories.
- The groundwater flow conditions are similar to those in the far field of repositories (i.e., the Darcy velocities are in the range of a few meters per year).
- Observed uranium concentrations are in the range of 10^{-9} mol/l, which are the typical concentrations determined in long-term safety analyses in the far field of repositories.

Other anthropogenic analogues are being explored. Since 1832, certain glassworks in southwest Bohemia produced glass colored by uranium-bearing pigment. This pigment ($Na_2U_2O_7$) was prepared from uranium ore mined in Jáchymov (Joachimstahl). The production of this type of glass ended in about 1945. Studies of this glass began in the past year (2000).

In 2000, a depot of scrap glass was investigated near to the former glassworks at Rejstejn. The layer of scrap is about 1.5 m thick and is covered by soil and vegetation cover of variable thickness. Uranium-colored glass of different colors was found, using beta probing and a UV lamp. According to preliminary results, the average chemical composition of the glass batch is as follows: SiO_2 75%, K_2O 15%, Na_2O 1%, CaO 4%, P_2O_5 up to 2%, and As_2O_3 up to 3.6%. The content of uranium oscillates between 885 and 2,858 ppm (measured by gamma spectrometry in the laboratory) in individual types of glass. The thickness of the altered zone on the surface of individual fragments usually varies from 140 to 200 µm.

Currently, a set of 13 samples of different types of uranium-colored glass is being studied with the following main goals:

- To get more detailed information on the chemical composition and its variability
- To get information concerning alteration and recrystallization of different glass types
- To select optimal methods for a complex and more detailed study of this glass.

In our opinion, old glass colored by uranium-bearing pigments could be used as the anthropogenic analogue for the study of glass leaching and degradation. The results obtained could be interesting for a study of vitrified-waste stability.

10.7. CONCLUSIONS

The Czech program for a deep geological repository started in 1993, and after completion of its generic, conceptual phase, further steps are now being planned. Initiation of technical activities has been postponed until the national radioactive waste and spent-fuel strategy (the Concept) is approved by the Czech government (which is anticipated by the end of the year 2001).

The Concept includes the identification of two final sites by 2015 and the confirmation of one by 2024. The repository should be commissioned after 2065, assuming a decision by the NPP operator to prolong the spent-fuel storage period beyond 2050.

It has been shown that the Czech nuclear waste storage program will be able, using the experience gained in the past 40 years and using internationally available know-how, to safely construct and operate a deep geological repository.

Chapter 11

Decision-in-Principle on Final Disposal of Spent Fuel in Finland

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11.1. INTRODUCTION

According to the present law in Finland, all spent fuel from the Finnish nuclear power plants must be handled, stored, and permanently disposed of in Finland. For the interim storage of spent fuel, both nuclear power plant owners, Teollisuuden Voima Oy (TVO) and Fortum Power and Heat Oy, have their own facilities. For permanent emplacement, the companies have established a joint company, Posiva Oy, to take care of the research and development needed for the geologic disposal of spent fuel, and later to construct and operate the disposal facility.

The guidelines and time schedules for the work towards final disposal were first set in a Finnish government decision of 1983. In 1991 and 1995, they were slightly modified, but the principal time schedule has been left unchanged, and so far it also corresponds to the progress made (Figure 11.1). The main target is the year 2020, when Posiva should be ready to start disposing spent fuel. However, the most important milestone so far has been the selection of a site for the final repository, which, according to the guidelines, was to be made by the end of the year 2000.

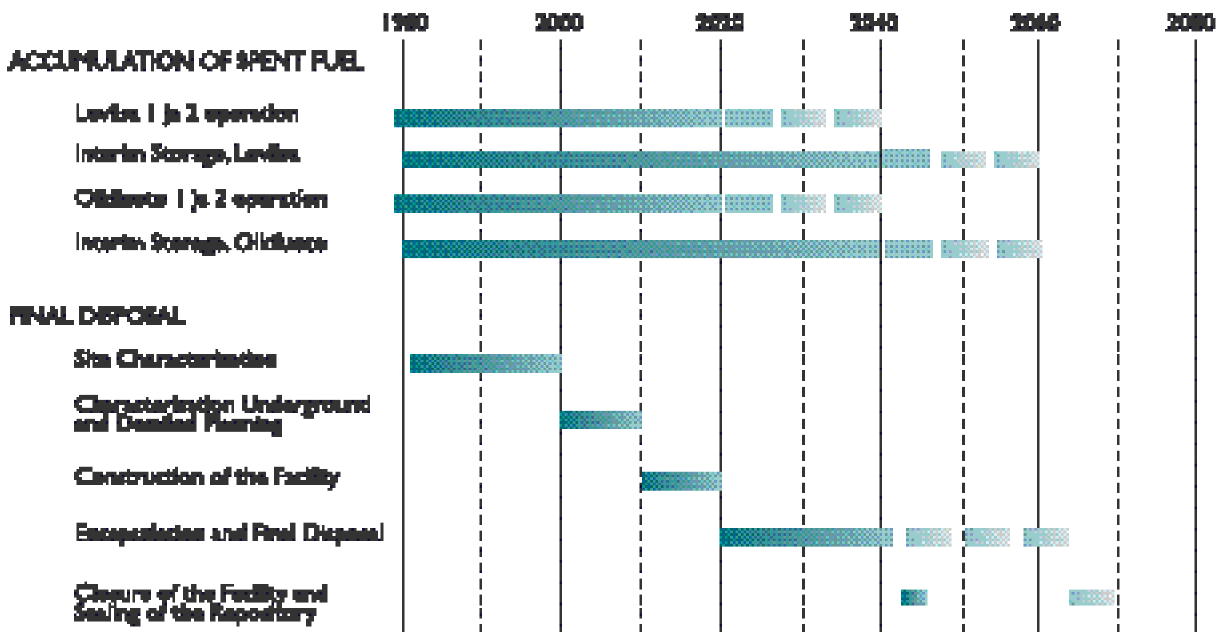

Figure 11.1. Time schedule for spent-fuel management in Finland

In May 1999, Posiva submitted an application for the so-called Decision-in-Principle (DiP) on the final disposal facility of spent fuel to the Finnish government (Posiva, 1999a). According to the application, the spent fuel from the Loviisa and Olkiluoto nuclear power plants would be disposed of in a KBS-3-type final repository to be built at Olkiluoto in the municipality of Eurajoki, where the TVO nuclear power plant is also situated. The main purpose of the DiP is to judge whether the facility is in line "with the overall good of society," but another important part of the decision is the judgment on the suitability of the intended facility site. The decision is made by the government, but has also to be ratified by the Finnish parliament.

The application was based on the outcome of some twenty years of research and investigations, and on the environmental impact assessment (EIA) that was carried out in the years 1997–1999 for the final disposal facility (Posiva, 1999b). The law requires the DiP before any significant commitment (e.g., investments) to a nuclear facility is made in Finland. In this case, the decision was needed before making a decision on an underground rock-characterization facility.

The Finnish government approved Posiva's application in December 2000, and the debate in parliament started

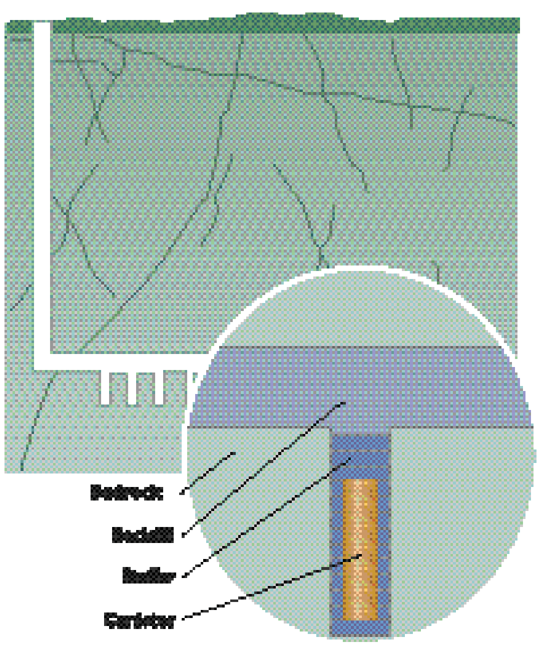

Figure 11.2. The KBS-3 concept for disposal of spent fuel

in February 2001. The final decision is expected before summer 2001. However, Posiva has already started the planning for the next program phase on the assumption that a positive decision will be made. A document containing a program for research, development, technical planning, and design work for the period preceding the construction license was published in early January 2001 (Posiva, 2000). In this program, the emphasis is on the needs arising from the application for construction license. According to the current official guidelines, Posiva should prepare for submitting the application in 2010.

In this report, the progress in Finland since the *Second Worldwide Review* in 1996 (Vira, 1996) is summarized, and the program for the near future is outlined.

11.2. DISPOSAL CONCEPT

Posiva's plans for final disposal of spent fuel are based on the KBS-3 concept developed by the Swedish Nuclear Fuel and Waste Management Company (SKB). The spent-fuel elements will be encapsulated in metal canisters and emplaced at a depth of several hundreds of meters in the bedrock (Figure 11.2). Between rock and canisters, a buffer of bentonite clay would be installed; after completion of the disposal effort, all access routes from the surface to the disposal galleries would be closed and sealed, and the repository would need no further control or maintenance.

Since its conception, the KBS-3 method has been subjected to a considerable amount of research and development, and during this time it has undergone some changes. In particular, the canister structure has evolved from the original lead-filled copper canister design of the late 1970s. The present canister concept consists of two parts: a massive inner structure of nodular cast iron to withstand the mechanical loads and an outer hull of copper to act as a corrosion shield (Figure 11.3).

In spite of "fine tuning," the basics of the idea are still much the same as 20 years ago. In the early 1990s, an extensive study (PASS) was carried out by SKB to compare possible alternatives to KBS-3, and comparisons of different concepts have also been made by Posiva (e.g., Autio et al., 1996; Posiva, 1999b), but so far KBS-3 has turned out to be the most promising alternative. Nevertheless, some variants to it (like horizontal tunnel emplacement) are considered as possible alternatives, both in Sweden and in Finland.

The long-term performance of the concept is, of course, still a subject of lively research. However, growing emphasis is being laid on the practical implementation of the concept. In particular, it needs to be demonstrated that a capsule with the desired long-term properties can really be manufactured and closed as assumed in safety assessments. To date, Posiva has manufactured two copper canisters using two different methods. In addition to manufacturing technology, future studies will strongly emphasize methods of canister sealing and nondestructive testing of the seals.

The present reference design for the copper canister has 5 cm wall thickness. In the future, the feasibility of thinner, 3 cm walls will be assessed.

11.3. SITE-SELECTION PROCESS

The planned disposal site is situated at Olkiluoto, in the vicinity of the Olkiluoto nuclear power plant. Olkiluoto is an island belonging to the municipality of Eurajoki, near the city of Rauma, on the western coast of Finland. Eurajoki has a population of slightly more than 6,000 inhabitants.

Near the power plant, a repository for low- and intermediate-level operating waste from Olkiluoto reactors (VLJ Repository) already exists. This repository is built at a depth of 60–100 m and consists of two vertical silos. The planned area for the spent-fuel repository is a few kilometers east of the existing repository.

Olkiluoto was selected as one of the candidates for the spent-fuel repository site in 1987, together with four other candidates: Romuvaara in Kuhmo, Veitsivaara in Hyrynsalmi, Syyry in Sievi, and Kivetty in Konginkangas (later part of Äänekoski). The site-selection process between 1983–2000 is outlined in Figure 11.4.

Preliminary site investigations were started in 1987. In 1992, an interim assessment was made in which three of the candidates were selected for further studies. Olkiluoto was one of them. In 1996, after the change of law that terminated the transport of Loviisa spent fuel to Russia, site investigations were also started at Hästholmen in Loviisa. Consequently, the choice of the repository site in 1999 was made among four candidates: Romuvaara, Kivetty, Hästholmen, and Olkiluoto.

Each of the four candidate sites has been subject to an extensive program of surface-based investigations and modeling studies. At least ten deep drillholes (of depths between 500 and 1,000 m) have been made at each site; at Olkiluoto already twelve. The investigations and interpretations from the preliminary site investigations were summarized and concluded in 1992 by TVO (1992). In 1996, an interim report was published by Posiva (1996) and four separate site reports were compiled for the site assessment in 1999 (Anttila et al., 1999a,b,c,d). The final site selection was based on the outcome of the EIA conducted in 1997–1999 (Posiva, 1999b).

The entire site-selection process, including the investigations already carried out, has been described in a report by McEwen and Äikäs (2000). In addition to the investigation history, the report makes some conclusions regarding the lessons learned for further studies by Posiva and also for other programs planning site-characterization work.

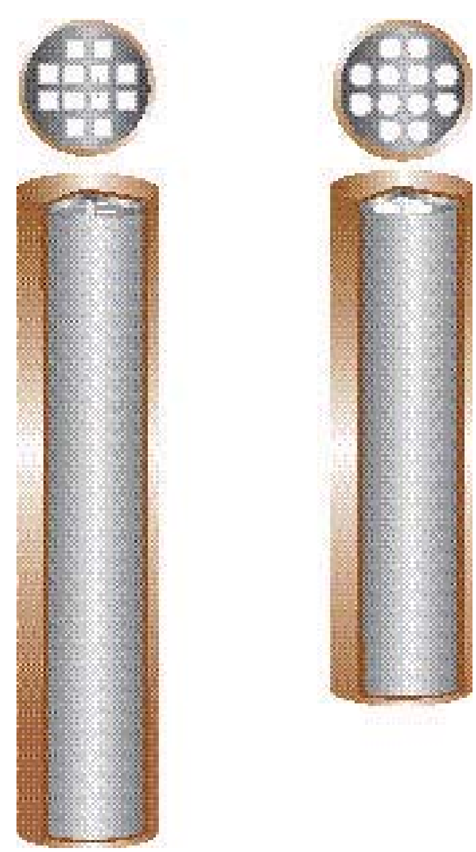

Figure 11.3. Section views of the disposal canisters: the BWR type (left) and the VVER 440 type (right)

11.4. ENVIRONMENTAL IMPACT ASSESSMENT

11.4.1. Assessment Program

The basis for the DiP application was included in the report from the EIApublished in May 1999. In general, the purpose of the EIA legislation has been to bring more transparency and interaction among potential stakeholders into the planning of projects that may have a significant impact on their physical or social environment. It is believed that the EIA should look particularly at different alternatives for project implementation and also consider the impact of not implementing the project at all (i.e., the so-called zero-alternative). Altogether, the emphasis should be on the impacts that the public finds of greatest concern.

In the scoping stage, Posiva's EIA process paid considerable attention to the interaction with the public in the communities that were candidate sites for the facility (on the basis of the site-selection process). The program for the EIA presented in 1998 was particularly focused on concerns expressed by local people in various meetings or in written comments and statements.

As could be expected, safety (both operational and long-term), including transportation safety, was among the top concerns. In addition to articulated safety concerns, one of the issues most frequently brought up in the public discussion was the question of "image": How would the project affect the public image of the home municipality or the whole region? It was evident that a multitude of issues (such as prices of real estate, marketability of agricultural produce from the region, and attractiveness to tourism) was at stake in the concerns of the public. Most likely, however, the image concerns can also be considered as overall negative feelings about the planned facility.

The assessment program was submitted in January 1998 for public review (Posiva, 1998). The contact authority's (Ministry of Trade and Industry) review statement was obtained in June 1998. The statement included the opinions of neighboring countries (Sweden, Estonia, and Russia).

The main concerns brought up in the official statements on the EIAprogram from various institutions were related to the assessment of alternatives to Posiva's proposed spent-fuel disposal concept. A special topic introduced into discussion during the EIAprocess was the question of reversibility. In March 1999 the Finnish government made a decision on general safety requirements for the final disposal of spent fuel that included the requirement that spent fuel must be retrievable, even after the repository was closed.

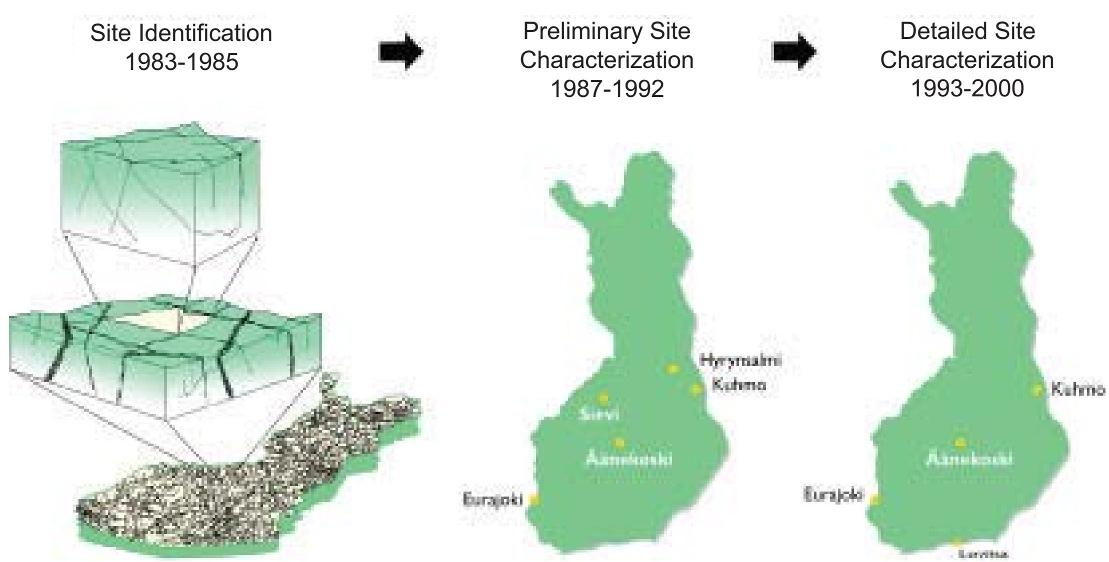

Figure 11.4. The stepwise site characterization and evaluation process for siting the deep repository in Finland

11.4.2 EIA Studies

Research into the long-term safety of final disposal has been carried out in Finland since the early 1980s. This work has already produced several comprehensive safety assessments since 1985. For the purposes of the EIA, the scope of the research program was substantially extended to address the assessment topics and issues identified in the assessment program.

A great many of the new studies were devoted to the expected social impact. In the general approach, it was admitted on the one hand that the local people were experts in matters related to their social environment and in their own perceptions of changes in their living environment. On the other hand, it was also considered that the eventual social impact of a project spanning over several decades could not be judged on that basis alone. Therefore, it was decided that rather than making firm statements about the likely impact, a spectrum of possible impacts would be described and analyzed. In this way, the assessment sought to combine the subjective experience of local citizens with the more general information obtained from other projects of similar nature or from published literature on social research. Of course, the difficulty with this approach was that no experience on the social effects of high-level nuclear waste repositories is available, and the relevance of any suggested analogy can easily be contested.

The areas addressed in the social-impact assessment included:

- Effects brought about by changes in the physical environmental (related to every-day living conditions and general living standards, amenities, and social well-being)
- Effects on community and population structures and infrastructures
- Effects on local and regional economics and well-being
- Psychosocial effects (anxiety, psychological effects).

The socioeconomic assessment was based on traditional input-output analyses, taking into account the direct and indirect effects on local and regional economic activity. The sociological and psychological aspects of the planned activity were discussed both in terms of the outcome of interviews and opinion surveys, and in terms of recent scientific studies and literature. Common to the whole assessment of social impact was the issue of image. A considerable amount of effort was spent on attempts to elaborate the meaning of image and constituents thereof, because it was clearly affecting people's opinion, both in likely economic impact and future social well-being.

Hence, safety as perceived was a part of the social-impact assessment. Of course, safety was also studied on the basis of technical and scientific analysis. Anew long-term safety assessment, "TILA-99," was produced as part of the EIA activity (Vieno and Nordman, 1999). TILA-99 considers the long-term safety of disposal against the general regulations issued in a March 1999 government decision (STUK 1999). The discussion is specific to the four candidate sites, taking into account all the results from site investigations conducted since 1987.

In addition to the long-term safety assessment, new evaluations were prepared on operational safety (Rossi et al., 1999) and the safety of spent-fuel transportation from the power plants to the encapsulation facility (Suolanen et al., 1999), which in the reference design is planned to be co-located with the repository. A review was also made of the possible chemical hazards arising from the disposed materials (Raiko and Nordman, 1999).

11.4.3. Assessment Results

The general conclusion of the EIA was that the likely impact on public health and natural conditions would always be small. The facility will have some impact on the local economic activity and employment, but this effect will most likely be positive. There may also be some negative social effects, mainly arising from the negative image and risk concerns that some people associate with nuclear waste.

The assessment of long-term safety concludes that most likely there will never be releases that would lead to meaningful radiation doses to any individual. Even in the case of the worst future scenarios, the doses would not exceed the background level. All in all, the regulatory requirements can be met with considerable safety margins, independent of which site is chosen. The conclusions of the studies into operational and transportation safety are similar: there should never be any significant health detriment from the planned operations, and even the likelihood of a significant radioactive release from the plant or transport casks is minimal.

Table 11.1. The final disposal options examined from the perspective of various ethical and

	Base Alternative and Variations	Hydraulic ca ge
A. Ethical and Ecological Principles		
1. Protection of man and nature	Wastes are isolated from nature so that active maintenance is unnecessary.	Wastes are insultated against nature so that active maintenance is unnecessary.
2. Protection of future generations	The multiple barrier principle supports the isolation of wastes as long as they may present a danger.	Wastes remain separated from nature as long as hydraulic insulation is operationally effective.
3. Avoidance of burden on future generations	Does not require action on the part of future generations but does not prevent it either.	Does not require actions as long as hydraulic insulation is operationally effective. Retrievability is dependent on packaging.
4. Operational safety of facilities	Can be implemented by means of strict release criteria.	Can be implemented by means of strict release criteria.
5. Prevention of misuse of nuclear materials	The illicit seizure of nuclear wastes would be arduous,costly and easily discernible.	The illicit seizure of nuclear wastes would be arduous, costly and easily discernible.
B. Technical Implementation		
1. Technical maturity level	Based on technology available in Finland.	The technology required is already in existence.
2. Readiness for selection of disposal site	The required investigations have, for the most part,already been carried out.	Research already undertaken can be utilized to its advantage.
3. Costs	Costs have been anticipated.	Much costlier than the base alternative.
4. Need for transportation	Transporation shall be needed to a certain extent regardless of disposal site.	Transportation shall be needed to a certain extent regardless of disposal site.
5. Suitability for other energy system(s)	Appropriate for the nuclear energy system respective to a small nation.	Appropriate for the nuclear energy system respective to a small nation.
C. Applicability from the Legislation Perspective of Present and Regulations		
1. Compliance to the Nuclear Energy Act and its regulations	Fulfills the requirements.	Fulfills the requirements in the hydraulic insulation is operationally effective without maintenance.
2. Compliance to the safety regulations of the Finnish Radiation and Nuclear Safety Authority	Can be shown to comply.	Long-term indication of functionality in respect to hydraulic insulation is problematic.

ecological principles, technical feasibility, and legislation		
"Deep Hole"	Reprocessing and Final Disposal	Nuclide Separation, Trans- mutation,and Final Disposal
Wastes are isolated from the environment in such as manner that active maintenance is not needed	Useful materials are separated for further use;the rest are isolated from nature.	Useful materials are separated for further use;the rest are separated from nature
Reliable assessment is not possible given the available data.	Final disposal can be implemented in the same manner as in the base alternative.	Final disposal as above;period of danger associated with wastes is shortened.
Does not require action,but retrieval of wastes is virtually impossible.	Final disposal can be implemented in the same manner as in the base alternative.	Requires future generations to develop the technology required.
Errors in handling may result in consequences difficult to control.	Reprocessing results in more releases than power stations.	Nuclide separation results in more releases than power stations.
Illicit seizure of nuclear wastes would be very difficult and dangerous.	The possibility of nuclear waste seizure is dependent on supervision.	Would also create potential for the production of nuclear weapons.
May rest in practice on available technology.	Based on technology employed abroad.	The technology needed is non-available.
Required new sorts of site characterization and the development of investigation methods.	Depends on method of implementation. Siting of reprocessing plant as with power plant.	Depends on method of implementation.Siting of partitioning facility as with power plant.
Cost difficult to assess.	Much costlier than the base alternative.	Costs unknown as the technology is non-availble.
Transportation shall be needed to a certain extent regardless of disposal site.	Many kinds of transport required.	Many kinds of transport required.
Appropriate for the nuclear energy system respective to a small nation.	Poorly appropriate to a small nuclear energy system.	Mandates long-term commitment to the use of nuclear energy.
Fulfills requiredment in the event that safety can be ensured.	Both reprocessing plant and final disposal facilty should be built in Finland.	Separation,transmutation and final disposal facility should be built in Finland.
Hardly complies to requirement for complete isolation.	Regulations are lacking in respect to reprocessing.	Regulations are lacking in regard to separation and transmutation.

Table 11.2. Comparison of implementation and nonimplementation

	Base Alternative and Deriviatives	Continuation of Water Pool Stora ge
A.Ethical and Ecological Principles 1.Protection of man and nature	Dangerous materials are insolated in such a manner that no active maintenance is required.	Dangerous materials are maintained in active operation,separate from nature and man.
2.Protection of future generations	Safety analyses indicate that adequate isolation shall function as long as the wastes are a danger man or nature.	As long as the storage areas are supervised and maintained,no danger shall be posed to man or nature. Neglect of care could lead to environmental contamination.
3. Avoidance of burden on future generations	Does not require action from future generations but at the same time does not pr event it.	Continued maintenance of storage areas, supevision and renewal are relegated to future generations.
4.Safety of facilities while in operation	Safety can be guaranteed by means of strict release criteria.	Safety can be guaranteed by means of currently available principles of operation.
B. Environmental Impact as Assessed 1.Effects on nature and utilization	Effects would be minimal and restricted to the immediate vicinity of the facility.	Do not deviate from present power plant-related impact.
2.Effects on land use and landscape	Required a few dozen hectares of land.Efects on landscape quite limited.	Do not deviate from present power plant-related impact.
3.Effects on human health	Aside from possible psychosocial influence, the project has no significance in respect to human health.The effect of stress would evidently be the most minimal in the current power plant sites.	Does not affect human health as long as the storage areas are maintained and supervised. Neglect of care could,with time, also result in health risks.
4.Social Effects	The project would exert positive socioeconomic influence. Fears, worries,contradictions and image problems would be at their minimum at the power plant sites.	No substantial positive impact.Concern for the condition of storage areas as well as their maintenance could,with time, result in social conflicts.
C.Technical Implementation 1. Technical development level	Does not require the development of new technical methods,but room is allowed for procedural development and optimization.	The technology is available in Loviisa and Olkiluoto.
2.Disposal site: readiness for selection	The investigations required in respect to site selection are, in the main, complete and allow assessment.	The current storage sites are appropriate for the purpose.
3.Costs	Costs are moderate and they have been anticipated in the price of electricity.	Cheaper than the base alternative, but over the long term the uncertainty of financing rises.

An important part of the EIA was the assessment of alternatives. This assessment included several levels:

1. Alternative spent-fuel management concepts
2. Different technical concepts for geologic disposal
3. Comparison of the proposed concept with the zero-alternative
4. Site alternatives.

On the first level, the past discussion of suggested alternatives was briefly reviewed, addressing different concepts of geologic disposal but also such exotic ideas as ocean and space disposal. On the basis of this review, four alternatives were first contrasted to the base alternative (KBS-3 and its variations). These included the idea of the hydraulic cage (WP Cave), deposition in deep boreholes, reprocessing (with subsequent disposal of the wastes from this process), and the partitioning and transmutation process. The conclusion of this comparison was that:

- The reprocessing alternative would also lead to disposal of high-level waste, the impact of which would be similar to direct disposal of spent fuel.
- The feasibility of partitioning and transmutation cannot be judged on the basis of present information and the alternative, therefore, reduces to the zero-alternative.
- The hydraulic cage and deep-borehole alternatives would require considerable effort without clear benefits; the retrievability of waste in the case of deep boreholes would be questionable.

The comparisons are summarized in Table 11.1. It was concluded that the base alternative would present the best promise for further development.

On the next level, the base alternative was contrasted to the zero-alternative, continued interim storage (Table 11.2). The conclusion of this comparison was that both alternatives were technically feasible and both alternatives could provide sufficient protection of man and environment. However, this would be true for the zero-alternative only if we assume continuous maintenance of the storage. The active development of a permanent, passively safe method of disposal was considered necessary to provide against the risk of discontinuation in the maintenance of temporary waste storage.

The assessment of alternative sites for the disposal facility was based on a number of factors, including:

- Long-term safety
- Constructability
- Possibility of expanding the repository
- Operation of the disposal facility
- Social impact
- Land-use and environmental loading
- Availability of infrastructure
- Costs.

Regarding long-term safety, an important conclusion of the TILA-99 safety assessment was that there is no possibility of objective ranking of the alternative sites in terms of the long-term performance of the repository. There were differences between the sites, but none of the sites was clearly superior or inferior to the other sites. Salinity of the groundwater was the most important single factor behind the differences. However, as regards long-term safety, both positive and negative aspects are associated with salinity.

An obvious difference between the sites was that of social acceptance and the expected social impact. In this respect, there was a clear difference between the inland and the coastal sites. The situation, as unfolded in the studies, interviews, and opinion surveys is summarized in Table 11.3. Accordingly, the expected social impact would clearly be smallest in Eurajoki. In Loviisa, opinions were polarized: the majority of the local population seemed willing to accept the siting of the repository and expected mainly a neutral or positive impact from the facility, but an active minority was strictly against the facility. There is apparent opposition in Eurajoki, as well, but a reconciliation is much more likely there than in Loviisa.

11.5. DECISION-IN-PRINCIPLE

In the application for the DiP, Posiva proposed that the Olkiluoto site be selected as the site of the disposal facility. The conclusion of the safety analysis TILA-99 was that none of the four candidate sites was either superior or inferior to the other sites, and all four sites were considered suitable for siting the repository. On the other hand, it was clear from the public interactions and opinion surveys that the repository would not get local acceptance in Kuhmo or Äänekoski. Finnish legislation makes it impossible to site the repository anywhere without the permission of the local municipal council. In practice, the choice could, therefore, only be made between Olkiluoto and Hästholmen. Among the factors supporting the choice of Olkiluoto was that most of the accumulated spent fuel was already at Olkiluoto, and that at Eurajoki, any major social controversies could probably be avoided (based on the EIA).

Since May 1999, the application for the DiP has been subject to an extensive review process. In addition to the local veto, a strong power of veto on safety matters is left with the regulatory authority, STUK. According to the Nuclear Energy Act, STUK has to make a preliminary assessment of the proposed facility vis-à-vis the legal requirement that the use of nuclear energy has to be safe. When the Finnish government considers the application, it has to state that no factors have been identified that would indicate the lack of sufficient provision for safety.

STUK's assessment was reported to the Finnish government in January 2000 (Ruokola, 2000). For the assessment, STUK requested a review by independent experts from the international scientific and technical community. In their statement, STUK considers that, first, the DiP is justified on safety grounds and, second, the DiP not only can be made but also should be made. STUK agrees that the KBS-3 concept is currently the most promising method to provide a permanent solution for spent-fuel disposal in Finland. In addition, STUK agrees that Olkiluoto is at least as good as the other site alternatives.

Shortly after STUK's assessment was published, the municipal council of Eurajoki made their decision. After a vote (20–7), the council approved the proposed siting of the disposal facility within its area. The council decision gave rise to two appeals to the regional court of administration and later to the Supreme Court of Administration, but all these appeals were rejected. In December 2000, the Finnish government approved the application.

The Finnish Parliament started the debate on the DiP in February 2001. In March 2001 the Parliament Committee of Environment gave their statement in support of the decision. On May 18th, 2001, the Parliament ratified the decision, virtually unanimously (only three Parliament members out of 200 voted against, with 37 members absent from the plenary session).

11.6. THE FUTURE

Posiva has started the planning of the next program phase. In January 2001, a program was published that describes the objectives and major items of research, development, technical planning, and design work for the period preceding the construction license (Posiva, 2000). According to current official guidelines, Posiva should prepare to submit the application for a license in 2010.

For the technical development and design work, the main target for starting the initial program phase is to reach an advanced stage in design and technical plans. Reaching such a stage will allow the specification of work packages for bid calls and provide sufficient confidence in the technical feasibility of planned operations at the encapsulation facility and in the repository. The main objectives for the complementary characterization work at Olkiluoto consist of verifying the present conclusions on site suitability, defining and identifying suitable rock volumes for repository space, and characterizing the target host rock for repository design, safety assessment, and planning of construction work.

The technical design and demonstration work, together with the results of complementary site characterization, will provide the basis of the safety case that is needed to support the construction-license application. An integrated safety assessment will be the cornerstone of the long-term safety case, but the argumentation for safety will also be based on broader technical and scientific evidence.

A substantial part of program activities will be staged according to the progress of the planned underground rock-characterization facility, ONKALO, at Olkiluoto.

Table 11.3. Significance of project effects in different municipalities, assessed from the inhabitants´ perspective

Expected size and significance of the effects	Safety worries and their consequences:effects on image, the character and pleasantness of the place	Direct and indirect economic effects and the significance of the project for the development of the municipality	Questions and conflicts relating to planning and decision making
Eurajoki	small	small	small
Kuhmo	great	great	great
Loviisa	great/small	great/small	great/small
Äänekoski	great	small	great

The plan is to start the construction of ONKALO in 2003–2004. Before that, Posiva will complete documentation concerning the:

- Baseline description of the Olkiluoto site
- Technical design description of the planned underground rock characterization facility
- Plan for underground investigations
- Review of rock characteristics that are important for the long-term safety of disposal, together with an assessment of how the ONKALO project would affect these.

Later, at the end of 2005, Posiva will publish an interim progress report of the disposal program. The report will also contain a more detailed description of the activities planned for the latter half of the program period.

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Chapter 12

Status of Research on Geological Disposal for High-Level Radioactive Waste in France

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12.1. LEGAL PROCESS OF RESEARCH

As mentioned in the *Second Worldwide Review* (Raynal, 1996), the French Parliament passed a law on December 30, 1991, that defined and regulated the management of high-level radioactive waste research programs in France. This law identified the research directions to be investigated, guaranteeing joint action with French citizens and transparency in assessing the research, and charged ANDRA with examining the options of reversible or irreversible disposal of high-level radioactive waste (HLW) in a geological formation. In January 1994, the French government authorized ANDRA to conduct investigations on three volunteer sites in the Gard, the Vienne, and the Meuse/Haute-Marne "départements," following the approval of the local authorities. This reconnaissance phase took place exclusively on the surface (detailed geological mapping, reflection seismic surveys in sedimentary areas, and gravimetry/aeromag surveys in granitic areas with deep boreholes).

These field operations were completed by summer 1996, and ANDRA then filed applications with the government for permits to install and equip underground research laboratories for each of the sites. A local public enquiry was then conducted, and the local authorities again expressed their approval through a new vote. A technical examination was carried out simultaneously by the Nuclear Safety Authority. The National Evaluation Commission, in charge of supervising the different research fields in connection with the 1991 law, held many hearings with ANDRA between 1996 and 1998. The overall process led the government (on December 9, 1998) to authorize ANDRA to construct an underground laboratory on the Meuse/Haute-Marne site.

The government also confirmed its interest in pursuing research in a granitic environment, but on a new site.

On August 3, 1999, the French government authorized two decrees: (1) to install and operate an underground research laboratory at Bure (Meuse), in Callovo-Oxfordian argillites, and (2) to again investigate the French granitic batholiths and propose a new list of potential sites. At the same time, the French government appointed three senior officials to communicate with the public (the "Granite Mission") to examine the possibility of installing and operating a second underground research laboratory.

These decrees have begun a new stage in the ANDRA research program for studying the options of reversible and irreversible disposal of HLW in a geological formation. The concept of reversibility, which was novel at the time, is a major objective of research today, at the same level of importance as the safety of the disposal concept. This decree is accompanied by a specification for the underground laboratory, which regulates its operation.

In response to the concerns expressed by the National Evaluation Board in its report (No. 4) on the quality of the radioactive-waste inventory, the government entrusted the President of ANDRA with the mission "to propose any suitable reform with a view to ensuring the reliability of the waste inventory and, notably, the medium and long-term extrapolation of that inventory." After one year of preparation, the report of that mission on "the methodology for a radioactive-waste inventory" was submitted to the government in June 2000. A new

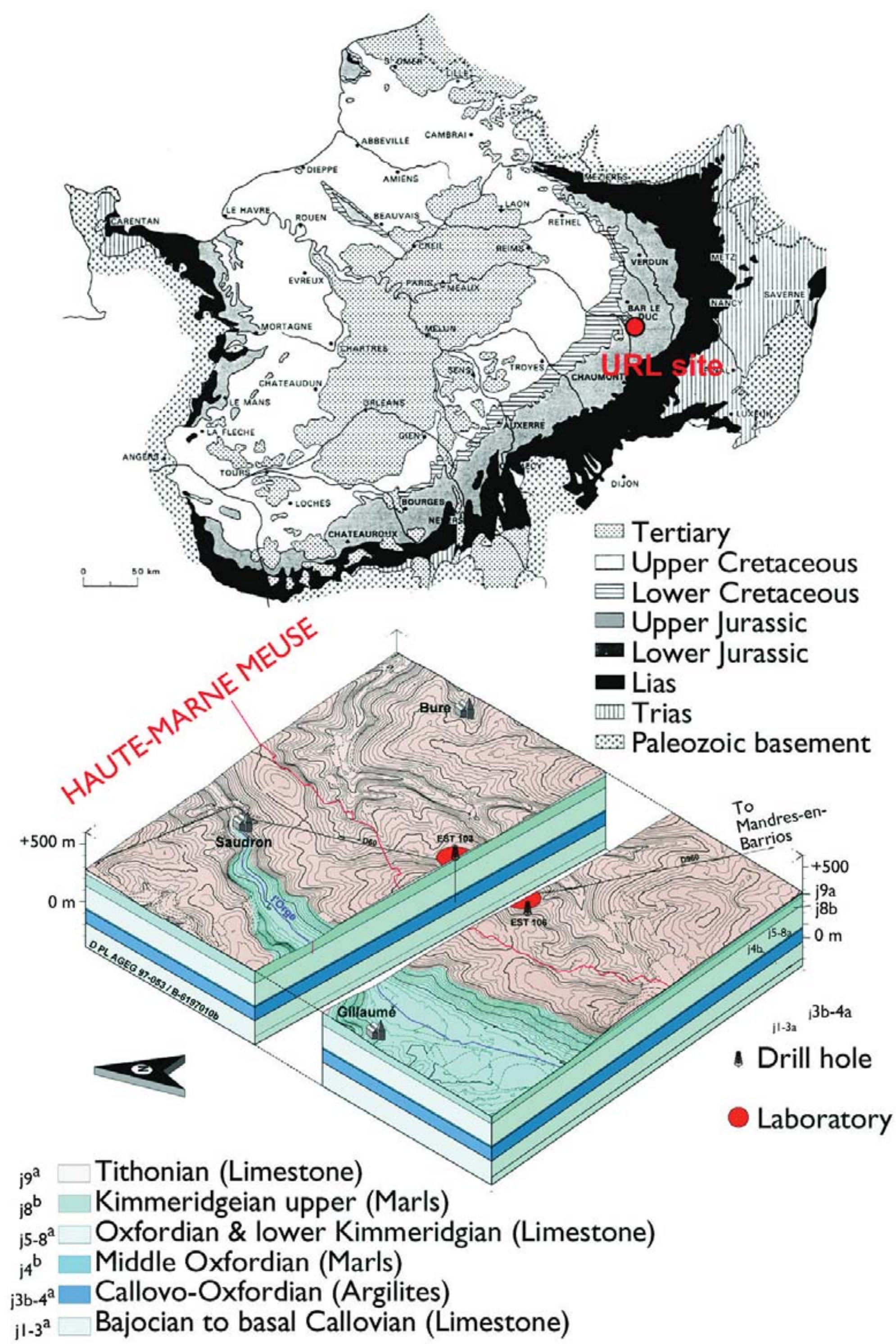

Figure 12.1. Location and block diagram of the Meuse/Haute-Marne site

national inventory, consistent with the proposed methodology, may be available in 2003, provided that the proposals contained in that report are accepted immediately and that the corresponding means to carry out the proposal are released.

12.2. EXECUTION OF RESEARCH PROGRAM ON MEUSE/HAUTE-MARNE SITE BY 2006

12.2.1. Main Features of the Site

The Meuse/Haute-Marne site is located on the eastern margin of the main French sedimentary basin, the Paris basin (Figure 12.1). The deep structure of this basin is well known from numerous industrial uses (extraction of water and oil, geothermal energy, gas storage, mining). The geology is very simple in the area of the URL's site: alternating carbonate and clay formations with uniform dip (2° NW) and thickness. ANDRA has targeted the Callovo-Oxfordian argillites lying at a depth of 400 to 500 m (130 to 135 meters thick over more than 10 km, Figures 12.1 and 12.2).

The Callovo-Oxfordian argillites are very tight (permeability below 10^{-13} m/s). Sedimentation cycles are clearly marked, particularly by the change in the carbonate content, and the slight influence of the physical-chemical properties of the formation (Table 12.1):

- A higher illite/smectite ratio in the lower half, as well as higher salinity of the pore water
- Lower mechanical strength (20 MPa) in the middle, more clayey portion than in the upper and lower ends (26 to 40 MPa), but sufficient for constructing galleries
- Anisotropy of the thermal conductivity (λ_v/λ_h=1.3) is lower in the middle, more clayey portion (1.35 $Wm^{-1}K^{-1}$ horizontally) than in the upper and lower ends (1.65 $Wm^{-1}K^{-1}$ horizontally).

Hydraulic overpressure exists with respect to the carbonate formations (this phenomenon has been recognized but is still unexplained). Hence, the predominant mechanism of transfer within the clay formation is diffusion.

The permeability of the carbonate formations surrounding the argillite is low (not more than 10^{-8} m/s) because they have undergone intense recrystallization of diagenetic origin.

12.2.2. Designing and Analyzing Disposal Concepts

At the current stage of studies conducted in the framework of the Law of December 30, 1991, ANDRA has selected a series of designs ("preliminary concepts") enabling analysis of the different factors to be taken into account (safety, reversibility, modeling, and costs). Those preliminary concepts are based on the state of scientific and technical knowledge and data collected so far. They do not necessarily prefigure the technical solutions that may be implemented in response to the radioactive-waste-management choices to be made starting in 2006.

The design follows these general rules:

- To separately dispose each type of waste, facilitating safety analyses and models while providing flexibility during the repository construction
- To assign specific drifts for each main function (construction, ventilation, transportation)

Table 12.1. Main characteristics of the Meuse/Haute-Marne's Callovo Oxfordian formation

PROPERTIES	EXTREME VALUES	MEAN VALUE
Natural density	2.32 – 2.61	2.42
Water content (%)	2.8 – 8.7	6.7
Porosity (%)	9 – 18	14
Thermal Conductivity ($Wm^{-1}K^{-1}$)	1.35 – 1.65/ 1.8 - 2.0	1.47/1.9
Heat capacity ($Jm^{-3}K^{-1}$)	0.9×10^6 – 1.2×10^6	2.1×10^6
Coefficient Thermal Expansion (K^{-1})	0.8×10^{-5}- 6.2×10^{-5}	1.7×10^{-5}
Uniaxial Compressive Strength (MPa)	12 – 49	26
Uniaxial Tensile Strength (MPa)	0.9 – 5.4	2.6
Young's Modulus (GPa)	2.3 – 11.0	4.9
Poisson's ratio	0.17 – 0.4	0.3

- To accommodate reversibility as much as possible, using a step-by-step closure.

In the case of the argillaceous Meuse/Haute-Marne site, the general architecture selected is designed on a single level in the middle of the clay formation. In principle, the architecture is designed according to a “dead-end” concept, theoretically to limit connections between disposal modules and cells.

Different cell types have been investigated, taking into account the possible diversity of current and future waste and also the state of knowledge concerning the site (Figure 12.3). One of the primary objectives of the

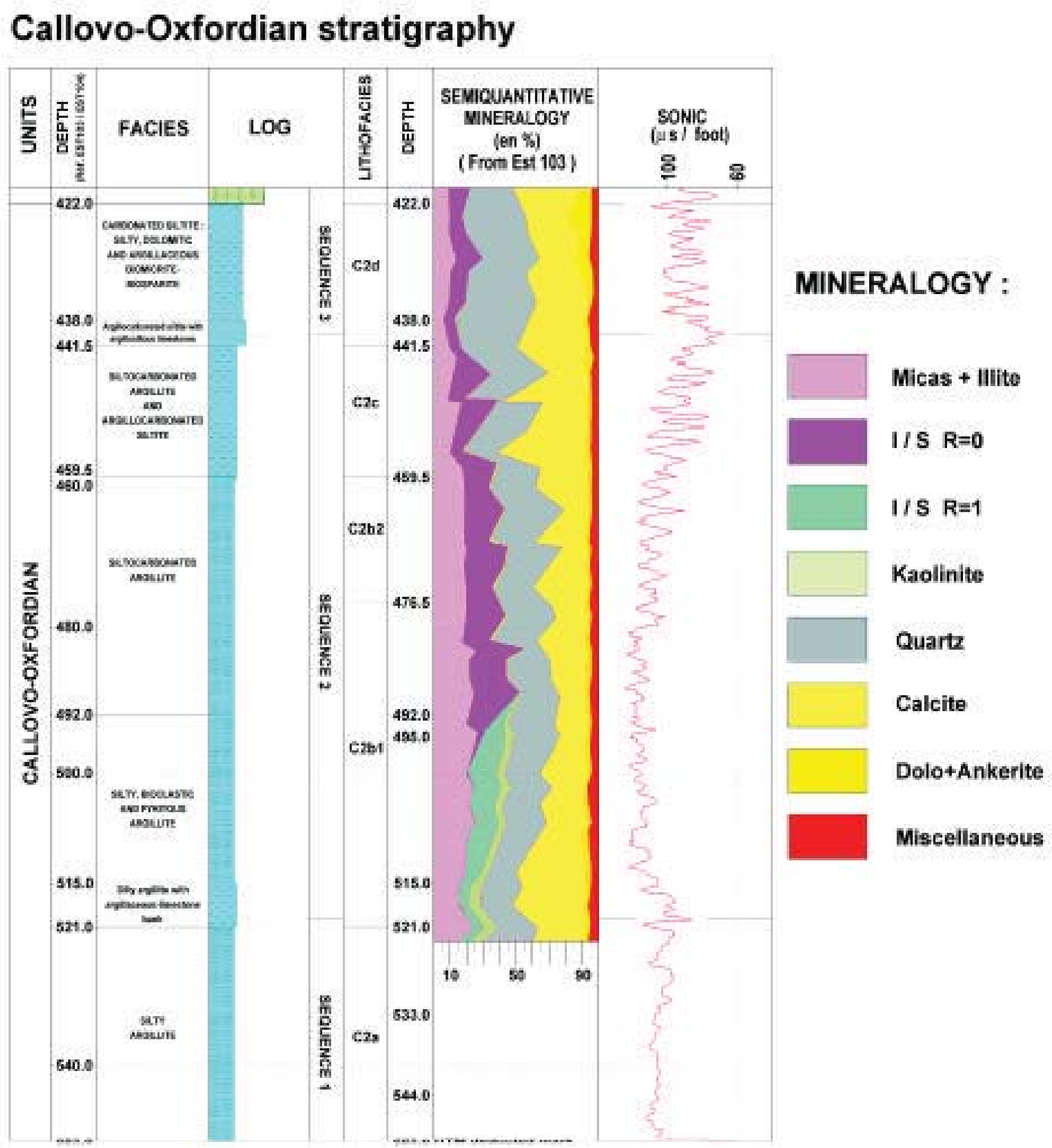

Figure 12.2. Stratigraphy of the Callovo-Oxfordian Formation

Meuse/Haute-Marne underground research laboratory is to eliminate current uncertainties concerning mechanical and thermal behavior of the clay formation.

With regard to heat-emitting waste, investigations have been undertaken on the basis of a maximum temperature criterion of 100–150°C at the contact points between packages and engineered barriers or with the geological environment. That criterion determines (in particular) the number of packages per cell and the distance between disposal structures.

An important part of this feasibility study is coping with the option of reversibility. Specifically, we wish to pursue the possibility of affecting the implementation and operation of the repository (in other words, to control or even modify its progression) as it is being built, without departing from the basic objective—protecting human beings and the environment.

Based on the preliminary concepts for the argillaceous Meuse/Haute-Marne site, ANDRA has carried out a Phenomenological Analysis of Repository Situations (PARS), with the purpose of identifying which phenomena characterize the different successive states of a repository and its geological environment. Analysis of the disposal system is structured according to the main components of the repository and its environment, as well as the various phases of the repository lifetime from the beginning of its construction (Figure 12.4). To be complete and continuous, the analysis must in fact include both the operational and post-closure periods of the repository, up to about one million years. This bound corresponds to a time frame for which a predictive geological evolution appears to be reasonable, based on the knowledge collected of the past and recent geological history of the site.

During the operational period of the repository, the successive operational phases determine in principle the different physical states of the repository. With regard to post-closure periods, as long as the geological and geodynamic context remains the same, the evolution of the repository is chiefly determined by the evolution of its own phenomenology. Eighty-three different situations have been defined. Each one has its own characteristic phenomenology and constitutes—as in a film—the successive specific images of repository life. At that stage of study, the analysis ignores, with justifying arguments, the transients between each situation.

The PARS constitutes the input data for the numerical simulations relating to the repository and for reversibility and monitoring program studies. It is also a reference

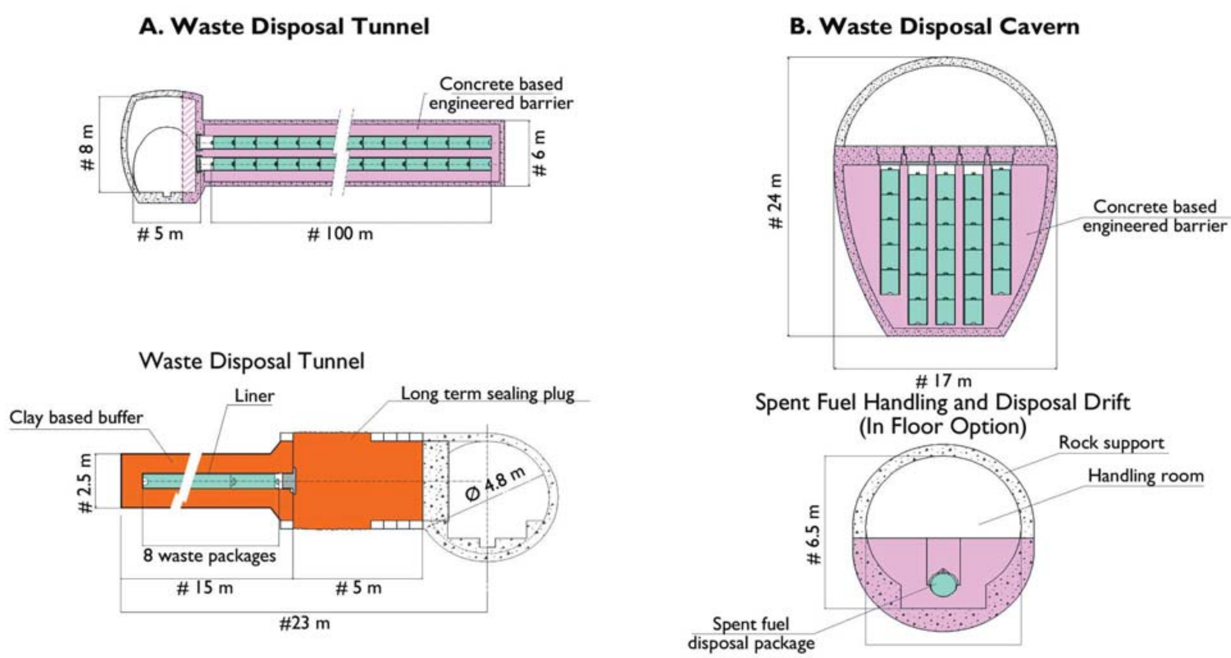

Figure 12.3. Design options for disposal cells

system for safety analyses. In addition, it enables us to identify which undetermined elements in the preliminary repository design limit phenomenological modeling. Similarly, scientific uncertainties (data, phenomenology) are specified and attributed to clearly defined fields. Those elements, together with the safety analysis, help to define the priorities of the research program (notably with regard to underground laboratories) and set the structure.

12.2.3. The Research Program of the URL

Objectives

The URL research program aims at providing reliable data for the performance assessment. At this stage of research, the program is limited to the geological medium and will attempt to answer the following questions:

- At what scale do the sedimentation cycles have a significant influence on the physicochemical properties of the formation?
- Is there a scale effect between the behavior determined on small samples and from drifts? Can the upscaling be formalized?
- Do discrete fractures exist as yet undetected by the seismic surveys, and if so, could they have a significant role in the transfer of radionuclides to the biosphere?
- Will the behavioral models of argillite (THM, HC) take into account the right mechanisms, and will their use in computer codes help quantify the effects of these mechanisms accurately or at least conservatively?

The research program includes the acquisition of sound scientific data and knowledge justifying the concept of repository reversibility. The collected information will determine the state of understanding concerning both the operation of a repository and its evolution over time, and answer two key questions:

- How can we ensure the possibility of safely getting back to the disposed wastes?
- How can we choose parameters for repository monitoring that will lead to clear criteria for further decisions?

Organization

The research program (for which the law of December 30, 1991, set a deadline of 2006) has been planned to account for the possible physical progress of the work

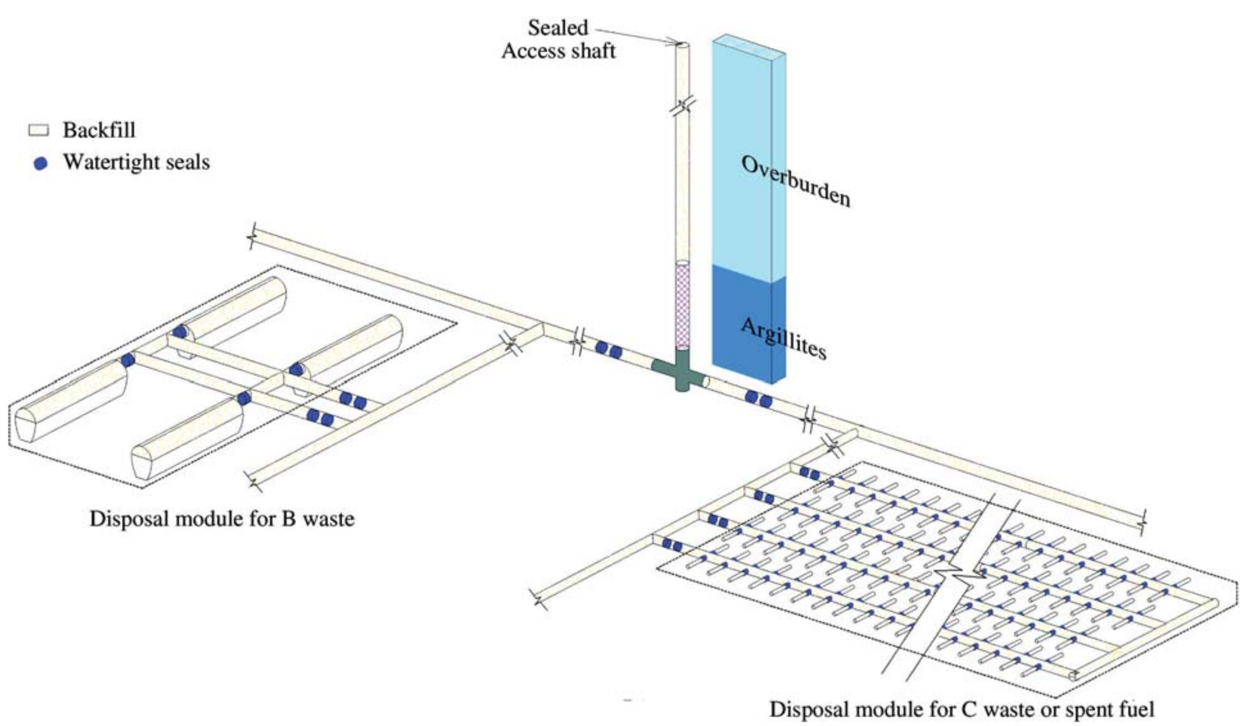

Figure 12.4. Scheme of a PARS "situation": a potential repository after sealing

on-site. The research projects are planned using the opportunities offered by different steps in the construction, but must in exchange adapt to the requirements of co-activity with the civil work. The construction of the underground laboratory (Figure 12.5) started in August 2000, and shaft sinking progresses according to schedule.

Before excavation of the underground installations, a reflection seismic survey in 3D was carried out on a 4 km area around the laboratory. This was done to obtain a detailed 3D geological model that can be compared with observations made in the different shafts and galleries of the underground laboratories. It confirmed the lack of faults within the Callovo-Oxfordian Formation and above.

During sinking of the shafts, the piezometric drawdown into the Oxfordian carbonates is monitored by special observation boreholes, and the dewatering flowrate is measured in the shafts. Extensive geological and geotechnical surveys are carried out at the same time.

The underground laboratory includes a series of experimental galleries to be excavated in the mid-plane of the formation (the most clayey) and two drifts (one upwards, one downwards) that allow characterization of other horizons (Figure 12.5).

The research program consists of a panel of experiments dealing with key questions. Thus, the organization into several experimental phases helps take account of new facts that could appear during the early experiments:

- Repetition of geological, geomechanical, and geochemical characterization measurements in the different structures of the underground laboratory
- Possibility of conducting experiments on containment at several points within the laboratory
- Splitting into at least two phases for the other experiments, located in dedicated galleries and aimed at loading the geological environment to a condition similar to that caused by excavation of galleries and the operation of the repository (mine-by-test type mechanical test, thermal test).

ANDRA has established teams at the laboratory site whose tasks are to prepare and control the experiments, guaranteeing fulfillment of the scientific objectives, the quality of the methods used, and the data obtained, as

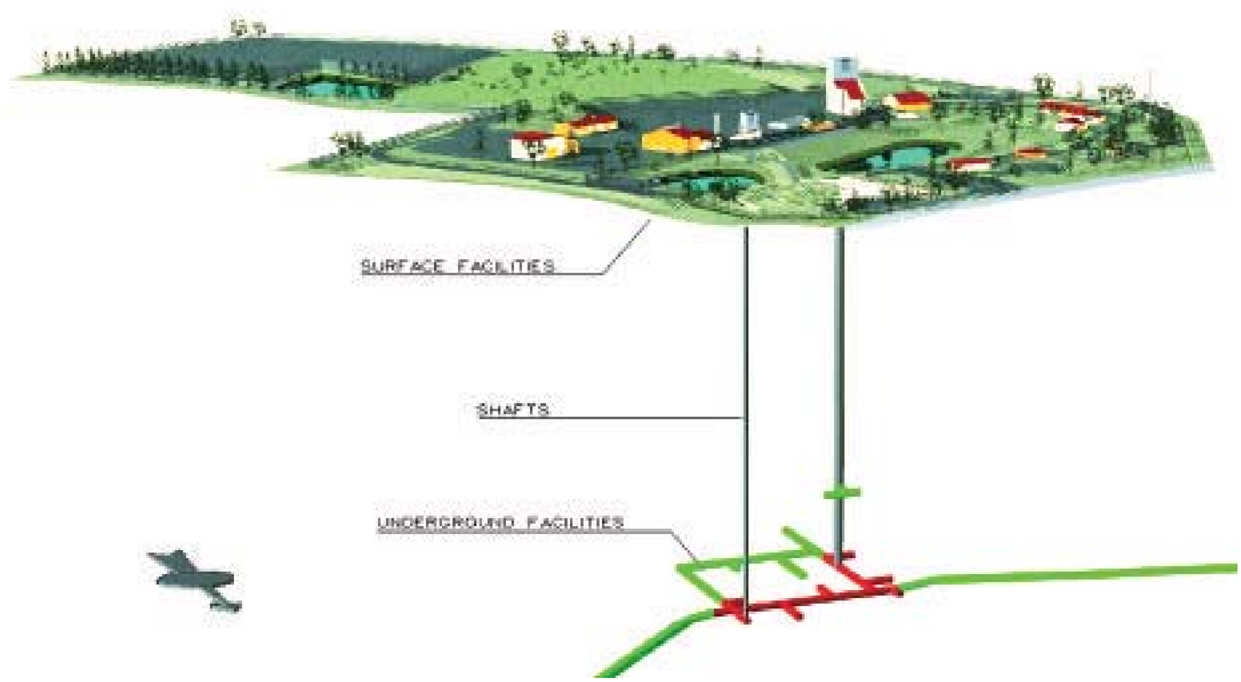

Figure 12.5. Meuse/Haute-Marne laboratory architecture project

well as guaranteeing safety during experimental operations. Each investigation will be monitored from start to finish by a group of international experts.

The execution of such a research program demands a major R&D investment. ANDRA has largely mobilized the French academic community and has also signed partnership agreements with major French research institutions (CNRS, BRGM, CEA). It also relies on international cooperation, particularly in connection with the 5th European RTD Framework Program, and anticipates participation in its international program from foreign teams experienced with investigations in underground laboratories.

Detailed Program

A series of samplings and measurements, as well as solute migration tests simulating radionuclide transfer, will be carried out from these galleries, on a statistical basis and at different scales.

The trickiest operations concern the sampling and chemical analysis of pore water, to avoid the effects of oxidation or bacteriological contamination. The low water content and permeability of the Callovo-Oxfordian Formation make the collection of representative pore-water samples very difficult. These operations have already been performed by ANDRAin the HADES laboratory (Mol, Belgium) within the Archimede experiment. They are also the subject of new procedures in the Mont Terri laboratory (Switzerland). where the characteristics of the argillites are quite similar to that of the Callovo-Oxfordian formation in Bure. Specific sampling, conditioning, and analysis techniques are being developed to minimize any potential changes to the natural chemical composition of the water and rock. A geomechanical model has been undertaken to estimate parameter values that cannot be directly measured, but that may be activated by known reactions and/or determined experimentally. An international task force for geochemical modeling is being formed to intensively interact with the experiment.

Another purpose of the program is to determine *in situ* the parameters governing convective flow in the host formation. A dedicated experiment will yield information to explain the origin of the overpressure developed within the formation and the difference between the permeability measurements. Hydraulic pulse tests will be carried out at intervals of 20 to 120 m in deep boreholes (from 10^{-13} to 10^{-11} m/s) and then compared to tests results obtained from cores (from 10^{-15} to 10^{-13} m/s), and even calculated on the basis of the pressure increase of the Wireless Data Transmission gauge sealed in the formation (mean value: 4×10^{-14} m/s). The underground laboratory will help to resolve various problems related to measurements made in deep boreholes or on the basis of cores collected in those boreholes, as follows:

- Verifying if osmotic pressures resulting from an imbalance between borehole fluid and the water of the formation can develop
- Allowing pressure conditions to stabilize before beginning *in situ* permeability tests, and then correctly measuring the dissipation and reconstitution times of pore-water pressures around the structures.

The experimental zone at level 490 is initially dedicated to support the mechanical-behavior models of the argillite. This will be done in three steps:

1. *Calibration of the models from measurements performed during the excavation of the access shafts*
   It is the first experiment to be conducted in the underground laboratory. It will be achieved by a vertical "mine-by test" consisting of observing, in real time, the excavation effects of the main shaft of the underground laboratory on the clay massif once the shaft has been equipped with instrumentation. A niche at the 445 level provides a location to install the required instruments before a zone is excavated at a depth varying between 460 and 473 m. This experiment also supports an international benchmark on modeling argillite geomechanical behavior.

2. *Borehole measurements of metric scale parameters*
   Two series of geomechanical parameters that were difficult to measure in previous deep boreholes (elastic moduli, creep, *in situ* stresses) are still to be obtained in the URL. Those consist of several random measurements related to intrinsic characteristics of the undisturbed massif. Those measurements will be made in different parts of the underground laboratory. The results will enable better design and interpretation of some subsequent experiments, for which predictive modeling will be necessary (e.g., the drift-opening experiment), and development of better measurements and observations made during

structural excavation (converging, scaling, breakouts, etc.).

3. *Comparison of the results between a mine-by test in a specific gallery and blind modeling of the exper - iment.*
   The study of the hydromechanical response of the argillite formation to drift opening is one of the main experiments to be conducted in the underground laboratory. It will take place on the main level of the laboratory and will consist of observing (in real time) drift-opening effects on the most clayey lithological unit of the Callovo-Oxfordian Formation. This experiment will complement the results obtained from the shaft experiment. Its horizontal geometry and its location at the center of the underground laboratory's experimental zone will allow for more complete and precise instrumentation.

A series of dedicated experiments will then be used to determine the extent of our understanding of the mechanisms that lead to disturbances in the near field:

1. Changes in the redox front under the influence of ventilation will be observed and monitored during and after the mine-by test in the drift (survey of geochemical and biological markers of the redox front, monitoring of the dehydration).
2. A specific experiment will study not only the phenomena occurring in argillites when subjected to temperature increase (thermal-mechanical damage on the transport and retention properties), but also the parameters obtained from cores (thermal properties of argillites and their evolution with temperature, thermal-hydraulic and thermal-mechanical coupling parameters, as well as effects on the mineralogy of argillites and the chemistry of the pore water). This *in situ* experiment will allow for conditions closer to reality, notably because of its scale and the minimal disturbance of the geological environment.

This URLis site-specific. One of the main challenges of the research program will be to extrapolate all of the results up to the typical scale of a repository. Therefore, the program also includes acquisition of geological data outside the perimeter of the URL. We have recently completed detailed studies of boreholes and a 2 km^2 3D-seismic survey. Another part of the program will be to perform a broad geological survey during excavation of the shaft and galleries to provide a lithological and petrophysical framework for the experimental results. Specifically, this survey will yield:

- Links between the petrofabric and properties
- Evolution of the petrofabric in space and time
- Representativeness of the measurements.

12.3. PROGRESS IN GRANITIC-SITE RESEARCH PROGRAM

Two main criticisms were expressed against this site:

- The existence of aquifers tapped in the overlying sedimentary layers
- The difficulty, judged serious, of accurately determining the layout and hydraulic connectivity of the network of transmissive fractures. (Experts agree that only an underground laboratory can provide the relevant information, but are reluctant to run the risk of a failure.)

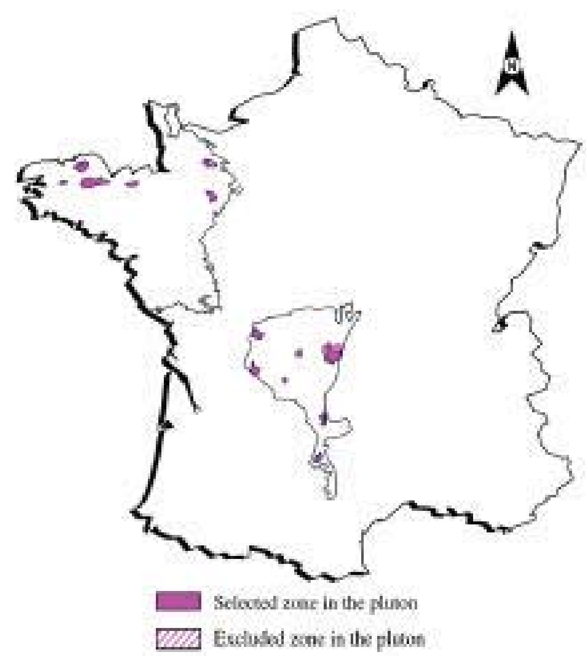

Figure 12.6. Map of the proposed granitic areas after the 1999 screening

In early 1999, at the government's request, ANDRA, with the help of BRGM, looked for a new granitic site capable of hosting an underground laboratory. This site screening was based on the 1983 national inventory of potentially favorable sites, incorporating recent knowledge and methodology advances in quantitative analysis of the granitoids (particularly their fracturation). The purpose was to answer the essential criteria set by Fundamental Safety Rule III-2f to guarantee the safety of a radioactive repository in a deep geological formation after the repository operation phase.

The selection criteria required:

- Geodynamically stable areas (e.g., western Hercynian batholiths)
- Outcropping batholiths without aquifers
- Surface up to 20 km^2, excluding the main neighboring faults
- Kalk- or sub-alkaline granitoids with self-sealing capability and no exceptional mineral resources.

A Geographical Information System was used to compile all the information. In September 1999, ANDRA submitted a report to the French government identifying 15 potential sites for a URL in a granitic area (Figure 12.6). The National Evaluation Board approved this list.

The siting project must be carried out with the approval of the elected officials and populations of the potentially favorable sites, under conditions to be set by decree as specified by the law of December 30, 1991.

After publication of the French government's report in July 2000, the "Granite Mission" was suspended, but since then, the government has reiterated several times its firm resolution to pursue research on reversible disposal at two different geological sites. For relevant elements to be available in 2006 to prepare a report on the feasibility of a repository at a granite site, ANDRA has designed an action plan based on its participation in foreign underground-laboratory programs and a review of alternative repository designs. The participation in foreign underground laboratories involves five experiments in progress:

(1) The Tunnel Sealing Experiment at the URL (Canada) is designed to provide a first assesment of the performance of seals in a gallery, tested at full scale.

(2) The TRUE experiment at the HRL(Sweden) provides the opportunity to test our ability to model ground water flows and solute transport in a fracture network at the decametric scale.

(3) The group of three experiments GAM, HPF and CRR at the Grimsel Test Site (Switzerland) will allow ANDRA to characterize mechanisms that disturb fluid transfer in a fracture network (e.g., gas or colloid transfer, alkaline effects).

The study of repository design relies on concepts, developed abroad, for granitic formations, mainly the KBS3 Swedish concept. Their applicability to French sites and waste inventory is being examined with respect to the characteristics of French granite massifs. Finally, the inventory of surface reconnaissance methods for outcropping granite sites is being compiled to prepare potential reconnaissance campaigns.

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Chapter 13

Nuclear Waste Disposal in Germany: Background, Status, and Future Research

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ABSTRACT. Since the early 1960s, the radioactive waste disposal policy in the Federal Republic of Germany has been based on the decision that all types of radioactive waste are to be disposed of in deep geological formations. According to the Atomic Energy Act, the siting, construction, and operation of the repository is a national task. Considering the favorable properties of rock salt, German geoscientists had (at an early stage) already recommended rock salt as an attractive host rock for the permanent isolation of nuclear waste. Therefore, the experimental emplacement of nuclear waste in rock salt started in Germany in the Asse mine in 1967, supported by many geoscientific *in situ* and laboratory studies. Since 1976, the former Konrad iron ore mine has been explored as a potential repository site for radioactive waste with negligible heat generation. The plan-approval procedure was started in 1982. The licensing procedure for the Konrad repository project was practically finished in 1992, but the license has not yet been granted. In 1979, the Gorleben salt dome was designated to be investigated as a repository for all kinds of radioactive waste. Selection of Gorleben in 1977 was mainly a product of geoscientific and economic criteria. Public involvement in a transparent site-selection procedure (the norm for site selection nowadays) was not included. Site investigations from the surface and underground exploration, as well as an extensive laboratory program, resulted in a comprehensive database that confirmed the potential suitability of the site. In addition, an abandoned salt mine at the Morsleben site, selected as a repository by the former German Democratic Republic Government, was used for the continued emplacement of low and intermediate nuclear waste after the reunification of the two German states. Operation of the Morsleben repository (ERAM) ceased in 1998. The plan-approval procedure is limited to the backfilling and sealing of the repository.

The German Federal Government has recently signed an agreement with national utility companies to end electricity generation by nuclear power. As for repository projects in Germany, the utility companies agreed to discontinue exploration at the Gorleben site. To examine further sites in Germany, a group of experts has been appointed by the Federal Minister for the Environment to develop a new comprehensive procedure for the selection of a repository site, built upon well-founded criteria and including public participation. The final recommendations of the committee are expected in 2002. In addition, further research has been initiated and other host-rock formations are to be taken into consideration. With this in mind, basic research on alternative host rocks is currently being carried out in international projects. German research institutes are participating in the Underground Research Labs of Grimsel, Äspö, and Mont Terri under the terms of bilateral agreements. A co-operative project with ANDRA, at the French URL Meuse/Haute Marne, has just started.

13.1. INTRODUCTION

On June 11, 2001, the German Federal Government signed an agreement with representatives of the national nuclear energy utility companies. The essential part of this so-called "consensus paper" was the decision to phase out the use of nuclear energy. In addition to the consequences for the energy policy, the entire German radioactive waste isolation strategy and especially the repository projects were affected by the agreement.

In the following, the current status of radioactive-waste-disposal-related investigations in Germany will be described, along with an outline of further activities in R&D programs.

13.2. HISTORY OF GERMAN NUCLEAR WASTE POLICY

In the Federal Republic of Germany, peaceful use of nuclear energy started with the operation of the first nuclear power plant in 1960. Since the early 1960s (i.e., from its very beginning), the German radioactive waste disposal policy has been based on the decision that all types of radioactive waste are to be disposed of in deep geological formations to (reasonably) assure the long-term and safe isolation of radioactive waste from the biosphere. Disposal of radioactive wastes is defined as maintenance-free and temporarily unlimited. Thus, retrievability is not considered within German radioactive waste disposal. Liquid and gaseous wastes are excluded from disposal in such a repository; only solid or solidified radioactive wastes are accepted.

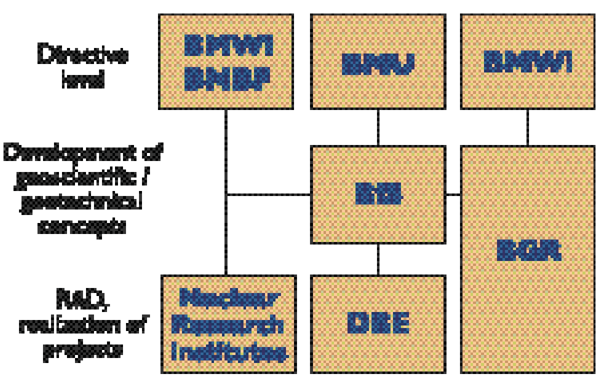

Figure 13.1. Authorities involved in nuclear waste disposal in Germany (explanations in the text)

The Atomic Energy Act of 1976 stipulated the reuse of residual substances and thus upheld the principle of reprocessing for spent-fuel elements. This was taken up in the German radioactive waste management and disposal concept, which originally included the construction of a reprocessing plant. In 1989, the national utilities renounced the construction of the planned German reprocessing plant and, instead, decided upon the reprocessing of spent nuclear fuel in the COGEMA and Brtish Nuclear Fuels Limited (BNFL) facilities in France and Great Britain, respectively.

Independently of this development, the emplacement of spent-fuel elements in a repository (the so-called "direct disposal" of spent fuel elements) has basically been developed to maturity. Both the feasibility of this technology and its licensibility have been demonstrated by research and development work. Thus, the Atomic Energy Act was amended in 1994, providing the legal basis for direct disposal of spent nuclear fuel.

13.3. INSTITUTIONAL FRAMEWORK

The Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU) is responsible for nuclear safety and radiation protection (Figure 13.1). Within this field, it has the competence to issue directions and to supervise the legality and expediency of the acts of authorities responsible for enforcing the Atomic Energy Act and the Radiation Protection Ordinance.

Apart from the BMU, the following federal ministries take part in radioactive waste management, according to their specific responsibilities:

- The Federal Ministry of Economics and Technology (BMWi)
- The Federal Ministry of Education and Research (BMBF)

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 (including establishing and operating the federal installations for its safekeeping and final disposal). It supports BMU technically and scientifically in these fields.

Under German nuclear and radiation protection law, the Federal States (Länder) execute administrative duties (licensing and supervision) not performed by the federal authorities. Thus, the Federal States are the competent licensing authorities for all nuclear installations concerning their territory, except interim storage facilities for spent nuclear fuel. They supervise all nuclear facilities except repositories. To ensure the uniform implementation of the Atomic Energy Act, the Federal States are subject to federal supervision by the BMU.

The Federal Institute for Geosciences and Natural Resources (BGR) advises the German Federal Government in all scientific work. Because geoscientific research in the field of nuclear waste disposal concerns the realization of repository projects, BGR also performs geoscientific research in this field. Another organization, the Company for the Construction and Operation of Waste Repositories (DBE), is assigned to plan, design, construct, and operate the repository. DBE, on behalf of BfS, has also explored the Gorleben site.

13.4. LEGISLATIVE FRAMEWORK

The disposal of radioactive waste in a German repository is specifically governed by the following specific acts and regulations (Brennecke et al., 1999)

- Atomgesetz (Atomic Energy Act)
- Strahlenschutzverordnung (Radiation Protection Ordinance)
- Bundesberggesetz (Federal Mining Act)
- Sicherheitskriterien für die Endlagerung radioaktiver Abfälle in einem Bergwerk (Safety Criteria for the Disposal of Radioactive Wastes in a Mine)
- International recommendations (e.g., IAEA Safety criteria) and regulations (e.g., The Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management of September 5, 1997).

The protection objective of radioactive waste disposal in a repository is prescribed by the Atomic Energy Act and the Radiation Protection Ordinance. The Federal Mining Act regulates all aspects concerning the operation of a disposal mine. The Safety Criteria specify the measures to be taken to ensure that this objective is reached.

By the Atomic Energy Act, radioactive waste disposal is assigned to the Federal Government as a sovereign task. BfS is responsible for establishing and operating those federal installations for radioactive waste disposal, acting on behalf of the Federal Government.

The required safety of a repository must be demonstrated by a site-specific safety assessment, which includes the respective geological situation, the technical concept of the repository (including its scheduled mode of operation), and the waste packages to be disposed. In the post-closure phase, the radionuclides that might reach the biosphere (via transport processes not completely excludable from scenario analyses) must not lead to individual dose rates that exceed the limiting values specified in Section 45 of the Radiation Protection Ordinance (0.3 mSv/a concept) The safety criteria permit some latitude of judgment. This process is predominantly determined by a site-specific safety assessment, within the scope of which the required safety of the repository must quantitatively be demonstrated (including the derivation of requirements for facility design as well as for the waste packages).

13.5. NATIONAL REPOSITORY PROJECTS

Konrad

In 1982, an application for the 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 Konrad mine is an abandoned iron ore mine and considered to be suitable for the emplacement of radioactive waste. The great depth of the iron ore formation (between 800 m and 1300 m), effectively sealed off from near-surface groundwater by the overburden of several-hundred-meters-thick claystone and marly formations, is the most important advantage of the Konrad mine. Thus, there is only limited availability of water as transportation medium for radionuclides into the biosphere.

The licensing documents, the so-called “Plan” Konrad, was submitted in 1990. In 1991 the legally required public participation took place, and the plan was publicly displayed. The public hearing (which lasted for 75 days) took place in 1992/1993. In September 1997, the licensing authority, the Ministry for the Environment of Lower Saxony (NMU), submitted a draft for a license. According to this draft, the fundamental licensability of the project was assured. In December 1997, the NMU submitted a second draft for the license and announced that the license would be granted early in 1998. Because of political reasons, the license has not been granted as yet.

Morsleben

In the former GDR, the abandoned Morsleben salt mine was selected as site for 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 geological formation of Zechstein/Thuringium (upper Permian). The salt structure consists of folded rock-salt, potash layers, and imbedded anhydrite blocks (Figure 13.2). The overall thickness of the salt formation is between 350 and 550 m. The 525 m deep Bartensleben shaft connects four floors on various levels between 386 and 596 m. In this former mine, many cavities exist with dimensions up to 100 m in length, 30 m in width and 30 m in height.

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. The plan-approval procedure is limited to backfilling and sealing the repository.

Gorleben

The Gorleben salt dome has been investigated since 1979 as the potential repository site for all types of radioactive waste. The underground exploration started in 1986 with the sinking of two shafts, followed by the construction of the underground infrastructural area. The BGR and BfS developed a comprehensive geological and geotechnical exploration program for site characterization.

In addition to the geological exploration, *in situ* experiments have been performed for the geotechnical characterization of the potential host rock. The results of both the *in situ* and laboratory tests provided the database to separate homogeneous areas and to perform numerical calculations. The results of the geological investigations so far show a large area of potentially suitable host rock in a simple anticline structure (Figure 13.3). To date, there exists no evidence that the Gorleben salt dome is unsuitable for a repository (Bornemann and Bräuer, 1999). But the exploration of the Gorleben salt dome was stopped in 2000.

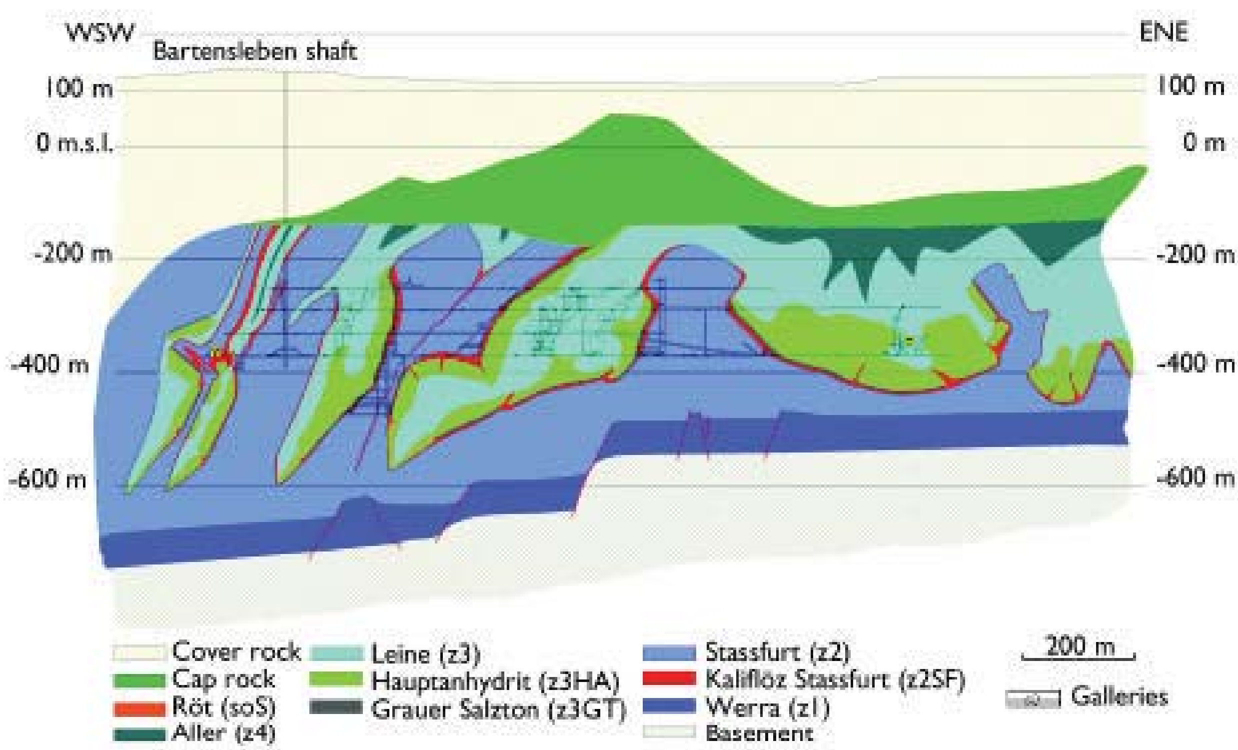

Figure 13.2. Cross section of the Morsleben salt structur e

13.6. NEW DEVELOPMENTS IN THE GERMAN WASTE MANAGEMENT POLICY

According to the phase-out policy of the German government, the entire waste-management strategy had to be reviewed. The first step was to sign an agreement between the Federal Government and the utility companies. The key points of the agreement with respect to radioactive-waste storage and disposal are as follows (Vereinbarung, 2000):

- New interim storage facilities will be built at reactor sites to minimize transport to the existing central interim storage facilities at Ahaus and Gorleben.
- Reprocessing will end by mid-2005.
- Exploration of the Gorleben salt dome as a possible repository site shall be interrupted for at least 3 and at most 10 years to clarify design and safety-related issues. The moratorium started on October 1, 2000.
- The competent authorities shall conclude the plan-approval procedures for Schacht Konrad in accordance with legal provisions. The applicant withdraws the application for the plan-approval decision's immediate enforceability, to allow a legal review in a main-action proceedings.

In the context of the new energy policy, the Federal Government has initiated the development of a new disposal concept with the political objective of constructing one simple repository for all types of radioactive waste around the year 2030 (Nies, 2000). Included in this initiative is the final decision on both the existing disposal projects and on new potential sites in different host rock.

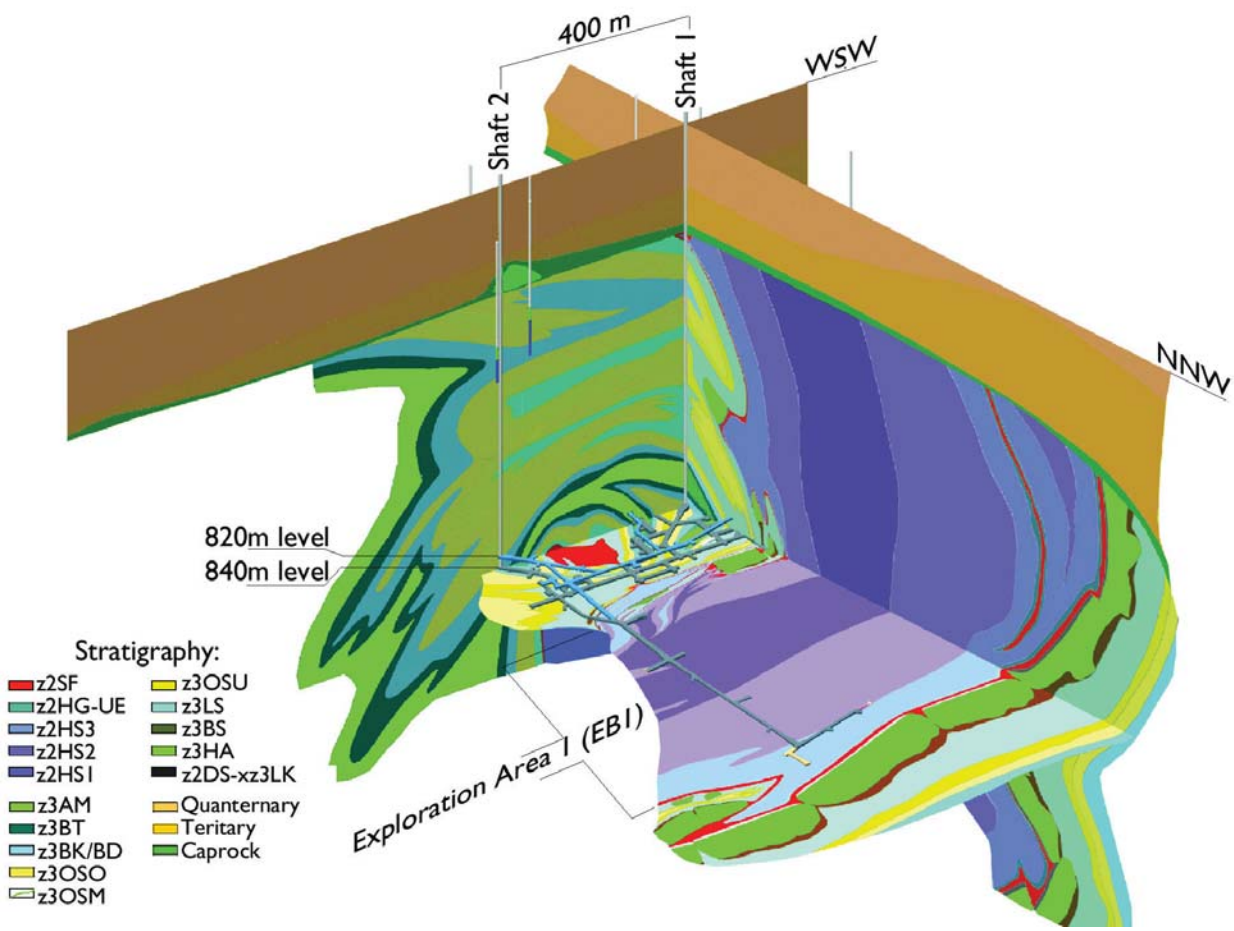

Figure 13.3. Three-dimensional view of the Gorleben salt dome, with shafts and exploration drifts

13.6.1. The Committee on a Selection Procedure for Disposal Sites

The Federal Ministry for Nature Conservation and the Environment has appointed a group of experts to develop a comprehensive procedure for the selection of sites that are both suitable for safe disposal and acceptable to the public. The procedure shall be based on well-founded criteria, with a clear and transparent structure. In contrast to previous site-selection procedures, public participation shall be an indispensable part from the very beginning (AKEND, 2000).

The committee on a selection procedure for disposal sites started its work in 1999. This committee has arrived at some general decisions, illustrated by the following examples:

- The "Concentrate and Contain" principle, rather than the "Disperse and Dilute" principle, is the basis for the committee's work.
- Only disposal in deep geological formations, i.e., at last several hundred meters below ground, is to be considered.
- The disposal facility shall be constructed as a mine, in accordance with the state of the art, with cost considerations to be taken into account.
- The isolation time frame should be on the order of one million years.
- A robust multibarrier system in a favorable geological setting determines the relative importance of geological, geotechnical, and technical barriers.

Including active public involvement, the committee distinguishes three phases for the site-selection procedure, from development to implementation:

- Phase 1: Development of a site-selection procedure and corresponding criteria.
- Phase 2: Political/legal obligatory establishment of a site-selection procedure.
- Phase 3: Implementation of the site-selection procedure.

Final recommendations of the committee are expected in 2002.

13.6.2. R&D Programs

In addition to the investigations of rock salt as a suitable candidate host rock for an HLW repository, German research organizations have (for many years) also conducted generic R&D in alternative geological formations like crystalline and argillaceous rocks. The motivation for this work was to:

- Have an alternative solution at hand
- Gain a better understanding about the positive and negative aspects of selected candidate formations
- Strengthen the international cooperation and information exchange.

In the past, several bilateral agreements (e.g., with Switzerland in 1982, with France in 1991, and with Sweden in 1995) have been signed. In the framework of these agreements, research is mainly performed in specific underground research laboratories (URLs).

Investigations related to crystalline rock are currently being conducted in the Swiss Grimsel Test Site (GTS) and in the Swedish Hard Rock Laboratory (HRL), Äspö. Investigations of argillaceous formations are being carried out in the Belgian HADES Underground Research Facility in Mol, in the Mont Terri-URL in Switzerland, in France at the Tournemire site, and (starting this year) in the planned URL in the Meuse/Haute-Marne district at Bure.

In Germany, R&D on argillaceous rocks started relatively late. Topics addressed were mainly related to the characterization of the host-rock formation. Examples for such R&D are the investigation of gas release and migration in the Boom Clay (Mol) in Belgium, two-phase flow and diffusive transport in geotechnical clay barriers and clay rock, investigation of the EDZ, and stress measurements and gas- and water-release investigations at Mont Terri.

German research institutions were invited to participate in the experimental R&D program of ANDRA, the French waste management organization, at the new URL being built at Bure in the Meuse/Haute-Marne district. Issues addressed are the study of thermal-hydraulic-mechanical properties of clay, the characterization of this specific clay material, and participation in the design, execution, and evaluation of the heater and ventilation tests.

Under the terms of bilateral agreements, the closest collaboration between German scientists and scientists from other countries exists in the Swiss GTS and the Swedish

HRL Äspö on the following issues: (1) Grimsel Test Site (relatively dry rock)—colloid and radionuclide migration, gas migration in the engineered barrier system and the interface to the geosphere, determination of effective parameters, and the FEBEX II-project, a Spanish-coordinated 1:1-scale project to test the emplacement concept for HLW; and (2) HRLÄspö (saturated fractured rock)—groundwater flow, radionuclide transport, geochemistry, development and testing of instrumentation and methods for underground rock characterization, development of numerical models to predict flow and transport processes in engineered and natural barriers, and the Prototype Repository project. Both programs also offer the opportunity to collaborate with research groups from non-European countries (e.g., Japan, Taiwan, and the U.S.).

In 1998, a scientific agreement between US-DOE and BMBF was renewed. On the basis of this agreement, cooperation between German institutions and the DOE/Carlsbad Area Office (CAO) will be intensified. It will focus on several issues of mutual interest, such as geochemistry, flow and transport phenomena, rock mechanics, disposal-room processes, and seals and monitoring. The main goal is to utilize and exchange knowledge related to implementation and operating repositories in rock salt, particularly with regard to the Waste Isolation Pilot Plant Site.

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Chapter 14

Geological Disposal as the Preferred Option in the Hungarian Radioactive Waste Management Program

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14.1. INTRODUCTION

Nuclear power plants produce radioactive wastes as unavoidable by-products of electricity generation. Hungary has neither significant fossil-fuel deposits nor renewable energy supplies, and the contribution of nuclear power is crucial to the national economy. Currently, almost half of the electricity generated in Hungary is produced at the four units of the Paks Nuclear Power Plant (NPP). The electricity is being generated in the national interest. Unfortunately, one cannot expect to enjoy the benefits of nuclear power without also having to cope with some of its problems. Perhaps the most significant of them is the disposal of radioactive waste.

In Hungary, the main source of radioactive waste is the Paks NPP, which operates four reactors, each rated at 460 MW. From the time the first unit started operation at the end of 1982 to the end of 2000, the Paks NPPhas generated approximately 200 billion kWh of electricity. Over the projected life of 30 years, the plant will produce 20,000 m^3 of waste, and nearly the same volume is planned to be produced by decommissioning the plant. This means that about 40,000 m^3 of low- and intermediate-level radioactive waste (LILW) of NPP origin must be disposed of in the country. In addition, the power plant will generate about 100 m^3 of high-level radioactive waste (HLW) over its operating life, and decommissioning will add some 3,700 m^3 more of HLW. On average, Paks NPP replaces 400 spent-fuel assemblies annually, which is equal to approximately 46.5 tons of heavy metal waste. The number of spent-fuel assemblies that will be generated by the end of the anticipated lifetime of the power plant will amount to 11,100 assemblies, excluding the quantity already shipped back to Russia.

The development of methods for long-term management of radioactive waste is a necessity in all countries with nuclear programs. The scale of this problem, in terms of volume, radioactive content, and diversity of physical and chemical forms of the waste, depends on the size of the country's nuclear program. In Hungary, one important component of the problem is the waste that already exists. A second important component is "committed" waste, which is the waste that is bound to accumulate from the operation or decommissioning of a plant. This legacy of waste—existing and committed—is very much greater than any current projections of wastes from any future nuclear program.

In Hungary, the Public Agency for Radioactive Waste Management (PURAM) has been designated to carry out the multilevel tasks in the field of radioactive waste management. As a result of concurrent changes in infrastructure, Hungary is about to make significant strategic and technical decisions. There are five technical priorities for the coming years:

1. Improving the existing LILW repository
2. Constructing a new repository for LILW
3. Expanding the interim storage capacity for spent fuel
4. Setting up a revised back-end policy, taking into account the new circumstances
5. Preparing for decommissioning of the nuclear facilities.

It is accepted in a number of countries that adequate protection of public health and safety against the hazards associated with short-lived LILW can be achieved with

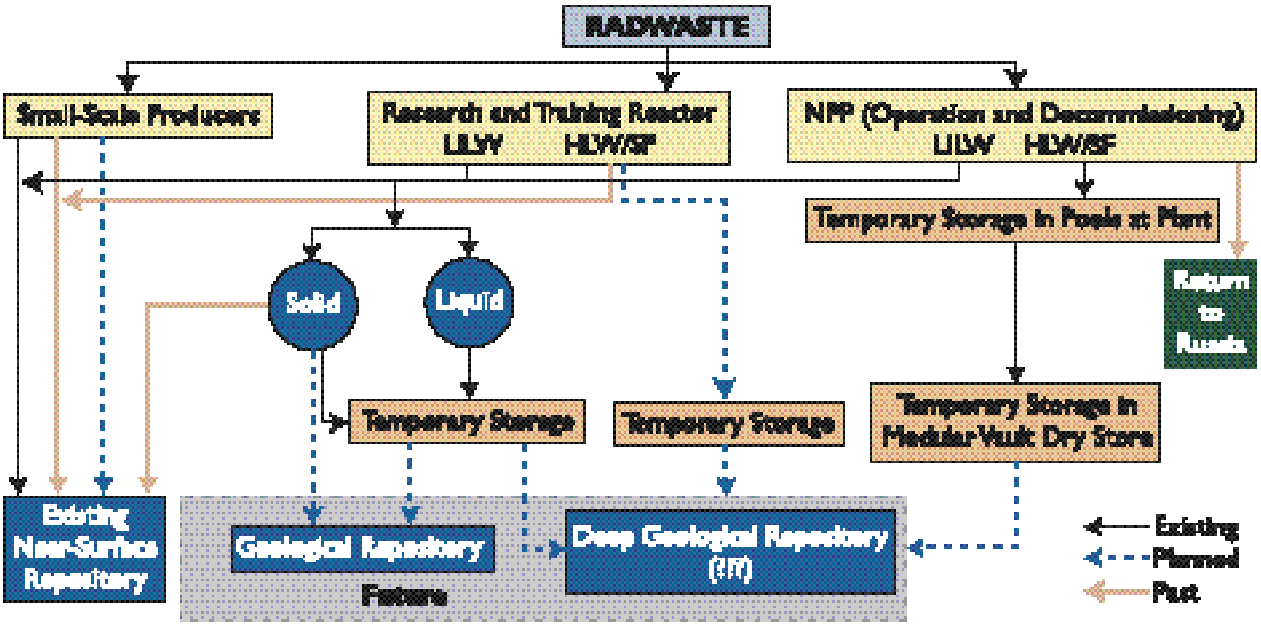

Figure 14.1. Radioactive waste disposal routes in Hungary

engineered facilities on the surface. Placing such waste in a well-chosen and well-engineered underground repository provides additional protection from surface hazards, both man-made and natural (e.g., the avoidance of extreme weather conditions). Underground structures are also inherently less vulnerable to seismic events.

Underground repositories are designed to provide the necessary long-term safety without surveillance measures, as opposed to near-surface facilities for which institutional control over a few hundred years is usually envisaged. Consequently, a well-engineered underground repository can provide a greater measure of public protection and safety than a comparable surface facility. Existing and planned radioactive disposal routes in Hungary are illustrated in Figure 14.1.

14.2. SITING OF AN LILW REPOSITORY

Within Hungary, the two granite massifs (Velence and Mórágy Granites), the West Mecsek Argillite, and the Tertiary clays and marls are the most suitable for radioactive waste disposal, because their volume is sufficient and their hydraulic conductivity is low. The clay and marl are frequently underlain by karstic rocks; they often occur in protected areas or areas used for recreation, and earthquake hypocentres are concentrated at depth below the surface. The area of the West Mecsek Argillite and most of the area of Tertiary clays and marls have been reserved as regions for further exploration for HLW disposal.

14.2.1. Selection of a Rock Mass and Area

Since 1993, as part of the framework of a National Project, the Geological Institute of Hungary and many other institutions have performed site explorations for a new LILW repository. Near-surface and underground disposal was evaluated through a four-stage investigation. First, the whole of Hungary and, second, a selected area within Hungary were investigated by desktop studies. In the initial phase of site exploration, using exclusionary criteria, all areas were ruled out that must be protected for political, economical, geological, etc. considerations or where the disposal site has to be protected from industrial or natural influences. Figure 14.2. provides examples of this approach.

The next phase was the positive screening, in which geological prospects were evaluated from a suitability point of view, which means 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 worth doing more research related to near-surface, and about 23,000 km^2 related to subsurface, disposal, as shown in Figure 14.3.

A suitable site could be expected in areas where the

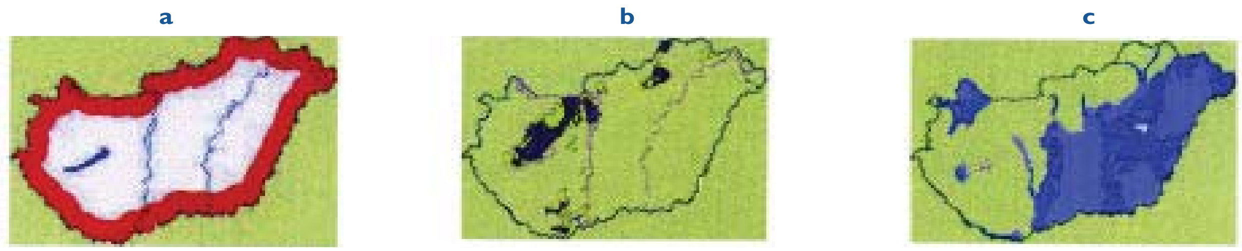

Figure 14.2. Negative screening concerning the whole country:(a) Excluded 30 km area from the border, (b) Excluded karstic areas, (c) Excluded danger of inundation area

number and density of potential prospects proved to be high. Using this approach, an area of 5,000 km^2 designated in Figure 14.2. was selected for further exploration. Numerous potential prospects were identified: 128 for near-surface and 193 for subsurface disposal, as illustrated in Figure 14.3. and Figure 14.4, respectively.

Public approval was given to just a few dozen out of several hundred potential prospects. 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) and one underground (granitic) sites. The granite site proved to be more suitable when compared with the loess site. Based on the first series of investigations, a granite prospect of Lower Carboniferous age in the village of Bátaapáti (Üveghuta) in southwest Hungary was selected as a potential site for an underground repository. See Figure 14.5.

The area is a hilly region with elevation up to 300 m above sea level and with valleys 100 m deep. The groundwater table is 0–1 m below the valley bottoms and 50–80 m below the hilltops. The deepest valleys are not bottomed in sediments that cover granite except in the uppermost, weathered part of the granite. The topography of the groundwater table follows the surface relief.

The investigations summarized above ended the first phase of the National Project, and their results form the basis for designing studies in the second phase.

14.2.2. Site Selection and Exploration

The second phase of the National Project consisted of three stages: selection of a site in the granite massif, assessment of its geologic suitability, and characterization of the selected site. The exploration was planned in three phases: (1) site selection, (2) assessment of site suitability, and (3) an integrated evaluation of the site and its surroundings. This could be followed by the licensing process. These three phases correspond to normal international practice.

In the preliminary concept of the repository, it was assumed that it would be a system of parallel galleries at one elevation. For the purpose of site delineation, the shape of the gallery system was set as a 300 × 600 m rectangle. There were two main options for the exploration:

1. The whole area would remain acceptable until the exclusionary analysis has been completed, and a 300 × 600 m rectangle was chosen for further exploration.
2. Preliminary hydrodynamic modeling would determine which topographical situation was most favorable for the site, when all orientations of the 300 × 600 m rectangle satisfying this criteria are fixed. Then the exploration is concentrated on the potential sites defined in this way. The site selected in this manner should potentially be the most suitable one. It was believed that the second option required much less work and expense; hence, it was accepted. Preliminary hydrodynamic modeling determined that the most favorable conditions could be expected under big hills. Five hills large enough for a 300 × 600 m site were found within the study area.

At the Üveghuta Site, six boreholes were drilled between 1997 and 1999 (four to 300, one to 382, and one to 500 m). In a surrounding area within 5 km, 24 shallow (13–83 m) boreholes were drilled. All boreholes were cored, walls were scanned using geophysical methods, and hydrogeological tests were performed. Core samples of the granite were examined for composition, structure, and fracturing. The spatial orientation of fractures, fissures, and veins were precisely defined. This information is needed in investigations of water migration. The geophysical logs provided data on densi-

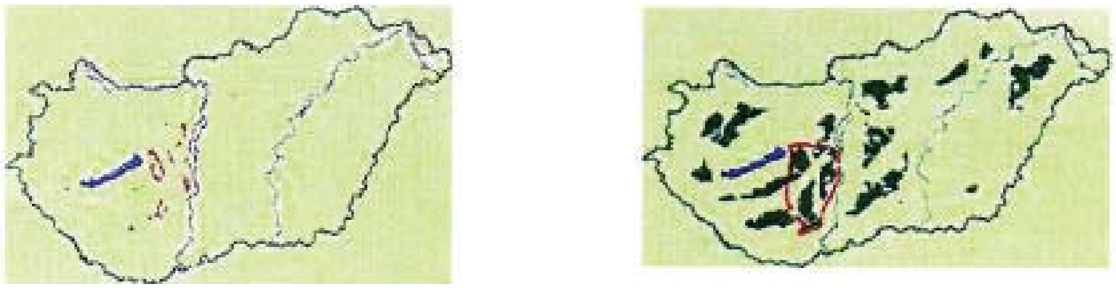

Figure 14.3. Perspective areas for near-surface and subsurface disposal.

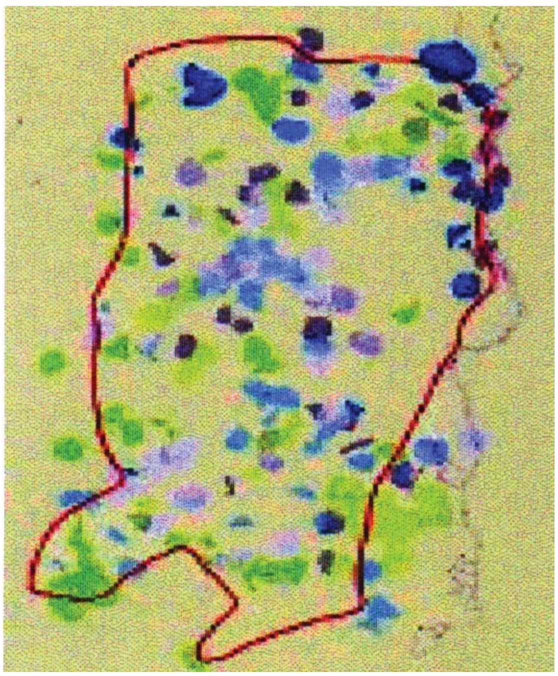

Figure 14.4. Suitable areas for subsurface disposal

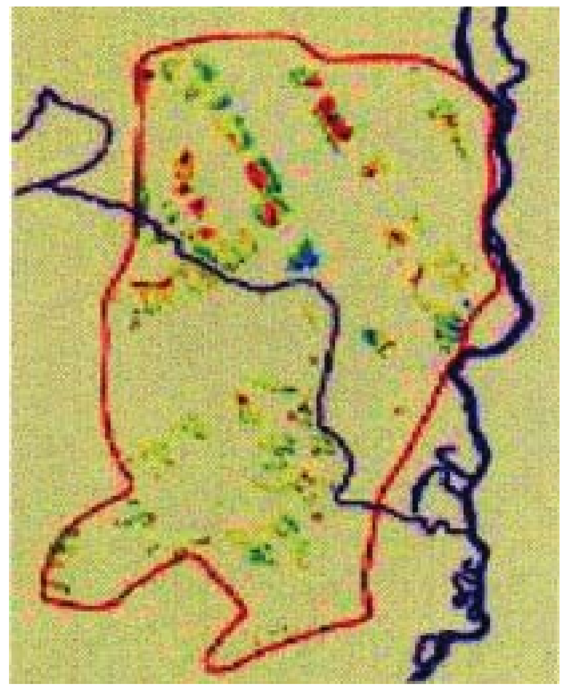

Figure 14.5. Suitable areas for near-surface disposal

ty, porosity, resistivity, and mechanical properties.

In the course of hydrogeological testing, hydraulic conductivity and yield of selected borehole intervals were determined. In this investigation, the boreholes were subdivided into intervals of 10–20 m. Individual studies of these intervals were made to determine changes in water properties as a function of depth. Isotopic compositions measured on some samples allowed us to estimate groundwater ages in the deeper horizons that are old, indicating very slow migration.

The rock mass between boreholes was investigated by seismic and geoelectric methods. Results produced a detailed picture of the granite surface, which has not been displaced by any faults or fractures. Two interference pump tests were also carried out in the middle of the site by pumping at a constant rate from two distinct intervals in one borehole for 8 and 9 days, respectively. The effects were measured at over 25 intervals in five surrounding observation boreholes. In addition, heat-pulse flow testing was carried out in all boreholes by recording points of inflow along the borehole axis.

The next phase of the program was site characterization. The general picture of the Üveghuta Site can be outlined as follows. The thickness of the sedimentary cover reaches 62 m and contains three main units. The upper unit is loess (maximum 52 m), which is underlain by red and variegated clay (1–5 m); and the lower unit is represented by granite debris (rubble) and sand (1–11 m). Loess started accumulating about 700,000 years ago, and the red to variegated clay about two and a half million years ago. Below the sediments, granite is weathered to a surprising thickness (21–76 m), which is interpreted as ancient (more than 2.5 million years old) superficial weathering. The groundwater table is 50–75 m below ground level, usually in the uppermost (4–9 m) disintegrated zone of weathered granite, where the hydraulic conductivity is a thousand to a hundred thousand times higher than fresh granite. This is one of the most important aquifers of the area, and flow in the uppermost level is mainly directed laterally. Distribution of water levels in the boreholes indicates that, at depth, the water flow is directed downwards within the granite body or at least has a significant downward-directed component.

The Üveghuta granite is much more fractured than many foreign granites. The hydraulic conductivity of unfractured granite between fracture zones (5.8×10^{-10} m/s in average) is very low, in the same range as Finnish granite. The water conductivity in the fracture zones is also surprisingly low (5.5×10^{-9} m/s), only ten times more than that of the solid granite between fracture zones. Thus, from a hydrogeological point of view, this granite is similar to Finnish granite despite the significant difference in age and geology. Overall, it has a suitably low hydraulic conductivity.

The low hydraulic conductivity of the fracture zones can be explained by the fact that the rock debris in them is cemented by clay, and the fissures are filled with clay minerals. The principal clay mineral is montmorillonite, from bentonite. This is very important for the repository, because any released radioactive pollution will be strongly absorbed in the montmorillonite.

Groundwater flow rates into boreholes are generally less than 0.7 l/min, and the inflow can be related to mapped fissures. The highest inflow measured by heat-pulse flowmetry was 6.5 l/min, and this was associated with an isolated fissure not part of a large fracture zone. During one interference test, a 20 m interval (which included this fissure) was pumped with an inflow of only 9.2 l/min. This required a significant drawdown in the pumping well. As a whole, the Üveghuta granite including the fracture zones has a very low permeability, and the mobility of water in it is very limited. The estimated radiometric age of groundwater in this formation varies between 4,000 and 20,000 years. This confirms that the migration of water is very slow (much of the precipitation was during the last ice age). Mean groundwater velocity is estimated to be only 10 cm/year.

The results of interference tests indicate that conductive hydrogeological features are present on the site, with an ENE–WSW orientation over distances of 152–255 m. However, no perpendicular connections were observed with these features at distances up to 86 m to the north. An analysis of the results indicates that the domain of connectedness in these features is almost vertical, dipping at more than 80º. Other than this feature of uncertainty, at least 100 m thick in the vertical direction, there are two big fracture zones at the site. Both are isolated from this special zone of hydrogeological connections by insulating sheets. The latter have probably originated from the filling of fissures close to the fracture zones by clay minerals, much more intensely than is usual. These insulating sheets should result in much less danger, for the safety of the repository, from the fracture zones than could be concluded on the basis of fracture intensity only. Note also that these sheets are not connected and are separated by impermeable zones of good quality rock.

In addition to characterizing the site, we must consider the future stability of the surroundings. For the 600-year period prescribed by law, climatic changes, uplift, and erosion obviously do not influence the safety of the planned repository at a depth of 200–250 m.

As far as earthquakes are concerned, the Üveghuta area is located in one of the lowest risk regions of Hungary. Furthermore, the effect from earthquakes at depth in hard massive rocks is always much less than on the surface, above a thick section of sediments. Consequently, the effect of earthquakes at the Üveghuta Site should be small and is negligible from the point of view of site suitability. A quantitative evaluation of the seismic risk will be calculated in the future.

It is legally required that faults should not exist at a site with evidence of surface displacements within the last hundred thousand years. Hence, specific attention was paid to this question. Sediments that have accumulated at the site during the last two and a half million years have been mapped in detail, and no faulting of these sediments has been identified. Thus, not only has no faulting occurred during the last hundred thousand years, it hasn't occurred over even a much longer portion of the last two and a half million years. On the basis of borehole evidence, this conclusion can be extended to distances of 3-4 km around the site. Fracture-infill minerals are about 100 million years old, and no evidence exists of an episode of fracture mineralization in the recent geological past.

A comprehensive image of the Üveghuta Site was developed on the basis of the interference testing, heat-pulse flowmetry, and an integrated interpretation of the hydrogeochemical data. In other aspects, only two new boreholes were interpreted, and the results were compared with those from earlier boreholes at a very general level. It was shown that the new results confirm the positive conclusion on site suitability, but it was clear that additional exploration will still be needed to achieve the level of characterization required. Four of the six boreholes on the site have been used to characterise a 461 m section, with a north-south orientation, along the hill ridge. The section is open at both ends, so no possibility for contouring any rock mass for the repository exists. In the perpendicular direction, two of the boreholes are out of the section plane by about 100 m, which obviously means they are insufficient for spatial characteristics. The six boreholes are enough for a positive conclusion on the suitability of the site as a whole, but they do not allow anyone to select and contour a distinct rock volume for the repository.

14.2.3. Principal Properties of the Üveghut a Site

Over the entire Üveghuta Site, granite appears only in the valley and slope bottoms. On the hills, the granite is buried by Pliocene to Quaternary sediments up to 50–60 m thick. It was essential to know the thickness of the sedimentary overburden and its internal structure. Granite was drilled in three of the five potential sites. Refraction seismic and geoelectric surveys were performed along the hill ridges to detect and map the granite surface and the sediment stratification. Correlation of borehole logs and geophysical profiles revealed that the granite surface is very even and the sediments are rather continuous. On this basis, a new map of the granite topography has been constructed that clearly displays steep valleys cutting into a smooth surface. The groundwater table, at depths of 50–70 m, is mostly in weathered granite, and the sedimentary overburden usually falls within the three-phase unsaturated zone.

In the subsurface flow system, the hilltops form recharge areas. Of course, any investigation of the whole system must extend over the discharge zones and cover the valleys as well. That is why valleys between, and in the neighborhood of, the potential sites have been studied with shallow boreholes and geophysical surveys. The latter included all the methods used on the hilltops, together with additional methods to detect zones of ascending water. Although these profiles displayed a certain heterogeneity, distinct zones of rising water could not be detected. Altogether 24 shallow boreholes were drilled, three of them on hilltops, five of them on slopes in a transitional position, and the rest in valleys at known or assumed discharge points. At four locations, two boreholes were drilled, one to 50 m and the other to 14–20 m. Head differences in these wells directly confirmed an upward flow of groundwater.

Seismic stability can be evaluated on the basis of earthquakes. In Hungary, there are enough data on earthquakes, from the 18th century onwards, to cover a time span almost half of what is needed for repository purposes. During that 250-year period, the Üveghuta region was one of the quietest seismic areas in Hungary. The strongest earthquakes in Hungary, with no exceptions, occurred in areas where before- and aftershocks were much more frequent than elsewhere. Therefore, this general distribution can be regarded as the “norm.” Thus, it can be assumed that no significant earthquakes will happen in the Üveghuta region for the next 600 years. It is also important to recognize that the strongest earthquakes in Hungary occurred in areas where the thickness of soft sediments was considerable, and the groundwater table was at shallow depths. Both factors increase the effects from earthquakes, to the extent that those effects are observed on the ground surface. The Üveghuta repository will probably be 200 m below the ground surface, in hard rock, deep below the groundwater table. It can therefore be concluded that even the strongest earthquake recorded in Hungary would not seriously damage the repository.

A separate question concerns the kind of damage that would be expected at the Üveghuta Site in connection with earthquakes. Two kinds of damage are possible in our opinion: damage to engineered barriers and changes in the groundwater-flow system. Two types of damage to the engineered barriers must be taken into account; one of them results from shaking and the other from shear in connection with a fault displacement. Protection against shaking is simple; the predicted acceleration can be considered in the repository design. This is not a suitability problem. The effects of shear connected with a fault displacement probably cannot be eliminated. In Hungary, however, fissure opening was only observed within 1 km inside the epicenter areas, and only on the ground surface. As was discussed above, the location of the Üveghuta Site near the epicenter of a strong earthquake is highly improbable.

Changes in the groundwater-flow system can be estimated from the evaluation of the data gathered in the past. There are two independent data sets. First, at the repository level, several isotope age dates for water range between 10 and 20 thousands years. This indicates the absence of any significant hydrogeological changes over this time span that could destroy the groundwater-flow regime. Secondly, the K/Ar ages for samples from argillaceous veins are Cretaceous, demonstrating high stability. The ^{14}C ages for the calcite veins, without exception, exceed 45,000 years (detection limit), which can be interpreted in the same way. Consequently, there are no indications of any hydrogeological readjustments within the last 50,000 years. This forms the basis for the conclusion that significant changes cannot be expected during the next 600 years.

The legal requirement concerning tectonic stability is that, within the last 100,000 years, no fault with a displacement on the ground surface shall be present on the site. The sedimentary overburden (variegated clay and loess) has been subdivided in detail, and all the horizons have been traced in borehole columns over the whole study area (Figure 14.2). It has been clearly shown that no interruption in the entire sedimentary overburden exists within 3-4 km from the Üveghuta Site. Since the basal horizon of the variegated clay is about 2.5 Ma old, the absence of displacement can be declared a positive finding, not only for the last 100,000, but also for the last 2.5 million years.

Within the detailed and strongly exaggerated sections, a 25-30 m downthrow has been located 3 km north and 4 km east of the site. It is connected with a zigzag-like fault traced in the topography and dated to the Holocene. However, if the section is drawn with no exaggeration, it can be stated that a deflection of less than 1° is enough to explain the actual results observed in the borehole columns. The loess-forming material, falling through air, obviously can cover gentle slopes, and thus the deflection could have arisen even before the loess accumulation. On the other hand, the fact that the deflection arose during the loess accumulation also cannot be excluded—there is no basis for dating. It should be clear that the existence of faults drawn in such strongly exaggerated sections is subject to doubt.

14.2.4. Safety Assessment

A preliminary safety assessment for the Üveghuta site has been prepared in cooperation with Belgian and Finnish institutions within the framework of a PHARE project initiated in 1998. This assessment focused on scenarios that did not include any disruptive event (the normal evolution scenario). Extreme or disruptive events (climatic change, undetected fault beneath the surface, failure of the backfill or seals) have been considered as separate scenarios. In addition, inadvertent human intrusion has been considered as a result of possible exploration boreholes drilled into the disposal area. Activity concentrations of radioactive isotopes, calculated for the vicinity of the disposal tunnels, do not significantly exceed concentrations in the natural environment. To calculate the biosphere concentrations, one has to investigate the effects of transport (delay, dilution, and dispersion) through the geological media. The model area discussed earlier has been used in a preliminary safety assessment. Results from the hydraulic modeling of this extended area show that groundwater velocities, at depths of 250–280 m, are only a few centimeters per year.

In 2000, a short geological summary was prepared, providing a synthesis of our geological understanding, and on this basis, an updated version of the safety analysis was performed. This new version was based on a preliminary safety analysis undertaken as part of an international cooperation. Results of the preliminary safety assessment for the Üveghuta subsurface disposal facility illustrate that the 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 normal and altered evolution scenarios. Because of the deep location and hydrogeological conditions of the site, the proposed concept of the subsurface disposal is not affected significantly by changes in external conditions.

14.2.5. Engineering Work

On the basis of available investigation results, the facility would be constructed 200–250 m below the surface on the outskirts of Bátaapáti village, which is 0–50 m above sea level. The exact location of the disposal area will be defined after additional geological investigations and experience are gained during tunnel construction.

Layout of the subsurface facility is affected by the geological environment and by the amount of waste. Presently, a tunnel-type arrangement seems more favorable. Both the waste drums and the disposal containers will be emplaced in disposal tunnels, so that radioactive isotopes that (after a long time) escape from the waste packages will be sorbed by the clay (in bentonite) backfill material, either around the waste packages or inside the containers. Thus, a significant release of radioactivity will not be possible, even after several hundred years. The backfill will limit access of groundwater to the waste packages. Granite pillars (10–20 m thick) will separate the disposal tunnels (6 or 10 m wide), ensuring the mechanical stability of the repository. The design of the layout and tunnel characteristics must be refined after further geological investigations.

14.2.6. Public Outreach

In Hungary, progress with LILW siting to date reflects a gradual realization of the need for public acceptance. In the early phase of the new siting process, the fundamental aim of all actions, events, and programs was to establish a long-term relationship between the local communities and the project management, and to continually keep local residents interested and confident in the ensuing developments. The siting process should also ensure that the community acceptance of the facility is compensated in a way that offsets all costs and leaves the community better off than it was previously. The repository has been presented as a mutually advantageous solution, based on voluntary cooperation of a number of partners that serves the interests of the whole nation—without forcing anything on anybody. The basis of the partnership resides in the trust of the local residents. During the course of the on-site investigations, a public-information program was maintained. In addition to the local municipalities involved, neighboring villages were also kept informed. Information was provided to the press and the scientific community in an organized manner.

According to the Hungarian Atomic Energy Act, information on fundamental scientific, technical, and other issues (e.g., related to risks) shall be disseminated to the public through news services and an education program. To regularly provide information to the communities in the vicinity of the facility, the licensee of a radioactive waste disposal facility must promote the establishment of a public control and information association and can subsequently provide assistance to this association. The law established the legal basis for the provision of financial incentives to any supportive group in the municipalities.

In April 1997, six municipalities in the immediate vicinity of the potential site organized their own Public Control and Information Association, under the acronym TETT. Although two neighboring municipalities have not joined the TETT, the level of acceptance can be judged as very good. Of course, full consensus and support is not a realistic objective. In the future, additional effort should be made to further improve this support. Since its establishment, the TETT has closely followed the site investigations, maintained contact with the project management, and kept their own and adjacent municipalities informed about developments. Building good public relations is a long process, one that will not end with site selection. Maintaining contact with the public during construction, operation, closure, and even the post-closure period of the repository is of paramount importance.

It is also a basic requirement to win the support of politicians and other decision-makers through the consistent implementation of an open information policy. If such people are not fully aware of the significance of the local situation, then demagogy can flourish and easily result in the failure of the project in a tense political atmosphere (at election time, for example). Socio-political issues relating to radioactive waste management are discussed in the Act on Environmental Protection (LIII, 1995). The Act requires assessment of the impact of major waste management activities in the form of an Environmental Impact Assessment. The Act also calls for consultation with citizens in local and neighboring municipalities and other interested groups.

14.2.7. Future Program

This next stage involving detailed site characterization and licensing procedures can only be started after approval by the Hungarian Parliament. This so-called preliminary decision-in-principle was scheduled to be requested in 2000. In the mean time, however, some experts have called into question the completeness of the research work conducted and the conclusions drawn from the results of the research in certain areas (e.g., hydrogeology, seismology). This scientific dissension has made the situation quite uncertain, and a political debate has emerged. Consequently, it was decided that further consultation and examination should be carried out in the interests of creating a broader consensus, both in the professional fields and in the general public. To promote the achievement of this consensus, the International Atomic Energy Agency (IAEA) was approached to organize a Waste Management Assessment and Technical Review Program (WATRP) to carry out a peer review for the validation of the activities and results of the site selection, and to give recommendations based on good international practice.

The principal conclusions of the WATRP team's review were that the process that led to the selection of the Üveghuta site appears reasonable and has properly considered both the Hungarian geology and public acceptance. The Üveghuta site appears potentially suitable for development of a safe repository for disposal of operational and decommissioning LILW from nuclear power generation. The site characterisation and repository design, however, should continue.

Based on existing information about the geological situation at the site, the WATRP team recommended that, for a rock mass of the type to be expected at Üveghuta, a "design as you go" approach is to be followed, adapting the design of the repository to the geological situation as revealed during excavation. The safety assessments based on limited, early geologic investigations should be updated. There is a need for an integrated safety assessment using the currently available site and conceptual design information, and including a broader spectrum of scenarios. This integrated safety assessment should form the basis for a continuing site characterization.

In accordance with the principles of international practice, the Üveghuta Site can be regarded as geologically suitable for the disposal of radioactive waste. This means that it is worthwhile to perform a detailed exploration, which is a prerequisite to licensing the facility. The suitability requirements of Hungarian regulations, which are much more prescriptive than the international ones, are fulfilled by the Üveghuta Site in a qualitative sense. Quantitative fulfillment of some of the requirements still awaits confirmation. The final confirmation, or rejection, of suitability is in both cases the objective of the safety-assessment study. The preliminary safety assessment in 1999 did not give rise to any doubts concerning suitability.

The geological exploration of the Üveghuta Site has not yet been finished. Future exploration—besides the quantitative confirmation discussed above—must fix the boundaries of the rock mass in the vertical and horizontal sections in which the repository can be located. The number of existing boreholes is obviously insufficient, and thus, only some generalizations can be given:

- The rock mass suitable for location of the repository will probably be situated between the steeply dipping fracture zones; hence, its size in the vertical direction can prevail over that in a horizontal direction or can, at least, be comparable with it. This differs significantly from the original engineering concept, which considered the repository in one single horizon. Thus, a new engineering concept is needed that defines the size and shape of the repository, taking into account the real geological situation. The shape and size of the host rock block will be derived from this concept.
- In delineating the host rock block, all hydrogeological and rock-mechanical conditions must first be taken into account. Hydrogeological conditions mainly influence safety, whereas the rock-mechanical conditions determine the technical circumstances and the cost of excavating drifts. The granite block to be selected must be optimal in both aspects.

The main tool in future exploration can only be drilling, because until the host rock block has been delineated, it is unclear where to orient the mining drifts. The cost and the associated risk of drift excavation is rather high. In any case, galleries will disturb both the hydrodynamic and hydrogeochemical systems, and make it impossible to continue exploration of the main geological barrier for the site. These systems in turn make up the groundwater flow system, which is the basic element in the hydrogeological evaluation of the site. Until the flow system has been studied in sufficient detail, the galleries cannot be constructed, not even in the neighborhood of the site.

Furthermore, a geological evaluation summary will have to be prepared. Based on the new geological exploration plan, supplementary on-site investigations will be pursued. In parallel with this work, an integrated safety assessment will also be performed. For licensing purposes, further geological and engineering examinations as well as safety assessments are required. According to the IAEA Expert Group and the Hungarian Geological Survey, further decision-making related to Üveghuta must emphasize safety.

Preparation, implementation, and evaluation of each task need very careful professional control and inspection. Further steps could be undertaken only after appropriate administrative approval. There is a need for an integrated safety assessment using the currently available site and conceptual design information, and including a broader spectrum of scenarios. This integrated safety assessment should form the basis for continued site characterization. During the authority licensing process, a dozen approvals are needed in order to obtain the construction license. From the legal standpoint, the Atomic Energy Act stipulates that preliminary approval of the Parliament is also required.

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 only exceeds by one order of magnitude the value for the background granite. The biggest influx came from a relatively insignificant fissure, whereas the usual influxes from thick fracture zones did not differ from those from small fractures.

Any future exploration of the Üveghuta Site must answer questions concerning the exact boundaries, in maps and sections, of the granite block in which the repository should be located. Only boreholes can serve as the main exploration tool, while the groundwater-flow system is not yet studied in sufficient detail. Galleries that would form an alternative to boreholes would undoubtedly disturb both the hydrodynamic and hydrogeochemical situation, making it impossible to perform a hydrogeological evaluation of the site on which the safety assessment can be based.

14.3. HLW MANAGEMENT

14.3.1. Strategy for the Back-End of the Fuel Cycle

With regard to the back-end of the fuel cycle, there are two main solutions: (1) reprocessing of the spent-fuel elements and disposal of the resulting high-level radioactive waste; and (2) use of the resulting plutonium or direct disposal of the spent fuel following conditioning. Smaller countries, such as Hungary, have chosen to delay making a final decision on these options, and this position was adopted by the Hungarian Atomic Energy Commission in 1993 and confirmed in 1998. The facility for the interim storage of spent fuel allows for the storage of fuel assemblies for a period of 50 years in Hungary. Hence, it is not necessary to reach an immediate decision on final disposal of spent fuel, and the decision may be delayed. This offers the possibility of monitoring experience in other countries, in particular the results of international research on the transformation of HLW. However, it is expedient to develop a policy and strategy, as well as a working program, to support the 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.

14.3.2. Interim Storage of Spent Nuclear Fuel

As a result of the operation of the Paks NPP, an average of 400 spent-fuel assemblies are generated annually. This is equal to approximately 46.5 tons of heavy metal. A certain part of the spent-fuel assemblies—2,331 pieces—were shipped back to Russia in the period between 1989 and 1998. The shipments to Russia became more and more difficult and expensive at the beginning of the 1990s. Because of this, and to assure the continuous and reliable operation of the power plant, it became necessary to provide for interim storage of spent nuclear fuel in Hungary. The Interim Storage of Irradiated Fuel (ISIF) was constructed on site at Paks NPP using a British license, and the facility was commissioned in 1997. The system is modular, vault dry storage (MVDS), as shown in Figure 14.6. Because of its modular design, the storage facility can be expanded in increments according to the operational needs of the power plant. At present, there are seven vaults in operation, each containing 450 assemblies. The present capacity of the facility is 3,150 fuel assemblies, which corresponds to 365 tons of heavy metal. As of April 1, 2001, the quantity of spent-fuel assemblies stored in the ISIF was 2000 (i.e., 4.5 vaults are filled).

The expansion of the storage facility is in progress at the present time as well, and it is anticipated that by the end of 2002 the 11th chamber will be completed, bringing the total capacity to 4,950 assemblies. In terms of the amount of spent-fuel assemblies that will be generated by the end of the anticipated lifetime of the power plant, there will be a need to construct an additional 25 vaults

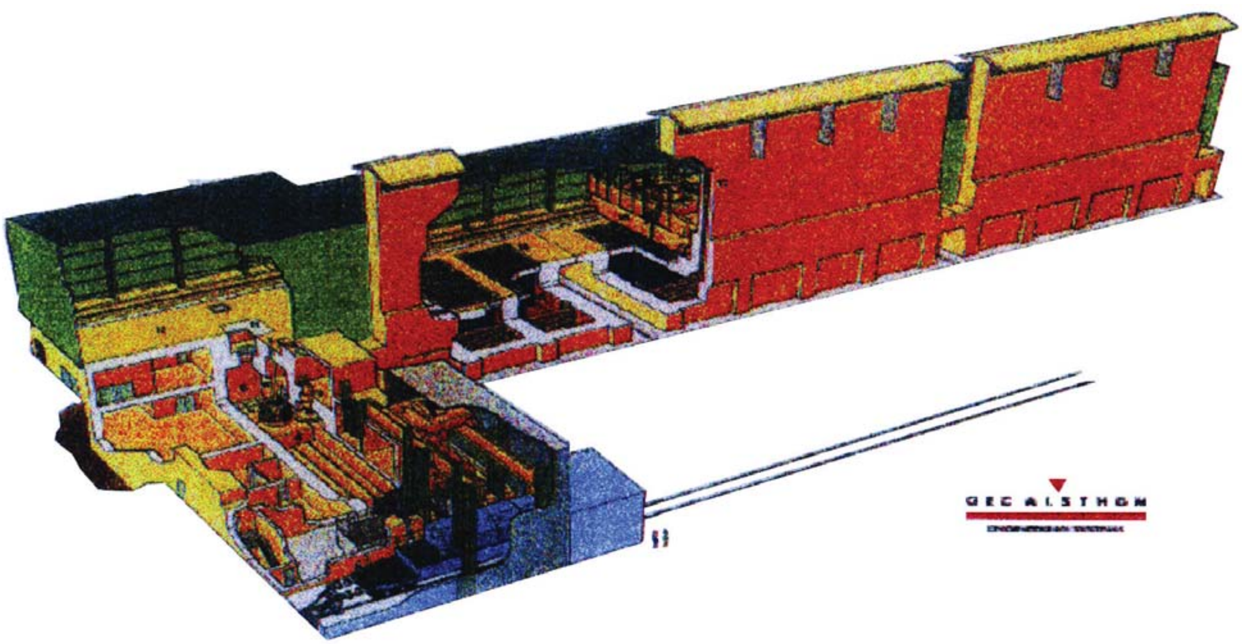

Figure 14.6. Modular, vault dry storage

for the interim storage of spent fuel. The schedule for the erection of vaults will follow the rate of spent-fuel generation.

14.3.3. Attempt To Site an HLW Repository

Among the small countries faced with the problem of HLW disposal, Hungary was in an outstanding position due to an ambitious project launched in 1993. In a "desktop" review of the territory of the country, it became clear that the most promising location for disposal of HLW is the claystone (aleurolit) formation near the village of Boda. Geologists became familiar in detail with this formation during the operation of the Mecsek uranium mine.

In 1993, the Mecsek Ore Company prepared an exploration program for the operation of the mine. In mid-1994, from details of the underground installations and the infrastructure of the uranium ore mine, the Boda Claystone Formation (BCF) was disclosed between 1030 and 1080 m below the surface. With this information, *in situ* investigations could be started with a minimum of cost and time, and the results were more economic and much more efficient than investigations conducted only from the surface. This has produced a huge amount of indispensable data that could not have been obtained in any other way.

The BCF is developed in SE Transdanubia, west of the town of Pécs (see Figure 14.7). This sedimentary formation of Permian age (250 to 260 Ma) is known over an area of 150 m^2 with a thickness of 700 to 900 m. Excavation for an exploration drift was started from the uranium mine and reached the clay-stone formation in 1994. That started the underground data collection in this area. A Canadian-Hungarian intergovernmental agreement was set up in December 1994, in which the AECL (Atomic Energy of Canada Limited) and a number of Hungarian specialists prepared a study of this formation. This study systematically collected all available information, and the ways it could be acquired, that would be necessary for a full-fledged assessment of a geological formation potentially suitable for an HLW repository.

Almost at the same time, a Governmental Decree was issued in Hungary concerning the cessation of uranium ore mining in Hungary. It specified that the investigation of the BCF had to be continued, using the underground facilities, until the definite termination of the mining operations (scheduled for December 31, 1997). With this Governmental Decree in mind, a conceptual plan for an approximate three-year "short-term" investigation phase was developed. The 1995–1998 investigations were performed according to this plan.

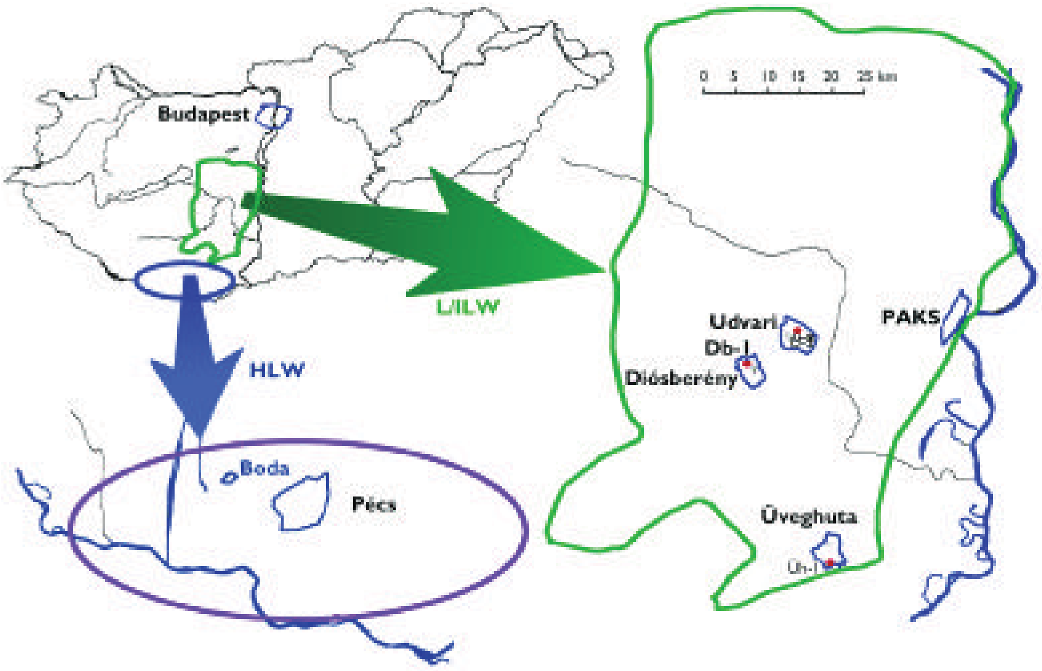

Figure 14.7. Candidate areas for radioactive waste disposal in Hungary

The short-term program was focused on investigations of those critical factors and parameters, which, in case of unfavorable results, could lead to an "unsuitable" qualification for the claystone formation. The plan defined four groups of investigations:

1. *In situ* investigations by means of geotechnical tests to be executed in underground exploration. Because of the principal aims of the short-term program, the majority of investigations were organized in this group. During the years 1995–1998, about 255 m of drifts were excavated and 1,950 m of underground drilling were done. In this phase, there was neither *in situ* testing with radioactive isotopes nor heat-load testing.

2. Geological and hydrogeological investigations on the surface. These concerned mainly the outcrop of the BCF and the surrounding areas (about 25 km^2). Geological mapping and data collection were complemented by drilling six shallow boreholes (total length 400 m) and a few geophyiscal measurements destined to support the study of neotectonism. A Geographic Positioning System (GPS) suitable for monitoring shallow crustal movements was also set up.

3. Comprehensive laboratory testing of water and rock samples from underground locations and from the area of surface mapping, including the determination of isotope migration paths.

4. Other tasks, such as collecting and processing archive data, development of a computerized database and a Geographic Information System (GIS), operation of a seismological observatory in the area, as well as the interpretation and assessment of data using geomathematics and numerical modelling.

The Hungarian geoscientific community was to a great extent involved in this work. Special care was taken to comply with internationally accepted methods of investigation and assessment, as well as with the recommendations of the international projects supported by the IAEA and the Nuclear Energy Agency of the European Union and of the OECD. The technical level and top quality of the investigations were also guaranteed by the involvement of highly experienced foreign companies (e.g., AECL, Golder Associates).

The detailed planning and technical coordination of the Mecsek-West Investigation Program was performed by the Mecsek Ore Mining Company and, from May 1998 on, by its legal successor, Mecsekérc Environmental. The investigations have been funded by Paks NPP. The program was progressing according to schedule and without problems from 1995 through 1998. However, serious problems were caused by the fact that uranium ore mining was abruptly terminated three months earlier than scheduled.

On the basis of the collected results, the formation turned out to be suitable for further investigations, and there was not a single result that could be used to question the suitability of the rock body for the purposes of a HLW repository. On this basis, a proposal was made in 1998 for further development and operation of the research laboratory. However, a high-level decision, in the summer 1989, directed the closure of the mine to be continued in accordance with the original schedule, and it was not possible to continue operation of the research laboratory.

14.3.4. Achievements

An assessment was made, according to the following four main points, using information obtained during the previous phases, as well as the short-term investigation program itself:

1. Primary isolation properties of the formation (unaffected by anthropogenic interventions)
2. Effects of anthropogenic interventions on the isolation properties of the formation (secondary isolation)
3. Long-term stability of the formation and its geological environment, possibility of extrapolation of the results in time
4. The rock body as a medium for a technical structure designed for the long term.

The primary and secondary isolation behavior and long-term stability of the BCF was compared with those of other potential host rocks. The results are summarized as follows.

- At granite sites, the primary aim is to avoid high-permeability fracture systems that could lead to an undesired rate of contaminant migration, even if the dimensions of the repository have to be increased. The requirement is to have an intact rock body of appropriate size.
- The concept of installing repositories in Neogene clays or salt bodies is that, for the depth interval in question, they possess the feature of complete self-closing. Owing to the huge difference in concentrations, the geometry of the host rock body has to be designed to take into account the very slow, diffusive transport of nuclides in the pore structure. In this manner, a realistic aim may be the disposal of HLW at shallow depths (eventually 200 to 300 m) and in host rocks of relatively reduced thickness (100 m).
- According to the results of our investigations, in the case of BCF, the formulation of an intermediary concept seems to be justified. The isolation parameters are in general favorable. The reformation of clay minerals, the isolation capacity resulting from the "sandwich" structure and hydrogeological block structure of the intact zones, and the parameters of isotope transportation (favorable strain state, etc.) of the BCF are close to those of clay sites. However, in the investigated depth interval, the phenomenon of self-closing, although observable, is far from being complete. Nevertheless, the geometrical dimensions of the sequence lead to an enhancement of disposal security, based on the granite concept.
- The nature and rate of the geological processes hitherto known in the surroundings of the BCF (e.g., the uplift rate and relative movements) suggest that, in the next hundred thousand years, no such changes will occur here that would radically modify the hydrological, hydrodynamic system of the area. From this point of view, no site-specific risk could be cited as compared to other potential European sites. However, in comparison with soft, young clays, a fundamental advantage is that a cavern system excavated in BCF seems to possess a long-term stability, according to instrument measurements, in spite of the considerable depth.
- According to the final report summarizing the results of the program, the investigations have not revealed any fact that would rule out BCF as an appropriate host rock. Its geometry, size, isolation properties, and geotechnical parameters meet international standards and are suitable for the envisaged function. This statement is supported by the results of a country-wide screening performed in the meantime. The Boda Claystone Formation has remained in first place among a succession of potentially suitable geological formations in Hungary.

A drift system (called drift Alfa) was developed in subsequent stages of the mining operations, 1990–1998, using the geological data and the installations and infrastructure of the uranium ore exploration, which had been going on for more than forty years. The drifts were excavated starting from a ventilation shaft at a depth of 1030–1080 m below the surface (–690 m b.s.l.). The total length of drift Alfa that was developed in sandstone in 1990 and 1991 is 742 m.

Chambers were constructed in sandstone at 250, 426, and 740 m in drift Alfa and in the BCF at 279 m and 422 m. These drifts made it possible to carry out drilling and special investigations. In connection with the Alfa drift system, 70 cores were drilled (total length about 34,000 m), out of which about 2,300 m. intersected the BCF. During excavation and drilling, detailed geological and hydrogeological records were compiled.

A few examples of methods that have yielded important results are as follows:

- On the basis of previous knowledge, a first step in the geological documentation system was to develop a uniform description for beds and rocks of the BCF.
- The geological and hydrological documentation of the drifts and borings, as well as the geophysical logging of the latter, have been accomplished. In addition, laboratory testing and analyses have been made on the mineralogical, petrological, chemical, and isotope geochemical composition, as well as the mechanical, petrophysical, hydraulic, isotope transportation properties of the main rock types and fissure fillings.
- A comprehensive program of repeated measuring, sampling, and laboratory testing of the chemical composition, temperature, geochemistry (e.g., Eh-pH, water age, O-H-S-C stable isotopes, noble gas content and isotope ratios) and microbiological parameters of the waters derived from the overlying sandstone and the BCF itself, including changes in tracers with time.
- Investigation of the geological-structural setting, features, and sedimentological characteristics of the BCF beds and their relation to the overlying sandstone formations within the Alfa drift system. Details of the structural elements and features, and their hydrodynamic and hydraulic characterization, rely upon the results of the hydrogeological testing program.
- Multipacker and double packer hydraulic, hydrodynamic tests performed on one or more wells in different directions, to identify and determine the tectonic and stratigraphic boundary zones, faulted zones inside the BCF, and parameters characterizing the primary state of intact sections.
- *In situ* investigations of strain and stress (overcoring, fracturing, hydroblasting, etc.)
- Rock mechanical measurements (convergence, extensometer, CSIRO-cell measurements), geophysical measurements (absorption tomography, spontaneous acoustic emission, ultrasounding) aimed at understanding the mechanical changes occurring in the excavation-disturbed zone (EDZ).
- Methods of investigating changes in water content of the EDZ and the features of the hydraulic potential field around the drift (multipacker systems, "climatic measurements," geo-electric measurements, heat photography).
- Methods of investigating the efficiency and suitability of rock-bolting and excavating technologies, and deformation monitoring to observe the long-term stability of drifts.

The complex interpretation of the results, using numerical modeling, was considered one of the most informative elements of the short-term program. As mentioned earlier, on the basis of the collected results, the BCF formation was determined to be suitable for further investigations, and there was not a single result that could be used to question the suitability of the rock body for the purposes of a HLW repository.

14.3.5. Future Plans and Outlook

The rejection by the government in 1999 of the PURAM plan for an underground research laboratory at Boda, as a step towards the development of a deep repository, left Hungary with no practical plan for the disposal of high-level and other long-lived radioactive wastes. In this new situation, the lessons learned from the first effort indicated that a different policy would be needed in developing a new HLW project. This policy must contain an action program to guide developments from the new start to the end of the project. In the first phase (in mid-term), we need to elaborate a strategy concerning the closure of the fuel cycle and disposal of the HLW. The policy will be circulated for debate with the aim of gaining widespread professional and public consensus.

PURAM has contracted with ENRESAin Spain for consulting services to help develop a strategy for final disposal of high-level and/or radioactive wastes, and the management of spent nuclear fuel. The main milestones for the acceptance of this strategy could be as follows:

- Elaboration of the potential topics for relevant scenarios
- Analysis and evaluation of the scenarios by appropriate professional panels
- Reducing the scope of the scenarios by selecting those few that are potentially realistic and best suit domestic circumstances
- Development and execution of a work program for the selection of a final strategy, i.e., a detailed analysis and evaluation of the realistic scenarios
- Approval of the strategy.

After the first discussions, the following SF/HLW management strategies appear to be better suited to an analysis of the Hungarian program:

1. *Deep Geological Disposal*
   Nearly all countries with nuclear power programs have adopted this solution, although with different standards and timeframes. It is generally accepted as a final solution for HLW, SF, and LLW, because there is compliance with the principles of radioactive waste management, as well as ethical and environmental considerations. At the same time, the solution can be conceived and designed with sufficient flexibility so as to follow the evolution of new technology in a reasonable timeframe.

2. *Supervised Extended Storage*
   Extended or long-term storage is increasingly being considered as an option for HLW/SF management. In general, long-term storage is intended to provide additional time to allow for the development of improved options, delays in design and construction programs for planned disposal facilities, or to increase public acceptance of proposed options. In the case of Hungary, this could provide flexibility for the future consideration of other options, in particular the possibility of regional or international repositories, or the consideration of possible contractual arrangements for the reprocessing of spent fuel in Russia, either with or without the return of the waste.

The development and approval of this new strategy will require 5–7 years. 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.

Geological disposal at the Boda site remains one of the preferred options. In parallel with the development of a new policy, we plan to conduct a country-wide screening to identify potential new host rock areas for the HLW repository, to continue the surface investigations of the Boda area, and to continue the mathematical processing of the accumulated geological parameters.

One of the prerequisites for any further activities is the allocation of the necessary funds. Unfortunately, this has recently become one of the weak points in the Hungarian waste management program. In the past, activities related to disposal of radioactive waste were conducted within the framework of the State budget for waste not originating from power generation, while the Paks NPP was responsible for financing the disposal of its waste. On January 1, 1998, the Atomic Energy Act established the Central Nuclear Financial Fund to be based on payments from parties using nuclear energy. The goal of this fund is to provide for the disposal of radioactive waste, interim storage, and final disposal of spent fuel, and the decommissioning (dismantling) of nuclear facilities. Thus, operators of a nuclear facility must accumulate, during the effective life-cycle of the power plant, the necessary funds to cover the costs of decommissioning the facility and disposing of the waste, as well as any costs arising over decades following such decommissioning. The funds are to be managed in such a manner that their value remains stable, and they may only be used for the aforementioned purposes.

The amount of payments by nuclear facilities (primarily the Paks NPP) to the fund is set forth in the annual Act on the State Budget, based on a cost projection provided by the Hungarian Atomic Energy Authority and the Hungarian Energy Office. In the interest of maintaining the value of the fund's resources, the Hungarian government should ensure the contribution of an appropriate amount to be paid from the State budget. Unfortunately, as a consequence of a "negative campaign" against the radioactive waste program by some vocal politicians, the Parliament cut back the budget for the HLW program for 2001 and 2002. Moreover, they annulled the government contribution, which seriously jeopardizes the future of the fund. It is hoped that, after the Parliamentary elections in May 2002, the waste management projects will get a new impetus from sober decisions by the policy makers.

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Chapter 15

Evaluation of a Plutonic Granite Rock Mass for a Geological Repository in India

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15.1. INTRODUCTION

The Indian program for a geological repository to handle the disposal of immobilized high-level radioactive waste is based on two major considerations. The repository must be located in granitic host rock in a tectonically stable region, and the selection of such a repository must proceed by narrowing down the choice from large areas (thousands of km^2) to a 4–6 km^2 area, adopting certain essential attributes and criteria. A number of plutonic and gneissic granites lying in Seismic Zones I and II (as described in Kumra et al., 1991) have been considered for assessment. These zones are characterized by a horizontal seismic coefficient of 0.01 and 0.02 respectively, and are comprised of relatively stable cratonic areas of the Indian Shield.

Out of many favorable granitic areas, the Sankara pluton, northwest of the Rajasthan State, was considered for the start of detailed investigations based on the nature of its granite, remoteness of the area, low rain fall, very low population density, thin vegetation cover, and low groundwater potential. In the first stage of our site selection, areas of 100–150 km^2 (“promising zone”) were identified, based on data collected from available literature and remote sensing. The areas were geologically mapped on a 1:50,000 scale, and an area of 52 km^2 was identified with lesser frequency of basic dikes. In the second stage, this area was geostructurally mapped on a 1:5,000 scale to select a subzone of 25 to 30 km^2 in area (“potential zone”). This subzone was the main-thrust area for carrying out geological, structural, geophysical, and hydrogeological investigations.

In the third stage of site selection, the subzone was geologically mapped on a 1:1,000 scale, geophysically surveyed at 1 km grid intervals, using resistivity, electron microscopy (EM), and magnetic methods as well as drilling of shallow boreholes down to a depth of 150 m. Based on the results, a rock mass of about 6 km^2 was further investigated by deep drilling to a depth of 500 m, and a geophysical survey was then carried out at a closer grid interval of 200 m in a 4 km^2 block, using the above-mentioned methods. All boreholes were geologically and geophysically logged, and core samples were examined for physical-chemical, mechanical, thermal, mineralogical, petrographic, and microstructural properties (along with recording of core recovery and rock quality designation [RQD]). Based on all these investigations, a granite rock mass of about 2.5 km^2 was found to be sufficiently massive and uniform, with a minimum number of dikes and fracture zones.

15.2. GEOLOGY OF WESTERN RAJASTHAN

In geologic history, the emplacement of Erinpura granites as S-type magmatism is related to the collision of the Bundelkhand craton with the Marwar craton along the South Delhi Fold belt. This collision was followed by large-scale bimodel volcanism in the NW Rajasthan, known as the Malani Igneous Suite. The magmatism is genetically related to doming and arching of the crust as a result of plume activity prior to Gondwana rifting in an essentially nonorogenic tensional tectonic regime. The Malani Igneous Suite, which is spread over 50,000 km^2, extends for about 255 km from south of Sirohi to north of Pokran, and for about 296 km from east of Jodhpur to the international border in the west. The Suite is comprised of both effusive and intrusive rocks and represents one of the largest igneous provinces in the world.

This suite has been classified into four distinct igneous episodes (Bhusan, 1984; 1985) with the second and fourth cycles corresponding to emplacement of intrusives as granitic plutons and dikes (respectively), whereas the other two are essentially marked by acidic lava flows. The granite-graodiorite-tonalite complex of the study area forms part of the Jalore Group and spreads over 1,500 km^2, marking the second cycle of magmatism in the suite. Geochemical studies on these granites (Bhusan, 1984) reveal an anorogenic nature related to a plate-tectonic setting, associated with doming and arching of the Gondwana rifting floor with a shallow subduction angle. The general geological succession of western Rajasthan is shown in Table 15.1.

15.3. GEOLOGY OF JALORE GRANITE

Jalore granites cover an area of 7,500 km^2 in the districts of Jaisalmer and Jodhpur, and are chiefly made up of pink and grey colored granites with basic intrusives. These granites are typically anorogenic in plate instrusions; however, relics of older granites, characteristic of collision tectonics, are also found within the pluton (Acharya and Bajpai, 2000).

Granites of the Jalore Group of the Malani Igneous Suite occur as plutons within the Karara Group. These granites, in our present area of study, cover an area of 1,500 km^2. The Sankara pluton of Jalore granites has been reported to be affected by third-stage volcanism and fourth-stage intrusives. The third stage is represented by the Sanawara Group of acid lavas, overlying the Jalore granites. The basic dikes mark the fourth stage of the igneous cycle (Bhusan, 1984). Jalore granites range in composition from alkali granite to sodic tonalite. The alkali granites are pink, whereas granodiorite and tonalite are grey. Both these units have been intruded by dikes of diorite, aplite, dolerite, and granodiorite.

Table 15.1. Geological succession of W. Rajasthan

Recent		Alluvium
Pleistocene		Laterites
Lower Cretaceous	Abur beds Limestone	Barmer sandstone/ Barmer
Jurassic	Jurassic	Lathi sandstone/Jaisalmer sandstone and Jaisalmer limestone
Lower Paleozoic		Marwar super group/ Trans Aravalli vindhyans
Lower Paleozoic	Malani igneous suite	Malani,Jalore and Siwana Granite/Malani Rhyolite
Upper Proterozoic	Jodhpur group	Vindhyan sandstone

The study area is mainly occupied by a variety of granites ranging in composition from meta-aluminus to peraluminus. Large granodiorite bodies have been mapped in the north and northeast parts of the area. Both these rock units are intruded by late-stage intrusives such as aplite/dolerite and diorite. The trend of these intrusives coincides with the three dominant joint sets in the area. The contact of granite and granodiorite with dolerite is very sharp, without much assimilation. This feature, coupled with a general lack of tectonic fabric in the host rock and the alkaline nature of basic magmatism, suggests a ring-type intrusion in the area. The granodioritic body is also characterized by some occurrences of xenoliths of basic and dioritic rocks. Surface geological mapping revealed the presence of three major sets of joints in the studied area, striking NNW-SSE, N-S, and NE–SW. These orientations, combined with subsurface measurements of joints/fractures in boreholes, indicate that a majority of these joints are cooling joints. However, the moderately dipping joints are shear joints developed mainly where intrusives contact the host granites.

The geological map of the investigated zone is given in Figure 15.1.

15.4. INVESTIGATIONS

The site-selection study was carried out in successive stages and considered a number of criteria (Mathur et al., 1996), including factors such as tectonics, lithology, hydrology and hydrogeology, and socio-economics. In Stage I, screening of the entire country was accomplished, and promising granitic zones of 100–150 km^2 were identified in the northwest, central, and south-central parts of the country. The identification was based solely on data collected and collated from available literature and remote sensing.

The Sankara pluton (composed of Jalore granites) in northwest Rajasthan was considered for further study in Stage II. An area of 52 km^2 in the JS-1 was investigated with geological and geophysical surveys, trenching, pitting, sampling, and collection of local data, narrowing the area of interest to a subzone of 25–30 km^2. To better understand the structure of the granite and dike rocks (with respect to their geological evolution), we also conducted geochronology and palaeomagnetic studies on rock samples. Shallow borehole drilling was initiated in the selected subzone.

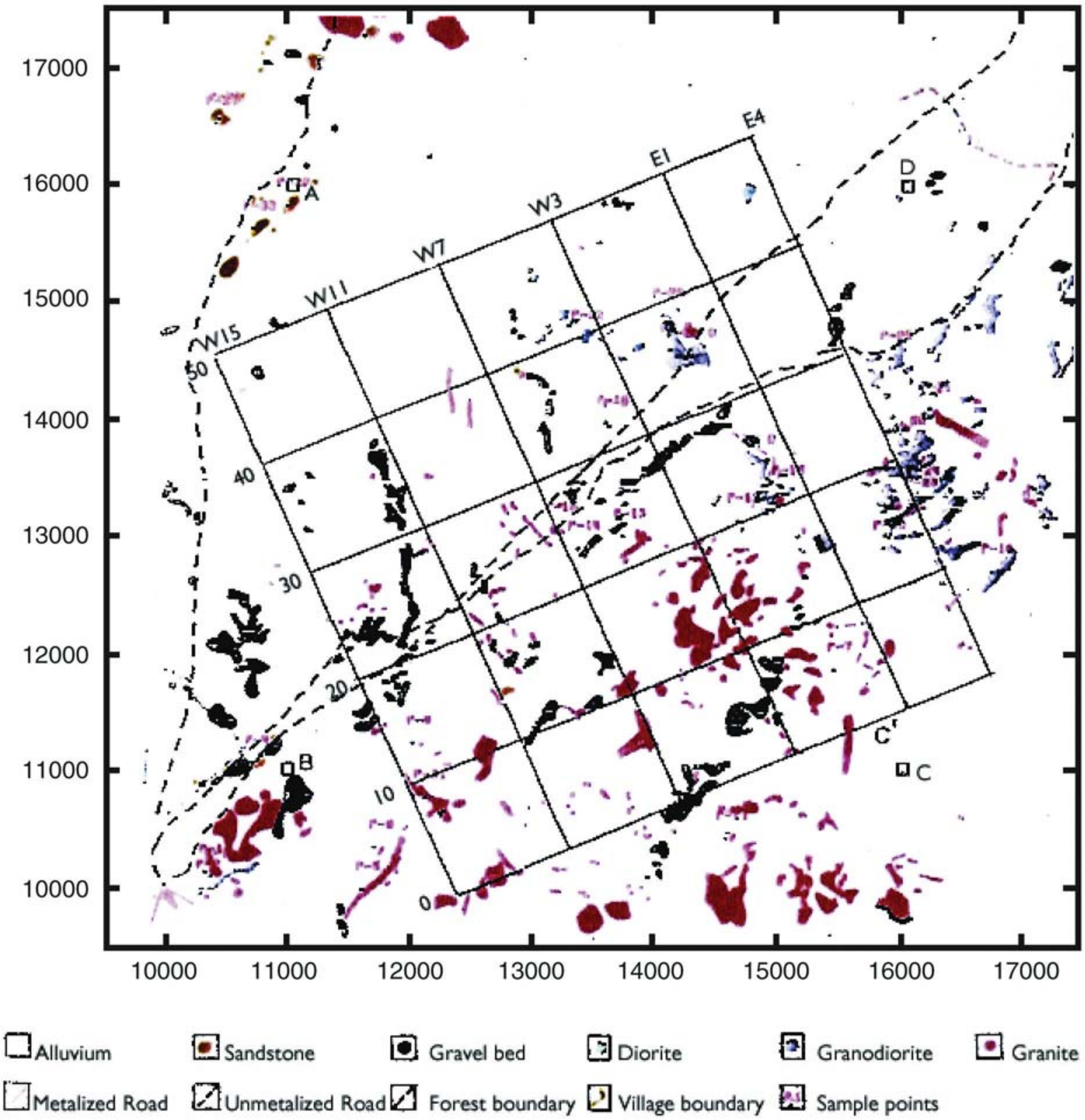

Figure 15.1. Geological map of study area showing outcrops of pink granite , diorite , granodiorite , etc .

Further investigations in Stage III involved geostructural mapping on a larger scale and close-grid geophysical surveys, deep borehole drilling, and associated core and borehole studies in a 6 km^2 area. Stage III also included geophysical logging and detailed study of core samples for homogeneity, variation in rock type, joint, and fracture systems, followed by a determination of physical, chemical, minerographic, thermal, and mechanical characteristics.

15.4.1. Geological Investigation

The main zone of 100 km^2 was geologically and structurally mapped first. Then, within this area, a smaller area of about 52 km^2 with a smaller density of basic dikes was mapped on a 1:5,000 scale. A further-reduced subzone of 25 km^2 was mapped on 1:1,000 scale. Trenching, pitting, and collection of samples continued throughout all stages.

Core samples recovered from shallow and deep boreholes, drilled in the subzone, were examined in detail for lithological variations, weathering patterns, structural features, and geochemical aspects. Borehole geophysical logging was also performed using gamma-gamma, natural gamma, density, caliper, SP, SPR, sonic, flow meter, and thermal methods.

Rock-mass suitability was evaluated by regrouping the data at intervals of 50 m for RQD, fracture-zone thickness, and number of fracture zones per 100 m drill length, etc. Results from this evaluation indicated that the rock mass was basically sound, with all borehole readings falling into excellent, good, and fair categories except for one (borehole MGJ-1), where the rock mass readings fell into fair, poor, and very poor categories.

From the synthesis of core recovery, RQD, and geological homogeneity, two zones were selected as possibilities, along with another zone of moderate confidence, totaling about 6 km2 in area.

15.4.2. Geophysical Survey s

15.4.2.1. First Stage (Test Traverse)

Geophysical surveys, using electrical methods, were carried out in the area of study to test the suitability of various methods and to evaluate the homogeneity of the granitic rock mass with respect to the location of fracture zones, water-bearing zones, intrusive rocks, etc. The observations were recorded over a path length of 12 km, at intervals of 100 m. The line was selected to dissect the entire area. The following methods were employed:

1. Direct current resistivity (DCR) method to obtain information at greater depth
2. Horizontal loop electromagnetic (HLEM) method to probe the shallow depths
3. Very low frequency (VLF) method to obtain near-surface information.

These investigations revealed the presence of semi-weathered granite at varying depths (7–60 m) below the surface/soil cover. Fresh granites were identified below these depths. A dike was also identified along the line of survey (Mathur et al., 1996).

15.4.2.2. Second Phase (25 km^2 Area)

In this stage, geophysical surveys were carried out, using deep resistivity soundings (31 locations) at intervals of 1,000 m and employing a Schlumberger configuration with electrode spacings up to AB = 8,000 m. Also, profiling using Schlumberger multi-electrode spacings AB/2 = 500, 1,000, 1,500, 2,000, 2,500, 3,000, 3,500, and 4,000 m were carried out along five traverses (W15, W11, W7, W3, and E1 grids—see Figure 15.1). The site was also surveyed by Wenner electrode configuration, with a spacing of 100 m for shallow profiling.

Isoapparent resistivity contour maps were prepared at depths of 100–1000 m to understand the lateral-resistivity distribution, and pseudo-depth sections were prepared to better understand the conducting zones along the sections. Acontour map of isoresistivity at a depth of 750 m is shown in Figure 15.2. Geological sections based on 1D inversion of deep resistivity data were also prepared.

Electromagnetic soundings and profiling (using frequencies of 222, 444, 888, 1,777, and 3,555 Hz) were conducted, by recording in-phase (IP) and quadrature values of the the vertical component for the resulting induced magnetic field, (OP) at a transmitter-receiver (T-R) separation of 100 m (with station spacing of 100 m). The contour trend of IP and OP at 222 and 444 Hz was drawn to understand the electromagnetic anomaly, which indicated a NNW-SSE trend consistent with the observed magnetic-anomaly contour map. Total magnetic-intensity measurements were carried out in a grid pattern of 20 traverses in a NNW-SSE direction, each 5 km in length and separated by a distance of 250 m, over an area of 5 km^2.

The resistivity data, interpreted using 2D inversion models, were obtained for various layers and groundwater conditions. This study helped in identifying locations for deep boreholes. The presence of dolerite dikes and sills, identified in the geophysical surveys, were confirmed by actual core drilling. A relatively homogeneous granite surrounded by a weak zone was identified in two small pockets. These zones were associated with closure of high-resistivity contours at depth, scattered contours of negative magnetic anomalies of –400 gamma, and very scattered contours of IP and OP of the EM anomaly. The deep-resistivity sounding curves rise sharply up to a maximum current with an electrode spacing of 8,000 m for relatively homogeneous granites.

In this study area, two prominent structural trends, NNW-SSE and NE-SW, were mapped for deep-seated structures/bodies (Figure 15.3) These trends appear to be broadly corresponding to weak zones through which basic and acidic dikes have been emplaced in the host

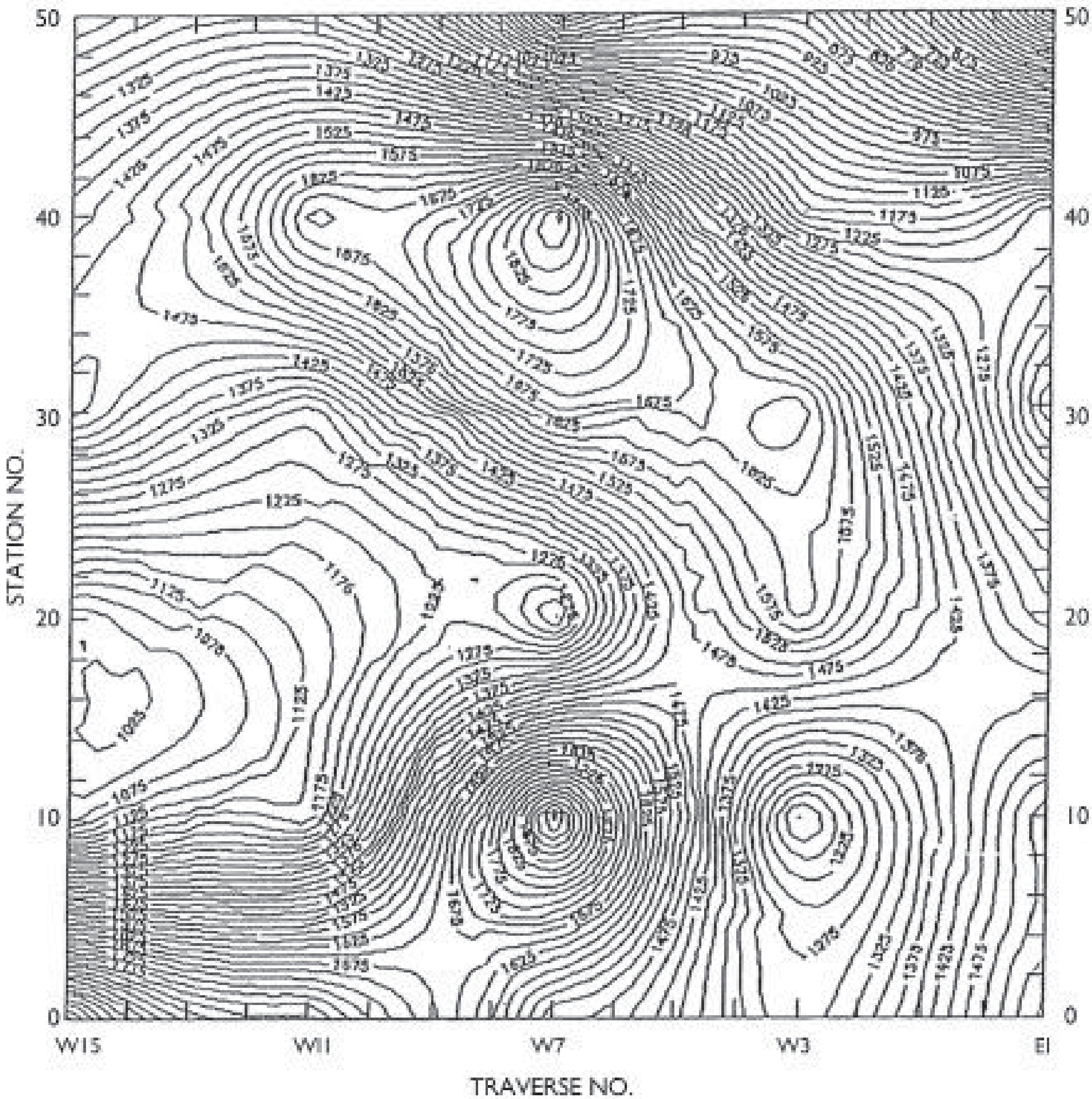

Figure 15.2. A typical isoapparent resistivity map at a depth of 750 m at contour interval of 250 ohm-m

rocks. The electrical resistivity measurements strongly suggest the presence of weak zones in the study area.

15.4.2.3. Third Phase (4 km^2 area—Closer Grid Interval)

From the earlier investigations using geological, geophysical, and geochemical methods (as well as drilling), an area of about 2 km × 2 km was identified as the most suitable block. To further understand the area with respect to intrusions, groundwater, fracture zones, etc., close grid investigations were conducted using an integrated geophysical methodology comprised of magnetic profiling, DCR and EM surveys (both in profiling and sounding modes). A total of 21 traverses (with a spacing of 100 m) were planned, with a total of 41 stations along each traverse (station interval of 50 m). Resistivity and electromagnetic-profiling measurements were taken at each 50 m station interval along 11 traverses, with a traverse spacing of 200 m. A symmetrical four-electrode Wenner configuration (with an electrode separation of 100 m) was employed for resistivity profiling. The horizontal loop electromagnetic (HLEM) method with T-R separation of 100 m was adopted for EM studies at 110, 220, 440, 880, 1760, 3520, 7040, 14080, 28160, and

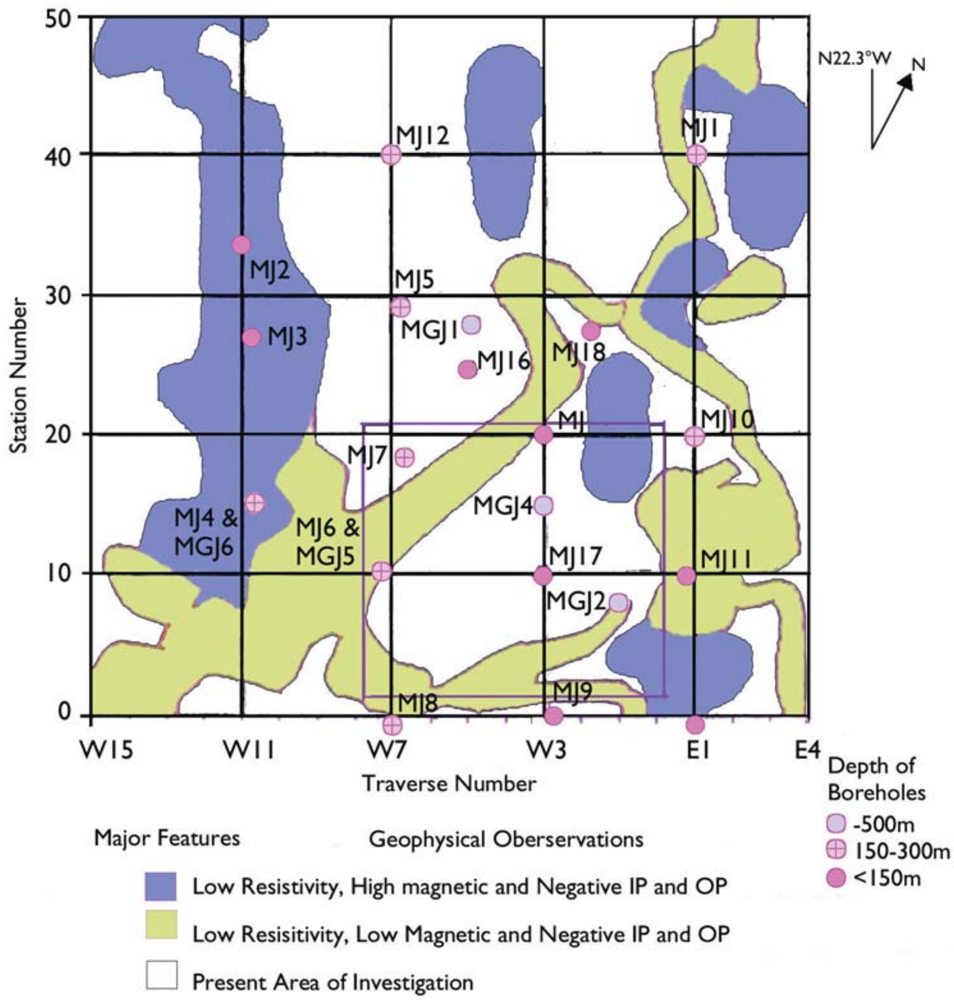

Figure 15.3. **Structural trends infer red through geophysical sur veys and location of boreholes**

56320 Hz. A total of 121 deep-resistivity soundings along 11 traverses were conducted with a traverse spacing of 200 m.

All the collected geophysical signatures were analyzed independently, and profiles and contour maps were prepared. Atotal magnetic field intensity map with a 50 nT contour interval indicated two major trends, in the NS and NW-SE direction. The integrated analysis of geophysical anomalies was attempted by plotting the apparent resistivity contours on the choropleth map of the total magnetic field intensities observed in the area. A comparison of both fields clearly shows the presence of a parallel set of major weak zones, one aligned in a NS and the other in a NW-SE direction. These are the major inferred weak zones along which the basic magma might have intruded the granitic host rock and now appears in the form of basic dikes. These weak zones are interconnected all along the periphery of the grid and are characterized by high magnetic field intensities, thereby suggesting a possible ring dike system in the area.

A quantitative interpretation of the magnetic-field anomalies indicates that the basic bodies are at shallow depths in the southern and northern parts of the study area. The main body in the northern part of the area shows nearly vertical to subvertical dips, while in the eastern part, it is slightly dipping towards the west. Similarly, in the southern part of the area, the bodies are dipping toward the north. These inferences support the existence of a ring dike system in the area as shown in Figure 15.4. By integrating all the results obtained so far from these studies, we have drawn geological cross sections along all the traverses to show the top weathered

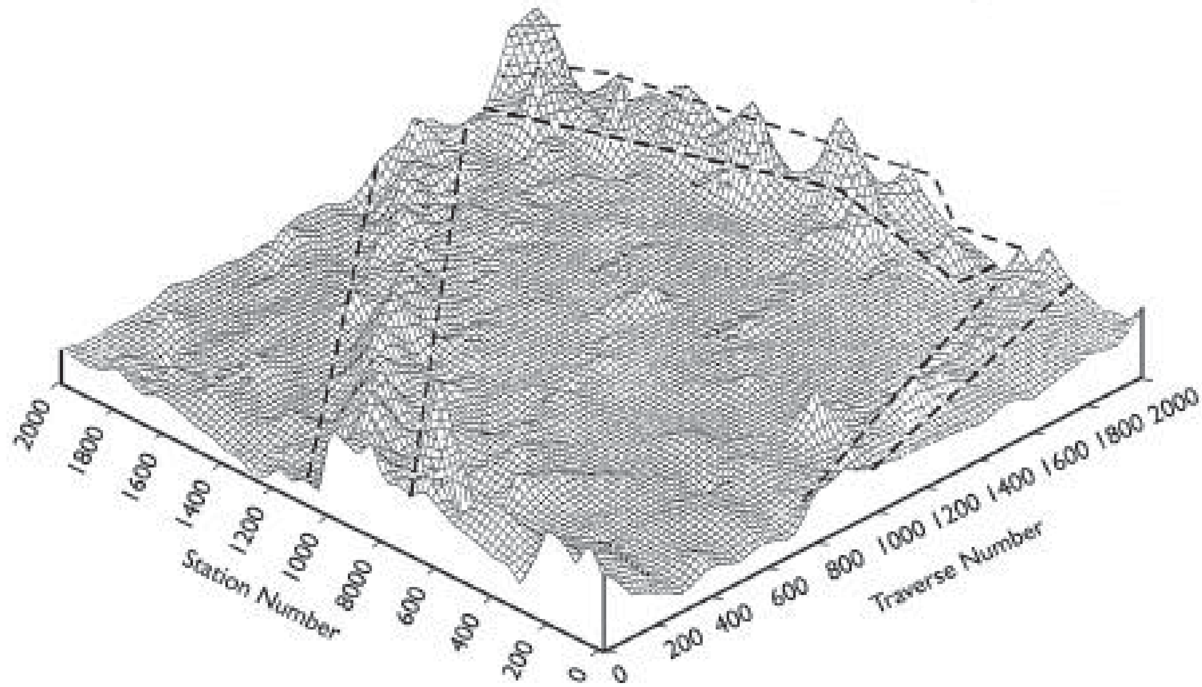

Figure 15.4. Sketch showing three-dimensional geometry and orientation of basic dikes

granite zone, fresh granite zones, and fractured zones along with the positions of dikes, both in vertical and horizontal layers.

15.4.3. Borehole Drilling

15.4.3.1. Shallow Borehole Drilling

Based on geological mapping and geophysical investigations, borehole drilling locations were finalized, and 18 shallow boreholes of NX size were drilled to a depth of 100–150 m, covering an area of 25 km^2. Detailed core logging for the rock types, joint systems, weathering pattern, and alteration of rocks were carried out, along with core recovery and RQD measurements. Using these data, a 3D diagram was drawn as shown in Figure 15.5, which indicates that the dolerite dike is quite thick (70 m) in one zone, while in others it is very thin.

15.4.3.2. Deep-Borehole Drilling

Further drilling operations were carried out at the site, using deep boreholes down to a depth of 500 m (NX size) to obtain more information about the granite rock mass (as well as the dolerite dikes). The borehole locations were spaced at a distance of 800 m from each other on a linear pattern. As was done in shallow drilling, data pertaining to core recovery, RQD, crystallinity, joint and fracture zones, fillings and dike and sills were recorded in detail.

15.4.4. Petrographic and Geochemical Investigations

Selective rock samples from surface outcrops (as well as cores from boreholes) were examined under the microscope for their petrographic character and geochemical composition, as detailed below.

15.4.4.1. Petrographic Studies

Tonalite
These are grey-colored porphyritic rocks chiefly made up of plagioclase and quartz, with minor biotite and amphiboles. The K-feldspars are as low as 4%. Tonalite is megascopically indistinguishable from granodiorite (see below) and often carries xenoliths of mafic rocks, a feature that, when coupled with late-stage basic dikes, suggests that these rocks evolved in an essentially tectonic environment.

Granodiorite
Fine- to medium-grained grey-colored plutonic rocks exposed in the northeastern part of the study area have been identified as granodiorite. Granodiorite is chiefly composed of quartz and plagioclase, with orthoclase, chlorite, sphene, apatite, biotite and hornblende as minor minerals, and is characterized by a porphyritic texture defined by phenocrysts of plagioclase and quartz in a matrix dominated by fine-grained hornblende.

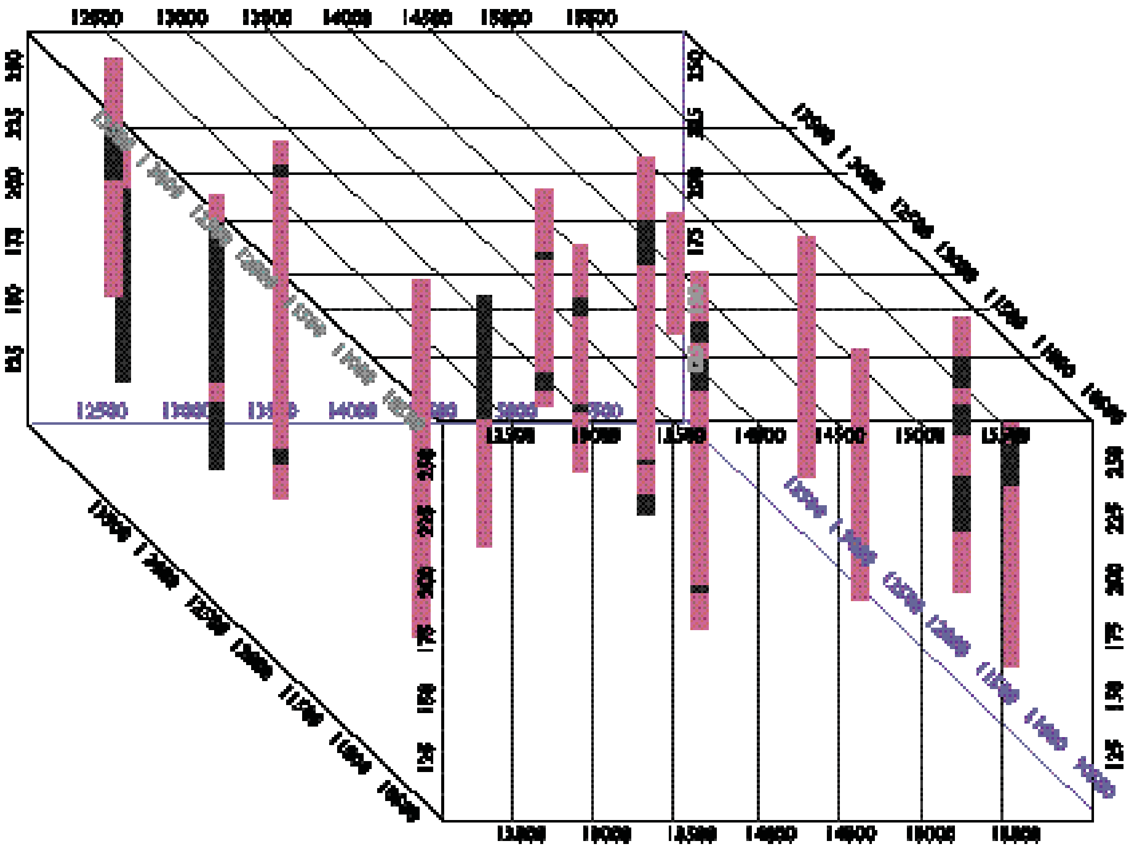

Figure 15.5. **Block diagram showing granite (pink) and basic (black) dikes encountered along the shallow boreholes drilled**

Granite

The greater part of the area is occupied by a variety of granites with varying grain size and texture. Mostly they are coarse-grained pink porphyritic in nature, composed mainly of quartz and orthoclase with biotite and chlorite occurring as minor minerals. Corundum also has been identified in a few samples. Accessory minerals include apatite, zircon, sphene and magnetite. Fine-grained varieties (mainly aplite) have been noticed as veins in the granites.

15.4.4.2. Geochemical Character

Detailed geochemical studies were undertaken on samples from the Jalore Pluton. The chemical composition of the granitoids varies significantly from that of the Jalore granites, which are known to have a distinct anorogenic geochemistry. Granitoids of the study area can be grouped into two distinct assemblages on the basis of their geochemical signatures.

1. *Tonalite-Granodiorite-Diorite-Granite Suite*

   These rocks occupy north, northeastern, and eastern parts of the study area. This suite is characterized by the meta-aluminous nature of member rocks and occupies granodiorite, tonalite and diorite fields in a Na_2O-K_2O-CaO ternary plot. On Harker diagrams, a decreasing slope shown by FeO(T), TiO_2, MgO, CaO and Al_2O_3, and a positive slope of Na_2O and K_2O when plotted against silica, indicate a magmatic origin for this suite, in which magmatic differentiation has played a major role. This is also substantiated by a negative correlation of V, Sr, Co, Cu, Zr and a positive correlation of Rb, Y, and Th with silica. The gradual depletion of the former elements suggests a calc-alkaline affinity for these rocks and is in conformity with an alkali-lime index of 53, indicating an alkali-calc type of magmatism. The suite shows a distinct I-type geochemistry.

2. *Peraluminous Hypersolvus Granites*
   Highly silica- and alumina-rich granites occupying the southwestern and central part of the study area are characterized by a very narrow compositional range; strong depletion of Ba, P, and Ti; and little (or almost no) definite trends (as defined by a cluster of major oxide and trace-element data sets on Harker diagrams). These granites plot in an A-type field and often show an overlap with S-type granites on plots of Rb/100 versus Sr/100 and Na_2O versus K_2O. This is mainly because both of these tectonic types are derived from recycling crustal material, most likely Erinpura-equivalent granites. The Erinpura granites have been identified as syntectonic S-type granites derived by partial melting of crustal material (Gangopadhyay and Lahiri, 1984). However, the granites associated with Malani magmatism have been widely accepted as within plate anorogenic granites. Ample evidence has been noted indicating the possibility of these being S-type granites. The mol Al_2O_3/Na_2O+K_2O values of these granites are almost 1.1, thus lying almost on the boundary of A and S-type granite. The appearance of normative corundum, coupled with low Zr+Nb+Ce+Y values, suggests that some of the granitic rocks earlier grouped as A-type may be S-type equivalents of Erinpura granite generated by Syn-collision magmatism. This rock is being considered as host rock for the geological repository.

15.4.5. Geochronology

Rubidium and Sr abundance (as well as isotopic composition) was determined by the isotope-dilution technique. The mean of a large number of isotopic-standard determinations for Sr was 0.71024 ± 4 (2_{-m}). Samples indicate an average precision of 0.8% for Rb and 0.2% for Sr concentrations. Similarly, the Sm–Nd concentration, by the isotope-dilution technique, was determined along with a direct measurement of the Nd isotope. During this study, the mean $^{143}Nd/^{145}Nd$ was 0.511861 ± 15. Accordingly, different isochrons were plotted that suggest a depleted mantle model age (T_{DM}) of 987 Ma (with $\in_{Nd}$ of + 5.78 for granites), whereas the Rb-Sr isochron gives a regression of 837 ± 9 Ma with an initial $^{87}Sr/^{86}Sr$ of 0.70317 ± 2. Dolerites of the area give a TDM of about 811 ± 2 Ma with ε_{Nd} of + 6.22. This remarkable similarity in Sm-Nd for granite and dolerite suggests a similar mantle source, as complete homogenization between dolerite and granite is extremely slow. Paleo-magnetic studies show post-emplacement thermal events in the area.

15.4.6. Hydrogeological Tests

In one of the deep boreholes drilled at the site, pump tests were carried out to evaluate hydrogeological conditions in the fractured-granite rock mass. The pump test was conducted for over 8 hours, and the well yielded groundwater continuously, with a drawdown of about 2 m. However, drawdowns were noted in neighboring wells and the open pit. This indicated that the rock mass is fractured as a result of dike intrusions, and that the fractures are hydraulically interconnected. Because of the large quantity of water present, studies on aquifer characteristics and the source of water recharge could not be confirmed.

15.4.7. Rock Mechanical Studies

Samples were collected from shallow and deep boreholes to determine rock mechanical properties at normal (28°C) and elevated temperatures of 70°C and 130°C. These properties included uniaxial and triaxial compressive strength, tensile strength, shear strength, Poisson's ratio, Young's modulus, stiffness, cohesion, and friction angle. The results obtained in these studies are shown in Table 15.2. The triaxial compressive strength was also determined at varying confining pressures under the above temperature conditions. When temperature and confining pressure increased, compressive strength was found to decrease.

The jointed rock materials were also characterized with respect to joint roughness coefficient (JRC), joint wall compressive strength (JCS), basic friction angle, normal

Table 15.2. Properties of intact rock samples of pink granite (average values)

Test conditions	Uniaxial Compressive strength (MPa)	Young's Modulus (Gpa)	Poisson's ratio	Tensile Strength (MPa)
28	189.66	67	0.27	12.08
70	212.50	66	0.25	11.45
130	192.50	60.50	0.22	10.59

stiffness, and shear stiffness. The studies conducted for jointed rock materials revealed that the rocks fall in two categories: (1) joints with surface staining and (2) joints altered (slightly) with nonsoftening mineral coating. The JRC was about 4 – 6, and the joints are planar and smooth. The residual friction angle is 28.01°, in the dry condition, which is less than the basic friction angle, 35.32°. Normal stiffness is 23 – 37 MPa/mm, with a shear stiffness of 1–4.5 MPa/mm.

15.5. CONCLUSIONS

Systematic studies using remote sensing, geological and structural mapping, borehole drilling, core logging, microscopic investigations, geochemical studies, geophysical investigations, and hydrogeological and rock mechanical testing of a pluton of Jalore granite (over an area of a few thousand km^2) have paved the way for further investigations.

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Chapter 16

Radioactive Waste Mangement in Italy: A Site for a Low-Level Radioactive Waste Repository and High-Level Radioactive Waste Long-Term Storage

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ABSTRACT . As a result of a 1987 referendum, nuclear energy has been phased out in Italy. In the 30 years of nuclear activity before then (covering electricity production, fuel cycle operation, and applied R&D activity), a considerable amount of radioactive waste of various kinds had been produced. Since 1996, a major effort has been made to provide the country with a repository for low-level radioactive waste (LLW). A near-surface LLW repository (based on a vault concept) is under design. It will contain a national total inventory of around 80,000 m^3 of conditioned waste. The conceptual design for this repository has been completed and submitted to the safety authority for preliminary evaluation. In addition, a long-term-storage system for high-level radioactive waste (HLW) will be located at the same site. The preliminary design of a storage system for vitrified HLW, spent fuel, and intermediate long-lived radioactive waste (ILW) is also under way.

Since 1997, a general site-selection process covering the entire national territory has been going on. An extensive nationwide geographic screening to identify suitable areas for the LLW repository has been completed. This geographic investigation, begun in 1998 and based on Geographic Information System (GIS) methodology, resulted in the development of a National Map of Suitable Areas. Further investigation, using an upgraded GIS methodology and a model for area classification, enabled us to select a number of areas within Italy that are suitable. Those areas are located in regions for which exclusion criteria do not apply. An institutional and financial framework is also being established for dealing with the nuclear legacy. In this report, the LLW repository, the HLW storage system, and the site investigation are briefly described.

16.1. INTRODUCTION

A considerable amount of radioactive waste has been produced in Italy over 30 years of R&D and commercial nuclear activities. Relevant facilities during this period included four power stations operated by Ente Nationale Energia ELettrica (ENEL), the national producer of electricity, and several laboratories and pilot plants (including two pilot reprocessing plants) operated by Ente per le Nuove Technologie l'Energia e l'Ambiente (ENEA), the former Commission for Nuclear Energy of Italy (now called the Agency for Energy and Environment). Since 1990, all the operational nuclear facilities in Italy have concentrated on conditioning of stored waste.

Since 1996, a major effort has been made to provide Italy with a repository for low-level waste (LLW). A near-surface LLW repository (based on a vault concept) is being considered. A total inventory of around 80,000 m^3 of LLW conditioned wastes is anticipated, mostly coming from the dismantling of power stations.

By governmental decision, the LLW repository site will also host a storage facility for unreprocessed spent fuel and vitrified high-level waste (HLW) returned to Italy after reprocessing at Sellafield (United Kingdom). The inventory of HLW is made up of about 8,000 m^3 of solid waste (mainly long-lived intermediate-level radioactive waste [ILW]) and an anticipated 300 canisters of vitrified waste from the reprocessing of spent fuel in the UK and the vitrification campaign beginning in 2005.

The criteria for site selection are based on the LLW repository location. On the institutional side, a special working group was set up by the Ministry of Industry in January 2000 within the Conference State-Regions, a permanent Italian institution created to deal with problems of common interest to the central government and local government administrations. This working group has been charged with identifying and proposing a procedure and methodology for site selection, along with developing the required level of support for the selection. How to implement a volunteer methodology is also being considered. The working group was to have presented its conclusion to the Conference by June 2000. This deadline has now been moved to the end of March 2001. During 2000, the working group has held several hearings with the organizations directly or indirectly involved in waste disposal (ENEA, ANPA, civil protection, universities, etc.).

In 2000, at the Italian National Parliament, the Special Permanent Commission for Waste Management (which covers conventional wastes as well) held several hearings on radioactive waste disposal with all involved parties. The institutional and financial framework for managing and carrying out decommissioning activities was also established and implemented during 2000.

The SOcietà Gestione Impianti Nucleari (SOGIN), the former ENEL(electricity producer) subsidiary, which in 1999 became an independent, state-owned company for managing power station shutdowns, became fully operational in 2000. A new board of directors was appointed in September 2000.

A decree by the Minister of Industry, issued on January 26, 2000, has established plans and procedures for funding back-end activities at the phased-out national nuclear facilities, from waste conditioning to dismantling. The costs incurred will be covered by levying a fee on electricity sales. The fee will be fixed annually by the Authority for Energy, an independent public body, upon presentation (by SOGIN and ENEA) of a detailed business plan. The first of such plans, dealing with fiscal year 2001, was presented to the Authority in September 2000.

16.2. THE LLW REPOSITORY

A near-surface LLW repository, based on a vault concept, is being considered for storing a national total inventory of around 80,000 m^3 of conditioned wastes, mostly coming from dismantling of the phased-out power stations.

As planned, the entire LLW repository includes 10 repository units, for a design capacity of 80,000 m^3 of LLW. Each repository unit contains a row of 9 cells, based on a 1 m thick concrete foundation and covered by a multilayer cap system. The waste packages will be disposed of in these cells (see Table 16.1) within a special disposal container (the "module"). Each cell is planned to house 240 modules. Modules are reinforced concrete containers housing conditioned drums and grouted steel boxes (for large components with low activity). The space between drums and boxes is back-

Table 16.1. Details of the Repository Cell

Cell dimensions	Internal	External
Length	25 m	26 m
Width	13 m	14 m
Height	9 m	10 m

Table 16.2. Details of the Module

Module dimensions	Internal	External
Length	2.75 m	3.05 m
Width	1.79 m	2.09 m
Height	1.37 m	1.7 m
Total module volume		10.84 m^3

filled with qualified grout. Table 16.2 shows the size of the module. The waste that does not require conditioning under the present Italian classification (VLLW)—mainly solid material generated in plant dismantling—will be disposed of in special containers (carbon-steel containers) having the same size as the module.

16.3. THE HLW LONG-TERM STORAGE SYSTEM

The centralized storage system for Category III waste (HLW) is designed for about 300 canisters of vitrified wastes (produced in the reprocessing of spent fuel in UK and in the vitrification campaign planned in Italy starting in 2002), 300 tons of spent fuel stored in dual-purpose casks, and 8,000 m^3 of long-lived ILW.

Spent fuel will be stored in 30 Castor-type casks,

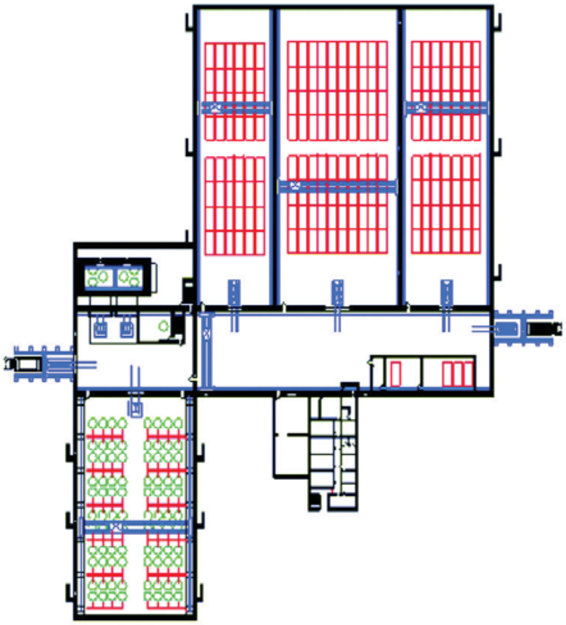

Figure 16.1. Layout of HLW long-term storage system

presently being designed for SOGIN by the German GNB. Ten casks will be used for the vitrified HLW. Storage area for casks is shown in the lower section of Figure 16.1 above. The storage area for ILW long-lived waste, in the upper section of the figure, is provided with dynamic containment. Most of the Italian inventory of long-lived ILW is made of transuranic waste and graphite unloaded from a gas-cooled power station.

16.4. SITE INVESTIGATION

A Geographical Information System (GIS) methodology has been developed to identify suitable areas for the location of the LLW repository, based on exclusion criteria derived from performance-assessment general principles and on a "point count system" developed as a suitability-index calculation for the selected areas. This methodology was implemented in three steps, using the ESRI Arc/Info and Arc/View platforms.

The first step (Level 1 GIS analysis) was carried out using data and maps with 1:500,000 scale resolution, to identify the most favorable regions in Italy for such a site. After this first stage of work and the application of the exclusion criteria, only 9% of the entire Italian territory was considered to be suitable for an LLW repository.

In the second step (Level 2 GIS analysis), a more detailed data acquisition was made (1:200,000–1:100,000 average scale) for regions selected on the basis of the first-step results. This GIS analysis screened out 88% of the remaining areas. Finally, after the second-step screening, less than 1% of Italian territory could be considered potentially suitable for the location of the LLW repository. A third-level GIS analysis is under way (beginning in 2001) on about 300 suitable areas identified in the second stage.

At the same time, a performance assessment (PA) procedure has been set up for four test sites (included within the suitable areas), using an upgraded AMBER code. The PA exercise in the test sites allowed the identification of the scenario and related features, events, and processes.

Exclusion criteria were adopted in Levels 1 and 2 of the GIS analysis. Criteria have been selected by considering the typical geographical, environmental, socio-economic and technical constraints for the vault repository. In particular, islands and areas within 50 kilometers from the national inland borders have been considered unsuitable *a priori* for the location of the LLW repository. Protected areas such as national parks, national and regional nature reserves, resort areas, and areas of natural beauty were also excluded. Socio-economic issues, as well as consideration of the Italian geography, led to exclusion of areas close to towns, roads, and railways, as shown in Table 16.3.

Table 16.3. Location criteria with respect to population and infrastructure

Exclusion Distance, km	Population or Infrastructure
15	> 100,000
10	20,000–100,000
5	10,000– 20,000
3	1,000– 10,000
2	< 1,000
2	(Motorways)
1	(National roads)
1	(Railways)

In Level 2 of the GIS analysis, some exclusion criteria were redefined or added. Specifically excluded were areas with:

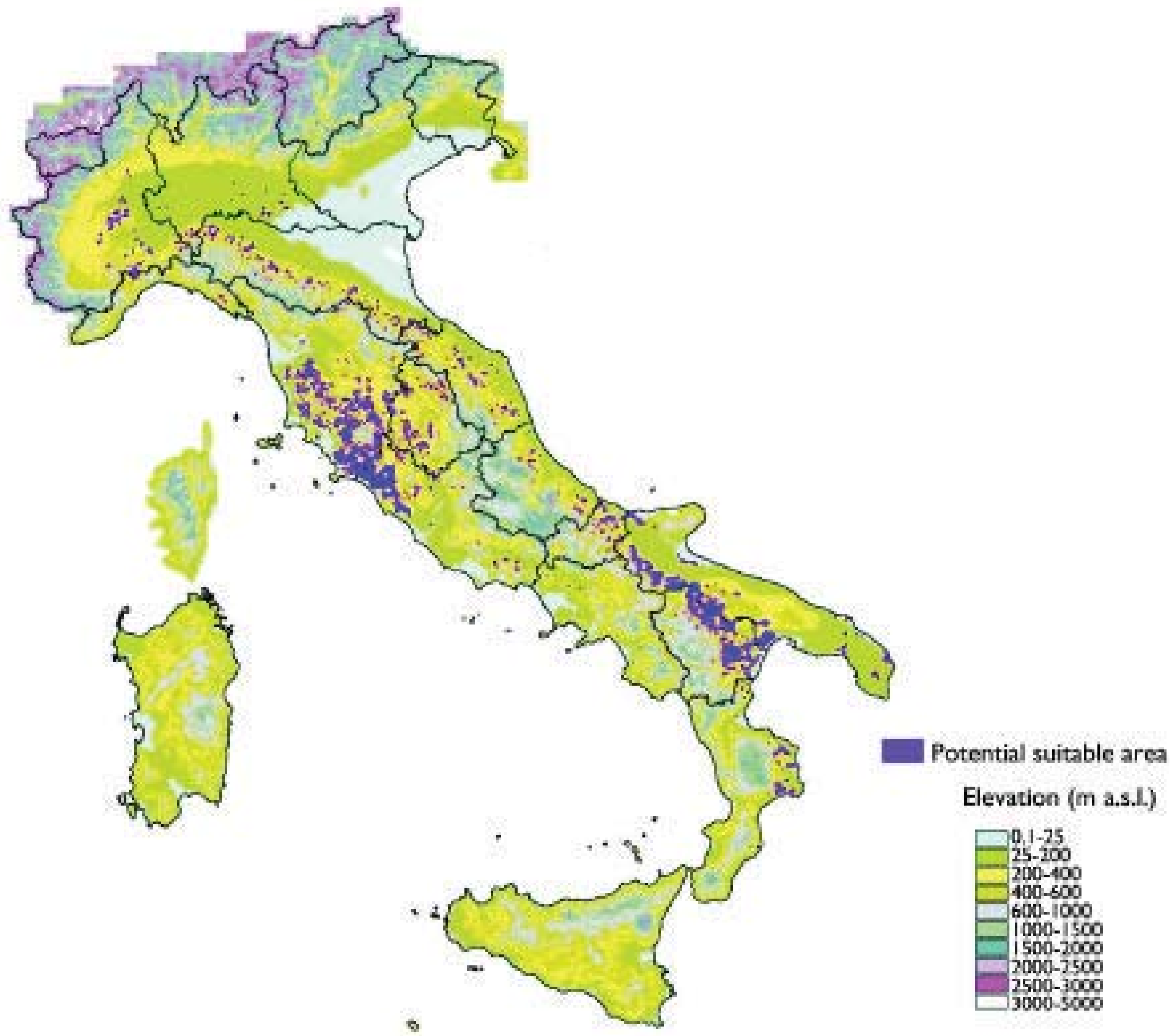

Figure 16.2. Map of Italy showing potentially suitable areas for the LLW repositor y

- Fractured and soluble rocks
- Slope >5°
- Elevation <20 m a.s.l. and >600 m a.s.l.
- Environmental constraints defined by law
- Wood or wet-lands
- Maximum amplified ground acceleration expected in 3,000 years >0.3 g
- Extension <3 km^2.

As result of the Level 2 GIS analysis, the previous 9% of suitable territory within Italy was reduced to about 1%, distributed over 214 areas larger than 3 km^2, as shown in Figure 16.2.

16.5. CONCLUSION

In Italy, the ongoing process for site selection is expected to be concluded by the end of 2002. Operation of the so-called National Site for radioactive waste (including the LLW repository and the HLW storage) is scheduled by 2009. Operational lifetime of the LLW repository will be 12 years, according to present planning. The National Site will include facilities for receiving and conditioning of nonenergetic LLW.

The next two years will determine whether this is a realistic schedule.

Chapter 17

The Japanese High-Level Radioactive Waste Disposal Program

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17.1. INTRODUCTION

As outlined in the overall high-level radioactive waste (HLW) management program defined by the Japanese Atomic Energy Commission (AEC), HLW from reprocessing of spent nuclear fuel is to be vitrified, stored for a period of 30 to 50 years for cooling, and finally disposed of in a stable geological environment deep underground (AEC, 2000a).

In the generic research and development (R&D) phase, the Japan Nuclear Cycle Development Institute (JNC) was assigned as the leading organization responsible for R&D activities. The aim of the R&D activities at that stage was to provide a scientific and technical basis for the geological disposal of HLW in Japan and to promote understanding of the HLW program safety concept, not only within the scientific and technical community but also within the general public. One of the features of the R&D program is that its progress is documented at appropriate intervals, with a view to clearly determine the level of achievement of the program and to identify further R&D issues. As a major milestone, the Power Reactor and Nuclear Fuel Development Corporation (PNC, now JNC) 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 has completed the second progress report (referred to as H12) (JNC, 2000) and submitted it to the AEC in November 1999. The primary objective of H12 (as specified in the guidelines published by the Advisory Committee on Nuclear Fuel Cycle Backend Policy [Advisory Committee] of AEC entitled "Guidelines on Research and Development Relating to Geological Disposal of High-Level Radioactive Waste in Japan" [AEC Guidelines] [AEC, 1997]) is to demonstrate more rigorously and transparently the technical feasibility and reliability of the specified disposal concept and to provide input for future siting and regulatory processes.

After receiving H12, the AEC organized an H12 review team in accordance with AEC Guidelines. As an overall conclusion of the review, the AEC noted in the review report (AEC, 2000b) issued in October 2000 that the technical basis integrated in H12 satisfies the technical requirements prescribed in the Guidelines. In H12, the long-term safety of a repository system is evaluated by a rigorous performance-assessment method that includes a comprehensive evaluation of the uncertainties involved. Despite remaining uncertainties at the generic stage of the R&D program, it was demonstrated that a geological repository would lead to negligible doses calculated to be sufficiently lower than the safety guidelines established in other countries and by international organizations.

In the year 2000, the geological disposal program in Japan moved from the phase of generic R&D into the phase of implementation, based on the progress so far achieved. The "Specified Radioactive Waste Final Disposal Act" was legislated in June 2000, taking account of the technical achievements of H12. Pursuant to the overall HLW management program, the Nuclear Waste Management Organization of Japan (NUMO), with responsibility for implementing the geological disposal of HLW, was established in October 2000 according to the Act. The assigned activities of NUMO include selection of the repository site, demonstration of disposal technology at the site, relevant licensing applications, and construction, operation, and closure of the repository. According to the present time schedule, repository operation will start as early as the 2030s.

This paper describes the Japanese geological disposal program, focusing on recent developments.

17.2. THE JAPANESE DISPOSAL CONCEPT : MAKING THE SAFETY CASE

The concept of geological disposal in Japan is similar to that in other countries: it is based on a multibarrier system that combines the natural geological environment with engineered barriers. The approach for development of a disposal system concept in the generic phase has been to consider the wide range of geological environments throughout Japan without targeting either a specific type of rock or a specific siting area. However, particular consideration is given to the long-term stability of the geological environment, taking into account Japan's location in a tectonically active zone. Because of Japan's complex geology, an engineered barrier system (EBS) with sufficient margins in its isolation functions to accommodate a wide range of geological environments was developed.

The major component of the disposal system's function to serve as an overall barrier is borne by the near field (the EBS and a limited volume of the surrounding host rock), while the remainder of the geosphere serves to reinforce and complement the performance of the EBS. The disposal concept is therefore to construct an EBS that, in a stable geological environment (Figure 17.1), provides sufficient margins in its long-term performance for isolation of the waste applicable to a range of potential geological conditions and their future evolution. The reference layout of the EBS involves either axial emplacement in a horizontal tunnel or vertical emplacement in a pit; in both cases, vitrified waste is encapsulated in a thick steel overpack surrounded by highly compacted bentonite.

This concept assumes that major disruptive events can be excluded by site selection. Stable geological environments identified as having favorable characteristics for disposal-system construction provide the basis for repository design. If the safety functions of the geological disposal system are assured, minor amounts of radioactivity released from the EBS in the far future will further decay, and concentrations will be reduced by

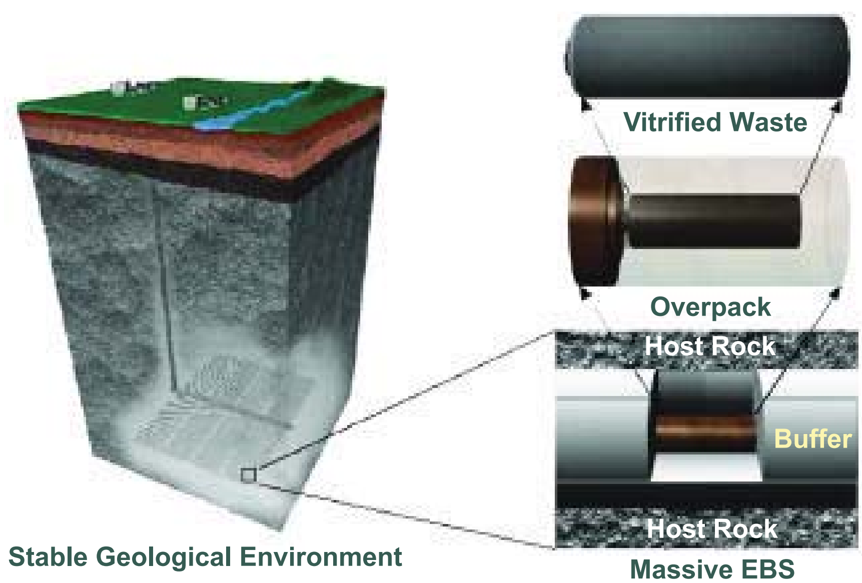

Figure 17.1. HLW disposal concept

dilution during the long migration period in the geosphere. HLW disposal can be realized in such a way that no significant detrimental influence is exerted on either humans or the environment (Umeki, 2000).

To support this safety concept, R&D activities must focus on the natural system attributes that optimize EBS performance, including relative tectonic stability, low groundwater flux, favorable geochemistry, and a low risk of disruptive events. Performance assessment is utilized to illustrate the robustness of the intrinsic safety features of the repository design, taking alternative future evolutions of the system into account.

To build confidence in the technical feasibility of the repository system, H12 was required to provide more detailed analyses to demonstrate how the intrinsic safety functions of the repository design, supported by appropriate quality assurance procedures and careful site selection, could provide a robust safety case. Demonstrating the repository system concept includes:

- Specifying the existence of geological environments that are stable over appropriate time scales and provide favorable conditions for EBS performance and for the radionuclide retardation function of the geosphere
- Illustrating an appropriate design for containment and retardation of radionuclides in the EBS for a wide range of geological environments.

It was also required that H12 provide results of performance assessment to estimate the long-term reliability of the repository system for a number of different geological settings.

To meet these requirements, R&D work since H3 focused on development of detailed and realistic near-field models and on improving the understanding of key processes and corresponding databases, taking into account a wide range of geological conditions. In JNC's R&D program, three major areas of research were set up following the AEC Guidelines. These are (1) evaluation of the geological environments for hosting a repository, (2) engineering technology for a repository and EBS design, and (3) performance assessment of the disposal system.

In the following section, the technical achievements presented in H12 are summarized.

17.3. ACHIEVEMENT IN RESEARCH AND DEVELOPMENT

17.3.1. Existence of Geological Environments Suitable for a Repository

Important natural phenomena that could influence the long-term stability of the geological environment include fault movement, volcanic activity, uplift and denudation, and climatic and sea-level changes. The occurrence of these natural phenomena and the extent of changes in the geological environment caused by them were investigated over time scales of several hundred thousand years or more, based on field studies in regions where the history of these phenomena could be observed. These studies showed that the location of sudden localized phenomena, such as volcanic activity and major fault movement, can be well specified and their effects can be avoided by selecting an appropriate disposal site. For example, tracing back the history of volcanic activity in the Quaternary shows that the locations where such activity occurred are restricted to distinct regions and that there is little change in these locations. In addition, the direct effects of volcanic activity that could significantly influence the repository performance were expected to be located at least a few tens of kilometers from the activity centers (Figure 17.2).

On the other hand, gradual phenomena such as uplift and denudation or climatic and sea-level changes are more ubiquitous. It is, nevertheless, possible to estimate their future trends and potential effects by extrapolating data obtained from the field studies. The conclusion from the studies is that it is possible to select a sufficiently stable environment for geological disposal.

The characteristics of the geological environment that are important in terms of its barrier function include groundwater flow rates, rock permeability, the geochemical features of groundwater, the thermal and mechanical properties of rock formations, and solute transport properties. Extensive data were obtained on these characteristics, particularly from geoscientific studies carried out in the Tono area and the Kamaishi mine. The data from measurements in Tono and Kamaishi refined our understanding of processes and enhanced the credibility of the databases.

From the revised databases, representative datasets were selected for crystalline and sedimentary rocks. These datasets were used for subsequent studies to determine appropriate repository designs and characteristics of the engineered barriers. The geological data also provided essential input to the performance assessment, for evaluating potential doses resulting from both expected evolution "groundwater scenarios" and perturbed "isolation failure scenarios."

17.3.2. Demonstration of Repository Design and Engineering Technology

The design requirements for the EBS and the general disposal facility were clarified based on utilization of currently available technology, taking economic aspects into consideration. Since the publication of H3, more reliable supporting data have been obtained from demonstration tests on both laboratory and engineering scales (carried out at JNC's ENTRY facility and other sites). Design requirements have been reviewed, the analytical design tools have been improved, and the design database has been extended to provide a better understanding of the barrier functions of the EBS. The practical feasibility of designing and emplacing the EBS and constructing the disposal facility was examined for a wide range of rock properties. This has provided a basis for a design of the EBS and disposal facility that is sufficiently flexible such that it could be tailored to the specific characteristics of a potential disposal site.

The basic material proposed for the overpack is carbon steel, because it appears to satisfy the design requirements and has been successfully used as a structural material. Composite overpacks using titanium or copper as a corrosion-resistant layer were also investigated as alternative designs. Overpack thickness can be estimated based on corrosion resistance, pressure resistance, and radiation shielding function (assuming an overpack lifetime of 1,000 years). The total thickness of the overpack is specified by adding the corrosion allowance (40

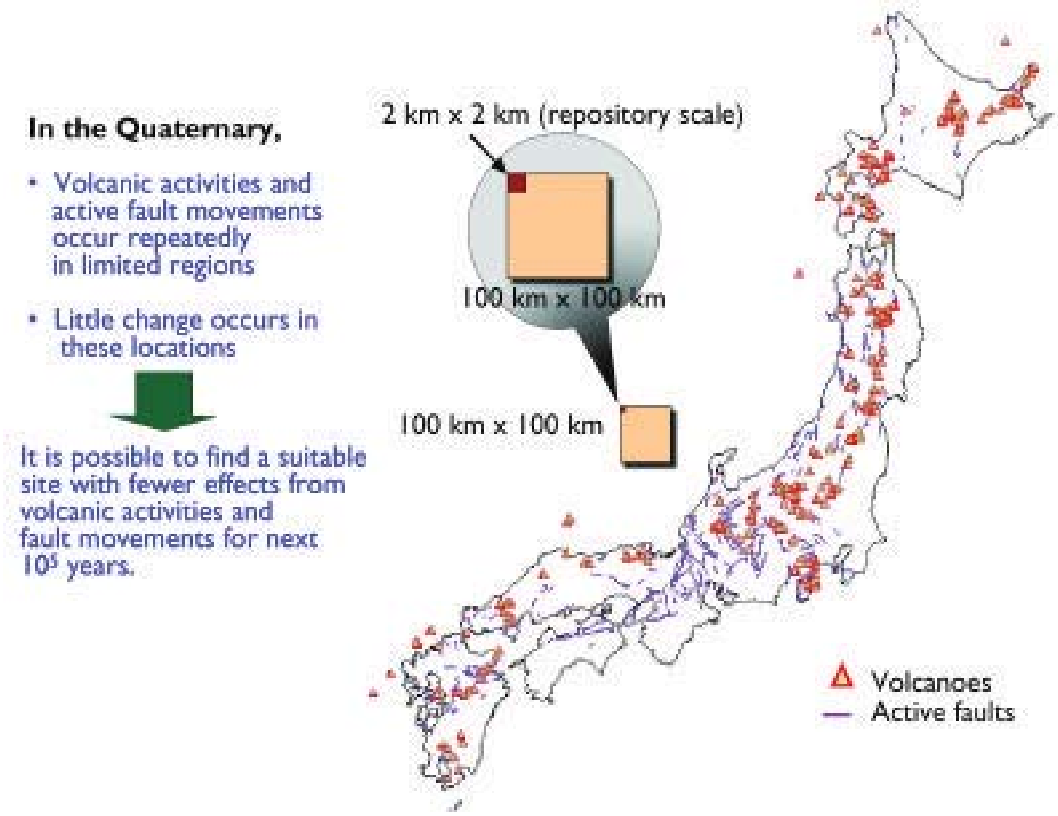

Figure 17.2. Stability of the geological environment—Distribution of volcanoes and active faults

mm) to the required radiation shielding thickness (150 mm). A uniform thickness of 190 mm for both the end and cylindrical shell sections can be adopted for both hard and soft rock systems.

The studies carried out indicated that pure bentonite could perform the required buffer functions. The quartz/sand mixing ratio and dry density that could meet the functional requirements of the buffer were also studied. It is considered reasonable to set the buffer thickness in the range of about 40 cm to about 70 cm from the viewpoint of the stress-buffering function.

If the thickness of the buffer material is set at 70 cm with performance margins for a dry density of 1.6 mg m^{-3}, it has to be verified whether this will also satisfy the other design requirements, such as self-sealing ability, thermal conductivity (buffer material not subject to thermal degeneration), workability, and colloid filtration. Of these requirements, thermal conductivity and workability must be satisfied with regard to dry buffer material (i.e., before water permeation and swelling). Self-sealing and colloid filtration capabilities must be met with respect to the buffer material after swelling.

Based on these studies, the thickness of both the overpack and the buffer material could be reduced by approximately 30% compared with the specifications in H3 (Figure 17.3). This leads to a reduction of around 50% in the total volume of EBS materials. Bentonite mixed with quartz sand was selected as the buffer material, bringing about a reduction in costs without compromising performance.

The mechanical stability of tunnels was investigated based on data obtained from relevant geological environments. Rough estimates were then made of the depth range in which construction of the disposal facility is feasible. In addition, a design concept for efficient emplacement of the vitrified waste and layout of the tunnels was developed, based on thermal analyses (Figure 17.4). It was shown that construction of the disposal system, emplacement of the waste forms, and backfilling of the tunnels can be realized using currently available technologies or technological advances expected in the near future.

This combination of engineering design studies shows that the intrinsic safety features of the repository design can be maintained and balanced against economic concerns. In addition, demonstration of feasibility on emplacement gives confidence that the repository can be safely constructed and that the resulting construction will be of very high quality (largely free of defects in workmanship that would affect its barrier functions). It is expected that the applicability of the engineering technologies will be further evaluated as part of the experiments planned for deep underground research facilities in Japan (see Section 17.5.3).

17.3.3. Performance Assessment to Illustrate Robustness of Barrier Functions

While the main safety functions of the repository are intrinsic to the design and location within a deep geological system, understanding the level of robustness of the proposed system in various geological conditions or under changes in system conditions can only be evaluated through studies of long-term repository performance. An assessment method has been developed and applied in H12 to evaluate the safety functions and the level of robustness in the proposed system under various

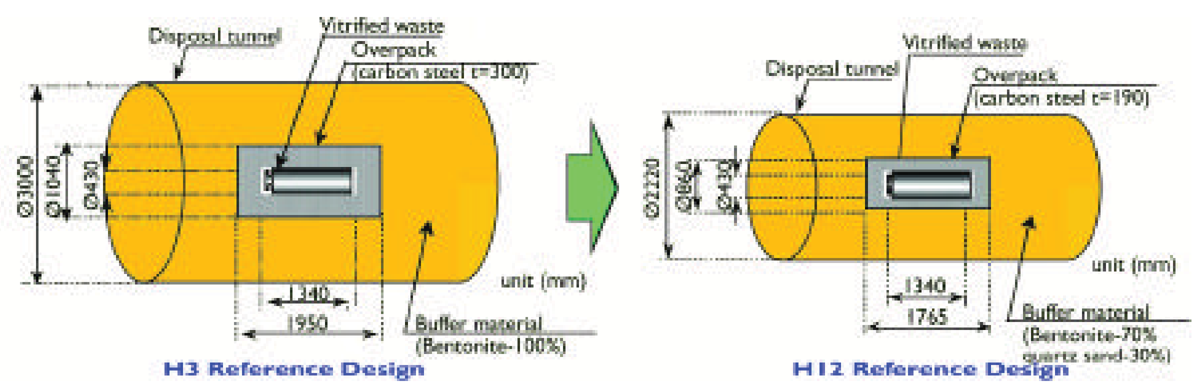

Figure 17.3. Evolution of the EBS design

conditions.

Scenario Development

To reduce the risk of overlooking potentially important scenarios, a systematic methodology has been developed and applied in H12. A comprehensive list of features, events, and processes (FEPs) was first developed by collating the FEP lists developed in other projects [e.g., Office of Economic Cooperation and Development/Nuclear Energy Agency (OECD/NEA), 1997; Nagra, 1994)]. The only scenarios modeled in detail in H12 are "groundwater scenarios" (scenarios in which moving groundwater provides the pathways for transfer of radionuclides from the repository to the surface environment). These include:

- A Base Scenario, in which external events and processes such as natural geological and climatic phenomena, initial defects, and future human activities are excluded
- A set of perturbation scenarios, in which the potential impacts of external events and processes are examined.

A Reference Case was defined for the Base Scenario, incorporating a particular set of geological characteristics, design features, model assumptions, and parameter values. The nuclide transport pathways considered in the Reference Case are illustrated in Figure 17.5. Alternative cases were also defined for the Base Scenario (with alternative geological settings, design features, model assumptions, or parameter values) and for perturbation scenarios.

FEPs that could generate "isolation failure scenarios" (scenarios in which the human environment is affected because of the physical isolation of the waste being compromised), such as human intrusion and scenarios associated with natural phenomena, were screened out (e.g., on the basis that they could be excluded by siting). Nevertheless, some "what if?" analyses have been carried out to illustrate the magnitude of potential consequences, and thus the importance of siting the repository in a suitable environment.

Comparing the scenarios considered in H12 with those considered in recent safety assessment reports in other countries (e.g., Nagra, 1994; Vieno and Nordman,

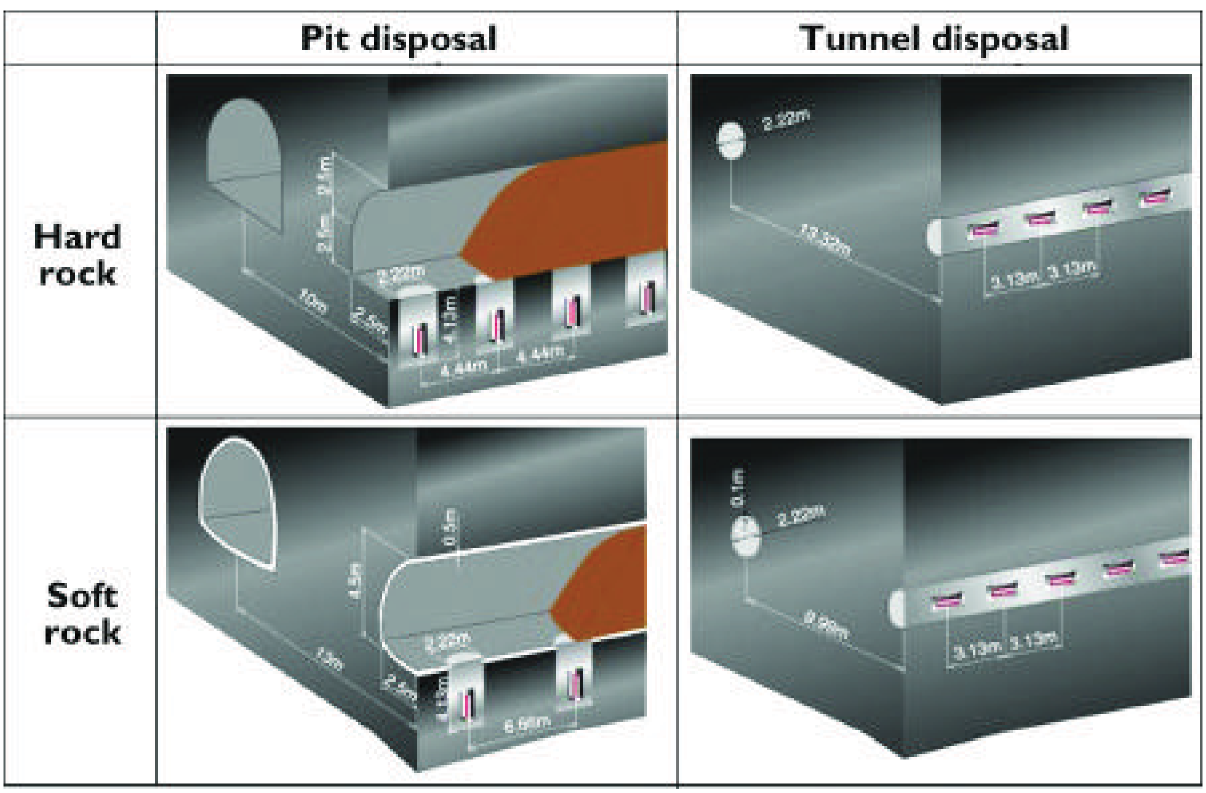

Figure 17.4. Emplacement of the vitrified waste and layout of the tunnels

1999), it was concluded that no significant scenarios relevant to disposal in Japan have been overlooked.

Development of Models and Databases
Based on a list of feasible scenarios, models simulating relevant phenomena in detail, together with associated databases, were established to quantify selected scenarios. Models were developed to simulate the evolution of the EBS and subsequent radionuclide migration in the rock surrounding the buffer material. These models are more detailed and realistic than those used in the H3 assessment and thus improve our understanding of key processes. The same can be said of the corresponding databases.

Models and datasets for near-field and geosphere modeling have been derived from, and tested against, the results of a laboratory and field experimental program in Japan. In particular, use has been made of engineering-scale experiments at the ENTRY facility, experiments using radionuclides at the QUALITY facility, and geoscientific investigations (mainly in Tono and Kamaishi). Extensive use has also been made of international scientific literature and international validation projects (e.g., OECD/NEA and SKI, 1994).

The main transport model used to represent the EBS performance was based on one-dimensional, diffusive transport with linear, reversible, and instantaneous sorption. Shared solubilities and precipitation of each radionuclide are also accounted for during actinide chain-migration through the buffer. The lifetime of the overpack is assumed to be 1,000 years and the long-term dissolution rate obtained experimentally for glass dissolution was used at the waste glass-bentonite interface. Radionuclides are assumed to be congruently dissolved with glass and limited in their release by solubility at the glass surface. The radionuclides released at the outer boundary of the buffer are assumed to be instantaneously mixed within the excavation-disturbed zone.

Several earlier safety assessments in Japan and abroad have considered a single fracture or channel to be representative of all transport paths within the host rock (e.g., Nagra, 1994; SKB, 1992; PNC, 1992). To assess the performance of the host rock surrounding the reposito-

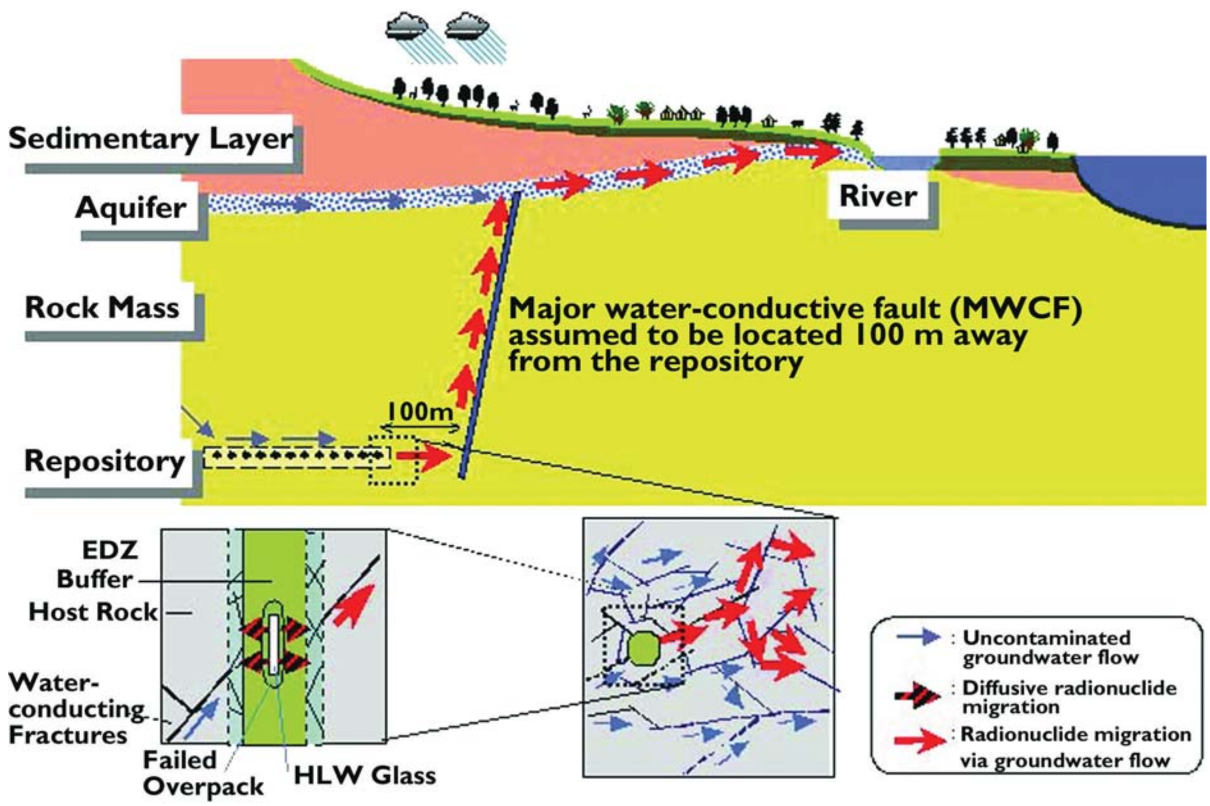

Figure 17.5. Groundwater scenario reference case and conceptual models for radionuclide migration

ry, H12 considered transport along a set of representative channels, taking into account the heterogeneity of real fractures and channels with respect to transmissivity. In this one-dimensional multiple-pathway model, the distribution of transmissivities was discretized, with each model pathway representing a set of channels of similar transmissivities (Figure 17.6).

Advection and dispersion, matrix diffusion, sorption onto surfaces within the rock matrix, and radioactive decay were taken into account in the modeling of transport within a single channel. It has been confirmed by numerical experiments that the multiple-pathway model conservatively approximates nuclide transport in a more complex, stochastically generated three-dimensional fracture network.

Radionuclide releases from the various canisters were assumed to flow towards a single, major water-conducting fault (MWCF) located downstream 100 m from the repository. All radionuclides released from the repository were assumed to migrate upward through the MWCF to a shallow aquifer, which in turn discharges to a river. Significant dilution occurs as a small amount of groundwater from the aquifer enters the river. Retardation in the MWCF is included in the performance assessment, but is conservatively ignored for the aquifer.

A different approach was taken to biosphere modeling. No attempt was made to model the evolution of the surface environment and the lifestyles of future generations, because of uncertainties that are inherently irreducible. Rather, certain sets of assumptions were made about these aspects of biosphere modeling, giving rise to stylized representations of the biosphere for dose calculations ("Reference Biospheres," e.g., BIOMASS, 1999). The biosphere model represents the components of the surface environment using compartments between which fluxes of material (solid/water) and radionuclides are defined by transfer factors. A range of exposure pathways through which such radionuclides could enter the food chain, along with uptake and concentration factors, was also defined. The resulting dose (from both ingestion and external irradiation) to a hypothetical critical group was then calculated. Parameters describing the processes in this system were based on

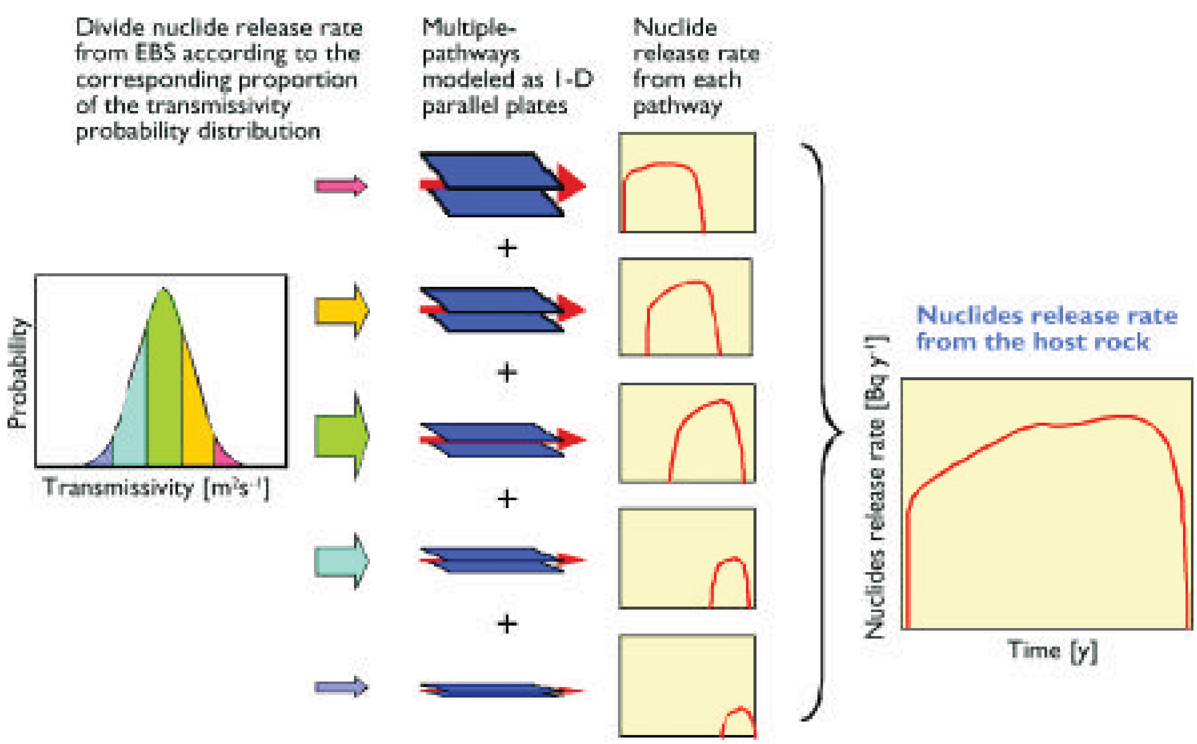

Figure 17.6. Conceptual illustration of a one-dimensional multiple pathway model

estimates of present-day environmental and lifestyle conditions.

Assessment Results

The models mentioned above were linked together in a safety-assessment-model chain, which allows the performance of the entire geological disposal system to be assessed. The Reference Case was analyzed using this model chain for a repository containing 40,000 vitrified waste packages. The calculation result for the Reference Case indicates that sufficient containment of radionuclides can be achieved by the EBS and the near-field host rock, provided that the groundwater flow rate is reasonably low (Figure 17.7).

Model and data uncertainty cases were analyzed for the Base Scenario by considering a wide range of alternative models and parameter values. In addition, a number of calculation cases have been conducted for scenarios in which the geological disposal system is perturbed in the future by natural events and human activities. Based on the understanding of system sensitivity acquired through these calculations, a set of cases focusing on the key factors has been analyzed for self-consistent combinations of alternative geological environments and designs. Owing to uncertainty effects in models and data, alternative disposal systems, and unlikely disruptive events, calculated values of the maximum annual dose vary significantly. However, none of them exceeds the radiological protection levels proposed in foreign regulatory criteria or guidelines (100–300 μSv y^{-1}).

17.4. BUILDING CONFIDENCE IN GEOLOGICAL DISPOSAL

Confidence building is a key issue in promoting the geological disposal program. In this section, the confidence-building activities that have been carried out so far are summarized.

17.4.1. PEER REVIEWS FOR H12 DOCUMENTATION

The H12 was reviewed in a stepwise manner during its documentation. The draft of H12, in Japanese, was prepared based on comments and discussions concerning

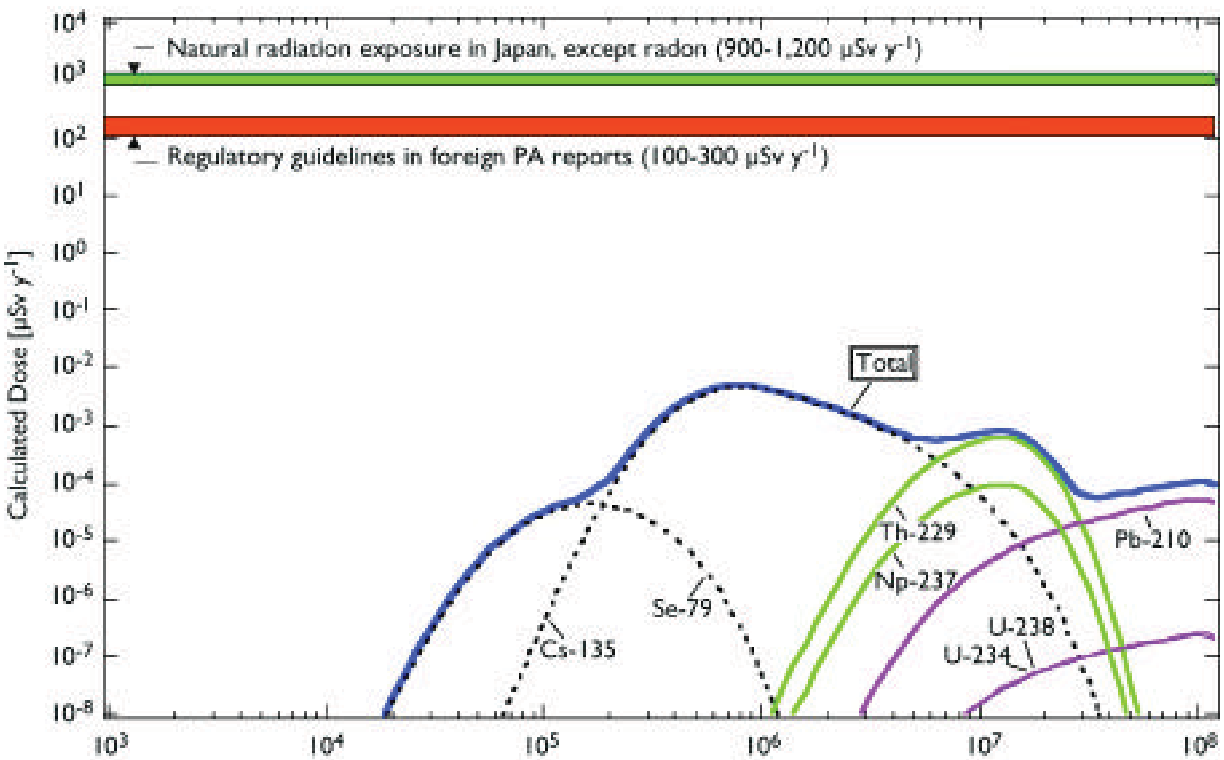

Figure 17.7. Reference Case dose evaluation. The repository is assumed to contain 40,000 vitrified waste packages, and it is also assumed that all packages fail 1,000 years after disposal.

the preliminary draft, as well as on the results of research and development obtained after the publication of the preliminary draft. The draft consists of a project overview report and three supporting reports that cover the three major fields described in Section 17.2 above. These documents were submitted to the Advisory Committee and opened to the public on April 21, 1999, to further solicit comments from Japanese experts in various fields. It was also required in the AEC Guidelines that to assure quality, the draft H12 would be reviewed by international experts before submission of final reports to the Japanese Government. In compliance with the AEC Guidelines, JNC approached the OECD/NEA to carry out an independent, international peer review of the draft H12 Project Overview Report.

Taking account of the review comments made by these reviews, JNC finalized the H12 and submitted it to the AEC on November 26, 1999.

17.4.2. Assuring Transparency and Traceability

As one of the quality assurance activities for H12, a centralized information management database system called "Internet Library" was developed to increase transparency and traceability of H12 documentation (Figure 17.8). The Internet Library is designed to support the technical basis of H12 and consists of over 1,000 technical memoranda summarizing literature referred to in H12, and approximately 300 sets of numeric data used for calculations in H12. The Internet Library is intended to be available through the JNC web-site at http://j2kyrp01.jnc.go.jp/db_2krp/cgi-bin/tbLogin.cgi, so that anyone can check and trace arguments used in H12. The Internet Library is also designed to function as a bidirectional communication tool to answer questions or comments from the audience on H12 through the Internet. It was put into operation at the end of November 1999, just in time for the publication of H12.

17.4.3. Development of Presentation Technique

To promote public understanding of the H12 safety case, JNC developed a special demonstration tool named "Geofuture21" (Figure 17.9). It has been in operation at the JNC Tokai PA center since December 1999. In the Geofuture21, people can enter a virtual repository extending 1,000 m underground by combination of scientific simulation, 3-D visualization, and a motion system, where they encounter and experience an earthquake deep underground of magnitude 7.5 on the

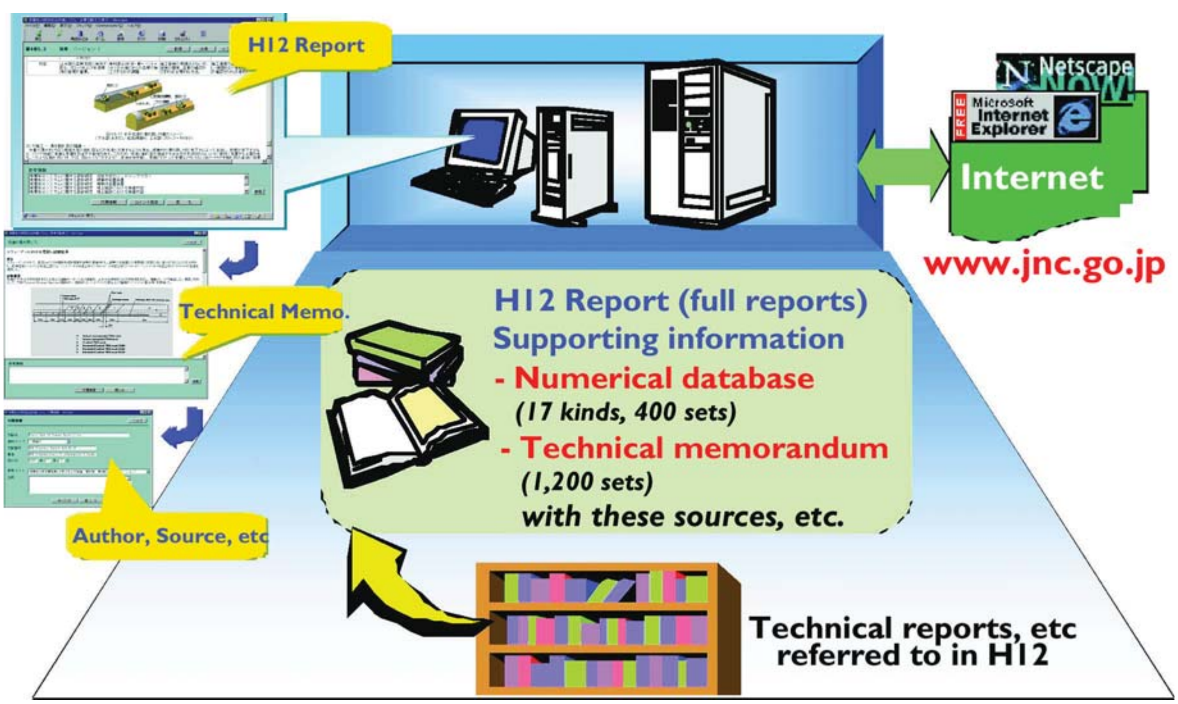

Figure 17.8. Configuration of the Internet Library for H12 report

Richter scale in comparison with that on the ground surface. Then they can observe the behavior of the EBS (such as the swelling of bentonite and degradation of overpack). People can also witness a nuclide movement through the bentonite buffer. About 90% of over 12,000 visitors so far responded to the questionnaire that they could understand geological disposal quite well.

17.5. NEW FRAMEWORK FOR THE PROGRAM

17.5.1. Implementation

The "Specified Radioactive Waste Final Disposal Act" was legislated in June 2000. The Act specifies the overall implementation scheme and defines the roles and responsibilities of the Japanese government and relevant organizations including NUMO, the funding management organization [i.e., the Radioactive Waste Management Funding and Research Center (RWMC)] and the owners of power reactors (Figure 17.10).

Under the Act, the Japanese government (i.e., Ministry of Economy, Trade and Industry [METI]) is responsible for setting the basic policy and final disposal plan for a 10-year term every 5 years. NUMO and RWMC are supervised by METI. NUMO is responsible for planning and conducting site selection (followed by site characterization at the selected site) and the relevant licensing applications for repository construction, operation, and closure. As the first step of the siting process, NUMO will announce an overall procedure for selection of potential candidate sites, followed by identifying siting factors and a repository-concept catalogue to be provided to all municipalities in Japan (Figure 17.11). The repository-concept catalogue consists of a set of repository concepts developed for patterns of geological environments expected in the potential candidate sites, taking account of siting factors. The performance of individual repository concepts will also be overviewed in the catalogue. This catalogue is aimed at providing information for the discussions within municipalities by increasing their understanding of the planned repository.

Potential candidate sites will be selected based mainly on information in the open literature, with particular atten-

Figure 17.9. Tour of the virtual repository constructed by a combination of scientific simulation, 3D visualization, and a motion system (Geofuture21)

tion to the potential long-term stability of the geological formation. Candidate site(s) are then selected by surface-based investigations, including boreholes, carried out to evaluate characteristics of the geological environment at the potential candidate sites. At the final step, detailed site characterization will lead to selection of a final disposal site. Investigations at this stage are mainly conducted in an underground facility constructed at the candidate site(s). The series of siting processes, carried out by NUMO, will be supervised by METI. METI will call for opinions from the governor and leaders of concerned communities prior to site selection. These opinions shall be taken into account in the final disposal plan.

Particularly in the siting procedure, it is important to promote public understanding of geological disposal and to obtain trust. Thus, the site-selection decision-making process must be kept transparent. NUMO will make available a variety of information obtained from their siting activities through (for example) the publication of documents and Websites, and will provide opportunities for people living near the (potential) candidate sites to voice their opinions. The government is also responsible for publishing various types of information on final disposal and for clarifying its policy to ensure the safety of final disposal.

As producers of HLW, the owners of the nuclear power plants are responsible for sharing the costs of final disposal. To cover these costs, they are required to make contributions to a disposal fund in accordance with the amounts of electricity generated. The budget for the NUMO program will be authorized by METI and assigned from the fund. Management of the funding system is conducted by RWMC and its activities are also supervised by METI. The total cost of disposal is currently estimated by the Advisory Committee for Energy, MITI (Ministry of International Trade and Industry, now METI) (MITI, 1999) at approximately 3 trillion yen (corresponding to 0.13 yen/kWh) for a repository with 40,000 waste forms.

17.5.2. Regulatory Aspects

The Advisory Committee on Radioactive Waste Safety Regulations of Nuclear Safety Commission of Japan (NSC) has been carrying on discussions about safety regulations for the geological disposal system referring to the output from H12. Taking public response into account,

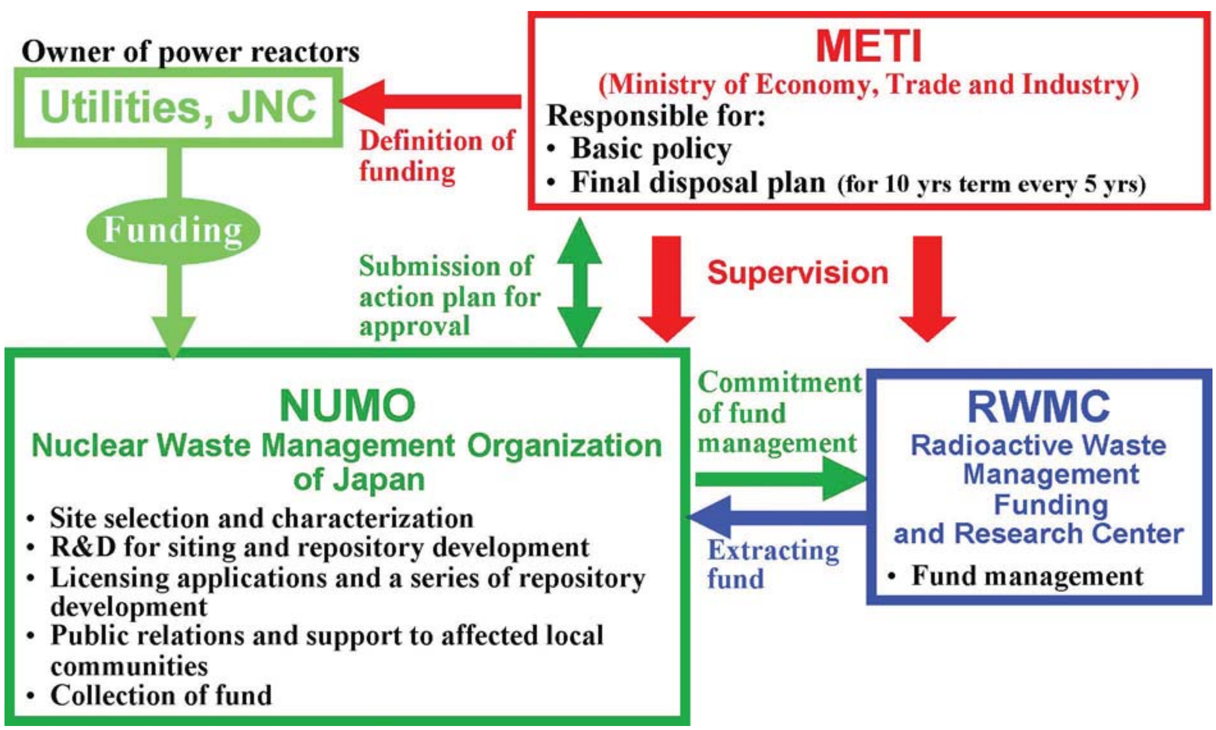

Figure 17.10. Frame work of implementation scheme

this group published its first report on safety standards for HLW disposal in November 2000 (NSC, 2000).

The report mainly specifies safety fundamentals, guidelines for site selection, basic considerations for safety assessment, and management of the disposal site:

- Regarding safety fundamentals, the safety case should be based on the intrinsic safety features of the system provided by appropriate siting and design, coupled with appropriate safety assessment to illustrate the robustness of the overall barrier functions.
- Guidelines for site selection specify favorable geological conditions: a stable geological environment with no underground natural resources of economic value at the time of site selection.
- It is stressed that two types of scenarios should be developed for safety assessment, namely a groundwater scenario and an isolation failure scenario.
- Concerning management of the disposal site, it is emphasized that a QA system for repository construction should be established. The report also emphasizes the need for monitoring during repository construction and operation to confirm the baseline conditions for safety assessment, as well as the need for retrievability over the period during which assessment of repository safety will be confirmed, taking into account such monitored information.

According to the report, NSC is planning to further issue the safety guidelines for license application prior to potential disposal-site selection.

17.5.3. R&D Activities

Concerning R&D in the implementing phase, AEC specifies its framework in the revised National Guideline issued in November 2000 (AEC, 2000a). According to the guideline, the R&D program is now being reorganized to conform better to the newly established implementing scheme. NUMO is responsible for conducting R&D for safe implementation of the repository with improved technology from economic and practical perspectives. The Japanese government and relevant organizations should proceed with R&D for establishing the safety-regulation framework, other fundamental issues related to safety assessment, geoscientific studies, and improving repository technology from the viewpoint of increasing confidence. With a substantial amount of experience and expertise in this field, the JNC continues to bear the responsibility for R&D activities aimed to enhance the reliability of repository technology and to establish safety-assessment methodology and a relevant database.

Among these, the demonstration of a site-characterization methodology from the ground surface in two ongoing URL projects (at Mizunami and Horonobe) is essential for supporting NUMO's program at an early stage. Basic studies and experiments to be conducted in ENTRY and QUALITY will contribute to better understanding of the observed phenomena in two URLs (Figure 17.12).

17.6. CONCLUDING REMARKS

Throughout the H12 project, a tremendous amount of experience has been accumulated, both in technical and

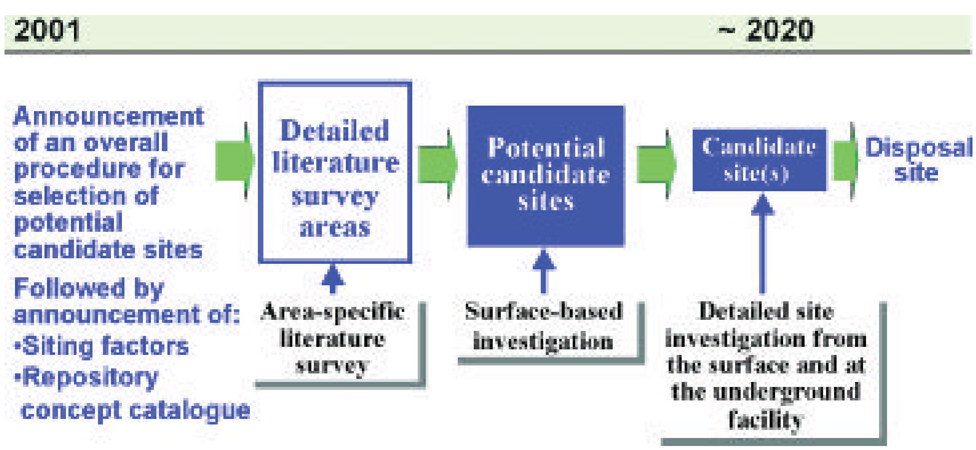

Figure 17.11. Stepwise site-selection procedure

societal domains. Both these domains are integral parts of the inventory for our future activities. Lessons learned from these experiences can be summarized as follows:

- Stakeholder confidence is the basis for public trust.
- A robust safety concept, based on a combination of an appropriate geological environment, an effective engineered system, and a reliable safety assessment is essential, but may not always be enough, by itself, to justify stakeholder confidence.
- An independent, technically competent regulator, working as a bridge between stakeholders and the repository-implementing organization, may help to gain increased public trust in a fair, equitable, and safe process.
- Dialogue is desirable to identify stakeholder concerns and to increase public trust in the disposal concept in advance of starting the site-selection process.

Taking account of lessons learned from the generic phases and the newly established institutional framework for the geological disposal program, we recognize that the following aspects are of prime importance for moving forward:

- Assuring traceability and transparency of the safety case
- Evaluating options such as retrievability and extended institutional control
- Demonstrating the repository technology developed in H12
- Using an incremental approach as a basis for siting and repository development.

Japan has been active in promoting international cooperation in connection with its R&D program, based on both bilateral and multilateral approaches. Such collaboration has proved to be very valuable in improving the R&D structure by identifying areas of strength and weakness, and also by generally enhancing R&D credibility. International cooperation is obviously another key element in the future program.

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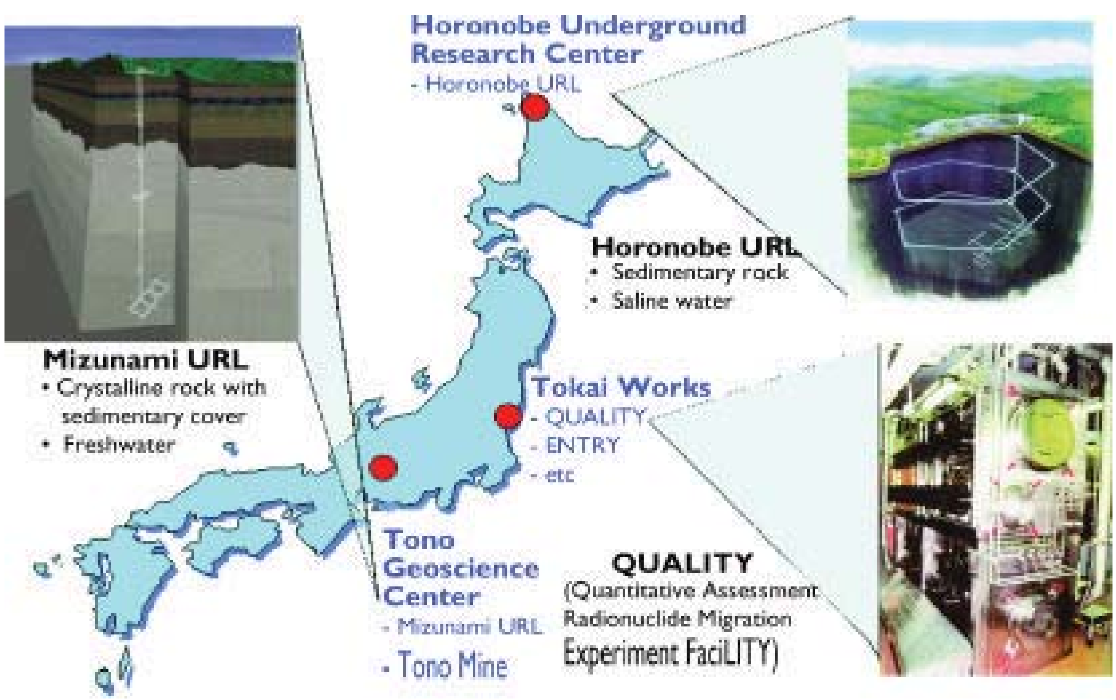

Figure 17.12. JNC's R&D facilities

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Chapter 18

Radioactive Waste Disposal of LILW and HLW in Korea

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ABSTRACT. The radioactive waste disposal program in Korea consists of a low- and intermediate-level waste (LILW) component and a high-level waste (HLW) component. For the LILW component, the Korea Hydro & Nuclear Power Co., Ltd., (KHNP) is seeking a repository site through voluntary participation of local governments and will determine the proper site through screening of candidate sites. According to the new plan of radioactive waste management established in 1997, the repository for LILW will be in operation by 2008, and the initial disposal capacity of LILW will be about 100,000 drums (based on a 200-liter-per-drum capacity). Conceptual design and generic safety assessment have been completed for both rock cavern and engineered vault disposal options as reference disposal methods. Preliminary probabilistic risk assessment and deterministic dose assessment were undertaken with the appropriate computer code scheme set up for the purpose of performance assessment for each disposal concept. The results of a radiological impact assessment showed good compliance of the conceptual design to performance objectives. The results indicate, however, that the performance will be strongly dependent on the geology and hydrogeology of the disposal facility location, and it should be noted that the assessment was carried out with limited understanding of the site-specific characteristics. These preliminary assessments of the conceptual disposal facilities provide a firm foundation for future site-specific assessment activities. A refined and comprehensive safety-assessment approach, along with site-specific data, would be required for the next stage of the LILW disposal project implementation.

For the HLW component of the Korean radioactive waste disposal program, a ten-year, three-step R&D program, launched in early 1997, is being conducted by the Korea Atomic Energy Research Institute (KAERI). This program will develop a reference geologic repository system for the disposal of high-level radioactive waste (HLW) coming from nuclear power generation in Korea. As the first step in developing the reference repository system, Mesozoic plutonic rocks were screened as the primary potential host rock for further study. With plutonic rock as a natural barrier, a repository concept was established by combining an engineered barrier system consisting of the waste form (CANDU and PWR spent nuclear fuels), corrosion-resistant metal canisters, and a high-density bentonite buffer. To assess the relevance of the established concept, we developed a performance-assessment code, MASCOT-K. In parallel, a set of radionuclide migration studies were conducted to investigate sorption mechanisms (e.g., surface complexation models [SCM]), a sorption database (i.e., SDB-21C), and fracture migration. In the second step, this R&D is focusing on further development of the established repository concept by incorporating data and information from an *in situ* investigation of crystalline rock masses.

18.1. LILW DISPOSAL

18.1.1. INTRODUCTION

In Korea, as of the end of 2000, 16 commercial nuclear power plants are in operation, and application of radioisotopes in industry has increased year by year. But a disposal site for radioactive waste has not been selected yet, and the final disposal of radioactive waste is regarded as a national issue. In 1995, Korea, which had been trying to establish a disposal site for safe radioactive waste management since 1989, selected the Gulup Island. However, the Gulup Island repository plan was subsequently abandoned because of the unexpected discovery of active faults near the island. As a result of restructuring in the national nuclear industry, responsibility for the management of radioactive waste was transferred from Korea Atomic Energy Research Institute (KAERI)/Ministry of Science and Technology (MOST) to Korea Hydro & Nuclear Power Co., Ltd. (KHNP)/Ministry of Commence and Industry and Energy (MOCIE). The Nuclear Environment Technology Institute (NETEC) of KHNP was established as the special organization for radioactive waste disposal.

The new plan of radioactive waste management was established by MOCIE and approved by the Atomic Energy Committee in 1997. According to the new plan, the repository of low- and intermediate-level waste (LILW) will be in operation by 2008, with an initial disposal capacity for LILW of about 100,000 drums (based on a 200-liter-per-drum capacity). Conceptual design and generic safety assessment had been completed for both rock cavern and engineered vault disposal facilities as the reference disposal methods in 1993 and 1999, respectively. KHNPis seeking approval for a repository site through a voluntary participation of local governments and will determine the proper site through screening. A preferred disposal type will be determined on the basis of site conditions. KHNP’s offer for public approval of the disposal site is from June 2000 to June 2001.

In this report, we summarize the results of a conceptual design and generic safety assessment carried out for the reference Korean LILW disposal facility. In Section 18.1.2.1, we describe the conceptual design and the radiological-risk-assessment calculations undertaken to address radionuclide transport along the groundwater pathway for a rock-cavern repository option. A screening-level safety assessment, based on a deterministic approach, for a potential engineered vault facility is described in Section 18.1.2.2. Three main sets of scenarios, namely waste-package-handling accident scenarios for the operating period, normal-evolution scenarios through groundwater pathways, and human-intrusion scenarios for the post-closure period, have been developed for the assessment and are also described.

18.1.2. CONCEPTUAL DESIGNS AND PERFORMANCE ASSESSMENTS FOR REFERENCE KOREAN LILW DISPOSAL FACILITIES

18.1.2.1. Rock Cavern Disposal Facility

Conceptual Design Description

For the initial phase of the disposal facility construction, five disposal caverns (four low-level radioactive waste [LLW] caverns and one intermediate-level radioactive waste [ILW] cavern) were designed, as illustrated in Figure 18.1. Each cavern is connected with operation and construction tunnels. Three types of caverns for LLW will be constructed: an LLW Type I cavern for dry active wastes, LLW Type II caverns (two caverns) for dry active wastes and concentrated wastes, and an LLW Type III cavern for spent resin wastes, spent-filter wastes, and concentrated wastes. The LLW caverns have an inclination of 1% toward the cavern entrance to facilitate the drainage of inflowing water to the water basin, with the LLW handled by a forklift truck. ILW, defined as waste that has to be handled remotely in the disposal facility, will be emplaced in a cavern that contains large concrete compartments with the same inclination as the LLW caverns. An overhead crane will remotely handle the waste package (B.M. Kang et al., 1998).

The general direction of groundwater flow in the region around the hypothetical repository location would be from the high ground inland towards, and roughly perpendicular to, the coast. The major formations identified in the geological structure around the repository are andesite and fractured andesite. It is assumed that two fault zones will exist within the repository site boundary. The caverns will be located in the depth interval from +50 m to –50 m relative to sea level.

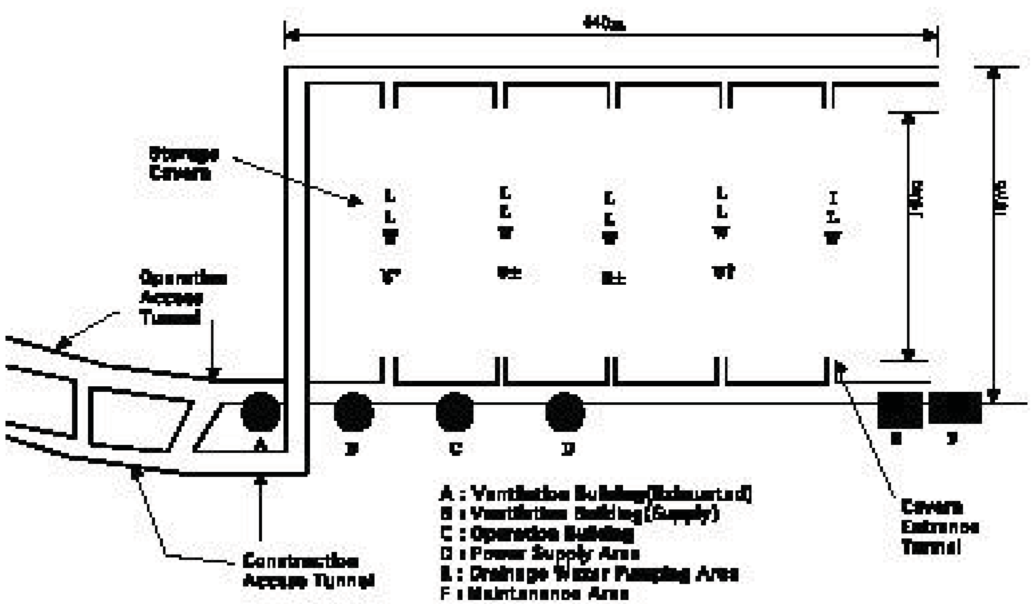

Figure 18.1. Proposed layout of Korean rock cavern disposal facility (initial phase)

Assessment of Groundwater Pathway

Assessment Scenario

As a first step in performance assessment, a study of the radionuclide-release scenario development was carried out for the rock-cavern disposal concept. From these results, it was assumed that the groundwater would have a normal release scenario. After closure and resaturation of the repository, it was assumed that radionuclides would dissolve in groundwater flowing through the waste within the cavern and be transported with the groundwater through the geosphere to the biosphere (i.e., river, well or spring, or sea water).

Groundwater Pathway Modeling

To calculate the risk from the groundwater pathway, we constructed an overall model of the system for groundwater-mediated return of radionuclides to the biosphere. Source term, geosphere, and biosphere models were implemented using the MASCOT program (Sinclair et al. 1994). Table 18.1 lists the radionuclide inventory of the facility considered in the safety assessment. Radionuclides with half-lives less than 5 years are eliminated, but long-lived radionuclides with a high potential for mobility (e.g., ^{14}C, ^{99}Tc, ^{129}I, and alpha-emitting transuranics such as ^{239}Pu) are included. In addition, radionuclides with a relatively high dose conversion factor and/or significant ingrowth of daughter radionuclides (e.g., ^{238}U) are included in the inventory.

To estimate the rate of groundwater flow in the vicinity of the repository (which largely determines the flux of radionuclides leaving the repository) and the nature of the regional groundwater flow system (which determines the subsequent transport through the geosphere and return to the biosphere), a Numerical Assessment Method for Modeling Migration Underground (NAMMU) program was used (Hartley and Jackson, 1993). A series of calculations with variant parameter values were performed to generate the probability density functions (PDFs) required by the geosphere-transport submodels of the MASCOT PSA program. PDFs are required for the travel times from the repository to the biosphere, the specific discharges at the repository locations, and the path length.

Because the six types of waste are distributed in different caverns, we adopted an approach in which nine different source-term submodels were used. The MASCOT containment submodels were used to specify the initial inventory of each radionuclide and the length of time for absolute containment of the waste.

These data feed into source-term submodels that calcu-

Table 18.1. Radionuclides, half lives, and in ventories considered in the assessment

Radionuclide	Half Life(years)	Inventory(Ci) Rock cavern type	Inventory(Ci) Engineered vault type
H-3	1.24E+1	1.16E+2	7.02E+2
C-14	5.73E+3	1.11E+2	4.54E+2
Co-60	5.27E+0	1.66E+4	4.68E+3
Ni-59	7.50E+4	2.75E+2	9.86E+1
Ni-63	9.60E+1	8.42E+3	2.57E+3
Sr-90	2.91E+1	2.15E+2	3.75E+1
Nb-94	2.03E+4	4.09E+0	2.71E+0
Tc-99	2.13E+5	9.02E+0	1.10E+0
I-129	1.57E+7	5.21E-1	3.39E-1
Cs-137	3.00E+1	9.21E+3	1.65E+3
U-235	7.04E+8	3.00E-4	5.15E-3
U-238	4.47E+9	1.06E-2	1.28E+0
Pu-238	8.77E+1	4.29E+0	3.46E+0
Pu-239	2.41E+4	8.86E+0	1.59E+0
Total		3.50E+4	1.02E+4

Note:Inventory of rock cavern type disposal facility includes that of HIC (High Integrity Container).But engineered vault type facility comprises RI (radioisotope) wastes instead of HIC.

late the release of radionuclides. Distribution submodels enable the total source-term output to be fed into the geosphere section of the system model. Input into the geosphere network is the combined flux from all source terms: the fluxes from the two paths, which move through each fault zone, are recombined to provide a total flux to the biosphere. Three exposure pathways were considered, in which the flux from the geosphere is released to (1) deep soil, (2) a river, and (3) a well. Three identical compartment-biosphere submodels were used in MASCOT to represent the three exposure pathways, each taking as its input the entire flux from the geosphere. The difference between the three submodels was that the inlet to different compartments was appropriate for the three exposure pathways. The outlet from the biosphere was, in each case, the total dose from concentrations in soil, water (river or irrigation), and seawater.

Results and Discussion

For both choices of repository location and for all the alternative assumptions about the biosphere release point, it was found that the predicted risk remained below the target of 10^{-6} yr^{-1}—throughout the period of detailed quantitative assessment and up to 1,000 years after closure. Beyond this time, no dramatic increases in risk were found, the greatest calculated mean risk being 1.4×10^{-6} between 10^4 and 10^5 years post-closure. The contributions of important radionuclides to the total risk were examined for each repository position and biosphere model. Figure 18.2 shows the total annual individual risk and the contributions from important radionuclides as functions of time for release to the well biosphere for the lower repository location. Up to 1,000 years after closure, only ^{129}I makes a significant contribution to risk. Other radionuclides are either effectively contained or else delayed in their passage through the geosphere. Beyond the 1,000-year period, the only significant additional contributor to risk was found to be ^{59}Ni.

18.1.2.2. Engineered Vault Disposal Facility

Disposal Facility Description

The climatic setting for this disposal facility is a humid environment, with annual average rainfall of 1.5 m. Native soils at the site are assumed to be relatively permeable. The disposal facility consists of three types of disposal vault, depending on the durability and/or size of waste packages: Vault I (waste packages with long durability and backfilled with gravel), Vault II (standard-sized waste packages with short durability and grouted with cement mortar), and Vault III (large size

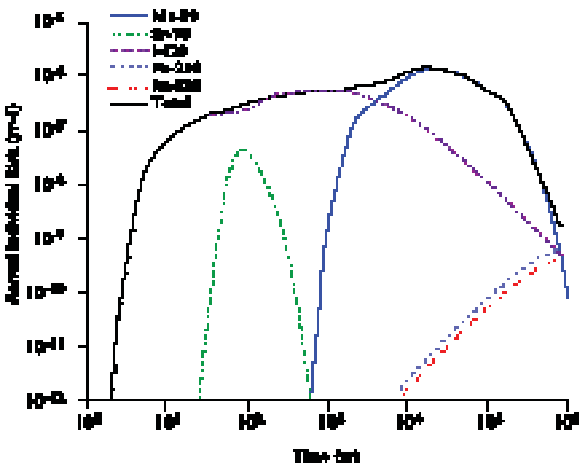

Figure 18.2. Total annual individual risk and the contributions from the important radionuclides (well, lower repository)

waste packages with short durability and grouted with cement mortar) (B.M. Kang et al., 1998).

The capacity of a vault will be 5,000 drums (based on a 200-liter-per-drum capacity). Seventeen grouted vaults (10 for Vault II and seven for Vault III) and three backfilled vaults will be constructed in the initial phase. The disposal facility will be oriented such that the longitudinal direction of the facility is perpendicular to the main direction of the aquifer flow. During waste-package loading, the disposal vault is covered with a mobile roof equipped with a waste-package-handling crane that can be moved to the next vault for another loading operation. The final disposal cover will be constructed when the disposal vaults in a disposal area are completely filled. Final cover consists of a 6.2 m thick multilayer system to ensure low percolation, water drain, and intrusion resistance.

Assessment Modeling and Simulation Results

Waste-Package-Handling Accident Scenarios

We also considered the exposure to radioactivity resulting from postulated dropping accidents during the disposal operation. Two kinds of dropping scenarios were studied: dropping of a waste package onto another waste package and dropping of a waste package that included a secondary collision on the floor. In these scenarios, it is assumed that a spent filter drum with equivalent dose rate of 10 mSv/hr at the surface is dropped 6 m from the crane. Radiation exposure dose rates for an individual crane worker from the postulated dropping accident, calculated by MICROSHIELD computer code, were far below the annual allowable limit for radiation workers, which is 20 mSv/yr (Grove Engineering, 1998).

Groundwater Scenario

For a conservative analysis of the radiological impact via groundwater pathway, we considered a well-drilling scenario near the site boundary. During the institutional-control period, waste forms are contained in nondegraded disposal containers, and would not come in contact with water. At the end of this period, the multilayer cover is assumed to be totally deteriorated, so that the water infiltration rate is almost the same as that of natural soils at the site. All engineered barriers, including disposal containers and waste matrices, are also assumed to be completely degraded. The radionuclides leached from waste forms begin to migrate into soil and aquifers, and finally reach wells located near the site boundary. The nearest well is assumed to be 200 m away from the centerline of the waste vaults. The dose is calculated by assuming that a person consumes 2 liters/day of well water. This generally yields the highest dose in the absence of irrigation.

For the groundwater pathway assessment, an estimate of water influx into the vault is required for the source-term component of performance assessment. The results are also used to determine the upper-boundary conditions of a detailed two- or three-dimensional fluid flow model for the portion below the cover. Water-balance simulations for the cover profile are conducted using the HELP code (Schroeder et al., 1994). Long-term performance of the final cover is simulated by considering the degradation of artificial barrier materials after 100 years under three precipitation conditions: (1) ambient precipitation, (2) doubling of the ambient precipitation, and (3) design storm conditions. Radionuclide release rates through the bottom of a concrete vault are calculated with the DUST-MS code (Sullivan, 1993). Homogenization by repository averaging is made in the source-term modeling. Three time intervals are considered—the first 100 years, 100 to 300 years, and after 300 years—to take into account the degradation of the engineered barrier. Finally, the GWSCREEN code is used to calculate radionuclide transport in unsaturated soil and aquifers, human uptake, and doses to the individual in a critical group (Rood, 1999). Radionuclide inventory of the facility considered in the safety assessment is listed in Table 18.1, where the same radionuclides with rock-cavern disposal (except radioisotope wastes) are included. Soil-water-retention parameters, hydraulic-conductivity parameters, and sorption characteristics were determined from the relevant previous studies.

Figure 18.3 shows peak ingestion dose and time of peak for each radionuclide release, calculated by GWSCREEN. Considering the peak-dose and assessment time scale, the contribution of ^{14}C and ^{129}I to the final dose is significantly higher than that of other radionuclides. Therefore, these radionuclides are identified as important. The time evolution of the concentration of radionuclides (Bq/m^3) extracted from the well is shown in Figure 18.4. Sensitivity analysis was carried out with the variation of different input parameters. The objective of the sensitivity analysis is to focus attention on important parameters by determining the relative contributions to the resulting dose. Findings show that the dose is most sensitive to Darcy velocity in aquifer for ^{14}C. The distribution coefficient also shows a high degree of sensitivity for ^{129}I release (Park et al., 2001).

Human Intrusion Scenarios

In another set of scenarios, it was assumed that human intrusion into the repository occurs deterministically 300 years after closure of the facility, in other words just after the end of the institutional-control period. Seven kinds of scenarios are postulated as potential intruder events: drilling, post-drilling, excavation, post-excavation, gardening, biotic, and farming. Each scenario definition and evaluation methodology followed the cases considered in the U.S. DOE LLW disposal facility (Aaberg et al., 1990). The GENII dose-assessment code was used to calculate the equivalent dose rates to be received by an intruder, considering exposure pathways that cause radiation exposure to humans (Napier et al., 1988). Estimated exposure dose rate includes external irradiation from the soil, inhalation of contaminated dust, and consumption of food produced in contaminated soil. It was calculated from these scenarios that the maximum effective dose to the intruder resulting from a post-excavation scenario in Vault Type II is 0.83 mSv/yr (C.L. Kim et al., 2000).

Comparison with Performance Objectives

Performance objectives for operating personnel and individuals required for the licensing process need to be defined in two stages: operational period and post-closure period. Performance objectives applicable to the LILW disposal facility in the post-closure period are defined in the MOST Notice No. 96-11 in terms of radiological risk to any individual in a critical group. It states that the predicted radiological risk for the post-closure period shall not exceed 10^{-6} per year (or its dose equivalent of 0.02 mSv/yr). It is recognized that this criterion means risk or dose constraints to be met after closure. While this Notice does not give specific guidance

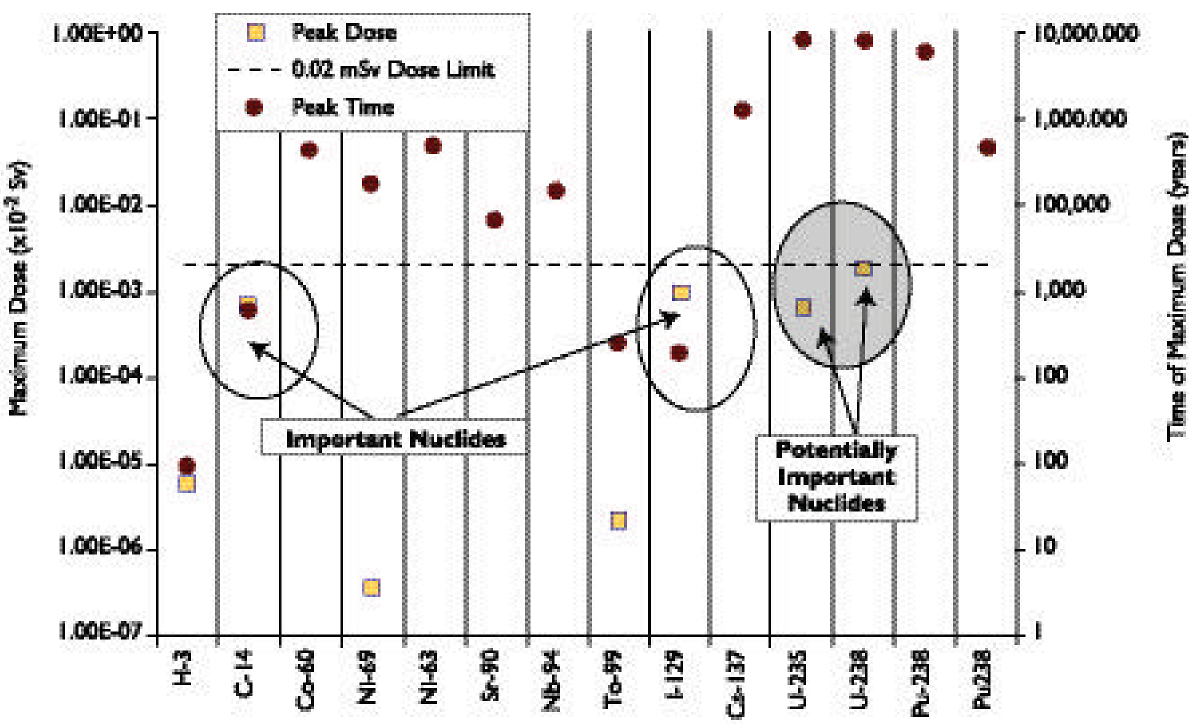

Figure 18.3. Peak dose and the time of the peak for each radionuclide

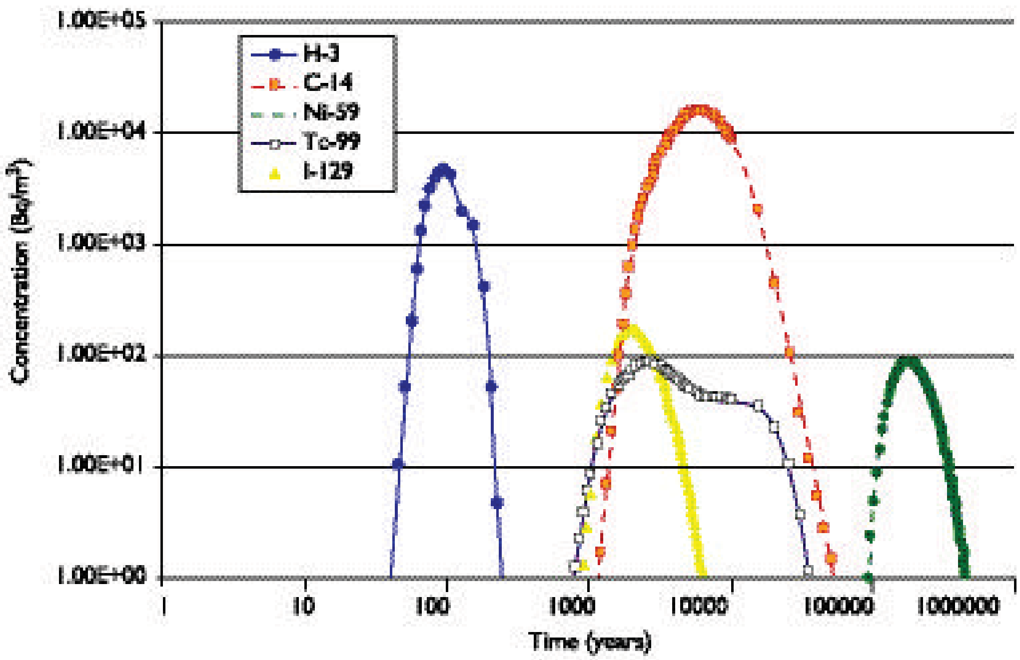

Figure 18.4. Radionuclide concentrations in well water

on radiological-protection criteria for the operational period, the dose limit of 20 mSv per year for operating personnel is stipulated in the MOST Notice No. 98-12 applicable to all nuclear facilities in Korea. The dose constraint for operating personnel will be determined with an appropriate fraction of the above dose limits. The effective dose equivalent to an intruder after the loss of active institutional control of the facility will also be determined by the Korea Institute of Nuclear Safety (KINS). In this study, the dose limit of 1 mSv per year to an inadvertent intruder was assumed.

The radiological consequences from the safety assessment of two disposal concepts for main scenarios, as summarized in Table 18.2, were below the dose limits set up in the national regulations on radiological protection.

18.1.3. Conclusions

In Korea, the conceptual designs of a representative disposal concept for LILW, both for rock-cavern and vault disposal facility options, have been implemented. Assuming a disposal facility at a hypothetical site, radiological safety assessments based on both probabilistic and deterministic approaches for each type of disposal were performed to confirm compliance of the proposed disposal system to performance objectives. Main scenarios covering waste-package-handling accident scenarios for the operating period, normal-evolution sce-

Table 18.2. Summary of preliminary safety assessments

Evaluation Scenarios	Safety Assessment Results: Rock Ca vern	Safety Assessment Results: Engineered Vault	Performance Objectives (Post Closure Period)
Groundwater pathway	9.45×10^{-7}/yr	0.01 mSv/yr	10^{-6}/yr or 0.02 mSv/yr
Inadvertent intruder	-	0.83 mSv/yr	1 mSv/yr (assumed)
Operational exposure (waste package drop accident)	-	0.29 mSv/yr	20 m/Sv/yr

narios through groundwater pathways, and human-intrusion scenarios for the post-closure period were developed. Depending on the nature of the scenarios, computer tools for quantitative analyses were applied in an appropriate manner. Risks and effective doses were calculated for various conservative exposure pathways and scenarios.

The results of radiological impact assessment showed that the doses for all pathways assessed are below the current regulatory limit in Korea. This means that the conceptual design of disposal concepts could be properly implemented from the point of view of radiological safety. The results indicate, however, that the performance will be strongly dependent on the geology and hydrogeology of facility location, and it should be noted that the assessment was carried out with limited understanding of site-specific characteristics. It will be necessary to obtain site-specific data at the intended disposal facility location to enable a more detailed assessment to be undertaken. These preliminary assessments of conceptual disposal facilities provide a firm foundation for future site-specific assessment activities. A refined and comprehensive safety-assessment approach, along with site-specific data, would be required for the next stage of LILW disposal project implementation.

18.2. HLW DISPOSAL

18.2.1. Introduction

Korea has 16 nuclear power plants (12 PWRs and 4 CANDUs) in operation and plans to build 12 more plants by 2015 (H.S. Park, 2001). Operation of the existing and planned power plants until 2015 is expected to produce 36,000 tHM of spent nuclear fuel, which must be managed safely. Such management will inevitably involve disposal of high-level waste (HLW), which may be in the form of spent nuclear fuel itself or vitrified waste. In this context, the Korea Atomic Energy Research Institute (KAERI) launched a three-step R&D program in 1997 to develop a reference geologic repository system for HLW by 2006.

As a first step, the R&D program set out a reference geologic repository concept using generic geologic information and data (Chun et al., 2001; Han et al., 2000). The program has now entered into its second step to develop a preliminary reference geologic repository system focusing on the near-field components. Second-step activities will include deep borehole investigation of a crystalline rock body to produce data required to establish the preliminary system. The third and last step will be devoted to further R&D of the preliminary system using more refined information about reference site characteristics, domestic materials applicable for the engineered barrier system, etc.

The R&D program involves a number of activities, i.e., geoscientific environmental research, repository system development and performance/safety assessment, and migration studies.

18.2.2. Geoscientific Environmental Research

The Korean peninsula is located in the marginal area of the stable Eurasian continent, which links to the west Pacific mobile belt. Thus, the geological environment of Korea differs from the Japan archipelago and China platform. For the geotectonic history of the peninsula, the Mesozoic orogeny is most important, an orogeny that includes folding, fault block movement, and igneous intrusion. The lithology of the Korean peninsula consists of a complex structure of 29 rock types, from Archean to Quaternary. Geoscience research work has focused on the production of basic data accompanied by the technical development of geologic and hydrogeologic characterization (C.S. Kim et al., 2000).

The regional fracture system has been classified and described from a lithological and geotectonic point of view, using available literature, shaded relief maps, and aeromagnetic survey data. The hydrogeological setting in Korea was classified by preliminary topographic characteristics and considered in terms of geological structures and lithology. Through data collection and fieldwork supported by laboratory studies, three rock types—plutonic rocks, crystalline gneisses, and massive volcanic rock—were selected as potential preferred host rocks.

In the second step, field studies began with deep borehole tests on a plutonic rock body, which produced data on mechanical properties, fracture characteristics, and geochemical properties of the rock and fracture-filling materials. While waiting to obtain deep groundwater samples, we set a tentative reference groundwater condition for the planned preliminary safety assessment and the production of required input data, i.e., sorption and diffusion of radionuclides with respect to the rock.

Further studies from deep boreholes will focus on

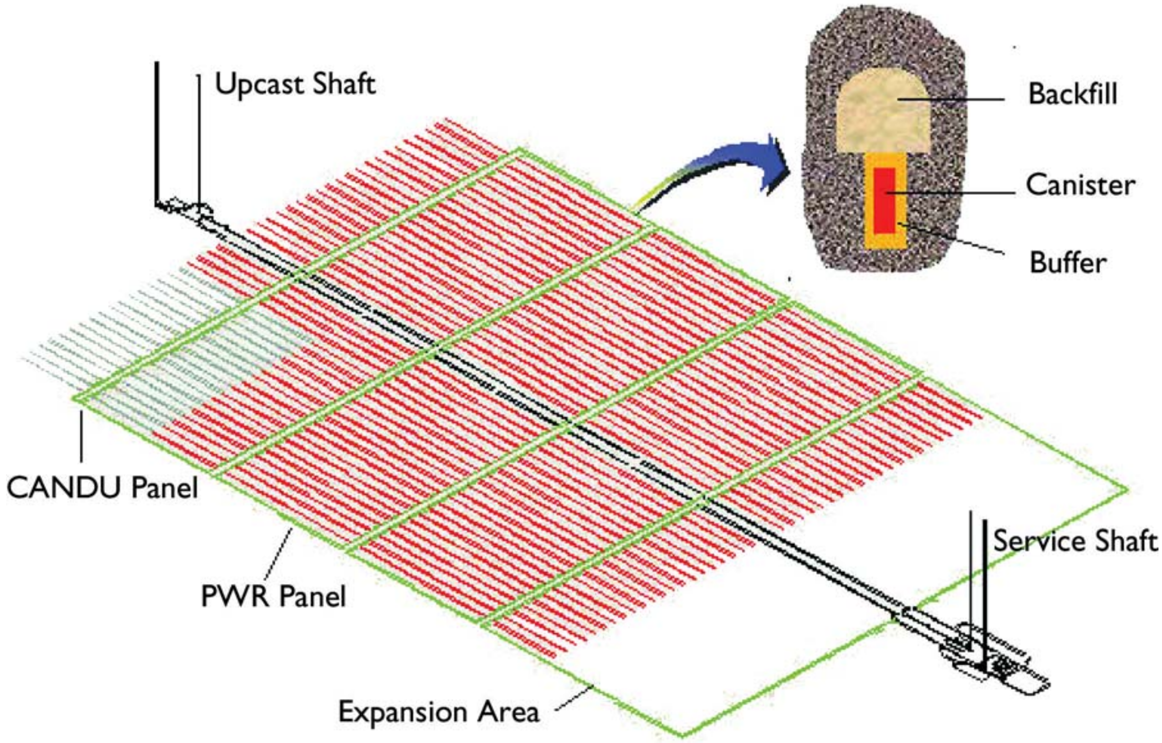

Figure 18.5. Geologic repository concept under development by KAERI

groundwater chemistry and flow-path analysis for more specific information and data to establish a preliminary reference geologic repository system and its safety assessment.

18.2.3. Repository System Development

The development of a Korean reference repository system was based on the internationally accepted fundamental principle that radioactive waste shall be managed to secure an acceptable level of protection for human health and the environment, as well as to impose no undue burdens on future generations. The required level of protection was interpreted in terms of safety criteria (C.H. Kang et al., 2000), while the post-closure safety target is under revision by the regulatory authority. In applying such criteria, we recognized that particular attention should be given to describing protection for the period up to the closure of the repository and the first 1,000 years thereafter, with a special focus on nearby residents. We also recognized that due consideration should be given to the difficulties in assuring compliance with safety criteria over a long time scale. To comply with the criteria, we introduced the concept of passive multiple barriers into the development of a reference repository system and established a set of functional requirements for each barrier (C.H. Kang et al., 2000).

As the first step, a geologic repository concept was generated, taking into account the established criteria and requirements, waste characteristics, and generic site characteristics in Korea. The basic repository concept is to encapsulate HLW in corrosion-resistant containers (high-Ni alloy, stainless steel, or copper), and place the encapsulated waste packages into deposition holes with high-density bentonite buffers. The deposition holes will be located in tunnels constructed at a depth of about 500 m in a stable plutonic rock body.

Within the framework of the basic concept, seven different options were scrutinized and ranked to come up with the most feasible option, i.e., vertical emplacement of waste packages with separate disposal rooms for spent PWR and CANDU fuels). A preconceptual design study was conducted on the selected option. At the same time, basic properties of Kyungju bentonite (Ca-bentonite containing 70 wt% montmorillonite) mixed with sand were characterized and assessed against the established functional requirements for a buffer (Cho et al., 1999). As a result, bentonite, with a dry density greater than 1.6 Mg/m^3, was suggested as a candidate buffer material. Figure 18.5 presents the repository concept established through this study.

The second-step study began by quantitatively evaluating important parameters of the established repository concept, focusing on the near-field system (i.e., canister thickness, buffer thickness, spacing of deposition holes and tunnels). This study will lead to the proposal of a preliminary reference repository system, the target of the second step. Also, further study will be conducted to characterize the overall long-term performance of the candidate material in support of system development.

18.2.4. Performance /Safety Assessment

A total system performance-assessment code (MASCOT-K) was developed by modifying a one-dimensional probabilistic safety assessment (PSA) code, MASCOT, which was originally intended to assess LILW repositories. To make MASCOT able to accommodate the transport phenomena in HLW repositories, first-step research focused on the development of source-term release modes such as a gap with backfill and congruent release (C.H. Kang et al., 1997; 2000). Also included in the research was identification of relevant features, events, and processes (FEP) and subsequent establishment of reference scenarios.

Using MASCOT-K, a preliminary post-closure safety assessment for the case of a well scenario was conducted with respect to the reference repository concept. To acquire the needed input data for the assessment, we collected in-house data sets from other R&D fields, and the remaining data came from a literature survey of overseas-related projects. The preliminary result is indicated in Figure 18.6.

At present, in the second-step study, a set of preparatory work is underway to establish a preliminary safety/performance assessment of the reference system. This is to facilitate the final assessment of the second step. At the moment, a methodology for probabilistic assessment of the groundwater flow and a system for input data management and quality assurance have been developed. In addition, submodels are being developed to support the assessments using MASCOT-K.

18.2.5. Migration Study

The migration study is intended to identify the chemical behavior of radionuclides in the deep underground environment as well as to produce input data for the safety assessment and validate the transport models used in the assessment (Jung et al., 2001). A geochemical code named MUGREM was developed to investigate the behavior of radionuclides in the system of deep groundwater and geological media. The code has been extended to introduce an SCM. To investigate these models, sorption experiments have been conducted with different radionuclides on single minerals (e.g., quartz, hematite, goethite, kaolinite, illite, montmorillonite). Initially, the experimental work began with the sorption of alkaline earth elements, transition metals, and anions under aerobic conditions. Then it expanded to deal with actinides under anaerobic conditions. This investigation is to be extended yet further to deal with more actinides as well as with the effects of mineral alteration and high-salinity groundwater conditions.

The input parameters for safety assessment include sorption and diffusion. A sorption database system (SDB-21C) was developed that incorporated data from a literature survey and from the KAERI program experiments (Jung et al., 2001). The system has been updated to accommodate sorption data produced from recent experiments with actinides on granite rock under anaerobic conditions. Thus, the database covers a large number of radionuclides with different geologic media under various groundwater conditions. On the other hand, diffusion data for some radionuclides in granite rock have been empirically determined. More sorption and diffusion data are to be produced for the safety assessment.

The migration of radionuclides in fractured rock has been investigated using an artificially fractured rock block (Hahn et al., 2000). The study has been expanded to include a granite block with a natural fracture, and it will consider such effects as sorption on the fracture surface and on fracture-filling minerals. Formation and migration of colloids is another area of research to be undertaken.

18.2.6. Summary

A three-step R&D program is being carried out by the Korea Atomic Energy Research Institute to develop a reference geologic repository system for disposal of high-level waste coming from nuclear power generation in Korea. In the first step, a geologic repository concept was established using broader geoscientific information and data from the Korean peninsula.

The second-step research program is now in progress,

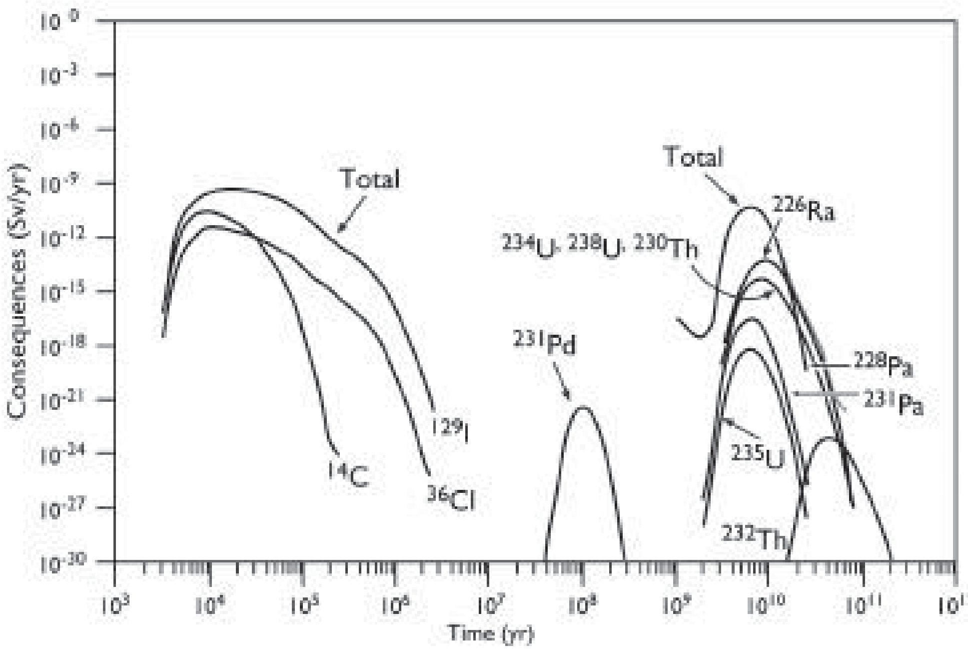

Figure 18.6. Result of a preliminary performance assessment using a well scenario in the proposed repository concept

with sensitivities of major near-field system parameters having recently been evaluated. Parameter values will be determined to establish a preliminary reference repository system, using data from ongoing borehole tests on a plutonic rock body, experiments on buffer materials, etc. Extensive progress will also soon be made in safety-assessment technologies and associated research on radionuclide behavior under geologic repository conditions.

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Chapter 19

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

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19.1. 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 is projected to be 2010–2015.

After the nuclear accident in Chernobyl, safety systems at the Ignalina NPPwere 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.

The Ignalina NPPis and (for the foreseeable future) will be 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. Invoking Article 9 of the Energy Law, a new National Energy Strategy was approved by the Seimas (Parliament) on October 5, 1999. This National Energy Strategy establishes that Unit 1 of the Ignalina NPP will be closed down before the year 2005, taking into consideration substantial long-term financial assistance from the European Union, G7, and other states as well as international institutions. Regarding Unit 2, the conditions and precise date of closure shall be decided upon in the next National Energy Strategy conference in 2004, when more detailed information on the work of Unit 2 is available. There are no other nuclear facilities in Lithuania.

More than 99% of Lithuania's radioactive waste is generated at the Ignalina NPP. The remaining radioactive waste (a few m^3/year) is generated by radioactive radiation sources in research, medicine, and industry. About 450 such institutions are authorized to handle radiation sources; however, most of them are not generating any radioactive waste, but rather using x-ray equipment. (Some hospitals are using radioisotopes for diagnostic and therapeutic purposes.)

19.2. LOW- AND INTERMEDIATE-LEVEL WASTE

19.2.1. Current Situation and Plans for Conditioning and Interim Storage

The Ignalina NPP produces about 2,000 m^3 of radioactive waste (except spent fuel) 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 Ignalina NPP 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 mixture was started in 2001, with incineration of the solid combustible waste also planned. The modernization of the entire management system for solid short-lived low/intermediate level waste (LILW-SL) and long-lived intermediate level waste (ILW-LL) is planned to be implemented at Ignalina NPP during the preparatory period for decommissioning of Unit 1.

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 each building's operation. Buildings 155 and 155/1 are intended for storage of LLW-SL, and they are assembled-monolith ground buildings. Buildings 157 and 157/1 are intended for storage of LILW, and they are ground facilities of concrete with 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. These storage facilities are not yet licensed, but a safety analysis report on these facilities was prepared during 1999–2000 and is now under review by regulatory authorities. Waste from research, medicine, and industry are transported to the Ignalina NPP and are also stored in these buildings.

Liquid waste is collected into big concrete tanks and temporarily stored there before evaporation and bituminization.

19.2.2. Activities Related to the Disposal of Low and Intermediate Level Waste

There are two old radioactive waste disposal facilities in Lithuania. Of Russian design, both of them could be classified as near-surface disposal facilities. Neither of them has yet been licensed.

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

The repository is designed for institutional waste and is a typical "Radon" type facility constructed in the early 1960s throughout the former Soviet Union (in 1964, Lithuania). It was closed in 1989.

Waste at this site is disposed of in a reinforced concrete vault with internal dimensions 14.75 × 4.75 × 3 m (200 m^3 in volume). The vault was only partially filled (about 60% of the volume) with waste during 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 have been disposed of with 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 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. They consist of moraine subclay and subsand or sand/gravel.

Safety assessment of this facility was performed by SKB (Sweden) with participation of the Lithuanian Energy Institute (Poskas et al., 2000). If a preferred retrieval solution were identified, the already-disposed waste would be retrieved for conditioning and characterization.

19.2.2.2. Storage/Disposal Facility for Bitumenized Waste at Ignalina NPPsite

A storage/disposal facility exists at Ignalina NPP for bitumenized evaporator bottoms of liquid radioactive waste generated there. This is an aboveground two-story assembled-monolith concrete building (Building 158). It consists of 12 inside steel-lined vaults for loading of 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 sediments by a series of tectonic breaks. The dimensions of these tectonic blocks are approximately 2 × 2 km. In the area of the Ignalina NPP, the surface sediments are very heterogeneous. They were formed during the retreat of the last period of glaciation as a result of different glacial and water-glacial processes. Later, alluvial, marsh, and lake sediments were formed. The lithological structure, filtration, and engineering-geological properties of 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 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 have a peat, sand, gravel, sandy soil, sandy loam and clay composition. 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.

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). The analysis led to the conclusion that this radioactive waste facility could be converted into a disposal facility if a multilayer earth cover were used, but further, more detailed analysis is necessary.

19.2.2.3. New Near-Surface Disposal Facility

A recent safety assessment, performed by SKB (Sweden) with the participation of the Lithuanian Energy Institute, revealed that the existing storage facilities for solid waste at the Ignalina NPP site cannot be converted into repositories. So it is very important to start activities related to construction of a new repository for LILW–SL in Lithuania soon. As a first step, a new project has been started in 2000, under the leadership of SKB and with the Lithuanian Energy Institute's participation. The prime objective of this project is to prepare the reference design for a near-surface repository for LILW–SL generated in Lithuania. The reference design shall be applicable to the needs in Lithuania, considering its hydrogeological, climatic, and other environmental conditions. During this project, the overall plan for the implementation of the repository will also be prepared.

19.3. SPENT NUCLEAR FUEL AND LONG-LIVED WASTE

19.3.1. Interim Storage of SNF and Long-Lived Waste

According to Lithuania's Law on Nuclear Energy, spent nuclear fuel is a radioactive waste, so it will not be reprocessed. In 1992, the decision was made to build an interim dry spent-nuclear-fuel storage facility at the Ignalina NPP site with a lifetime of about 50 years. A proposal by the German company, Gesellschaft für Nuclear-Behalter mbH (GNB), was accepted to store the spent nuclear fuel outdoors in 60 sealed metal (CASTOR RBMK-1500) casks filled with helium. Twenty of such cast iron CASTOR RBMK-1500 casks have already been delivered to Ignalina NPP. A contract for the delivery of 40 metal-concrete CONSTOR RBMK-1500 casks (instead of CASTOR RBMK-1500) casks was signed later with GNB. Some of these casks have already been delivered to Ignalina NPP, and the loading process has been started. Now, in conjunction with the decommissioning of Unit 1, another tendering process will be started to reload all spent nuclear fuel from the pools.

As was indicated above, long-lived waste is also stored in big vaults within concrete structures at the Ignalina site. We plan to modernize handling of this waste and store it for 50 years in concrete containers at the interim storage facility, and also to retrieve the waste from the vaults.

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

The draft strategy on management of radioactive waste in Lithuania defines a number of activities related to spent-nuclear-fuel disposal: collaboration with international organizations and other countries for selection of a regional repository, negotiations for agreement with supplier of fresh fuel to return back spent nuclear fuel, analysis of the possibilities for extending the interim storage period for spent fuel, and the start of investigations on the disposal of spent nuclear fuel in Lithuania.

Recently, some activities by the Lithuanian Energy Institute and Geological Survey of Lithuania have been started (with the support of Sweden). For instance, the Lithuanian Energy Institute has started research activities designed to select a potential concept for disposal of spent RBMK-1500 nuclear fuel. The aim is to develop competence in performance assessment and to select a robust and flexible technology that could be adapted to different sites. In parallel, prospective investigations have been started at the Geological Survey of Lithuania. All available data (borehole data, geological and geophysical mapping data, raw material prospecting data, and results of special geological investigations) concerning the geological structure of Lithuania will be analyzed.

On the basis of some earlier investigations of the geological structure of Lithuania, several geological forma-

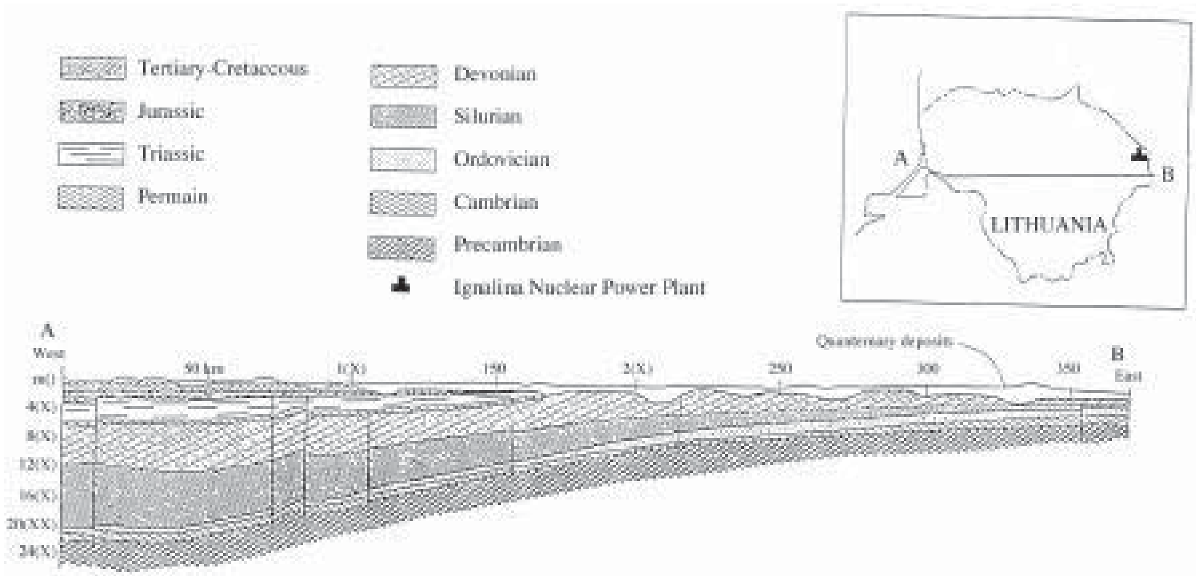

Figure 19.1. East-west geological cross section (Line A-B) of Lithuania bedrock

tions could be selected for a deep repository of radioactive waste (Kanopiene and Marcinkevicius, 2000): rocks of the crystalline basement, Lower Cambrian clay, Permian sulphate deposits, Permian rock-salt, and Lower Triassic clay. The territory of Lithuania is in the northeastern part of the East European platform. A crystalline basement exists at a depth of between 200–2,300 m below the land surface (Figure 19.1). The sedimentary cover consists of the deposits of all geological systems.

The crystalline basement is composed of Proterozoic rocks. The geotechnical properties of this massif have been investigated in the southeastern part of Lithuania. A Lower Cambrian clay formation occurs in Eastern Lithuania at a depth of at least 300–500 m. Also, a layer of Permian sulphate deposits (40–60 m thick) occurs in the southern and southwestern parts of Lithuania, with most of these deposits consisting of anhydrite. The northern edge of the upper Permian rock-salt basin can be found in the southwestern part of Lithuania, with salt (according to seismic data) in the form of salt domes. Lower Triassic deposits occur in the western and southwestern parts of Lithuania at a depth of 250–350 m and are more than 200 m thick.

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Chapter 20

Retrievable Disposal of Radioactive Waste in The Netherlands

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20.1. INTRODUCTION

In The Netherlands, a research program on retrievable disposal of radioactive waste was initiated in line with the 1993 policy directive of the Dutch Government. This directive states that underground disposal of highly toxic waste (including radioactive waste) is permitted, provided it is retrievable for a long period of time. The Commission of Radioactive Waste Disposal (CORA) has been established to coordinate such a program aimed at a comparison of the merits of long-term surface storage with (deep) underground disposal in various host rocks. In addition, attention had to be given to the prospects of transmutation of actinides and direct disposal (without reprocessing) of spent fuel from the research reactors as well as to any other risk reduction methods.

Research focused on several options for retrievable storage or disposal: long-term aboveground or underground in either rock-salt formations or deep clay deposits. For each of these options, the retrievability and safety aspects have been evaluated. Obviously, safety is a crucial prerequisite for any option. Amultibarrier system consisting of artificial and natural barriers has to ensure isolation of the waste to avoid the creation of a hazard. Preferably, a disposal facility should be fail-safe, in case of the possible loss of human control over the facility.

Retrievability of the waste allows future generations to make their own choices, but is dependent on the facility being kept accessible for a long time for inspection and monitoring. However, it entails a greater risk of exposure to radiation and requires a long-term organizational effort involving maintenance, data management, monitoring, and supervision. Particularly in the case of underground disposal, retrievability will make construction and operation more complex and more costly.

This report includes an estimate of the costs and briefly addresses the prospects of other risk-reduction methods. The program has greatly benefited from collaboration with research institutes in Belgium and Germany. International collaboration is also considered essential for the future. The task of the CORA Commission was of a technical-scientific nature; however, in view of public awareness and sensitivity over the waste-disposal issue, the Commission included two studies on the social and ethical aspects.

The CORA research program is concerned primarily with highly active waste, because this forms the greatest challenge. The main types of waste are vitrified high-level waste (HLW) from the nuclear power plants at Dodewaard and Borsele, and spent fuel (SF) from the Dutch reactors for medical and scientific research in Petten and Delft. The total volume of this highly active waste is approximately 125 m^3. The bulk of nuclear waste is low-to-medium active and has short half-life times and decays within a few hundred years to nonradioactive industrial waste. As a result of the high percentage of enriched uranium present in the SF, the International Atomic Energy Agency (IAEA) nonproliferation regulations also apply to the storage and disposal of this material.

The research program includes 21 projects, with contributions from 20 research institutes both in The Netherlands and abroad. The program was carried out during the period 1996–2000 at a cost of around € 3.5 million. The Dutch Ministry of Economic Affairs contributed € 2.5 million, the European Commission € 0.3 million, and the research institutes themselves approximately € 0.7 million.

20.2. LONG-TERM SURFACE STORAGE

The study of long-term retrievable storage at the surface has been based on the existing Central Organization for Radioactive Waste (COVRA) facility at Borsele (near Flushing) and examines the technical feasibility of extending the storage period from the currently envisaged 100 to approximately 300 years.

20.2.1. Retrievability

Individual waste containers from a surface storage facility such as at COVRA can be retrieved comparatively fast and simply. Storage of HLW and SF may require complete or partial replacement of the building every 100 years. It is believed that adequate maintenance and replacement will indeed enable the life span of the existing and planned buildings to be extended to approximately 300 years.

20.2.2. Safety

A scoping study analyzing the consequences of a possible inundation of the COVRA facility within the next 300 years indicated that inundation would have only a minor effect on the radiation levels in the biosphere. Inundation will increase the radiation levels by only a fraction of the average ambient dose rate prevailing in The Netherlands (~2.4 mSv/y). The study assumes that only containers with low and intermediate-level waste (LLW/ILW) end up in the sea, remain intact, and only small amounts of the waste dissolve by diffusion. The consequences of a HLW-storage facility becoming flooded are also believed to be manageable. The storage facility is watertight up to a water level of 10 m above mean sea level. Sea-level rise resulting from climate changes is expected to occur gradually, such that additional protective measures can be taken in time.

20.3. RETRIEVABLE DISPOSAL IN A ROCK-SALT FORMATION

An underground facility is similar to a conventional mine and consists of shafts and galleries in a rock-salt formation at a depth of approximately 800 m. Such mines exist at many places over the world and are considered proven technology.

20.3.1. Retrievability

Salt shows creep under high pressure and will gradually compress any voids, thereby ensuring that the waste will eventually be sealed off regardless of human intervention. In other words, long-term disposal in salt offers the advantage of a fail-safe situation. However, creep is a slow process, and experience shows that, with adequate maintenance, mines in rock salt can be kept open for more than 100 years.

The disposal concept contains short horizontal disposal cells drilled into the side-walls of a gallery, each accommodating one HLW container. The annular space around the container is filled with crushed salt, and the cell is sealed off with salt blocks. Retrieval requires a special machine to drill out the salt around the waste container. It is strongly believed that retrievable disposal of HLW in a rock-salt formation is technically quite feasible.

20.3.2. Safety

A worst-case scenario of inadequate maintenance was considered. This scenario involved flooding of the underground disposal facility, causing the HLW to be dissolved and dispersed in the groundwater. It was calculated that under these conditions, the annual dose rate-level to a representative individual in the biosphere will gradually increase and reach a maximum after 100,000 years. This level, however, is still much lower than the ambient dose rate of approximately 2.4 mSv/y, even if the facility contains SF from research reactors.

Criticality (spontaneous nuclear chain reaction) does not present a safety hazard for the proposed concept for disposal of SF containers from research reactors. Water ingress in the form of saturated brine will actually reduce the criticality factor to a value below 0.4, because of the high absorption cross section of chlorine for thermal neutrons.

Laboratory investigations indicate that ionizing radiation from the waste may create damage to the crystal lattice of the salt by the creation of minute voids, giving rise to sudden energy releases. It appears that this effect can be reduced by sealing off the waste containers with materials that are less prone to radiation damage. Another study indicates that backfilling of waste containers with a hygroscopic material such as calcium chloride can reduce corrosion and gas production.

20.4. RETRIEVABLE DISPOSAL IN A CLAY DEPOSIT

In the past 20 years, Belgian research has gathered considerable experience in constructing underground caverns in clay deposits for radioactive-waste disposal. Since 1980, we have gained experience in constructing underground caverns in clay deposits for the disposal of radioactive waste in Belgium. The Belgian design therefore has served as a prototype, albeit adapted to permit retrieval of HLW from clay deposits at a depth of at least 500 m.

20.4.1. Retrievability

The limited technical data available for clays in The Netherlands were supplemented with data on Belgian clay deposits. The disposal and retrieval concept is essentially the same as for rock salt, except that in this case clay/bentonite is used to seal off the disposal cells and the galleries must be supported by a concrete wall. Moreover, the containers must either have an extra overpack or be placed in a lined disposal cell as a protection against corrosion by moisture from the clay. Technical implementation of the adapted design seems feasible.

20.4.2. Safety

Assuming the same worst-case scenario as for rock salt, it was found that, for HLW disposal, the maximum annual dose rate to a representative individual in the biosphere occurs after about 200,000 years and will be higher than for salt, but remains still low in comparison with the ambient dose rate. The calculations are based on assumptions that may be on the pessimistic side, but which were adopted for want of reliable data on the properties of deep clay layers in The Netherlands.

In disposing of SF containers from the research reactors, ingress of water cannot be ruled out. Criticality can be avoided by using small containers holding a low number of spent fuel elements, or by filling the spaces in the containers with special materials.

20.5. OTHER OPTIONS TO REDUCE RISKS

Internationally, reduction of the effective half-life of the waste is investigated by separating highly radiotoxic, long-lived radionuclides (such as the actinides plutonium, americium, uranium, and neptunium) from the bulk of the waste and transmuting them into radionuclides with a much shorter half-life. For radiotoxicity reduction, we are doing research on the possibility of immobilizing those radionuclides that are hard to transmute and relatively mobile (such as caesium, technetium, and iodine). The immobilized materials should keep their properties during long-time disposal. These investigations show some promise, but are yet in an early stage of development.

At present, the transmutation or "burning" of plutonium is effected by using mixed oxide (MOX) fuel (a mixture of uranium and plutonium) in existing nuclear reactors. Eventually, however, accelerator-driven reactors would be required to burn the actinides more efficiently. Most of the Dutch HLW is reprocessed, i.e., the uranium and plutonium has been subtracted from the waste for potential reuse. The remaining waste is presently embedded in molten glass or cement, and separation and transmutation would therefore be difficult.

20.6. SOCIETAL ASPECTS

In the community debate on disposal of radioactive waste, risk perception is a key element. The advantages of retrievable disposal are that no irrevocable decisions have to be taken, that it allows continuous monitoring, and that the possibility of alternative solutions remains open as long as the facility is accessible. Moreover, the community at large has to be involved more closely in the technical and societal aspects of the waste issue.

A scoping study into the societal aspects, including an inquiry among Dutch environmental organizations, confirms that the waste problem is associated with the negative image of nuclear power and with the fear that solving the waste problem could imply renewed deployment of nuclear energy. These organizations express a lack of confidence in the feasibility and safety of underground disposal. Experience in other countries where stakeholders were invited to express their opinion on disposal options and sites indicates that it is very difficult to reach a consensus.

20.7. COST ASPECTS

The cost of extending the life span of a surface storage facility such as COVRA(from approximately 100 years to about 300 years) amounts to some 90 M€ in capital and operating expenses. Constructing, operating, and closing down a retrievable underground disposal facility in rock salt will cost about 250 M€, and in clay deposits about 600 M€. Keeping an underground disposal facility open for retrieval of the waste would entail an annual expenditure of around 1.8 M€ for maintenance, management, and inspection.

20.8. INTERNATIONAL COOPERATION

Joint international research on the disposal of radioactive waste is essential in view of the complexity of the problem. Proving the technical feasibility and safety of disposal concepts requires extensive and expensive testing under conditions. Countries studying the feasibility of underground disposal in similar rocks (granite, clay or salt) therefore often combine forces, so as to optimize the use of knowledge and resources and enhance the quality of research.

To demonstrate the safety of underground disposal options, borehole samples from salt or clay deposits are necessary, as well as tests in an underground laboratory. The Netherlands lacks such a laboratory, and collection of data for this purpose by drilling is not permitted. Collaboration therefore is proposed with countries where these restrictions do not exist, such as Germany (salt) and Belgium (clay). This is also desirable in view of our relatively small volume of radioactive waste. Such a collaboration also supports the European Union concept, but will have considerable financial implications. Other reasons for cooperation on a European level are the prospects of a regional European approach that might lead to an optimum solution, both from a safety and an economic viewpoint, and optimizing environmental protection beyond national borders.

Another issue requiring a joint European approach stems from the more stringent regulations regarding radioactive material recently issued by the European Commission. This would lead to large volumes of residues of a natural origin being classified as low-level material, for which no storage or disposal concept as yet exists.

Most of the current international cooperation projects were initiated through international organizations such as the European Union (EU), the International Atomic Energy Agency (IAEA/United Nations) and the Nuclear Energy Agency (NEA/Organisation for Economic Co-operation and Development). It will be desirable for The Netherlands to extend this co-operation in the future. Our particular interest lies in probabilistic analysis of retrievable disposal, participation in experiments in underground laboratories, developing monitoring techniques and strategies, and in further development of ways to assess the societal/ethical aspects.

20.9. CONCLUSIONS AND RECOMMENDATIONS

20.9.1. Conclusions

Retrievability
The studies performed did not reveal any factors prohibiting the technical feasibility of the options for retrievable disposal (long-term surface storage and deep underground disposal in either salt or clay deposits). For the underground options, retrievability requires extra facilities. In addition, a fall-back surface storage facility may be needed to accommodate any waste retrieved from the underground disposal facility. Based on today's knowledge, retrievability can only be guaranteed for a few hundred years.

Comparison of the Options
The data used are more reliable and the retrieval operation simpler for surface storage than for underground storage. However, in contrast with underground disposal, the surface option lacks the natural multibarrier and fail-safe features. Therefore, eventually, underground disposal will be necessary in any case. Moreover, compared to the data for rock salt, the basic data for clay are less reliable and the galleries must be supported.

Safety
For the case of long-term surface storage, only the risks of flooding of a properly maintained COVRA facility have been studied. On the basis of comparatively favorable assumptions, a very low annual dose rate was found. A worst-case scenario considered for underground disposal is that of inadequate maintenance, resulting in flooding of the facility. Under these conditions, the annual dose rate to a representative individual in the biosphere would reach a maximum after some 100,000 years that is higher for clay than for rock salt, but still much lower than the ambient dose rate in The Netherlands (of about 2.4 mSv/y).

The calculations did not take into account the probability of these scenarios, which can be expected to be much smaller than 1. On the other hand, part of the basic data is subject to a large margin of uncertainty.

Risk Reduction through Waste Treatment
In the long-term, processes for separation, transmutation, and immobilization of actinides and fission prod-

ucts may help to solve the waste problem. However, these processes are at an early stage of development and far from proven. Moreover, in The Netherlands, HLW remaining after reprocessing of the spent nuclear fuel is generally embedded in molten durable glass, most likely precluding application of the processes mentioned.

Societal Aspects
The public perception of the waste problem is influenced by the negative image of nuclear energy, the fear of renewed nuclear power generation, and a lack of trust in the feasibility and safety of underground disposal. Also, an evaluation of the various options for retrievable disposal against a set of ethical/societal criteria has not yet taken place. This is considered necessary to reach an acceptable solution to the radioactive waste problem.

Costs
The long-term waste management process would probably include a phase of surface storage, followed by retrievable underground disposal and final disposal. Total costs can only be calculated for a specific combination and timing of the different storage and disposal phases.

International Cooperation
Joint international research into disposal of radioactive waste is essential, because of the complexity of extensive and costly experiments in an underground laboratory. These facilities are available in (for example) Germany (for rock salt) and Belgium (for clay), but not in The Netherlands. Nor is it permitted in The Netherlands to collect data by means of borehole sampling.

Cooperation is also desirable with a view to a possible regional solution of the waste problem, common environmental problems, and the fact that all EU countries are confronted with the more stringent regulations of the European Commission.

20.9.2. Recommendations

On the basis of these conclusions, the Commission recommends continuation of the research program, covering both technical and societal aspects, in cooperation with other countries, notably Belgium and Germany.

Technical Aspects
Further research should focus on:

1. Analysis of all potential hazard scenarios for both the surface and the underground options, taking into account the probability of occurrence
2. *In situ* experiments in underground laboratories to study the responses of salt and clay to the combined effects of pressure, temperature, and radiation under *in situ* conditions
3. Developing, constructing, and testing the monitoring systems for the period of retrievability, for all options
4. Investigation of an integrated waste management concept incorporating storage and disposal facilities, and providing for different final destinations of the various types of waste (including large low-level waste volumes of natural origin)
5. Further evaluation of the long-term risks associated with flooding of the COVRA site; assessment of long-term surface storage elsewhere in The Netherlands
6. Evaluation of storage in shallow underground bunker-like structures
7. Investigation of the consequences of radiation damage within rock salt under *in situ* conditions.

Ethical and Social Aspects
An eventual acceptable solution for the waste problem will only be achieved if, in a public debate, the social and technical aspects of nuclear waste disposal are considered on an equivalent basis.

This calls for:

1. An inventory of all aspects and stakeholders involved in the decision-making process
2. Development of relevant societal/ethical criteria to evaluate the options for retrievable disposal
3. An incremental decision-making procedure under independent supervision, involving all stakeholders at an early stage, based on all information available, avoiding any preconceived ideas, and supported by sociological/ethical expertise.

Chapter 21

The Polish Concept of Radioactive Waste Disposal

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21.1. INTRODUCTION

The demonstration of proper management of radioactive waste, including high level waste (HLW) and spent nuclear fuel (SNF), is a prerequisite for the acceptance of nuclear power in Poland. Poland is one of the few European countries without nuclear power. Some amounts of HLW and SNF have been generated by two research reactors. However, when considering the need to diversify energy sources for the Polish energy system, the authorities cannot ignore the nuclear power option.

A Strategic Governmental Program entitled "Radioactive Waste and Spent Nuclear Fuel Management in Poland" was conducted from 1997 to 1999 to develop the national strategy for solving both current and future problems of radioactive waste management. This program was coordinated by the Polish National Atomic Energy Agency and covered the following issues:

- Changing and updating the existing national regulations
- Siting for both near-surface and deep geological repositories
- Development of technologies and techniques for safe storage of SNF from research reactors, elaboration of new technologies for radioactive waste processing and management
- Prognosis (the analysis of radioactive waste and SNF management assuming the development of nuclear power in Poland)
- Public information and socio-economic issues (information to society about radioactive waste management).

21.2. SITE SELECTION FOR A DEEP GEOLOGICAL REPOSITORY

Following the example of countries with advanced radioactive waste management, Poland is considering disposal of HLW and SNF in a deep underground repository (Janeczek and Wlodarski, 2000). Criteria for selecting the site suitable for the geological repository were determined in accordance with International Atomic Energy Agency (IAEA) regulations. The site-selection procedure began with exploring the possibility of radioactive waste disposal in now-operating underground mines after their closure. That idea, however, was soon abandoned on the grounds of environmental safety.

After reviewing the geology of Poland from the perspective of radioactive waste disposal, 44 rock formations were selected for further examinations, including 17 sites in igneous (mainly granitic) and metamorphic rocks, 7 sites in shale, and 20 sites in salt deposits. More detailed evaluation of the acceptability of those rock formations eliminated most of them from consideration as a site.

The most promising regions selected during the first stage of siting included crystalline rocks in the basement of the East-European platform in northeast Poland, shale in the Foresudetic Monocline, and salt domes within the Zechstein salt formation in the Polish Lowland (Figure 21.1). Unfortunately, knowledge of the geology of northeast Poland is not detailed enough at present to gauge its suitability for radioactive waste disposal. In particular, hydrogeological conditions in crystalline rocks are not well known. Therefore, while not entirely rejected, the crystalline rocks of northeast Poland were not considered further in the evaluation procedure. However, they may (if necessary) be reconsidered as potential host rocks for radioactive waste disposal, provided that more is learned about their geology.

During the second stage of siting for potential sites, four geological structures were chosen as promising. These are Triassic clay rocks in southwest Poland and three salt domes in central Poland. Preliminary geophysical investigations and study of archival geological and hydrogeological data allowed us to select candidate sites within each of those geological structures.

Based on the general criteria for radioactive waste disposal in shale (a minimum thickness of 200 m for shale beds and at least 300 m for the thickness of overlying rocks), the candidate site was selected near Jarocin (Figure 21.1) in the Triassic shale, also known as the Upper Gypsum Beds (UGB) (Slizowski, ed., 1999a; 1999b). The UGB occur at a depth of 535 m. Thickness of the UGB ranges from 185 m to 240 m, with the shale composed mainly of illite and Mg-chlorite. The UGB

Figure 21.1. Location of candidate sites for the geological radioactive waste repository in Poland shown on a simplified geological map: (1) salt domes; (2) shale; (3) igneous rocks (granites and other)

are overlain by a 290 m to 331 m thick sequence of mudstones and shale. Above them, up to the ground surface, there are Tertiary and Quaternary clay rocks. The presence of a thick sequence of clay rocks above the UGB results in favorable hydrogeological conditions.

The following initial criteria were considered for the selection of candidate sites in salt domes:

- Depth of the occurrence for rock salt down to 600 m below the ground surface
- Minimum thickness of 400 m for overlying rocks
- Minimum thickness of 250 m for homogeneous rock salt
- Disposal zone ranging between 20 and 200 m thick
- Maximum depth of the repository <1,200 m below ground surface.

The above criteria were fulfilled by seven salt domes within the Zechstein salt formation in the Polish Lowland. However, four of them were rejected based on their unsuitable geological, hydrogeological, and socio-economic conditions. The most promising salt domes occur in Damaslawek, Klodawa, and Lanieta (Figure 21.1). Lanieta has been explored in greatest detail because of its economic importance for salt mining.

The Damaslawek salt dome is cylindrical in shape with an elliptical cross section. It covers an area of about 13 km^2 at a depth of 650 m (Garlicki, ed., 1998). The average depth of the salt occurrence is 470 m below ground surface. Clay, gypsum, and anhydrite cap the salt. The average thickness of the caprock is 200 m. The salt dome is overlain by Tertiary and Quaternary deposits, the thickness of which ranges from 178 m to 260 m. Country rocks consist of Upper Cretaceous sandstones and sands to a depth of at least 1,000 m. Seismic data suggest that the internal structure of the caprock is complex and its thickness varies considerably (Krzywiec, ed., 1999). The difference in the elevation of the caprock is as high as 100 m.

Rock salt in the Damaslawek dome is coarse-grained and contains lenses of K-Mg salts and anhydrite. In terms of waste disposal, the best host rocks are Z2 salts because they form a thick and homogeneous body; the radioactive waste repository may be located at a depth of at least 800 m below ground level. The central part of the salt dome seems to be the most suitable for a

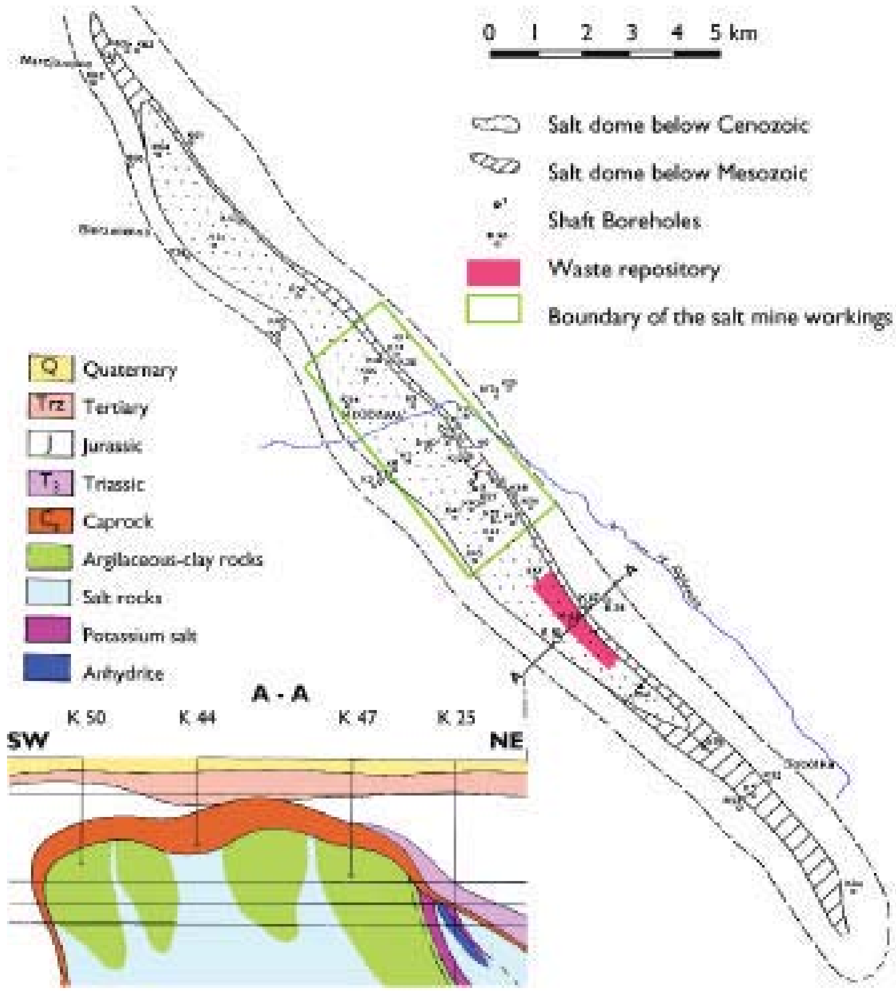

Figure 21.2. Geological map and cross section of the Klodawa salt dome (modified from Werner et al.and Poborski [Janeczek and Wlodarski,2000])

radioactive waste repository. The anhydrite-gypsum caprock is the thickest there (more than 150 m), and it is water-free. Two potential waste repository sites were suggested in the central part of the salt dome, based on geophysical data. They would be situated far from faults that border the salt dome down to a depth of at least 1,000 m, and they are far from the geologically young faults within the anhydrite-gypsum caprock.

The Klodawa salt dome is part of a 63 km elongated salt body, in the middle of which is the active Klodawa salt mine (Slizowski, ed., 1999a)—see Figure 21.2. This salt dome is one of the best known among salt domes in the Polish Zechstein Formation. The prospective repository location is suggested in the southern part of the Klodawa salt dome. Natural hazards in the southern part of the salt dome are probably the same as those experienced in the Klodawa salt mine, and the southern part of the dome has not been affected by mining activity. The geological structure in the southern part of the salt dome can be deduced from the geology of the mined portion in the middle section of the salt dome.

Klodawa is composed of folded rock salt intercalated with thin layers of potassium salts, anhydrite, and a thick sequence of argillaceous-salt rocks. Clay minerals make up about 50% of the argillaceous salt. The roof of the salt dome, in the area suggested for the repository site, is between 200 m to 300 m below ground surface. This salt is beneath an argillaceous gypsum caprock, which is overlain by Jurassic and Tertiary water-saturated rocks. Average thickness of the gypsum-argillaceous caprock is 100–150 m. The potential repository may be located some 2 km away from tunnels in the Klodawa salt mine (Slizowski, ed., 1999a), and it could be constructed at a depth between 600 m and 1,000 m below ground surface within the argillaceous-salt rocks, which are 500 to 600 m thick. Hydrogeological conditions in the Klodawa salt mine are rather safe. Presumably, similar hydrogeological conditions can be expected in the southern part of the dome. A two-level repository could conceivably be built in the Klodawa salt dome because of the large thickness of salt and argillaceous-salt rocks.

The Lanieta salt dome is a cylindrical body with steep walls. Its circular cross section has an area of 8.5 km^2, with the roof at a depth of 235 to 282 m below ground surface. Argillaceous-gypsum or anhydrite-gypsum rocks cap the salt dome. Based on observations in boreholes, the repository could be located within salt rocks of Z2 or Z3 and Z4 cyclothems, at a depth of 600 m to 1,000 m below ground surface. The gypsum-anhydrite cover is water-saturated in the fractured parts, and the water circulation system is complex. Water-saturated Tertiary sands and Quaternary sediments pose major hydrogeological threats. However, the hazard of the water inflow from the gypsum-anhydrite caprock is mostly local. Hydrogeochemical indicators suggest the mixing of meteoric water with ground water in the Tertiary rocks and with saline waters in the gypsum-anhydrite cover.

At this time, no ranking of candidate sites has been done because of the lack of necessary information. To obtain such information would require detailed and costly geological and geophysical investigations. Instead, the conceptual model for the geological repository was developed for both rock types, bearing in mind the specifics of the local geology.

A project for an Underground Research Laboratory (URL) was outlined in one of the salt domes. The unique feature of the Polish salt domes is the abundance of argillaceous-salt rocks, which combine the sorptive properties of clay minerals with the mechanical properties of salt (halite) (Slizowski, ed., 1999a). The construction of a URLwould enable *in situ* examinations of mechanical and thermomechanical properties of those rocks, which are considered a potentially excellent geological barrier. Furthermore, the hydrogeological conditions in salt domes, as well as the effectiveness of engineering barriers and filling materials, could be inspected, in addition to the experimental verification of the mathematical and geochemical models.

21.3. CONCLUDING REMARKS

Apart from solving several current and urgent issues related to radioactive waste management in Poland, the results from the Governmental Strategic Program provided the decision-makers in Poland with the assurance that the problem of disposing of HLW and SNF can be successfully solved. The candidate sites for disposing of radioactive waste have been selected, and further investigations will be necessary to evaluate their acceptability. The unique quality of the Polish salt domes, which consist of mixed clay and salt, is to be explored as potential host rocks for nuclear waste, as an alternative to clay and salt by themselves.

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Chapter 22

Long-Term Safety Assessment for Repositories in Rock Salt in Romania

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ABSTRACT. A long-term safety assessment of a repository has been performed for spent CANada Deuterium Uranium (CANDU) and spent light-water-reactor (LWR) fuel elements in a salt formation. A hypothetical repository site has been considered, using data from the European Union project Protected Area Geographic Information System (PAGIS) for all parts of the system (near field, overburden, and biosphere). Three scenarios have been taken into account: (1) subrosion as the normal evolution of the salt dome, (2) human intrusion into the cavern (representing future human actions) and (3) a combination of brine intrusion from the overburden and undetected brine pockets. Spent-fuel elements are assumed disposed of within large storage casks inside drifts.

For the sake of comparison, the same source-term model has been applied for both waste types but with different inventories of radionuclides and different heat productions. The key parameter for assessing long-term safety was the radiation exposure in the biosphere. The results of the calculations demonstrate that both types of waste can be disposed of safely. For the human-intrusion scenario and the combined brine-intrusion accident scenario, radiation exposures are below the limit of the German radiation protection law. In the subrosion scenario, however, for conservative values of the input parameters, the limit is exceeded in some calculations. This reflects the situation of a flat emplacement area that is quickly uncovered by subrosion, leading to high radionuclide concentrations. Because the subrosion model is very simple and overconservative, this result must not be thought definitive. Compared to PAGIS results, which were derived from the disposal of high-level waste in deep boreholes, these scenarios show higher consequences than were obtained in the PAGIS studies.

22.1. INTRODUCTION

In Romania, nuclear power is generated by Canadian-type reactors, which use natural uranium as nuclear fuel (in the following called CANDU fuel). The spent-fuel elements of these power plants are planned to be disposed of either in a salt or a hard-rock formation (Buhmann et al., 2000; Storck et al., 2001). This report investigates the long-term safety consequences of direct disposal in a deep repository located in salt. Results from this report are compared to those of a hypothetical direct disposal of spent light-water-reactor (LWR) fuel elements in salt. All of these investigations are closely related to the models and results of the European Union PAGIS exercise, which were published in 1988 (Storck et al., 1988).

Salt was chosen as the host rock because of some of its favorable long-term performance characteristics. Among these are low permeability and “creeping” behavior. Given its low permeability, the environment of the waste containers is usually dry because water flow is negligible. The creeping behavior of salt, also called convergence, results in the sealing of any faults (i.e., potential pathways for radionuclides) that may appear by mechanical stresses. Thus, under normal conditions, no release of radionuclides from the salt to the biosphere is expected.

Nevertheless, over a long time period, the uplift of the salt dome and subsequent subrosion by groundwater are possible mechanisms that may result in contamination of the biosphere. Also, some exceptional situations may occur that may result in a release of contaminants from the salt to the geosphere. We have modeled these possiblities with altered-evolution scenarios related to human activities in the future or to the presence of water near the waste containers. Three consequence calculations are performed: (1) Normal evolution of the repository, modeled by a subrosion scenario, (2) human intrusion, modeled by a storage-cavern scenario, and (3) a combined accident, consisting of a combination of brine intrusion from the overburden (through a fault in the rock salt) and from brine pockets near the emplacement drifts. The repositories for CANDU, or LRW fuel elements, are assumed to be different, mainly in the source-term data for spent fuel and the emplacement-field temperatures. Differences in the amount of waste for both countries are also taken into account. A common model for the source term is applied to both fuel types to improve comparability.

Model calculations were performed using modules of the computer-code package EMOS5. The older version of EMOS5 was used for calculation of the subrosion and human-intrusion scenarios. A new version, EMOS7, was used for the calculation of the combined-accident scenario (i.e., combined water intrusion via main anhydrite and via undetected brine pockets). Consequences are discussed mainly in terms of effective doses to future human beings. A purely deterministic approach was applied, and local sensitivity analyses were performed.

In this report, the three scenarios are described, the basic data for the repository systems are presented, and the most interesting characteristics of the modeling approaches for each scenario are briefly shown. A detailed discussion of the results for each of these scenarios follows afterwards.

22.2. MODELING

To assess the long-term safety of a repository system, scenarios have to be defined to describe the possible future evolution of the system. These scenarios are developed from combinations of features, events, and processes (FEPs) of the repository system. A systematic approach for scenario development has not been followed in this exercise; instead the procedure described in PAGIS (Storck et al., 1988) has been adopted. In PAGIS, two categories of evolution have been identified, "normal" and "altered," with the latter referring to disruptive events that may interfere with the normal evolution. In these types of evolution, time scale plays an important role. This results mainly from the different nature of influences that govern the development over a certain time period (and that require different kinds of models for evaluation), but also from the strong time dependence of the radionuclide inventory. The normal evolution that develops over a long time (over millions of years) is represented by the subrosion scenario. For medium-term development, in a time range between approximately 100 to approximately 1,000 years, possible human actions are taken into account (i.e., the human-intrusion scenario involving cavern leaching). For the short term, within the next 100 years, brine flow into the drift system of the repository is assumed (i.e., the combined-accident scenario). The scenarios selected for the present investigation are discussed in detail in the following sections.

A detailed discussion of the simulated long-term, short-term, and medium-term events at the repository site is provided in Buhmann et al. (2000) and Storck et al. (1988). Undisturbed rock salt is impermeable to water. Thus, it is (in principle) an ideal medium for the storage of wastes, because leaching of waste is not possible and no transport of contaminants can occur. However, because of mining activities during the emplacement period, the system could be disturbed, and the access drifts in the mine could become possible pathways for contaminant transport. Furthermore, heat-producing waste could lead to mechanical stresses in the host rock that could result in cracks. However, because of the creeping behavior of salt, open voids are reduced over time, a process that is accelerated by high temperatures that can be expected to occur in a repository containing high-level waste. Thus, such a repository tends to a final state in which all wastes are enclosed by impermeable rock material.

Nevertheless, release of contaminants is possible by normal evolution processes and by disruptive events. Three such scenarios are investigated in this exercise (Buhmann et al. 2000; Ionescu and Pavelescu, 1997a; 1998): (1) subrosion, (2) human intrusion, and (3) combined accident. These scenarios differ mainly for the near-field models. In the subrosion scenario, the geosphere is not explicitly modeled. In the human-intrusion and combined-accident scenarios, the geosphere and biosphere are modeled in the same way. Radionuclide transport through the overburden is modeled by a one-dimensional convection-dispersion equation, taking into account sorption processes (Buhmann, 1999). Calculation of radiological consequences starts from the radionuclide concentrations in the near-surface groundwater, which are converted to radiation exposure by means of dose conversion factors (Ionescu and Pavelescu, 1997a; 1998).

22.2.1. Subrosion Scenario

This scenario assumes that, over time, the salt dome is dissolved by groundwater in the caprock region. The subrosion of the top of the salt dome is balanced by halokinetic uplift, so that the depth of the salt dome top is unchanged. As a consequence, the emplacement field is continuously elevated and finally reaches the top of the dome. Waste is degraded and dissolved with salt and transported through the aquifers of the overburden to the biosphere. The emplacement sites of the repository are located some 500 m below the top of the present salt dome. Dissolution of the rock salt above the emplacement sites will take an estimated millions of years. Once this has occurred, the waste will come into direct contact with groundwater, and the remaining radionuclides will pass into the water. Residual barrier effects of the rock salt and the overburden determine the radiological consequences to the population.

Predicting the radiological consequences of such a scenario is complicated by uncertainties in predicting the subrosion rate, the long-term evolution of the future overburden and groundwater movement, and the consumption habits of future populations. Therefore, a simple model that enables calculation of the radiological consequences has been proposed for the subrosion scenario (Boese et al., 1996; Ionescu and Pavelescu, 1997b; Buhmann, 1999). The aforementioned uncertainties are dealt with by conservative assumptions that overestimate the consequences of the scenario. A range of parameter values, used in local sensitivity analyses, represents these uncertainties in the subrosion rate and in the dilution of the contaminated brine. To describe the processes in the future overburden, we selected modeling approaches that do not require detailed knowledge of future groundwater movement or the structure of the future overburden. Present-day consumption habits are used as a basis for intake of radionuclides via drinking water and food chains.

Evolution of the scenario can be divided into two time periods. The qualitative and quantitative description of these periods, as well as the mathematical modeling, is presented in Buhmann (2000) and Boese et al. (1996). According to Figure 22.1, the total volume of the emplacement area (i.e., the volume of salt dissolution) occurs evenly across the whole salt dome, while the dissolution of emplaced canisters surrounded by rock salt can be estimated by the dimensions of the original emplacement area prior to subrosion. All data are given in Buhmann et al. (2000), Ionescu and Pavelescu

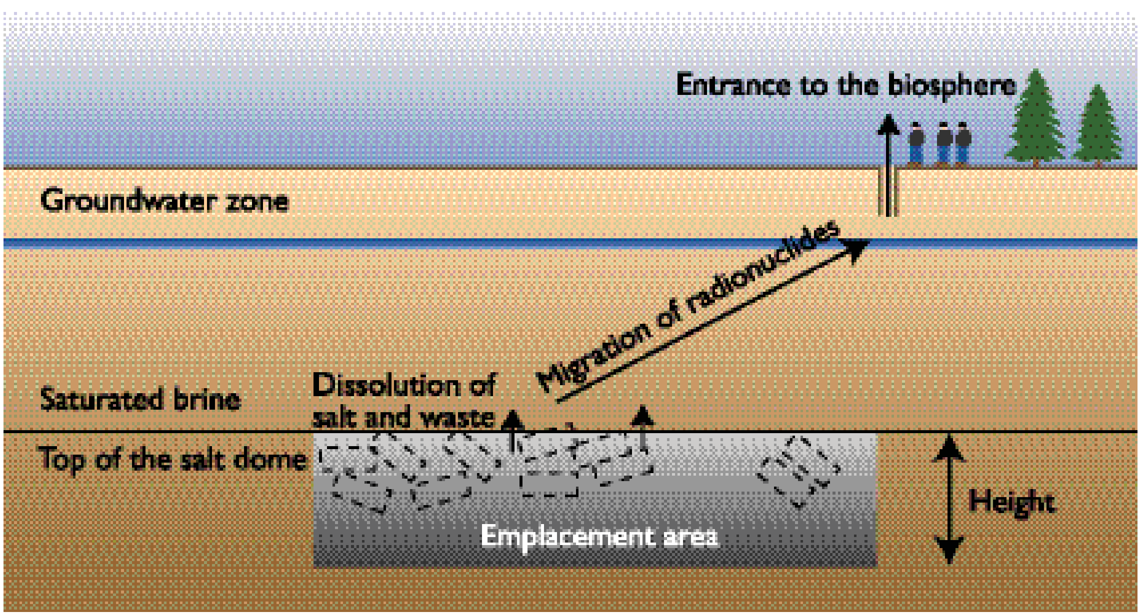

Figure 22.1. Subrosion scenario:model of radionuclide migration

(1998), and Boese et al. (1996). Release of radionuclides stretches over a time period that depends on the subrosion rate and the height of the disposal field. The range of the subrosion rates is: 2.2 mm/y–0.0005 mm/y, with disposal-field height of 10 m and release durations between 4.5×10^3 years and 2×10^7 years. A release duration of 3.0×10^5 years results in the best-estimate value of 0.033 mm/y.

22.2.2. Human-Intrusion Scenario

During the solution mining of a storage cavern, we assume that parts of a 1,000-year old repository containing radioactive waste would be laid bare. Waste containers in the affected region would fall to the ground of the excavated volume and be buried in the sump of insolubles at the bottom of the cavern. It is assumed that all containers would then be defective, and that corrosion of the waste matrix would start immediately. As a result of the continuing excavation process, an additional layer of insolubles would cover the containers. Thus, the sump would be divided into two parts: the bottom part containing insolubles and waste, and the upper part containing only insolubles. After the mining process, the brine in the cavern would be replaced by the medium in which it is to be stored. The operational phase of the cavern is assumed to last for about 50 years, after which the cavern would be abandoned. At this point, the storage medium is replaced by brine to support the mechanical stability of the cavern. Afterwards, the access borehole is sealed with concrete.

During the operational phase and afterwards, the sump volume and the open part of the cavern are reduced by convergence. The radionuclides that have been released from the containers in the bottom part of the sump are transported into the cavity by the convergence-driven flow of contaminated brine and also by molecular diffusion. Owing to the creeping of salt and the different densities of brine and rock salt, the cavern would not be mechanically stable after sealing of the access borehole. Consequently, the cemented casing-shoe of the access borehole would be exposed to high stresses and assumed to fail. As a consequence, a fracture may occur from the cavern to the overburden, establishing a pathway for contaminated brine. During the operational phase of the storage cavern, the pressure will be kept as high as possible to avoid a reduction of the cavern volume. However, since the duration of the operational phase is much lower than that of the post-operational leakage phase, for simplification purposes the hydraulic pressure assumed for the post-operational phase is applied to both phases.

The transport of radionuclides occurs mainly by advection in the brine and (to a smaller extent) by diffusion. Radionuclides could be transported from the bottom part of the sump into the top part of the sump and into the open volume of the cavern. As a retention mechanism for radionuclides along this path, sorption of the insoluble radionuclides is possible. (This possibility is not integrated into the model because of a lack of data.) Contaminated brine from the cavern would then intrude into the aquifers of the overburden. For simplicity, the same assumptions made for the combined-accident scenario (described below) are adopted with respect to the intrusion location and the pathway through the overburden.

Model calculations are performed with the computer code EMOS 5.02 (Boese et al., 1996) (a code that contains two special models for the simulation of the solution-mining scenario: one for the sump and one for the storage area of the cavern). The sealing of the cavern, with its continuous leakage, is modeled as a special physical effect by means of a resistor that limits the brine pressure to a fixed value.

The cavern is modeled as a series of three sections. The storage volume, called the cavity, is represented by only one section. The sump of the cavern is divided into two parts of equal size: the bottom part contains the wastes embedded in insolubles and brine, and the top part is filled with brine and insolubles only (no waste). For simplicity, it is assumed that the cavern has a cylindrical shape. Numerical values of the parameters for the best-estimate calculation, where they differ from those used for the combined-accident scenario, are given in Buhmann et al. (2000) and Storck et al. (2001).

22.2.3. Combined-Accident Scenario

The repository is modeled according to a hypothetical repository similar to that described in Ionescu and Pavelescu (1999) and Buhmann et al. (1991; 1994). The following is a short overview of the general modeling procedure. The source term for the release of radionuclides from spent-fuel elements is assumed to be the same for spent CANDU and LWR fuel elements, but some data are assumed to be different. The modeling of the source term is presented in detail in Buhmann et al. (2000). Finally, some relevant input data for the near field are compiled. The modeling and input data for the geosphere and biosphere are taken from Buhmann et al. (1991) and Johnson (1996) without changes.

The section structure of the near field is schematically shown in Figure 22.2. The main field, with its infrastructure, is followed by seven emplacement fields, each with 20 emplacement drifts for spent-fuel containers. It is assumed that in each emplacement field, four brine pockets occur, two pockets connected to the drifts at the end of the field and two pockets connected to the drifts in the center of the field. The infrastructure region of the main field is modeled as a drift system with an additional open void. Drifts are modeled as porous media with a flow resistance according to the permeability, and the additional voids are assumed to be accessible to brine but not contributing to the flow resistance. Because of the symmetry (as shown in Figure 22.2), a tree-like structure of the near field can be modeled.

22.3. INPUT DATA

All the input data are based on those given in Storck et al. (1988). The data for the geosphere and the biosphere are adopted without changes, because the radiation exposures are calculated only as a tool to compare the results. The intention is to compare the influence of different fuel types, and this influence is only given for the near field. For the near field, most of the data are taken from Buhmann et al. (1991; 1994). First, the data that are common to all scenarios are presented, including temperature data, container and fuel design data, the geometrical data of the repository, waste inventories, solubilization data, element-specific solubilities, and dose conversion factors. Additional data concerning the geometry of the repository (i.e., the section systems of the repositories and the specific geometrical section data) and the sorption data are presented in Ionescu and Pavelescu (1999) and Storck et al. (1988).

Finally, specific data for every scenario are presented. The subrosion scenario requires knowledge of start time of release from the repository, salt mass in the emplacement area, and concentration of salt in drinking water. The human-intrusion scenario requires geometrical data (radius and height of the cavern and also of the cavity, and final pore volume of the cavity), maximum brine-pressure value, initial and reference value for sump porosity, and time of spontaneous cavity fill-up. Two more specific data are necessary for a complete description of the combined scenario: time of brine intrusion from the anhydrite vein (unlimited brine intrusion) and time of brine intrusion from the brine pockets (limited brine intrusion).

22.4. RESULTS

The consequences of the three scenarios are calculated in two ways by: (1) best-estimate calculations and (2) local-sensitivity analyses. Best-estimate calculations are deterministic calculations, with best-estimate values for the input parameters and some conservative assumptions in the modeling. Results are presented as doses to individuals for each scenario, neglecting the probability

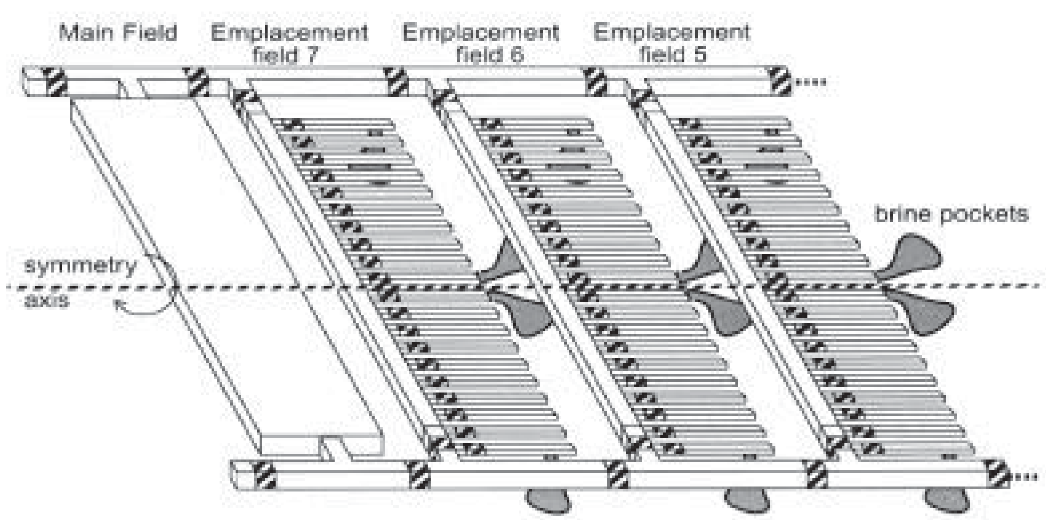

Figure 22.2. Part of the section system of the near field. The entire repository consists of seven emplacement fields.

of occurrence. Local sensitivity analyses are best-estimate calculations, with one single parameter being varied over its assumed 3-s range to study the behavior of the system. Results of local sensitivity analyses are presented as released nuclide masses or as individual doses for each scenario as a function of the considered input parameter. Performance assessments are made for both spent CANDU and spent LRW fuel repositories. In the following, the discussion is mainly focused on CANDU fuel. LRW fuel results are not discussed in detail, but are noted throughout for comparison purposes.

22.4.1. Best-Estimate Results

22.4.1.1. Subrosion Scenario

As described in Section 22.2.1, the subrosion rate determines the time of release of radionuclides as well as their radioactivity. In the best-estimate case, a subrosion rate of 0.033 mm/y is assumed, which results in a release time of 1.5×10^7 years. Time of release means the onset of a radionuclide flux into the groundwater. With this value, the resulting maximum of the radiation exposure from CANDU fuel is 3.7×10^{-4} Sv/y; for the LRW case it was 1.5×10^{-4} Sv/y. The main contributor to dose is U-234 (Ra-226) in both cases. Although these radiation exposures are rather high compared to the results of the combined- accident scenario (see Section 22.4.1.3 below), the conservative assumptions in the model have to be kept in mind. The results of the best-estimate calculations are listed in Table 22.1.

22.4.1.2. Human-Intrusion Scenario

In this scenario, temperatures near the waste containers after 1,000 years were roughly estimated. In these preliminary calculations for CANDU fuel, three different sets of temperature were chosen for the cavern sections. These calculations were performed to demonstrate the negligible influence of detailed modeling on the temperature fields. The results are listed in Table 22.2. Because of the generally low temperatures of the emplacement field after 1,000 years, it turned out to be of minor importance which of the three options was used. Thus, to be conservative, the calculations were performed with the first set of temperatures, which yields the highest radiological consequences. The temporal evolution of the scenario starts at t=0 years by leaching the waste matrices, because containers are assumed to be defective from the beginning of the scenario. After 50 years, when the cavern is sealed, the porosity of the sump has gained a value of 0.365 (0.217 for LRW fuel) owing to convergence. After that time, the brine pressure increases within less than 0.5 year, from the hydrostatic level of 10 MPa to the given maximum value of 15 MPa. At this pressure, it is assumed that the sealing of the cavern cracks and the contaminated brine starts to pass into the geosphere. The increase of brine pressure slows down the convergence, which reduces the brine flow.

The corresponding temporal evolution of the annual radiation exposure is shown in Figure 22.3. With CANDU fuel, the annual radiation exposure attains a maximum of 6×10^{-5} Sv/y at t = 494,000 years. The main contributions to the radiation exposure are from Np-237, followed by U-233, I-129 and Tc-99. For LRW fuel, the maximum is 1.26×10^{-4} Sv/y at t = 553,000 years. In this case, the main contributions to the radiation exposure are again from Np-237, followed by I-129 and Tc-99. In both cases, Se-79 dominates the radiation exposure at early times.

22.4.1.3. Combined-Accident Scenario

The best estimate values of the input parameters are the same for both fuel types (CANDU and LRW), with some exceptions regarding radionuclide inventories and (for some nuclides) solubility limits, as well as distribution of nuclide inventory between gap and matrix of the

Table 22.1. Subrosion scenario: Maximum total doses [Sv/y] for CANDU and LWR fuel as a function of subrosion rate and salt concentration in drinking water

Subrosion rate [mm/y]	Salt concentration [mg/l]					
	1		31(best estimate)		1000	
	CANDU	LWR	CANDU	LWR	CANDU	LWR
2.2	$1.7 \cdot 10^{-5}$	$4.8 \cdot 10^{-5}$	$5.3 \cdot 10^{-4}$	$1.5 \cdot 10^{-3}$	$1.7 \cdot 10^{-2}$	$4.8 \cdot 10^{-2}$
0.5	$1.4 \cdot 10^{-5}$	$1.9 \cdot 10^{-5}$	$4.4 \cdot 10^{-4}$	$5.9 \cdot 10^{-4}$	$1.4 \cdot 10^{-2}$	$1.9 \cdot 10^{-2}$
0.033 (b.e.)	$1.2 \cdot 10^{-5}$	$4.7 \cdot 10^{-6}$	$3.7 \cdot 10^{-4}$	$1.5 \cdot 10^{-4}$	$1.2 \cdot 10^{-2}$	$4.7 \cdot 10^{-3}$
0.005	$1.2 \cdot 10^{-5}$	$4.1 \cdot 10^{-6}$	$3.6 \cdot 10^{-4}$	$1.3 \cdot 10^{-4}$	$1.2 \cdot 10^{-2}$	$4.1 \cdot 10^{-3}$
0.0005	$1.0 \cdot 10^{-5}$	$3.5 \cdot 10^{-6}$	$3.1 \cdot 10^{-4}$	$1.1 \cdot 10^{-4}$	$1.0 \cdot 10^{-2}$	$3.5 \cdot 10^{-3}$

fuel elements. The time history of some relevant segments and the amount of water entering the entire repository is listed in Buhmann et al. (2000) and Storck et al. (2001). It can be seen that all the emplacement fields contribute to the radionuclide release, because brine from the brine pockets keeps most of the emplacement drifts open. Thus, the intruding brine from the main anhydrite can flow from the central field into the emplacement areas. Only those emplacement drifts not connected to brine pockets do not contribute to the release of radionuclides. Because no brine delays the convergence process, these emplacement drifts are closed by convergence, with the entering brine remaining trapped inside the drifts. Almost 15,845 m^3 of water enters into the repository, 11,347 m^3 intruding from the geosphere and 4,498 m^3 from brine pockets. About 11,857 m^3 of contaminated water is squeezed out from the repository over the time period of 1 million years, which is a little bit more than the total inflow from the geosphere. By convergence, almost 80% of the volume of the brine pockets is squeezed out into the emplacement drifts. The main contributions to the total dose are by I-129, followed by Ra-225, Np-237, and U-233. The temporal evolutions of the radiation exposures for several of the most important nuclides are shown in Figure 22.4.

The total dose rises to about 9×10^{-6} Sv/y at 440,000 years and is more than one order of magnitude below the limit of the radiation protection law.

22.4.2. Local Sensitivity Analyses

22.4.2.1. Subrosion Scenario

For local sensitivity analyses, the subrosion rate and the dilution factor have been varied (Buhmann et al., 2000; Storck et al., 2001. The calculated radiation exposures depend strongly on the subrosion rates because the times of release and thus the actual activities at these times change. The dilution of concentrated brine on the quality of drinking water has a similar strong influence, because (in agreement with the model assumptions) the radionuclide concentrations are diluted by the same factor (high dilution means low concentration).

22.4.2.2. Human-Intrusion Scenario

In this scenario, the results of varying the following parameters according to their given ranges were performed: reference convergence rate (Kr), maximum brine pressure (p max), diffusion coefficient, exponent in permeability-porosity relation (q), and solubility limits. The results were performed mainly for CANDU fuel, since the overall behavior is similar for LRW fuel. If larger differences between these fuel types occur, the results for LRW fuel are presented as well (Buhmann et al., 2000). The main concern with parameter variations is the influence these parameters have on the maximum of the radiation exposure. For some parameters, like the maximum brine pressure and the reference convergence rate, additional results are presented (Buhmann et al., 2000) concerning the released masses of representative

Table 22.2. Preliminary tests on the influence of section temperatures to radiological consequences: CANDU fuel

No.	Temperature Data	Time of Cavern Closure [y]	Time of Dose Maximum [y]	Maximum Dose [Sv/y]
1.	Sump temperature corresponds to a field with highest temperature Cavern has rock temperature at the reference level (i.e. no height dependence)	$8.89 \cdot 10^5$	$4.94 \cdot 10^5$	$6.80 \cdot 10^{-5}$
2.	Sump temperature corresponds to a field with highest temperatures Cavern has rock temperature at the centre of the cavity	$9.02 \cdot 10^5$	$5.05 \cdot 10^5$	$6.46 \cdot 10^{-5}$
3.	Every section has rock temperature at the corresponding depth	$9.03 \cdot 10^5$	$5.05 \cdot 10^5$	$6.47 \cdot 10^{-5}$

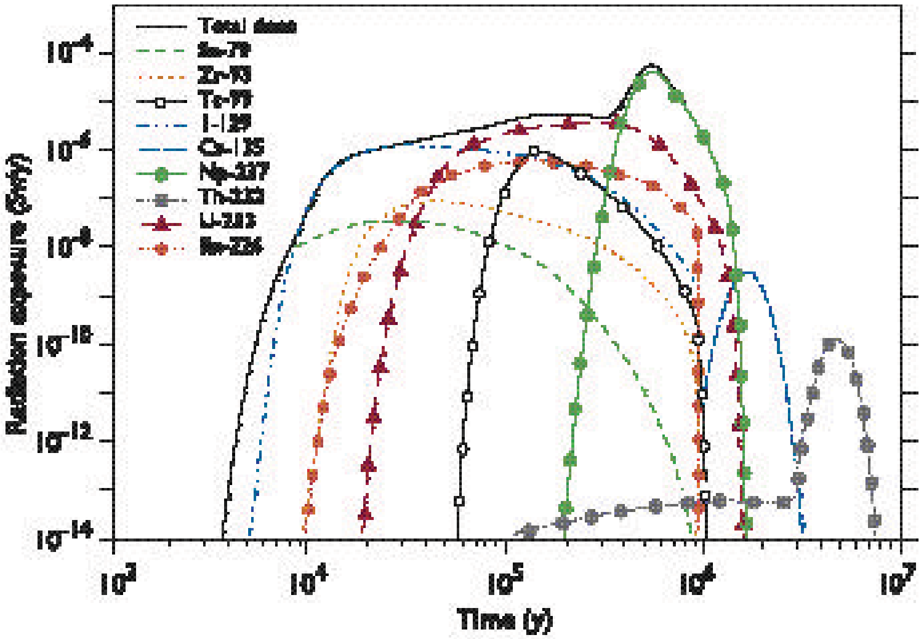

Figure 22.3. Human-intrusion scenario (CANDU Fuel)
Temporal e volution of the radiation exposure in the reference case

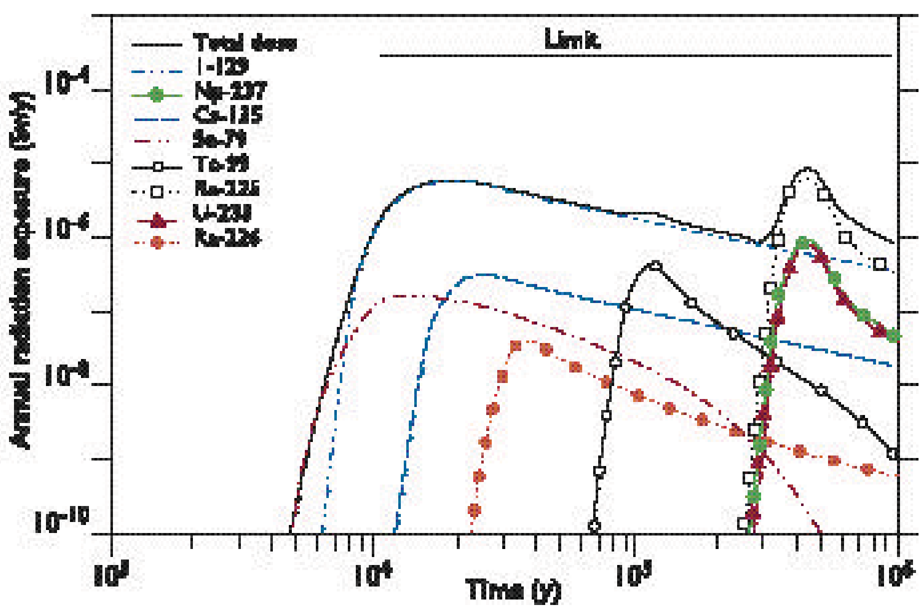

Figure 22.4. Combined accident scenario (CANDU fuel):
Temporal e volution of the radiation exposure in the reference case

radionuclides, the release rates out of the cavern, and the times at which sections reach the impermeable state.

22.4.2.3. Combined-Accident Scenario

Brine pockets are important because an early intrusion of brine into emplacement drifts results in an early mobilization of radionuclides and a slower convergence process (because of support by brine pressure). In recent probabilistic investigations, it was already shown that the volume of brine pockets is sensitive to the release of radionuclides. In the present calculations, at volumes of less than 25 m^3 for CANDU and less than 200 m^3 for LRW fuel, no release of radionuclides occurs at all. The time of brine intrusion is of particular importance, because the intruding water reduces the convergence rate, thus keeping portions of the repository open for longer times. The time of the brine intrusion resulting from the overburden into the repository has ranged between 1 and 1,000 years. The influence of the brine-intrusion time on radiation exposure is discussed in Storck et al. (2001). If the brine intrudes before 100 years, the main drifts are already filled with brine from the brine pockets. In this case, there is almost no influence on the radiation exposures compared to the reference case, the maximum total dose being about 1×10^{-5} Sv/y for CANDU fuel and about 1×10^{-7} Sv/y for LRW fuel. However, if the brine intrusion from the overburden comes after about 500 years for CANDU fuel (or 300 years for LRW fuel), the main field closes by convergence, and no release of radionuclides occurs at all, because there is no interaction between contaminated brine and the biosphere. The main contributors to radiation exposure are I-129, Cs-135, Ra-225, and Np-237.

22.5. SUMMARY AND CONCLUSIONS

Performance assessments for a hypothetical repository in a salt formation revealed differences in the calculated consequences, depending on whether spent CANDU fuel or spent LWR fuel is disposed of. Deterministic calculations and local sensitivity analyses with respect to some of the most influential parameters have been performed for three scenarios: subrosion, human intrusion, and combined accident. The differences in the modeling for CANDU and LRW fuel, respectively, have to do with the inventories and temperatures of the waste emplacement fields. Results have been mainly discussed in terms of effective doses. The calculated consequences for the LRW case are similar to earlier calculations of the PAGIS project of the European Union.

For the subrosion scenario, with best-estimate values of the input parameters, the radiation exposures are at the radiation protection limit of 3×10^{-4} Sv/y. A maximum radiation exposure of 3.7×10^{-4} Sv/y has been calculated for CANDU fuel and 1.5×10^{-4} Sv/y for LRW fuel, respectively. For both fuels, the main contributor to radiation exposure is U-234.

In the human-intrusion scenario, with best-estimate values of the input parameters, the maximum radiation exposure for CANDU fuel is 6.8×10^{-5} Sv/y and 1.3×10^{-5} Sv/y for LRW fuel, and is mainly caused by Np-237. Local sensitivity analyses have been performed to investigate the influence of some input parameters on radionuclide release to the overburden. In some variants, the maximum radiation exposures can rise to values higher than the best estimate value, but they are generally lower than about 2.5×10^{-4} Sv/y for both CANDU and LRW fuel. Thus, the consequences of the human-intrusion scenario are acceptable, especially because its probability of occurrence is low.

In the combined-accident scenario, an intrusion of brine from the overburden into the repository via the main anhydrite is considered in combination with a limited brine intrusion from brine pockets located near the emplacement galleries. Applying best-estimate values of the input parameters, the maximum radiation exposures for this scenario are 9×10^{-6} Sv/y for CANDU and 1×10^{-7} Sv/y for LRW fuel. The most relevant nuclides are I-129 and Ra-225 for CANDU fuel, and I-129 and Cs-135 for LRW fuel, respectively. Three parameters have been assumed to be important to radiological consequences in the combined-accident scenario: volume of brine pockets, time of brine intrusion from the overburden, and reference convergence rate. Maximum radiation exposures increase with increasing volume of the brine pockets. Below a threshold value of 25 m^3 for CANDU fuel (200 m^3 for LRW fuel), no release of nuclides occurs at all. The time of brine intrusion also has a threshold effect: if the brine intrudes later than about 500 years, no release of radionuclides occurs at all. The highest radiation exposures are close to the reference values and occur for brine intrusion immediately after closure of the repository. The variation of the reference convergence rate yields a maximum of the calculated radiation exposures of about 8 × 10^{-5} Sv/y for CANDU fuel and 4 ×10^{-5} Sv/y for LRW fuel at values of about 1 × 10^{-3} y^{-1}.

The results of the combined-accident scenario are different from the results of the subrosion scenario in that the radiation exposures for CANDU fuel are generally higher than for LRW fuel. The difference is that, in the combined-accident scenario (in the LRW case), the high inventory is “shielded” because only parts of the repository are involved in a radionuclide release (as described above). In the subrosion scenario, for both fuel types, the total inventories contribute to the release, and consequently the radiation exposure is lower for CANDU because of its lower inventory. The calculated radiation exposures for the human-intrusion scenario are between those for the subrosion and the combined-accident scenarios and are of comparable magnitude for CANDU and LRW fuel. In these cases, the inventories contributing to the release are almost the same, although the total inventories are different. But the dimensions of the cavern have an affect on inventories, which has a greater impact on CANDU than on LRW containers. In the combined-accident scenario, layout temperature turned out to be the most important parameter. It strongly affects the release from the repository and, finally, the radiological consequences to humans. Nevertheless, it must be emphasized that the temperature field for the CANDU repository is based on some rough approximations and should be recalculated in future studies.

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Chapter 23

Disposal of Radioactive Waste in Deep Geological Formations in Russia: Results and Prospects

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23.1. INTRODUCTION

In Russia, the main principles in the technical policy for radioactive waste management are first defined by the requirement to protect the environment and safety of present and future generations from the negative effects of radioactivity. In practice, the waste management also considers costs as an inevitable factor, owing to the severe economic situation in Russia.

The ultimate stage in the management of radioactive waste is its emplacement as solid waste in geological formations, with a high degree of multibarrier isolation from the environment. Our research program has been developed along these lines over the last two decades. Corresponding legal rules have been set up that consider international agreements and experience in other countries, and research programs have been developed and carried out. Disposal of liquid radioactive wastes in porous geological formations (reservoirs) has been carried out in Russia since 1963. This will continue, at the sites in operation, at least until 2010; after that, the storage sites will be shut down.

In parallel with the development of repositories in deep formations, radioactive waste disposal at currently available shallow sites, and any new ones of that kind, will be continued. At present, the design of shallow and deep repositories in permafrost is in progress. During the intervening stages before the final disposal of solid and solidified radioactive wastes, it will also be possible to store them in geological formations for long periods of time. Furthermore, these wastes may be buried or extracted for disposal at other sites. These storage sites may also contain spent fuel.

This report is concerned only with geological disposal, which implies the emplacement of radioactive wastes at depths of more than 50–100 m, in suitable geological formations and without any future withdrawal.

23.2. DEVELOPMENT OF RESEARCH ON THE DISPOSAL OF RADIO ACTIVE WASTE IN RUSSIA

The history of radioactive waste disposal in geological formations within Russia began as early as the middle 1950s. At that time, the creation of deep-injection storage projects for liquid radioactive wastes began. Problems with radioactive contamination of the environment in regions where large atomic industry enterprises are located, and the explosion of a surface tank with radioactive wastes in the Southern Urals in 1957, created the necessity to take urgent measures to isolate liquid radioactive wastes.

These facts compelled the Soviet Government to look for solutions to the radioactive waste problem, and disposal in deep geological formations was one of them. Experience with injection of wastes in porous formations was available from research and development of oil fields where the high confining ability of clays had been observed. Experience in investigations of uranium deposits had also demonstrated the high retentive capacity of sands and clays relative to radionuclides. A wealth of experience in the disposal of nonradioactive industrial wastes has been accumulated in the United States.

Initially, geological exploration was carried out in the areas of proposed disposal, to establish a database for storage projects and to obtain data necessary for their design. The geological exploration included geophysical studies, drilling wells and examining geologic columns, filtration tests, study of samples, underground waters, and wastes and their interaction with the geological environment.

It was established that deep storage projects could be created in the following regions of the atomic industry: the Siberian Chemical Combine (in the town of Seversk or Tomsk-7, Tomskaya obl.), the Mining and Chemical Combine (in the town of Zheleznogorsk or Krasnoyarsk-26, Krasnoyarsk region), the Research Institute of Atomic Reactors (in the town of Dimitrovgrad, Ulyanovskaya obl). The region of the Combine "Mayak" (in the town of Ozersk, Chelyabinskaya obl.) proved unsuitable for safe disposal of liquid radioactive wastes in limestone horizons at a depth of more that 1,000 m.

In 1963, an experimental disposal project commenced at the Siberian Chemical Combine, and results from this test have been used to construct other commercial storage projects for liquid radioactive wastes. Disposal of liquid radioactive waste at the Mining and Chemical Combine started in 1967, and at the Institute of Atomic Reactors in 1966. Some 47 million m^3 of liquid radioactive wastes, containing 30 million θBq (~9×10^8 Ci), have now been injected into deep reservoir formations. Analysis of the injection results has shown that these operations may be continued until 2010–2015. Then, the storage projects are to be shut down.

Simultaneously, with the operation of the above liquid radioactive waste projects, scientists were engaged in searching for alternative ways to dispose of radioactive solids and solidified wastes, and in developing technology for the solidification of the liquid wastes. Awide variety of geological conditions was considered, including:

- Available openings (shafts, adits) after mining is completed
- Openings driven for disposal of radwastes
- Natural cavities in geological media (caves)
- Cavities created by nuclear explosions in different kinds of rock (including rock salt)
- Deep disposal in permafrost.

Some unconventional ways of radioactive waste disposal were also considered, such as disposal in openings using radioactive decay heat to generate rock melt (in which the waste becomes submerged) and disposal in the immediate vicinity of volcanic vents.

The method finally accepted for disposal in geological formations was emplacement of solid and solidified wastes in underground workings driven especially for this purpose. Requirements for the formations that are suitable for this purpose have been formulated, and the possible disposal schemes and order of the principal steps in creating such schemes have been defined. In analyzing the problems of the atomic industry, the solutions to these problems were that the storage sites should be located in the immediate vicinity of the atomic power plants. This allows us to avoid the significant difficulties involved in waste transportation to the disposal site, public protests, etc.

Investigations for the purpose of developing repositories to dispose of solidified wastes in the region of the Southern Urals started in 1975. Drilling of wells and examination of core samples were carried out, and the geological structure and hydrogeological conditions were refined. The geological conditions were judged to be suitable for disposal of solidified radioactive wastes (SRW).

In 1990, investigations began at the Mining and Chemical Combine to create a deep repository for SRW. Geophysical examination and well drilling established the areas of hard rock convenient for a deep repository. Currently, a proposal is being considered to create a deep repository for SRW in shutdown mines (in mined-out deposits). Waste disposal in permafrost is also being considered as another possibility.

23.3. CRITERIA FOR SAFE DISPOSAL OF RADIO ACTIVE WASTE

The safety requirements that are imposed on the disposal of all kinds of radioactive waste in geological formations are based on legislative acts, including the act "On the Use of Nuclear Energy." Article 48, entitled "Storage or Disposal of Radioactive Wastes," states that the disposal process is to secure a reliable isolation (i.e., protection of present and future generations and biological resources from radiation effects exceeding established limits). The safety criteria and requirements take into consideration the principal propositions from documents and solutions of international agencies. The basic idea is protection of future generations and minimization of radiation exposure and risks (the "as low as reasonably achievable" [ALARA] principle).

Legislation by the Russian Federation (RF) has established the imperative, accepted by international organizations, that individuals living in the future are to be protected, and the protection is to be every bit as important and effective as that provided to individuals of the present.

The use of dose criterion is the current method for assessing radiation safety. The utilization of this procedure permits the realization of the fundamental safety criterion to be confirmed (or not confirmed). The dose criterion is established by the Radiation Safety Standards (NRB-99) and Fundamental Sanitary Rules for Providing Radiation Safety (OSPORB-99). On the basis of these standards, the effective dose equivalent limit for geological disposal of radioactive waste is not to exceed 0.01 mSv/y per person. The dose criterion is used simultaneously with the confining (localization) criterion, based on the RF Act "On Mineral Resources," which states that wastes are to be confined within the boundaries of the subsurface exclusion zone of the disposal site for a long period of time (more than 1,000 years).

The disposal safety assessment, using dose and confining criteria, consists of: defining the transport of waste components for different periods of time, evaluating actual and expected radiation doses, and comparing the results with the regulated values. The assessment is performed through direct measurements (in disposal-site operation) or predictive calculations and modeling (after the disposal site has been closed).

Probabilistic safety criteria are also used. This involves the probability for an individual to be subjected to excess exposure as a result of unlikely "destructive" events in the storage project. The exposure dose exceeding the threshold of the effects determined is to be limited by the value of 2.5×10^{-6} y^{-1}. It is difficult to use the probabilistic criteria when applied to geological radioactive waste disposal because predicting the probability of events in the future (hundreds and thousands of years) is very difficult, especially when related to the activities of mankind.

Most cases use qualitative and comparative evaluations for the probability of hypothetical events. The scenarios of the events are considered in detail, the modeling of waste migration and other processes is performed, and the dose of expected possible radiation is calculated. The requirements and criteria under consideration form a basis for norms, which are used to regulate the disposal of radioactive wastes in Russia.

23.4. EXISTING GEOLOGICAL STORAGE PROJECTS FOR LIQUID RADIO ACTIVE WASTE

The Siberian Chemical Combine, as well as the Mining and Chemical Combine, use sandy-clayey horizons that contain fresh water for their storage projects. The Research Institute of Nuclear Reactors uses limestones containing brines. The geological structure, hydrogeological conditions, and rock properties ensure that the wastes are confined in the storage formations.

The reservoir horizons are overlain by low permeable beds of clays that prevent vertical migration of the wastes. The travel velocities of the underground waters are slow and provide waste localization for a long period of time. Table 23.1 gives the main characteristics of geological conditions, volumes of wastes disposed, and areas over which waste components have been distributed in the reservoir horizon

There are two storage projects at the Siberian Chemical Combine: Site 18 for disposal of low-level wastes in Horizon II (340–386 m) and Horizon III (270–320 m), and Site 18a for disposal of intermediate-level waste (ILW) and high-level waste (HLW) in Horizon II (314–341 m). The natural groundwater velocity is 3–6 m/y. The radionuclides within the radioactive waste are retained by the rocks due to physical-chemical interactions. As a result, the migration of wastes is limited effectively and efficiently.

Only one storage project at the Mining and Chemical Combine (Severny site) uses two horizons. The first, at a depth of 355–500 m, is used for disposal of HLW and ILW; and the second, at a depth 180–200 m, is used for disposal of low level wastes (LLW). The natural groundwater velocity is 5–6 m/y in Horizon I and 10–15 m/y in Horizon II. The radionuclides are also retained by reactions with the rocks. To the west, the reservoir horizons are limited by a hydrodynamic barrier, formed by a fault plane, which acts as an obstacle in the movement of groundwater toward the Yenisei river.

Table 23.1. Description of deep well injection storage sites for liquid radioactive waste

Storage	Year Began	Type of Rock	Depth (m)	Natural Travel of Underground Water (m/yr)	Volume (Mln m^3)	Contour Area (m^2)	Storage Area (m^2)
Siberian Chemical Combine							
Site 18, Horizon II	1967	Sand	349-386	3-6	18,200	5.5	11.0
Site 18, Horizon III	1967	Sand	270-320	3-6	18,300	5.5	11.0
Site 18a, Horizon II	1963	Sand	314-341	3-6	5,050	1.5	3.5
Mining and Chemical Combine							
Site "Severny," Horizon I	1967	Sandstone	355-500	5-6	2,200	1.9	6.5
Site "Severny," Horizon II	1968	Sandstone	180-280	10-15	3,100	3.0	6.5
Scientific Research Institute of Nuclear Reactors							
Permeable Zone III	1966	Sandstone	1440-1550	<1	600	6.0	18.0
Permeable Zone IV	1977	Limestone	1130-1410	<1	1,660	1.5	18.0

In the Scientific and Research Institute of Nuclear Reactors, there is only one storage test site. In the initial stage of its operations, the third permeable zone at depths of 1440–1550 m was used, and in a subsequent stage, the fourth permeable zone at depths of 1130–1410 m was used. The horizons are located in a stagnant zone, where the groundwater velocities are less than 1 m/y.

Liquid radioactive wastes contain salts and radionuclides with short half-lives, ranging from tens of days to one or two years (isotopes of zirconium, niobium, ruthenium, cerium), and longer half-lives of about 30 years (strontium-90, caesium-137). The radionuclides with very long half-lives (plutonium-239, neptunium-237, etc.) are present only in microconcentrations, and prior to disposal, an additional extraction of them is made. The activity of long-lived fission products (with half-lives of hundreds and thousand of years: technetium-99, cerium-135 etc) is 10^{-6} of the total activity of wastes.

Note that with HLW, the short-lived nuclides contain 80–90% of the total activity. After 5–10 years in the underground reservoir, these wastes transfer into the category of ILW. The volume of HLW is less than 1% of the total volume of all wastes.

Systems for monitoring the geological formations were developed in the operation of the deep storage projects. The total number of monitoring wells is quite large: 273 at Siberian Chemical Combine, 70 at Mining and Chemical Combine, and 30 at Scientific and Research Institute of Nuclear Reactors. Monitoring includes hydrodynamic and geophysical observations, and groundwater sampling to identify the locations of waste components. From the monitoring results, the radioactive waste components are localized within predicted boundaries as well as the boundaries of subsurface exclusion zones. Modeling and predictive calculations have shown that the waste components will remain inside these boundaries for a period of approximately 1,000 years and more. Temperature monitoring in the reservoir horizons with high-level wastes shows that, under subsurface conditions, temperatures do not exceed the boiling point.

In the period 1995–1999, a number of international projects were developed to carry out an assessment of deep injection disposal of liquid radioactive wastes in Russia (Compton et al., 2000; EUR, 1997; EUR, 1999; Shestakov, 1999). From the data obtained, a conservative method of analysis revealed that the possible radiation burden on the general public, after thousands or tens of thousands of years, should not exceed ~0.01 of mSv.

Complications and preconditions of accidents in the waste disposal projects were caused by a deterioration in the condition of engineered systems (chiefly wells and pipelines) and improper choice of injection conditions and waste treatment. The most important of the complications were as follows:

- A serious gas release in a monitoring well at the experimental site of the Siberian Chemical Site in 1963
- Nonuniform distribution of wastes in the third permeable zone at the Scientific and Research Institute of Nuclear Reactors site and in Horizon II of the Mining and Chemical Combine, caused by high injection pressures under hydrofracturing conditions.

The consequences of these phenomena did not result in contamination of the environment outside the sanitary-protective zones. Analysis of hypothetical accidents in radioactive waste disposal has shown that the probability of their occurrence is low and the after-effects do not pose threats. The only risk from disposal of liquid radioactive wastes in geological formations is the migration of these liquid wastes from these formations into local drinking water.

After the injection of liquid radioactive wastes is completed, the storage projects are to be shut down to avoid their negative impact on the environment, and the technology for shutting down radioactive waste storage is now being developed.

23.5. REPOSITORIES FOR SOLID AND SOLIDIFIED RADIOACTIVE WASTE IN GEOLOGICAL FORMATIONS

To select a site for deep storage of high-level wastes in a repository, we examined five possible areas at the combine “Mayak” (Chelyabinskaya obl.) located near the enterprise. Volcanic rocks are widespread in the region, in the form of tuffs and porphyritic lava breccias that are low in permeability, heat resistant, and stable. All the areas are located within unique tectonic blocks with minimal jointing.

Construction of an underground research laboratory is considered the first step in developing a repository for solid and solidified wastes. The investigation of rock-massif properties *in situ* will be performed in this underground laboratory. The site proposed for the laboratory construction is located in the immediate vicinity of an area used for surface storage of vitrified wastes. Repository construction and commencement of waste disposal are planned to begin in 2030. Before that date, the wastes are to be stored at the surface.

A repository for solidified HLW from the Mining and Chemical Combine (Krasnoyarsk region) is to be constructed in the Nizhne-Kansky granitoidal massif of variable composition with prevailing biotite granites and granodiorites. Five areas have been selected for more detailed examination. It was established that the main factor defining the possibility of repository construction is the degree of tectonic disturbance of the granites and the surrounding metamorphic rocks. Tectonic fracturing should be minimal because it controls the hydrogeological conditions of the site and the possibilities for radionuclide transport. The available workings are planned to be used as an underground laboratory. A large volume of data has been accumulated on changes in hard-rock parameters under the influence of temperature and radiation. Complex investigations of the Nizhne-Kansky massif are in progress.

Proposals are also being considered for construction of a deep repository for solidified radioactive wastes in a region of mined-out uranium deposits at the Priargunsky-Mining Combine (Krasnokamensk, Chitinskaya obl.). Mining will be completed in 2020, and the available infrastructure can be used for repository operation. It is planned for new structures to be created for solidified waste emplacement and not to utilize available workings. The solidified wastes will be placed in storage sites of different types (e.g., chambers with access through shafts or adits, boreholes of large diameter drilled from the surface or from underground workings). The multibarrier protection of buried wastes will be designed to include the waste form (glass, mineral-like compositions) and packages (containers, sealed packages).

At the present time, a design for near-surface storage of solid radioactive waste in permafrost is being developed for the Novaya Zemlya archipelago. Isolation of solid radioactive wastes in deep boreholes of large diameter in permafrost will be one of the methods of disposal. Table 23.2 gives a general description of repositories for solid and solidified wastes that are currently being planned.

Table 23.2. Supposed deep repositories for solid and solidified radioactive waste

Repository	Year Begun	Type of Geological Formation	Depth Intervals (m)	State of Work	Eatimated Date for Putting the Repository into Operation
Combine "Mayak", Chelyabinskaya obl.	1975	Hard rock,porphirites	100-500	Geological exploration. URL being designed.	2025
Mining and Chemical Combine, Krasnojarsk region	1990	Hard rock,granites	100-1000	Detailed geological exploration	2030
Priargunskii mine	2000	Hard rock,diabase	100-800	Preliminary discussion	2030
Archipelago Novaya Zemlya	1995	Hard-rock,permafrost	30-100	Designing	2005-2010

23.6. CONCLUSIONS

Experience with radioactive waste management in Russia (including deep well injection of liquid radioactive wastes) has shown that disposal of wastes in geological formations is the safest and most reasonable method, in terms of protection from radiation effects, for the environment and the general public. However, the disposal of solid and solidified wastes in a hard-rock massif and rock salt according to this “classic” scheme requires significant costs and long periods of time.

Resolution of the repository construction issue will have to take into account the attitude towards waste disposal of broad sections of society, public organizations, and local governments. In this connection, locating storage projects in the area, or in the immediate vicinity, of enterprises of the atomic industry is favored today. The prospects for developing waste disposal in geological formations are that disposal of liquid radioactive wastes is to be completed in 2010–2015, and the storage projects will then be shut down. Geological disposal will not begin until 2025–2030; instead, the accumulation of solid and solidified waste in temporary storage projects will be carried out on the surface. Near-surface disposal of solid and solidified wastes will be carried out simultaneously with the construction of geological repositories that are expected to be put into operation after 2025.

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Chapter 24

Status of the Deep Geological Disposal Program in the Slovak Republic

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24.1. INTRODUCTION

During their operation, Slovak nuclear power plants (NPPs) have in total produced approximately 2,500 metric tons of spent fuel (expressed as heavy metal) as well as approximately 5,000 metric tons of radioactive waste considered unfit for the near-surface repository at Mochovce. This radioactive waste is obtained mainly from the operation and decommissioning of nuclear facilities as well as from medical, industrial, and research applications. These volumes of spent fuel and radioactive waste include that which is generated by two reactors operating at the Mochovce site. Construction of two additional reactors at the same location is uncertain at this time because of the loss of Slovak governmental support. In addition, the Slovak government recently decided to reduce the scale of NPPV1-operation, with a proposed shutdown scheduled for 2006 and 2008.

The Slovak national policy for management of radioactive waste (including spent fuel) that is unacceptable for storage in a near-surface repository follows Governmental Decision No. 930/1992. According to this document, these materials can only be disposed of in a deep geological repository after appropriate treatment. A more recent governmental decision (No. 5/2001) proposes two alternatives for the end of the fuel cycle:

- Construction of a deep geological repository for spent fuel and high-level radioactive waste (HLW) in Slovak territory
- Shipping and final disposal of spent fuel externally.

From an economic point of view, the first alternative—direct disposal after 50 years of interim storage—would seem to be more advantageous.

The Institute of Nuclear Research, Plc., has studied the possibility of siting a deep geological repository in the former Czechoslovakian Federative Republic since 1993. At that time, it was highly probable that the future Czecho-Slovak deep geological repository for spent fuel and high-level radioactive waste would be located in Czech territory because of much more suitable geological conditions. After the separation of Czechoslovakia in 1993, R&D for deep geological disposal of high-level radioactive waste and spent fuel in the Slovak Republic began in 1996 in close relation to previous activities. As a first step in development activities, 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.

Program activities are coordinated by Decom Slovakia, Ltd. The project formally consists of the following key areas (current contractors are given in parentheses):

- Design and implementation (EGP Invest, Ltd. Uhersky Brod, Czech Republic)
- 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 (VUJE Trnava Inc., 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), and
- Coordination (Decom Slovakia, Ltd., Trnava, Slovak Republic)

Crystalline and sedimentary host environments were investigated in selected areas. 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 deep geological repository is expected by 2037.

24.2. PROJECT STATUS

24.2.1. Design and Implementation

EGP Invest, Ltd., has performed the preliminary technical design of a deep geological repository for a hypothetical site and two alternative geological host environments (sedimentary and crystalline). Conceptual ideas for the deep-geological-repository operational phase focused on various aspects of surface and underground activities presented from technological, economic, and feasibility perspectives. These activities focused on transportation and reception of spent fuel 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 a deep geological repository 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 in the Slovak Republic. 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 more appropriate solution at this time. A preliminary feasibility study based on knowledge and experience acquired in the Slovak disposal program, current status of mining technologies, and worldwide experience has also been prepared by EGP Invest, Ltd.

24.2.2. Source Term

Nuclear Research Institute, Plc., has described the physical and chemical properties (radionuclide species and their quantities) of the WWER-440 spent-fuel assemblies after interim storage. Activity of selected isotopes, total spent-fuel assembly activity, fuel-assembly thermal power, and the contribution of selected isotopes were calculated (using the ORIGEN 2.1 code with appropriate data) over times of 5, 10, 50, 70, 100, 1,000, and 10,000 years after refuelling. The neutron source from actinides and their daughter products in a spent-fuel assembly, in part resulting from (alpha, n) reaction and spontaneous fission, were determined as well. 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/t U up to 46,000 MWd/tU. These data will be used as input for the repository concept, disposal container design, and the spent-fuel-disposal safety assessment.

Possible mechanisms of radionuclide leaching from spent fuel 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 deep geological repository). Bibliographical 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 spent-fuel cladding in a deep geological repository environment have also been summarized. It was found that zirconium-niob cladding could be considered an effective barrier against radionuclide release.

Current activities are focused on investigating mechanisms of radionuclides release from spent-fuel cladding and HLW glass and cement matrices in a repository environment, as well as identifying a critical group of radionuclides (activation products ^{14}C, ^{300}l, ^{59}Ni, ^{63}Ni, and ^{10}Be, fission products ^{129}I, ^{135}Cs, ^{79}Se, ^{99}Tc ^{107}Pd, ^{126}Sn, ^{146}Sm, ^{93}Zr, ^{94}Nb, and ^{93m}Nb, and actinides ^{210}Pb, ^{226}Ra, ^{229}Th, ^{230}Th, ^{231}Pa, ^{233}U, ^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{237}Np, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am, ^{243}Am, ^{245}Cm, and ^{246}Cm) and their characteristics. The instant release fraction is shown by ^{129}I, ^{135}Cs, ^{99}Tc, ^{79}Se, ^{126}Sn, ^{3}H, ^{14}C, ^{137}Cs, ^{90}Sr, ^{39}Ar, ^{42}Ar, ^{85}Kr, ^{81}Kr, and ^{40}K.

24.2.3. Near Field

The first step in a near-field study was a critical review of the current status of this assessment, engineered-barrier modeling, available information on the materials suitable for engineered barriers, and the disposal container. Physical and chemical properties of 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., pre-

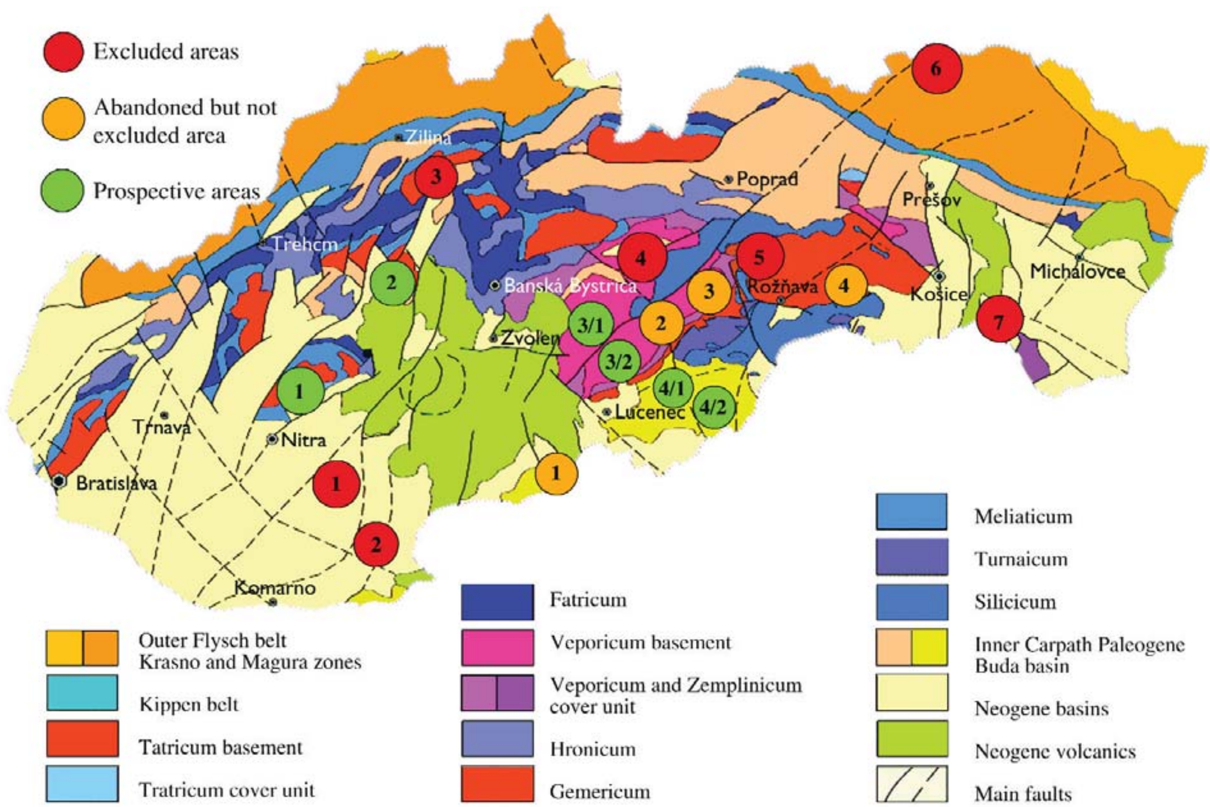

Figure 24.1. **Simplified tectonic sketch of Slovakia and status of deep geological repository siting in 2001. Excluded areas (red circles): (1) Zlate Moravce-Vrable , (2) Zeliez ovce , (3) Lucanska Mala Fatra Mountains, (4) Fabova hola Ridg e, (5) Nizna Slana, (6) Zbor ov, and (7) Bysta. Abandoned but not excluded areas (orange circles): (1) Ipelska kotlina Basin, (2) Revuca, (3) Rochovce , and (4) Prak ovce–Poproc . Prospective areas (green circles): (1) Tribec Mountains, (2) Ziar Mountains, (3/1) Veporske vrchy Mountains, (3/2) Stolicke vrch y Mountains,(4/1) Rimavska kotlina Basin,and (4/2) Cer ova vrchovina Upland**

pared the first proposal of a disposal container with seven WWER-440 spent-fuel assemblies. 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 spent fuel would be 7.7 metric tons. Container handling will require additional shielding to ensure a surface effective dose of 0.1 mS/h. The total number of disposable spent-fuel assemblies is expected to be 3,280.

24.2.4. Far Field

Worldwide experience in modeling geological barriers and groundwater flow (dominant factors in radioactive transport) were reviewed by the Geological Survey of the Slovak Republic and Nuclear Research Institute, Plc. 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. These models reflect the current status of knowledge about geology, petrography, seismicity, neotectonics, hydrogeology, and the geochemistry of given sites. Interactions between host environment (granites and clays) and engineered barriers, as well as the possible alteration of the host-rock environment and engineered-barrier materials (induced by expected hydrogeochemical processes), were analyzed.

24.2.5. Siting

24.2.5.1. Brief Outline of Geological Conditions for Repository Siting in Slovakia

Much of Slovak territory (49,016 km^2) is located in the mountain chain of the Western Carpathians. The Carpathian arc is a tectonically complicated Alpine-type geological structure within the Alpine chain of Europe. From a geologic point of view, the Western Carpathians are structured into several tectonic units (Figure 24.1). Only a few of these units contain rock environments potentially suitable for an HLW/spent-fuel repository site.

The Tatricum and the Veporicum large-scale tectonic units of the Central Western Carpathians include the Hercynian crystalline basement with autochthonous Late Palaeozoic-Mesozoic sedimentary cover (Figure 24.2). The Tatricum unit contains isolated cores of the crystalline basement (granitoid and metamorphosed rocks) partly covered by autochthonous and allochthonous 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 environment of this unit (mostly granitoids) can potentially provide suitable sites for construction of a repository.

A characteristic feature of the Western Carpathians is the Neogene post-tectonic basins infilled with Miocene sediments (predominantly clays, claystones, sands, and sandstones). The overall thickness of Neogene sediments is several thousand meters. The Neogene pelitic sediments are also potentially suitable for siting a repository in Slovakia.

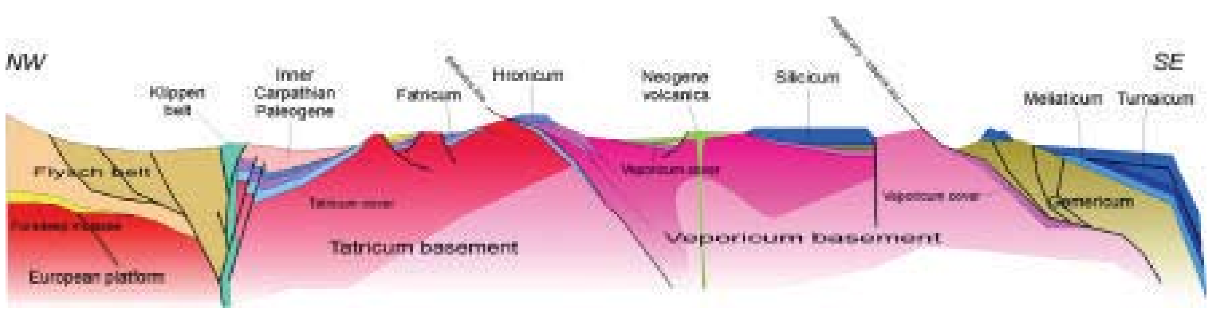

Figure 24.2. Simplified tectonic cross section of the Western Carpathians (not to scale)

Tectonics
In the Western Carpathians, tectonic processes resulted in a network of discontinuities (faults and fissures) affecting the bedrock. Down to a depth of ~100 m, fissures provide good conditions for groundwater flow. At deeper levels, they are usually gravitationally closed. The future repository must avoid these regionally important tectonic discontinuities (see Figure 24.3).

Geomorphologic Evolution of the Relief
Tendencies for vertical movement of the Earth's crust, along with denudation processes, decisively affect the depth of a repository. Long-term geodetic measurements indicate the block structure of the Western Carpathians. The average rate of sinking blocks is approximately 0.5–1.5 mm/ year (Figure 24.4). To assess the possible effects of relief, we are developing geomorphologic models to predict the future relief for a period with potential negative effects on the repository (from 10^4 up to 10^5 years).

Seismic Conditions
Seismic activity is usually dangerous for engineering works situated on loose sediments (e.g., sands), whereas hard rock (e.g., granitoids) is relatively safer. In Slovakia, the generally accepted limit of seismic intensity is 6° MSK-64. The majority of Slovakian territory roughly averages this value. Prospective sites for the repository are located away from recorded earthquake epicenters (see Figure 24.5).

Climatic and Hydrogeological Conditions
Situated in Central Europe, the Slovak Republic experiences temperate climatic conditions. Annual average precipitation 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 (at this point) little known at greater depth. In terms of a deep-geological repository, the main deficiency of these complexes is their relatively considerable petrophysical heterogeneity, resulting from tectonic effects. An unfavorable result of tectonic disturbances (besides manifestations of brittle tectonics associated with local increases of permeability) is the frequent occurrence of ductile zones, which may indicate (in some sections) an environment with increased permeability even at greater depths (Jetel, 1996). Neogene pelitic formations are generally poor in groundwater. Higher concentrations may occur only in interbedded sand lenses and layers.

24.2.5.2. Siting Criteria

The site for a deep geological repository, along with repository design and the engineered barrier system,

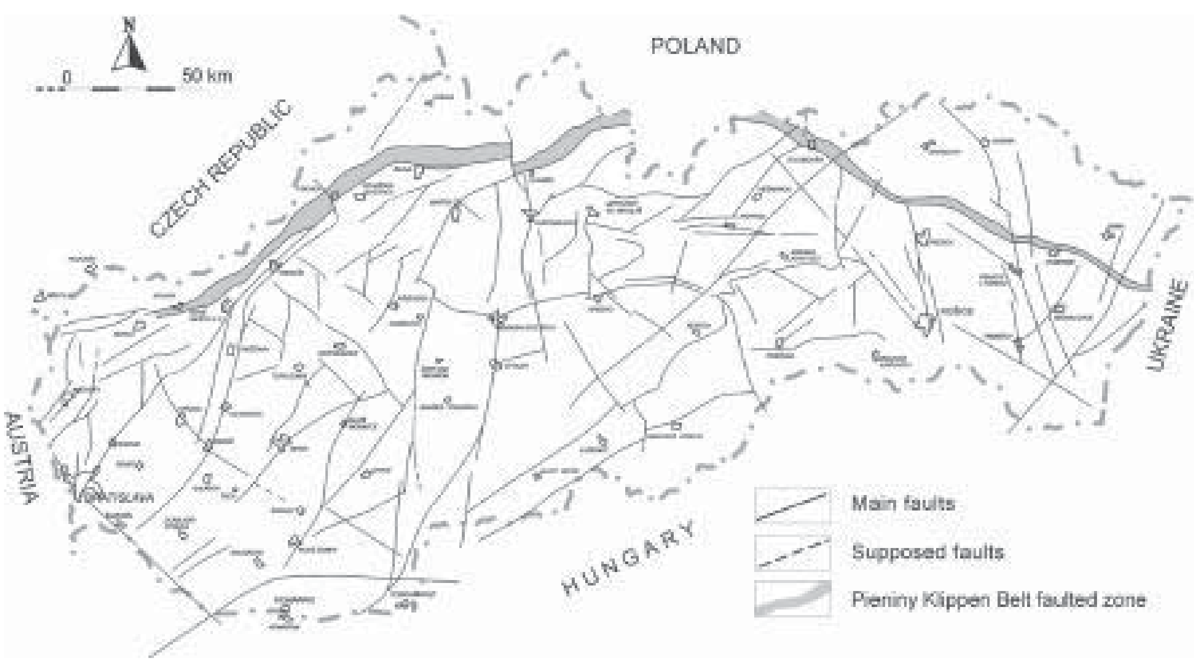

Figure 24.3. Simplified map of the main tectonic faults in Slovak Territory

Figure 24.4. Map of vertical-movement tendencies and thickness of the crust in Slovak territory

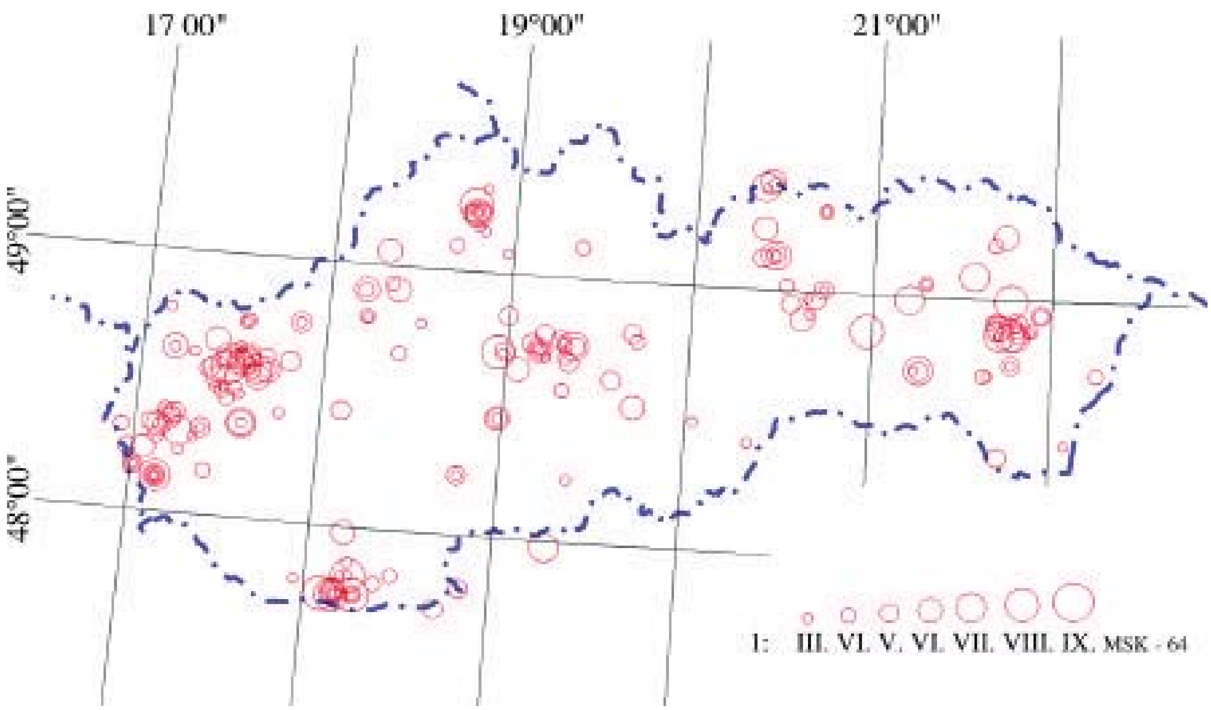

Figure 24.5. Map of epicenters of macroseismically observed earthquakes within the territory of Slovakia for the period 1304–1990

must ensure long-term safety of the disposal system. Disposal in a deep repository serves two basic purposes:

- Long-term isolation of high-level wastes and spent fuels from the environment, without relying on future generations to maintain the integrity of the disposal system
- Long-term radiological safety of humans and the environment in compliance with present, existing regulations and internationally accepted recommendations.

Worldwide experience shows that no unambiguous requirements exist in selecting suitable geological environments, just as no rock type can be clearly declared 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 geologic settings are suitably complemented by properly engineered barriers and by flexibility in repository construction and technology, numerous geological environments could guarantee a similar level of long-term safety.

A preliminary set of site-selection criteria for a deep geological repository was proposed, based on worldwide experience and consistent in principal with International Atomic Energy Agency (IAEA) recommendations (Safety Series No. 111-G-4.1). Three groups of criteria were proposed for site selection:

1. Geological and tectonic stability of prospective sites (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 of well or thermal waters).

Supplementation and adaptation of siting criteria are expected in the future, owing to progress in program activities and a growing knowledge from exploration of different localities.

24.2.5.3. Site-Selection Process

In 1996, this program started with a critical review (without field investigations) of information relevant to site selection. This review included a survey of published and archival regional geology, hydrogeology, engineering, and geophysical data, and led to identification of 15 sites potentially suitable for a deep geological repository within Slovak territory. Seven areas featured granitoid rocks, four clayey formations, three metamorphosed rocks, and one flyschoid rocks. The accuracy of the assessment corresponds to a map scale of 1:200,000. The next four-year phase focused on screening the 15 potentially suitable areas for further investigation (Figure 24.1). Limited field verification and some technical measures (geophysical profiles, shallow drillholes) were performed during this stage. A series of maps on a scale of 1:50,000 were compiled for each area (including important geological and hydrogeological factors and data on mineral resources). With respect to preliminary criteria, each area was assessed for its geology, tectonics, vertical movements, seismicity, mineral resources, geothermal potential, geochemistry, and engineering geology. The result of this stage was the ranking of the selected areas into three groups:

1. Areas recommended for further investigation—these areas are not expected to contain excluding factors (green circles in Figure 24.1).
2. Abandoned but not excluded areas—these areas may be considered as backup if other sites prove unsuitable or for any reason unacceptable (orange circles in Figure 24.1.). At these sites, lithologic structure, tectonic classification, and presence of ore indicators probably rule out (though not conclusively) selection of a sufficiently large and homogeneous rock block.
3. Excluded from further investigation—these areas contain utilizable geothermal energy below the prospective host rock; thin, hydraulically homogenous rock formations; and/or lithologic inhomogeneities (red circles in Figure 24.1.).

Based on the requirements mentioned above, four distinct areas (six localities) were determined as prospective sites for construction of a deep repository. Their total extent is 361 km^2. The objective at this stage was to perform more detailed geologic characterization of sites, possibly reducing their extent and number. 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 are to be compiled for each site (scale 1:25,000).

Four localities are situated in granitoid rocks (tonalites, granodiorites, granites) of Palaeozoic age and two in argillaceous Neogene complexes. Further reduction of the prospective site number is expected, but two alternative host environments should be considered, at least up to 2005. Decision on a selection of candidate sites is expected around 2010.

24.2.5.4. Prospective Sites for a Deep Geological Repository in Slovakia

Prospective sites in granitoid rock formations:

- Central part of the Tribec Mountains (46 km^2)
- Central part of the Ziar Mountains (41 km^2)
- Southern part of the Veporske vrchy Mountains (78 km^2)
- Southwestern part of the Stolicke vrchy Mountains (24 km^2)

Prospective sites in argillaceous and pelitic formations:

- Eastern part of the Cerova vrchovina Upland (87 km^2)
- Western part of the Rimavska kotlina Basin (85 km^2)

Central Tribec Mountains (Tatricum Unit)

This prospective site is an area of granitoid rocks in the southern Tribec-Zobor block in the Tribec Mountains. The Zobor Massif, one of the largest in the Western Carpathians, is formed of granitoids characterized by a monotonous and almost invariable composition, corresponding to massive medium-grained tonalites. Tectonic deterioration of the site is low in general, and thus hydrogeological conditions for a repository seem to be favorable. At present, there are no known limiting structural-tectonic phenomena in the present relief of the plutonic body. An ongoing drilling survey (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 mineral concentrations or geothermal potential have been discovered in this region. Hydrogeological conditions in the deeper horizon are little known at present.

Central Ziar Mountains (Tatricum Unit)

Granitoids of the Ziar Mountains are considered to be a prospective site. They form the predominant part of the crystalline complex and are represented mostly by varieties with porphyric K-feldspars. Leucocratic aplite-pegmatite granites occur mostly in marginal upper parts of the granitoid body, while two-mica granitoids occur in the central part. From a tectonic point of view, only some marginal faults with a NE-SW strike are important. The granitoid core is the most eroded one among the core mountains of the Central Western Carpathians. During the Palaeogene and Neogene, an upper strongly alpinotype-affected (1,000 to 1,300 m in vertical cross section) granitoid horizon was eroded away. As a result, the deepest plutonic levels have been exposed, within which protomagmatic structures may be identified. The granitoid core of the Ziar Mountains has (on radar images) the character of a rock mass affected only by disjunctive tectonics, which makes it an exception in the Western Carpathians.

The prospects of finding economically exploitable mineral resources in the Ziar Mountains are negligible. Information about the hydrogeological conditions at depths of 500–800 m is lacking.

Southern Veporske vrchy Mountains and Southwestern Stolicke vrchy Mountains (Veporicum Unit)

These two sites are adjacent to each other, though they belong to different geomorphologic units. A Divin tectonic line shifts them into two separate localities. The Vepor granitoid pluton is the largest one in the Western Carpathians (~60 km in length). Even though it is a complex pluton, consisting of several granitoid rocks, Alpine deformation (recrystalization) has a regional character here. Because of the pluton's size, it has been recommended for further investigation. The Vepor pluton is built by deformed porphyric granitoids, with several-km-large K-feldspar phenocrysts. A smaller area is occupied by biotitic granodiorites to tonalites of "Sihla" type and by a belt of hybrid granitoids in the southern part. Small bodies or veins of leucocratic granites, aplites, and pegmatites are typical features.

Veporicum granitoids are more tectonically affected in comparison to those of the Tatricum period. There are no indications of economically important mineralization.

Eastern Cerova Vrchovina Upland and Western Rimavska Kotlina Basin (Sedimentary Basins)

As with previously mentioned adjacent sites, these belong to different geomorphologic units: Cerova vrchovina Upland and Rimavska kotlina Basin. From a lithological, structural, and spatial perspective, the best prospective host rock appears to be two lithostratigraphic units: the Secen schlier of the Lucenec

Formation (Egerian) and Lenartovce Beds of the Ciz Formation (Kiscellian). These lithostratigraphic units form the principal mass of the basin filling. The predominant lithologic type in both formations is a mixture of siltstones and claystones. Maximum thickness of the Ciz Formation is 300–400 m, while the maximum thickness of the Lucenec Formation in Cerova vrchovina Hills is 1,200 m (1,100 m in the Rimavska Kotlina Basin). The thicknesses of both formations increase from the northern margin toward the south. Cumulative thickness of both formations varies between 1,400 and 1,600 m.

Considering hydraulic permeability, we must investigate the possible permeability of the Secen schlier siltstones and the permeability along faults (acidic-water springs). In some parts of these areas, there are also sandstone lenses of modest thickness and extent, which are saturated with fossil marine water.

24.2.6. 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 VUJE Trnava, Inc. 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. 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. We are also preparing to select 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.

The importance of natural analogues for deep geological disposal has been briefly described, especially in selecting engineered-barrier materials, in stressing the importance of safety in spent-fuel 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 methodology.

24.2.7. 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 of public awareness and involvement. 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 slow, there is evidence that prospects are enhanced by closer and earlier involvement of host communities. Key features of that involvement are:

- 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.

Present activities are focused on:

- Informing the public on radioactive waste management (presentation of nuclear energy and radioactive waste management in 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 (now in preparation) 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.

24.2.8. Legislation

Decom 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 in the

Slovak Republic.

Present activities are focused on evaluation of the existing radioactive waste management infrastructure in the Slovak Republic, ensuring that necessary activities, task definitions, and responsibilities (as prerequisites of quality assurance for the repository program) are known.

24.2.9. Coordination

One of the basic activities of a coordination center is transferral of information among contractors; subcontractors; managing, administrative, and regulatory bodies; and the public. An archival system for storage and maintenance of documents and data for the deep geological repository 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 in the Slovak Republic. Reports are intended for managing and administrative bodies and the public. Detailed plans for future periods of the repository program have been prepared by Decom.

We also 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. 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 (EIA) process. Attention was given to 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). Information to be released to the general public about deep geological disposal of spent fuel and HLW is under preparation.

24.3. 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) participate in the Slovak disposal program. Plans for further development of a deep repository have been reviewed by Nagra. Seminars and meetings dedicated to information exchange between our Czech colleagues involved in the Czech disposal program are organized yearly. Slovak Electricity, Plc., and Belgatom have signed an agreement for cooperation on deep geological disposal plans, especially in the study of clay-environment geochemistry, scenario development, and hydrogeological modeling in porous media. International cooperation is considered to be an effective way of increasing progress and benefiting from experience in other countries. Results from other disposal programs may also enhance the credibility of the Slovakian disposal program with the general public.

24.4. FUTURE ACTIVITIES

Future research and development activities should lead to the proposal for a first-reference disposal concept, establishment of a public-involvement program, and an information database. It should also lead to investigation of prospective localities (using boreholes), revision of siting criteria, a preliminary performance assessment based on available data, and selection of materials for an engineered barrier system.

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Chapter 25

Present Status, Objectives, and Preliminary Geological Suitability Assessment for LILW Disposal in Slovenia

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25.1. INTRODUCTION

In Slovenia, the main producers of radioactive waste and spent fuel are the nuclear power plant in Krsko and the research reactor TRIGA in the vicinity of Ljubljana. Other applications of radioactive sources and isotopes in research, industry, and medicine generate minor quantities of waste. The present quantities of waste amount to 2,250 m^3 of low- and intermediate-level radioactive waste (LILW) and 250 tU (~90 m^3) of spent nuclear fuel. Further generation of radioactive waste by all of these producers is expected in the future. Unfortunately, no final solution for this waste has been found so far.

The site selection for an LILW repository that was conducted in the early 1990s was unsuccessful—the process was presented in detail in Petkovsek et al. (1996). A few years ago, the Slovenian Agency for Radioactive Waste Management (ARAO) re-initiated the process on a completely new basis. Site selection and the development of an LILW repository remains the most important tasks of ARAO. The growing need for final disposal of LILW is the key issue for radioactive waste management in Slovenia at the present time. The ARAO is intensely involved in the re-initiated site-selection process for an LILW repository.

In this new process, we are trying to combine (as smoothly as possible) the technically (geologically) driven and stakeholder-driven site-selection processes. By combining technical and stakeholder approaches to site selection, we hope to guarantee strong public involvement and sufficient flexibility for the process (including the ability to adapt to specific conditions or new circumstances during the project). In the technical phase, our inclination is to retain a larger number of potential areas/sites, while keeping open the possibility of choosing from a number of repository types. The decision between the surface and underground option will only be made only after the repository site has been agreed upon.

In accordance with the International Atomic Energy Agency (IAEA) recommendations, the site-selection process is divided into four stages:

(1) Conceptual Design and Planning
(2) Area Survey
(3) Site Characterization
(4) Site Confirmation.

Last year, an area survey was started. In the preliminary geological suitability assessment, the natural features of the Slovene territory were assessed to locate geologically suitable formations. Performed with ARC/INFO technology, the assessment of natural conditions was based on consideration of the main geological, hydrogeological and seismotectonic conditions. The results have been compiled in a map showing potential areas for underground and surface disposal of LILW in Slovenia. It has been established that there is a potential suitability for underground disposal on about 3,050 km^2 of Slovenian territory, which represents approximately 15% of Slovenia. Areas potentially suitable for both surface and underground disposal amount to 9,040 km^2, or almost 45% of the entire country.

These preliminary results are now being carefully reexamined. As a result of this stage, a number of potentially suitable areas are expected to emerge. The final confirmation of site suitability will be carried out in detailed field investigations during the site-characterization and site-confirmation stages. The progress of the site-selection process will strongly depend on the response of local communities where potentially suitable areas will be identified, and on the success and efficiency of the mediator conducting the negotiations with local communities. According to the most optimistic scenario, the final site will be selected and confirmed by 2004–2005.

Along with the site-selection process, the disposal concept is also being developed. Because of the small amount of long-lived LILW compared to short-lived LILW, it was decided to develop a disposal solution for short-lived LILW only. Basic conceptual designs have been prepared for surface and underground repositories. Furthermore, the detailed engineering design for both types of repositories is being developed for the most likely combinations of geological environment and disposal facility. Performance assessment for both surface and underground LILW repositories at a generic location is also under preparation.

25.2. SITE-SELECTION PROCESS

The most demanding and sensitive part of this process is repository site selection. To fulfill this task successfully, ARAO has prepared a site-selection procedure with great care. Before making a decision on the right approach, international practice and experience were studied in detail. Three different possibilities were analyzed: (1) technical approach using explicit criteria; (2) local siting, based on voluntary participation of the local community in the siting process; (3) a mixed approach, basically a combination of the first two approaches. The mixed approach was selected as most appropriate (Mele and Zeleznik, 1998) because of its flexibility, transparency, and public involvement from the beginning. This mixed site-selection process will incorporate cabinet investigations and a general technical screening of the territory. Usually, the inclination during this part of the process is to retain a large number of potential sites. This initial step is followed by a period of negotiation with the local communities identified in the previous, pre-selection phase. Only after negotiations with the local community are successful will more detailed research follow, which would be field investigations to assess the suitability of the potential location.

As mentioned in the introduction above, after considering the IAEA recommendation (IEAE, 1994) for site selection, the proposed mixed-mode site-selection procedure was divided into four stages: conceptual and planning, area survey, site characterization, and site confirmation.

Until 1999, at the conceptual and planning stage, the entire process was prepared and defined as shown schematically in Figure 25.1 (Urban Planning Institute of Slovenia, 2000). Both technical and social aspects of site selection were considered. Recommendations and methodology for ranking the areas, according to their suitability for LILW disposal, were defined; and a program of cooperation with the public was established. Basic design requirements for the planned repository were also prepared.

Special attention was devoted to the inclusion of the local community in the site-selection process, which was recognized as essential to the process. It was decided that the best way to communicate with the local communities was through an independent mediator, who would conduct the negotiations between the community and the investor. The mediator would represent the link between the two parties and facilitate the communication and negotiations between the investor and the local community.

For the area-survey stage, the suitability of the Slovenian territory for a surface or underground repository of LILW was examined by cabinet investigations. The territory was assessed using different technical recommendations. The most important recommendations were related to the integrity and safety of the repository, which were evaluated through study of the geological properties of an area.

The biggest problem in this area-survey stage was the lack of an appropriate geological database. Together with the Geological Survey of Slovenia, the working group prepared a sound, expert basis for the site-selection process. For this purpose, the lithological map of Slovenia was digitized (on a scale of 1:200,000), thus enabling regional geological data to be processed by means of GIS technology. For more precise evaluations, more specific geological data were needed. With additional effort, we succeeded in providing hydrogeological, tectonic, and engineering-geological data. The quality of the newly prepared data and their compatibility with other spatial data used for the area-survey stage have also been examined. All these data will be used in determining the geological suitability of any potential LILW repository site in Slovenia.

After the suitability assessment is concluded in the areas of interest, contacts with local communities and municipalities will be established for those potential site areas.

With the assistance of a mediator, we hope to gain some positive response to continue the selection of potentially suitable locations. By the most optimistic scenario, the site-characterization activity is planned for 2004–2005.

25.3. INCLUSION OF RADIOACTIVE WASTE DISPOSAL INTO THE NATIONAL PHYSICAL PLAN

Along with the preparation of the area-survey stage, we also made an extensive effort to integrate our plans for LILW repository development into the National Physical Plan. At the moment, a new National Physical Plan for the period 2000–2020 is in preparation. In the existing spatial documents of Slovenia, the use of land to dispose of dangerous goods must also be defined. The LILW repository site-selection procedure must be treated as part of the spatial planning project. Moreover, the National Physical Plan (including LILW repository development plans) could be used as a crucial document of consensus and political support for project execution (Urban Planning Institute of Slovenia, 2000). The recommendations and proposals confirmed by the Physical Plan can also represent a starting point for the negotiations with local communities.

25.4. PRELIMINARY GEOLOGICAL-SUITABILITY ASSESSMENT FOR LILW DISPOSAL

This stage started in late 1999 and is to be concluded in mid-2001. Before that, based on the geological experience of other countries and according to our own specific geological conditions, a set of the six most probable combinations of geological environment/type of disposal facility (near surface and/or underground) was defined for Slovenia (Geological Survey of Slovenia, 1996):

- Type 1: Surface type of LILW repository above an open aquifer
- Type 2: Surface type of LILW repository on rock of low permeability
- Type 3: Underground type of LILW repository in plastic rock of low permeability

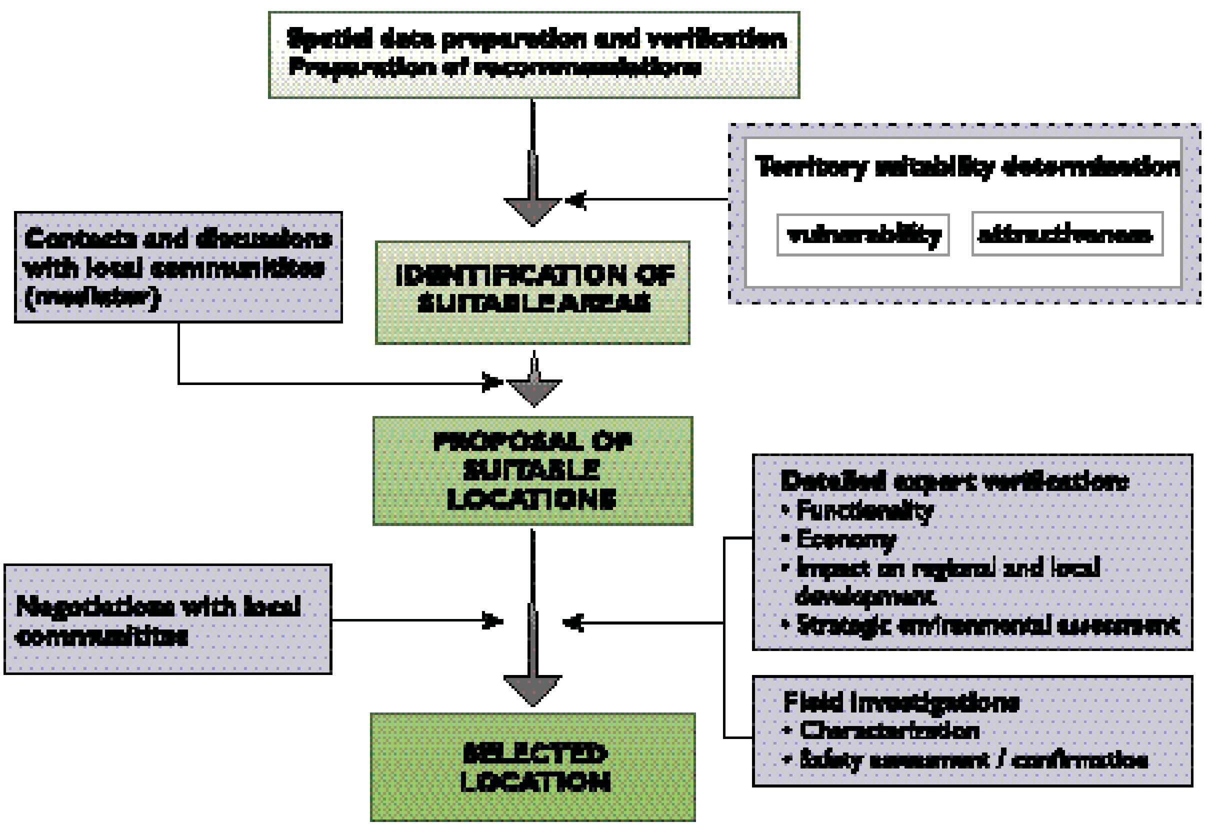

Figure 25.1 Schematic presentation of the mixed site-selection process

- Type 4: Underground type of repository for LILW with α-emitters in plastic rock of low permeability
- Type 5: Underground type of LILW repository in hard rock
- Type 6: Underground type of repository for LILW with α-emitters in hard rock.

Because of the decision to concentrate on the disposal of short-lived LILW, Types 4 and 6 were dropped from further consideration.

25.4.1. Differences in the Suitability Assessment in Comparison with the Previous Site-Selection Process

Compared to the previous site-selection process, we were no longer looking for the best specific host rock, but rather for suitable geological formations. The previous exclusionary criteria no longer applied. Instead of rigid criteria for site selection and elimination of territories, we preferred to use those recommendations that provided flexibility for the site-selection process.

The site-selection process has been extended to underground formations (as much as 100 m deep), and recommendations have also been made for underground LILW disposal. The decision to consider the underground disposal option provides an additional possibility of finding more suitable areas and locations. It is also believed that such a solution could gain higher public acceptance because of the high population density in Slovenia.

Using our mixed site-selection approach requires establishing sufficient flexibility for the technical effort. The flexibility of the process was assured by the following decisions:

- Recommendations with a higher level of uncertainty must be more flexible.
- Recommendations with minor influence/impact on repository safety can be more flexible.
- Recommendations that, at the attribute level, include not only expert judgment but also value determinations, prove to be more flexible.

25.4.2. Methodology

As a first step in the area-survey stage, the potentially suitable areas for LILW disposal are being identified on the basis of a “first group” of recommendations (see Table 25.1). The main guideline for selection of suitable areas and locations remains their natural properties (although repository safety could also be enhanced by engineered measures. Natural properties that enhance the safety and integrity of the repository cannot be influenced either by different technological solutions or by larger financial investments and other measures. From this point of view, this first group of criteria was defined (geology, seismicity, water). Areas identified through these criteria will be further evaluated by a second group of criteria (population density, natural resources, natural heritage, land use). But it is very important that the second group of recommendations is neither explicit nor rigid. For example, the population-density recommendation criterion can be applied in different ways. We are able to search for areas with low population density, but we are also able to search for industrial zones within areas having a higher population density. The evaluation will focus on the possible consequences for the suitable areas, considering both areas with low population density and industrial zones within densely populated areas.

The suitability assessment was performed by the multicriteria decision-making evaluation program within a Geographic Information System (GIS). This special application of the GIS was developed (Geological Survey of Slovenia, 1998; Sinigoj, 1998) to identify suitable areas with natural properties that are promising for the isolation of radionuclides. The application allows the user to model the relative suitability of areas for LILW disposal, as well as modify the categorization of the environmental datasets and their relative importance.

A predictive model on a small area of Slovenia was developed to perform sensitivity analyses and to predict and envisage the problems that might arise in the siting procedure. To develop this model, first-group criteria were compiled and prepared on a general level, and the multicriteria evaluation method called “weighted summation” was integrated into the application. The result was a list of priorities, with sites showing the highest value (sum of points) preferred for further, detailed investigations. The resulting map shows the potentially suitable areas with high value and less suitable areas with low value. Then the weighted summation method was applied using the basic screening method of “final score reduction.”

Table 25.1. First-group criteria in identifying suitable areas for LILW disposal

Recommendation	Basis for Recommendation Consideration	Explanation	Recommendation Flexibility
Volume of geological formation	Sufficient volume	Distance from the nearest geological boundary (from lithological map)	low
Simplicity of geological formation	As simple as possible and homogeneous	Area of geological structure (from lithological map)	low
Lithology	Lithological composition which prevents radionuclide migration	(Un)Suitability for disposal of all units of lithological map were defined (from lithological map)	very low
Geochemical properties	Stability of geochemical barrier. Minimization of reaction with waste	(from lithological map)	medium
Endogenic processes - Seismicity	As low as possible	Less suitable if higher than VIII degree MSC (from map of max.expected local intensity in a time period of 1000 years)	medium
Endogenic processes – Active faults	Outside active fault areas	Less suitable areas of five main faults dividing main seismo-genic zones (from tectonic map)	low
Surface stability	Hard rocks Low slope	Less suitable areas with surface unstability (from engineering geological map and from digitalized relief model)	low / medium
Water degradation processes	Outside flood areas	Less suitable areas of flooding (from map of flood lines)	very low
Extreme climate	Outside extreme weather conditions	(from digitalized relief model)	medium

25.4.3. Preliminary Results

Initially, we performed a preliminary geological suitability assessment to identify suitable areas for a surface LILW repository. After that, the dataset was supplemented with additional data about the underground geological structures of these areas, and an assessment was also made for underground disposal suitability. Both results were then compiled in a map showing potential areas for underground and surface disposal of LILW in Slovenia. With these preliminary results, we established that a potential suitability for underground disposal exists on about 3,050 km^2 of Slovenian territory, which represents approximately 15% of Slovenia. Areas potentially suitable for both surface and underground disposal amount to 9,040 km^2, or almost 45% of the entire territory. The most suitable natural predisposition for LILW disposal was identified as surface disposal in rock with low permeability; for underground disposal, it was plastic rock with low permeability.

During the area-survey stage, all methodologies were once more examined in detail, and the data were additionally verified and complemented with the results of new investigations. The lithological map of Slovenia was also supplemented and revised, and some sensitivity analyses and simulated use of different reduction factors were also used. After that, we started the final assessment to identify potentially suitable areas. The most recent preliminary result is shown in Figure 25.2.

Suitable areas were first divided into ten classes (according to their suitability). These classes were then grouped into three classes (shown on the map). The final decision on the number of classes (and the final result) will be defined at additional workshops to be held in late 2001. According to the preliminary results, the areas of main interest will be the first two classes, colored red (very suitable areas) and blue (suitable areas) on the map. The potential for other areas, colored green on the map, will be further investigated only if the local communities in these areas propose an investigation.

According to the preliminary results, the geologic structures identified as potentially suitable for LILW disposal can be described as:

1. *Argillaceous unconsolidated or mainly unconsolidated sediments*
   Represented by only a small part of the youngest geological formations. They are mainly of lower Pliocene and upper Miocene ages and deposited on formations of hard clays. They are suitable mainly for surface LILW disposal. These sediments were deposited in different parts of the Pannonian basin, and their lithological composition could vary. They are present mostly in the northeast and east parts of Slovenia, with a small portion in the central part.

2. *Argillaceous rocks—hard clays*
   The second group of suitable structures is represented by various rocks consisting of clay minerals—low permeable marls, claystones, mudstones and shales. Huge areas of these rocks are in the southwest, northeast, and central part of Slovenia. The youngest sediments of this group are of

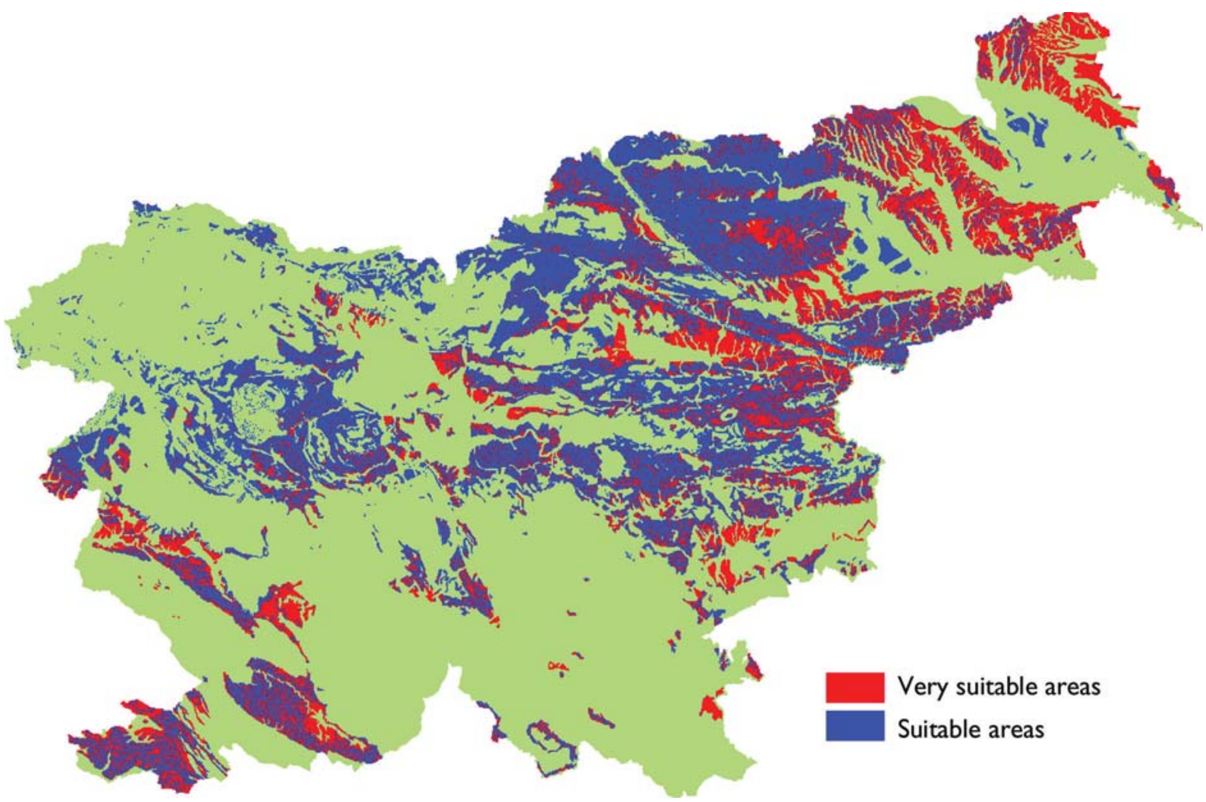

Figure 25.2. Suitability for LILW disposal—preliminary results (Prepared by Geological Sur vey of Slo venia,2001)

Miocene age; the oldest are of Permian/Carboniferous and Carboniferous age and (because of this) are very well consolidated.

3. *Crystalline rocks*
The third group of suitable formations consists of igneous rocks (granite, tonalite, diabase, basalt) and different metamorphic rocks forming a smaller part of northern Slovenia (Central Alps).

25.5. REPOSITORY CONCEPTUAL DESIGN AND PRELIMINARY PERFORMANCE ASSESSMENT

Concurrently with this site-selection process, the conceptual design of a repository for LILW is being developed. Because of the small amount of long-lived LILW (less than 2%) compared to short-lived LILW (98%), we decided to concentrate on a repository for short-lived LILW only.

Geological conditions in Slovenia are suitable for the disposal of LILW both in surface and underground facilities. Both alternatives have been considered feasible during the site-selection process. The repository type will be defined in accordance with site characteristics and local community interest, once the location is selected and agreed upon. The basic conceptual design for both types of repository has been prepared for a generic location. To facilitate the preparation of technical solutions while the location is still unknown, we defined four types of geological environment/type of disposal facility for short-lived LILW.

The conceptual design for the surface repository is similar to the Centre L'Aube (France) and El Cabril (Spain) facilities. The underground disposal system consists of facilities for disposal of conditioned waste and surface-waste acceptance, conditioning, and treatment facilities. Two alternatives for access to the disposal units were considered: horizontal access through plastic rock and vertical access through hard rock.

Since 1997, ARAO has also been working on a preliminary performance assessment of the LILW repository. In 1998, this was carried out through comprehensive analyses of radioactive waste's impact on the long-term integrity of disposal and its impact on the geosphere and biosphere. A systematic and generic list of all possible features, events, and processes (FEPs) predictable for surface or underground LILW disposal in Slovenia was prepared. The scenarios found to be most reliable for LILW disposal, under normal and emergency conditions,` were recommended and selected. A comprehensive database of parameters significant for surface and underground disposal was also made.

Scenarios and conceptual models for generic surface and underground LILW repositories have been developed[1]. Numerical modeling started last year. For this purpose, a surface repository over an aquifer of low water permeability and an underground repository in plastic rock were modeled. The modeling was carried out in three successive steps:

1. Near-field modeling (runs using the PORFLOW computer code), giving the mass fluxes and concentrations of radionuclides leaving the near field
2. Geosphere modeling, first for the hydraulic field and then with transport calculations for the entire mass flow entering the geosphere through the near field (GMS computer code)
3. Biosphere modeling, taking as a starting point the radionuclide concentrations in groundwater and in the stream and calculating the radiological impact to a member of a critical group. Calculations were run using the AMBER code.

This was the first attempt to perform LILW repository modeling in Slovenia.

25.6. CONCLUSION

As a result of the preliminary suitability assessment within the area-survey stage, a number of potentially suitable areas for an LILW repository have been identified. The final confirmation of site suitability will be made through detailed field investigations during the site-characterization and site-confirmation stages. The progress of this phase of the site-selection process will strongly depend on the response of local communities where potentially suitable areas have been identified, and on the success and efficiency of the mediator conducting the negotiations with these local communities. According to the most optimistic scenario, the site selection can be accomplished by 2004/2005. However, in the negotiation phase, some delays may occur, which may also affect the planned construction of the repository.

[1] For this purpose, an expert team from different organizations was formed (National Slovenian Civil Engineering Institute, Geological Survey of Slovenia, University of Ljubljana–Faculty of Mechanical Engineering).

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Chapter 26

Current Status of a Potential Future Geological Disposal Facility in South Africa

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26.1. INTRODUCTION

South Africa is currently in the process of developing a national policy and strategy for radioactive waste management. A draft policy has been formulated and is currently under review. A national strategy for spent-fuel management and other long-lived wastes should become a reality in the near future.

The main generators of spent fuel and other long-lived waste destined for a potential geological disposal facility in South Africa are the South African Nuclear Energy Corporation (NECSA) and the Koeberg Nuclear Power Station (Eskom). Mining companies are also expected to utilize a future geological repository for disposing of their long-lived waste. Current projections show that NECSA (Safari MTR research reactor) will produce about 5 m^3 of spent fuel and Koeberg will produce ~3,000 spent-fuel assemblies (~1500 tU) during their lifetimes, respectively. About 10,000 m^3 of long-lived bulk waste and a large number of industrial and medical sources could potentially also be earmarked for geological disposal.

The Vaalputs National Radioactive Waste Disposal Facility in the Northern Cape Province is licensed for the disposal of low and intermediate level radioactive waste (LILW). However, only limited investigations have been carried out with regard to Vaalputs' suitability as a geological disposal site. The impact of a future Pebble Bed Modular Reactor (PBMR) program has not yet been quantified.

26.2. HISTORY OF VAALPUTS AND SUBSEQUENT INVESTIGATIONS FOR GEOLOGICAL DISPOSAL

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, now NECSA, was given this responsibility and the following mandate, namely to:

- Construct an LILW facility by 1986
- Investigate the feasibility of a future deep geologic facility at the selected site.

The selection criteria that established the commissioning of Vaalputs in 1986 (Corner and Scott, 1980) therefore had the additional purpose of not only considering LILW, but also long-term high-level radioactive waste (HLW). The main studies centered around 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.

The final site selection was made according to geological criteria, which targeted a gritty clay formation that is mostly derived from the underlying granite-gneisses of the Namaqualand Metamorphic Complex. The gritty clay formation forms an excellent geological barrier for the shallow land repository at Vaalputs.

A working group on the disposal of HLW in South Africa, consisting of members of various government departments, Eskom, and the Atomic Energy Corporation (now NECSA), was established in 1987. This working group made the following recommendations (Toens, 1988) concerning spent-fuel 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 spent-fuel storage options at Vaalputs.

NECSA decided in the 1990s to commence with "low-key" investigations at 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 aero-magnetic and ground-magnetic surveys formed the backbone of a structural geological program to identify suitable areas for deep diamond boreholes. Three boreholes (between 550 m and 1,000 m deep) were drilled in selected areas 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) exposed rock between depths of 200 m and 1,000 m that showed excellent geotechnical qualities. This granite gneiss has virtually no weathered zones or faults, and very few joints were observed.

Because of the lack of cohesion between the major role players and the lack of a national strategy for radioactive waste management, NECSA halted any further investigations at the end of 1996.

26.3. POSSIBLE FUTURE SCENARIOS

If a site-selection program for geological disposal should form part of the national policy, which is currently being drafted, 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 very 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.
- Various options such as preliminary site selection with the view to long-term storage should be considered. Final site confirmation could then be done at a later stage.
- The prevailing economic and political climate will be a major factor and may permit the consideration of a regional repository involving various other countries.

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Chapter 27

Geological Disposal of High-Level Radioactive Wastes in Spain

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27.1. INTRODUCTION

The Empresa Nacional de Residuos Radiactivos, S.A. (ENRESA), a public company founded in 1984, manages all radioactive waste produced in Spain. All activities of ENRESA are approved by the Spanish government through the Radioactive Wastes General Plan (PGRR). The fifth PGRR was approved by the government in July 1999 and is presently in force. In this PGRR, it was established that no decisions will be made about the final management of high-level radioactive waste until the year 2010 (Ministerio de Industria y Energia, 1999). Until that date, ENRESAmust set up all the necessary techniques and methodologies for deep geological disposal, considering (in parallel) R&D activities related to separation and transmutation techniques that could reduce the inventory of waste.

Given this charge, ENRESA is developing an intense R&D program in geological disposal, with significant participation in the "European Underground Laboratories." In addition, generic designs for a repository located in granites and clays and corresponding long-term safety evaluations have been performed, and will be optimized until 2010. These activities include siting, designing, and evaluating the long-term components of repository performance. A performance assessment of repository designs is presented in this report.

27.2. ACTIVITIES AND AIMS OF ENRESA IN GEOLOGICAL DISPOSAL

ENRESA's strategy for deep geological disposal of high-level radioactive waste is conditioned by:

- The directions indicated in the fifth PGRR
- The references to possible lithologies focused on granites and clays
- The unavailability of underground laboratories in Spain.

To cover the aims established in the fifth PGRR, ENRESA has developed its activities in geological disposal through:

- Identifying key generic processes for disposal in granites and clays
- Establishing instrumental and numerical technologies needed to characterize these processes
- Verifying these technologies at representative scales and conditions and obtaining quantitative data for the processes that support the performance assessment exercises
- Establishing solid knowledge bases for the key functioning processes associated with repository components and their long-term behavior, developing basic tests in conventional laboratories and demonstration tests and verification in underground laboratories
- Using analogues and natural-systems studies to complete knowledge verification and characterization of technologies associated with key safety processes
- Developing (through R&D) the repository designs and safety-assessment exercises that allow integrating the knowledge obtained and identifying those aspects that should be improved

- Organizing the geological information obtained during the last PBE (Siting Program) and updating it with the information to be generated in Spain.

To develop these activities, ENRESA has created a scientific and technological infrastructure through research centers, universities, and companies in Spain, and has closely cooperated with international programs. In this sense, intense and productive participation with key European laboratories has been successfully nurtured.

This cooperation takes place on many fronts. For example, ENRESA is participating in the underground laboratory of ASPO (Sweden) in the Prototype, Buffer and Plug Test, and True Block Scale projects, focusing its activities on the thermal-hydrologic-mechanical behavior of clay barriers (instrumentation and modeling), and on migration tests and modeling of such tests. Also, in the underground laboratory at Grimsel (Switzerland), ENRESA is participating in the FEBEX, GAM, CRR, and GMT projects. These projects are investigating the development and verification of technologies related to construction and evaluation of compacted clay barriers (FEBEX), flow and multiphasic transport (GAM), radionuclide transport through clay colloids (CRR), and modeling gas effects in a clay barrier (GMT). Moreover, in the Mt. Terri underground laboratory, ENRESA participates in the ED-B, VE, HE, DI, EB, and FM-C projects. These projects aim to identify technologies associated with the hydraulic, mechanical, geochemical and thermal-hydrologic-mechanical characterization of compacted clay materials. These activities complement the plastic clay projects (RESEAL and CLIPEX) in the Mol (Belgium) underground laboratory. Finally, ENRESA also takes part in the MODEX-REP project in the Bure underground laboratory (France).

The results obtained so far have been highly successful, and ENRESA now has the capability to perform the characterization and predict the long-term behavior of all different repository components. It also has the experience and methodologies to perform the safety assessment. This combination of capability and experience will be improved and updated up to the year 2010, when the information gathered will be presented to the Spanish government. In this way, ENRESA will ensure that support is provided to the government in the decision-making process for final management of high-level radioactive waste in Spain.

27.3. SITE-SELECTION PLAN

The previous PGRRs selected several sites for detailed characterization in the period from 2000 to 2010. Even though this selection was cancelled, from 1986 to 1998 a detailed analysis of potential repository sites in Spain was made within the Site Selection Plan (PBE). A sequential analysis was performed, at different scales in the PBE, of the geological and socioeconomic characteristics of the territory, applying (at each scale) several feasibility and suitability criteria and considering all existing underground and surface geological information, as well as new information specifically obtained for the PBE. The successive working scales were: Favorable Formations Inventory (IFA) in the total country at a 1:500,000 scale, included in the European Catalogue of Favourable Formations. From the IFA information, 150,000 km^2 were studied at a 1:200,000 scale, which forms the Inventory of Regional Areas (ERA). Finally, more detailed studies were undertaken in 22,000 km^2, which represent the information at a 1:50,000 scale (AFA-ZOAProgram).

The site-selection plan has generated a large volume of geological information. For the management of this information, we have used a geographic information system (GIS) enhanced with software tools (e.g., Data Management System) to select the information by area, criterion, and scale. This system permits access to stored information and, more importantly, permits the construction of maps with the kind and amount of information desired at any moment.

The database generated during the PBE is now being completed (with some updated local information), and all information has been integrated in a fully GIS-supported database. Until 2010, the database will be updated including geological information to be produced for ENRESA by universities, research centers, companies, and others.

27.4. REPOSITORY DESIGN

We foresee the disposal system containing both spent fuel from Spanish nuclear power plants and long-lived intermediate–level wastes arising from the dismantling of these installations, as well as small amounts of vitrified waste from fuel reprocessing.

The number of spent-fuel elements expected in the current Spanish nuclear power plant program, assuming a

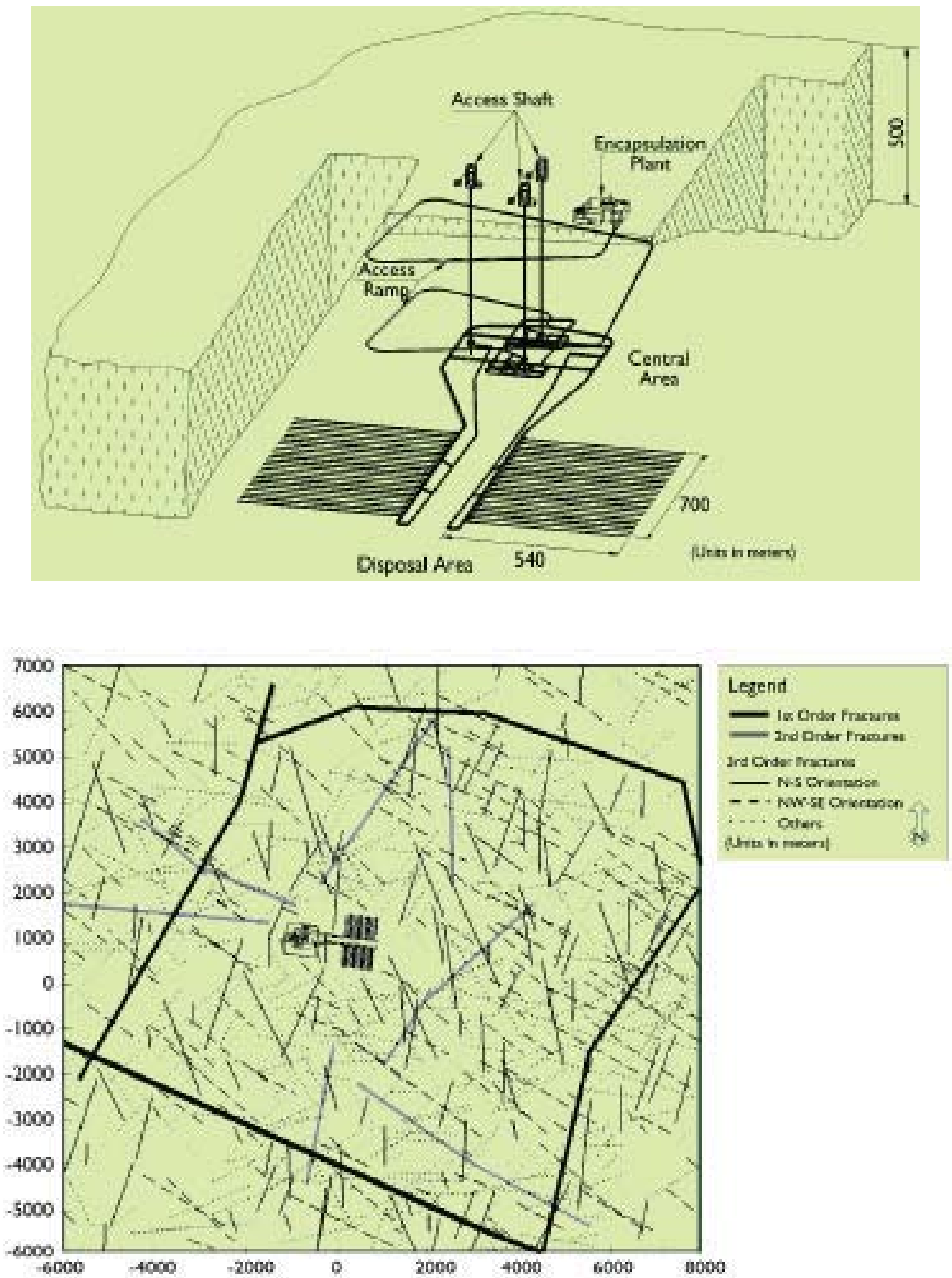

Figure 27.1. Repository design for granite

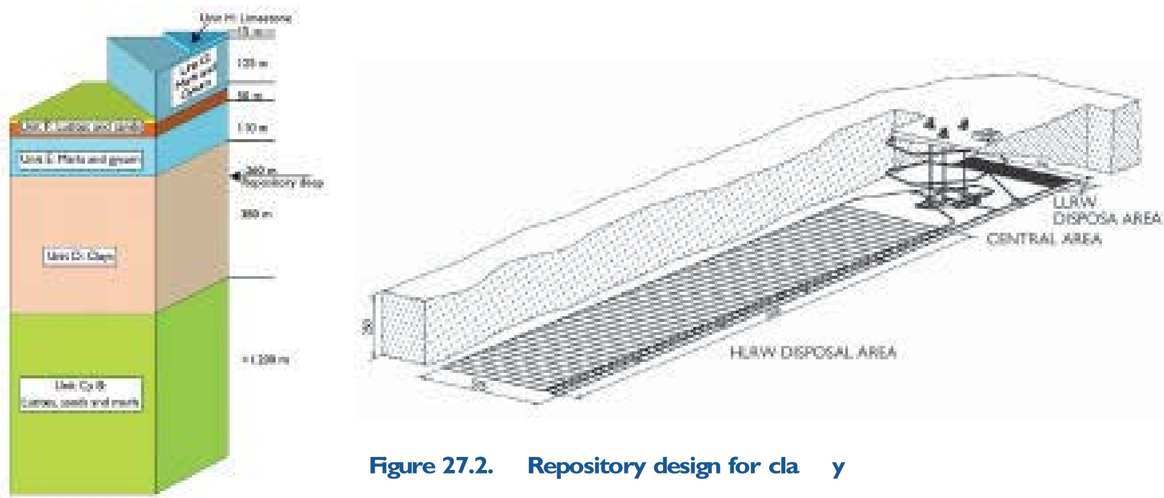

Figure 27.2. Repository design for clay

plant operating lifetime of 40 years and a burn-up cycle of 18 months, is 11,600 PWR type elements and 8,400 from BWR plants. In view of the diversity of existing fuel elements, a representative reference element has been selected, this being a standard 17×17 Westinghouse PWR element with a 4.1% enrichment in ^{235}U and a burn up of 40,000 MWd/tU. Likewise, an equivalence of 3 BWR elements to 1 PWR element has been adopted, and as a result, the total amount would be 14,400 PWR equivalent elements. Since the reference element's weight in uranium is 461.41 kg, the total quantity of uranium in the repository will be approximately 6,600 tons.

Up to now, ENRESAhas developed two reference concepts, one for granite and another for clay. Their main characteristics are described in Figures 27.1 and 27.2, respectively.

27.5. REPOSITORY TECHNOLOGY COMPONENTS

According to ENRESA's established strategy for high-level waste disposal, most of the effort up to 2010 will be focused on designing and characterizing the behavior of long-term repository components within ENRESA's R&D Program. The repository components considered are spent fuel, metallic canisters for disposal, compacted clay barriers, geological barriers, and biosphere. The R&D program seeks to identify and characterize the long-term safety processes and key parameters of the different components, which implies the design and verification of specific instrumental technologies. Another R&D objective (specifically for deep geological disposal) is to have adequate numerical models to make reliable estimates of the long-term behavior of different components, under a wide range of possible repository conditions.

27.5.1. Spent Fuel

The long-term behavior of spent fuel is an important aspect of repository safety analysis, since this is the source term for radionuclides. Within the framework of its R&D program, ENRESA considers the acquisition of knowledge fundamental for the following issues:

- Characterizing the physical-chemical properties of fuel with different degrees of burn-up, since this physical state controls radionuclide release to some extent.
- Characterizing radionuclide distribution in the fuel as another factor controlling the evolution of radionuclides which might be released over time.
- Characterizing the leaching mechanisms for radionuclides contained in the fuel. This state—the most important—includes the development of a leaching model for the fuel.

In performing these activities, and in view of the lack of appropriate facilities in Spain, ENRESAhas undertaken these studies through the following actions:

- Use of UO_2 simfuel as a fuel analog
- Establishment of cooperation agreements with Forschung Zentrum Karlsruhe and the Karlsruhe JRC for use of their installations

- Performance of joint projects with other European organizations within the framework of the European Union's R&D programs
- Use of natural uranites to verify the long-term behavior of fuel under natural conditions.

In 1999–2003, ENRESA's R&D plan, activities will be focused on:

- Physics of surfaces—their characteristics and modification
- Presence of volatile gases and elements, distribution of radionuclides in the matrix, instantaneous release of radionuclides via fissures or grain boundaries, and the analysis of metallic inclusions
- Response to leaching and dissolution
- Mobility and retention of products released
- Fundamental activities and fission products
- Analyzing the formation of insoluble products; sorption in steady phases
- Effects of the presence of iron or other components in the confinement systems
- Modeling of the evolution processes, leaching, and durability.

Figure 27.3 shows the leaching spent-fuel model.

27.5.2. Metallic Containers

Metallic containers are a fundamental component of waste disposal. On this issue, ENRESA is involved on two fronts: (1) developing a metallic container for temporary storage, and (2) securing transport licenses from the U.S. Nuclear Regulatory Commission (NRC) and the Spanish Nuclear Safety Council (CSN). R&D in this field has been focused on the study of different metallic materials and their long-term behavior, with a view toward selecting the best methodology for manufacturing canisters in accordance with the current design of the repository and safety analyses. Specifically, the activities include studies of corrosion, the behavior of generated secondary phases, and gas generation.

In previous R&D plans, activities were focused on studies of corrosion. The materials studied have been as follows: nonalloyed TStE 355 steels, steels alloyed with Ni and V TSt E460 and 15 Mn Ni6.3, and alloys of titanium Gr-7. The corrosion conditions have included the use of brines and solutions with salinities representative of the waters in the clay engineered barriers, stress corrosion (SC) using SSRT (Slow Strain Rate Testing) techniques, generalized corrosion, and analysis of mechanical parameters and generation of gas.

On the basis of the results obtained, ENRESA has selected carbon steel as the best candidate material for disposal canisters. Although the corrosion rate is faster than for other materials, the process is nevertheless continuous and the longevity of the material can be established with known uncertainties. Pitting corrosion in titanium alloys is more difficult to predict; despite the greater duration of these materials, the choice has been to adopt one for which the best prediction can be made, which is carbon steel. (Cost is another factor to be taken into account.)

The following activities have been set up in the 1999–2003 R&D plan:

- The completion of ongoing studies for long-term behavior of carbon steel and generation of associated gases. In parallel with this, more detailed studies will be performed on natural archaeological analogues, allowing processes of corrosion to be studied in steels more than 2,000 years old under several environmental conditions.
- Studies of the long-term behavior of corrosion-resistant material (stainless steels, titanium and copper alloys), with the intent of limiting problems associated with gas generation and facilitating possible recovery of wastes.
- Studies of metallic and ceramic coatings to improve durability in response to corrosion.

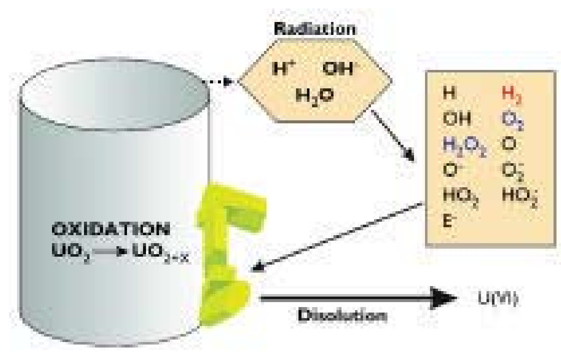

Figure 27.3. Leaching model of spent fuel

27.5.3. ENGINEERED CLAY BARRIERS

The engineered clay-barrier concept is one of the most important parts of the repository system. ENRESA has been working in this field since 1989, acquiring wide experience and developing improved efficiency. The main testing activity in this field is the Full-scale Engineered Barriers Experiment Project (FEBEX and FEBEX II) developed in the underground laboratory of Grimsel (Switzerland) in collaboration with the EU (ENRESA, 2000a). FEBEX consists of: (1) an *in situ* test performed under natural conditions and at full scale; (2) a "mock-up" test at almost full scale; and (3) a series of laboratory tests to complement the information from the two large-scale tests. (Figure 27.4)

The aim of the project is to study the behavior of near-field components for a high-level radioactive waste repository in crystalline rock. The following three objectives were established to investigate the:

- Feasibility of handling and constructing an engineered barrier system
- Thermal-hydrologic-mechanical (THM) processes in the near field
- Thermal-hydrologic-chemical (THC) processes in the near field.

The complexity of this project is justified as a follow-up to the characterization studies in a granite rock mass and other investigations on the thermal, hydraulic, mechanical, and geochemical behavior of materials for a clay barrier. In previous R&D investigations, ENRESA (1999) arrived at the conclusion that relevant progress in developing knowledge of near-field behavior could only be achieved by means of a comprehensive experiment, such as FEBEX.

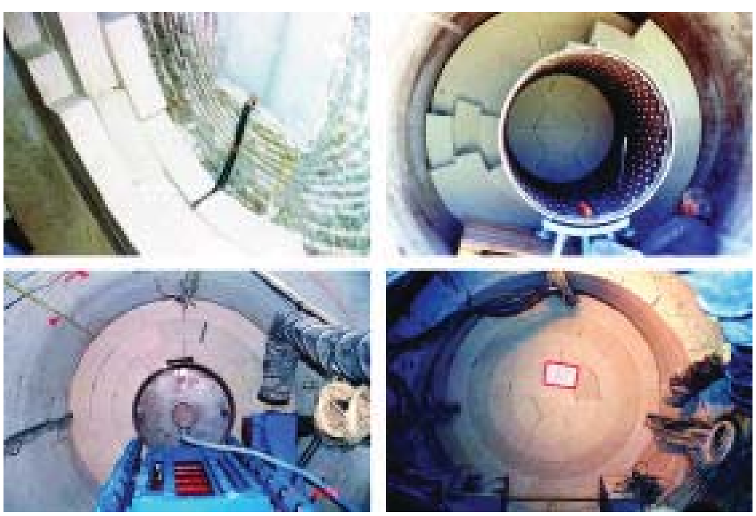

Figure 27.4. FEBEX Project: Underground research laboratory at Grimsel (Switzerland)

The principal objective of all this work is expected to be attained primarily in the *in situ* test. The THM and THC processes also need to be studied in this test, since it represents a real repository. However, the study of processes and variables, and the development, verification, and validation of the numerical models, requires less complex systems than the natural one: the "mock-up" test, and very simplified and controlled laboratory tests, should be adequate.

This *in situ* experiment is based on the Spanish reference concept for disposal of radioactive waste in crystalline rock (AGP Granite), in which the canisters enclosing the conditioned waste are emplaced horizontally in drifts and surrounded by a clay barrier made up of highly compacted bentonite blocks. The project was initially scheduled for a period of 7 years (1994–2001). However, in view of the experience acquired to date, the decision has been made to extend the project to the end of 2003 (FEBEX II). Its performance has been divided into four stages: (1) pre-operational (plan, design, characterization of the bentonite, installation, and modeling); (2) operational (heating, monitoring, cooling, and verification of predictions); (3) dismantling (extraction, inspection, and sampling and study of the materials); and (4) final evaluation of the results and verification of the models. The dismantling phase is now ongoing (FEBEX II).

The conclusions obtained in the first phase are as follows:

- The feasibility of constructing engineered barriers for the horizontal storage of canisters placed in drifts has been demonstrated. It has also been demonstrated that a quality assurance system is applicable, not only to manufacturing and installation of the physical components of the experiment, but also for the research work on processes, parameters, and modeling.

- The CODE-BRIGHT numerical THM model is capable of reasonably approximating the measured results of the two large-scale tests. Fundamental progress has been made in the development of laboratory apparatus, techniques and methods for the determination of the constitutive laws, and parameters required by the model. Thus, although complete validation is never possible, the performed tests have significantly increased the degree of confidence in the capacity for performance assessment of THM behavior in the repository near field.
- Great progress has been made in the development of THC numerical tools and analysis methods, from both the experimental and modeling points of view. Two THM codes have been developed and verified: CORE-LE and FADES CORE-LE. These codes reproduce fairly well the geochemical patterns of a large number of laboratory tests, and thus have generated reasonable expectations for their predictive capacity. (This capacity will be checked further in the dismantling phase.)

The engineered-barrier activities have been completed with the HE test in the Mt. Terri Underground

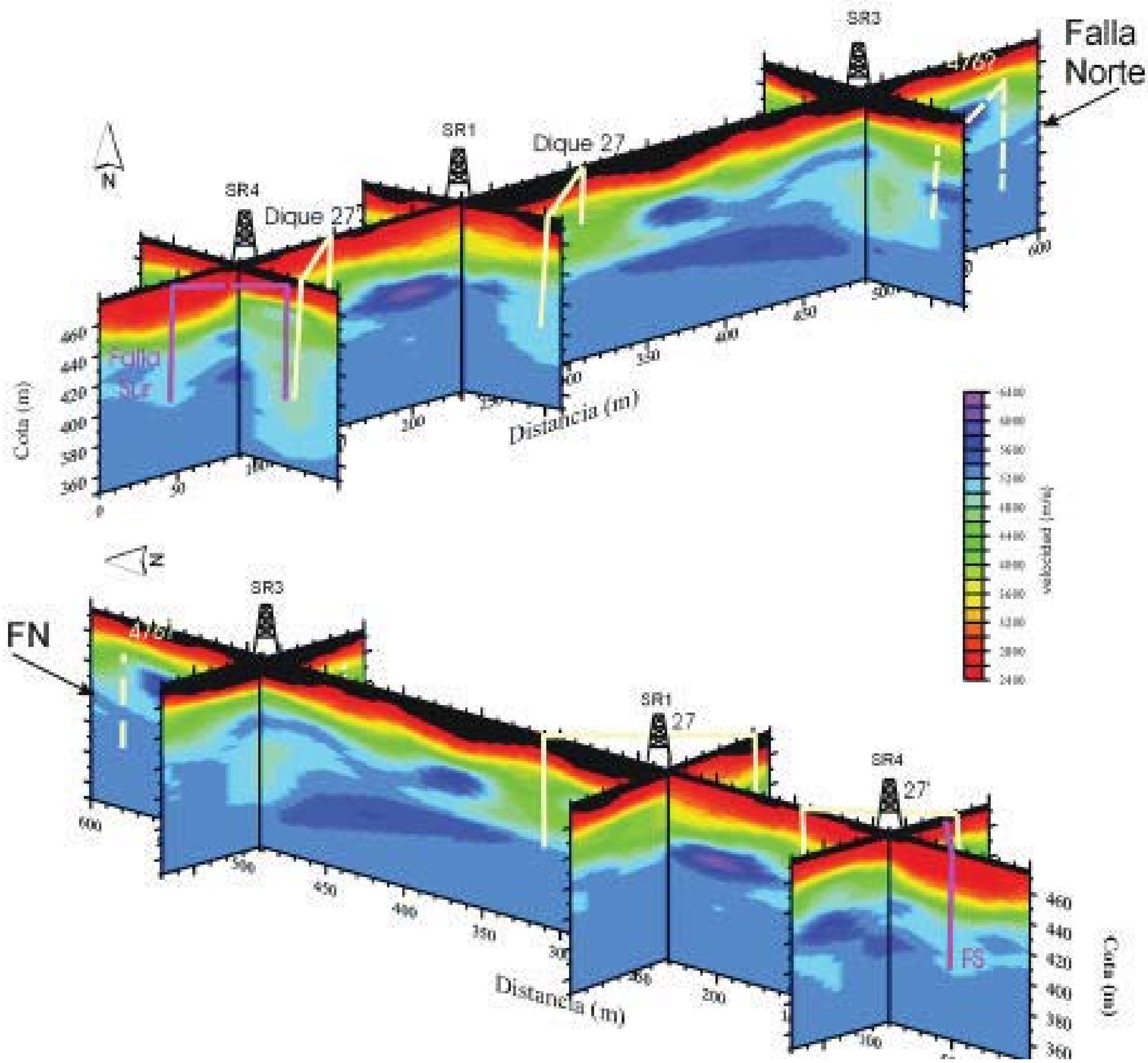

Figure 27.5. Geophysical tools applied in granite lithologies

Laboratory (related to thermal-hydrodrologic-mechanical behavior of clay engineered barriers). This test is similar to the FEBEX II, but much shorter than FEBEX. It is similar to a previous THM test in compacted clay formations.

27.6. GEOLOGICAL BARRIER

The effort in regard to the geological barrier has been directed toward developing, at different scales, the instrumentation and numerical models relating to lithostructural, hydrogeological, geochemical-hydrogeochemical and geomechanical characterization, involving the heterogeneity and anisotropy characteristics of the site. These techniques have focused on granitic media, with some sporadic work in clay media.

The technologies developed have been applied to underground laboratories, natural analogues, and the restoration of old uranium mines in granite media in Spain to identify the key processes for safety analyses of geological disposal.

27.6.1. Lithostructural Characterization

These activities have been focused on developing and verifying geophysical techniques to locate possible groundwater flow paths in a granitic batholith. The seismic reflection methods, combined with analysis and mapping of the stressed field in the different groups of existing fractures, were the main techniques under development. (Figure 27.5) The results have been particularly good and will be applied in the future to the analysis of deep granitic structures and the evolution of possible hydraulic behavior of fractures at depth.

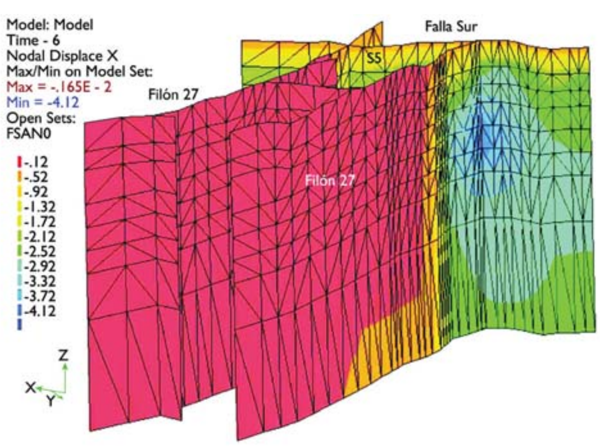

Figure 27.6. **Results of the hydrogeological numerical codes performed in the ENRESA's R&D plan**

27.6.2. Hydrogeological Characterization and Flow and Transport Modeling

ENRESA has oriented its activities along the following lines:

1. *Hydraulic-testing instrumentation in low-permeability media*
   Two mobile units have been developed for this purpose. One allows testing to be performed on the main hydraulic parameters of the formation down to a depth of 500 m, selecting the most permeable zones and continuously monitoring the boreholes. The other, more complex unit allows for hydraulic testing down to depths of 1,000 m and is currently in the verification phase. The instrumental developments are being applied at the Grimsel Facility (FEBEX Project) and in the restoration of ancient uranium mines in Spain.

2. *Numerical models*
   ENRESA is currently improving its inventory of codes with R&D plans, including flow, reactive transport, and the application of geostatistical techniques. The approach adopted in modeling groundwater flow is the so-called geostatistical inverse problem, with the resulting codes known as TRANSIN SERIE. This methodology resolves problems of heterogeneity within the medium, using geostatistical techniques. The fractured medium is considered as an equivalent porous medium in which discrete fractures are incorporated.

   In parallel with the above, geostatistical support models have been developed for both quantitative and qualitative data, making it possible to assign values of transmissivity continuously throughout the entire domain of application in the model (INVERTO SERIE). Models are currently being developed to simulate the heterogeneity of the coupled medium, along with models of tectonic evolution and others for flow and radionuclide transport.

   Another key aspect is reactive transport. For this, two families of codes have been developed, RETRASO and CORE-LE, which are applicable to problems in 1 or 2 dimensions with heterogeneity and anisotropy. The codes resolve heat and solute-transport equations using the finite-element method.
3. *Specific applications*
   Hydrogeological characterization methodologies have been applied to verification problems in four

different projects. One of these was completed in 1996 (Berrocal); the other three are underway (Ratones Mine, FEBEX II, and Matrix natural analogue), with a view toward supporting performance-assessment exercises in granite (Figure 27.6). A regional hydrogeological model has been developed, along with another local model using the hydraulic and characterization parameters derived from R&D.

Given that the performance-assessment site is generic, geostatistical techniques have been applied to use existing data and create a synthetic physical medium with realistic property values.

27.6.3. Geochemical and Hydrogeochemical Characterization : Radionuclide Migration

These activities have been focused on providing suitable instrumentation for *in situ* sampling and testing, and characterizing processes related to migration in granite and clays, water-rock radionuclide interaction, geochemical modeling, etc. Activities specifically include:

- *In situ* acquisition of physical-chemical parameters, major components, and certain *in situ* trace elements, along with others in the different hydrogeological units
- Sampling and analysis of colloids and associated transport elements
- Continuous monitoring of boreholes in selected sections, depending on transmissivity
- Measurement of parameters controlling radionuclide migration in the matrix, fracture fillings and separated minerals
- Measurement of distribution coefficients under different physical conditions and diffusion coefficients in clay material
- Model of current hydrogeochemical system
- Pore-water extraction, analysis and modeling.

A highly sophisticated mobile laboratory has been developed for this purpose, making it possible to perform *in situ* sampling and analysis of low-permeability media to depths of 700 m. This has been employed in different projects with highly satisfactory results. An important analytical laboratory support has also been developed for the characterization of colloids and analysis of associated elements, as well as extraction and analysis of pore waters (Figure 27.7).

In parallel with the above and for hydrogeological testing, highly sophisticated equipment has been developed to perform tracer tests on the surface and in underground laboratories, under natural flow, dipole, and forced-circulation conditions. This equipment, which has been used in different international projects, will be used in diffusion experiments in clay materials (Mt. Terri Project).

Analysis of retention mechanisms is a very important aspect for safety analyses. With this in mind, a specific laboratory has been developed for migration studies, enabling us to identify the transport/retention mechanisms for the main radionuclides, in different materials of a granitic repository (matrix, fissure infills, clay barrier, concretes, etc). Our aim is to replace the term "Kd" in flow and transport modeling with the most relevant processes for transport or retention. The problems of spatial variation and upscaling are also to be addressed in the present plan.

Figure 27.7. Pore-water extraction equipment

27.7. NATURALANALOGUES

Natural analogues are an important group of our activities. This work has the following objectives:

- Qualitative and quantitative identification of relevant processes in the long-term operation of a repository
- Application under real conditions of instrumental and numerical developments performed with a view to verification
- Information for the general public on the feasibility of the repository concept and on how nature offers numerous examples of the long-term operation of repository components.

The ENRESA natural-analogue program has been involved with the following fields of investigation:

- Long-term behavior of radiated fuel: Natural Uraninites Project and Oklo Project
- Long-term behavior of canister metallic materials: ARCHEO Project
- Long-term behavior of bentonites for an engineered barrier: BARRAProject
- Characterization of radionuclide migration processes in granitic media: BERROCAL Project, PALMOTTU Project, RATONES Project
- Behavior of radionuclides in different redox environments and interaction with capsule degradation products: MATRIX. Project (Figure 27.8).

This work has led to the following achievements:

- Verification of methodological approaches from the surface for hydrogeological characterization of a site in fractured media

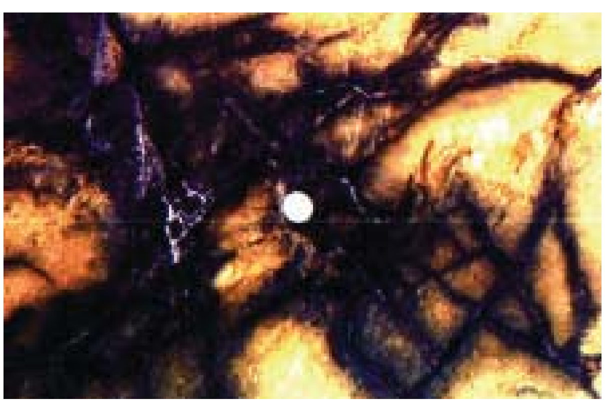

Figure 27.8. Natural analogues:Matrix Project

- Verification of methodological approaches for geochemical and hydrogeochemical site characterization
- Verification of hydraulic and hydrogeochemical testing instrumentation
- Verification of the TRANSIN, RETRASO and CORE-LE codes
- Development and verification of methodologies for geochemical modeling in granitic media.

It is considered necessary to maintain activities in this field throughout the period 1999–2003 via two basic projects:

- ARCHEO, focused on the evolution of archeological metallic material (Pre-Roman steels (>2,000 years), Roman steels (<1,700–2,000 years), Arabic steels (1,000–1,500 years) and iron-copper alloys of variables ages
- BARRA, focused on the long-term behavior of Spanish reference bentonitic materials, will facilitate analyzing the effects of thermal and saline water on the long-term mechanical and retention properties of the bentonites.

27.8. BIOSPHEREAND CLIMATIC CHANGE EVOLUTION

The biosphere constitutes the ultimate receiver of those radionuclides that hypothetically might be released from a repository. It also constitutes a barrier, given the effect it might have on radionuclide dispersion and dilution. Our related R&D activities have been addressed as follows:

- Characterization of the most relevant radionuclide migration processes in the different compartments of the biosphere (and improvements of characterization and modeling methods)
- Biospheric models and biospheres of reference via the BIOMASS and BIOMOVS projects. ENRESA has developed an in-house methodology for modeling the biosphere and for constructing reference biospheres that might be used in performance assessment.

It is fundamentally important that there be specific databases dedicated to the behavior and distribution of radionuclides in the biosphere, as well as data on the main geosphere-soil-biosphere transfer systems. These can be applied to those biospheres that may potentially be included in performance-assessment exercises.

Palaeo-environmental studies are considered a key element of R&D, with a view to establishing lower-uncertainty predictive models for the long-term evolution of the biosphere and the geological barrier. In the Spanish case, the main effects of long-term repository operation will be as follows:

- Verification of recharging conditions and, therefore, of the hydrogeological-geochemical operation of the barrier
- Variation in geomorphological surfaces affecting recharge and associated biotics
- Biosphere variations adapted to different climatic conditions.

Our R&D has concentrated on developing a coupled set of technologies, making it possible to establish the palaeo-environmental evolution of the Iberian Peninsula over the last million years, in an effort to establish geo-prospecting forecasts.

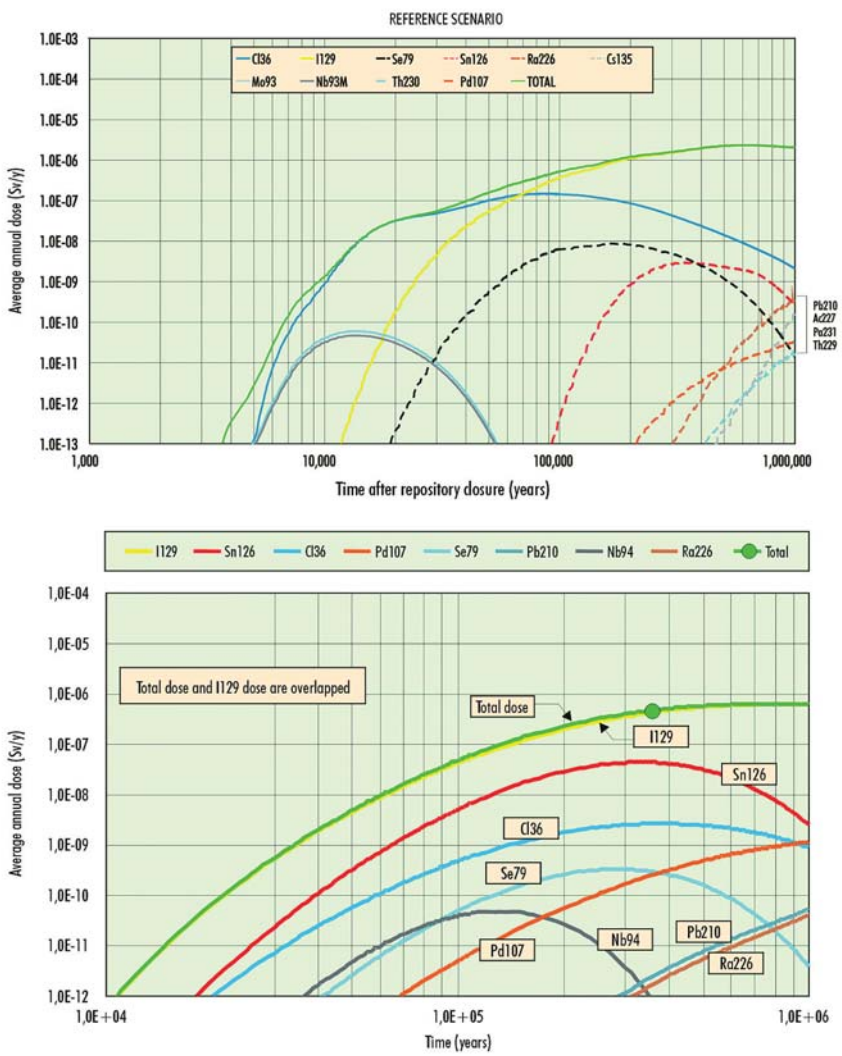

Figure 27.9. Performance-assessment exercises in granite and clay: Results for the reference scenario

27.9. REPOSITORY PERFORMANCE ASSESSMENT

ENRESA has performed two generic exercises related to long-term safety of deep geological disposal in granites and clays, the main characteristics and analysis of which are described in this report (ENRESA, 1998). However, two more advanced performance-assessment exercises are in progress (ENRESA, 2000b), ending on December 2001 (for granites) and December 2003 (for clays).

These exercises include both the degree of knowledge reached within ENRESA's R&D and the main international developments performed in this field. The main criteria and methodology adopted in the performance-assessment exercises are the following:

Criteria

- The repository concept for granite and clay has been previously described (Section 27.2).
- The disposal facilities are located at a generic site, defined on the basis of the known characteristics of Spanish granites and clays.
- Priority is given to the analysis of scenarios that envisage the normal evolution of the disposal system, but a few alternative scenarios have also been considered.
- Simplified models of the different processes and subsystems are adopted.
- A probabilistic approach in dealing with the uncertainties associated with different input parameters and spatial and time-related variability was used.

Methodology

- Establishment of the safety criteria and input data.
- Development of scenarios: identification of features, events, and processes that may affect the safety of the disposal facility.
- Modeling of the reference scenario, including the development of conceptual models and the subdivision of the general influence diagram into 12 submodels.
- Analysis of consequences in which the input parameters and the probability functions, adopted to represent possible variations in each parameter, are provided. Calculations are performed to evaluate the potential radiological consequences for reference scenarios and alternative scenarios (production-well and poor-sealing scenarios).
- Analysis of results and conclusions regarding the performance of the engineered barriers, the geosphere, the biosphere, and the overall safety of the disposal concept.

In Figure 27.9, the results of reference scenarios for clay and granite are described. After these exercises, several conclusions have been drawn related to engineered-barrier, geological-barrier, and biosphere performance, as well as to dose calculations. In all these analyses, final results demonstrate satisfactory levels of performance for the geological disposal of radioactive waste.

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Chapter 28

The Swedish Program for Spent-Fuel Management

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ABSTRACT . The Swedish concept for a deep geological repository involves encapsulating the fuel in copper canisters with cast iron inserts, embedding each canister vertically, and surrounding them with bentonite clay at a depth of about 500 m in the bedrock. The siting process has now reached the end of the feasibility study phase. Three municipalities have been selected from the original eight. These are Oskarshamn in southeast Sweden, and Tierp and Östhammar in the region of Northern Uppland. It is now up to the municipalities concerned to decide if they want to participate in the next step of the siting process. If the decision is positive, site investigations will start in 2002.

So far, the Swedish Nuclear Fuel and Waste Management Company (SKB) has tested three different methods to fabricate the canister copper shells. Trial fabrication shows that seamless fabrication methods produce a fine-grained and homogenous structure. Much of the work on the sealing method is being done at SKB's Canister Laboratory in Oskarshamn, where we are developing new procedures for welding. The dress rehearsal for repository technology in SKB's Hard Rock Laboratory on the island of Äspö, outside Oskarshamn, has entered an intensive phase. Among the main projects are the retrieval test, the backfill and plug test, and the prototype repository.

28.1. BACKGROUND

Sweden has been generating electricity using nuclear power for almost 30 years. The first commercial reactor was put in operation in 1972, and the latest in 1985. A referendum in 1980 limited the nuclear program to twelve reactors. One of these, Barsebäck 2, was closed down in 1999 because of political reasons. The 11 remaining reactors are operated by four utilities and supply almost half of the electricity produced in Sweden. Their total capacity amounts to 9,600 MW. Eight of the reactors are boiling-water reactors (BWRs) and three are pressurized-water reactors (PWRs).

Up to now, almost 4,000 tons of fuel have been used in power production. As mentioned above, the Swedish government has decided to start phasing out reactors. Because of this phase-out, it is difficult to estimate the total amount of spent fuel to be managed in the future. A 25-year operating period will likely result in a total of about 6,300 tons. If, on the other hand, reactors are operated for 40 years, spent-fuel totals will likely be about 9,000 tons.

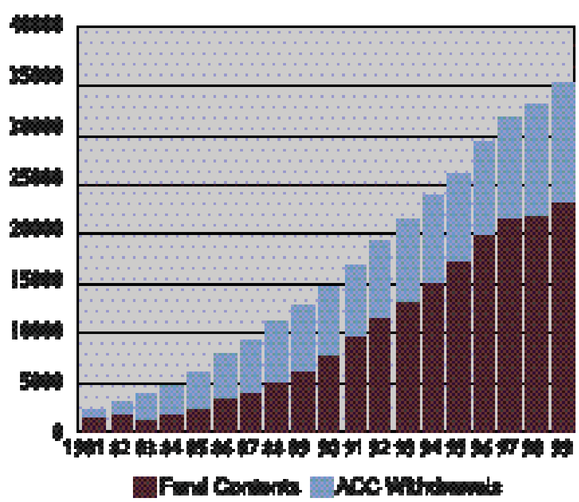

Figure 28.1. Graph showing fund balance and accumulated withdrawals for the financing of waste management

In the 1970s, legislation charged the nuclear power industry with the responsibility for managing and disposing of all its radioactive waste in a safe manner. For this purpose, the owners of the nuclear power plants formed the Swedish Nuclear Fuel and Waste Management Company (SKB). SKB´s task was and 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, the development of a deep repository, a canister factory, and an encapsulation plant are all that remain for the system to be complete.

28.2. FUNDING

The costs of managing radioactive waste are financed by a special charge, which varies between one and two öre (100 öre = 1 SEK; 1 USD = 10.30 SEK) per kilowatt-hour. This charge is determined each year by the Swedish government and is based on the cost calculations submitted by SKB to the Swedish Nuclear Power Inspectorate. The calculations are based on the assumption that all nuclear power plants will be operated for 25 years. Starting in 2001, the assumed operating time will be changed to 40 years. The charges vary from owner to owner. The total sum depends on how long reactors at different power plants have been in operation.

The money is paid to the Swedish Nuclear Waste Fund and is deposited with the National Debt Office. As shown in Figure 28.1, approximately 800 million SEK are added to the fund in this manner every year. In addition, around 1,500 million SEK per year is earned in interest on the money already set aside. We estimate that the total expenses for nuclear power will amount to 48 billion SEK. The outlays will be spread out over a period of more than 50 years, which means that we can expect to receive additional interest income. The distribution of costs is shown in Figure 28.2.

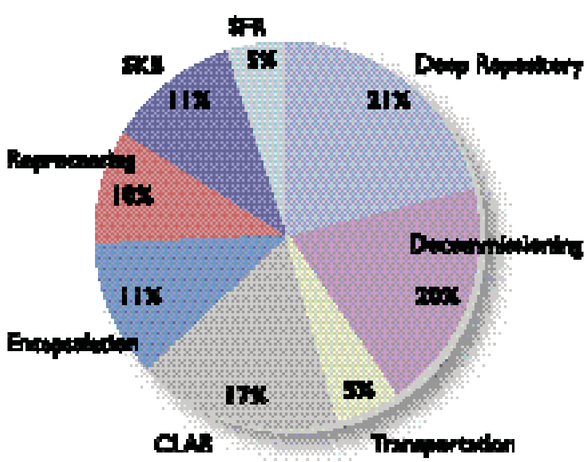

Figure 28.2. Distribution of total costs for nuclear waste management

28.3. THE DEEP REPOSITORY

Over the past years, SKB has built up a system to manage various kinds of radioactive waste. The structure of the Swedish system is shown in Figure 28.3. We have a specially built ship to transport the waste, a final repository for various types of radioactive waste, and an interim storage facility for spent nuclear fuel.

SKB has been considering what the repository will look like and what materials and technology will be used for almost 30 years. The principal alternative involves encapsulating the fuel in copper canisters and embedding each canister vertically in bentonite clay at a depth of about 500 m in the bedrock, as shown in Figure 28.4. The deep 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 spent fuel 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.

28.3.1. A STEP-WISE PROCEDURE

It will take at least 40–50 years to carry out all measures needed to dispose of all long-lived and high-level nuclear waste 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 waste that has already been disposed. This will ensure freedom of choice for the future, while at the same time demonstrating the deep-disposal method on a full scale and under actual conditions.

Around one-tenth of the total quantity of nuclear waste will be deposited in the repository to begin with. This is equivalent to about 400 canisters out of a projected total of 4,000. The numbers are based on the 25-year operation scenario previously described. After an initial period of operations, we will conduct a thorough evaluation of the repository. If the evaluation of this 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 licence to begin regular operation. Then the remaining

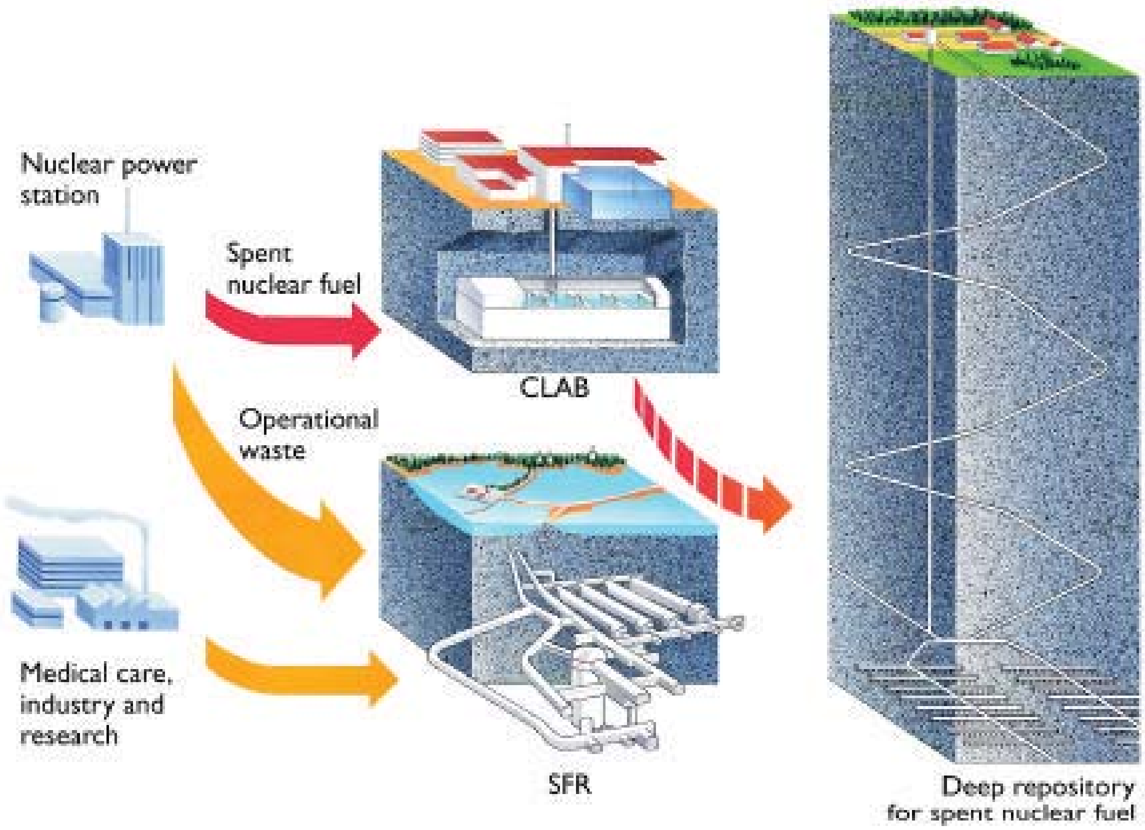

Figure 28. 3. The Swedish system for handling radioactive waste

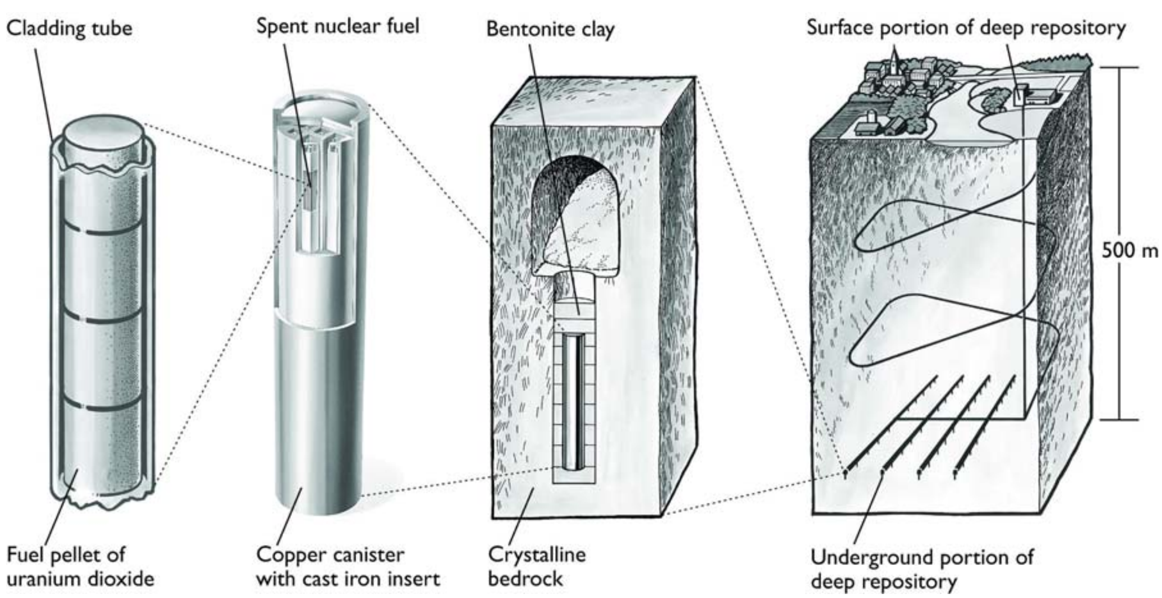

Figure 28.4. The spent fuel will be encapsulated in copper , embedded in bentonite cla y, and emplaced at a depth of 500 m in the rock.

3,600 canisters will be disposed of as well. The program of regular operations is projected to last for between 20 and 30 years (Figure 28.5). Decisions regarding siting, construction, and operation of an encapsulation plant and a deep repository will also be taken in steps, based on progressively more detailed information.

28.3.2. Repositor y Technology

To prepare for the siting and construction of a deep repository, SKB has built the Äspö Hard Rock Laboratory (HRL) on the island of Äspö outside Oskarshamn. The laboratory is designed to meet R&D requirements and consists of a 3,600 m long tunnel going down in a spiral to a depth of 450 m, as shown in Figure 28.6. The dress rehearsal of the deep-disposal technology in our Hard Rock Laboratory has entered an intensive phase. In the coming years, we will try out the technology and the methods needed in a deep repository. Some of the activities at Äspö HRL are described below.

Boring Machine

One of the first steps in the Äspö HRL dress rehearsal was the boring of 13 deposition holes, which was concluded in 1999. The holes are 175 cm in diameter and eight meters deep. This is exactly the same size as will be used in the future deep repository. Boring such holes vertically requires new technology, because of the difficulty in operating a boring machine in tunnels with a roof height of only five meters. This was accomplished by modifying a traditional TBM machine, and these special holes are now being used in different tests.

Deposition Machine

A copper canister is nearly five meters long and weighs between 25 and 27 tons when it is filled with spent fuel. In other words, it is not particularly easy to handle—especially not in the confined spaces that will exist in a deep repository. To be able to emplace a canister in a deposition hole, we have developed a prototype of a remote-controlled and radiation-shielded deposition machine. The machine runs on rails and is being used in the demonstration repository. There we are practicing the sequence of emplacing and retrieving canisters. We will also develop and test procedures for what to do if something breaks or power fails. The machine is shown in Figure 28.7.

Backfill and Plug Test

SKB is testing the technology for backfilling and plugging tunnels. The main goals for the project are to test (a) materials and methods for their compaction, and (b) the function and interaction of these materials with the surrounding rock. In 1999, we backfilled and plugged a 30 m long test area in a drill-and-blast tunnel in the

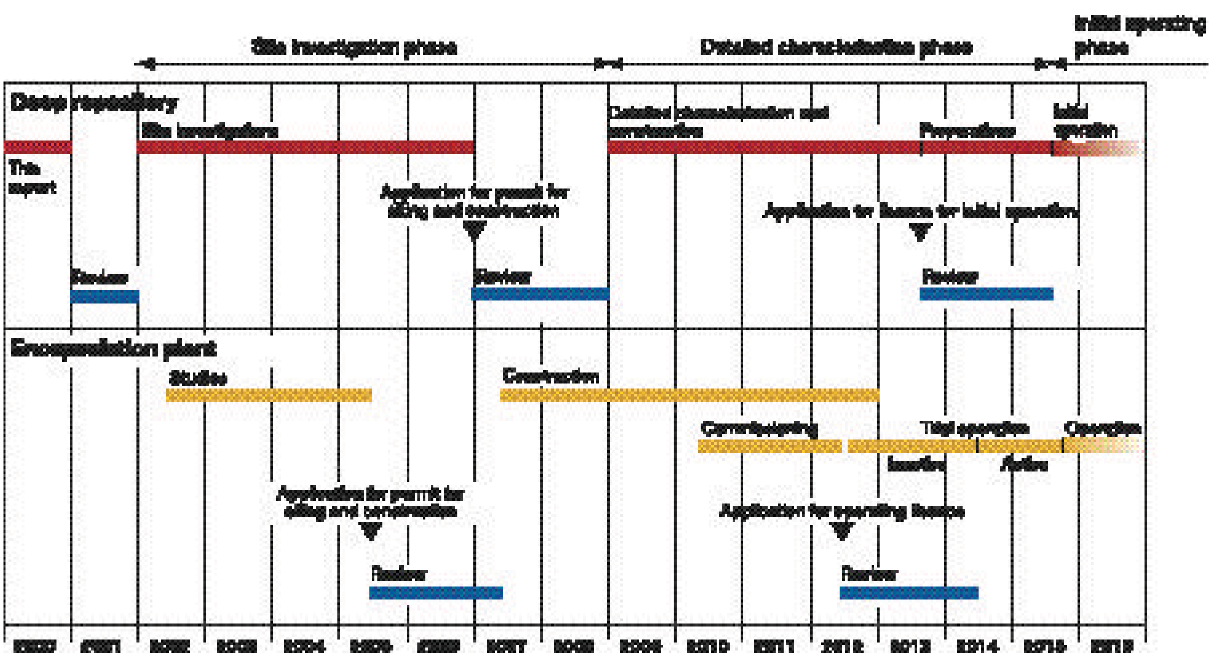

Figure 28.5. Reference time schedule for deep repository and encapsulation plant

Äspö HRL. The inner part of the tunnel was filled with a mixture of 30% bentonite and 70% crushed rock, whereas the outer part was filled with crushed rock only. The backfill was compacted in layers with a technique developed in earlier tests. Permeable mats between the different layers ensured a controlled saturation process. The test area was sealed with a thick concrete plug, and during the next few years, we will monitor the sealing capacity of the backfill and this plug. The results will show whether the method works and how well the numerical models we have used agree with reality.

Figure 28.6. The Äspö Hard Rock Laborator y

Retrieval Test

Since the repository is designed in such a way that it is possible to retrieve deposited canisters, we must develop and test the method for retrieval as well. The main goal for this test is to develop the method for freeing a canister from water-saturated bentonite. One full-sized canister is placed in a deposition hole and surrounded with bentonite. The canisters are equipped with electric heaters, and the bentonite blocks are equipped with some instrumentation. When the bentonite is saturated with water, it will be disintegrated with saline solution or frozen. We believe that it will take between three and five years for the bentonite to reach saturation. Meanwhile, techniques and equipment for freezing the canister will be developed. Aschematic layout of the retrieval test is shown in Figure 28.8.

Figure 28.7. SKB has de veloped a prototype of a remote-contr olled and radiation-shielded deposition machine

Prototype Repository

Particular aspects of the repository concept have previously been tested in a number of *in situ* and laboratory tests. We also must test and demonstrate a full-scale repository with state-of-the-art technology. This is done in the prototype repository at Äspö HRL, shown in Figure 28.9. The purpose of the prototype repository is to simulate the integrated function of the repository components and to provide a full-scale reference for comparision with models and assumptions. We also want to develop and test appropriate engineering standards, quality criteria, and quality systems. As shown in Figure 28.9, the test area consists of six deposition holes. It is divided into two sections: one inner section with four canisters and one outer section with two canisters. All conditions in a KBS-3 repository, with respect to geometry, materials, and rock conditions, are identical to a real repository. However, to provide the thermal field, electric heaters are used in place of nuclear fuel. The outer part of the repository will be excavated after five years, while the monitoring of the inner part will continue for another five to ten years.

28.3.3. The Siting Process

Choosing a site for a deep repository is no easy task. Many conditions must be met. During the last few years, SKB has devoted a great deal of effort to formulating requirements and criteria on what kind of rock is suitable for a deep repository and to finish the ongoing feasibility studies. The next effort is to start investigations on three sites in year 2002.

Feasibility Studies

Feasibility studies have been conducted in eight municipalities (see Figure 28.10). In two of these, Storuman and Malå, local referenda followed the feasibility studies in 1995 and 1997, on the question of further participation in the siting process for a deep repository. In both cases, the referenda resulted in a rejection of further investigations. The other six municipalities—Hultsfred, Oskarshamn, Nyköping, Tierp, Älvkarleby, and Östhammar—became the selection pool. Areas where the bedrock is judged to be potentially suitable for a deep repository have been found in all municipalities except Älvkarleby.

SKB has evaluated the alternatives with respect to requirements and preferences that can be evaluated today with regard to the bedrock, industrial establishment, and societal aspects. The results are given in the report I*ntegrated Account of Method, Site Selection, and Programme Prior to the Site Investigation Phase* (Swedish Nuclear Fuel and Waste Management Co., 2000). The feasibility studies have resulted in eight siting alternatives for the deep repository. These provide a selection pool for the choice of sites for subsurface investigations. The evaluation has been focused on the properties that are of importance for selection at this stage.

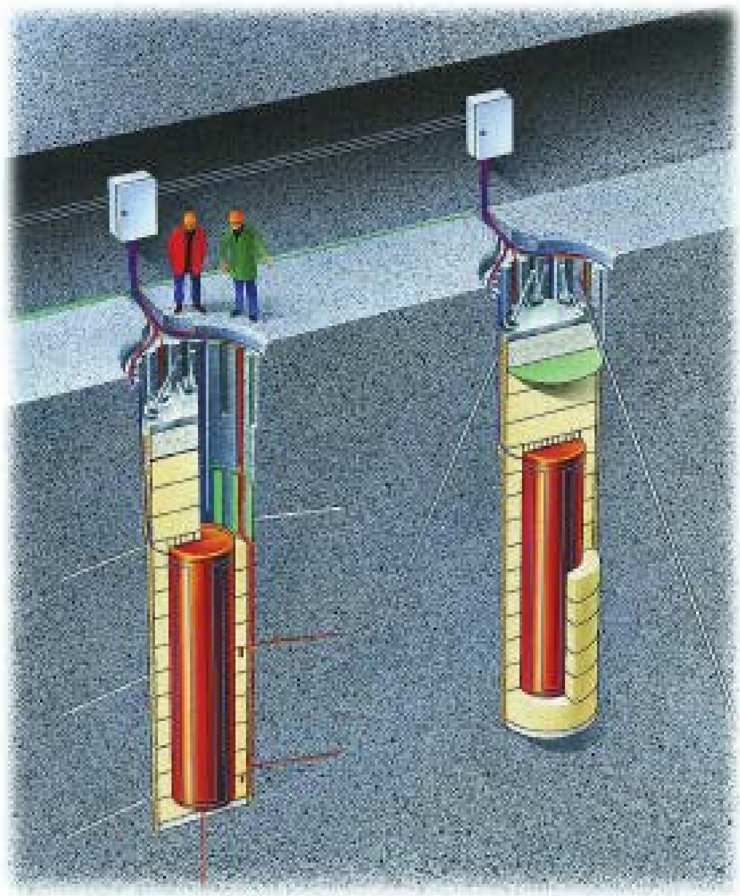

Figure 28.8. Schematic layout of retrieval test to demonstrate that canisters can be freed from water-saturated bentonite under realistic conditions

From our evaluations to date, we note that:

- All alternatives in the selection pool that can be checked meet (for the present) the safety requirements.
- The alternatives cannot be ranked in terms of safety based on what is known about the bedrock today.
- Test drilling is required to verify whether the bedrock satisfies the safety requirements. There is a risk that this drilling will reveal conditions leading to abandonment of a site.
- Simpevarp (close to Oskarshamn NPP) and Forsmark (close to Forsmark NPP) stand out as particularly suitable with respect to industrial establishments.
- There are good establishment prospects at several other siting alternatives, but the uncertainties are significant, because of (for example) the need for overland transport of spent nuclear fuel and/or development of new land areas for industrial purposes.
- The prospects of gaining public confidence and support in most municipalities for the deep repository project are difficult to assess and may change.
- It should be possible to gain such confidence and support in most localities, but the prospects are deemed to be particularly good for a siting at Simpevarp and Forsmark, provided the bedrock satisfies the safety requirements.

Against the background of the above evaluations and the current state of knowledge, SKB has prioritized the

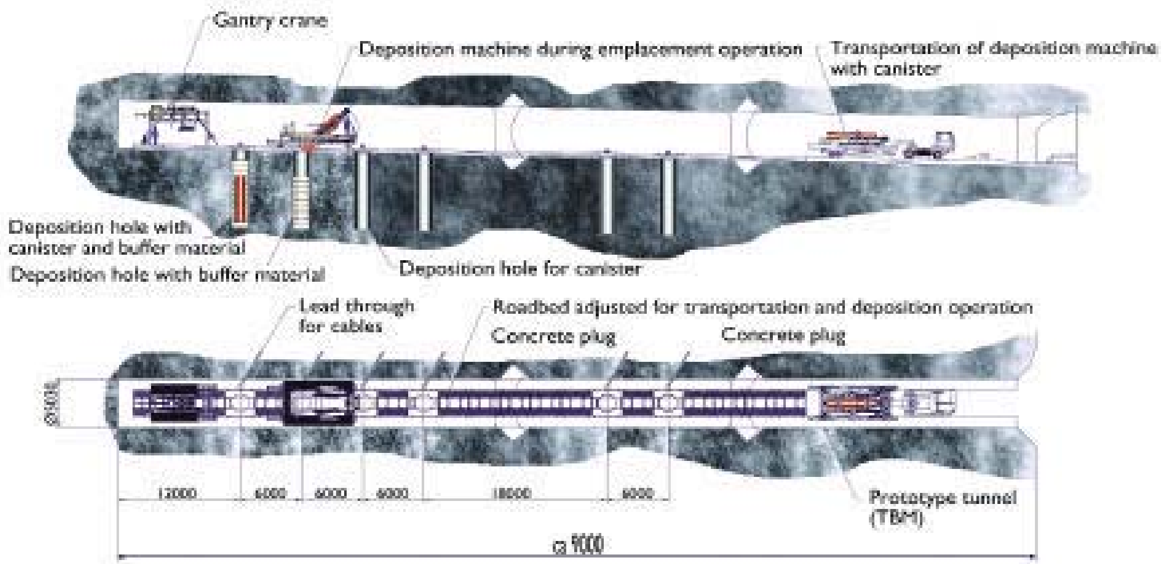

Figure 28.9. The prototype repository

site investigations with test drillings to be carried out at the following three sites:

- Forsmark in the municipality of Östhammar. The priority area is well defined and it should be possible to commence site investigations with test drilling shortly after a decision to proceed.
- Simpevarp in the municipality of Oskarshamn. Large areas with potentially suitable bedrock exist west of Simpevarp, but the area nearest the nuclear power plant is the most suitable. It should be possible to commence site investigations with test drilling after a decision to proceed.
- Tierp north/Skutskär. The priority area of north Tierp is relatively large, so that initial efforts must be focused on defining a satisfactory subarea for continued investigations and finding suitable locations for surface facilities and a transportation system. After that, SKB intends to initiate test drilling within the priority subarea. This alternative requires the continued participation of the municipalities of both Tierp and Älvkarleby.

Site Investigations
The next phase, site investigation, is aimed at gathering the data needed to site the deep repository and the encapsulation plant with associated support functions. Siting of the deep repository requires geoscientific investigations, with test drilling at selected locations and more in-depth local studies to establish the overall layout for the entire deep-disposal system. The goal of this phase is to obtain all permits needed to build the planned facilities. It will take an estimated 7–8 years to assemble the requisite supporting material, carry out consultations, compile siting applications, and have these applications reviewed by the appropriate authorities.

The siting investigations are divided into two main parts. Initially, investigations are performed to identify a specific site within a specified area that is deemed most suitable for a deep repository, and then to determine whether the initial judgment of its suitability is confirmed by in-depth data. This is expected to take 1.5–2 years. If the initial assessment is confirmed, the second part follows with complete site investigations for a period of 3.5–4 years. The purpose of these investigations is to gather the material required to select one of the sites as the main alternative and to apply for a permit for the deep repository.

Programs for the site-specific work, during the site-investigation phase, will be prepared during the first half of 2001. The programs will be designed to provide a complete database, as well as answers to specific questions, for each site. The database must also take into account the viewpoints of the municipality, landowners, and nearby residents, as well as important conservation measures and other interests.

28.4. CANISTER DESIGN AND FABRICATION

A reference canister consists of a 50 mm thick copper cylinder with welded-on top and bottom. Inside the cop-

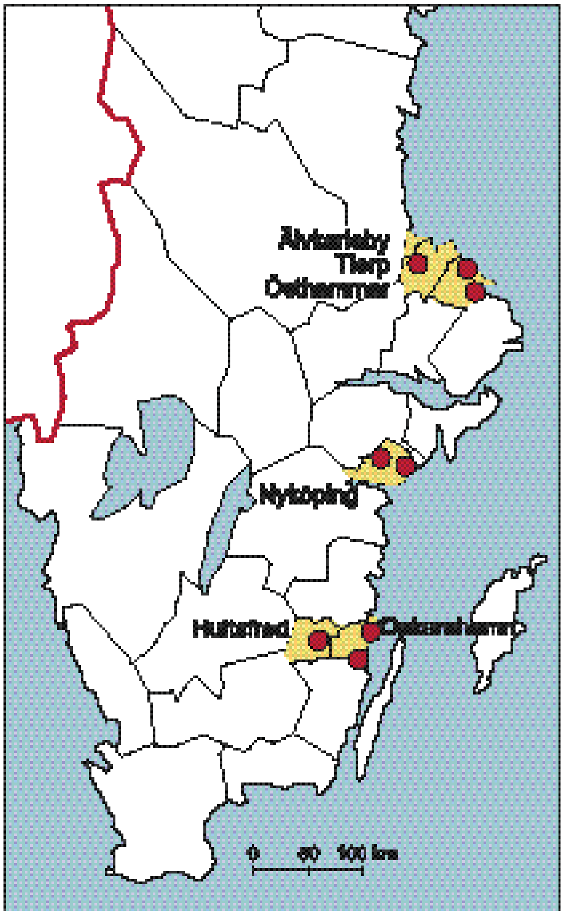

Figure 28.10. Locations of feasibility studies

Figure 28.11. Copper canister with nodular cast-iron insert for BWR fuel

per shell is a pressure-bearing insert of nodular iron (see Figure 28.11). The canister is not yet fully optimized. According to current plans, the thickness of the copper might be reduced to 30 mm at the same time as the insert is reinforced. A canister holds either 12 BWR assemblies or four PWR assemblies. Its outer diameter is 1,050 mm and its length is 4,830 mm.

In the last few years, fabrication trials with copper canisters and cast inserts have been conducted on a full scale by means of different methods. Full-size copper cylinders have been manufactured using roll forming of tube halves that are welded together, and two seamless fabrication methods: extrusion and pierce, and draw processing. The specified grade is equivalent to UNS 10100 (Cu OFE) or EN133/63:1994 Cu-OF1, with an addition of about 50 ppm phosphorous to increase creep ductility. All three fabrication methods have shown satisfactory results, although additional development work is needed. So far, seamless pipe manufacturing by extrusion seems to be the most promising method.

The principle and the technology for fabricating the insert have been finalized, and we have produced a number of inserts. The insert is cast in one piece. The channels in which the fuel assemblies are placed are fabricated of square steel tubes. Then the tubes are welded together to form a cassette, which is embedded in cast iron. Copper lids and bottoms are machined from material that has been performed by forging. To complete an empty canister, the bottoms are welded to the machined copper tubes and, after that, the insert is lowered into the copper shell.

A preliminary study of how a canister factory could be designed has been performed. The location of the canister factory has not yet been decided. Questions that must be taken into consideration in siting the factory include shipments to and from the factory, as well as availability of labor and the industrial infrastructure. Among the alternatives studied is a siting in the same region as the encapsulation plant or the deep repository, but other alternatives may also be considered.

28.5. ENCAPSULATION TECHNOLOGY

After loading the canister with fuel assemblies, the final sealing of the canister will take place in the encapsulation plant. About 200 canisters per year will be delivered from the canister factory to the encapsulation plant. Our first option is to join the encapsulation plant with the CLAB, the interim storage for spent nuclear fuel in Oskarshamn.

SKB's Canister Laboratory in Oskarshamn serves as a center for development of the encapsulation technology and training of personnel for the encapsulation plant. The main purpose is to test equipment for sealing and inspection of canisters on an industrial scale. This testing is needed to provide a good basis for the continued planning and design of the encapsulation plant. Results from the Canister Laboratory will provide an important part of the technology needed in the application for a license for the encapsulation plant. Much of the work is done in cooperation with TWI, a British welding institute. In 2004, the preferred sealing method will be identified for production purposes in the encapsulation plant.

28.6. RESEARCH

SKB has been conducting extensive research since the late 1970s. The main purpose has been to develop methodology and gather the scientific data needed to design the multi barrier system and assess the long-term safety of the repository. This research has laid the foundation for proceeding with the initial design of a deep repository. This does not mean the research is over; continued research can further improve the knowledge base.

The research program has several goals. The most important goal is to provide as complete a basis for safety assessment as possible. Research should also contribute towards optimizing the repository design so that its function is achieved as efficiently as possible. Moreover, the research should provide a basis for judging the development of alternative methods. The current research program is presented in *RD&D-Programme 98—Treatment and Final Disposal of Nuclear Waste* (Swedish Nuclear Fuel and Waste Management Co., 1998a) and *Detailed Programme for Research and Development* (Swedish Nuclear Fuel and Waste Management Co., 1998b). The next program is due in September 2001.

28.6.1. Long-Term Safety

The goal of the research on long-term safety is to understand the processes that occur in a deep repository and that affect its ability to isolate the waste. Further, the most essential processes must be quantified. Results from this research will serve as input for safety reports that provide a reliable basis for analyzing the long-term performance of the barriers. The final step is to assess and report the total safety of the repository.

In addition to the R&D required to meet safety-assessment needs, R&D is also needed for the repository design and for the site–investigation program. For example, materials must be developed for use in the repository, and the site investigations require development of methods and equipment.

28.6.2. Alternative Methods

R&D is also required for odd waste forms and for investigations on alternative methods. Direct disposal of spent nuclear fuel is the main strategy, but SKB is also keeping track of, and supporting, the development of alternative methods. It is as difficult for us, as for anyone else, to know what will be technically possible to do in the distant future, but considering the possibilities of partitioning and transmutation, SKB will follow progress in the field to determine whether the alternatives have any future potential.

28.7. SAFETY ANALYSES

In preparing for the forthcoming site investigations for a deep repository, SKB presented a new safety report, *SR 97—Post-Closure Safety*, at the end of 1999 (Swedish Nuclear Fuel and Waste Management Co., 1999). The calculations in *SR 97* are based on specific data from three actual sites. The sites differ when it comes to important conditions such as climate, groundwater flow, land uplift, and natural environment. The purpose of choosing sites with these differences is to give us an idea of how sensitive the performance of the repository is to variations between the different sites. We also want to know what site conditions are most important for safety in different situations.

The methodology in the assessment entails first describing the physical condition of the repository when it has just been closed and then analyze how that condition changes with time as a result of both internal processes in the repository and external forces.

The evolution of the repository is analyzed in terms of five scenarios. The first is a base scenario, in which the repository is postulated as having been built entirely according to specifications and where present-day conditions in the environment, including climate, are assumed to persist. The four other scenarios reveal how the performance of the repository may differ from that in the base scenario, if there are: (1) a few initially defective canisters, (2) climate changes, (3) earthquakes, or (4) future inadvertent human intrusions. Repository performance is analyzed in terms of thermal, hydraulic, mechanical, and chemical processes, with the ultimate purpose of evaluating the repository's capacity to isolate the waste in the canisters and to retard any releases of radionuclides if canisters are damaged. In accordance with preliminary regulations, the time domain for these analyses is at most one million years.

The principal conclusion of the *SR 97* safety assessment is that the prospects of building a safe deep repository for spent nuclear fuel in Swedish granite are very good. The three sites investigated reflect reasonable variations of conditions in a granitic bedrock. The analysis does not provide support for attaching any particular significance to differences in long-term safety between sites, when all the factors that influence performance are weighed together. Another conclusion is that the methodology used in *SR 97* provides a good foundation for future safety assessments, which will be based on data from complete site investigations. The results of the assessment also serve as a basis for: formulating requirements and preferences regarding bedrock in site investigations, designing a program for site investigations, formulating functional requirements at the repository's barriers, and prioritizing research. The next stage in siting a deep repository will involve an investigation of the bedrock at three candidate sites. It is SKB's judgment that the scope of the safety assessment and confidence in its results must satisfy the requirements that should be made in preparation for such a stage. *SR 97* has been subjected to an international review (OECD/NEA, 2000) and has also been reviewed by the Swedish authorities.

28.8. CONCLUSIONS

The Swedish program for spent-fuel management is now focused on the siting process for the deep repository. The nearest date for starting site investigations is 2002. SKB has opened discussions with the concerned municipalities regarding the additional information they may need to arrive at a decision about participation in the ongoing program. Provided that the material now being presented by regulatory authorities and the Swedish government is in agreement with the stipulated timetable and has a positive outcome, it is SKB's opinion that the municipalities should be able to arrive at a decision on the Swedish program by the end of 2001.

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Chapter 29

Swiss Geological Studies to Support Implementation of Repository Project: Status 2001 and Outlook

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29.1. 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; about 40% of the electrical energy is supplied from nuclear plants, with almost all of the rest being generated by hydropower.

Nuclear power production is the main source of Swiss radioactive waste, although waste results also from medical, industrial, and research applications. Switzerland currently has five nuclear power plants (pressurized-water reactors [PWR] and boiling-water reactors [BWR]) with a total capacity of 3 GW(e). Spent fuel, which contains most of the waste radionuclides produced by fission, 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. Currently, however, the preferred strategy of the utilities is to keep both options open (reprocessing or direct disposal of spent fuel), although a revision of the nuclear law (under review at present) proposes to ban future reprocessing.

The legal and regulatory background to nuclear waste management was described in some detail in a previous review (McKinley and McCombie, 1996). The legal framework is, however, under review as part of the revision of the nuclear law expected to be promulgated within the next year or so. Apart from stopping reprocessing (as mentioned above), this new law will modify the decision-making framework for future disposal projects. It also envisages a concept of "monitored" geological disposal consistent with the Entsorgungskonzepte für radioaktive Abfälle (EKRA, 2000) committee proposals.

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 waste. The responsibility for spent-fuel reprocessing and transport, for waste conditioning at power plants, and for interim storage remains directly with the utilities. In 1994, a separate organization was founded to actually construct and operate a low- and intermediate-level radioactive waste (LILW) repository at a site selected and characterized by Nagra—the Genossenschaft für die Nukleare Entsorgung Wellenberg (GNW).

29.2. CHARACTERISTICS AND 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 (Nagra, 1992), one for LILW and another for high-level waste (HLW) and intermediate-level waste containing higher concentrations of long-lived or alpha-emitting radionuclides (transuranic [TRU] waste).

Highest priority at present is allocated to the LILW repository, which is intended to be implemented in horizontally accessed rock caverns with a few hundreds of 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 development of Wellenberg as a repository site was accepted in public referenda in the local community, but was blocked by a narrow margin at the cantonal level.

A further cantonal referendum, expected in 2002, will decide whether or not to allow an exploration tunnel to be constructed at this site. This tunnel could provide an expanded geological database to support subsequent application for a repository construction license. In the interim, the repository concept has been modified following input from an independent expert group (EKRA, 2000). Introduction of a "pilot facility," in which some waste is disposed and which is monitored intensively during (and possibly after) operation of the main repository, now provides the basis to make the final decision on when to close and decommission the facility.

For the present limited nuclear program in Switzerland, operation of all plants for 40 years will result in around 3,000 tons of spent fuel. Existing reprocessing contracts cover about 1/3 of this inventory, and it is expected that all remaining spent fuel will be conditioned for direct disposal. Vitrified waste and some LILW from reprocessing abroad will be returned to Switzerland for disposal. It is planned to store HLW for at least 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 spent fuel and vitrified HLW and for other reprocessed wastes has recently been constructed by the Zwischenlager Wuerenlingen AG (ZWILAG) organization, a daughter company of the utilities.

The existence of ZWILAG removes much of the time pressure from the Swiss disposal program. HLW cooling (especially high-burnup UO^2 and MOX spent fuel if directly disposed) and economic arguments suggest an operational date for an HLW repository around 2050. Nevertheless, there is a requirement to demonstrate the fundamental feasibility of siting a repository in Switzerland. This demonstration—termed "Projekt Entsorgungsnachweis"—is planned to be submitted to the authorities in late 2002.

Site selection for HLW is very much constrained by the small size of Switzerland and by its geological setting. The current geological consensus is that the orogeny which built the Swiss Alps is still continuing, and there is still net uplift in this area of ~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, two host rock options are considered—either the crystalline basement or one of the overlying, low permeability sediment layers (first priority Opalinus Clay, Lower Freshwater Molasse as a reserve).

The current conceptual repository design was developed taking into account the potential host rocks, the very low volumes of HLW expected, and the government requirement for an early, convincing demonstration of waste-disposal safety as a condition of extending reactor operating licenses. These factors together led to designs that are certainly robust (or even overdesigned), but which are not yet necessarily optimized in an economic or operational sense.

The concept (see Figure 29.1) has the following features:

1. Deep disposal (about 500 m to 1 km below surface) in a specially constructed facility
2. In-tunnel emplacement of 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 may (especially for Opalinus Clay) act as a very efficient radionuclide transport barrier
3. Massive engineered barriers: in addition to the vitrified waste in its steel fabrication canister or spent UO_2/MOX fuel within its cladding, a thick steel overpack is envisaged, surrounded by compacted bentonite clay.
4. Co-disposal of TRU in a separate part of the repository (not shown in Figure 29.1).

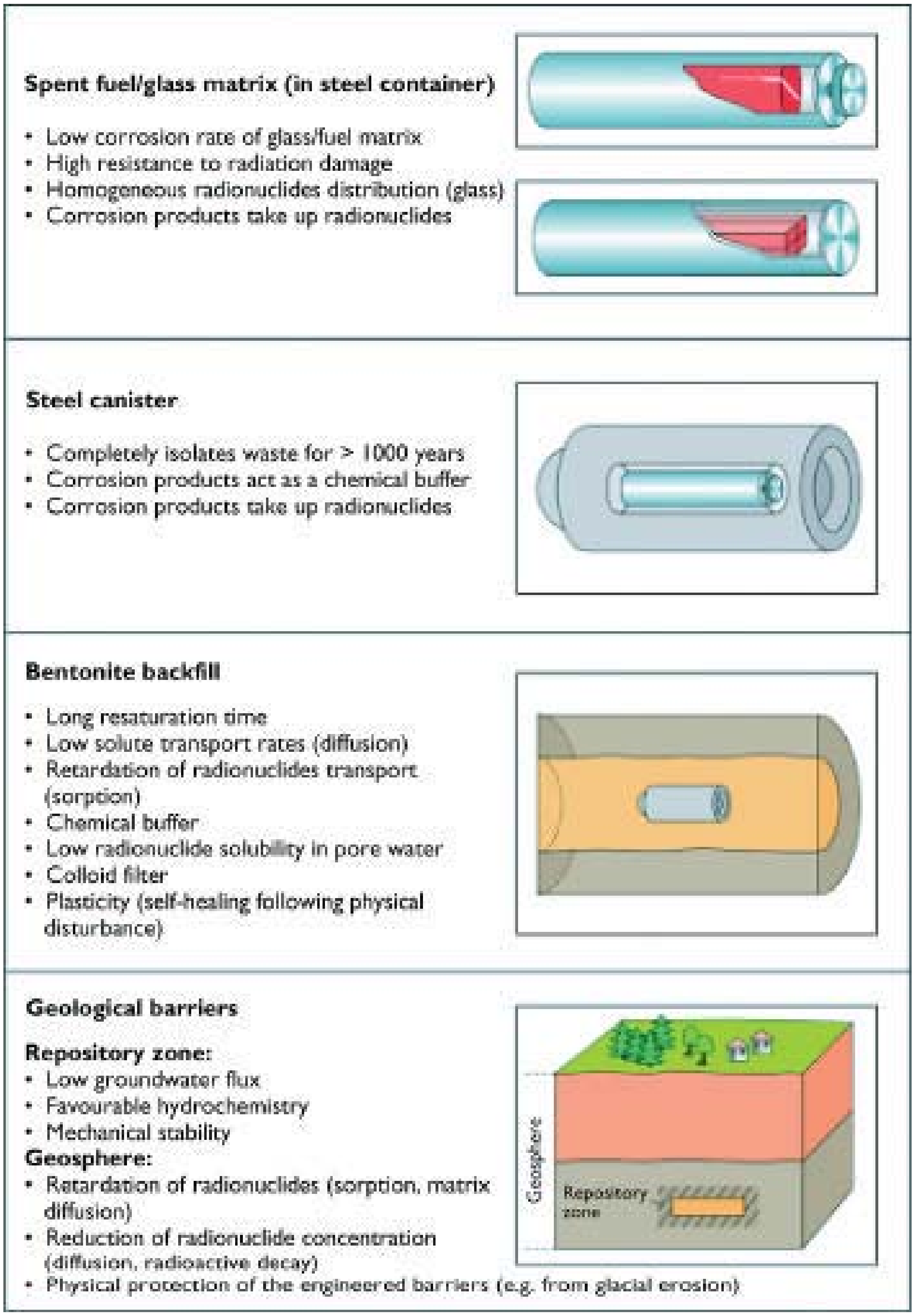

Figure 29.1 Safety barrier system for HLW/SF

Analysis of this 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 review also strongly recommended that the option of disposal in sedimentary formations be considered in more depth. Despite these caveats, the waste disposal issue was no longer tied directly to operation of existing power plants.

Since 1985, the regional investigation of the crystalline basement has been completed and documented (Nagra, 1984a; 1984b). 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 Permo-carboniferous trough that cuts the region. Only two restricted areas remain for selection of a possible site, each covering about 50 km^2. Nevertheless, there are strong arguments that it would be feasible to find a suitable repository for the required low volume of waste (Nagra, 1994a).

In parallel, investigations of the sedimentary options have proceeded from a desk study to select potential host formations through to identification of specific potential siting areas. The two sedimentary host rocks investigated in detail were Opalinus Clay, which exists in a laterally extensive but rather thin layer in Northern Switzerland, and Lower Freshwater Molasse, where the formations are large but somewhat heterogeneous (Nagra, 1994c). For Opalinus Clay, which was identified as the higher priority option for Nagra, a field program in the "Zürcher Weinland" potential siting area has recently been completed (see below).

The next major milestone in the HLW program will be the demonstration of repository siting feasibility (Project Entsorgungsnachweis) scheduled for 2002. This will focus on the Zürcher Weinland/Opalinus Clay option mentioned above, but the crystalline basement is maintained as an alternative host rock.

29.3. SITE-CHARACTERIZATION STUDIES TO SUPPORT PROJECT ENTSORGUNGSNACHWEIS

Following the work on the crystalline host-rock option described previously (Nagra, 1994b), site-characterization work since about 1995 has focused on Opalinus Clay. 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 Züricher Weinland exploration area. Because it shows a good velocity contrast with neighboring formations, the site characterization focused on close-mesh 3D reflection seismics (Nagra NTB 00-03).

In a densely populated country like Switzerland, this presented a considerable logistical challenge. Within the 50 km^2 investigation area, 9,000 excitation points had to be distributed over a total of 4,400 land parcels, taking into account all service facilities (pipes, cables, etc.), structures, and protected areas. In fact, 98.4% of all planned shotpoints could be implemented (with only 2% of around 1,700 landowners refusing access to their land).

The resulting database was subject to detailed processing, which allowed direct visualization of displacements within the Opalinus Clay of a magnitude $\geq$ 10 m. Further state-of-the-art image processing allowed continuous structures with much smaller displacements to be clearly recognized (Figure 29.2).

The interpretation of the seismic profiles was calibrated by an exploration borehole drilled at Benken in the middle of the study area (Nagra NTB 00-01). This not only gave confirmation of the 3D seismic results (via check shots, sonic logs, and vertical seismic profiling), but also provided basic information on the rock-mechanical, hydrogeological, lithological, and geochemical properties of the Opalinus Clay and surrounding formations. These characteristics are further supported and complemented by studies off site at the Mont Terri underground test site (see below) and observations in other deep boreholes (e.g., hydrocarbon exploration) and tunnels intersecting these sediments.

Documentation on the 3D seismic survey and the Benken borehole will be integrated with input from Mont Terri and other national/international studies in the "Geosynthesis" to support Project Entsorgungsnachweis (documentation in production; final references will be available on the Nagra homepage: www.nagra.ch).

29.4. 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.

The Grimsel Test Site (GTS) is situated below ~ 500 m of overburden in granite/granodiorite of the Swiss Alps. Over a period of almost 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 (Figure 29.3), 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.

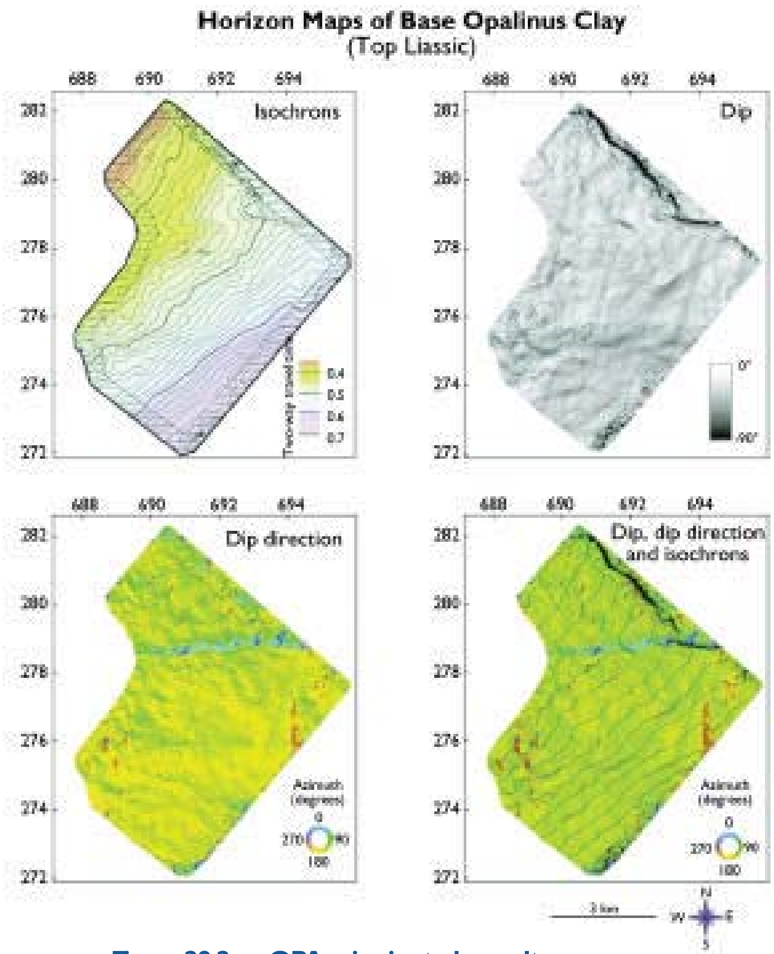

Figure 29.2. OPA seismic study results

Nagra and 17 other organizations from nine countries involved in work at the GTS are now investigating options for a further phase of work when the present "Phase V" ends in 2002/2003. There are indications that GTS could serve as an international center of excellence for specific *in situ* studies where considerable experience has been gained, such as:

- *In situ* studies with radionuclides—part of the GTS is established as an IAEA class B laboratory for handling a wide range of isotopes (Table 29.1)
- Large-scale long-timespan studies of demonstration/optimization of waste handling, emplacement, monitoring, and retrieval.

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 (managed by the Swiss Federal Office of Water and Geology) is now entering a seventh one-year phase involving Nagra and 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 (Figure 29.4). Again, in principle, other organizations could join future phases of work at this site (see www.mont-terri.ch).

29.5. THE SWISS PROGRAM IN AN INTERNATIONAL CONTEXT

The Swiss waste management program, although relatively small in terms of budget and manpower, is very wide in scope—with one site currently selected for establishing an LILW repository and two types of host rocks under investigation for co-disposal of spent fuel, vitrified HLW, and long-lived ILW. 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 and the Nuclear Energy Agency, Nagra has formal agreements with the EEC, Belgium (CEN/SCK), the Czech Republic (RAWRA), Spain (ENRESA), France (CEA/ANDRA), Germany (GSF/BGG), Japan (JNC, CRIEPI, NUMO, JNFL and Obayashi), Taiwan (AEC/FCMA and INER), Sweden (SKB), Finland (Posiva), USA(DOE and NRC) and the UK (NIREX). Informal collaborations extend the list further.

Since 1997, Nagra has also expanded the provision of 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 cost (now ~800 M SFr. over ~30 years) within the Swiss national program can be made available for other purposes, while Nagra staff has the possibility to further widen their experience, which provides a perspective that is essential for such a small program.

29.6. 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 have now been solved. Key milestones in the near future—the Referendum on the Wellenberg LILW site and the demonstration of HLW siting feasibility in Northern Switzerland—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. Also, as we move toward implementation, the demonstration of the practicability of handling and emplacement—and, if required, monitoring and retrieval—are also of importance. Finally, for very low volumes of HLW/TRU, it is policy in Switzerland to investigate options for multinational or regional repositories after we have demonstrated that disposal within Switzerland is possible. This idea may gain prominence over the next decade.

Table 29.1.: List of isotopes used / planned at GTS

Isotopes/radiotracers already applied	Additional radiotracers in discussion
H-3	Co-58
He-4	I-129,I-131
Na-22,Na-24	Cs-134
Co-60	Eu-154
Se-75	Th-232
Br-82	Pu-238,Pu-242,Pu-244
Sr-85	Am-241,Am-243
Rb-86	
Tc-99	
Sn-113	
I-123	
Cs-137	
Eu-152	
U-234,U-235	
Np-237	

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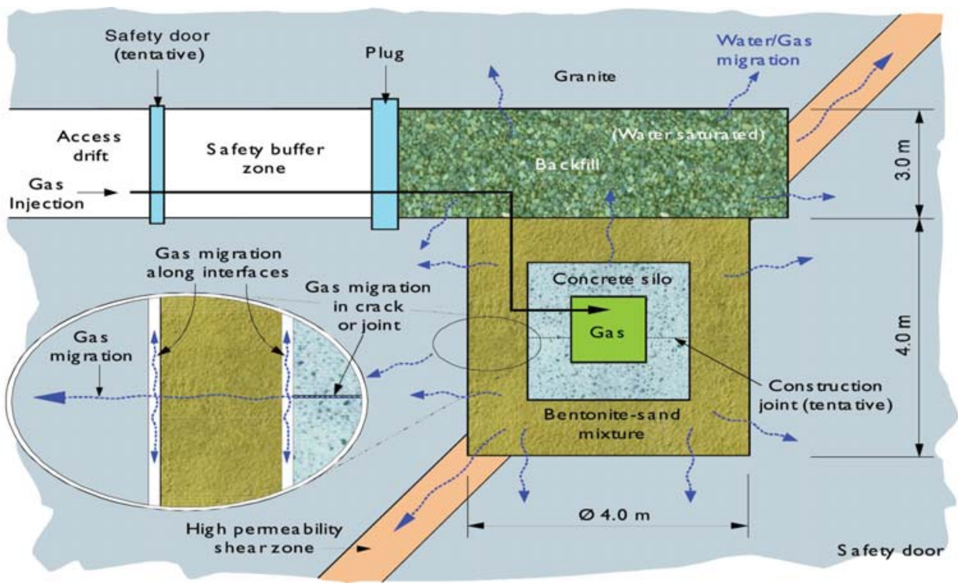
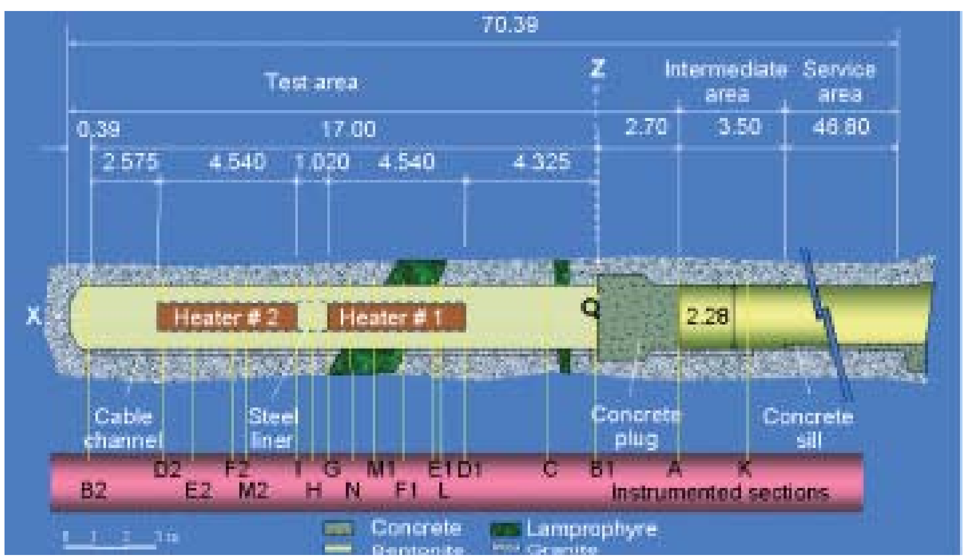


Concept GMT-Silo disposal (L/ILW, TRU)
Gas Transport through the EBS and geosphere, scale 1:10



Concept FEBEX–Horizontal emplacement of HLW canisters, scale 1:1
Thermal-hydro-mechanical & geochemical processese

Figure 29.3. GMT/FEBEX: Examples of large-scale tests of the EBS

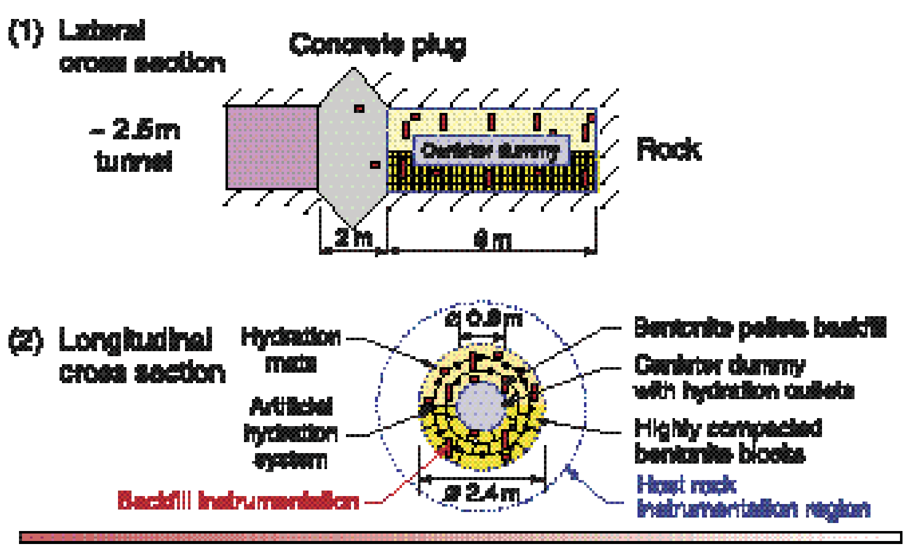


Figure 29.4. Plan of Mont Terri EB experiment

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Chapter 30

Current Status of Radioactive Waste Administration in Taiwan

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ABSTRACT. Taiwan is a country lacking in natural energy resources, and thus nuclear energy is at present an important energy source, contributing up to 25% of Taiwan's total electricity production. Radioactive waste management necessitated by nuclear energy applications must be carefully planned and credibly implemented. This paper introduces government policies, relevant laws and regulations, the status of low-level radioactive waste (LLW) management in the national nuclear power plants (NPPs), and the development of LLW final-disposal and spent-fuel management programs. As in many other countries, public acceptance is a key factor in the future development of nuclear energy, and intelligent, careful radioactive waste management is an important part of earning public support. Measures to improve public participation and to enhance public acceptance of nuclear energy in Taiwan are now under serious discussion.

30.1. INTRODUCTION

The commercial operation of Chinshan NPP Unit One in 1978 marked the beginning of Taiwan's nuclear power program. Currently, there are three NPPs in operation, each consisting of two units. Collectively they represent a generating capacity of 5,144 MW, contributing 23.6% of national electricity supplies in 2000. Detailed information about Taiwan's nuclear power program is shown in Table 30.1.

As far as low-level radioactive waste (LLW) is concerned, the Taiwan Power Company (TPC) is the principal producer, contributing more than 90% of the total volume of waste in Taiwan. Small producers (medical facilities, research institutes, and universities) are responsible for the remaining 10%. In terms of spent-fuel management, TPC is the only organization that generates spent fuel from commercial operations. In this report, we address Taiwan's radioactive waste management policy, the structure of radioactive waste management organizations, and regulations for radioactive waste treatment, storage, transportation, and disposal (as well as spent-fuel disposal).

30.2. RADIOACTIVE WASTE MANAGEMENT POLICY AND ORGANIZATIONAL STRUCTURE

30.2.1. Policy

The Atomic Energy Application and Development Policy (AEADP) and the Radioactive Waste Management Policy (RWMP) (as proclaimed in 1991 and 1997, respectively) set out the principal guidelines for management of radioactive waste in Taiwan. The two policies set down the following basic positions:

- The responsibility for safely treating, transporting, storing, and disposing of radioactive waste should rest with the producer. Therefore, the producer is responsible for all necessary expenses.
- Radioactive waste administration must consider the safety of citizens and the protection of the environment, and observe related international conventions.
- Radioactive waste producers should strive to minimize waste generation and reduce its volume.
- The feasibility of regional/international cooperation in radioactive waste disposal should be assessed in

Table 30.1. Information on nuclear power plants in Taiwan

Unit	Reactor Type	Installed Capacity (MWe)	Commercial Operation	Status
Chinshan 1	BWR/4	636	1978	Operating
Chinshan 2	BWR/4	636	1979	Operating
Kuosheng 1	BWR/6	985	1981	Operating
Kuosheng 2	BWR/6	985	1982	Operating
Maanshan 1	PWR	951	1984	Operating
Maanshan 2	PWR	951	1985	Operating
Yenliao 1	ABWR	1,300	2004	Under Construction (30%)
Yenliao 2	ABWR	1,300	2005	Under Construction

parallel with a domestic option. A domestic option must be available even if extraterritorial disposal becomes possible.

30.2.2. Structure of Radioactive Waste Management Organizations

Organizations involved in radioactive waste management are shown in Figure 30.1. Both the Atomic Energy Council (AEC) and the Ministry of Economic Affairs (MOEA) are under the Executive Yuan. The Fuel Cycle and Materials Administration (FCMA), a subordinate organization to the AEC, has regulatory control over radioactive waste management matters. The Institute of Nuclear Energy Research (INER) was empowered by AEC to take responsibility for collecting radioactive waste generated by small producers and to treat the waste if necessary. In TPC, the Nuclear Back-end Management Department (NBMD) and the Nuclear Operation Department (NOD) are responsible for the off-site and on-site radioactive waste management of NPPs, respectively. NOD's major responsibility is to supervise treatment and storage of LLW within NPPs, whereas NBMD is responsible for radioactive waste transportation, the operations of both the Lan-yu Storage Site and the Volume Reduction Center, and (most importantly) the final disposal of LLW and spent fuel in Taiwan.

30.3. DEVELOPMENT OF RADIOACTIVE-WASTE-RELATED REGULATIONS

The Atomic Energy Act, enacted in 1968, established the legal basis for regulating nuclear energy activities in Taiwan. Over the years, regulatory guidelines and standards have been put into effect by AEC to keep pace with international developments and domestic needs, focusing on enhancing public health and safety and environmental protection. A law put into effect in 1999 stated that all regulations without approval or authorization by the Legislative Yuan will lose legal effect after January, 2001. Since 1999, a reorganization of all regulatory bylaws and guidelines was carried out to meet this deadline.

Following one and a half years of extensive effort, AEC submitted a draft of the Radioactive Materials Management Act (RMMA) to the Executive Yuan for review. This act was enacted to regulate nuclear source material, nuclear fuel, and radioactive wastes; to prevent potential radioactive hazards to the public and the environment; and to fulfill our commitments to international standards, including nuclear nonproliferation. On March 2000, the Executive Yuan gave approval in principle to RMMA and transmitted it to the Legislative Yuan. It has passed the first run of legislative review and is expected to come into force later this year.

The AEC, after consulting with stakeholders over a two-year period, developed 10 regulations to supplement the implementation of the RMMA. RMMA and its supplementary regulations are now posted on the FCMA Web site for public comments. Once the RMMA is approved by the Legislative Yuan, all these regulations will automatically be enacted into law. The ten regulations are as follows:

1. Regulation for the Detail Implementation of the Radioactive Materials Management Act
2. Regulation for the Operational Safety of Nuclear Source Materials
3. Regulation for the Operational Safety of Nuclear Fuels
4. Regulation for the Operational Safety of Radwaste Treatment Facilities
5. Regulation for the Operational Safety of Radwaste Storage Facilities
6. Regulation for the Operational Safety of Low-Level-Radwaste Final Disposal and Facilities

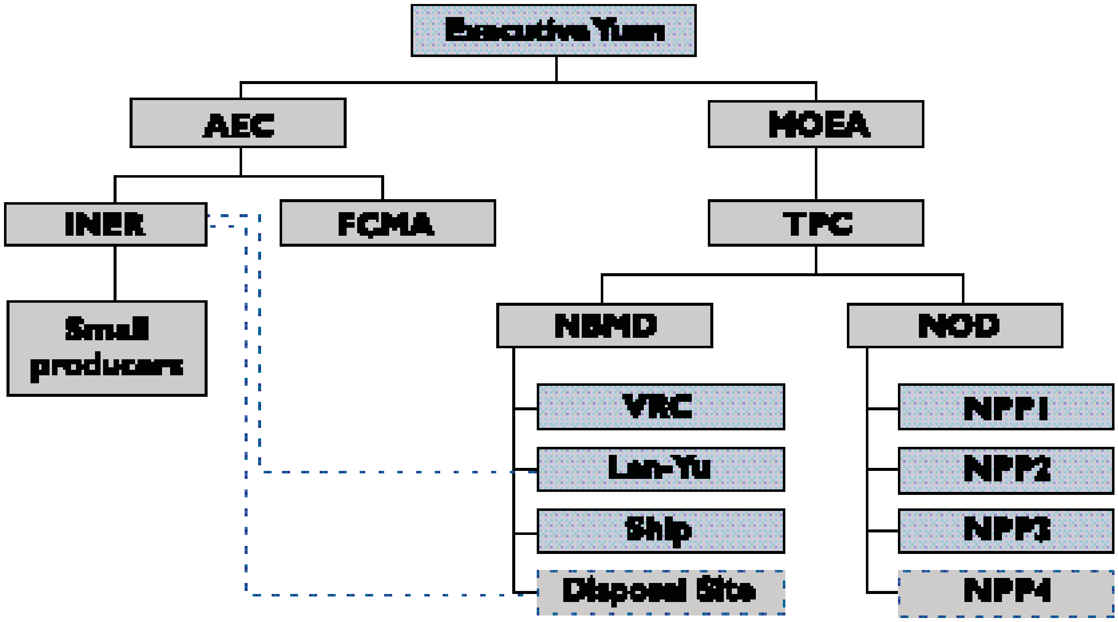

Figure 30.1. Organizations involved with radioactive waste management in Taiwan

7. Regulation for the Operational Safety of High-Level-Radwaste Final Disposal and Facilities
8. Regulation for the Management of Waste Derived from Naturally Occurring Radioactive Materials
9. Regulation for the Management of Radioactive Wastes Below Regulatory Concern
10. Regulation Governing Nuclear Safeguards Operations.

30.4. LLW MANAGEMENT

The management of LLW in Taiwan can be summarized as shown in Figure 30.2. Highlights of the program include plans for radioactive waste volume reduction, control over the Lan-Yu interim storage site, and on-site storage.

30.4.1. Radio active Waste Volume Reduction

LLW is divided into two main categories: wet waste and dry active waste. Wet waste (mainly evaporation residue, filter sludge, and spent resins) is first solidified inside galvanized steel drums and then stored in dedicated storage warehouses. Dry active waste (mainly waste paper, clothes, plastics, woods, and metal) is either segmented or shredded and then forwarded for volume reduction by incineration and supercompactor at the Volume Reduction Center (at the Kuosheng NPP). As of January 2001, the cumulative amounts of LLW in storage are as listed in Table 30.2. Cement is the most commonly used solidification agent for wet waste. While bitumen had been used for mixing with incinerator ash, we have recently switched to compaction by a super-compactor instead.

As a result of persistent efforts implemented by Taipower, annual solidified LLW production has been reduced from a high of 11,814 drums per year in 1983 to less than 5,000 drums in 1990, and further to 1,081 drums in 2000. The generation rate has been significantly reduced, to 11% over the past two decades (see Figure 30.3). In comparison with operational data provided by the World Association of Nuclear Operation (WANO), waste generation from our BWR plants has achieved a level slightly lower than the world average, and our PWR plant has achieved an excellent record, among the top three plants in the world.

The Volume Reduction Center at the Kuosheng NPP is equipped with a 100-kilogram-per-hour incinerator, at a volume reduction factor of 96.7% to 99%, and a 5-waste-drums-per-hour supercompactor at a volume reduction factor of 66.6% to 80%. Incinerators will also be built in the Chinshan and Maanshan plants within a few years. Research on plasma burner technology is now ongoing in INER.

The High-Efficiency Solidification Technology (HEST)

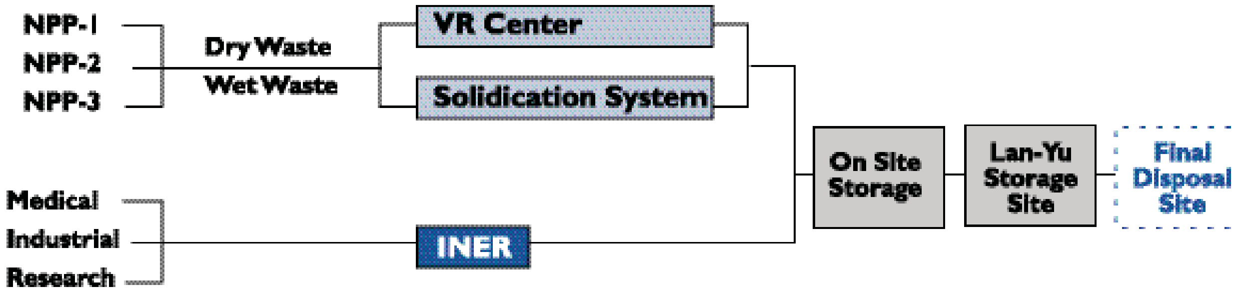

Figure 30.2. Diagram of low-level radioactive waste management in Taiwan

was first developed by INER and successfully implemented at the Maanshan NPP in 1998. HEST contributes greatly to volume reduction results because of its high-volume-efficiency characteristics. Expectation is high for further reduction of waste at the other two plants, if installed.

30.4.2. Control over the Lan-Yu Interim Storage Site

The Lan-Yu Storage Site has provided off-site interim storage for solidified radioactive waste since 1982. This site is located in a small Lan-Yu islet with a surface area of about 45 km^2 and contains a storage capacity of about 98,000 55-gallon drums in 23 semi-underground engineered trenches. The site had received 97,672 waste drums when it ceased receiving waste in 1996, mainly in response to local opposition.

30.4.3. On-Site Storage

Because of the difficulties in continuing to receive waste at the Lan-Yu site, TPC started the construction of a dedicated storage warehouse at each nuclear power plant. These include a 40,000-drum facility at the Kuosheng NPP, a 23,000-drum facility at Chinshan NPP, and a 40,000-drum facility at the Maanshan NPP. As a result, since 1999, the temporary warehouses for LLW storage have been completely replaced by air-conditioned, automated, and well-shielded storage facilities.

30.5. LLW Final Disposal

Permanent disposal of LLW presently stored on-site or in the Lan-Yu site is being planned. According to the RWMP, and in light of the fact that TPC contributes 90% of LLW generated in Taiwan, TPC has been charged with this responsibility.

30.5.1. Regulatory Requirements

According to the "Regulation for the Operational Safety of Low-Level-Radioactive Waste Final Disposal and Facilities," the annual dose to any member of the public resulting from radioactivity in a disposal site must not exceed 0.25 mSv. A set of siting criteria for LLW final disposal was outlined in the Regulation, including items such as:

- Repository site selection, operation, and closure must consider the re-utilization of the repository site.
- Sites should be situated in an area with low population density and low development potential.
- Areas where tectonic activity, geological processes, or hydrological and hydrogeological conditions could endanger the safety of the disposal facility should be avoided.
- Areas where geological and hydrological data are too complicated to be adequately evaluated should be avoided.

Table 30.2. LLRW storage in Taiwan

Waste	Chinshan	Kuosheng	Maanshan	Lan-Yu	INER	Subtotal
Combustible	9,061	514	4,437	0	2,508	16,520
Compatible	6,108	3,139	968	0	2	10,217
Spent Resin	2,074	1,720	1,134	0	136	5,064
Solidified	5,656	23,325	2,361	97,672	3,325	132,339
Others	7,605	7,321	0	0	6,503	21,429
Subtotal	30,504	36,019	8,900	97,672	12,474	185,569

Note: 1. Unit: 200-liter drum; 2. As of the end of January 2001.

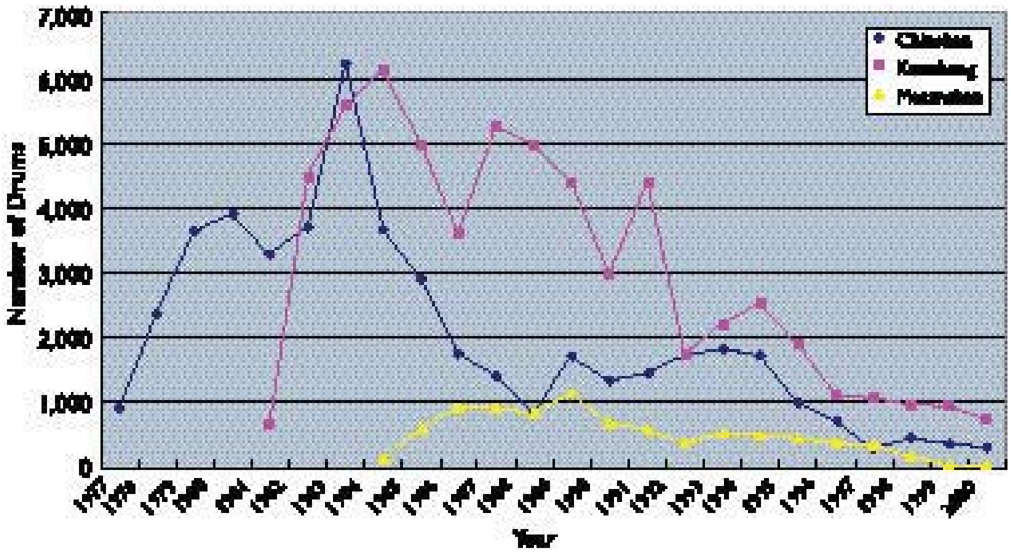

Figure 30.3. Annual solidified LLW production at nuclear power plants in Taiwan

- Areas acknowledged with important natural resources, ecological protection needs, and historical reservation values should be avoided.

30.5.2. LLRW Final Disposal Program

The TPC's program for LLRW disposal will be carried out according to the following phases:

Phase 1. Selection of disposal site and method
Phase 2. Environmental survey and assessment
Phase 3. Site characterization, engineering design, and licensing
Phase 4. Site construction
Phase 5. Operation
Phase 6. Post-operation monitoring.

TPC reported a preliminary site selection to the AEC on February 25, 1998. Little Chiu Yu at Wu-Chiu Hsiang was selected by TPC as the priority candidate site for investigation. The islet, under the jurisdiction of Kinmen County, is located off the China Mainland coast, about 80 nautical miles from Taiwan's west cost and about 10–20 nautical miles from mainland China.

TPC submitted an Environmental Impact Statement (EIS) to the Taiwan Environmental Protection Agency (EPA) in November 2000 for review. The Investment Feasibility Study Report (IFS) is scheduled for submission to the Ministry of Economical Affairs in Summer 2001, and the Safety Analysis Report is to be submitted to the AEC upon approval of the EIS and IFS reports. Little Chiu Yu can be formally qualified as a disposal site only after receiving approvals from these three organizations.

30.6. SPENT-FUEL MANAGEMENT PROGRAM

30.6.1. Interim Storage

Spent fuel is now temporarily stored in pools at each of the nuclear power plants. Although the re-racking projects for the spent-fuel pools were undertaken in 1992, 1995, and 1999 for the three NPPs, the first and second NPPs will lose their full core reserve by the years 2008 and 2007, respectively (see Table 30.3). In 1995, EPA reviewed and approved the Environmental Impact Statements for the Spent Fuel Interim Storage Program for the first and second NPPs. On-site dry storage was confirmed as a favorable option after a detailed study of the technical, economic, and environmental impacts.

30.6.2. Final Disposal

Regarding the final disposal of spent nuclear fuel, TPC first launched a two-year Phase 1 study project in May 1986. The information obtained from geological surveys indicated that potential host rocks, including granite, shale, and mudstone, can be found at appropriate depths in some parts of Taiwan. Asubsequent Phase 2 study project was started in November 1988. The scope of work included formulating a long-term R&D plan for the development of final disposal technologies and conducting geological investigations on the potential host rock. This study was completed in June 1991. A six-phase development plan (Figure 30.4) that spans 40 years was proposed, in recognition of the following considerations:

Table 30.3. Spent Fuel Storage Capacity in Nuclear Power Plants							
Unit		Year of Commercial Operation	Capacity Before/After Re-racking*	Year of Re-racking Operation	Current Storage Inventory*	Expected Year of Full Occupation	Discharge/ Fuel Cycle*
Chinshan	#1	1978	1,410 / 3,083	1987 / 1999	1,944	2008	130
	#2	1979	1,620 / 3,083	1987 / 1999	1,976	2009	130
Kuosheng	#1	1981	2,469 / 4,237	1992	2,628	2007	220
	#2	1982	2,469 / 4,237	1992	2,456	2008	220
Maanshan	#1	1984	746 / 2,151	1994	736	2025	72
	#2	1985	746 / 2,159	1994	745	2026	72
*Per fuel assembly unit							

- A long-term investigation plan is required to select a suitable candidate site with preferred geological characteristics for future development of a geological repository, with sufficient information needed for safety assessments.
- Spent-fuel interim storage for 40 years or longer is required to provide ample time for implementing the final disposal plan and ensuring flexibility in consideration of other options that are proven, beneficial, and feasible in the future.

Based on this plan, Taipower's main goal in the last few years was to develop the techniques, methodologies, quality assurance infrastructure, and equipment required for conducting the follow-on surface investigation; and field and laboratory testing for the area-investigation phase. However, based on accumulated experience, the long-term program was revised in 1999 into a four-phase plan that includes: (1) a potential host-rock investigation and evaluation phase, (2) a detailed site-investigation phase, (3) a detailed design and licensing phase, and (4) a construction phase and operation/closure phase. As a result of this program, we expect identification of the spent-fuel disposal site by 2016 and commissioning of the repository by 2032.

30.7. CONCLUSION

As mentioned above in the introduction, Taiwan is a country with scarce natural energy resources, and by necessity the operation of existing nuclear power plants currently continues. The management of radioactive waste arising from using nuclear energy has to be carefully planned and credibly implemented. Final disposal for the LLW and spent fuel is considered the top priority to be implemented. This includes the development of relevant technologies for site investigation and safety evaluation, as well as site-development and construction activities. However, in recognition of the geological or economic benefits that an international/regional program can present, Taiwan's AEC would like to keep such an option open, under the condition that international safety standards are met and regulatory control at the recipient country is established.

Public support is one of the most important factors in successfully selecting a repository site. There are many aspects to gaining such public support. Past experience indicates that public participation during the decision-making process is essential. Openness to the public and sharing information with it is the starting point for enhancing public communication and gaining public confidence.

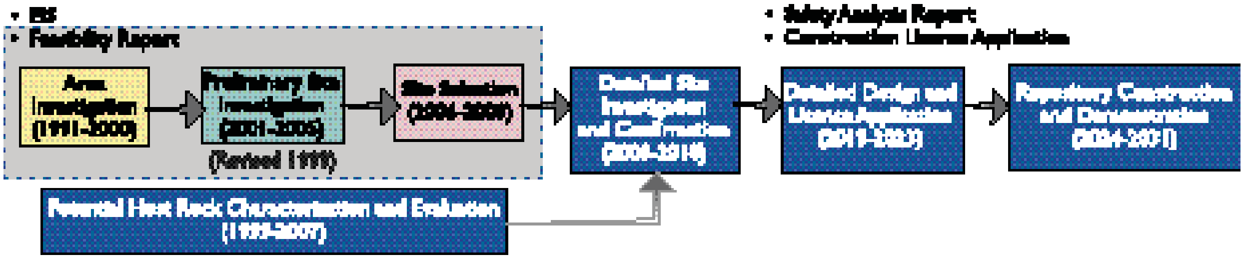

Figure 30.4. Schematic diagram of spent nuclear fuel disposal long-term plan

Chapter 31

Deep Geological Disposal of Radioactive Waste in Ukraine

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31.1. INTRODUCTION

Ukraine is among three countries possessing enormous quantities of radioactive waste. In Ukraine, these wastes have accumulated mainly because of the operation of five nuclear power plants (NPP) and the Chernobyl accident. As a result, Ukraine has been forced to develop a national program of radioactive waste isolation in geological formations. Research on radioactive waste isolation in underground geological formations was started in 1993, by an initiative of the State Committee on Nuclear Power Utilization (Goskomatom), the main waste producer.

In the period 1993–1996, the preparatory stage of R&D in an interdepartmental program for deep geological disposal was inaugurated. During this period, important decrees concerning radioactive waste management and deep geological disposal, in particular, were affirmed by governmental bodies, i.e., the Cabinet of Ministers and Verchovna Rada (Parliament) of Ukraine. The elaboration of these decrees was realized by ministries and state committees (Goskomatom, State Committee on Geology, Ministry of Health Protection, etc.) under the leading role of the Ministry of Environment Protection and Radiological Safety.

The main results of this preparatory stage were as follows:

1. Elaboration of a general concept
2. Development of a scientific methodology for R&D
3. Evaluation of the territory of Ukraine from the point of view of radioactive waste isolation
4. Selection of geological regions and formations potentially suitable for isolating radioactive waste
5. Initiation of regional studies aimed at site selection.

As a result of the regional studies, several preliminary candidate sites have been selected. At present, the coordination of the radioactive waste disposal problem has been delegated to the Ministry of Emergencies and Chernobyl Catastrophe Affairs.

31.2. PRESENT STATE OF THE PROBLEM

31.2.1. Spent-Fuel Production and Waste Handling

Nuclear power is the main source of radioactive waste in Ukraine. At present, there are five NPP's (including Chernobyl), comprising 14 units (including four at Chernobyl, whose closure was affirmed by a special State Decree in 2000). The total capacity for energy production is about 12,000 MW. The total volume of high-level radioactive waste (HLW) and intermediate long-lived waste (ILW) intended for disposal is made up of:

- Spent fuel stored in cooling ponds—3,000 tons (560 m^3)
- Operational waste at NPPs—12,000 tons
- Waste at the "Shelter"—34,000 m^3
- Total volume of high-level waste in temporary storage in the Chernobyl exclusion zone—223,000 m^3.

If a simple linear prognosis is used, the dynamics of NPP's spent-fuel production for future decades is as follows: 2010—8,200 tons, 2020—11,700 tons, 2030—15,200 tons. The existing storage facilities for spent fuel are now filled. During the period 2010–2013, waste from reprocessed spent fuel will be transferred back to Ukraine from Russia. The dynamics for receiving this waste are not yet defined.

The technologies for reprocessing long-lived radioactive waste and its vitrification for disposal are not developed in Ukraine. Containers for HLW transportation and their isolation in a geological repository do not yet exist. Thus, there are some uncertainties concerning radioactive waste management, namely:

- Status of spent fuel (waste or raw material)
- Amount of waste intended for underground disposal
- Management of waste in the "Shelter" facility and long-lived waste in the Chernobyl exclusion zone, etc.

The preparatory stage for a radioactive waste/deep geological disposal program has been set up. Subsequent steps in the program connected with final selection of a site and its characterization have been decided. This R&D effort will need significant financing, but in the late 1990s, the realization of the necessary funds for this program was delayed because of the economic state of Ukraine. In 1999, a new project for the disposal of HLW and ILW in geological formations (the deep-geological-disposal program) was formulated by the Ministry of Emergencies. Hence, the implementation of this program will depend upon the available financing.

31.2.2. Legislative and Regulatory Background

The legislative background for radioactive waste management in Ukraine is based upon the following state acts (decrees):

- "On radioactive waste management"
- "On nuclear power utilization and radiological safety"
- "On public health, sanitation, and disease control."

A regulatory basis for deep-geological-disposal issues in radioactive waste management is currently under development. The structure of the main documents being developed is based primarily on the published literature of the International Atomic Energy Agency (IAEA). Ukrainian regulatory documents containing issues related to deep geological disposal are as follows:

- HD 306.607.95, "Conditions for radioactive waste management prior to disposal. General issues."
- HD 306.698.95, "Containers for disposal of solid radioactive waste. Conditions for security of radiological safety" (prepared by Department of Radiological and Nuclear Safety, Ministry of Environment Protection)
- NRBU-97/D – 2000, "Regulations for radiological safety of Ukraine: Radiological protection from radiation sources."

In addition, there are two significant radioactive waste management programs containing special issues concerning deep geological disposal:

- State Program of radioactive waste management in Ukraine for 1998–2000 and up to 2005 (1995)
- Complex Program of radioactive waste management in Ukraine for 1999–2001 and up to 2005 (1999).

In accordance with a decision in the latter program, a program of ILW and disposal in geological repositories was compiled in 1999, but it has not yet been confirmed.

31.2.3. General Concept and Methodology

As a result of the deep-geological-disposal program, a preliminary stage in the elaboration of general concepts and scientific methodology for selecting and characterizing sites for radioactive waste/deep geological disposal has been developed. The general concept of deep geological disposal is based on the experience of advanced countries, IAEA basic principles, and technical criteria adapted to geological, socio-economic and ecological conditions in the Ukraine (Witherspoon, 1991; Khrushchov and Starodumov, 1996; IAEA, 1989; 1992). The principle of long-term (over 10,000 years) radioactive waste isolation is based on the idea of disposal as a geological engineering system that must satisfy a variety of conditions (final form of the radioactive waste, disposal in deep geological formations at an appreciable depth, special engineering barriers, etc.).

In developing a conceptual design for repository construction, world experience in underground methods and current projections was taken into consideration. The simplest and least expensive projections are preferable. According to this approach, the reference concept for the repository will consist of a wide transport tunnel and a system of galleries for waste emplacement (Krushchov and Starodumov, 1996). Different approaches to deposition are planned for different types of waste: operational, Chernobyl waste (including fuel with solid masses), and spent-fuel waste. The system of engineering barriers includes the matrix, buffer, containers, compactors, and bentonite backfill. Sometimes, a protective covering is needed on the cavern walls. Special nonblasting methods of excavation for maximum preservation of rock integrity have to be used. The construction of an underground research laboratory (URL) is planned as the first stage in developing the repository. Investigations in the URL are of a traditional nature, but the program may be shortened using results from actual world experience.

The scientific approach to site selection needs special consideration. A special methodology has been developed for the complex R&D needed in selecting and characterizing sites for deep geological disposal/radioactive waste. This methodology involves the following issues: scientific basis (philosophy) and basic principles, definition of main tasks, suitability of methods used in regional studies, site selection, and R&D connected with site characterization. The proposed methodology is based on basic principles and recommendations of IAEA, experience in advanced countries, and modern scientific achievements and approaches—all organized in a systems approach.

Using the multibarrier concept in the development of a radioactive waste repository, IAEAhas established a set of criteria for this approach that includes three groups: geological environment, repository, and waste. This set of criteria has been developed and refined, and is the basis for regional studies aimed at site selection and characterization (as well as the R&D connected with such activities). According to IAEA's general recommendations, the scheme for investigations and R&D, aimed at site selection and characterization, includes four stages: conceptual and planning, area surveys (including investigations and studies on a regional basis), site characterization, and site confirmation.

In accordance with this general scheme, our methodology of general regional investigations, aimed at evaluating and selecting regions and formations potentially suitable for radioactive waste disposal, is based upon a special theoretical background and criteria. The evaluation and selection of regions with suitable formations are carried out by estimating a geodynamic/energetic potential for a geologic section (i.e., set of formations) as well as the tectonic, structural, neotectonic, geomorphological, hydrogeological, climatic, and other geological and geophysical characteristics. This initial evaluation is completed using small-scale structural and lithological models considering other characteristics.

Regional studies aimed at site selection and R&D for characterization and confirmation of candidate sites are carried out in four stages:

1. Reconnaissance stage, with small-scale screening of a favorable geological formation spread over a significant region and its division into areas (conventionally corresponding to geological surveys at scales of 1:1,000,000–1: 500,000)
2. Middle-scale prospective studies of favorable areas (zones, groups of sites) selected within the limits of the region (corresponding to scales of 1:200,000–1:100,000)
3. Specialized characterization of candidate-sites, large scale and detailed (corresponding to scales of 1:50,000–1:25,000 and 1:10,000 and larger)—includes prospecting and preliminary exploration
4. Experimental studies of the geological environment and radioactive waste isolation conditions in the URL.

Stages 1 and 2 of this scheme include site-selection procedures. The whole procedure involving evaluation, comparison, ranking, and final selection of candidate-sites is based upon two main methodological tools: a set of criteria for selection (and one for exclusion) and a set of ecological-geological models. The ecological-geological criteria include: tectonic, structural, neotectonic, seismic, lithological, hydrogeological, geomechanical, geochemical, geomorphologic, hydrological, and climatic features, in addition to ecological conditions and prospectives for mineral deposits. The ecological-geological models include all of the same areas of investigation. Both static and dynamic aspects of these models are considered. The sets and content (as well as scales) of the models and criteria vary at different stages of the regional investigations and for different formations.

The tasks and scales of the models are defined for all stages of the regional studies and R&D. Methodologies for regional studies and R&D aimed at site characterization vary for different types of geological formations: crystalline, salt, and argillaceous. The methodology described above has been used for regional investigations and preliminary site selection in crystalline and salt formations.

31.3. RESULTS OF REGIONAL STUDIES

As a result of regional investigations, an evaluation of the territory of Ukraine and the selection of geological regions and formations that are potentially suitable for radioactive waste isolation have been completed (Khrushchov et al., 1993). From an evaluation and ranking of 12 geological regions in Ukraine, only two have been selected as suitable for radioactive waste disposal: (1) Ukrainian shield; (2) Dnieper-Donets depression (including Northwestern Donbass). Conditionally, the Volyn-Azov plate is under consideration. Hence, regional studies aimed at site selection were carried out in these two promising regions.

Using the methodology described above, the selection of promising zones (areas), subzones, and sites has been carried out. The following zones have been chosen:

- In the Ukrainian shield (crystalline formation), two zones: Korosten pluton and Middle Near-Dnieper area group of structures
- In the Dnieper-Donets depression (salt domes), three zones: northeastern and southeastern marginal zones, and the southeastern part of the depression
- In the Northwestern Donbass sector of the Dnieper-Donets Depression (bedded salt formations), three zones: Kamyshevakha, Kramatorsk-Chasovjar, and Kalmius-Torets synclines
- Argillaceous formations of sufficient thickness are spread over the southwestern slope of the East-European platform (Volyn-Azov plate, Cambrian, Oligocene), and in the Dnieper-Donets depression. Several zones have been chosen, but detailed investigations aimed at site selection were not carried out.

Within the limits of these zones, the following smaller units (i.e., subzones) have been selected:

- In the Korosten pluton, two subzones: Luginy massif composed primarily of granites, and Volodarsk-Volyn composed mainly of coarse grained anorthosites and gabbro-norite-anorthosites with granite intrusions
- Six areas in the Donbass region, within the limits of Kamyshevakha, Kramatorsk, Chasovjar and Kalmius-Torets synclines (zones). These areas are represented with bedded salt formations.

Within the limits of zones and subzones, several promising sites have been preliminarily selected:

- Malakhov and Doroginy sites (Luginy massif, granites), Novo-Borova and Zankovo sites (Volodarsk-Volyn massif, gabbro-anorthosite blocks), and the Veresnjany site in the northern part of Korosten pluton
- Eight salt domes within the limits of the Dnieper-Donets Depression.

A description of some promising geological formations is cited below.

31.3.1. Korosten Pluton

The Korosten pluton is a large (1,200 km^2) platform intrusive composed of anorthosites, gabbros, granites of rapakivi type, and other magmatic rocks surrounded with a complicated complex of metamorphic and magmatic rocks. Isotopic age of the rocks is 1,750–1,850 ma.

Porphyritic, even-grained granites and ovoid granites predominate over the granites of the pluton. They consist of orthoclase-perthite, quartz, oligoclase, biotite, hornblende, ilmenite, magnetite, zircon, fluor spar, sometimes fayalite, ferroaugite. Leucogranites, granophires, subalkalic granites and pegmatites are seldom found. Crystalline rocks of the pluton have high mechanic durability, and low porosity and permeability. The alternation of crystalline rocks is not significant.

The Luginy massif (see Figure 31.1) is located in the northern part of Korosten pluton, the Volodarsk-Volyn massif in its southern part. Analysis of mycrostructural-neonectonic conditions in the Korosten pluton shows that the Luginy massif has two parts: the Stepanov microblock in the northwestern sector and the Luginy morphostructural group of dislocations in the southwestern sector.

31.3.2. Malakhov Block

The Malakhov site (Institute of Geological Sciences, 1994) is located in the western part of the Luginy massive. The area of the block is about 120 km^2, elongated in the northwestern direction. According to geophysical data, the thickness of the block is about 3 km. The block is composed of biotite rapakivi-like granites, with small bodies of gabbro and anorthosites in the northern part and quartzites in the southern part. The block is confined from Ovrouch graben-syncline in the north by the deep Northern fault and from the southwest by the Central fault. The block is mostly homogenous, and faults are absent. Insignificant ruptures and fractures are

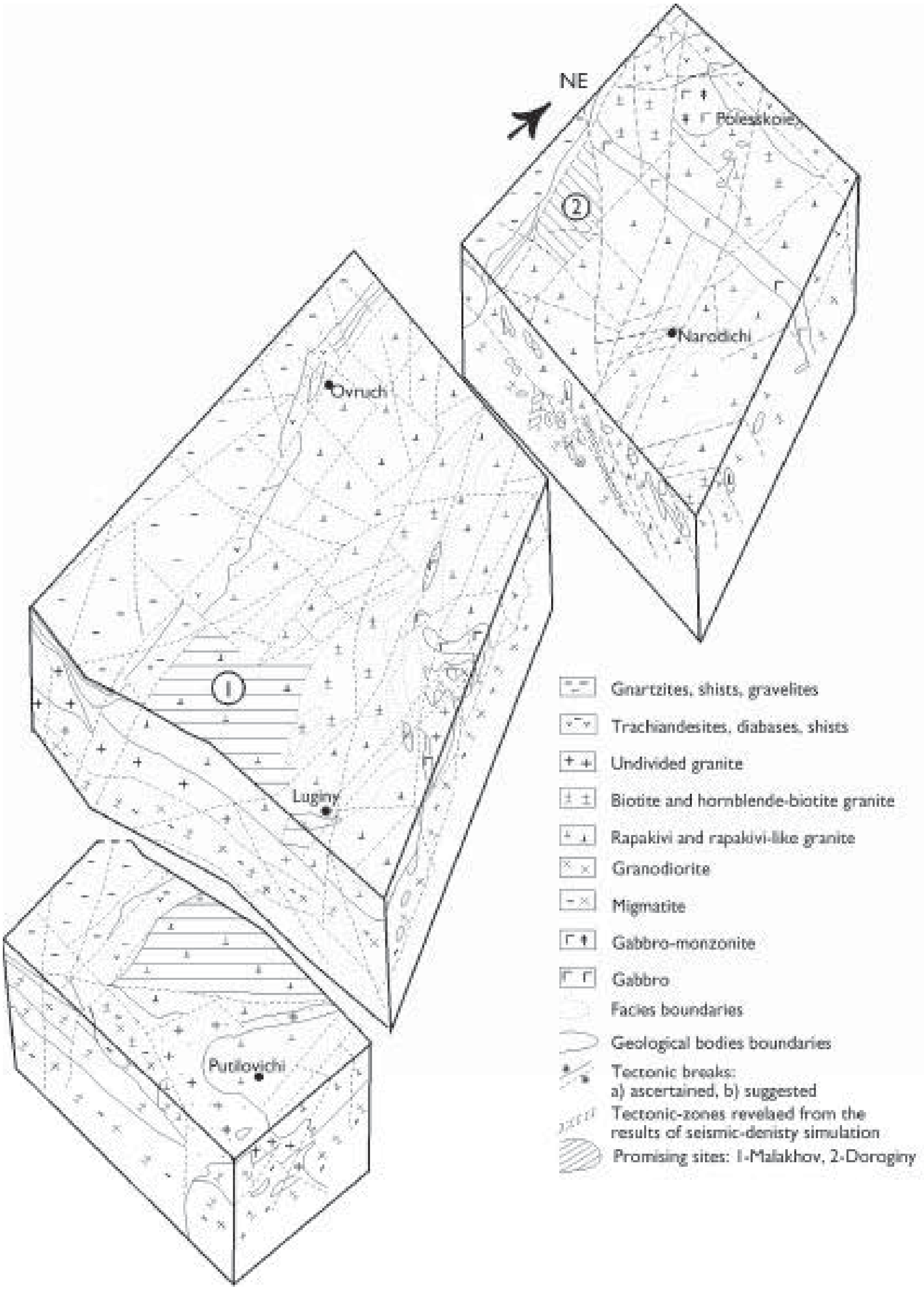

Figure 31.1. Block diagram of Luginy Massif

spread out at shallow depths. A crust of weathering is composed mostly of kaolin, up to 10 m thick, and a layer of disintegrated granites that range in thickness from 0.5 to 8 m. The block is located in the watershed of the Teterev and Liushonitsa rivers.

Aquifers in the sedimentary cover of the Malakhov block are Quaternary and locally include sands of the Poltava suite. Water supply from these aquifers is negligible; the capacity of wells is 4-5 m^3 per day. The chemical composition of the groundwater is mixed; total dissolved solids (TDS) range from 400 to 1,800 mg/L. Rocks in the disintegrated zone of weathered crust are also saturated. This zone, the weathered granites, and the saturated rocks of the sedimentary cover form a united, hydraulically connected aquifer.

Waters of the fractured crystalline bedrock are related to upper horizons in this zone. The fracturing extends downward to a depth of about 110 m, and its effectiveness decreases with depth. Depth to the water table ranges from 0 to 15 m. and more. Kaolin clays in the weathered crust form an impermeable layer for the fractured aquifer. From the standpoint of chemical composition, the underground waters are hydrocarbonate-calcium-sodium and hydrocarbonate-calcium-magnesium, with TDS of 400–1,200 mg/L.

The northwestern part of the Malakhov block, in the northwestern sector of the Stepanov neotectonic microblock, is more conducive to radioactive waste disposal. Meanwhile, in the western part (according to geophysical data), the manifestations of microfracturing zones are present. Structural, petrological, hydrogeological, thermophysical, and geotechnical schematic models have been elaborated. The location of the site in the neighborhood of the Chernobyl exclusion zone is an advantage, facilitating public acceptance.

31.3.3. Salt Dome N1

This site is a salt dome located in the southwestern marginal zone. The area of the salt body is about 14 km^2 (5.6 × 2.6 km). The salt is overlain by a zone of breccia (caprock) 300 m thick. The breccia is composed of debris from limestones, marls, argillites, aleurolites, gypsum, angydrites, diabase, basalt, and tuff cemented in a carbonate-clay matrix. Depth of the salt is 42.5–432.3 m. The salt mass contains blocks and debris from angydrites, limestones, argillites, and extrusive rocks. It is surrounded with rocks of the normal sedimentary series and marginal dissolution breccia. The thickness of the latter can extend to 300 m. In the uppermost part of the salt body, there are two beds of gypsum-anhydrite at depths 85–135 m and 210–300 m. Dimensions of the major debris and blocks in the salt mass range from 0.8 to 3.5 m. In cores from boreholes drilled deeply into parts of the salt dome, large blocks of undisturbed rock salt, interbedded with limestone, marl and anhydrite, have been encountered. The salt body is overlain with Paleogene, Cretaceous, and Jurassic sediments in marginal parts of the dome.

The Quaternary sediments are saturated. Water production ranges from 0.1–1 L/sec, and the waters are fresh, with TDS of 300–1,000 mg/L. The Paleogene aquifer is characterized with a productivity of 0.2 to 1 L/sec, and the waters are fresh, with TDS up to 1,000 mg/L. Near contacts with salt bodies, TDS increase up to 4–16.5 g/L. Waters of the Triassic aquifer are sodium-chlorite with high TDS (50–135 g/L). The carboniferous sediments are saturated (productivity up to 100–250 m^3/d), and the waters are high in TDS (up to 200 g/L). Neotectonic conditions indicate that the plug is being uplifted, but the rates are temperate. Structural, lithological, hydrogeological, neotectonic, and thermophysical schematic models have been developed to characterize the salt dome.

31.4. FUTURE R&D AND STRATEGIC PRIORITIES

The complete program of radioactive waste management (1999) includes two main stages of R&D related to the central program for the period 1999-2005:

- Selection of suitable areas and geological formations, 2000–2005
- Selection of candidate sites, 2003–2005.

These actions must be accompanied by development of a legislative and regulatory basis (1999–2005), concept for geological storage (2000), concept for an underground research laboratory (2005), methodological recommendations for R&D aimed at selection and characterization of sites, and a set of efficient technologies and technical measures related to radioactive waste management and repository construction (2000–2005).

In accordance with this program, a subordinate program for ILW and HLW isolation was developed in 1999. This subordinate program included three main stages:

1. Elaboration of a regulatory basis and completion of R&D related to selection and characterization of candidate sites (1999–2005)
2. Underground characterization of the selected site, experiments in the underground research laboratory, demonstration of site safety, licensing, and a state decision regarding repository construction (2005–2020)
3. Construction and use of the repository (after 2021).

According to this plan, the construction of the repository is scheduled for 2020–2025. But opposition has developed, primarily over existing agreements concerning the return of reprocessed spent fuel from Russia in 2010–2013, the prospects for a solution to the Chernobyl problem (especially the fuel and masses in the “Shelter”), and other arguments. As a result, the strategy of radioactive waste disposal is now being discussed in terms of three possibilities:

1. Construct interim surface storage and promote the deep geological disposal program (this does not seem realistic in the current economic situation).
2. Construct interim storage and delay the deep geological disposal program (this contradicts basic IAEAprinciples and jeopardizes the safety of future generations).
3. Place existing radioactive waste and reprocessed waste from Russia in temporary storage facilities at NPPs, centralize storage for low and intermediate waste in the Chernobyl zone, and develop the deep geological disposal program.

The last possibility is attractive from the standpoint of common sense, but its realization faces objections based on economic arguments. In view of this complicated situation, the search for other alternatives is continuing. The following possibilities from the deep geological disposal program are now being considered: (1) detailed study of reference sites in the vicinity of the Chernobyl zone and (2) a search for alternative sites and other possibilities/variants. The first of these possibilities is rather attractive from the point of view of public acceptance (the sites are located in the zone where the population has been excluded) and acceptable from the ecological standpoint. Its drawback is the high cost of underground construction in granite.

The proposal of a specialized investigation for radioactive waste disposal in reworked salt mines has met active public opposition. There are other alternatives—for example, studies and exploration of the Chernobyl exclusion zone for a repository location. This alternative is opposed however, by technical experts, based on the fact that a network of neotectonically active faults crosses this zone.

31.5. CONCLUSION

The fulfillment of the preparatory stage of the deep geological disposal program has resulted in the solution of the main initial tasks. However, no realistic steps have taken place during the preparatory stages of this program. Hence, Ukraine is still lagging considerably behind in the field of R&D, as compared to those countries that have been developing their programs over several decades. Meanwhile, there are results from the completion of the preparatory stage of the deep geological disposal program (i.e., preliminary selection of promising geological formations for subsequent target oriented studies), as well as existing legislation and a regulatory basis providing preconditions for program realization. Ukraine possesses the technological potential as well as the technical and scientific personnel capable of handling the tasks related to this program. Subsequent phases of the program are connected with more complicated R&D aimed at site exploration, which need significant investments. But the present economic situation in Ukraine has led to a delay in subsequent program development. The possibilities for program support will depend upon a significant increase in national funds and the organization of international cooperation. The latter could accelerate work on the deep geological disposal solution, while potentially decreasing costs.

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Chapter 32

The Long-Term Management of the United Kingdom's Radioactive Wastes: Current Status

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32.1. INTRODUCTION

The United Kingdom (UK) has significant 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 wastes still exist, of course. If we are to avoid passing this waste legacy on to future generations, the need for a long-term management solution remains. It is now no longer a foregone conclusion that the UK solution to this problem will involve geological disposal. The Department of Environment, Food, and Rural Affairs (DEFRA) has just launched a wide-ranging consultation to inform the development of national policy.

In consultation with others, Nirex has reviewed the events leading up to the RCF decision in the light of other countries' experience and recent academic research. The review shows that if a widely acceptable solution is to be found, there will need to be a widespread consensus on three key issues. These are: the *process* through which any solution is decided; the *structure* of the organizations charged with overseeing and implementing the solution; and the *behavior* of those organizations and the individuals within them.

32.2. BACKGROUND

In pursuit of a deep geological disposal facility for intermediate-level wastes (a high proportion of which are the equivalent of what many categorize as transuranic wastes) and certain low-level wastes unsuitable for near-surface disposal, an extensive program of geological investigations was focused on a site near Sellafield. Twenty-nine deep boreholes were drilled to investigate the properties of the geology around the site, and considerable modeling work was carried out to assess the suitability of the site for a repository. In 1992, a need was identified for a Rock Characterization Facility (RCF), an underground laboratory near Sellafield to investigate *in situ* the detailed properties of the potential host rock. Nirex applied for planning permission to build the RCF in June 1994 and the application was rejected by Cumbria County Council in December 1994. Nirex appealed against the decision, which resulted in a public inquiry that took place between October 1995 and February 1996.

The result of the public inquiry was that the rejection of planning permission for the RCF was upheld. The Secretary of State for the Environment announced on March 17, 1997, that he supported the decision not to allow the construction of the RCF, and consequently Nirex terminated its work at Sellafield.

The Secretary of State's findings are summarized in a report published by the Parliamentary Office of Science and Technology (Parliamentary Office of Science and

Technology, 1997). The Secretary of State noted that the poor design, layout, and arrangements for access of the proposed RCF, together with adverse impacts on visual amenities, a protected species (badgers), and the natural beauty of the English Lake District were serious enough to warrant refusal of the planning application. Further, he stated that he remained "concerned abut the scientific uncertainties and technical deficiencies" in the Nirex proposals, and that he was "concerned about the selection of the site and the broader issue of the scope and adequacy of the environmental statement."

32.2.1. Nirex's Strategy Following the Inquiry

Nirex' reaction to the decision was essentially defensive. Research at the Sellafield site was stopped and staff numbers were cut from 230 to 150, then to 88, and later to the current level of 67. Funding was reduced from around £50 million per year to £11 million per year. After the RCF decision, in the absence of any agreed-upon way forward with the British government on disposal of radioactive waste, Nirex had time to re-establish its priorities. It was recognized that, while a new consensus was needed on the way forward, the waste still existed, and deep geological disposal remained a potential long-term waste management solution. Nirex needed to continue providing packaging advice to waste producers (Nirex, 2000), making it essential that the valuable scientific research carried out so far was not lost. Consequently, research into the generic (i.e., non-site-specific) aspects of deep geological disposal continued. Nirex was also able to consider in depth the lessons learned from the RCF decision, as described in Section 32.4.

32.2.2. House of Lords Findings

Following the RCF Public Inquiry decision and the election of a new party into government in May 1997, the House of Lords' Select Committee on Science and Technology launched an enquiry into the management of nuclear waste, inviting evidence from a wide range of experts and stakeholders. Their report, *The Management of Nuclear Waste* (House of Lords, 1999), was published in March 1999. An extract of the executive summary of the report is presented below:

II. The bulk of nuclear waste that exists now and is certain to arise in the future originates from past military and civil nuclear programs. The problem exists and has to be solved. It could not be avoided by deciding today to discontinue nuclear power production or the reprocessing of spent fuel.

IV. The long time scales involved might be thought to be a reason for postponing decisions. The contrary is the case, since existing storage arrangements have a limited life and will require replacement and eventually the repackaging and transfer of stored waste. Reliance on supervision for very long periods increases the probability of human error.

V. We received a great deal of evidence on the technical issues and conclude that phased disposal in a deep repository is feasible and desirable….The phased approach which we recommend would allow decisions to be taken in a considered way as technical confidence and experience develop, and would avoid premature decisions which may be difficult to reverse.

VI. The future policy for nuclear waste management will require public acceptance…. Central to this is the need for widespread public consultation before a policy is settled by Government and presented to Parliament for endorsement.

VII. Present policy for nuclear waste management is fragmented. There are wastes for which no long-term management option has yet been decided and there are a number of significant materials, for which no use is foreseen, which are not categorized as waste at all. This leads to uncertainties in the planning of future facilities and to the continued storage of hazardous materials in an essentially temporary state.

VIII. These problems require changes in the present organizational structure for nuclear waste management.

32.2.3. Conclusions from National Consensus Conference

A "Consensus Conference" on radioactive waste management, held in London in May 1999, provided further input. Consensus Conferences are a new way of involving the public in the assessment of key issues of science and technology. Pioneered in Denmark, Consensus Conferences create a forum for a Citizens' Panel, made up of members of the public, to take part in an informed debate with expert witnesses of their choice.

The Panel, comprising fifteen citizens recruited from throughout Britain, came together in London to debate the issue of radioactive waste management, following two weekends of intensive preparation. At the end of the conference, the Panel produced a report (UK CEED, 1999) on its views as to the key issues for circulation to the Government, media, and other interested parties, thus opening up the debate in an area usually dominated by scientists and specialists. Most of the conclusions of the report were directly relevant to the possible development of a deep repository and are given in full below.

- Radioactive waste must be removed from the surface and stored underground, but must be monitorable and retrievable. Cost cannot be an issue. We must leave options open for future solutions.
- We recommend the appointment of a neutral body by the Government to deal with waste management, including the selection of a national storage site. The criteria for site selection should be open and publicized.
- All institutions handling radioactive waste should conform to the same high standards, which should include random scrutiny.
- Research and development must be continued on a much larger scale and international cooperation should be encouraged.
- We see no problem with privatization within the nuclear industry if done properly, with adequate safeguards.
- At present, there is a lack of trust and understanding, and public awareness must be raised. The public needs to be fully informed of the problems and solutions available. Decision-making must be open and transparent. Radioactive waste issues should be made part of the Government's education strategy.
- We are not fundamentally opposed to nuclear power, but it should not be expanded until a way is found to deal adequately with the waste problem.
- A new and internationally accepted method of waste classification is needed that clearly and openly communicates information about nuclear waste to the public as well as industry.
- Existing international reprocessing contracts should be honored, but no new ones should be taken up.
- Finally, while the industry has in the past had a well-deserved reputation for secrecy, we have in the course of the conference noted a welcome shift in culture and a new feeling of openness in dealing with these difficult issues.

32.3. LESSONS FROM THE RCF DECISION

Nirex has been reviewing the processes and actions that led up to the RCF decision to try to learn what contributed to the rejection and how things might be done differently in the future. Key parts of this review are examples provided by other countries' experience and recent academic research. The results of the review are far-reaching, but they can be summarized into three key issues: *process, structure, and behavior*. The following sections outline the findings in these areas in more detail. Section 32.4 describes some of the initiatives Nirex has undertaken to respond to them.

32.3.1. PROCESS

Discussions with a wide range of stakeholders have revealed that in the years leading up to the RCF Public Inquiry, decisions taken by Nirex were not transparent, and there was a lack of stakeholder involvement. In particular, Nirex used a closed process, the pace of which was driven by predetermined deadlines and not by the needs of stakeholders. To address these issues in the future, there must be a clear, phased decision-making process that:

- Has been developed in consultation with all stakeholders
- Has clear decision points
- Explains how decisions will be taken
- Provides opportunities for stakeholders to make inputs.

The whole process must be transparent and inclusive. The pace of progress—the speed at which the process moves from one phase to the next—should be determined by the time needed to obtain stakeholder inputs, not by a project plan. Only when there is sufficient consensus should the process move on to the next phase. The process should include "checks and balances," so that the behaviors (see below) of all the players can be analyzed and reviewed. This will require ongoing regulatory involvement, and a Commission may need to be set up to oversee implementation of the process. There should also be appropriate nuclear-industry involvement in formulating the process. Early in the process, there will need to be an independent review of technical options for the management of long-lived radioactive waste. This review should be open to input from all stakeholders and be based on dialogue and consultation. It is important to recognize that no part of the decision-making process can be considered in isolation: from the start, it is important to consider future steps and how current decisions may impact on them and vice-versa.

Whatever options are chosen for long-term waste management, a site-selection process will be required. This will need to be developed in consultation with stakeholders. Issues needing to be considered include the "contract" between the "UK" and the potential host community, which would define the right to volunteer for the project, community benefits, and any powers of veto.

In the belief that the UK can learn from other countries' experiences in this field, Nirex has been investigating how European Union (EU) directives on Environmental Impact Assessment (EIA) might be used as a mechanism for engaging stakeholders in dialogue.

32.3.2. Structure

Everyday experience demonstrates that the structure of an industry has a large impact on the way the industry needs to be regulated, public confidence in the organizations involved, the clarity of the issues that need to be addressed, and the ability of those in authority to make decisions. Radioactive waste management involves short-term decisions that may have very long-term impacts. It is essential that these short and long-term issues are visible, to the regulators and to others who have to make decisions. One way of achieving this visibility would be to maintain a separation between the short and long-term roles by having a separate organization focused on the long-term implications of radioactive waste management.

Work with focus groups (The Future Foundation, 2000; CSEC, 2001) has shown that the general public feels that the organization responsible for long-term waste management should be able to demonstrate its independence from all waste producers and from short-term political pressure; and gain respect by being technically competent, politically impartial, and resolute in considering the protection of future generations.

Given the EU directives on EIA, the organization that is to implement an option will need to demonstrate that it has sufficiently considered and evaluated other options to justify its solution. This gives the implementing organization the responsibility for looking at other options.

32.3.3. Behavior

In addition to a properly instituted process and structure, delivery of a long-term implementable solution for radioactive waste management would still be dependent on the behavior of those involved in the process. Research and experience (EPA [U.S.A.], 1995; Armour, 1996; O'Sullivan, 1999; Rabe, 1991) has shown that the behavior must be:

- *Open*—the debate must take place in the public domain and there should be free access to all the relevant information. Those involved should be open to influence from different people with different opinions and perspectives.
- *Transparent*—the reasoning behind actions, deliberations, and decisions should be made available. It must be clear from the outset how stakeholders and the wider public can be involved and how their opinions will be taken into account and used.
- *Accountable*—those responsible for the process should be accountable for their actions to all parties. This includes publicizing the reasoning behind decisions and giving people feedback on how their views have been taken into account.

32.4. NIREX RESPONSES TO "LESSONS LEARNED"

Nirex's responses to the "lessons learned" from the RCF decision are reflected in its current company objectives. These recognize the need to talk to stakeholders to understand the different perspectives.

Significant developments have been the publication of a Nirex Transparency Policy in 1999 (Nirex, 1999) and, in 2000, a formal internal inquiry into allegations of mismanagement made against the company before, during, and after the RCF Public Inquiry (Nirex, 2001a).

The Transparency Policy sets out the key elements of transparency:

1. Developing openness as a core value
2. Listening to people and taking their views into account
3. Making information readily available, in accessible formats

4. Making decisions in an open and traceable way
5. Giving people access to, and influence on, Nirex's future programs.

As part of the process of identifying and developing the behavior of the company, Nirex staff took part in a "Vision and Values" exercise. This identified the sort of conduct staff thought was important, what they believed Nirex should be aiming for, and the attributes the company should be developing. The issues identified included: openness, transparency, accountability, long-term focus, independence from political and short-term pressures, and competency. Nirex has been developing initiatives to address the issues outlined in the Transparency Policy and those identified through the Vision and Values work, as described below.

Developing Openness as a Core Value

To improve its interactions with stakeholders, Nirex has been developing its communication and consultation strategy. Openness within the company and between departments is also being actively promoted through regular company briefings and encouragement to employees to give briefings on their work to other members of staff.

Listening to People and Taking Their Views into Account

Nirex has been actively engaged with all stakeholders, including those who have been antagonistic in the past, and members of the general public. These meetings have taken several formats. To engage with the general public, Nirex commissioned a series of focus groups to identify people's issues and concerns about radioactive waste. Reports on the focus groups can be found on the Nirex website (The Future Foundation, 2000; CSEC, 2001).

Making Information Readily Available, in Accessible Formats

Nirex has a publication policy that commits Nirex to responding to a request for information within 20 working days. Nirex uses its website to publish its reports and has a bibliography that lists all the reports Nirex has produced.

Making Decisions in an Open and Traceable Way

To increase the transparency of the decision-making processes Nirex undertakes when giving packaging advice (Parliamentary Office of Science and Technology), Nirex has set up the Nirex Waste Management Advisory Committee (WMAC). This group consists of a review and advisory body drawn from senior Nirex staff and external members to consider any significant safety issues arising from Nirex's activities. These include issues relating to the waste packaging advice that Nirex gives to the industry and the suite of generic documents that describe the Nirex phased disposal concept.

In particular, the WMAC is expected to take a wider- and longer-term view of waste-packaging submissions, procedures, and changes to procedures—and will take account of regulatory issues, independent advice, and best practice from across the industry. The minutes of the meetings of the WMAC are sent to the nuclear safety and environmental regulators and to the waste producers.

Giving People Access to, and Influence on, Future Nirex Programs

Nirex has developed its concept for geological disposal to include a period of monitoring and the potential for retrieval, in line with the House of Lord's recommendations (House of Lords, 1999) and the results of the UK CEED Consensus Conference held in 1999 (UK CEED, 1999). It has also developed a Proposed Forward Program (Nirex, 2001b) that outlines the research planned in the next few years. By publishing the report and discussing it with people, Nirex hopes to encourage stakeholders to provide input into the development of the forward program and the research that is planned.

Nirex has also introduced a process of preview, designed to give stakeholders opportunities to access and influence Nirex's future work program. Preview can take various formats, including workshops, publication of work specifications on the Web, or one-to-one meetings. One of the main areas where preview has been undertaken recently has been on Monitoring and Retrievability. Two workshops were held with different groups of stakeholders to obtain their input into the work that Nirex should be undertaking in this area. Reports of the workshops (UK CEED, 2000; 2001) have been produced, and Nirex is integrating the feedback into its work program.

32.5. GOVERNMENT CONSULTATION

On September 12, 2001, the UK Government and the Devolved Administrations for Scotland, Wales, and Northern Ireland published a consultation paper entitled "Managing Radioactive Waste Safely: Proposals for developing a policy for managing solid radioactive waste in the UK," as a means of launching a national debate about the subject. The stated aim is to develop, and implement, a UK radioactive-waste-management program that inspires public support and confidence.

The paper sets out a proposed program for reaching decisions, which has five stages, as follows:

1. The current consultation on the proposed program; considering responses; planning the next stage: 2001–2002
2. Research and public debate to examine the different options and recommend the best option (or combination): 2002–2004
3. Further consultation seeking public views on the proposed option: 2005
4. Announcement on the chosen option, seeking public views on how this should be implemented: 2006
5. Legislation, if needed: 2007.

The paper notes, "The shape and speed of the program will depend on many factors, including public comments on this consultation paper. We must press ahead as quickly as we can. But we must also get the decisions right and ensure that the strategy wins public confidence."

32.6. CONCLUSIONS / SUMMARY

The management of long-lived radioactive waste is, in its essence, an ethical issue. The waste already exists, independent of whether nuclear power production stops or continues. This generation has benefited from the technologies that have produced the waste and should not leave it as a legacy for future generations. The solutions being developed to solve the problem will need to address social, ethical, political, and scientific issues, while also taking account of stakeholder views.

The keys to finding a long-term solution to radioactive waste disposal are:

- The process by which decisions are made
- The structure of the nuclear industry and the organizations involved in waste management
- The behavior of the parties involved in the process.

The short and long-term issues of waste management must be visible to the regulators and other decision-makers. This may be best achieved by keeping the short and long-term roles separate from each other.

Stakeholder involvement through all the stages of a clear and well-defined decision-making process will be essential if the UK is to find an implementable solution to the long-term management of its long-lived radioactive waste.

Those involved in waste management must be open and transparent in their dealings with other stakeholders and engage with stakeholders in all aspects of their work, so that a joint approach can be developed.

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Chapter 33

The Yucca Mountain Site Characterization Project for the United States

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33.1. INTRODUCTION

The United States high-level radioactive waste disposal program is investigating a site at Yucca Mountain, Nevada, to determine whether or not it is a suitable location for the development of a mined geologic repository. The U.S. has evaluated methods for the safe storage and disposal of radioactive waste for more than 40 years. The Nuclear Waste Policy Act of 1982 (NWPA, as amended in 1987) established the Office of Civilian Radioactive Waste Management within the United States Department of Energy (DOE) to develop and manage a Federal system for disposing of all spent nuclear fuel from commercial nuclear reactors and high-level radioactive waste resulting from atomic energy defense activities. The mission of the Yucca Mountain Site Characterization Office is to provide the basis for a national decision regarding the development of a repository for spent nuclear fuel and high-level waste at Yucca Mountain. The program is still in the site-characterization process; the DOE has a commitment to a site-suitability evaluation process that is objective, unbiased, and based on sound science.

The repository will be designed primarily for the emplacement of commercial spent fuel. The United States civilian nuclear power generation programs do not now involve reprocessing of spent fuel, and only minor amounts were ever reprocessed as part of a demonstration. However, the repository will also be designed to accept wastes, spent fuel, and excess fissile material from U.S. defense programs and previous reprocessing activities in the United States. As with other sites, geological disposal at Yucca Mountain is predicated on the expectation that the geologic setting and associated natural barriers will work in combination with the engineered barriers to limit radionuclide release, enhance the resiliency of the repository, and increase confidence in its performance.

As the DOE moves toward a possible site-recommendation decision, the DOE is continuing to evaluate uncertainties in performance-assessment models. Such uncertainties result from the long time frames over which performance must be forecast; the natural variability in features and processes at the site; inherent limitations on the amount of data that could be collected; and complexities in the processes studied—most notably the interrelated thermal, hydrologic, chemical, and mechanical processes that would occur in the underground emplacement drifts and the surrounding rock. A few questions remain regarding each of these factors, but these questions are understood and work to address them is underway. The DOE's scientific and technical understanding of the Yucca Mountain site will continue to evolve and improve as the DOE proceeds through completion of site-characterization activities, the site-recommendation process, and any license application process. The repository design is evolving to take better advantage of the conditions at Yucca Mountain. Current thinking underlying the design is that the natural system provides the setting to develop and protect the engineered systems.

33.2. REGULATORY REQUIREMENTS

The current United States Nuclear Regulatory Commission (NRC) requirements for a geologic repository, described in 10 CFR part 60 (U.S. NRC, 1997), embody a phased approach to construction and emplacement of high-level wastes in the repository.

Following successful completion of site characterization and recommendation of the site by the Secretary of Energy to the President, if it is found suitable, the DOE initially would submit documentation for a license hearing to authorize construction of the repository. Authorization of construction would be based in part on understanding of the long-term performance of the proposed repository, including an understanding of the importance of the site's physical attributes. After sufficient construction to affirm that the site conditions and underground excavation response are within the limits specified in the license to construct, the DOE would submit documentation for a hearing for a license to receive and possess waste. Amendment of the license to allow the authorization to receive and possess waste marks the first point in time that waste could be emplaced for the repository. After completing emplacement and operation of the repository, including specific monitoring, an application would be submitted for an amendment to decommission and terminally close the repository.

Taking advantage of advancements in computational techniques since the time their original regulation (10 CFR part 60) was promulgated, the NRC has issued for public comment a new regulation, 10 CFR part 63 (U.S. NRC, 1999). This regulation argues against the need for the specific subsystem performance objectives that are present in the original regulation. Instead, approaches based on an integrated safety assessment and a total system performance assessment (TSPA) are used for preclosure (operational) time and post-closure time, respectively.

The promulgation of proposed 10 CFR part 63 marks a significant change in the regulatory approach to compliance demonstration for a high-level waste repository in the United States. Whereas 10 CFR part 60 proscriptively set requirements for subsystem components and identified generic additional design requirements without regard to specific features of the repository system concept under consideration, proposed 10 CFR part 63 allows the applicant (in this case, the DOE) discretion to provide the technical basis for either inclusion or exclusion of specific features, events, and processes of the geological setting in the performance assessment. Specific features, events, and processes of the geological setting would only be evaluated in detail if the magnitude and timing of the resulting expected annual dose would be significantly changed by their omission. Similarly, the applicant is to provide the technical basis for inclusion or exclusion of degradation, deterioration, or alteration processes of the engineered barriers in the performance assessment, including those processes that would adversely affect the performance of the natural barriers. These processes would only be evaluated in detail if the magnitude and time of the resulting expected annual dose would be significantly changed by their omission. Generally, the applicant is required to identify those design features of the engineered barrier system, and the natural features of the geologic setting, that are considered barriers important to waste isolation. For these features, the applicant is to provide the technical basis for the rationale for inclusion and describe the capability of the barriers to protect public health and safety.

33.3. SOURCES OF MATERIALS CONSIDERED FOR DISPOSAL

By statute, the DOE is responsible for the safe, permanent disposal of spent nuclear fuel from commercial nuclear power plants and smaller quantities of spent nuclear fuel from weapons production reactors, research reactors, and naval reactors. The United States has generated about 40,000 MTHM of spent nuclear fuel from commercial nuclear power plants. This amount could more than double by 2035 if all currently operating plants complete their initial 40-year license period. By 2035, the United States will also have about 2,500 MTHM of spent nuclear fuel from research reactors, naval reactors, reactor prototypes, and reactors that produced nuclear weapons materials. The majority of this spent nuclear fuel is stored at DOE sites in Idaho, South Carolina, and Washington.

The DOE must also dispose of large quantities of DOE-owned high-level radioactive waste from the production of nuclear weapons. About 380 million liters (100 million gallons) of liquid waste from nuclear weapons production programs are stored in underground tanks at the same DOE sites. This high-level radioactive waste will be vitrified in stainless steel canisters. All the waste forms transported to and received at a repository would be solid materials. No liquid waste forms would be accepted for disposal.

A small portion of the material that would be disposed in a repository would come from surplus plutonium resulting from the production and decommissioning of nuclear weapons. A nominal 50 metric tons of surplus plutonium must be safely dispositioned. Current plans call for some of the surplus plutonium to be combined with uranium to form fuel that would be used in commercial reactors. The resulting spent nuclear fuel would be disposed as commercial spent nuclear fuel. Some of the surplus plutonium would be immobilized in ceramic, placed inside stainless steel cans, which in turn are placed in canisters. These canisters will be filled with molten high-level radioactive waste glass.

33.4. SITE CHARACTERIZATION DATA AND ANALYSES

The investigation programs at the Yucca Mountain site have been described in a Site Characterization Plan (U.S. DOE, 1988), which was required by the Nuclear Waste Policy Act. The logic supporting the definition of the investigation programs was developed from strategies to obtain the information needed to address regulatory requirements, primarily those of the NRC at 10 CFR part 60. The investigations address both pre-closure (operational performance) and post-closure (long-term performance) considerations. The pre-closure requirements include worker and public radiation protection; foundations for facilities; access construction, including both shafts and ramps; excavation stability, and the effects of seismicity and thermal loading associated with waste emplacement, including requirements to maintain the capability for retrievability for nearly 100 years; rock handling; and underground closure functions.

For post-closure, the requirements relate to long-term performance as assessed by TSPA. Generally, the test programs address the movement of very small amounts of water in the unsaturated zone rocks, thermal-mechanical-hydrologic-chemical coupled processes, and the corrosion behavior of highly resistant stainless steels that will be used for waste packages.

Yucca Mountain is unique among potential repository sites being considered worldwide, in that the repository is planned to be developed in unsaturated rock, well above the groundwater table. However, like all potential sites, the role of water is important at Yucca Mountain. The most important factors affecting performance of the repository system at Yucca Mountain appear to be limited seepage of water into the emplacement drifts; solubility limits of dissolved radionuclides in Yucca Mountain water; dilution of radionuclide concentrations in the geologic setting; retardation of radionuclide migration in the unsaturated zone; retardation of radionuclide migration in the saturated zone; performance of the waste package; and performance of the drip shield. There are concerns associated with limited seepage of water into the emplacement drifts; retardation of radionuclide migration in the unsaturated zone; performance of the waste package; and performance of the drip shield. Resolution strategies for the most important issues facing the Yucca Mountain Project are related to the determination of the amount of water that could seep into an emplacement drift.

During the site-characterization program, the DOE has performed extensive surface-based tests and investigations, underground tests, laboratory studies, and modeling activities designed to provide the technical information necessary for the evaluation of long-term repository performance. The site-characterization program has evolved in response to advancements in scientific understanding, proposed changes in regulatory requirements, and changes in program requirements, such as changes in design requirements for the potential repository. Computer models of the hydrologic, geochemical, thermal, and mechanical processes that would operate in the repository system over time have been developed from the data collected. These process models have been used to develop an overall TSPA model that evaluates how the potential repository may behave for 10,000 years. Analyses have also considered alternative conceptual models to assess the extent to which the results of performance assessment depend on the details of the underlying process models.

The potential repository would be located in volcanic tuff that was deposited between approximately 11 and 14 million years ago. The characteristics of the volcanic rock have been studied by geologic mapping of the surface, in boreholes, and in underground excavations. Mapping and other studies show that faults are present in the vicinity of Yucca Mountain. The location, timing, and amount of movement on these faults have been characterized as part of a seismic hazard analysis. The underground location for a potential repository was identified using several factors, including the thickness of overlying rock and soil, the characteristics of the rock that would host the repository, the location of faults, and the depth to groundwater. A repository at Yucca Mountain would be sited deep underground to prevent waste from being exposed to the environment and to discourage human intrusion.

33.5. THE VIABILITY ASSESSMENT

In appropriation legislation for 1997, Congress required the DOE to prepare a Viability Assessment of the Yucca Mountain site. The Viability Assessment (U.S. DOE, 1998) of the Yucca Mountain site described the following: (1) a preliminary design concept for the critical elements of a repository and waste package; (2) a TSPA, based on the design concept and the scientific data and analyses available at that time, that describes the probable behavior of a repository in the Yucca Mountain geologic setting; (3) a plan and cost estimate for the remaining work required to complete and submit a license application to the NRC; and (4) an estimate of the costs to construct and operate a repository in accordance with the design concept.

Yucca Mountain is located on federal land adjacent to the Nevada Test Site in Nye County, Nevada, about 160 km (100 mi) northwest of Las Vegas. The mountain consists of a series of ridges extending 40 km (25 mi) from Timber Mountain in the north to the Amargosa Desert in the south. The water table at Yucca Mountain is approximately 500 to 800 m (1,600 to 2,600 ft) below the surface of the mountain at the potential repository location. The zone of soil or rock below the ground surface and above the water table is called the unsaturated zone. The potential repository would be located in the unsaturated zone, about 200 to 500 m (660 to 1,600 ft) below the surface and, on average, about 300 m (1,000 ft) above the water table. The deep water table and thick unsaturated zone at Yucca Mountain result from the low infiltration rate of surface water due to low annual rainfall and high rates of evaporation and transpiration (the process by which water vapor passes from soil into plants, then into the air).

The Viability Assessment concluded that the Yucca Mountain site remains a promising site for a repository and that work should proceed to support a decision in 2001 on whether to recommend the site to the President for development as a repository. It also summarized the additional work that needed to be done to complete site characterization, to continue to evolve the design, and to make a decision on whether or not to recommend the site for approval.

33.6. REPOSITORY DESIGN

The DOE has developed a design for a monitored geologic repository at Yucca Mountain that could give future generations the choice of either closing and sealing the repository as early as allowable under proposed regulations, or keeping it open and monitoring it for a longer time period. The design for the proposed repository would not preclude the option for future generations to make societal decisions to monitor the repository for up to 300 years before making decisions to decommission and close the facility.

Figure 33.1 is a conceptual illustration of the proposed facilities and structures that would make up a potential repository at Yucca Mountain. It shows the facilities as they would appear after construction. In general, the operations that would be performed include receiving spent nuclear fuel and high-level radioactive waste in NRC-certified shipping casks from rail and truck transporters; unloading, handling, and packaging spent nuclear fuel and high-level radioactive waste into waste packages suitable for underground emplacement; transporting waste packages from the surface to the underground facility; emplacing waste packages in underground drifts; decommissioning and closure; and monitoring operations and repository system performance to ensure the safety of workers and the public.

The design that has been developed will maintain the flexibility to adapt to various construction and operational conditions and requirements. Four key aspects of design flexibility are the ability of the repository design to support a range of construction approaches (e.g., change in emplacement drift spacing, modular or sequential construction of surface and subsurface facilities); the capability to dispose of a wide range of radioactive waste container sizes; the ability to support a range of thermal operating modes (e.g., defining a larger waste emplacement area or varying ventilation duration and rates); and the ability to continue to enhance the design to best achieve performance-related benefits identified through ongoing analyses.

The general repository design concept provides flexibility for operation over a range of thermal operating modes. This range is being examined to identify the potential performance benefits of different environmental conditions (lower temperatures and associated humidity conditions) in the emplacement drifts. The temperatures at the drift wall and waste package surfaces can be varied, along with the relative humidity, by modifying operational parameters such as the thermal

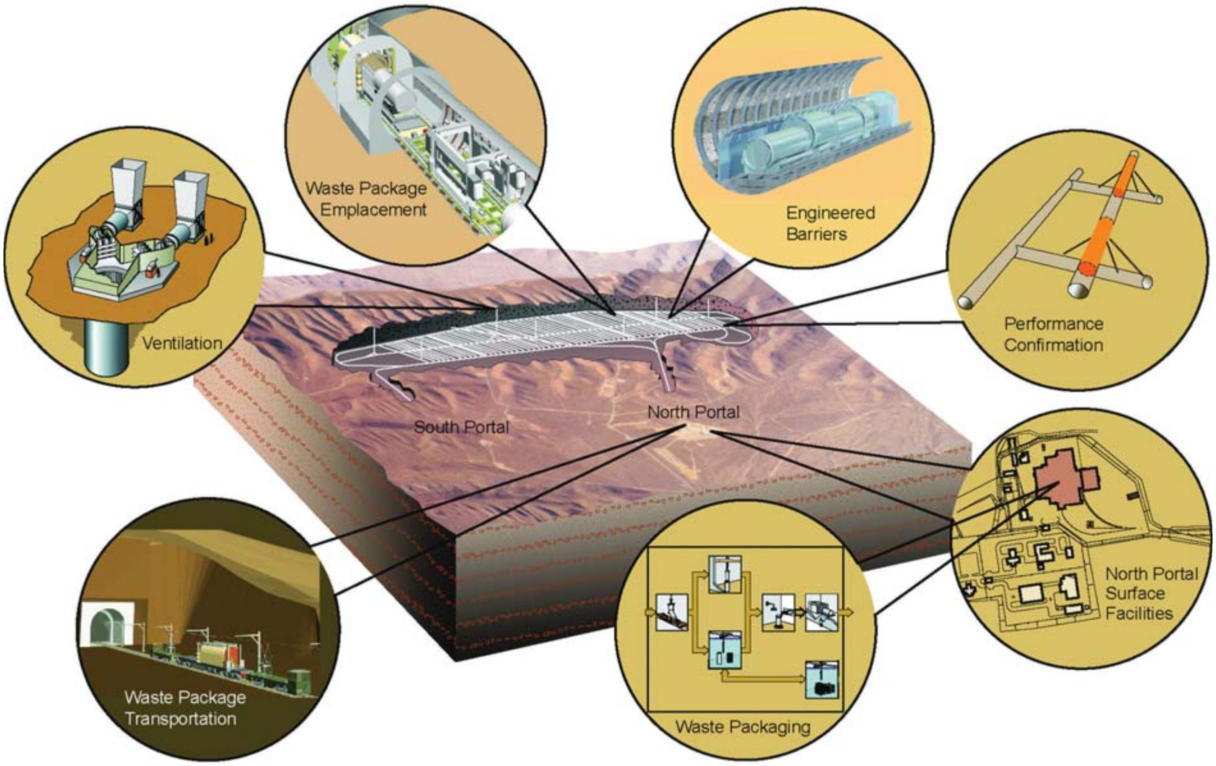

Figure 33.1. Conceptual illustration of the proposed repository facilities

output of the waste packages, the spacing of waste packages in emplacement drifts, and the duration and rate of ventilation (e.g., active ventilation, using fans or passive ventilation that relies on natural air flow). Higher temperatures (i.e., above the boiling point of water) would dry out the emplacement drifts, thereby limiting the amount of water available to contact waste packages. Lower temperatures might have less effect on rock properties and geochemistry, thereby reducing the complexities in modeling thermal effects. This, in turn, may reduce uncertainties in assessments of future repository performance. Lower in-drift temperatures may also reduce the potential for waste package corrosion. Analyses are continuing that will be used to evaluate the effects of lower temperatures (below the boiling point of water) on the in-drift environment and repository performance.

Waste packages would be moved, one at a time, from the surface to the emplacement drifts by way of a connecting rail system. Equipment in the Waste Handling Building would place a waste package into a shielded transporter. Two electric locomotives, one on each end of the transporter, would move the transporter down the North Ramp, through the repository's main access drift, to an emplacement drift. Once the transporter arrives at the assigned emplacement drift and the drift's isolation doors are opened, the transporter's shielded doors would be opened and the waste package would be moved out of the transporter using a retractable deck. An in-drift gantry would lift the waste package and its supporting pallet off the deck and deposit them in their designated position inside the drift. Before the repository is permanently closed, overlapping and interlocking drip shields would be placed over the waste packages to divert any water that might drip from the top of the emplacement drifts.

Emplacement operations would take place in finished emplacement drifts at the same time as future emplacement drifts are being constructed. During construction, separate ventilation systems operating on the development side and the waste emplacement side would allow separate regulation of airflow to accommodate different needs. During emplacement, ventilation would maintain temperatures within the range for equipment operation. Before closure, ventilation would remove most of the heat generated by the waste packages and keep the relative humidity low.

33.7. NATURAL BARRIERS

The barriers important to waste isolation, broadly characterized as natural barriers, are associated with the geologic and hydrologic setting as well as engineered barriers. The natural barriers at Yucca Mountain include surface soils and topography; unsaturated rock layers above the repository; unsaturated rock layers below the repository; and the volcanic tuff and alluvial deposits below the water table. The engineered barriers are designed specifically to complement the natural system in prolonging radionuclide isolation within the repository and limiting their potential release. They are discussed in the following section.

Natural barriers would contribute to waste isolation by (1) limiting the amount of water entering emplacement drifts, and (2) limiting the transport of radionuclides through the natural system. In addition, the natural system would provide an environment that would contribute to the long lives of the waste packages and drip shields.

The identification of a subsurface location for the potential repository was based on several factors that take advantage of the natural barriers, including the thickness of overlying rock and soil, the extent and geomechanical characteristics of the host rock, the location of faults, and the depth to groundwater. The host rock for a potential repository should be able to sustain the excavation of stable openings that can be maintained during repository operations, and that will isolate the waste for an extended period after closure. In addition, the rock should be able to absorb any heat generated by the waste without undergoing changes that could threaten the site's ability to safely isolate the waste. The host rock should be of sufficient thickness and lateral extent to construct a repository large enough to support the design's intended disposal capacity. Moreover, the amount of suitable host rock should provide adequate flexibility in selecting the depth, configuration, and location of the repository.

The distribution and characteristics of fractures at Yucca Mountain are important because in many of the hydrogeologic units, particularly the welded tuffs, fractures are the dominant pathways for water flow in both the unsaturated and saturated zones. By controlling where, and at what rates, water is likely to flow under various conditions, the fracture systems play a major role in the performance of the repository. The potential repository has been designed to take advantage of the free-draining nature of the repository host rock, which would promote the flow of water past the emplaced waste and limit the amount of water available to contact the waste packages.

Fractures are common in the Topopah Spring Tuff, the rock unit hosting the potential repository. These fractures provide the main pathways for water to flow through the rock unit. The water table below the potential repository is located within the rock units called the Calico Hills Formation and the Crater Flat Group. These units are less fractured; water movement is likely slowed as it moves through these units. Another important feature of the tuffs of the Calico Hills Formation is the abundance of zeolite minerals in the rock matrix and fractures. Zeolites are silicate minerals that have the ability to sorb many types of radionuclides and other ions that might be transported in the water.

33.8. ENGINEERED BARRIERS

The components of the engineered barrier system are designed to complement the natural barriers in isolating waste from the environment. The engineered barriers would contribute to waste isolation by (1) using long-lived waste packages and drip shields to keep water away from the waste forms, and (2) limiting release of radionuclides from the engineered barriers through components engineered for optimum performance in the expected environment.

Waste packages would have a dual-metal design containing two concentric cylinders. Figure 33.2 depicts waste packages within an emplacement drift. The inner cylinder would be made of Stainless Steel Type 316NG with a thickness of 5 cm (2 inches). The outer cylinder would be made of a corrosion-resistant, nickel-based alloy (Alloy 22) with a thickness ranging from 2.0 to 2.5 cm (0.8 to 1.0 inch). Alloy 22 would protect the stainless steel inner cylinder from corrosion, and Stainless Steel Type 316NG would provide structural support for the thinner Alloy 22 cylinder. Laboratory tests and analyses indicate that Alloy 22 would last more than 10,000 years in the range of expected repository environments at Yucca Mountain. Corrosion tests are con-

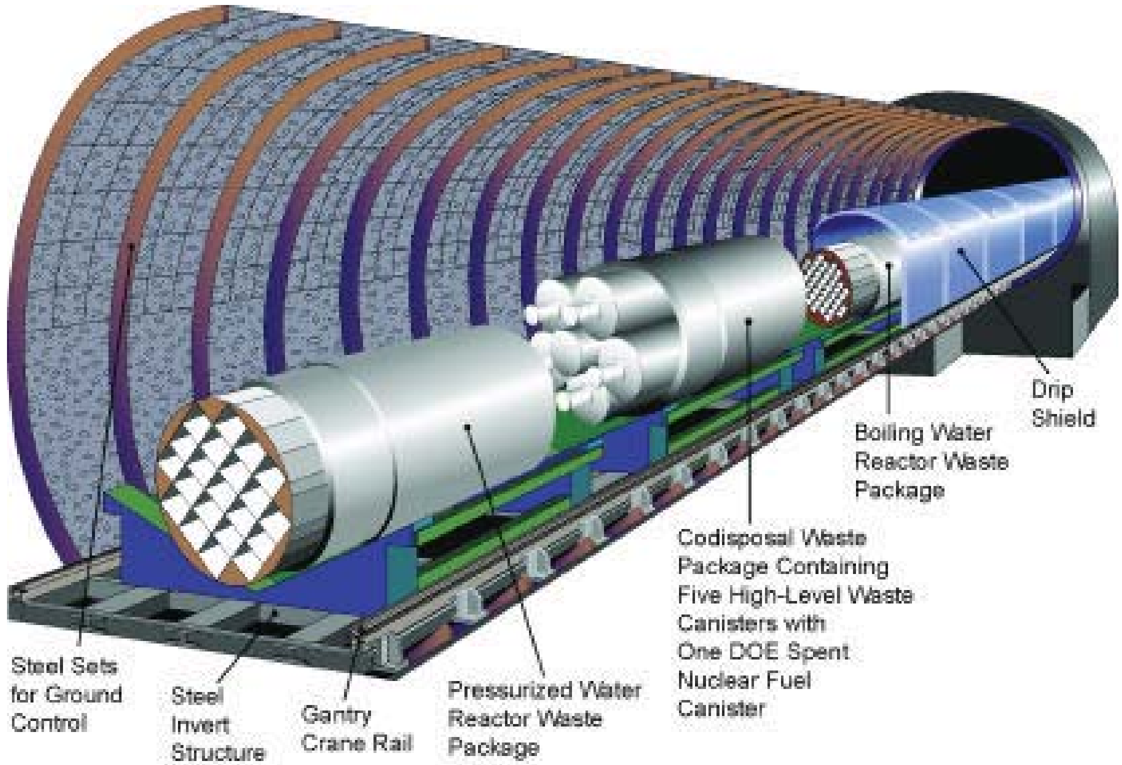

Figure 33.2. Schematic illustration of the emplacement drift, with cutaway views of different waste packages

tinuing in a variety of thermal and chemical environments to provide additional information on the corrosion rate of Alloy 22 and to address uncertainties.

Each waste package would have outer and inner lids at each end of the cylinder. The outer (closure) lids would be made of Alloy 22 with a thickness of 2.5 cm (1.0 inch). The inner lids would be made of Stainless Steel Type 316NG with a thickness between 6.5 cm (2.6 inches) and 13 cm (5 inches), depending on the waste package design. The loading end of the waste package has a third flat closure lid made of Alloy 22, which would be placed between the inner lid of stainless steel and the outer lid of Alloy 22. The flat closure lid provides an extra barrier against a potential release caused by cracks and corrosion in the closure weld areas. The basic waste package design is the same for all the waste forms. However, the sizes and internal configurations vary to accommodate the different waste forms. Figure 33.2 illustrates several common internal designs, including two for different types of commercial spent nuclear fuel and one for high-level radioactive waste and DOE spent fuel (a co-disposal package).

Drip shields would be installed over the waste packages prior to repository closure. The drip shields would divert any moisture that might drip from the drift walls, as well as condensed water vapor, around the waste packages to the drift floor. All the drip shields would be the same size, so one design could be used with all the waste packages. The drip shields would be made of titanium, which would provide corrosion resistance and structural strength. They are designed to divert moisture around waste packages for thousands of years. Tests continue on drip-shield materials to assess how well current data and models can be extrapolated over long periods of time. The drip shields would maintain their function in the event of rockfalls.

The invert includes the structures and materials that support the pallet and waste package, the drift rail system, and the drip shield. It is composed of two parts: the steel invert structure and the ballast (or fill) that consists of granular material. Following closure, one function of the granular material in the invert would be to provide a layer of material below the waste packages that would slow the movement of radionuclides into the host rock. Water is not expected to accumulate and flow beneath the drip shields, so the most likely way radionuclides could move is by diffusion (where particles migrate slowly from zones of high concentration to zones of low concentration) through thin films of water on the granular material.

33.9. PROCESSES IMPORTANT TO PERFORMANCE OF A REPOSITORY AT YUCCA MOUNTAIN

The processes important to the repository's performance after it is closed include those that control the movement of water through the mountain. These processes begin with precipitation, as rain and snow, at the surface, a fraction of which infiltrates into the mountain. This net infiltration would move through the unsaturated zone to the level of the potential repository, then downward from the repository level (still in the unsaturated zone) to the saturated zone. Within the saturated zone, water would move laterally away, where it could eventually reach a location where a receptor resides. At the repository level, water moving past the engineered barriers would be affected by the physical and chemical processes associated with heat from the emplaced waste, and could corrode waste packages and dissolve some of the waste. These processes could lead to the movement of radionuclides out of the repository. Figure 33.3 illustrates some of the processes that were considered and modeled for the TSPA. Certain disruptive events that could affect these processes are also considered in the following discussion. The treatment of uncertainties associated with these processes is addressed in a later section.

33.9.1. Unsaturated Zone Flow

Because of the present-day arid climate at Yucca Mountain and the surface processes of runoff, evaporation, and transpiration, the amount of water available to

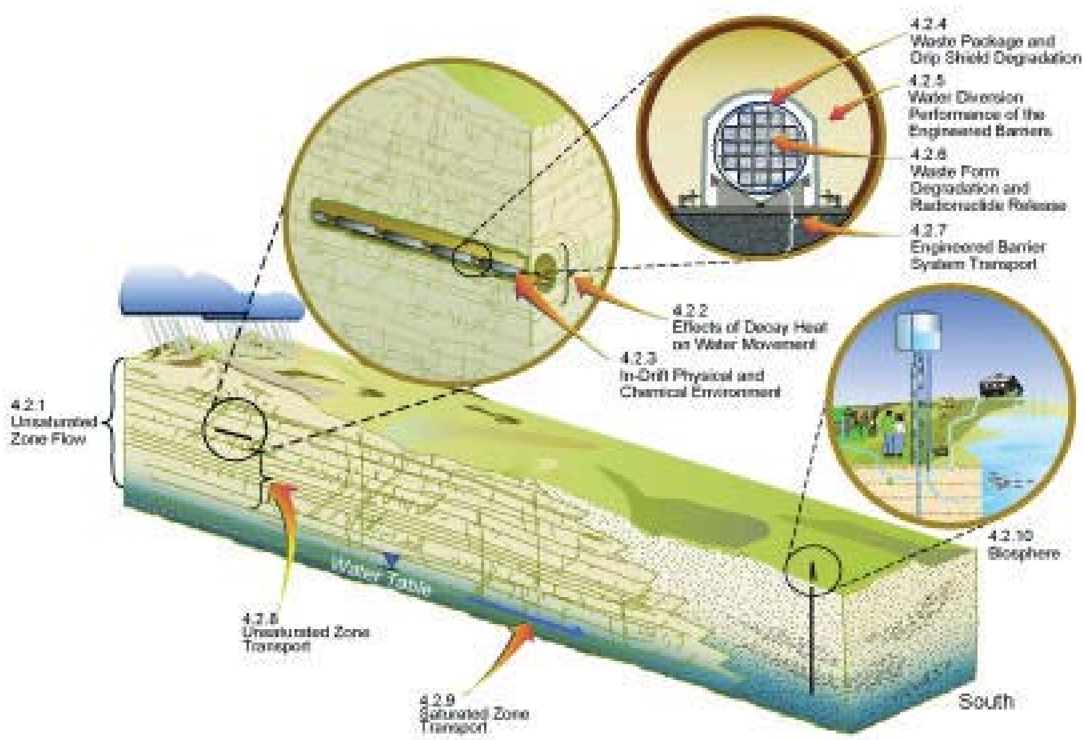

Figure 33.3. Schematic illustration of the processes modeled for Total System Performance Assessment

contact and transport radionuclides is expected to be small. However, the availability of water may increase as a result of wetter climates that may occur in the future. One potential operating mode of the current design uses the heat produced by the waste to effectively limit the potential for contact between water and waste packages for hundreds to thousands of years. It keeps most of the pillar area between the emplacement drifts below the boiling point of water to promote water drainage through the cooler portions of the rock pillars. In addition, the repository design takes advantage of natural processes that would divert water around drift openings. Most water moving in the unsaturated zone flows through fractures. Water flowing in narrow fractures will usually remain in them rather than flow into large openings, such as drifts, because of capillary pressure in the fractures. Thus, capillary forces and water flow in unsaturated zone fractures would limit seepage into openings, allowing most water to move past (not into) the emplacement drifts. If water does seep into an emplacement drift, most of it may flow down the drift wall to the floor and drain without contacting either the drip shields or the waste packages.

However, for modeling repository performance, a more conservative approach is used. The DOE is continuing to evaluate the extent to which unsaturated zone flow processes would impede the movement of water into emplacement drifts. A range of operating modes that would restrict temperatures in the rock surrounding the emplacement drifts to below the boiling point of water is also being examined. Regardless of the operating mode, capillary forces and water flow in unsaturated zone fractures would limit seepage into openings, allowing most water to move past (not into) the emplacement drifts. The ongoing studies are designed to provide additional information about water movement in faults and fractures, along with the rates and potential locations of seepage.

33.9.2. Effects of Heat on Water Movement

In higher-temperature operating modes, no liquid water can remain in the emplacement drifts, and very little can remain in the nearby rock, as long as the drift wall remains at temperatures above the boiling point of water. Even after hundreds to a few thousand years, when the waste packages have cooled below the boiling point of water, their continued, but reduced heat production will still cause evaporation in and near emplacement drifts, thereby limiting the amount of water in the rock near the waste packages. The heat from emplaced waste may change the flow properties of the rock, as well as the chemical composition of the water and minerals in the engineered barrier system and surrounding rock. The nature and extent of these effects, however, would depend on thermal loading, ventilation rates and durations, and thermal operating properties.

Because the repository would be ventilated during operations, the major effects of heat on water movement would take place during the post-closure period. Thermal-hydrologic processes in the repository environment will determine the conditions in the drift, including temperature, relative humidity, and seepage at the drift wall. In the lower-temperature operating modes that the DOE is evaluating, the heat from emplaced waste would still increase evaporation rates and, to some extent, dry out the rock near the emplacement drifts. Although liquid water could enter drifts, it is likely that only limited amounts of water would be available because of fracture flow and processes related to seepage. Ongoing evaluations are assessing the extent to which lower temperatures might limit adverse changes in rock properties and water chemistry, possibly reducing the complexities in modeling thermal effects and the uncertainties in assessments of future repository performance.

33.9.3. Physical and Chemical Environment

The lifetimes of the drip shields and waste packages would depend on the conditions to which they are exposed: the in-drift physical and chemical environment. Once water enters a degraded waste package, the transport of radionuclides released from the waste form inside also depends on the drift environment. For higher-temperature operating modes, the repository environment would be warm, with temperatures at the surface of the waste package initially increasing above the boiling point of water. The expected duration of temperatures above the boiling point of water on the surfaces of the waste packages would be hundreds to thousands of years. The precise time period varies for three main reasons: (1) location within the repository layout, (2) spatial variation in the infiltration of water at the ground surface, and (3) variability in the heat output of individual waste packages. The repository edges will cool first because they lose heat to the cooler rock outside the perimeter. Water percolating downward through the host rock in response to infiltration at the ground surface will hasten cooling of the repository; locations with greater percolation cool faster. The heat output of individual waste packages will vary, depending on the type and age of the waste they contain; however, this variability can be managed. Cooler operating temperatures have the potential to limit complex temperature-related couplings of physical processes, thus enhancing confidence in projections of performance.

The chemical environment is expected to be at near-neutral pH (mildly acidic to mildly alkaline) and mildly oxidizing. Under such conditions, Alloy 22 is expected to form a thin, stable oxide layer that is extremely corrosion resistant. Important processes affecting the chemical environment include the evaporation and condensation of water, the formation of salts, and the effects of gas composition. While the repository environment is warm, relative humidity will probably control the water chemistry.

33.9.4. Waste Package and Drip Shield Degradation

The lifetimes of the drip shields and waste packages will depend on the environment to which they are exposed and the degradation processes that occur. Corrosion is the most important degradation process considered in selecting the materials for the waste package and drip shield. A number of corrosion processes have been investigated in detail. The results have been used to support the selection of materials and the design of components.

Because most corrosion occurs only in the presence of water, and because highly corrosive chemical conditions are not expected in the repository environment, both the titanium drip shield and the Alloy 22 outer layer of the waste package are expected to have long lifetimes. Analyses based on laboratory tests and other evaluations indicate that, in the absence of a disruptive event or human intrusion, no waste packages are expected to be breached by corrosion or any other mechanism for more than 10,000 years.

Analyses to date indicate that the drip shields and waste packages will be long lived. Current models based on higher-temperature operating modes indicate that significant corrosion is not likely for thousands of years. Nevertheless, the DOE is evaluating whether keeping waste-package surface temperatures cooler would improve performance or reduce the uncertainty in the models used. Water could contact waste packages sooner in lower-temperature operating modes. However, lower-temperature operating modes would likely reduce the potential for corrosion susceptibility of Alloy 22. The DOE continues to perform materials tests and evaluate corrosion data to provide a stronger technical basis for the projections of waste-package and drip-shield lifetimes.

33.9.5. Water-Diversion Performance of the Engineered Barriers

Models of the movement of water within emplacement drifts focus on the process of seepage. For water to contact a drip shield or waste package, water droplets must form from seepage and fall to the surface of the drip shield. Other modes of flow, such as film flow on the drift wall, may divert some or all of the seepage around the drift. Seepage would probably occur at or near faults or fractures that could focus percolation into a drift. Analyses indicate that the emplacement drifts have enough drainage capacity to ensure that water does not rise above the level of the invert even under extreme seepage conditions.

The effectiveness of the engineered barriers in diverting water depends on the long lifetimes of the drip shields and waste packages. As discussed previously, lower-temperature operating modes may improve confidence that the barriers would function as expected over the life of the repository by reducing the uncertainty related to waste-package and drip-shield degradation. It is expected that the flow processes for lower-temperature operating modes would be nearly identical to those of higher temperature modes, except that water would begin to flow sooner because of the below-boiling temperatures in the rock.

33.9.6. Waste-Form Degradation and Radionuclide Release

Because of the characteristics of the natural system and the engineered barriers, the DOE does not expect water to contact waste for over 10,000 years. Even if water were to penetrate a breached waste package before 10,000 years, several characteristics of the waste form and the repository would limit radionuclide releases.

First, because of the high temperatures of the waste, much of the water that penetrates the waste package will evaporate before it can dissolve or transport radionuclides. Both spent nuclear fuel and glass high-level radioactive waste forms will dissolve slowly in the waste package environment. Data and analyses indicate that most of the radionuclides in the waste are not very soluble in the warm, near-neutral pH conditions that are expected. To dissolve radionuclides that may be soluble (technetium-99, iodine-129, neptunium-237, and all isotopes of uranium), water must also penetrate the metal cladding of the spent nuclear fuel assemblies. Although the performance of the cladding as a barrier may vary because of possible degradation, it is expected to limit contact between water and the waste.

Release of radionuclides from the waste forms is a three-step process, requiring: (1) degradation of the waste forms, (2) mobilization of the radionuclides from the degraded waste forms, and (3) transport of the radionuclides away from the waste forms. Radionuclides can be released only after the waste package is breached and air and water begin to enter. The rates of water flow and evaporation will determine when and if water accumulates in a waste package. Even if water is available to dissolve radionuclides, the chemistry inside the waste package influences the rate at which the waste forms degrade and the mobility of radionuclides from the degraded waste forms. The subsequent transport processes could occur only after temperatures have cooled below boiling, thereby allowing water to be available as a transport mechanism.

33.9.7. Engineered Barrier System Transport

The invert below the waste package would contain crushed tuff that would limit the transport of radionuclides from breached waste packages into the unsaturated zone. Transport could occur either through advection, which is the flow of liquid water, or by diffusion. The scarcity of water makes advective transport unlikely, but diffusive transport through thin films on the waste form, on the waste package, and in the invert ballast is possible.

33.9.8. Unsaturated Zone Transport

Eventually, components of the repository's engineered barrier system will degrade, and small amounts of water will contact the waste. Analyses indicate that this initial degradation is not likely to occur within the first 10,000 years following repository closure, but degradation will gradually increase over tens of thousands of years. Even then, however, features of the site geology and the repository will limit radionuclide migration to the environment and slow that migration by hundreds to thousands of years. Processes that could be important to the movement of radionuclides include sorption, matrix diffusion, dispersion, and dilution.

Rocks in the unsaturated zone at Yucca Mountain contain minerals that can adsorb many types of radionuclides. Matrix diffusion would increase both the time it takes for radionuclides to move out of the repository and the likelihood that they would be exposed to sorbing minerals. Dispersion also would result in lower concentrations of contaminants. Dilution occurs naturally as contaminated groundwater flows and mixes with non-contaminated groundwater, which reduces the concentration of contaminants.

Process models of transport in the unsaturated zone have incorporated the processes described above. The results of these models indicate that the movement of radionuclides from a breached waste package down to the water table would require hundreds to thousands of years, depending on the mobility of specific radionuclides. Many radionuclides would not move over much longer time spans because of their particular chemical properties.

33.9.9. Saturated Zone Flow and Transport

The same basic processes that apply to the movement of radionuclides in the unsaturated zone (sorption, matrix diffusion, dispersion, and dilution) also apply to transport in the saturated zone. Flowing groundwater transports radionuclides either in solution (dissolved) or in suspension (bound to very small particles called colloids). Any radionuclides released by water contacting breached waste packages would have to migrate through the unsaturated zone down to the water table and then travel through the saturated zone to reach a location where a receptor resides. Groundwater in the saturated zone under the potential repository's location generally moves southeast before flowing south out of the volcanic rocks and into the thick alluvium deposits of the Amargosa Desert. Analyses show that it would take thousands of years or longer (depending on the mobility of specific radionuclides) for radionuclides to move down through the unsaturated zone below the potential repository, into the saturated zone, and then to a likely receptor location.

33.9.10. Biosphere

Biosphere analyses of Yucca Mountain have been performed to develop conversion factors that enable analysts to estimate doses to a receptor from the transport and retention of radionuclides within the biosphere. The biosphere analyses scrutinize processes and pathways that could either disperse or concentrate radionuclides released from the repository. In calculating radiation exposure, biosphere analyses consider the environment around and the lifestyle (including diet and activity) of a potential receptor. Current biosphere analyses consider a potential receptor living in Amargosa Valley with a lifestyle similar to that of the current residents.

33.9.11. Disruptive Processes and Events Scenarios

Analyses of future repository performance must also consider events that could occur in the future that have the potential to compromise the repository's ability to protect public health and safety. Analysts have evaluated a wide variety of potentially disruptive processes and events that could affect performance. These range from extremely unlikely events to changes in processes that, although low in probability, could affect long-term repository performance. The potential for igneous (volcanic) activity in or near the repository has been specifically included in performance assessments in the disruptive-scenario case, and the effects of seismic activity (i.e., vibratory ground motion that might damage the cladding on spent nuclear fuel) have been considered in the nominal scenario case. The possibility of accidental human intrusion into the repository has been analyzed in a separate performance assessment in accordance with proposed regulations. The results of these analyses indicate that no potentially disruptive processes or events are likely to compromise the repository's performance.

33.10. UNCERTAINTIES IN DATA AND MODELS

Quantitative assessments of the long-term performance of the potential repository consider a comprehensive set of features, events, and processes that may have an effect on that performance. The features and characteristics of the site are incorporated into conceptual and numerical models. The likelihood of occurrence and consequences of processes and events that may affect repository performance are evaluated, then incorporated, as appropriate, into the numerical models. Although the DOE continues to evaluate ways to reduce uncertainties in repository performance models, uncertainties will always remain because of the long time frames over which the performance of a repository must be assessed, the natural variability in features and processes at the site, and limitations on the amount of data that can be collected. Features, events, and processes are generally represented probabilistically in a performance assessment to address this inherent uncertainty and variability.

Numerous analyses have been and are being performed, using existing data and validated models, to help the DOE understand the extent to which the results of performance assessments are robust. The DOE is currently conducting a study to assess the degree of realism in current process models, to quantify key uncertainties, and to improve the understanding of conservatism in the models and in performance assessment results. The DOE is also conducting studies to evaluate whether uncertainties (especially modeling uncertainties that are not easily quantified) can be reduced further by operating the repository at lower temperatures.

The DOE has adopted an approach that relies on multiple lines of evidence to build confidence in analyses of repository performance and assurance that the repository will meet applicable post-closure performance standards. Quantitative analyses performed to date indicate that the repository would perform well and offer both defense in depth and a significant safety margin. The DOE recognizes that independent lines of evidence may be strengthened by additional analysis. For example, ongoing evaluations of lower-temperature operating modes could increase confidence in projecting waste package corrosion behavior. The degree of safety margin (the difference between expected performance and the proposed regulatory limit) and the independence of the measures that provide defense in depth are being assessed, both qualitatively and quantitatively. A variety of analogue studies have provided several lines of evidence that suggest current models are representative and, in some cases, may be conservative. The DOE is continuing to study analogues that may provide important additional information about the reliability of the performance-assessment models. Performance confirmation activities will monitor the behavior of the repos-

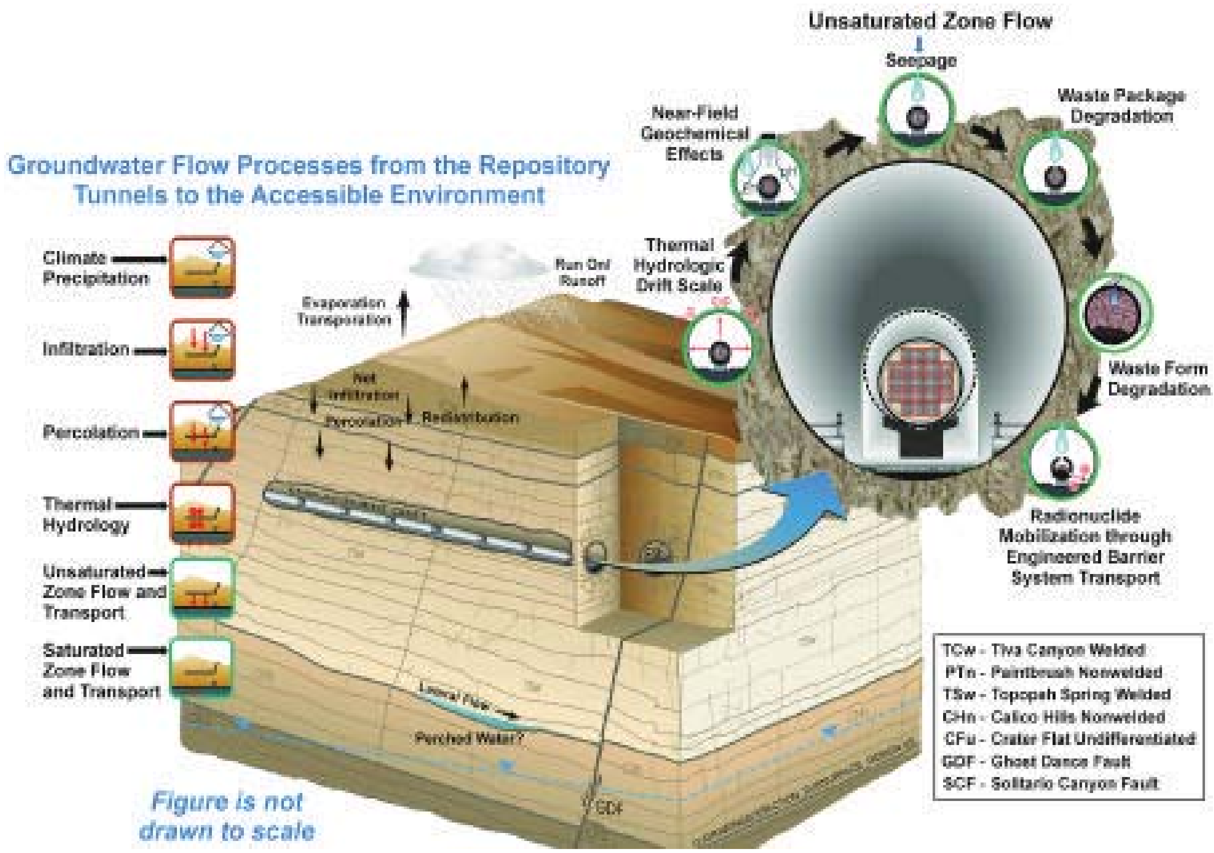

Figure 33.4. Conceptual illustration of the flow processes that affect performance at the Yucca Mountain site

itory and enable future managers to continuously assess the technical bases for decisions about the repository.

Collectively, these multiple lines of evidence are known as the post-closure safety case. Elements of the safety case include quantitative assessments of long-term performance; selection of a site and design of a repository that provides defense in depth—a system of multiple, independent, and redundant barriers designed to ensure that failure of one barrier does not result in failure of the entire system. The system would also provide a margin of safety against radionuclide releases and a margin of safety compared to health and safety requirements; qualitative insights gained from the study of natural and man-made analogues to the repository or to processes that may affect repository performance; and long-term management to ensure the integrity and security of the repository, as well as long-term monitoring (a performance confirmation program) to monitor the behavior of the repository and enable sound scientific and engineering bases for a later repository closure decision.

33.11. PERFORMANCE CONFIRMATION AND MONITORING

A performance-confirmation program established to monitor and confirm that the repository is performing as expected has been initiated during site characterization. The focus of the performance-confirmation program is to gather and analyze data on conditions and systems that will affect performance of the facility after closure and to evaluate their impacts on post-closure performance. Subsurface facilities, including performance-confirmation drifts and alcoves, would be constructed to facilitate monitoring of the emplacement environment and the performance of the engineered barrier system. Primary testing and monitoring activities will include seepage monitoring to evaluate flow of water into excavations and confirm expected waste package environ-

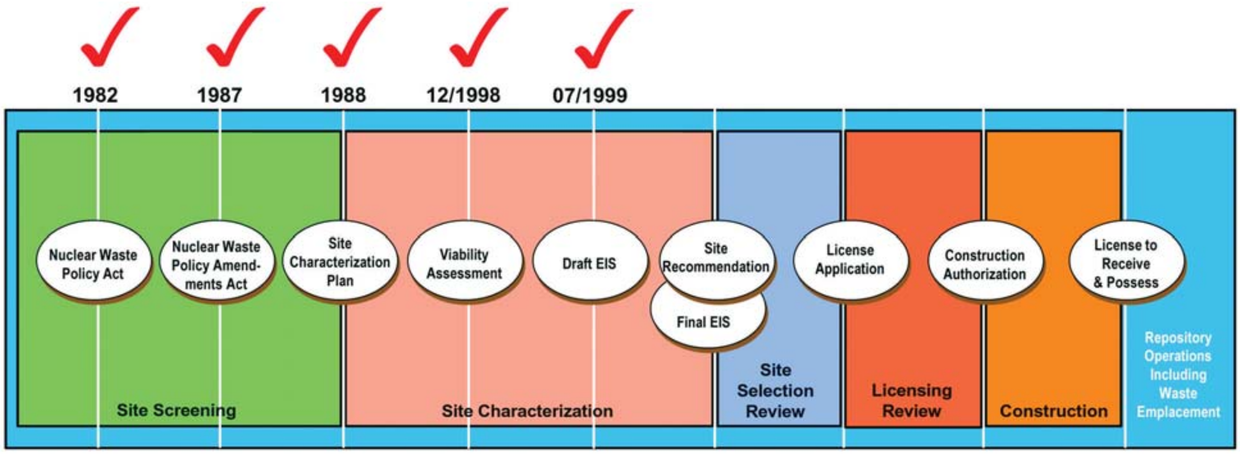

Figure 33.5. Reference schedule of important milestones for Yucca Mountain Project

ment; *in situ* waste package monitoring to verify cladding temperatures; rock-mass monitoring to confirm the conceptual understandings and numerical simulations of coupled processes considered in performance assessments; and other activities that focus on providing an increased understanding of processes most important to repository post-closure safety. Performance confirmation and monitoring activities would continue throughout the pre-closure period, which could be extended up to 300 years.

33.12. CONCLUSIONS

Geological disposal at Yucca Mountain is predicated on the expectation that the geological setting and associated natural barriers will work in combination with the engineered barriers to limit radionuclide release, enhance the resiliency of the repository, and increase confidence in its performance. The Yucca Mountain site setting comprises multiple natural features that are favorable for long-term performance as well as for the design of stable engineered barriers (Figure 33.4).

The important factors affecting performance of the repository system at Yucca Mountain appear to be: limited seepage of water into the emplacement drifts; solubility limits of dissolved radionuclides in Yucca Mountain water; dilution of radionuclide concentrations in the geologic setting; retardation of radionuclide migration in the unsaturated zone; retardation of radionuclide migration in the saturated zone; performance of the waste package; and performance of the drip shield. The most important technical issues facing the Yucca Mountain Project are related to the determination of the amount of water that could seep into an emplacement drift, corrosion processes in very long-lived stainless steels, and projecting performance far into the future.

The DOE intends to begin issuing documents related to recommending the site for development as a repository presently. In so doing, the DOE will open the public-comment period on the Secretary's consideration of the possible recommendation of the Yucca Mountain site as a potential repository for spent nuclear fuel and high-level radioactive waste (Figure 33.5).

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Chapter 34

An Update on the Geological Disposal of Radioactive Waste at the Waste Isolation Pilot Plant in Southeastern New Mexico, U.S.A.

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34.1. INTRODUCTION

The Waste Isolation Pilot Plant (WIPP), located in southeastern New Mexico, is currently being used by the U.S. Department of Energy (DOE) for the deep geologic disposal of defense-related transuranic (TRU) waste. TRU waste by definition contains alpha-particle-emitting radionuclides with atomic numbers greater than uranium (92) and with a half-life greater than 20 years, in concentrations greater than 100 nanocuries per gram of waste, excepting high-level radioactive waste. Two types of TRU wastes will be disposed of in WIPP: mixed wastes that contain radionuclides (along with hazardous substances such as PCBs, solvents, and lead) and nonmixed waste that contain only radioactive substances. Most of the TRU waste to be disposed of at WIPPis contaminated sludge and refuse, including rags, tools, protective clothing, and equipment. The bulk of these items are from activities associated with the production of nuclear weapons, including plutonium fabrication and reprocessing, research and development, decontamination and decommissioning, and environmental restoration programs at various sites.

Geologically, the WIPP repository is located in the Delaware Basin of southeastern New Mexico and consists of a labyrinth of rooms and tunnels excavated in a bedded salt formation of Permian age about 655 m below the land surface. The allowed 176,000 m^3 of waste will be mostly packaged in 55-gallon (208 liter) drums and standard waste boxes (each having a capacity of about 1.88 m^3), surrounded by a magnesium oxide (MgO) backfill. Formal characterization activities at WIPPbegan in 1975, and extensive site characterization data were gathered throughout the 1980s and early 1990s. Repository design, performance assessment, and site characterization/repository performance confirmation testing have been developed and designed to meet specific regulatory commitments/requirements, and continues to this date.

Radionuclide disposal at WIPP is regulated by the Environmental Protection Agency (EPA) to meet requirements in radioactive waste disposal regulations 40 Code of Federal Regulation Parts 191 Subparts B and C (40 CFR 191) and 40 CFR 194 and the WIPP Land Withdrawal Act. WIPP is further regulated by the New Mexico Environment Department (NMED) to meet the requirements of the Resource Conservation and Recovery Act (RCRA) under the Hazardous Waste Facility Permit for mixed wastes. Radioactive waste disposal began at WIPP on March 26, 1999. Re-certification of the facility must take place, by regulation, every five years on or before the anniversary date of first disposal, over the 35-year life of the project.

About 224 shipments (as of May 31, 2001) of radioactive waste have been received and disposed of at the WIPP site from various locations throughout the United States, at the recent average of 5–6 shipments each week. The DOE Carlsbad Field Office and its contractors are developing the technology and regulatory documentation required to increase the number and types of shipments to be received at WIPP, to meet increasing demands for disposal from DOE sites throughout the complex.

In this paper, the regulatory framework will be further defined, the current waste transportation system will be discussed, the design of the repository will be more completely described, and the potential future uses/enhancements to the repository will be presented.

34.2. THE WIPP LAND WITHDRAWAL ACT

In 1992, Congress passed the WIPP Land Withdrawal Act (LWA). The title of this crucial legislation underscores that Congress "withdrew" from public use the area devoted to the site, transferring jurisdiction of the site from the Department of Interior to the DOE. The Act also established an array of regulatory conditions and standards covering everything from limits on the kinds and quantities of waste the DOE could place in the repository to transportation safety. The LWA set requirements for oversight and regulation of the WIPP by federal and state agencies, for publication of information and documents, and for provision of economic assistance to the State of New Mexico. It also established the Environmental Protection Agency (EPA) as the WIPP's primary regulator.

The Act limited the waste sent to WIPP to DOE's defense-related waste. It also prohibited the disposal of high-level radioactive waste and spent nuclear fuel. In 1996, Congress amended the LWA, deleting requirements that the WIPP obtain a "no-migration" variance from the Environmental Protection Agency. This meant that the DOE would not need to submit a lengthy application showing why it should be exempt from land disposal restrictions under the Resource Conservation and Recovery Act. The rationale was that WIPPis not a shallow landfill of the kind typically used for waste containing toxic chemicals or metals, and that the requirements imposed on radioactive waste disposal and containment are conservatively adequate to also contain any hazardous constituents of waste disposed of at the WIPP.

In addition, the 1996 amendments confirmed a 1993 decision by the Secretary of Energy to cancel tests using radioactive waste at the WIPP. Instead, national laboratories conducted this research in existing laboratories, rather than underground at the WIPP.

The LWA is a landmark in the legal history of the site. It serves as a concise record of the essential steps required to establish the WIPP, the major institutions involved, and the basic requirements for disposal and decommissioning activities.

34.3. CERTIFICATION BY THE EPA

Since 1992, the EPA has been the WIPP's primary regulator, with responsibility for evaluating and verifying that the WIPPwill safely isolate TRU waste and protect human health or the environment. To carry out this responsibility, the EPA issued regulatory standards for waste containment during handling and after disposal (40 CFR 191).

Then, to determine whether the WIPP would meet these containment standards, the EPA formulated a set of WIPP-specific criteria (40 CFR 194) that required the DOE to provide certain kinds of information and set forth what the DOE must do to show that the WIPP would meet the containment standards.

These standards addressed several crucial aspects of the WIPP and the waste that would be placed in it:

- The longevity and potential dangers of TRU waste require any permanent disposal facility to be highly reliable. The nation's responsibility toward future citizens, who have no say in decisions made before their time, means that containment standards must be particularly rigorous.
- The WIPP is the world's first deep geologic disposal site designed specifically for TRU wastes, and it is one of a very small number of permanent repositories in salt beds for any type of waste. People have had no opportunities to observe such a site for more than a few decades. Therefore, EPA regulation could not be based upon actual measured performance over the short term. Instead, the DOE was required to perform research, probabilistic simulations of future performance, and independent reviews to demonstrate that the WIPP would satisfy containment standards.
- The EPA and the DOE must be very confident that the facility will perform as expected, because removing wastes from the salt bed becomes more difficult and costly as time passes.

In late 1996, the DOE submitted its WIPP Compliance Certification Application (CCA) to the EPA, representing the results of decades of research, review, and public comment. The EPA evaluated whether the application demonstrated that the WIPP could comply with the stringent containment requirements for TRU waste. On May 18, 1998, the EPA certified that the repository system would meet the standards.

The EPA's certification of the repository, followed by the Secretary of Energy's decision to proceed with waste disposal, completed one of the major steps in opening the WIPP. During the course of the disposal phase, the EPAwill review whether to continue or mod-

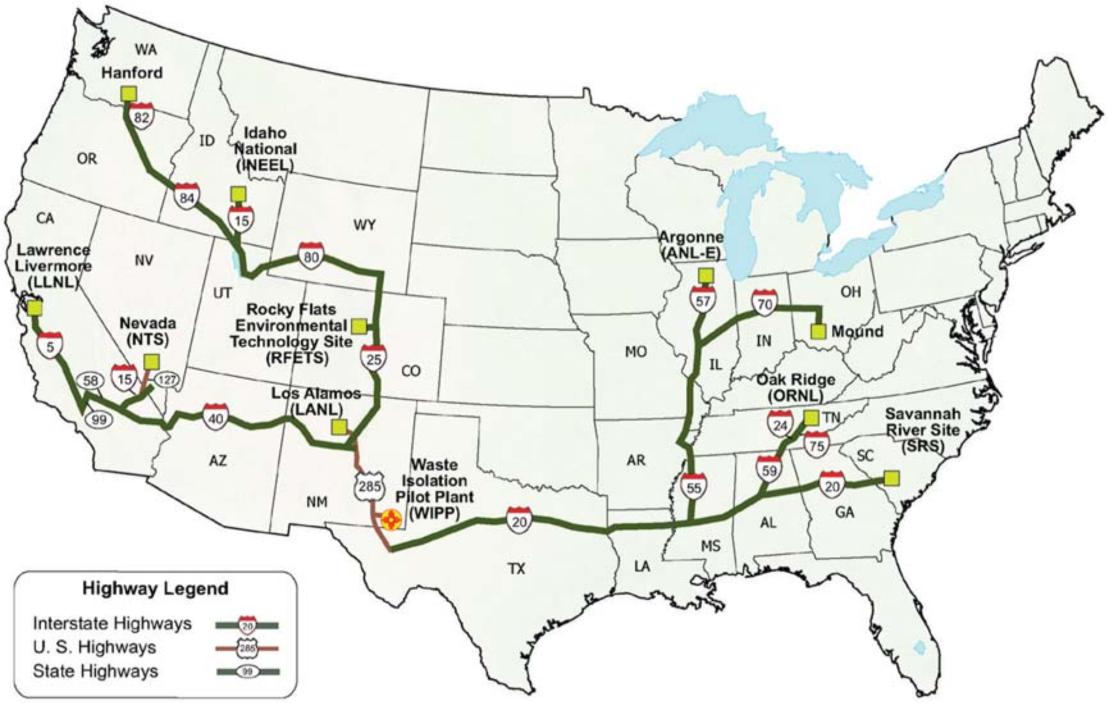

Figure 34.1. Location of TRU waste sites

ify its certification of the WIPP, at a minimum of every five years.

34.4. THE RESOURCE CONSERVATION AND RECOVERY ACT

Congress passed the Resource Conservation and Recovery Act (RCRA) in 1976 to establish requirements for the management of hazardous waste. The term “hazardous” refers to the chemical hazard of the material and is usually associated with chemical toxicity, corrosivity, ignitability, or flammability. Much of the waste to be disposed of at the WIPPis mixed waste, meaning that it contains hazardous waste in addition to TRU. Therefore, the WIPP must comply with RCRA to dispose of mixed waste. The EPA delegated its RCRA regulatory authority to the New Mexico Environment Department, which is responsible for enforcing its requirements and issuing the WIPP RCRA permit.

The RCRApermit application consists of two parts: Part A and Part B. Part A is a standard form that identifies the types and quantities of waste intended for disposal at the site. Timely submission of a Part A application and notification of hazardous waste activities usually qualifies owners and operators of existing hazardous waste management facilities for “interim status.” A facility with interim status is treated as having been issued a permit until the EPA or an authorized state makes a final determination on the facility’s Part B RCRA permit application. DOE submitted the WIPPRCRAPart A application to NMED in 1991. Almost immediately, the New Mexico Attorney General and others filed a challenge as to whether or not WIPP qualified for interim status in court. This resulted in an injunction that was not resolved until March 1999, when the U.S. District Court of the District of Columbia ruled that WIPPqualified for interim status.

Part B of the permit application imposes an extensive set of requirements on how the facility will operate to meet RCRA requirements. Part B includes waste characterization information on the hazardous wastes to be handled at the WIPP, a description of procedures for handling hazardous wastes, security procedures and equipment, seismic and flood plain information, and closure and post-closure plans, including groundwater monitoring. WIPP submitted its Part B RCRA Permit application to the State of New Mexico in 1995.

After three years of amendments, evaluation, and requests for additional information, NMED issued a draft RCRA permit for WIPP on May 15, 1998. After receiving an initial round of public comments, the NMED issued a revision on November 13, 1998, and held public hearings in Santa Fe and Carlsbad, New Mexico. A standard hazardous waste permit is issued for

Figure 34.2. TRUP ACT–II Container

a fixed term not to exceed 10 years. Several permit renewals will be necessary during the projected 35-year operation of the repository.

34.5. WASTE CHARACTERIZATION AT GENERA TOR SITES

Disposal operations at the WIPP require coordination with sites that will ship TRU waste to the WIPP and periodic re-certification of waste-handling procedures at those facilities. During the WIPP's operational phase, shipments will originate from six major sites and more than 12 small-quantity sites in 16 states (Figure 34.1).

Before shipments from a site can begin, the DOE's Carlsbad Field Office and the EPA must certify each site's ability to characterize, prepare, and package the waste. The Los Alamos, Idaho, and Rocky Flats facilities were the first three sites to obtain this certification and ship waste to the WIPP. Once sites are certified, they must then characterize and certify waste. They must identify what is inside the drums and ensure that it meets the WIPP Waste Acceptance Criteria and is packaged correctly for shipment. Both the EPA and New Mexico Environment Department must approve DOE's certification at each site for each waste stream.

In addition, the DOE must consider binding agreements at each of the sites. For example, Idaho and the DOE negotiated an agreement that required the first TRU waste shipment to depart the state by April 30, 1999. The RFETS clean-up agreement requires significant cleanup by 2006. As of this writing, only two other major generator sites (Hanford and Savannah River) have begun routine shipments for disposal. With a limited supply of shipping containers (see next section), shipments must be carefully coordinated from WIPP with all generator sites. The Carlsbad Field Office must ensure that the correct number and type of shipping containers are routed to the sites at the right times.

34.6. BUILDUP OF WIPP'S TRANSPOR TATION SYSTEM CAP ACITY

The LWArequired DOE to license any shipping containers carrying waste to WIPP by the U.S. Nuclear Regulatory Commission (NRC). DOE designed and tested a unique design called the Transuranic Packaging Transporter-Model II (TRUPACT-II), and NRC has issued several revisions to this license over the years. These unique shipping containers make the transportation of waste one of the most noticeable activities of the disposal phase. For the transportation of contact-handled (CH) TRU waste, each shipment consists of a truck and trailer with up to three TRUPACT-II containers (Figure 34.2). DOE recently received certification by the Nuclear Regulatory Commission for a shorter, lighter version of the shipping container, called the HalfPACT. It was designed to accommodate higher waste weight loads and may substitute for one or more TRUPACT-IIs.

As generator-site waste characterization programs mature, and more waste streams are certified, the eventual design shipment rate to WIPP will be 17 per week. In 1999, the DOE awarded contracts to manufacture additional TRUPACT-II containers, and these are being merged into service to accommodate the shipment

Figure 34.3. RH-72B Container

growth rate. The number of available shipping containers is just that needed to meet shipping demand.

DOE has also licensed a shipping container for transportation of remote-handled (RH) TRU waste called the RH-72B (Figure 34.3). This container (certified by NRC in 1999) was designed to shield 1,000 R/hr canisters down to an external dose rate less than 200 mR/hr. Shipments of RH TRU waste are planned to begin in 2003.

Recently, DOE evaluated the need for a new shipping container that would accommodate large payload packages currently stored around the DOE complex. About 35% of the total retrievably stored TRU waste is presently in the form of reinforced fiberglass boxes from 4 feet to 6 feet on each side. This waste would have to be repackaged into standard drums, standard waste boxes, or over-packs to be shipped to WIPP in TRUPACT-II containers. A larger shipping container (yet to be designed and licensed by NRC) in DOE's "fleet" could offer significant cost and risk (exposure) avoidance, by precluding the need for repackaging of this waste. DOE is evaluating options for such a container (generically called a "TRUPACT-III") and plans to make a decision to proceed with engineering design, testing, and certification application to NRC in 2002.

Finally, DOE has begun evaluating the use of rail transportation for shipments to WIPP. While rail was initially considered in the environmental planning process for WIPP, the economic viability never justified detailed planning and engineering designs for that option. Recent evaluation of rail costs and the suitability to supplement truck transport with some rail shipments (from a limited number of generator sites) has showed that rail transport may now be part of a balanced transportation program. DOE will continue to evaluate rail transport over the coming year.

34.7. WASTE VOLUMES AND CHARACTERISTICS

The capacity of the WIPP is limited by the WIPP Land Withdrawal Act to 6.2 million cubic feet of waste. The waste itself can be described in a variety of ways:

- It was generated at DOE defense facilities.
- More than half the waste is considered mixed TRU waste (meaning it contains hazardous components, usually metals or organic solvents) and is subject to regulation by the RCRA.
- About 97% of the waste is CH TRU, while the remaining 3% is RH TRU waste.
- The waste currently in storage at sites around the country was generated after 1970. Most of the remainder will be generated from activities such as environmental restoration, decontamination, and decommissioning.

34.8. REPOSITORY MINING

The WIPP repository will eventually consist of eight panels, each of which contains seven waste disposal rooms (Figure 34.4). Panel 1, which was completed in 1988, was not used for waste disposal until March 1999. Since 1988,

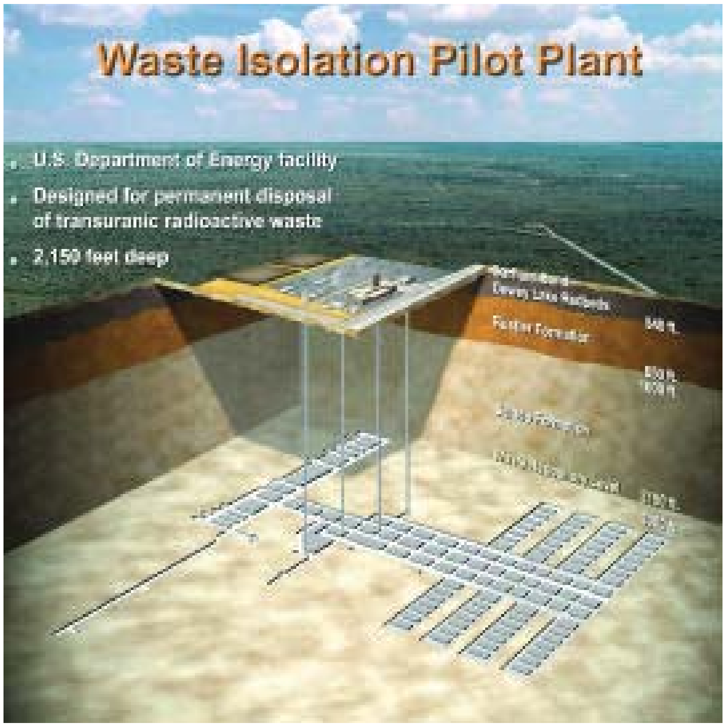

Figure 34.4. Repository plan provided in presentation materials

the walls and floors were maintained by grinding off the salt that creeps, or closes in on, open areas, caused by the plastic nature of the salt formation. In addition, roof bolts were installed in Panel 1 to reinforce the ceiling.

Panel 2 mining was completed in July 2000. Panels will be closed as they are filled, and waste disposal will continue in the next panel. When a panel is half-filled with waste, mining will begin on the next panel. Each panel is expected to take approximately 5 years to mine, fill, and close.

34.9. FUTURE RETURNS ON THE INVESTMENT IN WIP

The characteristics of the WIPP repository and its infrastructure make it uniquely suited for other research. In October 2000, the Carlsbad office of DOE was elevated to Field Office status, and the Secretary of Energy committed the DOE to host scientific research using the WIPP facility, as long as such activities would not compromise its waste-disposal mission. Anticipated uses of WIPP during its operating lifetime include experiments in underground particle astrophysics and low-dose radiation biology, studies to improve natural-resource extraction, repository science services to other nations, and demonstrations of nuclear nonproliferation and transparency technologies for the back end of the fuel cycle.

Such research, which can and will be conducted without compromising the primary disposal mission and the priority on safety, will achieve tremendous benefit from DOE's significant financial and intellectual investment in the WIPP.

34.10. CONCLUSION

After extensive site characterization in the 1980s and 1990s, the WIPP was established in southeastern New Mexico. It was licensed as a deep geologic repository for nuclear wastes in 1998 and began receiving wastes in 1999. Presently CH TRU waste is being transported from locations throughout the United States and disposed of at this site, and RH TRU waste is expected to be disposed of starting in 2003. Disposal operations at WIPP are expected to continue for at least 35 years. Scientific and confirmatory studies may continue well past the disposal period.

Chapter 35

International Repositories: An Essential Complement to National Facilities

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35.1. INTRODUCTION

Many countries have been developing concepts for national geological disposal facilities for decades, and a few of these are within a few years of implementing deep repositories (e.g., Finland, Sweden). Over the same period, many countries have suffered setbacks and long delays relative to their originally foreseen schedules (e.g., Canada, France, Spain, Switzerland, United Kingdom, U.S.A.). Others have begun national projects for geological disposal only relatively recently (e.g., Czech Republic, Japan, Slovak Republic, South Africa).

Developing a deep repository for long-lived wastes demands huge resources and is predicated on locating a suitable geological environment. Typical deep repository costs are estimated to be on the order of one to three billion U.S. dollars, and some sites/concepts may prove even more expensive. Countries with restricted land area or complex, unstable geology may have difficulties in finding a suitable site. There is broad agreement that national nuclear power programs are responsible for finding their own solutions for disposal of their wastes: however, this does not mean that they have to find solutions within their own countries. These factors have led to the concept of international or multinational repositories: shared, well-sited, and safe facilities operated for the benefit of a number of users, with effective use of shared resources. This may, in fact, be the only realistic option for some national programs, but there is also a more general, growing recognition that global, as opposed to national, environmental safety can be encouraged by implementing shared international repositories.

In addition, a further issue has become increasingly prominent: the potential role of geological disposal in reducing the threat of nuclear proliferation. Fissile materials from dismantled nuclear weapons (and also from commercial reactor operations) must be securely isolated to prevent their misuse by terrorist organizations. Deep repositories can be used to dispose of suitably conditioned ex-weapons materials, either in the form of spent fuel (following recycling of the fissile uranium and plutonium through nuclear power reactors) or as solid wastes that meet the so-called "spent fuel standard." International repositories could provide an additional level of fissile-material control.

Over the last few years, Pangea has been investigating and testing many of the technical and nontechnical issues raised in connection with international repositories. This paper begins by reviewing the arguments in favor of international repositories and their relationships to national programs, and concludes by looking briefly at some of the findings of recent Pangea studies.

There are two compelling justifications for international repositories:

1. They contribute to global environmental safety by ensuring that radioactive wastes from programs with limited resources or disposal requirements can be permanently removed from the human environment on similar time scales and to the same technical standard as in national programs.

2. They can enhance global security by providing secure storage and disposal options for surplus fissile materials, thus simplifying the task of preventing their malicious use.

35.2. RESPONDING TO THE NEEDS OF COUNTRIES WITH LIMITED RESOURCES OR LIMITED REQUIREMENTS

Achieving acceptable safety levels requires suitable geological environments to be available. However, repository designs are flexible, and requirements on the geology can often be relaxed by increasing the sophistication of the engineered barriers. This can be seen in the range of designs that is currently being developed by the advanced national programs. Thus, most countries should be able to find suitable sites. However, the availability of a worldwide choice of regions would make it easier to choose sites in stable, simple geological and hydrogeological settings so that the uncertainties in long-term safety assessments are reduced. (Recent Pangea studies on identifying such regions and investigating their safety performance are discussed later in this chapter). The key scientific advantage of a global or regional choice of geological environments thus concerns not the absolute level of safety, but rather the *confidence* with which we can predict the future safety.

Even where a national repository is technically feasible, it may be ruled out by economics. A deep geological facility for long-lived radioactive wastes will cost one to three billion U.S. dollars, no matter how small the volume of wastes to be disposed, and it is inconceivable that each country with such wastes will be able to provide adequate resources. Some countries have only a single reactor; some have no power reactors but still produce long-lived wastes from medical research and industry. For such countries, shared geological repositories are essential. For other countries, complex geology, intense land use, or economic optimization will justify pursuing international options, even if national disposal could be realized. Various small countries (e.g., Switzerland, Belgium, and Taiwan) are also pursuing a "dual-track" approach in which international options are kept open while a national option is also being considered.

There is a strong sustainability argument behind the drive to ensure that some geological disposal option is available to each country. Geological disposal is increasingly recognized as being the only sustainable solution to managing long-lived wastes, because it does not pass on problems to future generations who may have neither the resources nor the capability to handle them. This argument applies in any country and to any size of program. However, it is perhaps an even more cogent argument for those countries, including several European countries with small nuclear programs, where pressure on resources is already evident.

It is important for the global nuclear industry as a whole, and for the wider interests of environmental safety and nuclear security, that the radioactive-waste-management community demonstrates that the challenges of properly managing the wastes of "small users" and countries with limited resources are being properly addressed (i.e., not being ignored while larger countries progress with strictly national programs). Providing such solutions will also give the smaller programs the confidence to make best use of their nuclear power facilities and to have greater flexibility in choosing future power generation options. In short, there should not be a two-tier system for disposal of hazardous wastes, and hence for the application of nuclear technologies. It is in everyone's best interests that equivalent technologies are available on equivalent time scales to all those countries that need them.

These reasons have recently generated an increasing and much more focused interest in the international concept. Pangea, originally established as a commercial venture exploring the possibilities of international repositories, is responding to these developments by taking a much longer-term, more strategic, and less commercial view. The current key objective is restructuring to provide a co-operative "mutual assistance" forum for smaller or geologically complex countries to investigate practical possibilities for shared storage and disposal facilities.

35.3. CONTRIBUTING TO GLOBAL NUCLEAR SECURITY

The security justification for deep geological repositories as a means of safeguarding ex-weapons materials is specifically addressed in a further paper in this volume, by Pentz and Stoll. Current estimates are that Russian and American plans for reducing nuclear weapons arsenals could lead to a surplus of around 2,000 tonnes of highly enriched uranium and over 200 tonnes of weapons-grade plutonium. There are simple options for dealing with these weapons-grade materials. The plutonium can be fabricated into mixed oxide fuel (MOX) which is then burned in reactors, producing highly radioactive spent fuel that is much more proliferation resistant. Alternatively, it can be conditioned into a suitable form for disposal, (e.g., by incorporation into a glass or SYNROC-type ceramic matrix). Enriched uranium can be blended down to produce normal, low-enriched fuel for the current generation of reactors. Again, the result is spent fuel, which is more proliferation resistant but still requires safeguard measures. Thus, a deep geological repository provides a proper end-point for each option. Interim storage is not a permanent solution either environmentally or from a security perspective (as discussed below).

International facilities offer additional, specific features relative to the inherent safeguard advantages of a national deep repository; for example:

- Many countries with spent fuel may not have repositories soon, or ever.
- Host countries can be identified that have especially good safeguard credentials.
- Control becomes even more international than through the current International Atomic Energy Agency (IAEA) regime.
- Repository sites can be selected in regions that are extremely remote and more amenable to surveillance.

An international repository in a country acceptable to all Nuclear Weapon States could facilitate the process of obtaining the necessary political agreements to reduce the number of nuclear weapons further. It could also contribute to the release of the inherent economic value of these materials and provide a commercial source of financing to address nonproliferation goals that are currently difficult for governments to fund. This point is especially important to Russia. The supreme importance of ensuring full safeguards for fissile materials, in an era when dismantling of nuclear weapons is leading to increasing stockpiles, may be one of the more powerful arguments for a potential repository host state, as discussed in the following section.

35.4. INTERNATIONAL REPOSITORIES AND NATIONAL PROGRAMS

Countries with active national programs sometimes express understandable concern about discussions of international repositories because:

- People may fear that the national repository will later choose to—or be compelled to— accept foreign wastes. In Sweden, for example, opponents of national programs have deliberately attempted to use such arguments, despite the existing Swedish legislation ruling out waste import. For this leading program, which is on the way to developing one of the first spent-fuel repositories, public fears are more easily played upon.
- The prospect of a politically easier or economically better external solution might lead national politicians or waste producers to be less committed to preparing a national solution. This argument is irrelevant in those countries (e.g., The Netherlands), where national geological repository preparations are not being made, in any case. It is also not very important in those countries that have chosen a "dual track" strategy, investigating both national and international disposal options.

For those countries fully committed to national disposal, these concerns are real and can be countered only by firmly emphasizing the national strategy. We believe that national and international programs can be mutually supportive. The undeniable need for both is also openly recognized by independent bodies such as the IAEA and the U.S. National Research Council (Dyck and Bonne, 2000; NAS, 2001).

35.5. OPTIONS FOR INTERNATIONAL SOLUTIONS

A national program that has decided not to develop a deep geological repository of its own has a limited range of waste management options available to it:

- It can store its wastes indefinitely within its own borders.
- It can store its wastes at home or abroad until an international solution is available.

In the latter case, it must then either:

- Await the development of a national or international facility offering a disposal service, or
- Participate in pursuing and developing an international solution.

As a complement to its core studies on designing and siting international repositories, Pangea has also examined the basic question of the roles of storage and disposal in the long-term management of radioactive wastes.

35.6. STORAGE OR DISPOSAL?

Some interim period of storage is an inevitable aspect of any national program, whatever its long-term waste management policy. However, there has been much discussion specifically about the indefinite storage option, and Pangea recently carried out a study of attitudes to this option as part of its R&D program. The study (Hill and Gunton, 2001; Hill and Chapman, 2001) compared four long-term radioactive waste management options using a multi-attribute analysis:

1. *Surface storage*—indefinite storage in purpose-built surface facilities
2. *Near-surface storage*—indefinite storage in underground facilities at shallow depth (at most a few tens of meters below the surface of the earth)
3. *Early-seal repository*—disposal in deep geological repositories that are backfilled and sealed shortly (a few years) after waste emplacement has been completed
4. *Late-seal repository*—disposal in deep geological repositories that are backfilled and sealed a fairly long time (at least a few decades, perhaps a century or more) after waste emplacement has been completed.

The surface storage option is not the so-called "do-nothing" option. It is based on the assumption that a national decision has been taken to adopt indefinite storage as a long-term management method and that surface stores are constructed in each country specifically for this purpose. It was also assumed that these surface and near-surface storage facilities are kept under continuous surveillance for safeguards and security purposes, and are monitored and maintained for the indefinite future. Safety, ethical, and economic arguments were weighed for the different options.

Overall, the study indicated clear advantages for disposal over indefinite storage for all long-lived radioactive wastes. Only an emphasis on the very short term (0–100-year period) would lead to difficulty in choosing between storage and disposal, and even in this case, disposal could well be preferred. The explicit treatment of time variations in impacts and features led to the finding that an early-seal repository is preferred to a late-seal repository. It was shown that the results of the generic comparison hold for a range of national political and economic situations, and are not sensitive to whether repositories (or stores) are national or international. In countries with stable societies, there is likely to be less concern about the shorter-term disadvantages of storage options and of the late-seal repository in terms of monitoring and control requirements, extent of responsibilities (social and political), and stability with respect to socio-political changes. If such countries emphasize the short term, they could find it difficult to choose between the four storage and disposal options. An emphasis on the long term, or equal emphasis to all time periods, would lead to a preference for disposal.

In less stable societies, disposal in an early-seal repository would likely be preferred unless very short-term cost considerations were of overriding importance. If international repositories, or stores, were located in politically stable, developed countries, they would be viewed in the same way as proposed national facilities. Politically stable, developing countries with expanding nuclear programs might wish to host an international early-seal repository if the countries sending waste to it were prepared to contribute substantially to funding its construction, operation, and closure. In any country, reluctance to host an international repository (or store) might be overcome if the public became convinced that such a facility would make an important contribution to the protection of human health and the environment, and international security, on a worldwide scale.

35.7. OPTIONS FOR AN INTERNATIONAL REPOSITORY

There are two main options available for promoting and developing an international disposal solution. An international repository might make use of one of the "standard" siting solutions (hard rocks, clays, or salt) that have been under investigation in national programs for more than 25 years and which are now coming to fruition in some countries. This solution, as mentioned in the introduction to this report (Section 35.1), might be implemented in almost any country seeking an international solution. This has led to some discussions of "regional" solutions, where one country might offer disposal facilities to its neighbors. Little concrete progress has been made so far in this area.

Alternatively, one might seek a solution that is designed to make specific use of the flexibility allowed in a truly international approach. A global approach encourages a search for locations where it is easier to demonstrate (in a readily understood fashion) the factors that underpin a safety case, and where the disposal environment will

Table 35.1. Features of optimum regions for repository locations

Large regions identified on the basis of aridity index	Tectonic stability with respect to global seismic hazard map	Lack of volcanicity, on the basis of proximity to current intra-plate hotspots	Overall potential of finding large regions of stable, arid terrain
Northern S.America	Poor (high to very high seismic hazard)	Good	None
West Coast S.America (mainly Chile)	Poor (high to very high seismic hazard: near plate boundary)	Good	None
Southern S.America (Argentina)	Good (low hazard)	Good	Good
Central and Western N.America (mainly USA)	Moderate to poor (low to very high seismic hazard)	Poor	Poor
Southern Africa	Good (low hazard)	Good	Good
Northern Africa	Good (low hazard)	Variable	Moderate to Good
Arabian Peninsula	Good (low hazard)	Variable	Moderate to Good
Southwest Asia (Iran,Iraq)	Poor (high to very high seismic hazard)	Good	Poor
Southern Russia & Kazakhstan	Good to moderate (generally low but bordering on regions of high seismic hazard)	Good	Good
China and Mongolia	Good to poor (some low regions bordering on regions of high seismic hazard)	Variable	Moderate but good in some regions
Australia	Good (low to moderate seismic hazard)	Good	Good

allow great flexibility and economies of scale. Pangea began its existence by looking at such environments and developed the "high-isolation" concept that makes use of the beneficial attributes of remote, tectonically stable, geologically simple, flat, arid desert regions (Miller et al., 1999). The objective behind the concept is to achieve high isolation of the wastes by choosing repository locations where there is essentially no movement of water deep in the rock, around the wastes. Water could transport to the surface any waste that did dissolve, albeit in minute amounts. If there is no likelihood of deep water movement over a period of a few hundred thousand years, then the wastes can very easily be shown to be permanently isolated. In this type of environment, the rock and the site properties provide the bulk of the containment, and less emphasis need be placed on predicting the behavior of engineered structures and materials far into the future. Environments with the right characteristics to provide this simply demonstrable high isolation are to be found in geologically stable, flat, arid areas that have not been subject to major climate-change impacts during the last million years or so (when worldwide glaciations have radically affected many northern countries). Typically, these are found in central continental regions, particularly in the southern hemisphere, in regions such as Australia, southern Africa, and southern South America.

Recently, Pangea has looked in much more detail at where such regions might be found, how sites within these might be identified, and how a repository would perform in such an environment. The technical approach to siting a high-isolation repository is defined by Black and Chapman (2001). This study first identified large regions on the basis of aridity and then evaluated tectonic stability, using data on seismicity, seismic hazard, and volcanicity. The table above (Table 35.1) shows the regions of the world identified.

This information was used to identify areas of primary interest in:

- Australia (where pilot studies began in 1997)
- Argentina

- Southern Africa (Botswana, Namibia, and South Africa)
- Russia
- China and Mongolia.

Other areas could be added to this list, particularly if a search were made for smaller regions with similar properties. The study identified a six-stage siting procedure to narrow in from this worldwide study to potential sites:

- **Stage 1** is the global, high-level desk study carried out by Black and Chapman (2001) that resulted in the identification of regions of the world that are arid or semi-arid and are geologically stable over the time period of concern for waste containment.
- **Stage 2** involves more detailed, individual desk-study assessment of selected broad potential host regions of the world, introducing a first evaluation of topography, geological environment, and mineral and energy resources. It also involves a more detailed consideration of climate in the regions concerned. It produces a group of geological provinces (identifiable geological entities, such as major sedimentary basins) that match the principal high-isolation characteristics.
- **Stage 3** would involve initial access to the regions of interest and the application of much more detailed local knowledge derived from discussions with national organizations such as environmental agencies and geological surveys. Consequently, it requires the agreement and cooperation of the countries concerned. It produces a list of survey areas.
- **Stage 4** may be required, depending on the extent of information available within national databases, and would involve airborne geophysical surveys of survey areas to obtain improved information on large-scale, deep structures. It results in the definition of candidate siting areas.
- **Stage 5** involves the drilling of single deep boreholes (typically of the order of 1,000–2000 m depth) within candidate siting areas, linked to extensive ground geophysical surveys. The results from Stage 5, if carried out in a suitably simple geological situation, should allow the nomination of a group of preferred sites for more detailed study.
- **Stage 6** involves detailed drilling and testing to obtain a comprehensive understanding of site properties that would be sufficient to make a decision on whether to proceed to underground excavation. The decision on whether to carry out Stage 6 investigations in parallel at more than one preferred site within a country would need to be taken at the end of Stage 5, on the basis of many factors, including nontechnical matters. For example, it may be decided that the short list of sites from Stage 5 all display sufficiently similar technical properties that the risk of finding adverse geological features is small enough to warrant focusing on one site. That site might be identified on the basis of social factors, for example by a volunteer community approach.

In parallel with this look at siting factors and procedures, Pangea began preliminary studies of the performance and radiological safety of a high-isolation repository sited in such regions. This study (Apted et al., 2001 in press; Apted et al., 2001) evaluated in particular the performance of the natural (geological) barrier provided by high-isolation sites, where geochemical transport processes are expected to be dominated by diffusion. The study aimed to:

- Assess the effectiveness of the high-isolation concept, considering containment in the waste form itself, in "minimal" engineered structures (i.e., those structures that are required for operational reasons and to satisfy any requirements on retrievability) and in the geological environment.
- Identify and assess the potential significance of phenomena (i.e., scenarios) that might adversely affect the operation of such a system.
- Identify engineered-barrier design concepts in which the high-isolation system could be made more robust with respect to these scenarios.

For the basic concept, and for all the example radionuclides studied (other than the low-sorbing, long-lived ^{129}I), almost 100% of the inventory decays within the repository system. Some radionuclides decay within the waste matrix itself. For some radionuclides (^{99}Tc, ^{237}Np and ^{59}Ni), almost complete decay occurs within precipitates formed around the waste due to solubility limits being exceeded. For others, with higher solubilities (^{135}Cs and ^{79}Se), virtually complete decay occurs within the geosphere. Except for ^{129}I, no more than about one millionth of the initial inventory of any nuclide ever migrates through the 100 m diffusional rock barrier assumed in the study and into the surrounding rocks or biosphere. It is expected that the diffusional rock barrier would actually be more extensive than the 100 m conservatively assumed in this preliminary study.

35.8. GETTING WASTE TO AN INTERNATIONAL REPOSITORY

The high-isolation concept would locate disposal facilities in stable, arid environments, which tend to be remote and located hundreds of kilometers from the coast. A frequently expressed concern is that transport of highly radioactive wastes to an international repository would create unacceptable risks to the public. Over the operational life of a repository, many thousands of tons of waste will be shipped across the world, by sea, to the host country. This is not a new practice. There is considerable international experience in waste transport, and the hazards have been shown to be insignificant, by any standard. Nevertheless, Pangea launched a preliminary study in 2000 that set out to examine what the potential hazards might be and to quantify the actual risks, so as to provide a more specific answer to questions about public safety.

The study (Tunaboylu et al., 2001) involved evaluating the risks to individual members of the public that might result from the transport of a large inventory of spent MOX fuel by sea from Europe or the Northwest Pacific area to a southern hemisphere repository location (about 20,000 km) and then by road or rail to a site 1,000 km inland. The individual risks of fatalities to individuals from accidents involving radiation releases from transported spent fuel were estimated to be of the order of one in ten billion per year, or less. The results showed that:

- The radiological consequences resulting from possible transport accidents are significantly below the permissible exposure limits set in national and international regulations. They are also significantly lower than risks that are commonly regarded as trivial and acceptable to society and far below risks that can be attributed to natural background-radiation levels.
- The risks of sea transport are several orders of magnitude less than for land (rail/road) transport.
- The radiological risks arising from accidents during land transport of spent fuel are 1–2 orders of magnitude less than the radiological risks during incident-free transport. Incident-free transport risks are of the same order of magnitude as the nonradiological risks of the same transport, due to exposure of the population to vehicle exhaust gases.

35.9. THE WAY AHEAD FOR INTERNATIONAL REPOSITORIES

It has often been stated by national and international organizations that national repositories must “show the way” before any country will find it politically feasible to accept imported wastes for disposal. In practice, the commencement of operation of the first purpose-built deep geological repository, WIPP, has unfortunately not led to any obvious breakthrough in public or political acceptance of other facilities. Nevertheless, the greater the extent to which geological disposal becomes an accepted, widespread practice, the easier it should be to engage in objective dialogue on the pros and cons of international facilities.

Accordingly, it is in the interest of the entire waste management community to support national programs that are close to achieving the goal of repository siting and licensing. International technical cooperation can help here, and there are many channels open to encourage this. In addition, should the prospect of national facilities choosing (or being compelled) to accept foreign waste be an important argument, then the implementing country, other nuclear nations, and also international organizations can all take pains to emphasize the sovereign right of any country to refuse foreign wastes. The IAEA Waste Convention is a useful document in this respect.

On the other hand, even national programs that have firmly decided against waste import can be constructive, by acknowledging that shared international facilities will be needed by others less fortunate in their national geology or in their economic situation. Indeed, the countries that are advanced in their national programs could offer direct support of efforts aimed at implementing shared repositories elsewhere. As in the case of national repositories, it will be difficult to achieve acceptance of geological disposal if there is no widely recognized need for it. The sooner that credible, voluntary, international repository projects are established, the easier it will be for national programs to demonstrate to their publics, if they so wish, that there need be no compulsion on any country to import wastes.

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Chapter 36

The Contribution of Deep Geologic Repositories to Nuclear Nonproliferation

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36.1. INTRODUCTION

Discussions about the role of international or regional storage and disposal facilities for managing unwanted nuclear materials have become commonplace in a widening circle of conferences, meetings, and articles (Nunn and Ebel, 2000; Starr and Hafele, 2000; McCombie and Stoll, 2000; Bunn, 2000; and Finlay, 2000). While most discussions have focused upon safety and environmental benefits, recent international political events have reinforced the importance of the nonproliferation advantages of deep geologic repositories and the critical role of the private sector in contributing to international security. Proper safeguarding and disposition of excess nuclear weapon materials, particularly in Russia, are of immediate concern globally. This is a shared mission that both the nuclear waste management community and the nuclear nonproliferation community should embrace and enthusiastically advocate.

36.2. DISMANTLING NUCLEAR WEAPONS IS A SERIOUS GLOBAL SECURITY CHALLENGE

With the end of the Cold War, the reduction in the nuclear weapons stockpiles in the United States and Russia, for strategic as well as economic reasons, is creating large quantities of surplus weapons-grade materials. There are international security concerns that these materials, the bulk of which is Highly Enriched Uranium (HEU), but which also include significant quantities of plutonium, pose a serious proliferation threat if they fall into the wrong hands. In Russia alone, there are unclassified estimates of up to 160 tons of plutonium and 1,500 tons of HEU. The quantities of these materials are large enough to fuel a significant percentage of the commercial nuclear market for many years and thus, in addition to being a serious proliferation problem, they present a significant challenge to the market share of existing fuel-cycle suppliers. Progress with arms reduction treaties and agreements, especially in 2000, is driving the need for a solution to the disposition of these materials.

36.3. THE SOLUTION IS PERMANENT DISPOSITION OF THE NUCLEAR MATERIALS IN DEEP GEOLOGICAL REPOSITORIES

Currently, the two accepted methods for disposing of these nuclear-weapons materials are either to convert them into nuclear-reactor fuels or to immobilize them. However, because of its need for cash, Russia is insistent that, as part of any international disposition plan, its materials must be converted into nuclear reactor fuels, not immobilized. The United States has developed a Dual Track Plan for its material that includes both nuclear fuel conversion and immobilization, but Congress has mandated that U.S. disposition must proceed in parallel with stockpile reductions in Russia. Whatever disposition methods the U.S. and Russia use, all require eventual permanent disposal underground in a geologic repository.

36.4. COMMERCIAL SPENT FUEL ALSO MUST BE SAFEGUARDED

Commercial nuclear electricity generation throughout the world currently results in annual discharges of about 10,000 tons of spent fuel that contains about 1 percent plutonium. While this plutonium is not easily separated from the intensely radioactive spent fuel and is not of the same quality as plutonium removed from weapons, it can still be used as a threat or in a crude weapon. Smaller quantities of highly enriched uranium form the fuel for research reactors around the world. Because the radioactivity of spent fuel decays over time, it loses its natural proliferation protection, and thus long-term storage is not a permanent solution to assure security. Additionally, some spent fuel could be produced with inferior barriers to diversion through lower burn-up rates or unplanned early removal from reactors.

36.5. NATIONAL GEOLOGICAL REPOSITORIES HAVE NON-PROLIFERATION BENEFITS

Today, most commercial spent fuel is maintained under a strict, worldwide safeguards regime, and the security threat is less urgent than the danger from excess nuclear weapon materials. However, commercial materials must also (ultimately) be made as inaccessible as possible. As for the excess weapon materials, the answer is implementation of geologic repositories. For nations with good prospects for successful national repository programs, there are very useful nonproliferation advantages to concentrating fissile materials from many locations into a carefully selected national site technically easier to safeguard.

In any fuel-leasing scenario, a necessary requirement is the availability of a storage and disposal facility that can relieve the nuclear utilities from any future liability for the management and permanent disposition of their spent fuel. National geological repositories can provide an alternative to returning spent fuel to a Russian repository. The nation would be providing a valuable contribution to international security by increasing the disposition rate of Russian ex-weapon materials, while simultaneously enhancing public acceptance for its own disposal program and demonstrating the viability of a commercial-industry role in fissile-material disposition.

36.6. INTERNATIONAL REPOSITORIES CAN INCREASE GLOBAL SECURITY YET FURTHER

National repositories can contribute to nonproliferation, but will all nations with fissile materials have access to a geological repository? At least 33 nations currently have commercial nuclear power programs and others have research reactor programs, and it is very unlikely that every one of these countries will possess the political, economic, and geological factors necessary to create permanent geologic disposal programs for their materials soon or ever. For these nations, regional or international spent-fuel management solutions must remain as a viable option for both environmental and security reasons. Moreover, an international repository can be optimized for properly safeguarding the fissile materials from all users of the facility (Pellaud and McCombie, 2000).

In a future of continually increasing quantities of excess nuclear weapon materials from dismantling nuclear weapons and commercial nuclear-industry activities, there is a convincing requirement to create new spent-fuel storage facilities and develop diplomatic consensus about the role of international solutions in solving this security threat. A global system of a few disposal facilities in isolated areas under multinational scrutiny should be preferable to many small national facilities that often are located in less than ideal conditions (Carter and Pigford, 2000; Peterson, 2000).

36.7. CONCLUSION

Governments are seeking innovative, flexible, commercially sustainable solutions that can supplement current diplomatic efforts to dispose of these materials. As an assured disposal route in support of arms control objectives, an international repository may encourage faster conversion of nuclear weapon materials into nuclear fuel to achieve the spent-fuel standard that is the goal of many nonproliferation programs.

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Chapter 37

International Cooperation in Nuclear Waste Management

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ABSTRACT. The ultimate responsibility for ensuring the safety of spent fuel and radioactive waste rests with the State. Nevertheless, there are circumstances where safe and efficient management of spent fuel and radioactive waste might be fostered through agreements among States to use facilities in one of them for the benefit of the others. In general, the United States favors the idea of States in a region getting together to solve their nuclear waste problems, though each proposal for a regional repository must be evaluated individually on its own merits. The IAEA and the Joint Convention on the Safety of Spent Fuel Management and the Safety of Radioactive Waste have established technical considerations governing transfers of radioactive waste and spent fuel, but there are political considerations as well, and this is an issue that ultimately must be decided by States.

Most current proposals for international repositories involve spent fuel from countries where the U.S. has consent rights over the retransfer of much or all of the spent fuel. For the U.S. to consent to the retransfer of spent fuel to a repository, several things would be required. The U.S. would need an agreement for cooperation with the receiving State. The U.S. would need to know that the transfer was for eventual disposal, not reprocessing. The U.S. would need to know the material would be handled safely in transit. The U.S. would need to be assured that the storage and disposition facilities were suitable. Finally, the U.S. would want to be sure that mechanisms were in place to ensure that the large amount of money that would change hands was properly managed.

37.1. ENSURING THE SAFETY OF SPENT FUEL AND RADIOACTIVE WASTE

The ultimate responsibility for ensuring the safety of spent fuel and radioactive waste rests with the State. This is affirmed in the Preamble to the Joint Convention on the Safety of Spent Fuel Management and the Safety of Radioactive Waste. With the recent accession of the UK and Ireland to the Convention, it should enter into force on June 18, 2001. The U.S. was the first nation to sign the Convention, and it was submitted last year to the Senate for its advice and consent to ratification, but we have not yet ratified it. Nevertheless, the Convention incorporates some important principles that I will want to return to from time to time.

Reflecting our belief that responsibility for safety of spent fuel and radioactive waste rests with the State, the U.S. has maintained a strong program of international cooperation in waste management technologies so that our trading partners can manage their own waste arisings. My personal experience with our nuclear cooperation committees with Taiwan and the Republic of Korea goes back almost 20 years, and spent fuel and nuclear waste management have been on the agendas of most of our meetings. Many of you have first-hand knowledge of DOE's repository science programs and site evaluation methodology. But one thing we cannot expect to see is the U.S. giving consideration to taking irradiated U.S.-origin fuel supplied for electricity generation back for storage and disposition, in Yucca Mountain or elsewhere. The Nuclear Non-Proliferation Act of 1978 makes any plan for the return of such fuel subject to stringent conditions, including submission to Congress, which has the option to reject it. Subsequently, Congress prohibited the Executive Branch from even spending money to formulate or review such a plan.

While giving primacy to the responsibility of the State that generates nuclear waste to dispose of it on its own territory, the Waste Convention recognizes that in certain circumstances safe and efficient management of spent fuel and radioactive waste might be fostered through agreements among contracting parties to use facilities in one of them for the benefit of the other parties, particularly where the waste originates from joint programs. In general, the U.S. favors the idea of States in a region getting together to solve their nuclear waste problems. Conceptually, this is similar to the Waste Compact program in the U.S. in which several American states join together in compacts to locate a low-level waste repository in one of them, rather than to locate one in each.

Each proposal for a regional repository must be evaluated individually on its merits. The IAEAhas developed a Code of Practice on the Transboundary Movement of Radioactive Waste. The most important provisions of the Code found their way into the Waste Convention. This Code and the Convention provide some technical guidance for a State in determining whether or not to participate in a regional repository scheme, or in any other international waste shipment. The overriding principle is that a sending State should ship waste or spent fuel only with the consent of the receiving State and only after satisfying itself that the receiving State has the administrative and technical capacity, as well as the regulatory structure, needed to manage the waste or spent fuel safely. Similarly, the receiving State should only consent to receiving the waste or spent fuel if it can satisfy itself that it can meet those requirements. This means that shipments of spent fuel and nuclear waste are fundamentally a matter to be decided by States, rather than simply a commercial matter, just as is the rest of nuclear trade. While technical factors are important in each evaluation, political factors are sure to enter into the consideration of each proposal as well.

I wish I had the answer to public acceptance. I really do. If it were an easy problem, there would be a regional spent-fuel repository by now, because the concept has certainly been around for nearly 25 years, since the first proposal to put one on Palmyra Island during the Carter Administration as an alternative to reprocessing. But I think it is inevitable that at least in some areas of the world, regional repositories will be built. There are 30 countries plus Taiwan with operating nuclear power plants, two with a plant under construction, and two more with shut-down plants. That means that even if no more countries adopt nuclear power, 34 countries plus Taiwan will have to dispose of spent fuel and/or high-level waste from reprocessing. It is hard to imagine 35 separate deep geologic repositories. It is particularly hard to imagine them in regions of closely grouped States, each with fuel from only a few nuclear plants to dispose of (e.g., Eastern Europe). They might conclude that their environs would be better served by one repository than by several.

Similar considerations surround low-level waste repositories, although the need for them is more widespread, encompassing States that use radioisotopes for medical, industrial, or agricultural purposes. So far, States have had more success in siting them indigenously, but there have been several well-publicized proposals for transfers of low-level waste for disposal, and I would not be surprised to see more.

I don’t think I am saying anything new or unique. The reports before this one have discussed several ideas about repositories, and I do not want to comment on any one in particular. However, I do want to offer a few general observations.

Most proposals for a regional or international repository focus on the nuclear programs in Taiwan and the Republic of Korea (ROK) as potential customers. Both Taiwan and the ROK would be large sources of fuel and the money to make a repository proposal attractive to a potential host. The U.S. has retransfer consent rights on all the spent fuel on Taiwan and much of the spent fuel in the ROK. Consequently, U.S. consent would be required for a repository scheme involving one of these two customers to work. Switzerland and Japan have also been mentioned as potential customers, and U.S. consent is involved with much of this fuel as well.

37.2. SOME IMPORTANT CONSIDERATIONS IN TRANSFERRING SPENT FUEL AND NUCLEAR WASTE BETWEEN STATES

Under Section 131 of the Atomic Energy Act of 1954, as amended, U.S. consent rights over the retransfer of spent nuclear fuel are exercised by the Secretary of

Energy on a case-by-case basis through a process called a Subsequent Arrangement. The Act establishes procedures and standards for concluding Subsequent Arrangements. These include requirements that the Secretary of Energy obtain the consent of the Secretary of State to any Subsequent Arrangement and consult with the Department of Defense and the Nuclear Regulatory Commission. The Secretary of Energy must also make a written determination that such Arrangement will not be inimical to the common defense and security. A notice of the proposed Subsequent Arrangement and this determination must be published in the Federal Register for 15 days before the Arrangement can take effect.

The United States does not authorize retransfer of nuclear material to countries to which it could not transfer nuclear material directly. Therefore, a recipient of any retransfer for the purpose of disposal in a repository must have a Peaceful Nuclear Cooperation Agreement in force with the United States. Section 123 of the Atomic Energy Act details a series of requirements for such an agreement. The U.S. currently has Nuclear Cooperation Agreements with EURATOM, the IAEA, Taiwan, and 25 countries, but the list does not include Russia.

In addition, to avoid increases in civil stockpiles of separated plutonium, the U.S. would need to be assured that the transfer was for eventual disposal and not for reprocessing. We would not necessarily expect the permanent repository to be available immediately, and we can see a period of long-term storage as part of any scheme. But any scheme should involve specific plans for, and specific commitment of sufficient resources to, development of a geologic repository. All U.S. Peaceful Nuclear Cooperation Agreements contain specific clauses requiring U.S. consent for the reprocessing of U.S.-origin nuclear material, and we would expect to use such consent rights to enforce the disposition, *vice* reprocessing, of fuel transferred to a repository. We would also likely seek a specific no-reprocessing pledge from the receiving State.

Moreover, prior to providing consent for retransfer, the U.S. would have to be assured that the material would be handled safely in transit. Sea transport of radioactive materials is routinely carried out with an exceptionally high degree of safety and security, in compliance with stringent IAEAand International Maritime Organization standards. Nevertheless, such shipments are highly controversial, and some coastal and small island States are increasingly vocal in calling for greater regulation or an outright ban. Attempts to ship through international choke-points, like the Panama Canal, the Straits of Malacca, or the Bosporus and the Dardanelles could risk attempts to pose unilateral restrictions or even attempts at interception by protestors. Large-scale movement of nuclear material from a port to a repository, via road or rail, might prove to be a challenge for many nation's infrastructures, and (as has been seen in Germany) can be another focal point for protest. However, the technology for the transport casks is well established, and any foreseeable incidents are likely to be more a matter of inconvenience than safety.

The U.S. would also need to be assured that the storage and disposition facilities were suitable. The technology for long-term interim storage is well established, and we would expect any scheme to be environmentally sound. Many of the aspects of geologic containment are still under investigation in connection with the potential siting of a U.S. repository at Yucca Mountain. However, the 1999 opening of the Waste Isolation Pilot Plant (WIPP) in New Mexico marked the world's first geologic repository, and a giant step forward. We are making a broad range of efforts to share our experience with both WIPP and Yucca Mountain. We would want to assure ourselves first-hand that any facility storing irradiated fuel of U.S. origin is constructed on an environmentally sound basis.

Finally, the U.S. would want to be sure that mechanisms were in place to ensure that the large amount of money that would change hands, much of it up front, was properly managed and accounted for and remained available to manage the spent fuel for the life of the disposition program. The obligations being undertaken will be very long term, longer than what a commercial entity might be able to guarantee. An intergovernmental organization such as the Korean Peninsula Energy Development Organization might be one way to address that issue.

Now that I have listed these factors that might go into U.S. consideration of whether to grant consent to retransfer irradiated fuel of U.S. origin, I would like to conclude by saying that I hope we get the opportunity to work through these considerations. In each case, I think they are factors that we have to make work in order to have a successful repository program, not obstacles to a repository program.

Appendix A

Attendees at *Third Worldwide Review* Workshop

Last name	First Name	Affiliation
Ahn	Joonhong	U.S.A./UC Berkeley
Alford	Paula	U.S.A./NWTRB[1]
Astudillo	Julio	Spain
Bodvarsson	Gudmundur	U.S.A./LBNL[2]
Boyle	William	U.S.A./DOE[3]
Bradshaw	Les	U.S.A./ State of Nevada
Bredell	Piet	South Africa
Burkart	Alex	U.S. Department of State
Chou	C.K.	U.S.A./DOE[3]
Cochran	Tom	U.S.A./NRDC[4]
Dyer	Russ	U.S.A./DOE[3]
Evstatiev	Dimcho	Bulgaria
Gallagher	Kimberly	U.S.A./LBNL[2]
Halsey	William	U.S.A./LLNL[5]
Han	Kyong-Won	Korea
Hess	Ken	U.S.A./BSC[6]
Hok	Jozef	Slovak Republic
Isaacs	Tom	U.S.A./LLNL[5]
Kovacik	Milos	Slovak Republic
Krushchov	D.	Ukraine
Kurochkin	Vitaly	Russia
Landais	Patrick	France
Lebon	Patrick	France
Levich	Bob	U.S.A./DOE[3]
Levine	Mark	U.S.A./LBNL[2]
Lin	Wunan	U.S.A./LLNL[5]
Lundqvist	Berit	Sweden
Marschall	Paul	Switzerland
Masuda	Sumio	Japan
Matejovic	Igor	Slovak Republic
McCombie	Charles	Switzerland/Pangea Resources
Mele	Irena	Slovenia
Myers	Wes	U.S.A./LANL[7]
Neerdal	Bernard	Belgium
Nodora	Donald	U.S.A./LBNL[2]
Ormai	Peter	Hungary
Park	Hyun-Soo	Korea
Patterson	Russ	U.S.A./WIPP[8]
Pavelescu	Margarit	Romania
Pentz	David	U.S.A./Pentz Consulting
Poskas	Povilas	Lithuania
Pye	John	U.S.A./NWTRB[1]
Reiter	Leon	U.S.A./NWTRB[1]
Rui	Su	China
Sakuma	Hideyuki	Japan
Shiotsuki	Masao	Japan
Simmons	Ardyth	U.S.A./LBNL[2]
Stevenson	Karyn	U.S.A./NWTRB[1]
Stoll	Ralph	U.S.A./Pangea Resouces
Tsang	Yvonne	U.S.A./LBNL[2]
VanLuik	Abe	DOE
Villavert	Maryann	U.S.A./LBNL[2]
Vira	Juhani	Finland
Voegle	Michael	U.S.A./BSC[6]
Von Tiesenhausen	Engelbrecht	U.S.A./ Clark County, NV
Vujic	Jasmina	U.S.A./UC Berkeley
Wallner	Manfred	Germany
Wang	Joseph	U.S.A./LBNL[2]
Williams	Jeff	U.S.A./DOE[3]
Witherspoon	Paul	U.S.A./LBNL[2]

1 Nuclear Waste Technical Review Board
2 Lawrence Berkeley National Laboratory
3 Department of Energy
4 Natural Resources Defense Council
5 Lawrence Livermore National Laboratory
6 Bechtel SAIC Company
7 Los Alamos National Laboratory
8 Waste Isolation Pilot Plant