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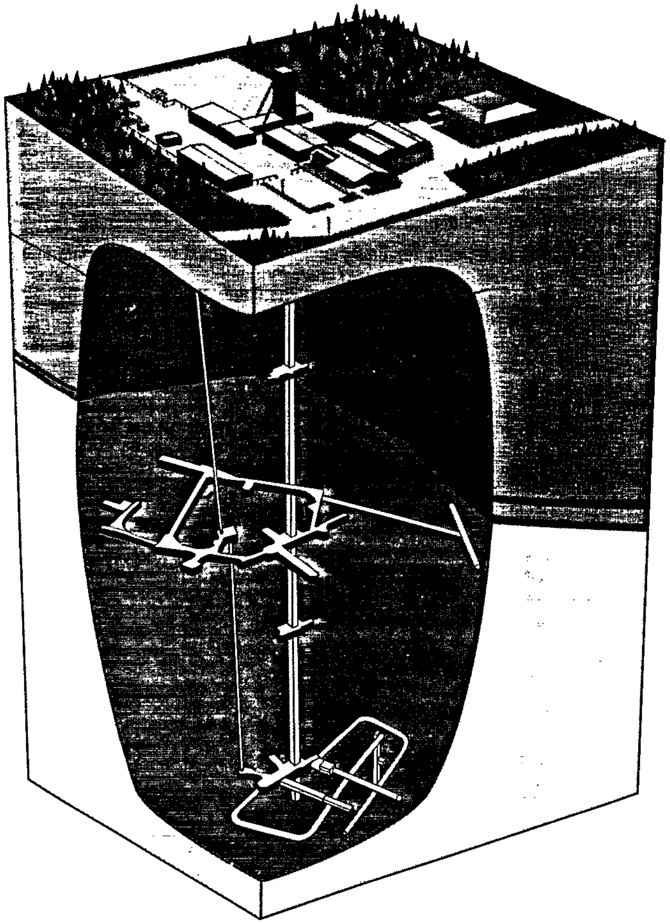
Geological Problems in Radioactive Waste Isolation

Second Worldwide Review



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GEOLOGICAL PROBLEMS IN RADIOACTIVE WASTE ISOLATION

SECOND WORLD WIDE REVIEW

P. A. WITHERSPOON, EDITOR

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CHAPTER 1

INTRODUCTION TO SECOND WORLD WIDE REVIEW OF GEOLOGICAL PROBLEMS IN RADIOACTIVE WASTE ISOLATION

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1.1 INTRODUCTION

The first world wide review of the geological problems in radioactive waste isolation was published by Lawrence Berkeley National Laboratory in 1991[1]. 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 this first review that the problem of providing a permanent and reliable method of isolating radioactive waste from the biosphere is a topic of great concern among the more advanced, as well as the developing, nations of the world. This is especially the case in connection with high-level waste (HLW) after its removal from nuclear power plants. The general concensus is that an adequate isolation can be accomplished by selecting an appropriate geologic setting and carefully designing the underground system with its engineered barriers.

There is 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, but important, problem for the nuclear industry.

Much new technology is being developed to solve the problems of waste isolation, and there is a continuing need to publish the results of new developments for the benefit of the international nuclear community. Thus, it was decided that after a five-year interval, it would be desirable to gather material on the latest developments and publish another review on the geological problems of radioactive waste isolation. As shown in Table 1.1, this second review contains reports from 26 countries.

1.2 SOME HIGHLIGHTS FROM THE SECOND REVIEW

1.2.1 Characterizing the Repository Site

Although no repository for HLW has yet been put in operation, significant progress has been made on this subject since the publication of the first review. 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 over ten years, and the wide variety of technologies that are described in this review reflects the fact that, in general, each country has its own internal constraints to satisfy. The process of site characterization can be significantly different depending on the particular type of rock that has been selected as a potential repository site.

The problem of locating a repository for LILW is not as difficult as for HLW, primarily because there is no heat release from the waste to cause temperature problems. Furthermore, a number of the reports in this review describe some well thought out procedures that have been developed to handle LILW. For example, in Slovenia[2], the siting process has been divided into four steps. In the first step, unsuited areas are omitted from consideration on the basis of certain exclusionary criteria. In the second step, the remaining acceptable areas are further reduced to potential sites according to land use, water resources, seismic and geological criteria. In the third step, several of the most suitable of the potential sites are chosen by comparing their locations on the basis of population, economic feasibility, transport, ecology, and public acceptance. In the final fourth step, a comprehensive analysis of the most suitable sites from

the third stage is made by applying the criteria of the previous steps and an additional factor involving the corrosive nature of the soils, and then a detailed field investigation is carried out to confirm site suitability.

1.2.2 Maximum Repository Temperature

The problem of characterizing the potential repository site for HLW is complicated because of the heat generated by the decay process. If the HLW is not stored at the surface for a lengthy period so as to lose most of its thermal generating capacity, the heat released in the underground can raise repository temperatures well above ambient for thousands of years. In the immediate vicinity of the drift in which the canisters containing the waste have been placed, the temperature of the rock walls may reach as much as 200° C. This depends on the canister spacing and the thermal generating capacity of the waste. When one considers that the area of the repository may be several square kilometers in size, the mass of rock that will be thermally perturbed is significant, and as a result, the problems of understanding the factors that control the coupled behavior (thermal, hydraulic, chemical, mechanical) of such a rock mass are formidable.

As a result, it will be noted in this review that most countries have been following the lead of the early workers in Europe, who have adopted the practice of storing the spent, as well as reprocessed, fuel in surface cooling ponds for 40 to 50 years. This will dissipate the great bulk of the heat load, so that after emplacement in the repository, maximum rock temperatures will not exceed 100° C. This procedure has been adopted by practically all countries except United States where the cooling period may be no more than 10 years. Currently, the United States is investigating a potential site at Yucca Mountain in the State of Nevada where the rock is a fractured tuff. If a repository is eventually built at this site, the current conceptual repository design would produce maximum emplacement drift wall temperatures of approximately 155° C at about 40 to 60 years after emplacement.[3]

1.2.3 Rock Types Under Consideration

Another point of interest is the variety of rock types that are under consideration in the different countries where a repository may be built. In the first review, whch was mainly concerned with HLW, granitic rocks were the primary rock type under consideration, but it is evident in Table 1.1 that a much wider range of rock types is now being evaluated. In those countries where detailed investigations have been carried out since the first review, excellent summaries are presented of the new technologies that have been developed for several different rock types.

1.2.4 International Waste Management Systems

The establishment of international waste management

Table 1.1. List of countries and rock types being investigated where radioactive waste repositories may be located.

Country	Rock Type for HLW	Rock Type for LILW	Country	Rock Type for HLW	Rock Type for LILW
Belarus	clay, salt	clay, salt	Japan	(1)	
Belgium	clay	clay	Korea		andesite
Bulgaria	granitic, marls		Netherlands	salt	
Canada	granitic		Poland	(1)	(1)
China	granitic		Slovakia	(1)	(1)
Croatia		(1)	Slovenia	marl	
Czech	granitic		Spain	(1)	(2)
Finland	granitic		Sweden	granitic	
France	(1)	(2)	Switzerland	clay, granitic	marl
Germany	salt	iron ore, salt	Taiwan		(1)
Hungary	claystone		Ukraine	granitic, salt	
India	granitic		United Kingdom	volcanics	
Indonesia	basalt		United States	tuff	

Note: (1) Not yet determined; (2) Surface facility.

systems has been suggested in the past and a number of studies has been undertaken[4,5,6,7] to address the inherent difficulties associated with the disposal of limited amounts of HLW exclusively within national borders. Such a system would be set up to accept and manage radioactive waste from countries with small nuclear energy programs and relatively small volumes of HLW. Bredell and Fuchs[8] and Lin[9] have recently discussed the feasibility of such systems, from technical, economical, institutional and ethical viewpoints.

The feasibility of an international waste management system is not discussed in detail in any report of this second review, but as one reads of the activities in the smaller counties (e.g., Switzerland[10], Taiwan[11], Ukraine[12]) it is clear that each one is becoming aware of a difficult situation in developing a viable program to handle the problems of isolating HLW. A somewhat different aspect of the need for an international facility may develop in Indonesia[13] because this archipelago is one of the regions of the world with active volcanisim.

The need to consider international waste management systems has been discussed in a recent editorial by Issler[14], who raises some important arguments. From the economic standpoint, countries with relatively small nuclear energy programs and relatively small volumes of HLW are faced with solutions that are essentially uneconomic. A large proportion of the disposal costs is independent of waste volume, particularly those related to concept development, site selection and characterization and, to a large extent, construction and operation of the facility. The costs of site characterization can be very large. For example, the cost of the total effort to characterize the HLW site at Yucca Mountain in the United States is currently about $2.5 billion[15]. This is atypical of costs in Europe, but it serves as an upper bound to illustrate the magnitude of costs associated with this complex problem.

From the practical standpoint, it may be very difficult for small countries to find a HLW repository site that is satisfactory geologically and, at the same time, can satisfy planning restrictions. On the other hand, the geological situation in a particular country may not be favorable, and the country has no choice except to store the waste for an indefinite period. In this respect, regional repositories could provide a safer solution. And from the technical standpoint, the more advanced countries can help others that are disadvantaged by lack of infrastructure, restricted financial means, insufficient technical capacity, or lack of relevant know-how.

Issler has a very good point that puts the situation in context. The United States is planning only one or two repositories for its HLW. Europe, which is comparable in geographic area, has 18 of the 26 countries on Table 1.1, that need disposal facilities. Should 18 national repositories for HLW be constructed, or do not the economic and geologic considerations indicate that two or three regional facilities would suffice? It is understandable that acceptance of international solutions, particularly in the host countries, will be difficult to achieve, but the option of international waste management systems should be kept open and needs careful consideration.

REFERENCES

1. Geological Problems in Radioactive Waste Isolation: A World Wide Review, Lawrence Berkeley Laboratory Report, LBL-29703, Ed. P.A. Witherspoon, 233 pp., 1991.

2. Petkovsek, B., M. Dusan and I. Osojnik, Geological aspects of site selection for low and intermediate level radwaste repository in Slovenia, this volume.

3. Dyer, J.R. and M.D. Voegele, High-level radioactive waste management in the United States, Background and Status: 1996, this volume.

4. Kühn, K., Are we ready to construct and operate an underground repository?, Proceedings Symposium on Siting, Design, and Construction of Underground Repositories for Radioactive Wastes, IAEA, Vienna, Austria, Mar. 3 - 7, 1986.

5. International Approaches on the Use of Radioactive Waste Disposal Facilities, OECD, Report, Paris, France, p. 13, 1987.

6. Lin, W., A global nuclear waste repository, Forum, EOS Transactions, American Geophysical Union, 69, 18, p. 561, 1988.

7 Technical, Institutional and Economic Factors for Developing a Regional Repository, IAEA Report, Vienna, Austria, Nov. 28 - Dec. 2, 1994.

8. Bredell, P.J. and H.D. Fuchs, An approach to international high level radioactive waste management, High Level Radioactive Waste Management, Proceedings Seventh Annual International Conference, Las Vegas, NV, pp. 486-488, Apr. 29 -

May 3, 1996.

9. Lin, W., International high-level radioactive waste repositories, High Level Radioactive Waste Management, Proceedings Seventh Annual International Conference, Las Vegas, NV, pp. 492-493, Apr. 29 - May 3, 1996.

10. McKinley, I.G. and C. McCombie, High-level radioactive waste management in Switzerland: Background and status 1995, this volume.

11 Tsai, C.M. and D.S. Liu, Low level radioactive waste management in Taiwan, this volume.

12. Khrushchov, D.P. and V.M. Starodumov, Programme and results of initial phase of radioactive waste isolation in geological formations in Ukraine, this volume.

13. Imardjoko, Y.U., H.B. Santosa, Sunarno, N. Prabaningrum, A. Muharini, E. Wijayanti, and M. Adiartsi, Critical data required to potentially investigate Genting Island as a high-level radioactive waste repository site facility in Indonesia, this volume.

14. Issler, H., International radwaste collaboration - View from a small country, Editorial, Nuclear Europe Worldscan, 3-4, 1996.

15. Witherspoon, P.A., R.A. Freeze, F.A. Kulacki, J.N. Moore, F.W. Schwartz, and Y.C. Yortsos, Peer Review Report on Thermohydrologic Modeling and Testing Program, Appendix D, U.S. Department of Energy Determination of the Use of the Results of the Peer Review Report on the Thermohydrologic Modeling and Testing Program, Yucca Mountain Site Characterization Project, August, 1996.

CHAPTER 2

DISPOSAL OF RADIOACTIVE WASTE IN BELARUS AND COMPLICATIONS FROM THE CHERNOBYL DISASTER

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2.1 INTRODUCTION

A very large area of the Republic of Belarus was exposed to radioactive contamination as a result of the accident at the Chernobyl nuclear power plant on April 26, 1986. Following the accident, the reactor which contained 190.2 t of nuclear fuel, released 1.85 x 10^{18} Bq of radiation into the environment, of which approximately 70% fell within the territory of Belarus. The released composition included a large amount of radionuclides of iodine, cesium, cerium, barium, strontium, plutonium, etc.

As shown in Figure 2.1, the prevailing winds carried contamination as much as 300 km away from the plant to the north and to the west in Belarus. A total of 46,450 km^2 (23% of the country) was contaminated, an area that was originally inhabited by 2.1 million people (over 20% of the population). The contamination has affected 1.8 million hectares of agricultural lands, of which 264,000 hectares have been excluded from use. The territory of the Polessie State Radioecological Reserve (131,400 hectares) has been turned into a "plutonium reservation" and is practically excluded from use because of the high levels of contamination.

Although Belarus has no nuclear power plant producing spent fuel, it has some very serious problems in developing appropriate remediation measures to collect and isolate the many kinds of contamination that are spread throughout the affected areas (Fig. 2.1).

Figure 2.1. Areas in the Republic of Belarus contaminated by radionuclides from Chernobyl.

2.2 DECONTAMINATION WASTE AND BURIAL METHODS

The Belarus territory has received a great amount of radionuclide fallout over a significant percentage of its arable land, and although the contaminated industrial and other objects are numerous and widespread, the activity of the decontamination products (DP) is rather low, of the order of 10^{-6} to 10^{-8} Ci/kg. In view of the specific character of the releases from Chernobyl and considering the DP and its associated materials as one category of radioactive waste, the National Committee for Radioactivity Protection (NCRP) of Belarus has adopted the levels of specific activity as shown in Table 2.1.

Table 2.1. Specific activities of radionuclides.

Radionuclide	Specific Activity Bq/kg
Cesium-137	8.5×10^3
Strontium-90	3.7×10^4
Other beta-emitters	7.4×10^4
Plutonium and other transuranic elements	3.7×10^2

An effective program of decontamination outside the 30-km zone (plutonium contamination zone around Chernobyl) was carried out in the period 1986-1988. As a result, 12 million m^2 of exterior surfaces of buildings and other constructions were cleaned, 13.3 million m^3 of contaminated ground were removed and buried, and 7570 old buildings were demolished and buried. These DP were stored far from populated areas, mainly in unprepared sites (such as flat areas, gullies, sinks, old quarries, etc.). About 980,000 m^3 of DP and other contaminated refuse are located in 69 partially equipped repositories and five makeshift burial grounds. All these sites are constructed in zones where the ^{137}Cs activity of the soils is 15 Ci/km^2 and above; the gamma radiation exposure dose rates in the storage sites vary from 0.03 to 0.23 mR/h. The cumulative activities total 1.9 TBq for ^{137}Cs, 0.14 TBq for ^{90}Sr, and 2.6 TBq for ^{239,240}Pu.

Most of these burials are ecologically unsuitable, and the radioactive wastes placed there need to be reburied. A concept and strategy for reburial have been developed, and sites with a suitable geology and geomorphology for constructing repositories have been selected. When selecting sites for DP repositories, the following factors have been considered:

- Location of testing grounds in evacuation zones and obligatory evacuation of people from areas where the activity exceeds 40 Ci/km^2;
- Correspondence of the geologic and hydrogeologic conditions of the locality to the requirements for safe radioactive waste isolation;
- Hydrogeologic conditions of the territory with reference to flooding and seasonal water levels, areas subjected to flooding, etc.;
- Structure of atmospheric circulation and seasonal wind patterns in relatively uncontaminated areas and populated areas; and
- DP transportation routes, quality of major and secondary highway routes (near cities), dust generation during loading, transportation, etc.

A model to forecast the migration of radionuclides away from waste repositories has been developed for recommended objects with given migration parameters.

2.3 PROGRAMME FOR DEEP DISPOSAL OF RADIOACTIVE WASTE

In the context of the government programme to construct nuclear power plants in the future, sites for their location have been selected, and a preliminary investigation of geological conditions has been carried out to locate potential sites for repositories for high- and medium-level radioactive wastes. Three rock types have been recognized as potentially being able to provide satisfactory conditions for radioactive waste disposal: (a) the crystalline basement, (b) salt beds, and (c) thick deposits of monomineral clays.

As shown on Figure 2.2, the territory of Belarus has been subdivided into zones showing various conditions to be considered in selecting sites:

- Areas marked Zone I are where the depth to basement ranges from 0.0 to 0.4 km and where conditions are considered very promising for engineering and constructing repositories in basement crystalline rocks;
- Areas marked Zone II are where the depth to basement ranges from 0.4 to 0.6 km and where conditions do not show much promise for repositories in basement crystalline rocks;
- Areas marked Zone III are where the depth to basement ranges from 0.6 to 1.5 km and where conditions

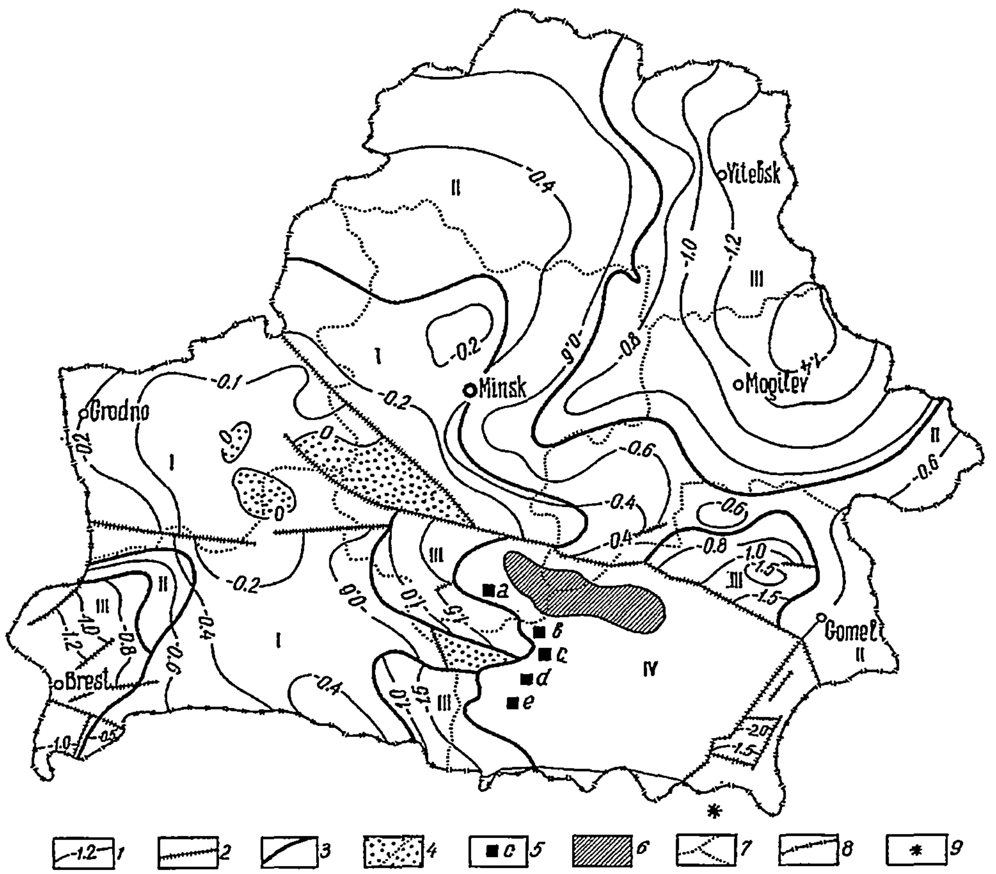

Figure 2.2. Geological division of Belarus into zones in accordance with conditions for medium- and high-level radioactive waste isolation. Legend: 1 - isolines showing depth to crystalline basement (km); 2 - tectonic faults penetrating through the sedimentary cover; 3 - boundaries of zones with different conditions and potential for waste isolation. Symbols for regions with geologic structures considered as primary candidates for repositories: 4 - crystalline basement of Bobovnya, Slonin and Mikashevichy uplifts; 5 - salt diapir domes of: a - Novaya Dubrova, b - Zarechie, c - Kopatkevichy, d - Konkovichy, e - Shestovichy; and 6 - palygorskite clay beds of Pripyat Graben. Other symbols: 7 - boundaries of districts; 8 - frontier of Belarus; and 9 - Chernobyl nuclear power plant.

show very little promise for radioactive waste repositories; and

- The area marked Zone IV is the Pripyat Graben that is considered to have highly promising geological structures for waste repositories in: (a) stratified saliferous formations of Middle and Upper Devonian, (b) domed salt diapirs of the Upper Salt strata, as well as in (c) palygorskite clay beds of Upper Devonian strata.

Crystalline rocks are exposed or overlain by Anthropogene deposits of limited thickness within the Central-Belarussian massif, Ukrainian shield and Mikashevichy 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 blastomilonite show the greatest promise for a site 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: (a) Upper Frasian (up to 1100 m thick), and (b) Upper Famennian (up to 3000 m thick zones of diaparisim) separated by terrigenous-carbonate strata. Salt formations occur in zones of diapirism in a depth range of 300 to 400 m and have the potential as sites for waste disposal (Novaya Dubrova, Zarechye and Kopatkevich uplifts on the northwest, and Konkovichy

and Shestovichy uplifts, on the southwest side of the graben).

Large deposits of palygorskite clays are found in the Pripyat Graben at depths of 80 to 120 m, and with thicknesses of as much as 140 to 150 m, they are of particular interest as potential sites for radioactive waste repositories.

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CHAPTER 3

GEOLOGICAL RADWASTE DISPOSAL IN BELGIUM RESEARCH PROGRAMME, REVIEW AND OBJECTIVES

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Executive Summary. In the early seventies, SCK·CEN launched a research programme on the geological disposal of long-lived radioactive waste. Investigations were concentrated on argillaceous formations because of their abundance at various depths in some parts of Belgium, their expected favourable properties and the virtual absence of salt and/or unfracturated hard rock formations in the country. Among the argillaceous formations, the Tertiary "Boom clay" layer was tentatively selected for site characterization studies. In this first phase, the Research and Development programme consisted of laboratory studies on clay core samples and observations on the geology and hydrology of the area. The initial results were very promising with regard to sorption properties of the potential host rock, its chemical stability, permeability, etc.

Early in the programme (1980), an underground laboratory (HADES) was constructed in the clay at the Mol site; a tunnel, 35 m long and 3.5 m in diameter provides a large number of access possibilities to the clay formation, at a depth of 223 m. The first objectives were related to the technical feasibility of gallery excavations in plastic clay under high lithostatic pressures and to the evaluation of the compatibility of vitrified HLW and candidate container materials with the clay, in close to real conditions. In 1988, an additional 60 m long drift was built for integrated tests, e.g. the impact of temperature and radiation. In the meantime, much experience has been gained on sampling of fluids and solids, in the characterization of clay and waste forms; an important scientific data base is now available at SCK·CEN, specifically for vitrified waste and "Boom clay."

This formation - and site specific approach - allowed SCK·CEN to collect, in a short period of time, valuable results and input data for required modelling work. Some preliminary observations could be made:

- using conventional civil engineering technology, it is technically feasible to drive horizontal galleries in plastic clay, at a depth of more than 200 m, with an internal diameter of at least 3.5m, without special ground conditioning, such as freezing;
- using available experimental data about permeability and retention properties of the host rock (Boom clay), preliminary safety assessments were carried out for many scenarios from natural evolution to important disruption. For the reference mix of reprocessing waste (HLW, MLW, hulls) from a 40 y nuclear programme in Belgium, the highest calculated dose to population, for a normal evolution scenario and water well pathway, is of the order of $3x10^{-7}$ Sv/a, which can be compared with a dose constraint of 10^{-4} Sv/a, in use in several countries; these doses would occur after a period of about 60,000 years.
- stabilizing factors in the Boom clay are its cationic exchange capacity, the strong reducing conditions due to finely dispersed pyrite and humic materials intimately associated with the clay minerals and a slightly alkaline environment caused by the occurrence of carbonates; and
- the methodology for risk assessment was established and tested in the framework of an international benchmark exercise in cooperation with the European Commission and the Nuclear Energy Agency of the OECD.

Observations and earlier assessments also illustrate remaining areas of uncertainty which require complementary R&D or new studies and demonstrations. Furthermore, the validity of present evaluations and preliminary conclusions, mainly in view of the very long term safety, will benefit from an iterated validation of

models and even comparisons with studies on natural analogues.

3.1 INTRODUCTION

In the present context, the R&D programme of SCK•CEN on radioactive waste management is focused on the characterization of conditioned waste and its underground disposal in geological formations.

Although in the past (1960 to ~1985), SCK•CEN has run several projects on waste treatment and conditioning (mainly low-level and alpha-bearing waste), very little is still being pursued in this area, given the commercial availability of industrial processes and the fact that the waste treatment and conditioning has been entrusted to a separate company, BELGOPROCESS. However, this trend may have to be adjusted some time in the future for reasons of optimization and the development of new waste types resulting from e.g decommissioning and/or site restoration programmes.

With regard to characterization of conditioned waste, the programme aims at quality control (verification of compliance with contractual quality criteria), characterization of the "source term," which includes studies of corrosion phenomena and compatibility of the conditioned waste form, and its package with the intended geological environment. Attention also has to be paid to "acceptance criteria" since the total radiological capacity of the repository is obviously limited, and non-radiological factors (e.g temperature and residual thermal capacity of the waste and potentially interfering components) may enhance the mobility of radionuclides in the repository formation. Furthermore, the occurrence of chemo-toxic components in conditioned waste or packages deserves a minimum of attention.

With regard to geological disposal, the R&D programme, initiated by SCK·CEN, deals mainly with the disposal of vitrified high-level waste from reprocessing and other alpha-bearing wastes. Other types of waste are to be considered in the future such as, for example, non reprocessed spent fuel or spent MOX fuel since it is clear that, even if reprocessing remains the reference scenario, at least some spent fuel is not suited for industrial reprocessing.

The R&D programme on disposal proceeds stepwise and cautiously since options have to remain flexible until adequate scientific and technical ground is available to support decisions.

In the exploration of geological formations (1974) for the implantation of a repository in Belgium and after consultation with the National Geological Survey, preference was expressed for a deep clay formation (called "Boom clay") at and around the site of Mol. This early preference, resulted from a number of facts, e.g. the absence in Belgium of any salt formations, the highly fractured nature of most hard rock formations, the natural retention capacity of clay, and the fact that major nuclear facilities were already installed in the Mol area. Consequently, R&D and assessment studies became mostly site-specific or at least formation-specific, without anticipating the conclusions of a more detailed site selection procedure.

The Boom clay belongs to the Rupelian formation and is underlain by the Berg sands (Rupelian aquifer) and overlain by the Neogene aquifer. These two water bearing formations are not considered as barriers, although they could have good sorption properties for the radionuclides. Rather early in the programme (1980), an underground laboratory was constructed in the clay at the Mol site at a depth of 223 m. The underground laboratory was intended to evaluate the technical feasibility of such a construction and to become an *in-situ* facility for performing tests in close to real conditions within the High Activity Disposal Experimental Site (HADES) project.

The current R&D programme is focused on site characterization (mechanical, physicochemical, and hydrogeologic properties) and performance assessments. Understanding the basic phenomena which control the retention and/or mobility of radionuclides in the clay received high priority. A first interim safety report (SAFIR) was produced for the Authorities in 1989. The main conclusion was the absence of counter-indications against the principle of disposal in clay. The next preliminary safety report is to be submitted around 1998 and is meant to be an important milestone in the selection, or decision making, process.

For the near future, investigations on basic phenomena of physical chemistry and geohydrology are to be continued; in addition, a number of areas may require much more study and investigation such as:

- impact of radiation and heat on retention/mobility of radionuclides in clay; these near-field effects may affect important factors such as the length of cooling time in surface storage facilities prior to transfer into the repository and/or the conceptual design of the

repository;

- influence or significance of non-radioactive components in the anaerobic environment of the repository (e.g decomposition products of organic materials). In case of adverse effects, they might lead to the consideration of alternative conditioning processes, depending on the acceptance criteria for packaged waste; and
- other issues are to be considered as well, e.g position with regard to retrievability, reduction of uncertainties with regard to some major characteristics and to the time scales applicable to geological disposal.

The above programme items require a major extension of the present underground facility to perform larger scale integrated *in-situ* demonstration tests. This important decision has been recently taken and after 1998 will lead to doubling the present capacity.

3.2 THE HADES PROGRAMME AT SCK·CEN

The R&D programme on geological disposal was initiated in 1974 by SCK•CEN and received scientific and financial support from the European Commission from the very beginning. Since 1985, it has been heavily supported by the National Radioactive Waste Agency (ONDRAF/NIRAS), which is in charge of the implementation of the disposal programme.

With a total installed nuclear power of 5.5 GWe and a 40-year operational period for the Belgian nuclear power plants, the volume of waste to be produced of low, medium and high activity can be estimated to be 150,000 m^3, 39,000 m^3 and 4000 m^3, respectively. These volumes do not yet include the decommissioning waste.

The first category of waste contains small quantities of short- or medium-lived radionuclides (half-life not exceeding thirty years); the current option is to dispose this waste at the surface; shallow land burial is considered in Belgium as the reference solution but will not be examined in the present document. Earlier disposal studies resulted in a report submitted by the waste management authority (NIRAS/ONDRAF) in 1993 to a national advisory scientific commission; they are being updated and extended according to the recommendations of this commission.

Two other types of waste are considered in Belgium for underground disposal:

- alpha-contaminated waste (concrete or bitumen matrix) containing significant quantities of medium-lived radionuclides (half-life exceeding 30 years); it results from specific fuel cycle operations (spent fuel reprocessing, fuel assembly manufacturing) and does not generate heat; and
- vitrified heat-emitting waste (glass matrix) containing a mixture of short- or medium-lived radionuclides in high concentrations, and long-lived radionuclides which are usually alpha emitters. It generates a significant heat power, owing to the fission products content.

Table 3.1 presents some more details on these waste types and volumes.

Table 3.1. Data on radioactive waste types in Belgium.

Waste Type	Matrix	Volume Total (m^3)
HEAT EMITTING WASTE		
Cogema high-level waste	glass	784
Cogema cladding waste	concrete	2,860
Eurochemic high-level waste	glass	265
	Total	3,909
ALPHA WASTE & OTHER		
Technological alpha waste	cement	8,419
MLW concentrates	concrete	20,430
Reactor operating waste	concrete	2,724
Coprecipitation sludge	bitumen	3,310
Eurochemic MLW concentrates	bitumen	2,765
Mox production waste	concrete	1,353
	Total	39,001

It is assumed in the present reference scenario that the operation of a repository for underground waste disposal could start in 2035 to last until 2070/2080 (closure phase). This requires, for vitrified waste, a temporary surface storage of about 50 years allowing the heat output to be significantly reduced. This limited heat output in the repository is intended to produce temperatures not exceeding 100° C in the host clay at the container/clay interface and is one of the basic assumptions for the definition of the R&D programme and associated processes to be considered.

3.2.1 Historical Review of Hades Program

Early studies on potential host formations for geological disposal have indicated that, in Belgium, clay and shale

layers can be taken seriously into consideration. One of the potential formations which was identified is the Boom clay at the Mol-Dessel site situated in northeastern Belgium, where the SCK•CEN and other nuclear facilities are located. The following guidelines for the provisional selection (for experimental purposes) of the Boom clay layer were considered: the anticipated future stability of this 30 million year old formation as well as its depth, thickness, relative homogeneity and sorption/retention capacity (Heremans and Baetslé, 1978).

The research programme was started about 20 years ago by SCK•CEN, which concentrated its activities on verification of basic phenomena, safety and technical feasibility. Several partners from Europe and abroad joined the programme during the second decade and, through their own competence, contributed to the integration of different scientific and technological skills and approaches (Bonne and Collard, 1992).

3.2.2 Approach and Major Achievements During First Two Decades of Programme.

Beside the selection of the formation (and experimental site), the first decade (1974-1984) was dedicated to verify whether the characteristics and properties of the Boom clay layer were really promising for hosting a geological repository.

The first drillings were performed on the SCK•CEN site from 1975 onwards; and analyses of core samples were made to determine lithological, chemical, mineralogical, ion exchange and geomechanical properties of Boom clay and, in some cases, of the surrounding strata. Geohydrological studies were undertaken; a preliminary repository design and a probabilistic risk assessment methodology were developed. Simultaneously, a catalogue of all potential formations occurring in the country was established. From these studies, it was concluded that the Boom clay satisfied the expectations; this plastic material has good sorption capacities and provides very low permeability and low but sufficient heat conductance. It is sufficiently thick and homogeneous and it is chemically and mineralogically stable. Preliminary experiments on corrosion/heat transfer were performed in parallel from an open clay pit in Terhaegen, near Antwerp.

The next step was to demonstrate that it was technically feasible to build a repository in such a plastic material and to strengthen our capabilities in the area of safety assessment; in the latter, a collaboration with JRC Ispra was an important feature (D'Alessandro and Bonne, 1981).

An underground research laboratory (URL) was built according to specifications which were based on very pessimistic hypotheses (afterwards found to be unrealistic) about the plasticity and perviousness of the Boom clay. For example, the freezing technique was not only used in sands to sink the shaft through these water-bearing formations but also in clay to lower the anticipated convergence rate.

The approach became fully site and formation specific; the site of Mol was placed in a larger geohydrological context, extending the regional observation network on a 2,000 km^2 area around Mol. Both clay and surrounding geologies were intensively investigated from various perspectives (geochemical, geomechanical, hydrological and geophysical). Due to an increasing amount of available information, the Boom clay became progressively a reference case.

For the critical issue of geomechanics in clay, collaboration was started with ANDRA, the French waste management agency; the digging of a small experimental shaft and gallery in non-frozen clay demonstrated the possibility of excavating and tunneling in the plastic Boom clay at such a depth.

During the second decade (1984-1994), the URL became fully operational; several experimental set-ups were installed in clay for studies on corrosion, hydraulics, migration and geomechanics. A new and challenging field of demonstration was opened with the CEC programme on demonstration and pilot facilities. The HADES URL was extended by tunneling a test drift for demonstration purposes as illustrated in Figure 3.1.

This facility, now called URF (Underground Research Facility,) consists of 100 metres of galleries lined with cast-iron, concrete or steel ribs, providing direct access to the clay mass. In particular, the test drift offered the opportunity to perform large scale integrated tests under conditions close to those expected in and around a final repository.

Validation exercises for the modelling of different processes were launched and extensive performance assessment exercises were carried out, as the "Performance Assessment of Geological Isolation Systems" (PAGIS) and the "Performance Assessment of Confinements for Medium-level and Alpha waste,"

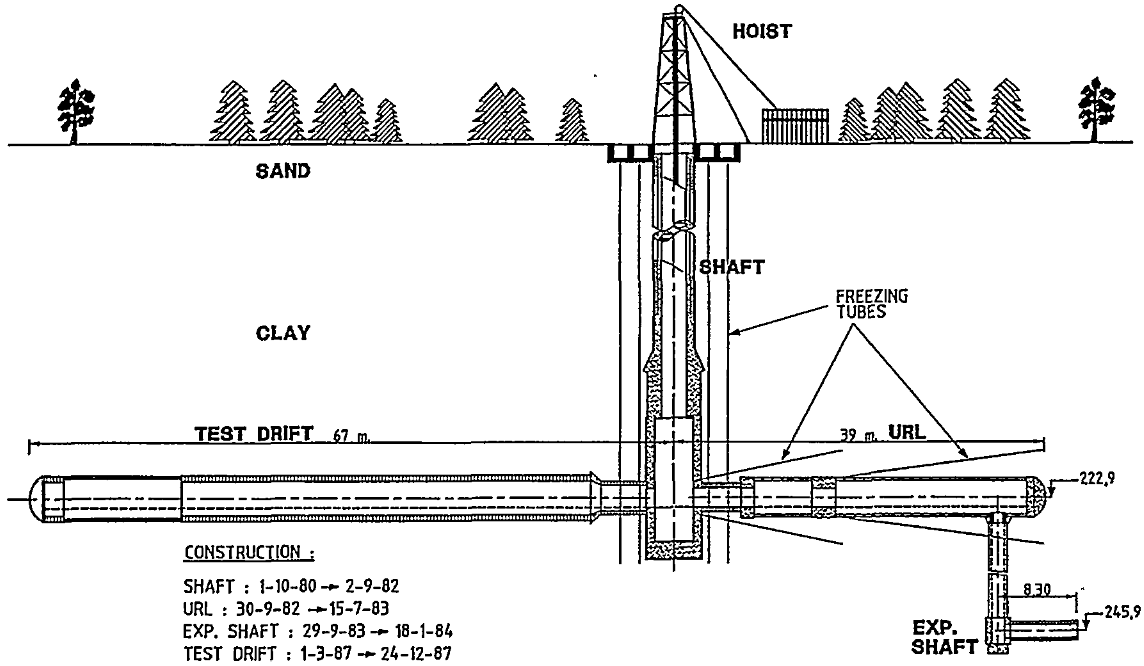

Figure 3.1. The underground research facility (URF) HADES.

(PACOMA) within the framework of the E.C. Both studies were updated, and results were published in the "Updating 1990" (Marivoet, 1992). A methodology was elaborated with the involvement and consensus of a large number of scientists in different member states, active in the various fields of this multidisciplinary activity; the broad contribution from the latter activity had a very positive impact on confidence building.

In 1990, an international commission of experts, formed by the Belgian Secretary of State for Energy, evaluated a Safety Assessment and Feasibility Interim Report (SAFIR) on the geological disposal of waste prepared by NIRAS/ONDRAF (1989); the results of the research carried out by SCK•CEN on final disposal in Boom clay were the main contribution to this report.

These successive actions confirmed the choice of the clay option in Belgium and the continuation of the site characterization programme on the Boom clay layer at the Mol-Dessel site as a potential host formation essentially for long-lived waste. They also define the R&D priorities developed in the current programmes on the different components of the disposal system. As a result of these continuing interactions, tangible progress has been made through improvements in the SCK's characterization programme, e.g. more recently regarding the study of the performance of the engineered barriers in the near-field. Present and future developments are detailed in the next chapters.

3.3 Objectives and Basic Assumptions of HADES Programme

3.3.1 Objectives

Safe disposal of radioactive waste is one of the key issues in the environmental concerns of the nuclear industry. In particular, the disposal of long-lived waste, specifically spent fuel and vitrified high-level wastes, is important. Disposal in deep geological formations is at present the most promising option, which requires one to meet the following objectives:

- provide reliable data on characteristics of waste and the geological environment;
- demonstrate the feasibility of disposal concepts e.g. in underground laboratories; and
- assess the safety and acceptability of disposal systems by developing and applying validated methods to the modelling of the various phenomena which control the release and migration of radionuclides from the repository into the biosphere.

3.3.2 Methodology

In order to assess the performance of individual components of the disposal system as well as that of the inte-

grated system, the current R&D programme covers mainly research on basic phenomena, demonstration tests and safety studies.

The geological disposal of long-lived waste is essentially based on a "multi-barrier" concept; several "engineered" barriers (overpack, backfill, gallery lining) are installed between the waste matrix itself (primary containment) and the host rock. The purpose is to delay (e.g. by 500 to 1000 years) the release of activity from the repository structure into the geological environment. The latter time period corresponds approximately to the thermal phase of the repository. Consequently, several coupled effects with temperature can be avoided in the host clay, provided the required performance of these engineered barriers can be guaranteed.

Phenomena relevant to repository performance are generally subdivided into those occurring in the near-field and those occurring in the far-field (geosphere). By near-field, one generally considers the engineered repository structure (including the waste packages and engineered barriers) and those parts of the surrounding rock whose characteristics have been, or could be, altered by the repository or its content (excavation, temperature, radiation).

Any assessment of the performance of geological disposal refers to a conceptual design of the repository and, in particular, to the characteristics of the near field. Information concerning the Belgian reference concept is described in a separate section (see Section 3.3.4).

3.3.3 Performance of Near-Field

An essential distinction for disposal in clay, with regard to disposal in granite, consists of the requirements and objectives of the near-field.

In hard rock, the components of the multi-barrier containment and essentially the backfill material are to be considered as primary barriers which ensure the isolation of the system for the time periods required. In this context, the backfill must preserve alkaline conditions during a very long time period in order to prevent dissolution of key radionuclides and to provide sorption capacity and controlled porosity and permeability. The host geological formation is supposed to provide structural stability, a predictable groundwater flow, and a suitable long pathway and travel time for radionuclides to reach the biosphere.

For argillaceous formations and particularly for the Boom clay formation, the host "rock" is the primary natural barrier against radionuclide migration; it provides good sorption capacities, very low permeability, and given the composition of the clay, chemically reducing conditions. The backfill has to provide geomechanical and geochemical stability and compatibility.

3.3.4 Repository Concepts in Belgium.

We are considering for geological disposal the long-lived medium level waste (MLW), heat producing vitrified high-level waste (HLW) and spent fuel (SF), but for the latter category, no detailed concept has yet been worked out.

Since 1978, different repository concepts have been considered for high-level and long lived radioactive waste in the Boom clay formation at Mol. The reference concept, now considered by NIRAS/ONDRAF, is illustrated in Figure 3.2. Heat producing high-level and long-lived, medium-level waste would be emplaced in separate disposal galleries. Approximate dimensions (expressed in terms of the 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. The length of the disposal galleries would be 800 m (4 sections of 200 m). Concrete would be used for lining the shaft and galleries. The lining is designed to ensure mechanical stability during the repository's operational phase.

For the non-heat producing long-lived MLW, the galleries would have an inner diameter of 3.5 m and the section would be filled with MLW canisters, the voids being backfilled, possibly with concrete (see Fig. 3.2).

The design of the HLW disposal galleries is more advanced (see Fig. 3.3) and will, during the next 10 years, be the main subject of the PRACLAY project. The latter requires a substantial extension of the present URF; it is managed by an Economic Interest Group (E.I.G.) where SCK•CEN and NIRAS/ONDRAF are represented. The PRACLAY project is a thermo-mechanical experiment intended to simulate a full scale disposal gallery for heat-emitting waste and to quantify the extent of the disturbance induced in the surrounding clay.

According to this concept, the HLW canisters, simulated by heating elements, are assumed to be placed in long thin metallic tubes in the centre of the disposal galleries. These tubes are designed to remain intact during the thermal transient of the heat producing waste. The thick-

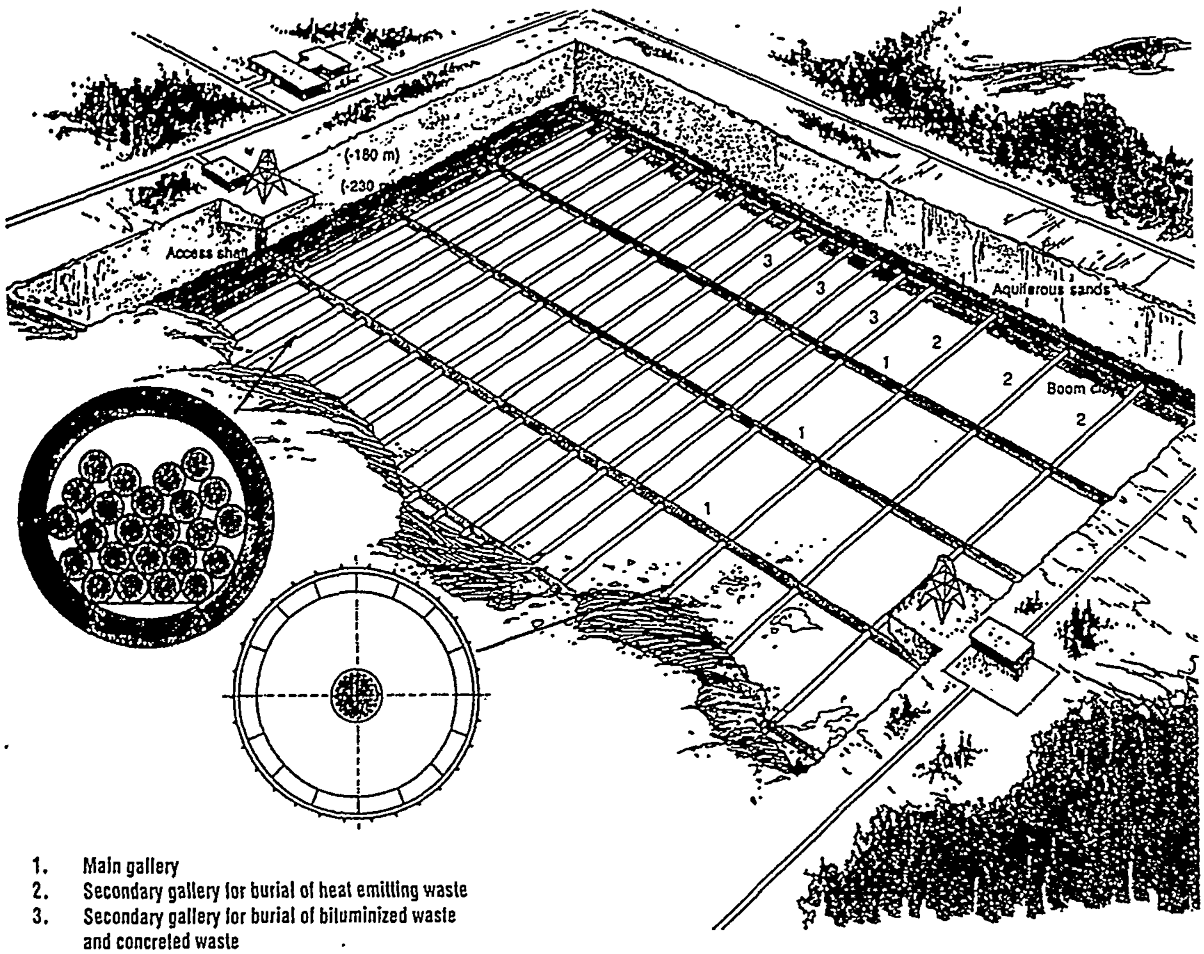

Figure 3.2. Present Belgian reference disposal concept (NIRAS/ONDRAF).

ness of this "overpack" will have to be sufficient to guarantee its role as a reliable barrier during the thermal phase of the disposal, and sufficiently low to limit the potential generation of gas, through corrosion, in the anaerobic phase. The void between the metallic tube and the concrete gallery lining would, according to the present concept of NIRAS/ONDRAF, be backfilled with pre-compacted calcium bentonite blocks.

The option of retrievability or "long term storage" is examined abroad in some disposal programmes (France, the Netherlands, etc.), possibly in connection with later "advanced reprocessing." However, in Belgium, easy retrievability is not considered, up to now, as a requirement in the different disposal concepts. It is assumed that, with the necessary capital and effort, the retrieval of the waste will always be possible during a limited time period, generally considered to be in the range of 50 to 100 years. Only preliminary studies related to the impact of retrievability on performance assessment might be launched in the near-future by SCK•CEN, providing European partners join the effort.

For the disposal of spent fuel, conceptual studies are in a preliminary stage. The main differences compared to the heat producing HLW are the longer thermal transient to be considered, the content of the total inventory of plutonium and the larger dimensions of the packages. Given the limited dimensions of the clay disposal infrastructure, a size reduction and a form of preconditioning of spent fuel might be required.

3.4 PROGRESS OF CURRENT RESEARCH AND STUDIES

An understanding of the migration processes in the host rock, together with the hydrogeological context of the host and surrounding formations are of prime importance for the long term safety of a reference geological disposal concept. Furthermore, one has to take into

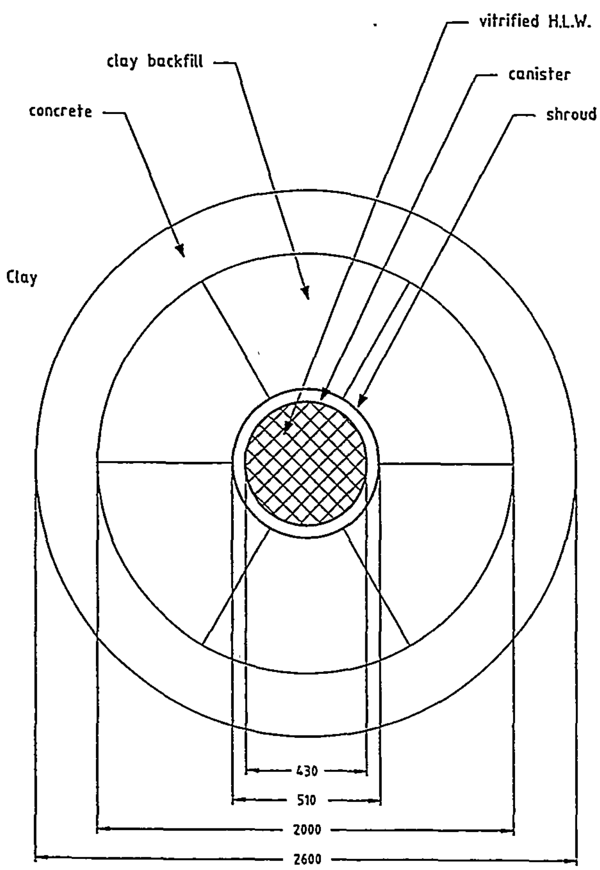

Figure 3.3. Detail of the Belgian reference concept for HLW (NIRAS/ONDRAF).

account the technical feasibility of the different components of the system, from the constructional and operational aspects to the assessment of the integrity of the volume of clay surrounding the disposal galleries. All the phenomena and processes involved in and around a deep geological repository (decompression, radiolysis, heat emission, gas production, etc.) and their evolution with time are dependent upon the waste types and the concept of the disposal infrastructure.

3.4.1 Research on Basic Phenomena

The safety of a repository for radioactive waste is conditioned by the long term behaviour of the various elements which constitute the engineered and natural barriers. Research on the basic phenomena expected to take place in the disposal system is therefore essential, in addition to the technical demonstration tests. Some of the research topics listed below are carried out in surface laboratories, others in the underground facility (URF). This research work is intended to better understand the phenomena, which control the release of radionuclides from the waste packages, and their migration through the various successive barriers to the environment, in order to model these phenomena and validate the models. We are considering the following succession of research issues:

- characterization of waste packages and their compatibility with clay;
- geomechanical/geochemical behaviour of clay-based materials
- gas generation and transport;
- thermal, radiation and chemical effects; and
- radionuclide migration.

3.4.1.1 Characterization of Waste Packages and their Compatibility with Clay

Waste package is a general term used to indicate the conditioned waste in its container; the conditioned waste can be homogenous (vitrified waste, bitumenized sludge) or heterogeneous (e.g supercompacted waste in a cement matrix). "Container" is here considered as a generic term covering both the canister, into which the conditioned waste is loaded, and the overpack which has to provide isolation of the waste for relatively long time periods.

Background

Studies on waste packages performed at the SCK•CEN over the past 15 years have dealt with:

1. characterization of conditioned waste forms, as part of the national QA/QC programme (qualification of the processes, verification - Van Iseghem et al., 1990); and
2. compatibility with the anticipated disposal environment. These studies involved both laboratory and *in situ* "corrosion" tests. The efforts so far mainly focused on glass and bitumen waste forms, C-steel container material and, increasingly, on cement waste forms and stainless steel. These studies were aimed at characterizing the source term and at providing information for the future elaboration of acceptance criteria for the disposal of waste in the host formation.

Results

High-level waste glass (Van Iseghem, 1994; Lemmens et al., 1993). The results of *in situ* experiments are in agreement with those of laboratory tests which show

that glass is an efficient first barrier for the radionuclides; dissolution rates of less than 0.1 μm per year were recorded at an ambient clay temperature (16° C) in direct contact with Boom clay. However, the dissolution of glass is strongly enhanced by increasing the temperature and in the presence of clay, which acts as a sink for many glass constituents such as Si, rare earths and actinides. The long-lived actinides (Pu, Am, Np) in general leach slower than the bulk glass, and for more than 90%, the fraction leached is sorbed onto the clay. These data are further used in impact assessment studies reported under Section 3.4.3.

Bitumenized reprocessing waste (Berghman et al., 1990). A high leaching rate of NaNO3 was noticed for Eurobitum, a reference Belgian bitumen, containing an average 35% of reprocessed sludge. By extrapolation, we can predict that a full size drum would be depleted in NaNO3 (25%) within a few thousand years leading to the release of important amounts of nitrates into the near field. Lower leaching rates were noted for the actinides (Pu, Am); their leaching appears to be controlled by solubility and is not enhanced by lithostatic pressures.

Container (Cornelis and Van Iseghem, 1994). An average corrosion rate of 5 μm per year was inferred from *in situ* tests on carbon steel lasting up to 7.5 years; pitting processes with an average growth rate of about 20 μm per year are dominating. Corrosion proceeds mainly by pitting attack in the aerobic phase and congruent corrosion would dominate in the anaerobic phase. Stainless steel is becoming the candidate overpack material.

Future actions

Additional or corrective actions are required due to not following specifications, the absence of reference product specifications or representative samples (QA/QC programme).

Development, application and standardization of quality checking for waste packages must identify R&D requirements and coordinate the development of new test methods.

For HLW glass, further investigation of the secondary phase formation is needed since it could affect the glass dissolution mechanisms in the long-term.

Investigations on the soluble concentrations of the actinides in clay media in the presence of bitumen degradation products, on the ageing behaviour of bitumenized waste and on the microbiological degradation of bitumen due to nitrate and sulphate reducing bacteria are scheduled. Microbiological activity could lead to the production of methane and complexing agents which need to be further studied as gas and chemical effects.

A new laboratory corrosion programme (steel) will be run to provide the necessary information in both aerobic and anaerobic environments.

Chemical stability of Synroc, a titanate ceramic proposed as an alternative to glass, is under evaluation for Boom clay disposal environments.

New programmes on other waste forms, such as cemented reprocessing sludge (BR2/Dounreay), cellulose containing MOX waste, and spent fuel have to be started.

3.4.1.2 Geomechanical/Geochemical Behaviour of Clay-Based Materials

The mechanical and chemical stability of the near field provide a set of controlling factors for groundwater movement, radionuclide transport and heat dissipation through the interface between near and far field. The excavation disturbed zone in clay and the backfill/sealing material might substantially affect the subsequent migration behaviour of radionuclides.

Background

Excavation disturbed zone. The measurements of pore water pressure and stress distribution around boreholes and galleries during and after their construction (Neerdael and De Bruyn, 1988) have shown that the extent of the disturbance depends on the excavation diameter, the excavation speed and the oversizing with respect to the lining diameter; the magnitude of the disturbance drops significantly with distance and decreases slowly with time thanks to the plasticity of clay.

Backfill and sealing material. In case of a repository in clay, a clay-based material is chosen as backfill material for the same reasons as put forward when dealing with the geological barrier. Boom clay has been characterized with a view to reusing the excavated clay spoils for backfill, since oxidation (pyrite) can be prevented easily by fast drying. The following results have been found for highly compacted Boom clay with regard to compacted bentonite: (a) slightly higher but sufficiently low hydraulic conductivity, (b) lower swelling but still convenient regarding the lithostatic pressures at the Mol

site, (c) better retention properties for some radionuclides, and (d) steam effect less dramatic than for some bentonites.

When dry compacted clay is used as a backfilling around HLW packages, its behaviour during saturation is complicated by the thermal transient caused by the heat emission from the waste. It must be shown that during this transient period, no (negative) influence on the long term behaviour of the repository can occur due to e.g. shrinkage or collapse of the clay backfill. Therefore, the thermo-hydro-mechanical (THM) and geochemical behaviour of the backfill must be studied. Typically hydro-mechanical and thermo-hydraulical models have to be coupled to describe stress/strain behaviour, moisture migration and heat transfer. The thermo-hydraulic model also has to be coupled to a geochemical code to describe the migration and formation of chemical species.

To avoid the effect that the natural occurring, or human induced, disturbance in one disposal gallery would have on the entire repository, the disposal galleries need to be properly sealed from one another. The shafts must also be properly sealed because they might act as preferential pathways to the biosphere, short-circuiting the Boom clay formation.

The main requirements for a sealing material are: low hydraulic conductivity and long term mechanical and chemical stability. Clay is also the more preferable option in this case. With high density bentonite, the hydraulic conductivity criterion can easily be fulfilled and with a proper composition, the stability requirements can also be fulfilled.

Results

Excavation disturbed zone. Small changes in mechanical stress and pore water pressure have only a minor influence on the migration properties of the Boom clay. Large changes in stress and pore water pressure are limited to the zone close to the gallery or borehole wall.

Backfill and sealing material. A fairly extended database concerning THM properties of unsaturated clays is a prerequisite to enable physical model testing, model calibration and validation. Impressive work has been done on this subject during the last four years in the framework of three research contracts of the present E.C. programme (Volckaert et al., 1994). The results obtained confirm that swelling clays are promising backfill materials.

Future actions

Potential influence of the stress changes on the migration parameters might be studied in combination with temperature, radiation and gas effects.

Quantification of the disturbed zone around a disposal gallery successively submitted to decompression and thermal loading is one of the main objectives of the PRACLAY demonstration test (see Section 3.3.4).

Efforts on the THM behaviour of dense unsaturated clay are continuing for the next five years. A large scale *in situ* demonstration test has been proposed within the next E.C. five year programme.

3.4.1.3 Gas Generation and Transport

In a radioactive waste repository, gas generation may occur due to several phenomena depending on a number of factors (waste types, packages, buffer/backfill, near field host rock). The gas release may increase the rates of flow of potentially contaminated groundwaters to the surface and two-phase flow may entrain radionuclides in solution at the gas-water interface.

Background

The main potential sources of gases in the case of deep geological disposal are methane from degradation of organic materials and hydrogen from anaerobic corrosion of iron. Of these gases, hydrogen is certainly the gas which can be released in potentially the largest amounts.

The objectives of the MEGAS (Modelling and Experiments on GAS migration in repository host rocks) project coordinated by SCK•CEN extend from the understanding of the consequences of gas generation to the validation, using a 3-D *in situ* gas injection experiment, of a gas migration model (TOPAZ-INTERA). Attention has to be paid to gas to ensure that the potential pressure build-up in the near-field of a repository would not lead to fractures in the host medium, which might act as new migration pathways affecting the long term safety of the whole disposal system.

Results

In the surface laboratory, special odometers were built to

saturate clay cores to measure its hydraulic conductivity and to perform gas injection; at the end of the gas flow experiment, the cores were still more than 95% saturated, as confirmed by X-ray tomographic analysis. The formation of preferential pathways and a low breakthrough pressure were observed during the field test. These observations are a consequence of the stress distribution around the test borehole. As explained in section 3.4.1.2 (excavated disturbed zone), the drilling of a borehole significantly reduces the stress close to the borehole wall and particularly the effective stress, i.e., the pressure keeping the clay particles together. As a consequence, it is relatively easy to create gas pathways in the excavation disturbed zone. Further *in situ* tests have shown that these fissures close very soon after the gas injection is stopped (Volckaert, et al., 1995).

Future actions

More experimental data are necessary to quantify the link between hydraulic and geomechanical parameters of Boom clay and further gas migration experiments are needed; the use of a swelling backfill material, which locally increases the lower effective stress distribution, would reduce the concentration of pathways in the excavated disturbed zone. Tests in the surface laboratory on larger clay cores will also contribute to increase confidence in the verification and validation of the model.

3.4.1.4 Thermal, Radiation and Chemical Effects

Radiolysis of water by gamma rays can lead to the production of oxidants which might seriously alter the near field chemistry and retention properties of the clay. The radiation effects need to be combined with heat effects, as they occur during the same period. Chemical species released from the waste might seriously alter the near-field chemistry and retention properties of the clay.

Background

The study of the combined effect of heat and radiation is one of the main objectives of the CERBERUS *in situ* experiment (Noynaert et al., 1992). The CERBERUS experiment simulates the in-floor configuration for the disposal of a COGEMA HLW canister after 50 years cooling. Instruments, or test specimens, have been installed in the periphery to obtain data on water chemistry, radionuclide migration, corrosion, radiolysis, water pressure and temperature distribution.

The thermal transient period will last about 300 to 500 years in the case of HLW and 3,000 to 5,000 years in the case of spent fuel. The formation of steam must be avoided at all times because it would lead to operational problems, enhanced corrosion, and could have a dramatic effect on the hydraulic and mechanical properties of clay backfill and sealing materials. A higher temperature will further increase the solubility of the waste form and radionuclides and might reduce the sorption capacity of the clay and/or increase the diffusion coefficients. These effects have not yet been quantified. The temperature cycle also has implications on the mechanical behaviour of the clay. The combined effect of all these influences on the migration in the near field is very complex to model; currently it can not be done with confidence due to a lack of knowledge.

Chemical effects have not yet been studied in detail; however some potential problems have already been identified. The presence of large quantities of degradable organic material and soluble salts, such as nitrates and sulphates, can have an important effect on the global near-field chemistry, the solubility of the radionuclides, and the sorption properties of the clay. The decomposition of e.g. cellulose could generate important quantities of complexing agents.

Results

Seven years of follow-up measurements in the clay around the CERBERUS experiment show a decrease in pH with about one pH unit and a very slight increase in redox potential so that the clay remained reducing. The presence of thiosulphates, which can seriously increase metal corrosion, and oxalate, which can act as a complexant, was observed (Put et al., 1993). The codes for the calculation of the thermal, hydraulical and radiation field were validated.

Selecting the appropriate concept, cooling time and overpack design allows one to avoid an investigation of high temperature and steam effects.

Future actions

Radiation effects will be further studied in the last phase of the CERBERUS experiment, together with the performance of a potential backfill material (Boom clay based) that has been submitted for more than five years to realistic temperature and radiation fields.

Chemical effects identified need further investigation, surely when elaborating acceptance criteria for geologi-

cal disposal systems. Irreversible complexation of actinides by small organic molecules would strongly increase their transport through the clay. Nitrates affect the redox conditions in the clay while high salt concentrations also influence its sorption properties.

3.4.1.5 Radionuclide Migration

Research on radionuclide migration through the Boom clay has provided data to support the performance assessment of geological waste disposal. Work was concentrated on methodological aspects, codes for flow and transport, theoretical studies, laboratory and field experiments on geochemical transport phenomena to develop and test conceptual and numerical models.

Background

Radionuclide transport in porous and fissured media is controlled by diffusion, advection, retardation (sorption, precipitation) processes and radioactive decay, as described in a general way by the advection-dispersion equation. In a rather stiff material, such as Boom clay, the radionuclide migration is dominated by diffusion. Advection plays a secondary role due to the very low hydraulic conductivity of this medium and the absence of preferential paths for the water (self-closure of fissures due to the clay plasticity).

Boom clay exhibits good sorption properties for cations due to its physico-chemical characteristics under *in situ* conditions. The presence of micro-dispersed pyrite and organic matter intimately associated with the clay mineral surface ensures strong reducing conditions while carbonate (calcite and siderite) maintains a slightly alkaline environment. These physico-chemical conditions are very favourable for the sorption of cations on the Boom clay surface.

From the performance assessment studies, three groups of radionuclides have been selected for further experiments in the present migration program:

- critical radionuclides: ^{14}C,^{99}Tc,^{129}I,^{135}Cs and ^{237}Np;
- less critical radionuclides: ^{79}Se, ^{93}Zr, ^{107}Pd, U-, Pu-, Am-, Cm-isotopes; and
- non-critical radionuclides used for the study of migration mechanisms in Boom clay (Sr^{2+}, Br^{-1}, HTO) are used as chemical analogues of relevant nuclides (Eu^{3+} as an analogue for the actinides).

The migration behaviour of radionuclides in the Boom clay is studied by means of different types of experiments (De Preter et al., 1992):

- fundamental physico-chemical studies of radionuclides in the clay;
- laboratory experiments for the measurement on clay cores of the migration (and hydraulic) parameters (diffusion accessible porosity, retardation factor, diffusion coefficient etc.); and
- large scale *in situ* experiments for model validation and confidence building.

Mathematical models have been developed for the design and interpretation of migration experiments performed both on small scale samples and *in situ* at a larger scale. Different analytical models taking into account the various kind of experiments, the type of source terms and the boundary conditions are presently available to interpret laboratory migration experiments and to determine the migration parameters of the radionuclides. The tridimensional migration model and the values of parameters obtained in the surface laboratory on small clay plugs (3 cm) for HTO (tritiated water) have been validated on the metric scale (2 m) by comparison with model predictions from experimental results with large scale *in situ* injection of tritiated water.

Results and future actions

Running studies confirm that the most critical elements to be considered for long term safety are iodine and fission products.

Laboratory experiments have to be performed under strictly anaerobic conditions. A small exposure to traces of oxygen alters the physico-chemical characteristics of Boom clay; many radionuclides (Tc, U, Se, etc.) are very sensitive to oxidation. In the case of technetium e.g., the strongly sorbed $Tc0^{2+}$ cation is rapidly converted to the non-sorbed $Tc0^{4-}$ anion if the oxidation potential of the interstitial solution increases too much.

Aqueous speciation, the clay surface properties, and the concept of anion exclusion or diffusion accessible porosity are fundamental in the migration of radionuclides.

Organic matter distributed between the solid phase (99.99%) and the interstitial solution (0.01%) plays a dual role in the sorption and the complexation processes of the trivalent actinides and lanthanides in Boom clay. The coating of organic matter at the Boom clay surface increases the surface complexation properties of the solid phase and enhances the retention of trivalent

radionuclides on the clay. On the contrary, small dissolved organic molecules might form soluble complexes with these nuclides increasing their mobility in water.

3.4.2 Demonstration Tests in Underground Research Facility.

The HADES underground research facility is presently the only one installed in a clay layer. The facility offers important opportunities for participation of foreign organisations and contributes to the scientific leadership of SCK•CEN for waste disposal in clay.

Objectives

Investigate and demonstrate the feasibility of disposal concepts for vitrified high level waste, long-lived waste and spent fuel and provide reliable data on the performance of repository barrier components; and allow testing and validation of models under representative conditions.

Research topics

The research carried out at SCK•CEN is related to the following scientific and technical aspects:

- testing and demonstration of disposal concepts;
- backfilling and sealing a repository;
- long term behaviour of repository components;
- ground water flow analysis and radionuclide migration; and
- monitoring the reconsolidation of the disturbed zone around an excavation.

All of these items were already introduced in Section 3.4.1. The corresponding demonstration tests now (or previously) running are briefly described below.

In disposal holes (in-floor concept), for demonstration purposes:

- a large scale integrated experiment simulating the radiation and thermal output of a COGEMA canister after 50 years cooling time (CERBERUS). Different experimental set-ups are installed in the periphery to get data on corrosion, migration, radiolysis and material performance; and
- full scale thermo-hydro-mechanical experiments under similar thermal conditions but according to other designs and instrumentation purposes and considering the use of several potential backfill materials (BACCHUS 1&2, CACTUS 1&2).

At the horizontal (in-gallery or axial concept), for validation purposes:

- large scale thermo-hydro-mechanical experiment (ATLAS), again under similar thermal conditions but without the use of backfill material between the clay wall and the heaters;
- 3-D gas migration test (MEGAS) using in-house designed injection and detection piezometers; and
- 2-D and 3-D radionuclide migration experiments using non-sorbed tracers (tritiated water, etc.)

The results of all these tests confirm Boom clay as a promising host rock for radwaste disposal.

In the framework of the fourth R&D programme on "management and storage of waste" of the E.C., we also developed two experiments for the French waste management authority (ANDRA):

- ARCHIMEDES project, now running for three years, aims at studying the acquisition and regulation of the water chemistry in a clay environment. The work programme consists of the following four main topics: (a) fluid and solid sampling and characterization, (b) *in situ* measurements, (c) microbial interactions; and (d) fluid-rock interaction modelling. Available results clearly show, e.g., that the microbial activity in the far field is at the limit of detection while the near-field exhibits the same microbial activity as a soil; and
- PHEBUS project aims at understanding the phenomenology of hydrical exchanges between ventilated underground structures (operational phase) and the host clay. After support tests and mock-ups in surface laboratories, an *in situ* ventilation experiment has been installed from the URF and has been operational since May 1995.

An important action to be developed for the next ten years is the PRACLAY project. PRACLAY is a demonstration test simulating the thermal output of a 30 m long HLW disposal gallery, 2 m in diameter (see Section 3.3.4). The experiment will be installed from an extension of the existing facilities to be built over the next 3 years. The experiment is planned to last until 2005.

3.4.3 Safety Aspects of Waste Disposal

The tools needed for a preliminary evaluation of the safety of disposal of radioactive waste in Boom clay have been described. Performance assessment is a phased process and the models and data used have to be adapted as results from the R&D programme become

available.

This can be illustrated by the study on regional hydrology, which is above all one of the support research items for performance assessments. In the period 1981-1984, a regional site characterisation programme has been carried out. It consisted mainly of drilling about 20 boreholes located in northeast Belgium and the elaboration of pumping and slug tests to determine transmissivities and hydraulic conductivities in the aquifers. These boreholes have been equipped as piezometers and together with existing piezometers they form a regional piezometric network that consists of 122 observation points. The water levels in the piezometers have been measured monthly since 1984. However, most piezometers are located in the overlying Neogene aquifer. The scarcity of data from the underlying Ruisbroek-Berg aquifer is a considerable drawback for the hydrogeological modelling of the regional aquifer system and urgently needs remedial action to enable improvement of the available models.

Objectives

Develop and apply a comprehensive approach and methodology for evaluating the long term safety of disposal in clay, including radiological as well as non-radiological (chemical-toxic) consequences for the purpose of increasing confidence in building and public acceptance by putting the long term risks into perspective.

Background

A performance assessment of a geological repository for radioactive waste disposal consists of three steps:

1. identification of scenarios that can lead to exposure of man and an estimation of the probability of occurrence of these scenarios;
2. evaluation of the radiological consequences of the identified scenarios; and
3. comparison of the estimated doses and risks using appropriate radiological protection criteria.

Since no criteria or guidelines applicable to radioactive waste repositories have been published in Belgium, the recommendations of the International Commission on Radiological Protection (ICRP) are used as a reference. The ICRP recommends the application of a total dose limit to individuals from all relevant sources. The annual effective dose limits are 1 mSv/a for a member of the public and 20 mSv/a for occupational exposure. These limits are considered by ICRP as the boundary between "unacceptable" and "tolerable". Concerning potential exposures, ICRP recommends that risk limits be of the same order of magnitude as the risks corresponding to the dose limits for routine exposure.
In order to meet the requirements of individual dose or risk limits for all relevant sources, ICRP proposes to determine dose or risk constraints related to sources. In several European countries, e.g. Switzerland, the dose constraint has been taken equal to 1/10 of the annual dose limit for the case of geological repositories. We recommend using the same value in Belgium. This dose constraint of 10^{-4} Sv/a is clearly indicated in the diagrams reproduced in this section. The corresponding risk constraint is of the order of $5x10^{-6}$/a for fatal radiation induced cancers.

A difficult aspect of performance assessment is the time scales that have to be considered. In France, the radiological protection authority requires forecasts of doses based on an explicit uncertainty analysis for the first 10,000 years. For a period of 100,000 years, the impact of expected changes in climate on the repository system have to be taken into account. For very long time scales, "more qualitative assessments have to be made to verify that the release of the radionuclides does not result in an unacceptable individual dose."

Results and future actions

Hydrogeological modelling. A new version of the regional hydrogeological model for the Mol site (Wemaere and Marivoet, 1995) has been achieved by considering the available data and computer techniques and after reviewing all the improvements that could be brought to the model developed in 1985. A detailed geometry is used to reproduce the geological structure in the model, and special attention is given to the fault system in the northeastern part of the modeled area. The calibration of the regional model, which was strongly hindered by the poor characterization of the underlying aquifers, led to a value of $3x10^{-10}$ m/s for the hydraulic conductivity of the Boom clay on a regional scale. This means that the calibrated value is much higher than the values measured *in situ* or on cores taken from the underground laboratory ($3x10^{-12}$ m/s). This discrepancy should be further investigated. The latter value is used for the clay migration model in the performance assessment, because in that model a local value of the hydraulic conductivity for clay is needed.

Additional hydrogeological investigations focusing on the spatial variability of the parameters of the Boom clay and on a better characterization of the aquifers,

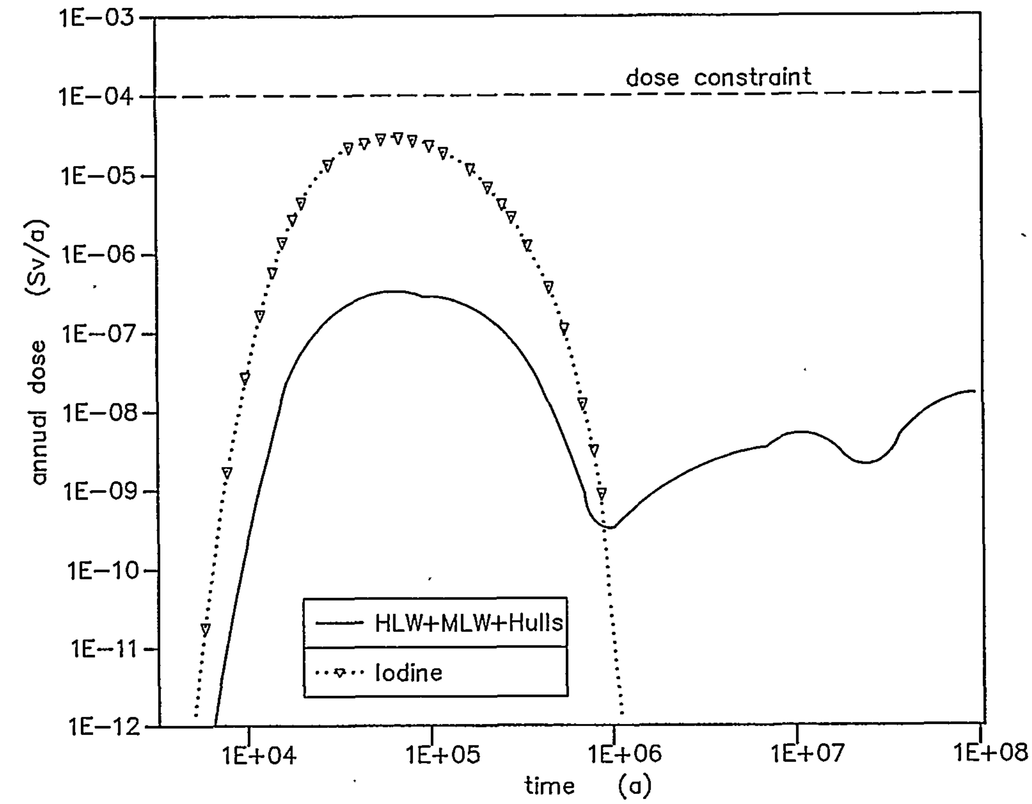

Figure 3.4. Normal evolution scenario - Annual doses via the water well pathway.

especially the deepest ones, underlying the Boom clay are highly recommended. Further, the hydrochemistry should help to understand the behaviour of the aquifer system and constrain its modelling since stable isotope analysis and major trace element analyses provide a way to understand the origins of the water and its evolution.

Performance assessments of reprocessing waste. Following the CEC's PAGIS and PACOMA studies, the "Updating 1990" study (Marivoet, 1992) was performed. A normal evolution scenario, which considers all phenomena that are certain or highly likely to occur, as well as some altered evolution scenarios (deep water well, climatic change, tectonic fault) have been identified in this way. Because the Neogene aquifer is an important drinking water resource, the pumping of drinking and irrigation waters from a well located in that aquifer is considered as a realistic pathway to the biosphere within the normal evolution scenario. The consequences calculated in the case of this scenario for various considered waste types are shown in Figure 3.4. For the reference waste types (HLW, hulls and MLW) together, the highest calculated annual doses, which are only 3×10^{-7} Sv/a, are due to the small fractions (about 1%) of I-129 present in the hulls and the medium-level waste.

The deterministic performance calculations are complemented with stochastic calculations in which the uncertainties in the model parameters are taken into account. The results of such calculations show that the expected value for the annual dose and the 95% percentiles also remain below the dose constraint. Figure 3.4 also indicates that a possibly problematic waste type consists of the iodine waste. This waste type is not included on the list of waste types that have to be considered in Belgium (NIRAS, 1989). This waste arises in the reprocessing plants from the capture of iodine from the dissolver gasses and contains 98% of the I-129 which is generated by nuclear fission. Since iodine is not sorbed on the Boom clay, the disposal of this waste form might, in case all of it were returned to Belgium, lead to some scenarios for annual doses of the order of the dose constraint.

In 1991, the CEC's EVEREST project was started. Complementary sensitivity studies are investigated within EVEREST to determine which elements of the repository system strongly influence the performance of the repository system. Stochastic calculations were per-

formed to assess the influence of the uncertainties in the model parameters on the calculated annual doses. Other sensitivity studies focused on conceptual model uncertainty in the aquifer modelling and on uncertainties in the scenario descriptions.

In 1992, the next Belgian performance assessment, ordered by NIRAS/ONDRAF, started with the description of a systematic scenario study. Recently two additional scenarios have been analysed (Wibin and Marivoet, 1995). The exploitation scenario considers the pumping of water from the Ruisbroek-Berg aquifer which underlies the Boom clay layer. The consequences associated with this scenario are more than two orders of magnitude higher than the doses estimated for the normal evolution scenario in which only pumping from the overlying aquifer was considered. The "poor sealing" scenario considers that the sealing of the transport gallery and the access shaft are not successful. The consequences of this scenario are not much higher than the ones estimated for the normal evolution scenario.

The next step will be a detailed analysis of the normal evolution scenario. In this analysis, all phenomena which are about certain to occur will be considered. Thus, the expected evolution of the climate for the next 100,000 years has to be taken into account. The impact of the climate changes on the aquifer system will be evaluated by applying geoprospective methods. In this time period, two glaciations are expected to occur that will result in drastic changes in the geography and shallow geology. Consequently, the uncertainties in the aquifer system become considerable for these time scales. However, it should be noticed that these climate changes will have very little influence on the migration of radionuclides in the Boom clay, which is about 32 million years old and would remain stable for another million years.

The case of spent fuel. SCK•CEN started a preliminary performance assessment of the direct disposal of spent fuel in a clay layer (Marivoet et al., 1995). The spent fuel types considered are uranium oxide fuels with burnups of 33 and 45 MWd/tHM and MOX fuel with a burnup of 45 MWd/tHM. The radionuclide inventories have been calculated with the ORIGEN2 code. A serious problem in the assessment is that no information is available from specific experiments on the behaviour of spent fuel in conditions similar to the ones prevailing in a repository in clay. Therefore, the near-field model is largely based on results obtained from research programmes (mainly Canadian and Swedish), on the dis-

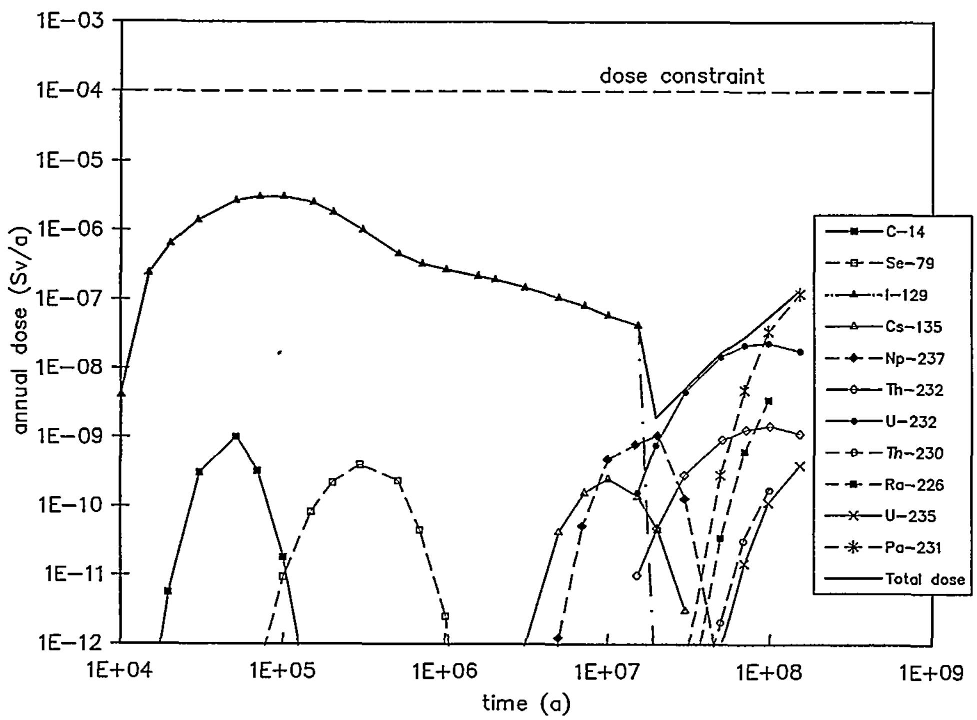

Figure 3.5. Preliminary performance assessment of the direct disposal of spent fuel in clay. Calculated annual doses via the water well pathway (normal evolution scenario).

posal of spent fuel in granite.

The annual doses calculated in the case of the normal evolution scenario are given in Figure 3.5. The maximum calculated annual dose is due to I-129 and should occur about 60,000 years after disposal. The maximum annual dose actually depends on the fraction of iodine which is present in the gap between pellets and zircalloy hulls (about 10%). The bulk of the iodine (90%) is dissolved in the uranium oxide crystals and is released over a 10 million year period. A second but lower dose peak due to actinides is expected to occur after a very long time period (several tens of millions of years). The preliminary assessment shows that, from a radiological point of view, the direct disposal of spent fuel in a clay layer is an acceptable option.

Chemical toxicity aspects in the R&D geological disposal programme. A preliminary assessment of the chemical toxicity aspects, related to the geological disposal of radioactive waste, has been carried out. A number of simplifying assumptions have been made for this assessment, and the study considered only the possible toxicity of heavy and other metals present in the waste forms or the packing materials. The elements that caused the highest concentrations in the overlying aquifer were uranium, molybdenum and nickel. However these concentrations are lower than the concentration limits given in a European guideline for water to be used for drinking water purposes. The calculated low concentrations can be attributed to the low solubilities of some metals and the fact that most of the expected species in solution are cations, which are strongly sorbed on the Boom clay.

However, this assessment is still preliminary. For many essential parameters like solubility limits and retention coefficients, rough estimates had to be used in the simulations because no experimental data were available. The consequences of beryllium, which arises from the beryllium matrices of SCK•CEN's research reactor BR2, were not analysed because beryllium is essentially toxic to the lungs; therefore, the analysis requires the application of a detailed biosphere model to assess the consequences of the inhalation of beryllium.

REFERENCES

Berghman, K., P. Van Iseghem and W. Timmermans, The physico-chemical stability of bitumenized Eurochemic medium-level reprocessing waste, Proc. 2nd Intern. Seminar on Radioactive Waste Products, BfS Schriften 1990, Ed. E. Warnecke, R. Odoj, R. Simon, pp. 479-498, 1990.

Bonne, A. and G. Collard, Scientific bases of SCK•CEN's programme on radioactive waste disposal in argillaceous formations. Contributions to the success and progress of national, foreign and international programmes, Proc. International symposium on "Geological Disposal of Spent Fuel, High-Level and Alpha-Bearing Wastes", pp. 339-351, IAEA-SM-326/38, Antwerpen, 19-23 October 1992, Ed. IAEA, Vienna, 1992.

Cornélis, B. and P. Van Iseghem, In-situ corrosion behaviour of candidate canister/overpack materials in clay, In-situ Testing of Radioactive Waste Forms and Engineered Barriers, Pre-Print of EUR 15629, Ed. T. McMenamin, 1994.

D'Alessandro, M. and A. Bonne, Radioactive waste disposal into a plastic clay formation - A site specific exercise of probabilistic assessment of geological containment, Harwood Academic Publishers, ISBN 3-7186 0084-6; ISSN 0275, 1981.

De Preter, P., M. Put, P. De Cannière and H. Moors, Migration of radionuclides in Boom clay: State-of-the-Art report, June 1992. NIROND 92-07 report. Contract FI 2W/0039. Study performed under cost sharing contract as part of the European Atomic Energy Community's programme on management and storage of radioactive waste, 1992.

Heremans, R. and L. Baetslé, Investigation of the use of a clay formation for terminal disposal of radioactive waste, Proceedings of the Nuclex 78 technical meeting "Spent fuel processing and waste management of nuclear power plants," Basel (CH), Paper B3/3, 3-7 October, 1978.

Lemmens, K., P. Van Iseghem and W. Lian, The leaching of Pu, Am, Np and Tc form high-level waste glass in clay media, Scientific Basis for Nuclear Waste Management XVI, Mat. Res. Soc. Symp. Proc. vol. 294, pp. 147-154, Mat. Res. Soc. Publ., 1993.

Marivoet, J., G. Volckaert, and A. Bonne, Performance assessment of a high-level waste repository in clay, Proc. Twelfth Internat. Symp. on "The Scientific Basis for Nuclear Waste Management," Berlin, 10-13 Oct., 1988.

Marivoet, J, UPDATING 1990, Updating of the performance assessments of the geological disposal of high-level and medium-level wastes in the Boom clay formation, SCK•CEN, Mol, Report BLG 634, 1992.

Marivoet, J., G. Volckaert, A. Sneyers and J. Wibin, Preliminary performance assessment of the disposal of spent fuel in a clay layer, CEC, Luxembourg, (report in preparation, 1995.)

Neerdael, B. and D. De Bruyn, Excavation response studies at the Mol facility, NEA Workshop on Excavation Responses in Deep Radioactive Waste Repositories - Implications for Engineering Design and Safety Performance, Winnipeg, Canada, April 1988, OECD Publ., pp. 465-479, 1988.

Noynaert, L., G. Volckaert, P. Meynendonckx, R. Beaufays, and A. Fonteyne, The CERBERUS test, Proc. Workshop on Pilot tests on radioactive waste disposal in underground facilities, Braunschweig (Germany), 19-21 June, 1991, Ed. CEC, EUR 13985, pp. 239-253, 1992.

NIRAS/ONDRAF, SAFIR, Safety Assessment Fea-sibility and Interim Report, NIRAS/ONDRAF, Brussels, 1989.

Put, M., P. De Canniere, H. Moors, A. Fonteyne, and P. De Preter, Validation of performance assessment model by large scale in-situ migration experiments, Proc. International symposium on Geological Disposal of Spent Fuel, High-Level and Alpha-Bearing Wastes, pp. 319-326, IAEA-SM-326/37, Antwerpen, 19-23 October, 1992, Ed. IAEA, Vienna, 1993.

Van Iseghem, P., K. Berghman, K. Lemmens, A.D., Stalios, W. Timmermans and S. Van Winckel, The characterization of nuclear waste glass, Rivista della Staz. Sper. Vetro No. 6-1990, pp. 179-183, 1990.

Van Iseghem, P. (1994), In-situ testing of waste glass in clay, Scientific Basis for Nuclear Waste Management XVII, Mat. Res. Soc. Symp. Proc. vol. 333, pp. 133-144, Mat. Res. Soc. Publ., 1994.

Volckaert, G., F. Bernier, E. Alonso, A. Gens, J. Samper, M. Villar, P.L. Martin-Martin, J. Cuevas, R. Campos, H. Thomas, C. Imbert, and V. Zingarelli, Model development and validation of the thermo-hydraulic-mechanical and geochemical behaviour of the clay barrier, Annual progress report 1993, CEC contract FI2W-CT91-0102 and FI2W-CT90-0033, SCK•CEN, Mol, R 3025, 1994.

Volckaert, G., L. Ortiz, and M. Put, In situ water and gas injection experiments performed in the HADES underground research facility, Accepted for presentation at the fifth international ASME conference on radioactive Waste Management and environmental remediation, Berlin, September 3-8, 1995, (to be published in proceedings, 1995.)

Wemaere, I. and J. Marivoet, Updated regional hydrogeological model for the Mol site (the northeastern Belgium model), SCK•CEN, Mol, Report R-3060, 1995.

Wibin, J. and J. Marivoet, Consequence analyses for two altered evolution scenarios, SCK•CEN, Mol, Report R-3045, 1995.

CHAPTER 4

THE PROBLEM OF SITE SELECTION FOR A RADIOACTIVE WASTE REPOSITORY IN BULGARIA

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Abstract. The approach to the solution of the problem of site selection for the construction of a high level radioactive waste (RAW) repository in Bulgaria has passed so far through two stages. The first stage includes the regional survey of the country on a 1:500,000 scale to select regions with suitable conditions for further investigations. At the second stage, 18 sites have been selected in these regions, to be subjected to more detailed analysis. Two types of areas, providing comparatively favourable conditions, are considered in the present work: the marl terrains in northwest and northeast part of the Fore-Balkan in north Bulgaria, and the Sakar granite pluton in southeast Bulgaria. The marls are clayey and have a thickness of up to 1000 m. They are situated in a zone of calm tectonics and seismicity of VII-VIII degree on the MSK scale. All available data for the Sakar granite pluton are also favourable. Additional *in-situ* and laboratory investigations have to be carried out for both types of formations with the view of collecting the necessary specific information, required for the in-depth RAW disposal.

4.1 INTRODUCTION

The Bulgarian Nuclear Power Plant (NPP) of "Kozlodui" is equipped with 6 nuclear reactors of the VVER type (produced in Russia) and has an output of 3760 MW of electric energy, which is 35 - 40 % of the total output of the country. The construction of the second Nuclear Power Plant of "Belene" has been stopped because of civil protests.

So far, the unsolved problem of radioactive waste (RAW) processing and storage is one of the reasons for the negative social reaction and for the attitude of ecological organizations against nuclear power generation. Vast quantities of unprocessed low and medium level wastes have been accumulated after nearly twenty years of operations at the Kozlodui NPP. Their total volume at the closing of the plant is estimated to be from 128,000 to 170,000 cubic meters in the cemented state.

The problem with the highly radioactive wastes is still more serious. During the last several years, Russia refuses to further process the waste fuel, which is now stored at a temporary repository. Negotiations are in progress for the signing of a new agreement, according to which the vitrified highly radioactive wastes obtained after the spent fuel processing are to be returned to Bulgaria for final disposal. In case that such agreement is not achieved, the spent nuclear fuel will be stored for several decades at a temporary repository in Kozlodui. In 1991, the Council of Ministers of the Republic of Bulgaria imposed the task of developing a concept for the construction of a national RAW repository on the Bulgarian Academy of Sciences (BAS) with the view of accelerating the solution of the problem. A large team of scientists from the Earth Sciences Division of BAS was organized to study foreign experience and to make a thorough analysis of the natural environment in Bulgaria. As a result, areas and geological complexes that appear favorable for further investigations concerning the construction of RAW repositories have been determined (Milanov et al., 1993).

The problem of selecting sites and rock systems, suitable for highly active RAW has been carried out in two stages. A preliminary screening of the territory of the country has been performed, and single regions distinguished where suitable geological formations are expected to be found. A subsequent screening has been carried out in these regions to select 18 sites offering considerably better conditions.

4.2 PRELIMINARY SCREENING OF THE TERRITORY OF THE COUNTRY

The territory of Bulgaria has a complex and diverse geo-

logical structure. Part of it is affected by young Alpine tectonics and is characterized by complex and difficult geological conditions from the point of view of the problem to be solved.

Rocks and formations ranging in age from the Precambrian to the Quaternary make up the geological structures of the country. The metamorphic and magmatic complexes are distributed mainly in south Bulgaria: the Rhodopes massif, the Sredna Gora and the Sakar-Strandzha zone, and the sediment complexes in the region of the Balkanides, the Kraishte and in the Moezian platform. The volcanogenic sediments and volcanic complexes are well developed in the Sredna Gora (of Upper Cretaceous age) and in the Rhodopes massif with Paleogenic formations (of Oligocene-Miocene age) situated on its periphery.

The Bulgarian territory falls entirely within the region of the Alpine-Himalayas orogeny (Boncev, 1971) and this is the prerequisite for the relatively complex tectonic conditions. Deformations were established which are related to the Precambrian mega stage in the development of the highly metamorphic complexes as well as the Caledonian Variscian deformations, early and late Alpine deformations and the neotectonic deformations related to rock complexes of the Fanerozic mega stage.

The superposition of various tectonic movements in the course of time has led to the formation of folded structures, varying in type and magnitude, and terrains of block or block/thrust structures. The rock complexes in them are strongly deformed and have hydrogeological and geotechnical engineering conditions which are difficult to model. This especially concerns the Balkanides, the Kraishte, the Strandzha zone and parts of the Rhodopes massif. Fault zones exist in these morphotectonic units, along which neotectonic movements of an amplitude of up to 3400 m occur, accompanied by rock deformations and the development of exogenic geological processes.

The selection of zones with low tectonics, from the Neogene to the present, and containing suitable rock complexes is facilitated by the carefully studied geology of the country. The whole territory of Bulgaria is presented on geologic, tectonic, engineering geology and hydrogeology maps on scales from 1:10,000 to 1:1,000,000. A map of the geologic hazards in Bulgaria has recently been published on a scale of 1:500,000 (Iliev et al., 1994). The new geologic map of Bulgaria on the scale of 1:100,000 has been very useful for the geologic screening of the country. It is based on the lithostratigraphic principle and provides a very good idea for the distribution, composition and structure of rock complexes. There is also available information for the in-depth structure of the lithostratigraphic units, which provides a means of choosing rock complexes of great thickness and suitable properties (granites, gneisses, volcanic rocks, serpentinites, marls and clays. Evaporitic rocks have not been considered since they are of industrial interest or are situated at great depth.

The geological analysis of the country has been supplemented by analyses from the other earth sciences. The lithostratigraphic information, combined with data from tectonic and seismic investigations, has provided the possibility of delineating zones of relatively less tectonics, where the number of active faults is smaller and the seismic centers are of smaller magnitude. The seismic characteristics of the Balkans peninsula and Bulgaria are reflected by a series of maps, showing locations of strong earthquakes up to 1990 with the magnitude-depth distribution, the possible epicenters of strong earthquakes, the seismic risk for a period of 1000 years, etc. The specificities of the temperature regime and of the geothermal field are also taken into consideration.

The contribution of engineering geological analysis consists of information about the texture and structure of the rock massifs, the physical, mechanical and chemical properties, and the conditions for the construction of underground openings in them.

The hydrogeological conditions in Bulgaria are summarized in maps of different scale, which provide information about the groundwater basins, their water transmissibility and rock filtration properties. Vast regions of the country have been excluded from further consideration on the basis of the available hydrogeological information, proving that RAW disposal in them is impossible.

The geographic analysis has taken into account the relief characteristics, the hydrographic network, the climatic specificities, the natural resources, the economic-geographic and conditions of population. A map has been created on the basis of the density of population, the network settlements and the average distance between settlements, on which five types of regions are distinguished from the point of view of the problem to be solved. The possibilities of utilizing about 260 mines for disposal purposes have been studied separately, and a selection has been made for the most prospective sites. The comparison and juxtaposition of the results of these

different earth sciences have led to the selection of regions offering the best conditions for investigating a site, suitable for the construction of a RAW repository. On this basis, a "Map of Categorization of Terrains in the Territory of Bulgaria" has been prepared on a scale of 1:500,000. A reduced copy of this map is presented in Figure 4.1.

The categorization has been performed on the basis of the following principles:

- All natural conditions of the country are considered but priority is given to the geological conditions;
- Preference is given to those regions where positive assessments of the geologic, seismic-tectonic and hydrogeological conditions coincide and the other conditions are not negatively assessed;
- Except for natural conditions, other factors are taken under consideration too—technogeneous loading, demographic conditions, ecological conditions, interests of tourism, etc.

The regions possessing favourable characteristics are concentrated mainly in the northwest and northeast parts of the country (Fig. 4.1). They contain: (a) rock complexes of considerable thickness and surface distribution; (b) no active faults; (c) level of seismicity that is lower than the VII - VIII degree according to the 1000 year map; (d) hydrogeological conditions that are acceptable; and (e) a density of population that is not too great.

4.3 METHOD FOR SELECTING SUITABLE SITES

The next step in the procedure of site selection has been to distinguish several tens of areas and mines in the regions with favourable conditions, where comparatively good geologic settings exist for the in-depth disposal

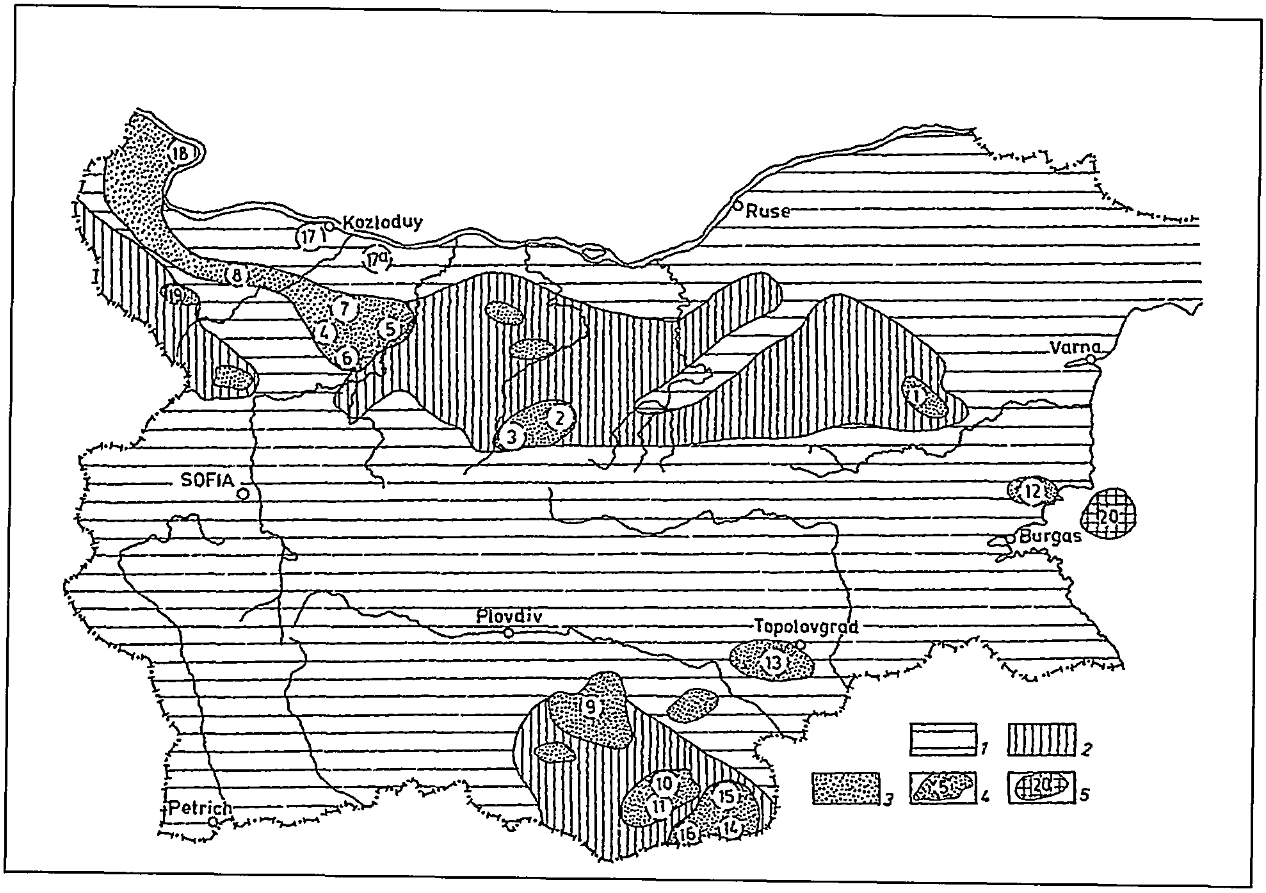

Figure 4.1. Map showing categorization of the Territory of Bulgaria into: (1) site unsuited for RAW repository construction; (2) prospective sites for further investigations; (3) regions with selected single sites for additional investigations; (4) a particular numbered site; and (5) a tunnel in the Black Sea offshore area (after Kostadinov and Kozhoukharov, 1992).

of highly radioactive wastes. Their number has been reduced to 18 by the method of successive elimination. The selected 18 sites have been investigated in more detail on the basis of the available information and minimum additional investigations. A report has been prepared for each of the sites, summarizing the geology, hydrogeology and engineering geology information. The reports are supplemented by detailed geologic maps and lithostratigraphic profiles.

The comparative analysis of the sites is performed on the basis of the criteria proposed by the special commission of the EC countries (European Catalogue 1979, Orlowski and Shaller, 1990), taking into consideration the specific conditions in Bulgaria. The criteria have been divided in four groups: (a) rock-linked parameters; (b) basic formation parameters; (c) formation - environment factors; and (d) supplementary selection factors. The rock-linked parameters include the rock's sorption capacity, thermal conductivity and solubility. The formation is evaluated by a larger number of parameters: thickness, minimum depth, surface area, homogeneity and uniformity, permeability and geotechnical properties. The environment in which the formation exists is evaluated by three important factors: hydraulic gradient, seismicity and tectonics. The additional factors include: the availability in the region of a rock formation and its potential use for economic purposes, the sensitivity of the zones to climatic and hydrologic changes and sociogeographic and population conditions.

The sites have been classified according to their suitability on the basis of an ordinary expert analysis, as well as by means of a system analysis (Vachev and Evstatiev, 1994). Priority scales have been developed for each criterion for the purposes of system analysis.

This classification has a preliminary character because the available information on the most important criteria such as sorption of the rock, its filtration properties and the hydraulic gradient, are insufficient for quantitative assessment. However, the sites composed of Lower Cretaceous marls in the Fore-Balkan, the Sakar granite pluton, and the gneiss and serpentinite massifs in the East Rhodopes, although at a preliminary stage, have been determined as the most favorable for further investigations.

4.4 CHARACTERISTICS OF SITES

Examples of prospective disposal sites from two different areas are presented in order to show the degree of investigations and the completeness of the available data at the conceptual stage. The first area is situated in north Bulgaria and consists of thick marl complexes, and the second is in south Bulgaria in the Sakar granite pluton.

4.4.1 Prospective Sites in Lower Cretaceous Sediments of Fore-Balkan

A vast depression was formed in the Fore-Balkan of north Bulgaria during the Lower Cretaceous where sediments, mainly of marl composition, are deposited (Nikolov and Khrischev, 1965). These sediments have been carefully investigated at the surface and at depth during explorations for oil and gas deposits, and their tectonic conditions and lithostratigraphic structure have been studied in detail (Nikolov et al., 1991). They have been subjected to additional analysis in three reports which are supplements to the Concept of BAS for a National RAW repository (Nikolov and Ruskova, 1992, Monov, 1992, 1993).

The information presented below is obtained from these reports. The results from the analysis show that several sites situated in northwest and northeast Bulgaria are most favorable for more detailed future investigations. These sites contain comparatively pure clayey marls of great thickness, up to 1000 m and more. The marls in central north Bulgaria contain thick accumulations of limestones and sandstones. Some of them are water-bearing, and for this reason are excluded from further consideration.

4.4.2 Prospective Sites in Northwest Bulgaria

Several sites situated between the Iskar and Ogosta rivers fall into this category (sites Nos. 4, 5, 6, and 7 in Fig. 4.1). They are situated at a distance of 60-80 km south of the Kozlodui NPP. They are in the northern strip of the west Fore-Balkan, which is characterized by relatively calm neotectonics. Intensive tectonic activity had ceased after the Pyrinean phases, when the Fore-Balkan was formed. There are no active faults in the vicinity of the sites. The seismicity is of the VII degree on the MSK scale for a period of 1000 years.

Gas deposits have been discovered in the region, and as a result, there are more than 150 deep boreholes. This has provided data for lithostratigraphic and geophysical investigations and the investigation of geological structures both at the surface and at depths down to 3500 m (Fig. 4.2).

The Sumer and Mramoren formations which consist of

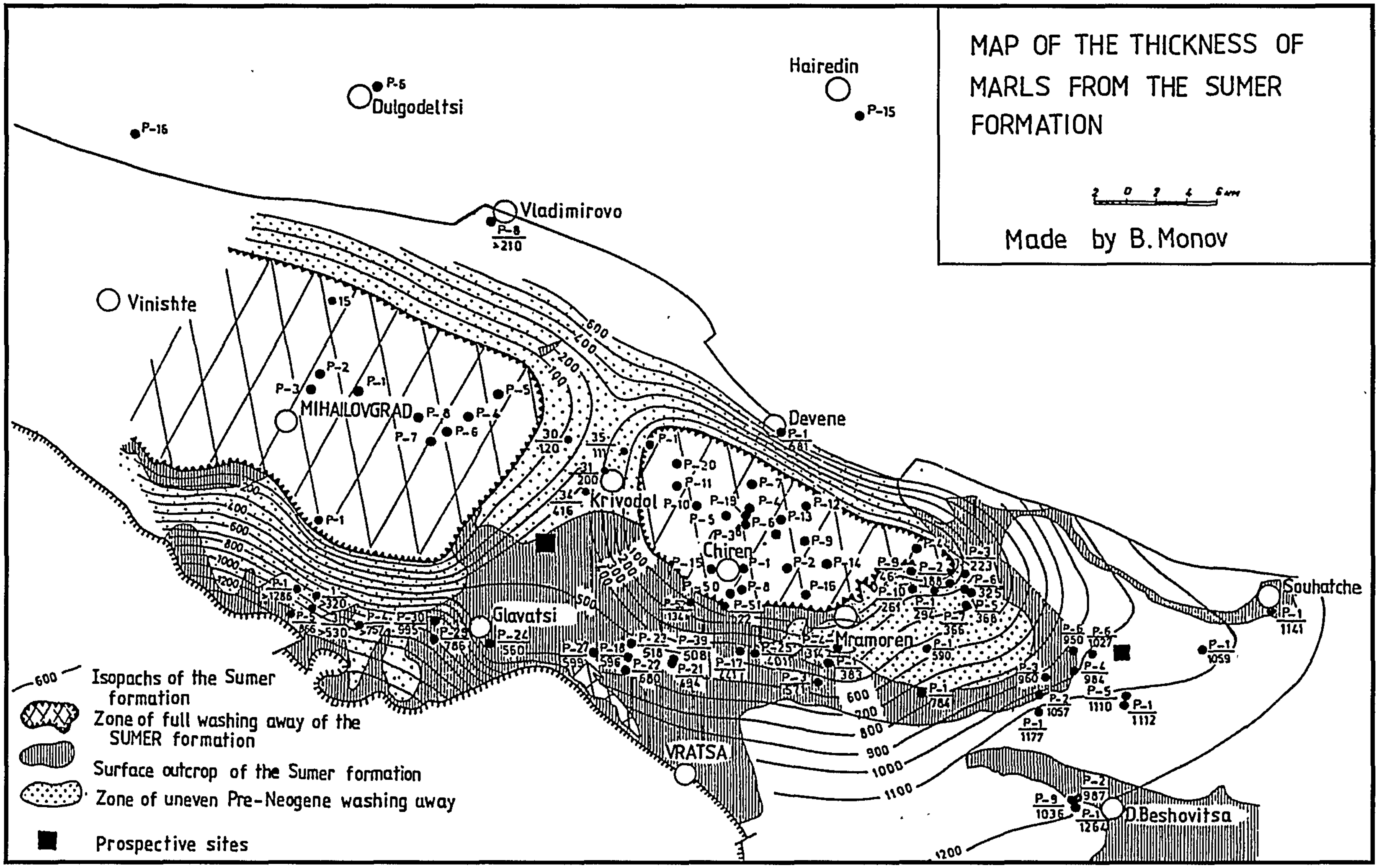

Figure 4.2. Map of the thickness of marls of the Sumer Formation (after Monov, 1993).

very thick clayey marls are of greatest interest from the point of view of the construction of an in-depth repository (Fig. 4.3). The available hydrogeological data indicate that the marls are dry and of very low permeability.

4.4.3 Prospective Sites in Northeast Bulgaria

The Gorna Oryahovitsa formation, in the Lower Cretaceous sediments of northeast Bulgaria, is most interesting as a prospective site for RAW disposal. (Nikolov et al., 1991). The formation consists of Lower Cretaceous clayey marls where terrigenous intercalations are almost absent (Figs. 4.4 and 4.5). The selected site (No. 1 in Fig. 4.1) near the village of Zlatar, Shoumen district, is situated in a shallow anticline and is characterized by a low level of neotectonics and lack of active faults. The marl thickness exceeds 600 m. The Ticha formation, which is a buffer zone also containing thick marl layers, occurs in the Gorna Oryahovitsa formation (Fig. 4.5). The seismicity of the region is of VIII degree on the MSK scale according to the seismic map for a period of 1000 years.

The Concept of BAS accepts marl terrains as possibilities for both in-depth disposal of high level RAW and the surface-type repository for low- and medium-level RAW. From the point of view of problems related with the attitude of the population concerning the RAW repository, it is advantageous to build repositories for both types of waste at one and the same place.

Two types of repositories are proposed for consideration for in-depth disposal of high level RAW: (1) mine disposal and (2) borehole disposal. The comparatively small volumes of this type of waste make it possible to consider the construction of the second type of repository, which is less expensive and possesses certain technological and ecological advantages. Moreover, considerable experience exists in Bulgaria on the performance of deep boreholes, in marl rocks, in relation to the exploration for oil and gas deposits.

4.4.4 Prospective Sites in Magmatic-Metamorphic Terrains

Several sites with magmatic and highly metamorphic rocks in southeast Bulgaria are proposed in the Concept of BAS. The following sites have been selected after comprehensive analysis of natural conditions in this part of the country: the Byala Reka structure, composed of Precambrian highly metamorphic gneisses; the Avren and Zhâlti Chal sites, composed of metaultrabasites; the Sakar site, composed of Paleozoic granites and two sites in the East Rhodopes Paleogene depression, composed of volcanic and volcanogenic sediments (Nos. 9, 10, 11, 13, 14, 15 and 16 in Fig. 4.1). Preliminary explorations on these sites have been performed by Kozhoukharova (1992), Kozhoukharova and Kozhoukharov (1992), and Kozhoukharova, et al. (1992).

4.4.5 Prospective Sites in Paleozoic Granites of Sakar Pluton

The Sakar site in southeast Bulgaria (No 13 in Fig. 4.1) is entirely within the area of the mountain of the same name. The region is composed of Precambrian high-crystalline metamorphites, Paleozoic metasediments and granites, Triassic metasediments and Pliocene formations (Kozhoukharov et al., 1994a, b). From a tectonic point of view, the region falls into the range of the Sakar unit, which is a component of the Sakar-Strandzha tectonic zone. About 60% of the site is composed of Paleozoic granites, known as the Sakar pluton. At the surface, the granites have been well investigated, but there are no data about their condition at depth. There are no active faults in them from the neotectonic stage; faults of this type are observed along the periphery of the mountain beyond the limits of the pluton. The seismicity is of the VII degree on the MSK scale for a period of 1000 years.

4.4.6 Characteristics of Sakar Pluton

The Sakar pluton was emplaced entirely among the Precambrian high-crystalline, ortho- and para-metamorphites of the Prazhodopian supergroup (Kozhoukharov, 1984, 1987) with presumed age of Archaic (?) - Lower Proterozoic. It intrudes and is partially metamorphosed by contact with the porphyric blastic migmatites of the Pânovo formation, the different gneisses, gneiss-schists and schists, amphibolites, graphite- and iron-bearing quartzites, etc., of the Zhâlti Chal variegated formation as well as the metaconglomerates of the Konstantinovo formation. A multitude of xenolites of these rocks occur in the peripheral parts of the pluton. At its western end, the Sakar pluton crosses and contact metamorphoses Upper Carboniferous sediments. The basic varieties of the granitoids in the Sakar batholith are three: uniformly grained; porphyric and porphyroid and aplitoid-pegmatoid (Alexandrovo) granites.

The porphyric granites are found in the western parts of

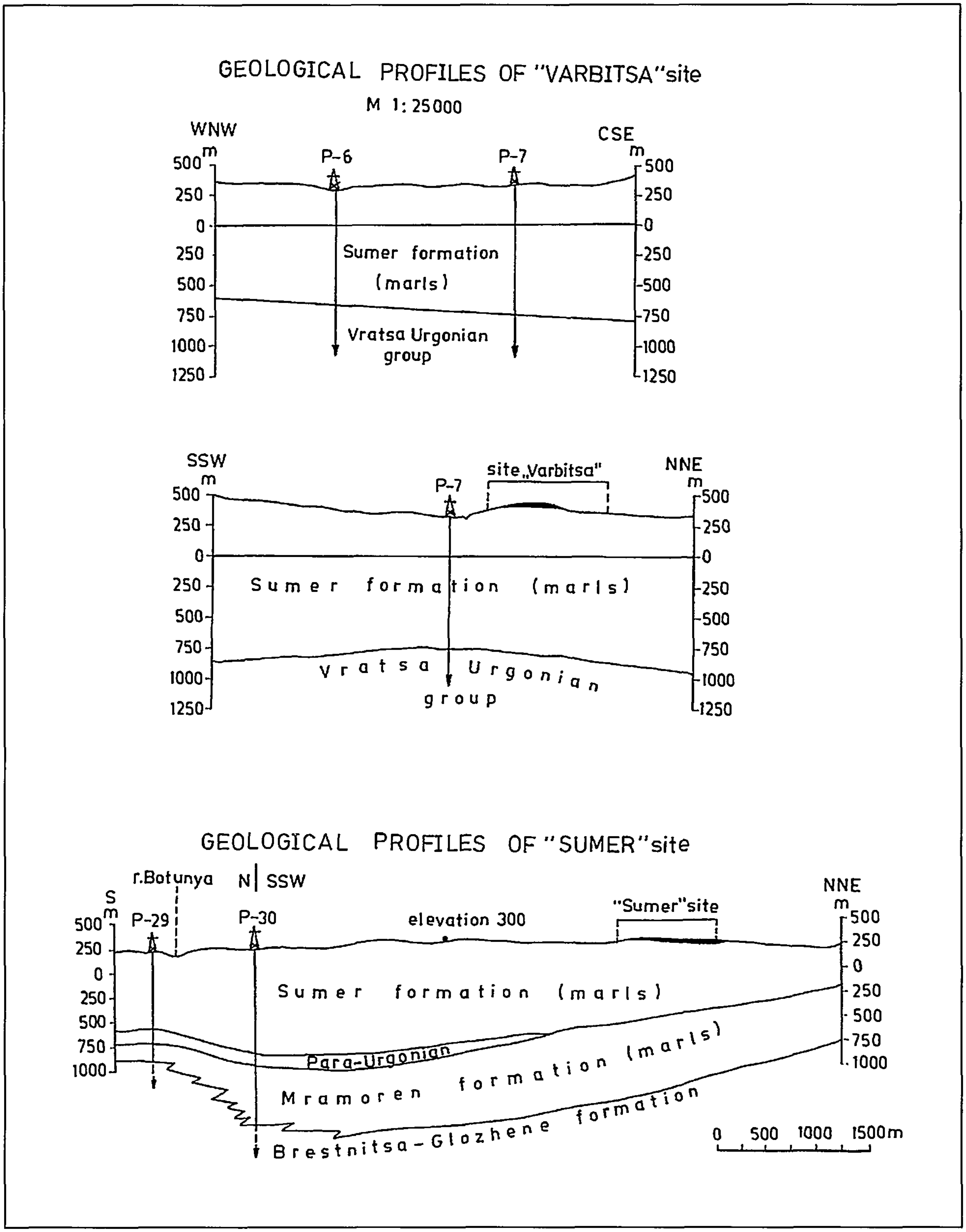

Figure 4.3. Geological profiles of the "Varbitsa" and "Sumer" sites (after Monov, 1993).

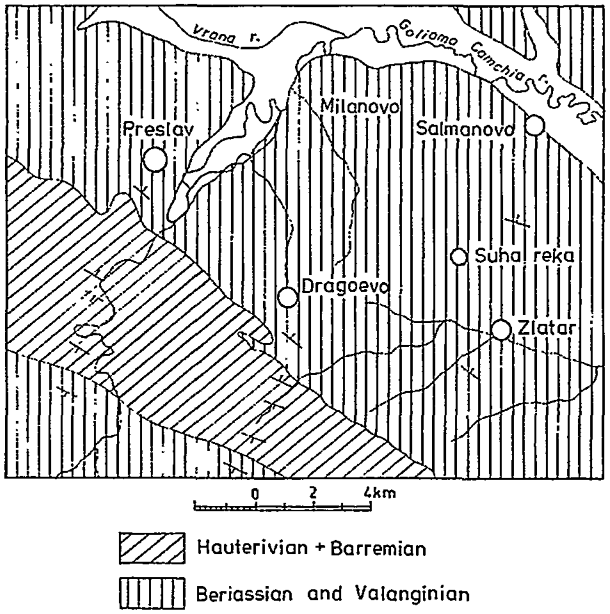

Figure 4.4. Geological map of the Zlatar site (after Nikolov and Ruskova, 1992).

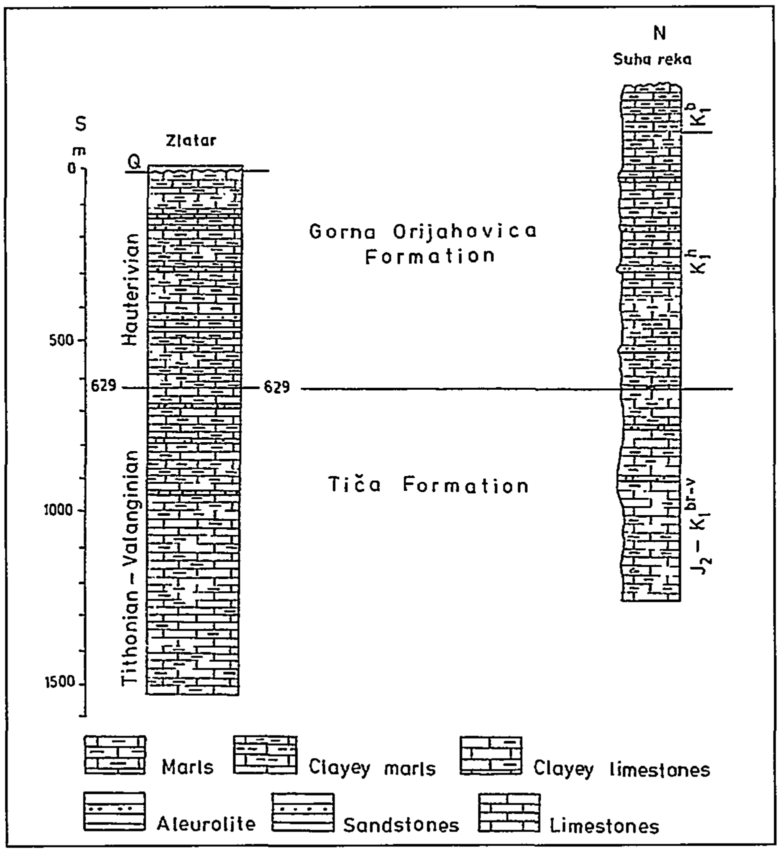

Figure 4.5. Lithostratigraphical columns of the Zlatar site (after Nikolov and Ruskova, 1992).

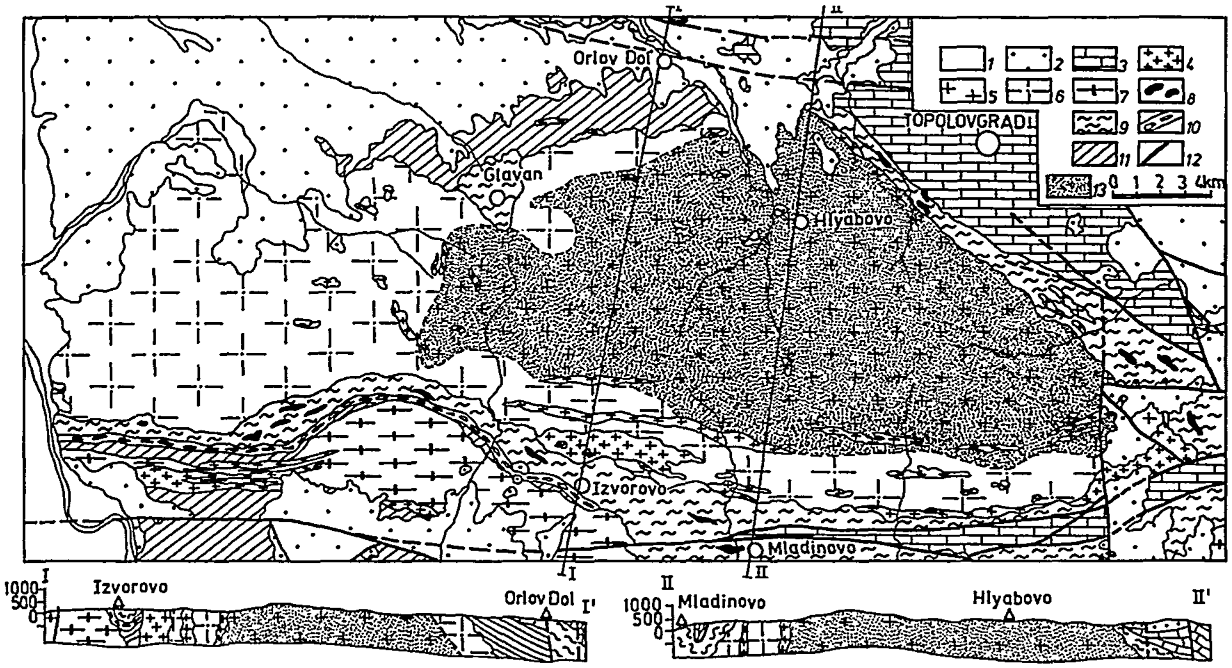

Figure 4.6. Geological map of the Sakar pluton: 1 - Quaternary; 2 - Pliocene sediments; 3 - Upper Paleozoic and Triassic metasediments; 4 - leucocratic (Alexandrovo) granites, Sakar pluton; 5 - medium-grained granites; 6 - porphyric granites, Precambrian; 7 - Lessovo gneiss-granites; 8 - metaultrabasites; 9 - Zhâlti Chal variegated formation; 10 - Konstantinovo metaconglomerate formation; 11 - Pânovo porphyric-blastic formation; 12 - faults; 13 - a site, suitable for high level RAW disposal (after Kozhoukharov, 1992).

the pluton. The porphyries are of potassium feldspar and are characterized by a great variety of sizes and rock saturations. Their length usually varies from several centimeters to 7-8 cm, sometimes reaching 12-15 cm. The porphyric granites are biotitic to two-mica ones. The uniformly grained granites are medium to coarse grained. They elevate the region between the villages of Glavan, Balgarska Polyana, Hlyabovo, Sakartsi and the neighbourhood of the Bairaka and Karaburun peaks. The micas are of the biotite type, but two-mica varieties also occur.

The mineral composition of the two facial varieties is analogous. They are composed of: quartz, plagioclase, microcline, biotite and occasionally of muscovite and accessory minerals - apatite, zircon, garnet, sphene and clinozoisite. Biotite is often replaced by muscovite and some microclines exhibit later crystallization and active erosion of the plagioclases.

The Alexandrovo leucocratic, aplitoid-pegmatitic granites have a wider distribution than that shown on the map sheet. They occur in the form of elongated bodies and veins mainly in the east-west direction in the southern parts of the pluton, where they cross the rocks of the Pânovo and Zhâlti Chal formation and the two facial varieties of the Sakar batholith.

The granites are fine- and medium-grained and occasionally coarse-grained. Three varieties are distinguished among them: medium-grained muscovite granites, coarse-grained pegmatoid granites and fine-grained aplitoid granites. The individual varieties are transformed into each other, but the dominating granites are medium grained, while the aplitoid ones mainly occupy the peripheral parts of the rock bodies or fill some of the veins. These granites do not differ in their mineral composition from the other two varieties of the Sakar batholith except for the biotite content, which is quite rare or missing in the Alexandrovo granites.

Different opinions have formed concerning the age of the Alexandrovo and Sakar granites, covering the range from the Archaic (?) to the Jurassic. The direct geological relationships between the Alexandrovo and Sakar granites, which intrude and contact metamorphose the

rocks of the Paleozoic Klokotnitsa Formation, as well as outliers of granite in the Lower Triassic metasediments (Sakar type) provide the basis for assuming that their age is Upper Paleozoic.

Three categories of terrains are distinguished in the Sakar pluton according to its internal structure. The site, composed of medium-grained granites, indicated by a special symbol on the geological map (Fig. 4.6) is considered to be the most suitable for further investigation and construction of a RAW repository.

REFERENCES

Boncev, E., Problemi na balgarskata geotektonika (Problems of Bulgarian Geotectonics), Sofia, Technika, 203 pp., 1971. (in Bulgarian)

Commission of the European Communities, European catalogue of geological formations having favourable solidified characteristics for the disposal of high-level and/or long-lived radioactive wastes, Introduction and summary. Volume compiled by BRGM, France, 79 p.,1979.

Evstatiev, D., and B. Vachev, Site selection procedure for high level radioactive waste disposal in Bulgaria, IAEA regional project for Technical Assistance PEP/9/010, Recommendations on management of radioactive wastes VVER, Meeting in Sofia, 1993.

Geological map of Bulgaria on scale 1:100,000, 96 sheets.

Iliev, I., ed., Map of the Geological Hazards in Bulgaria, Committee of Geology, Bulgarian Academy of Sciences, Sofia, 1994.

Kostadinov, V. and D. Kozhoukharov, Categorization of terrains on Bulgarian territory, In: Concept of BAS for a national RAW repository, 1992.

Kozhoukharov, D., Lithostratigraphy of Precambrian metamorphics from Rhodope Supergroup in the Central Rhodopes, Geologica Balc., 14, 1, 43-88, 1984.

Kozhoukharov, D., Lithostratigraphy and structure of the Precambrian metamorphics from the core of Biala-reka Dome, East Rhodope Mts., Geologica Balc., 17, 2, 15-38, 1987.

Kozhoukharov, D., Composition, structure and complex geological assessment of the Biala-Reka structure in the East Rhodopes, In: Concept of BAS for National RAW repository, 1992.

Kozhoukharova, E., and D. Kozhoukharov, Petrographic characteristics and complex geological assessment of the ultrabasite bodies in the region of the Zhâlti Chal village, Ivailovgrad district, In: Concept of BAS for National RAW repository, 1992.

Kozhoukharova, E., D. Kozhoukharov, S. Pristavova, and M. Ichev, Geological structure and complex geological assessment of the Sakar site, In: Concept of BAS for National RAW repository, 1992.

Kozhoukharov, D., S. Savov, G. Catalov, E. Kozhoukharova, A. Goranov, and I. Bojanov, Geological map of Bulgaria on scale 1:100,000, sheet Topolovgrad, 1994a.

Kozhoukharov, D., I. Bojanov, E. Kouzhoukharova, and A. Goranov, Geological map of Bulgaria on scale 1:100,000, sheet Svilengrad, 1994b.

Monov, B., Lithostratigraphy of the Lower Cretaceous in Northwest Bulgaria (the Lom depression and the West Fore-Balkan), In: Concept of BAS for a National RAW repository, 1992.

Monov, B., Prospective sites for investigations concerning the construction of a surface-type repository for low and medium active RAW in the Lower Cretaceous marls in Northwest Bulgaria, In: Concept of RAS for a National RAW repository, 1993.

Milanov, M., D. Evstatiev, and D. Kozhoukharov, Concept for National Repository for Radioactive Wastes in Republic of Bulgaria (Summary), Bulgarian Academy of Sciences, 40 pp, 1993.

Nikolov, T. and K. Khrischev. Osnovi na stratigrafiata i litologiata na dolnata kreda v Predbalkana (Principles of the stratigraphy and lithology of the Lower Cretaceous in the Fore-Balkan), Geology of Bulgaria, Series: Stratigraphy and Tectonics, 6, 77-175, 1965. (in Bulgarian)

Nikolov, T., and N. Ruskova, Geological structure and a complex geological assessment of the Lower Cretaceous in the Fore-Balkan between the rivers of

Ossam and Louda Kamchia with suggestions for the location of RAW repository, In: Concept of BAS for a National RAW repository, 1992.

Nikolov, T., N. Ruskova and K. Khrischev, Osnovi litostratigrafii nizhnemelovogo otdela v Bolgarii (Principles of the Lower Cretaceous lithostratigraphy in Bulgaria), Geologica Balc., 21, 6, 3-47, 1991. (in Bulgarian)

Orlowski, S. and V. H. Shaller, Objectives, standards and criteria for radioactive waste disposal in the European Community, Euradwaste series No 1, 81pp., 1990.

Vachev, B., and D. Evstatiev, Radioactive wastes management: Application of AHP, Proc. of the Third International Symposium of the Analytic Hierarchy Process, Session W2A, George Washington University, Washington D.C., July, 11-13, 1994.

CHAPTER 5

CANADA'S NUCLEAR FUEL WASTE MANAGEMENT PROGRAM: THE ENVIRONMENTAL ASSESSMENT OF THE DISPOSAL CONCEPT

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Abstract. Over the last 17 years under the Nuclear Fuel Waste Management Program, Atomic Energy of Canada Limited (AECL) has developed and assessed a concept to dispose of nuclear fuel waste from Canada's CANDU reactors in a vault excavated in plutonic rock of the Canadian Shield. A robust concept has been developed, with options for the choice of materials and designs for the different engineered components of the disposal system. AECL submitted in 1994 October an Environmental Impact Statement on the Concept for the Disposal of Canada's Nuclear Fuel Waste for review under the Canadian Environmental Assessment and Review Process. If the review is completed in 1996, as currently expected, and if the concept is accepted by the review panel and approved by government, disposal would not likely begin before about 2025.

5.1 INTRODUCTION

About 20% of Canada's electricity is generated using CANDU nuclear reactors. Three provincial utilities, Ontario Hydro, Hydro-Quebec and New Brunswick Power, own these reactors and the used fuel removed from them. The used fuel is currently stored in water-filled pools or in dry-storage concrete storage structures. Current storage practices, while safe, require continuing institutional controls, such as security measures, monitoring and maintenance. Thus, storage is an effective interim measure for the protection of human health and the environment, but not a permanent measure. Disposal is needed to manage nuclear fuel waste in a way that does not depend on institutional controls to maintain safety in the long term.

Canada, like other countries, is basing its plans for disposal of nuclear fuel waste on deep geological disposal in the rock of a continental land mass. The Nuclear Fuel Waste Management Program (NFWMP) was launched in 1978 as a joint initiative by the governments of Canada and Ontario. Under the program, Atomic Energy of Canada Limited (AECL) has been developing and assessing a concept to dispose of nuclear fuel waste in plutonic rock of the Canadian Shield. Ontario Hydro has advanced the technologies for interim storage and transportation of used fuel. The two governments stated in 1981 that selection of a nuclear fuel waste disposal site would not proceed until the concept had been reviewed and assessed. Thus, a generic rather than a site-specific concept has been developed.

Participants in the program have included AECL, the lead agency for research on nuclear fuel waste disposal; Ontario Hydro, which has advanced the technologies for storage and transportation as well as contributing financially and technically to the R&D on disposal; Natural Resources Canada (NRCan); Environment Canada; scientists at Canadian universities; and consultants in the private sector. An independent Technical Advisory Committee has provided advice on the scope and quality of the technical work.

During the past seventeen years, AECL has carried out detailed studies on the multiple-barrier disposal concept. The objective has been to develop a concept with flexibility in the choice of methods, materials, and designs for the components of the disposal system. The approach has focused on ensuring that the system as a whole meets safety standards by a large margin.

5.2 THE DISPOSAL CONCEPT

The disposal concept being investigated is a proposed method for the geological disposal of nuclear fuel waste in which:

1. The waste form would either be used CANDU fuel or solidified highly radioactive reprocessing waste;

2. The waste form would be sealed in a container designed to last at least 500 years and possibly much longer;
3. The containers of waste would be placed in rooms in a disposal vault or in boreholes drilled from the rooms;
4. The vault would be nominally 500 to 1000 metres deep;
5. The geological medium would be plutonic rock of the Canadian Shield;
6. Each waste container would be surrounded by a buffer;
7. Each room would be sealed with backfill and other vault seals; and
8. All tunnels, shafts and exploration boreholes would ultimately be sealed so that the disposal facility would be passively safe, that is, so that long-term safety would not depend on institutional controls.

After the disposal vault is closed, a series of engineered and natural barriers would protect humans and the natural environment from the radioactive and chemically toxic contaminants in the nuclear fuel waste. These barriers include the waste form; the container; the buffer, backfill and other vault seals; and the geosphere (the rock, any sediments overlying the rock below the water table and the groundwater flow system). Institutional controls would not be required to maintain safety in the long term.

The nuclear fuel waste, or waste form, would be either used CANDU fuel or, if the used fuel was reprocessed in the future, the solidified highly radioactive waste from reprocessing. The low solubility of used CANDU fuel under the expected disposal conditions would make it effective for retaining radioactive and chemically toxic contaminants, thus it is an excellent waste form in its current state. The liquid radioactive waste that would result if used fuel were reprocessed would not be suitable for direct disposal, but such waste could be solidified to produce an excellent waste form.

The waste form would be sealed in a container to facilitate handling and to isolate it from groundwater for a desired minimum time. The container would be designed to have a minimum life of 500 years following emplacement in a disposal vault. As the conditions at potential sites may vary, the container geometry, material and fabrication method would be developed for each particular waste form and site. Important considerations in the design of containers would include the temperature and pressure that would be imposed on the container at the depth and location selected for the disposal vault and the chemical and microbial environment expected at each site.

The containers of nuclear fuel waste would be emplaced in a disposal vault excavated nominally 500 to 1000 metres below the surface in the plutonic rock of the Canadian Shield. The disposal vault would be a network of horizontal tunnels and disposal rooms that would be excavated deep in the rock, with vertical shafts extending from the surface to the tunnels. Rooms and tunnels might be excavated at more that one level. The vault would be designed to accommodate the rock structure and the subsurface conditions at the chosen site. The method of emplacing the containers in the vault would be selected to suit the specific characteristics of the site and container geometry. Figure 5.1 shows two possibilities for emplacing waste in a vault.

Within the multiple barrier system, the vault seals would perform the following functions to enhance the isolation of the nuclear fuel waste:

1. The buffer would inhibit the movement and modify the chemistry of the groundwater near each container to limit the container corrosion rate, the waste form dissolution rate, and the movement of contaminants;
2. The backfill would fill the space in disposal rooms to keep the buffer and containers securely in place, and in shafts and tunnels to reduce the potential for human intrusion. It would also retard the movement of contaminants by slowing the movement of groundwater, enhancing sorption of contaminants and by chemically conditioning the groundwater;
3. Bulkheads would inhibit groundwater movement at the entrance of disposal rooms, contain any pressures exerted by the backfill, and prevent easy human access to the emplaced waste;
4. Plugs and grouts, located at hydraulically critical locations in the disposal vault, such as locations where shafts and tunnels intersect fracture zones, would inhibit groundwater movement and the potential for contaminant transport in the excavations and in the rock around excavations; and
5. Shaft seals would prevent the shafts from being a preferential pathway for groundwater movement and would reduce the possibility of human intrusion into the sealed disposal vault.

The materials selected for vault seals need to have low hydraulic conductivity (in the order of 10^{-10} m/s or

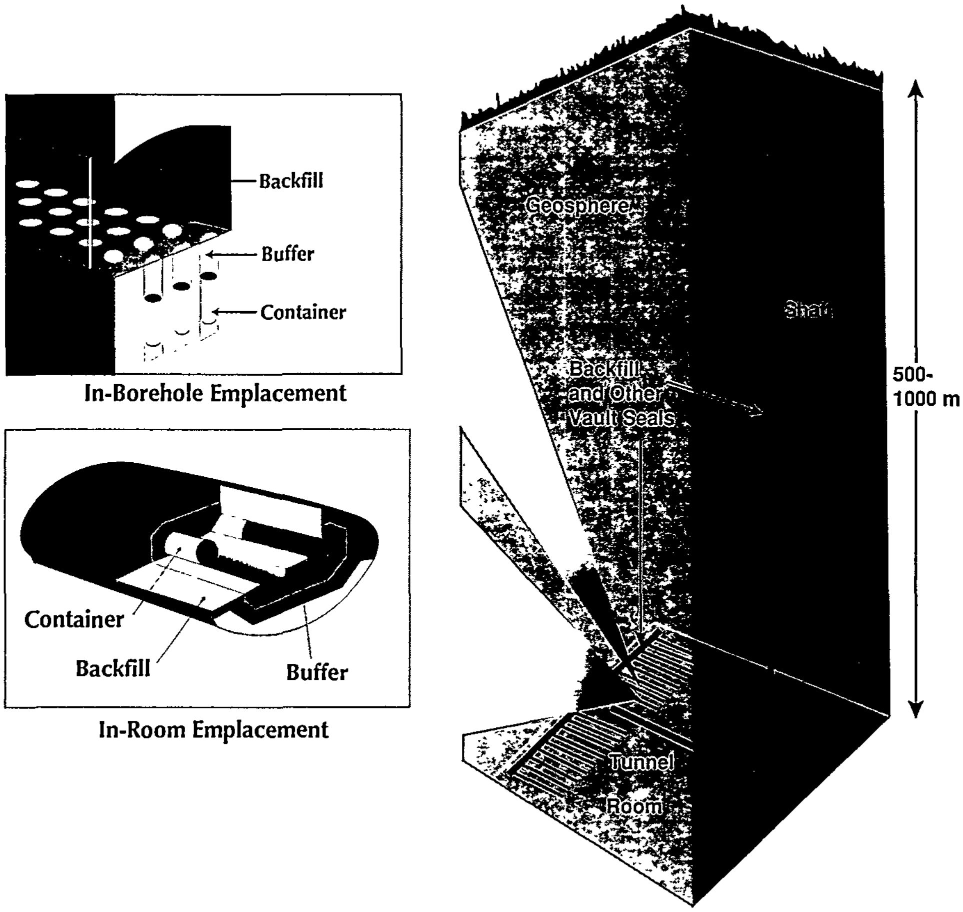

Figure 5.1. The disposal concept showing two examples of waste emplacement.

lower), must be available in the volumes required, must be workable and placeable with current technology, and must have predictable long-term performance.

The geosphere, comprising the plutonic rock, any sediments overlying the rock below the water table, and the groundwater flow system, would surround the disposal vault. The function of the geosphere, as a barrier, would be to protect the waste form, the container and the vault seals from natural disruptions and human intrusion; to maintain conditions in the vault favourable for long-term waste isolation; and to limit the rate at which contaminants from the waste could move from the vault to the biosphere. Any characteristics of a geological medium that support or enhance these functions would be considered favourable.

The plutonic rock bodies of the Canadian Shield are considered favourable as a disposal medium for Canada because they have many characteristics considered favourable including:

1. Wide geological distribution, allowing flexibility in siting, and wide distribution in regions of low topographic relief, where the driving force for groundwater movement in the rock is likely to be low;
2. Geological stability and the likelihood of remaining stable;
3. Very large size, in many cases, allowing the disposal rooms to be located away from discontinuities;
4. Fewer ore bodies associated with them than other types of rock bodies;
5. Relatively high thermal conductivity to dissipate the heat released from the emplaced waste;
6. Chemical composition and hydrogeological characteristics that favour retention and retardation of the type of contaminants expected in the waste form; and
7. Low frequency of permeable fractures, away from

the widely spaced fracture zones, below a depth of 200 to 500 metres.

The choice of methods, materials, and designs for an actual disposal system will ultimately be made on the basis of performance assessments taking into account the characteristics of the specific site on which the facility is to be developed, availability of materials, cost, and practicality. They could include, for example:

- the form of the waste: used fuel bundles or fuel reprocessing waste incorporated in a stable matrix, e.g. glass or ceramic;
- the disposal container material: titanium alloy, copper, or other durable material;
- the container design;
- the composition of materials used for the buffer, backfill, and other seals;
- the excavation method: blasting or boring;
- the depth, geometry, and the number of levels of the vault;
- the size and shape of the excavated openings; and
- the location of the waste containers: within disposal rooms or in boreholes drilled from the rooms.

These choices will not be made until a site for a vault has been selected.

5.3 ENVIRONMENTAL REVIEW

The federal Department of Energy, Mines and Resources (EMR) (now Natural Resources Canada, NRCan) referred the concept for review under the Environmental Assessment and Review Process (EARP) in 1988. As the "Proponent" for this review, AECL is responsible for preparing and submitting an Environmental Impact Statement (EIS) describing the concept. The Environmental Assessment Panel responsible for carrying out the review is chaired by Mr. Blair Seaborn. The Panel has appointed a Scientific Review Group (SRG), chaired by Professor Raymond Price and composed of eminent scientists from a variety of relevant disciplines, to assist it in judging the technical validity and acceptability of the disposal concept. The Canadian Environmental Assessment Agency (CEAA), formerly Federal Environmental Assessment Review Office (FEARO), provides administrative support to the Panel.

The Panel will review AECL's concept, along with a broad range of nuclear fuel waste management issues. These include the criteria for determining safety and acceptability; the approaches used in handling nuclear fuel waste both in Canada and other countries; the potential social, economic, and environmental effects of waste disposal; and the potential impact of recycling and other processes on waste volume. A general review of other aspects of the nuclear industry, such as energy policy and reactor operation and safety, is specifically excluded from the Panel's review.

All federal departments with a relevant interest in the concept are expected to participate in the review process. These include the Atomic Energy Control Board (AECB), NRCan, Environment Canada, Health and Welfare Canada, and Transport Canada. NRCan has assembled a team to review the results of AECL's R&D program, and Environment Canada has assembled two teams of experts to review in detail how well the concept protects the environment.

When the EARP review is concluded, the Panel will make recommendations as to the acceptability of the concept and the course of future action regarding nuclear fuel waste disposal. Government decisions will then follow.

In the spring of 1990, CEAA organized a series of "Open Houses" to inform interested parties, not directly connected with the nuclear industry or with the scientific review process, about how they could take part in the review. "Scoping Hearings" took place in the autumn of 1990 to identify issues of concern, and to assist the Panel in setting guidelines for the EIS. One hundred and thirty participants made presentations, including government departments, scientific and business organizations, special interest groups, and private individuals. Among the major issues raised were arguments for and against storage as compared with disposal, the adequacy of the regulatory criteria, and monitoring the performance of the disposal vault. Aboriginal land claims affect much of the land where a disposal vault could be sited. In view of this, an aboriginal representative was added to the Panel.

In June 1991 the Panel issued draft Environmental Impact Statement guidelines for comment. Over thirty different groups and individuals submitted comments. The Panel issued its final guidelines to AECL in March of 1992.

AECL responded to these guidelines by preparing and issuing the Environmental Impact Statement and nine primary reference documents to support the

Environmental Impact Statement. These documents provide a complete description of the concept and the technology that has been developed over the past 15 years. The EIS also provides additional information specifically requested by the Panel.

The nine primary references were issued in 1993 and 1994. The Environmental Impact Statement on the Concept for Disposal of Canada's Nuclear Fuel Waste, and the Summary of the Environmental Impact Statement on the Concept for Disposal of Canada's Nuclear Fuel Waste were completed and submitted to the EARP Panel in 1994 October. The Panel released these documents to the public and initiated a nine month review period for the public to assess their completeness and provide comments to the Panel. By 1995 January 31, AECL had distributed about 18,500 copies of these documents.

From 1994 November to 1995 March, CEAA conducted another series of Open Houses in 21 communities across the review provinces; New Brunswick, Quebec, Ontario, Manitoba and Saskatchewan. These were designed to familiarize the public with the review process, the disposal concept, and nuclear fuel waste transportation and storage and to encourage their participation in the review of the completeness of the EIS and in the Public Hearings announced for later. At many of the Open Houses, short formal presentations were made by CEAA, AECL and Ontario Hydro staff. These were often followed by a question period. About 2750 people attended the presentations or visited the AECL exhibit.

The environmental assessment panel released its approach for the public hearings in 1994 August. The approach is divided into three phases:

1. Phase I is designed to assist the panel addressing issues in the panel's terms of reference which go beyond the generic concept for deep geologic disposal including: the criteria by which safety and acceptibility of a concept for long-term management and disposal should be evaluated; the degree to which this generation should relieve future generations of the burden of caring for the waste; social, economic and environmental implications of a possible nuclear fuel waste management facility; the general criteria for site selection and a future site selection process; and the potential costs and benefits to potential host communities. This phase is scheduled for 1996 March and April in Toronto and other communities in Ontario.

2. Phase II of hearings will focus specifically on scientific and technical issues related to the safety of the AECL's generic concept for deep geologic disposal of nuclear fuel waste. This phase is scheduled for 1996 June in Toronto.

3. Phase III hearings will be held over six weeks in the autumn of 1996 in a number of communities in five provinces previously visited by the panel during the scoping phase of this review. This phase will involve presentations on: recommendations to assist governments in reaching decisions on the acceptability of the disposal concept; steps to be taken to ensure safe long-term management of nuclear fuel waste; criteria by which the safety and acceptability of a concept for long-term waste management and disposal should be evaluated; social, economic and environmental implications of a possible nuclear fuel waste management facility, including the impact of transportaion of nuclear fuel waste; general criteria for site selection and on a future site selection process; and the costs and benefits to potential host communities.

5.4 THE ENVIRONMENTAL IMPACT STATEMENT AND PRIMARY REFERENCES

The Environmental Impact Statement document provides an overview of AECL's case for the acceptability of the disposal concept, and provides information about the following topics:

- the characteristics of nuclear fuel waste;
- storage and the rationale for disposal;
- major issues in nuclear fuel waste management;
- the disposal concept and implementation activities;
- alternatives to the disposal concept;
- methods and results of the environmental assessments;
- principles and potential measures for managing environmental effects; and
- AECL's overall evaluation of the disposal concept.

The nine Primary References expand on particular socioeconomic and technical aspects AECL has evaluated in developing the concept:

Public Involvement and Social Aspects

- describes the activities undertaken to provide information to the public about the program and to obtain public input into the development of the disposal

concept;
- presents the issues raised by the public and how the issues have been addressed during the development of the disposal concept or how they could be addressed during the implementation of the disposal concept; and
- discusses social aspects of public perspectives on risk, ethical issues associated with nuclear fuel waste management, and principles for the development of a publicly acceptable site selection process.

Site Screening and Site Evaluation Technology

- discusses geoscience, environmental, and engineering factors that would need to be considered during a siting process; and
- describes the methodology for site characterization, that is, for obtaining the data about regions, areas, and sites that would be needed for facility design, monitoring, and environmental assessment.

Engineered Barriers Alternatives

- describes the characteristics of nuclear fuel waste;
- describes the materials that were evaluated for use in engineered barriers, such as containers and vault seals;
- describes potential designs for containers and vault seals; and
- describes procedures and processes that could be used in the production of containers and the emplacement of vault-sealing material.

Engineering for a Disposal Facility

- discusses alternative vault designs and general considerations for engineering a nuclear fuel waste disposal facility;
- describes a reference disposal facility design that was used to assess the technical feasibility, costs, and potential effects of disposal (The term "reference" is used to designate the disposal systems, including the facility designs, specified for the assessment studies. Different disposal facility designs are possible and might be favoured during concept implementation.); and
- presents cost and labour estimates for implementing the reference design.

Preclosure Assessment of a Conceptual System

- describes a methodology for estimating effects on human health, the natural environment, and the socioeconomic environment that could be associated with siting, constructing, operating (including transporting used fuel), decommissioning, and closing a disposal facility;
- describes an application of this assessment methodology to a reference disposal system;
- discusses technical and social factors that would need to be considered during siting; and
- discusses possible measures and approaches for managing environmental effects.

Postclosure Assessment of a Reference System

- describes a methodology for:
  - estimating the long-term effects of a disposal facility on human health and the natural environment,
  - determining how sensitive the estimated effects are to variations in site characteristics, design parameters, and other factors, and
  - evaluating design constraints; and
- describes an application of this assessment methodology to a reference disposal system.

The Vault Model for Postclosure Assessment

- describes the assumptions, data, and models used in the postclosure assessment to analyze processes within and near the buried containers of waste; and
- discusses the reliability of the data and models.

The Geosphere Model for Postclosure Assessment

- describes the assumptions, data, and models used in the postclosure assessment to analyze processes within the rock in which a disposal vault is excavated; and
- discusses the reliability of the data and models.

The Biosphere Model, BIOTRAC, for Postclosure Assessment

- describes the assumptions, data, and models used in the postclosure assessment to analyze processes in the near-surface and surface environment; and
- discusses the reliability of the data and models.

The EIS and the nine Primary References comprise some 6000 pages.

5.5 Reasons for Confidence in the Disposal Concept

Our confidence in the long-term safety of the concept

draws strength from a number of sources:

1. The technical approach, the use of multiple barriers for redundancy and defence in depth;
2. The adoption of an observational approach to site characterization and to disposal vault design, construction, operation and eventually closure;
3. An approach to the project which is based on ongoing review and decision-making and which recognizes that, throughout, the process must be flexible and responsive and that decisions can be modified; and
4. Active and effective involvement of the public in the process.

5.6 THE MULTIBARRIER SYSTEM

In common with the approach adopted in other countries, the concept developed by AECL involves isolating the waste from the biosphere by a series of engineered and natural barriers. AECL's approach to development of the disposal concept has been to consider the performance of the system as a whole, rather than focusing on performance requirements for individual components. This approach allows flexibility in implementation to be retained and it increases the likelihood of identifying any counterintuitive interactions or synergisms among system components that could adversely affect safety. Thus, the performance of individual components, such as waste containers, is analyzed in the context of the system. This contrasts with a design and safety approach that prescribes performance standards for individual components and evaluates safety by the analysis of the performance of each component independently. Our goal, therefore, has been to develop a thorough scientific understanding of the performance of the different components of a disposal system and how these components interact and influence one another, so that the overall system can be designed to provide defence in depth.

Acquiring and building the necessary knowledge base is a continuing process, and in implementing disposal, flexibility must be retained so that new information and understanding acquired over time can be integrated into the disposal system. An example is container life-time. The original target was to achieve a minimum container life-time of 500 years and early work established that this goal could be achieved with a thin-walled titanium container. Subsequent studies on the corrosion of titanium and copper indicated that, for the expected groundwater chemistry, thin-walled titanium containers can be designed to have a corrosion lifetime in excess of tens of thousands of years, and a 25 mm thick copper container can potentially provide containment in excess of 10^6 years. Such advances in understanding and in our ability to defend, scientifically, this understanding can have a profound impact on the approach taken to facility design and implementation, and on decision-making.

Much of the evidence needed to evaluate any site that would be considered for deep geological disposal can be obtained from geologic information developed as the site is characterized, i.e., from the record of past changes preserved in the native rock mass and the groundwater. In the Canadian Shield the record available for investigation can be as long as two billion years.

Investigations at our field research areas, for example, indicate that discrete zones of intensely fractured rock, intersecting otherwise sparsely fractured rock, are the dominant pathways for groundwater flow at depths greater than 300 m to 500 m in plutonic rock of the Canadian Shield. The flux of groundwater can be high in the fracture zones; however, the flux is very low in the sparsely fractured rock bounded by the fracture zones, due to its very low permeability, which is commonly less than 10^{-18} m^2. Such low permeabilities can limit the rate of contaminant movement and indicate that within the overall disposal system, the host rock can play an important role as a natural barrier to the transport of radionuclides.

The field evidence is supplemented and complemented by understanding derived from laboratory studies, Field studies, including studies of natural analogues, can extend the short-term evidence from the laboratory studies to the longer times of interest - tens and even hundreds of thousands of years - and provide verification of the understanding incorporated in predictive modeling. For example, studies of the Cigar Lake uranium deposit in northern Saskatchewan have been under way since 1984. The uranium ore in the deposit has essentially the same composition as used fuel. It was formed some 1.3 billion years ago and has been in contact with groundwater since its deposition. Yet the uranium has remained stable under the reducing conditions prevailing in the deposit. Similar conditions are expected to occur in a disposal vault.

5.7 THE OBSERVATIONAL METHOD

No organization in Canada has yet been given the mandate to proceed with siting a disposal facility. Nevertheless, we can anticipate that the approach that will be used in site characterization and disposal vault

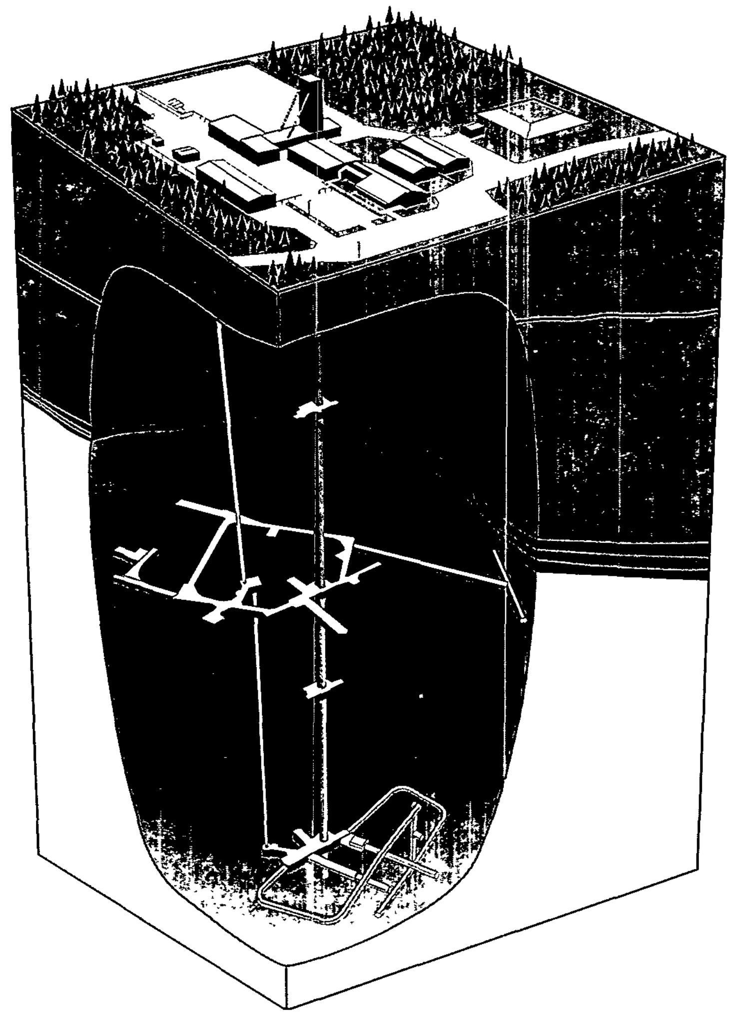

Figure 5.2. Underground research laboratory.

design, construction, operation, and eventually closure will be based on the observational method - the approach used in good geotechnical engineering. The observational method is a systematized approach to dealing with practical problems encountered when engineering in the sub-surface. In this approach, the results of modeling and computations during design are viewed as working hypotheses subject to confirmation or modification during construction. This approach provides a framework for decision-making in a situation where it is not practical, even if it were possible, to obtain all the detailed geotechnical information that would be needed for design of an underground facility prior to excavation at the site. As the project proceeds, the information is continuously acquired and incorporated into the design. A lack of detailed knowledge about local variability within a selected rock mass prior to excavation is common in underground construction. Nevertheless, major projects such as underground powerhouses and storage chambers, transportation tunnels, and dams are completed successfully using a design approach that accommodates observations made as construction is advanced. As part of its research program, AECL has constructed an Underground Research Laboratory for large-scale testing and *in-situ* engineering experiments on aspects of disposal (Figure 5.2). The observational method was used during the design and construction of this facility.

The observational method is also central to the use of performance assessment analyses as part of the design and implementation process. It is a continuously applied, iterative process. Beginning during the site selection phase, assessments are made based on all available data on the site conditions. The understanding of the site is incorporated into models for use in design and in performance assessment studies. Both the designs and the assessments become more refined as the knowledge of a site increases. Design and operational deci-

sions are made on the basis of the understanding of site conditions at the time. The maximum possible flexibility is retained to incorporate technological improvements. In addition, the potential impacts of conceivable deviations from site conditions as understood need to be assessed and contingency measures established, in advance, to address the deviations should they be encountered.

As work proceeds, observation and evaluation of the actual conditions encountered are compared with the previous understanding and, if necessary, the detailed design and the models used in performance assessment are modified. This cycle continues throughout site selection, construction and operation, so that at each point when significant licensing and operational decisions need to be made, a long record of observation and a series of increasingly refined performance assessments are available on which to base the decision.

5.8 ONGOING REVIEW AND DECISION-MAKING

AECL views the current review as the beginning of a continuing process. As the technology for managing the disposal of nuclear fuel waste is developed and applied to specific sites, further reviews and public consultation and involvement will be needed. Any facility will be subject to rigorous regulatory criteria, and it is anticipated that society will demand that a step-by-step process be followed. Thus, a decision to proceed on the basis of the current review would represent only the first of a series of decisions between distinct phases of the process.

Each phase should lead to increased confidence in the overall system, thus facilitating decision-making about how and whether to proceed to the next phase. We are currently nearing the end of the first phase - concept development and assessment. If the Panel shares AECL's view that we have adequately developed the concept, and there is a governmental decision to proceed, the next appropriate step would be the start of site-specific activities, beginning with site screening. The sequence of events could be as follows:

1. Site screening would lead to the selection of one or more sites for detailed characterization based on surface techniques;
2. Such site characterization studies could lead to a selection of one or more sites for exploratory excavation and more extensive in-ground characterization;
3. In-ground characterization could lead to a decision to initiate construction and operation of a disposal vault, possibly beginning with a demonstration phase;
4. Design, construction and operation of a facility would involve ongoing review, reassessment and recommitment, leading to continued operation and then eventually to a decision to cease operations and decommission; and
5. Decommissioning and post-operational monitoring would ultimately lead to a decision to close and seal the vault.

The process of site screening and of evaluating several sites will likely involve a further ten to fifteen years of work before a commitment would be made to initiate an underground excavation, followed by a further ten years or so of site exploration and characterization before construction could begin. Thus, waste would not be emplaced in a vault before about 2025. By then we would have accumulated many years of site-specific data and a series of increasingly refined evaluations on which to base a decision to begin to emplace waste.

The decision to close and seal the vault would be made on the basis of the accumulated evidence and experience gained throughout the siting, characterization and operational phases, a process extending over close to a century. Only with that decision will disposal based on the concept have definitively been judged as safe and acceptable.

Thus, at the current concept assessment phase of the process, "concept approval" represents a judgment that:

- sufficient understanding has been developed to continue with the process, with an expectation that we will eventually reach the end point of sealing a vault; and that
- at the appropriate time, we should proceed to the next phase of the program, the beginning of site-specific activities to resolve outstanding issues that can only be resolved on a site-specific basis.

5.9 PUBLIC INVOLVEMENT

The general public and potential host communities are important constituencies which contribute to the decision-making when identifying options for waste management. Building public confidence in a program is therefore an important part of its development. The process to be followed in reviewing the program and deciding on future steps should involve consultation with, and the active participation of, the communities and public affected. Decision-makers need to have a

mechanism to take public concerns into account when advancing major projects such as a disposal facility.

In Canada a formal mechanism for public involvement in the early part of project development is defined in environmental assessment and review legislation. The objective is to establish the scope of public concerns and interest early in the planning stage of a project so that steps can be taken to address the concerns in the project design. The public is asked to formally participate in the assessment and review and may be provided with the funds to do so.

In the Canadian program, no site will be selected for a disposal facility until the technology has first been evaluated in an environmental review. This review is currently under way. Because no directly affected community exists, public involvement at this stage is necessarily very broadly based. As part of concept development, AECL has carried out a public interaction program with the objectives of providing information to the general public and to those groups which have shown a particular interest in the program. At the same time we have endeavoured to identify the issues of concern to the public and to address these in the documentation describing the technology and our approach to disposal.

If the environmental review leads to a decision to proceed toward selecting a site, we anticipate that public involvement will continue and that it will become more community-specific. Beginning with siting, the organization selected to implement disposal, the implementing organization, would need to develop and maintain effective working relationships with potential host communities and with communities along potential transportation corridors. For these relationships to be effective, the implementing agency must demonstrate a commitment to principles of fairness, openness, shared decision-making, and above all to safety, so that affected communities can participate fully in the decision-making and so become empowered.

AECL proposes that the implementing organization adhere to the following principles:

Safety and Environmental Protection

In addition to complying with all applicable legislation, the implementing organization would keep adverse effects on human health, the natural environment, and the socioeconomic environment as low as reasonably achievable, taking social and economic factors into account.

Voluntarism

No community would be forced to host a disposal facility. A community would have the right to determine whether or not it was willing to be a host community.

Shared Decision Making

Implementation of the concept would occur in stages and would entail a series of decisions about whether and how to proceed. Each potential host community, and eventually the host community, would share in the decision making. In addition, the implementing organization would seek and address the views of other potentially effected communities.

Openness

Throughout the project, the implementing organization would offer information to the general public about its plans, procedures, activities, and progress. In addition, potentially affected communities would have access to all available information required to make a judgment about safety and environmental protection.

Fairness

In accepting a disposal facility, the host community would provide a significant service to the consumers of nuclear-generated electricity and to the public at large. In fairness, the net benefit to the host community should be correspondingly significant. The net benefit is not intended to induce a community to accept an unsafe disposal facility. As part of the negotiated program for managing environmental effects, measures would be taken to avoid, mitigate, or compensate for adverse effects; such measures would be enhanced or additional measures taken to ensure the betterment of the host community. Fairness also requires "due process," which would be provided by adherence to the principles of voluntarism, shared decision making, and openness.

The principle of safety and environmental protection would not be compromised, no matter how acceptable or desirable a site might be in all other respects.

5.10 CONCLUSIONS AND RECOMMENDATIONS

AECL believes that it has developed a robust and flexible concept for disposal of nuclear fuel waste that will

meet the regulatory requirements of Canada. In our EIS we conclude that:

1. Implementation of the disposal concept would protect human health and the natural environment from the potential adverse effects of nuclear fuel waste far into the future. In addition, human heath and the natural environment would be protected while the disposal concept was being implemented.
2. The disposal concept provides a means of minimizing the burden on future generations.
3. The disposal concept provides scope for public involvement during implementation.
4. Of the options that have been considered internationally, only geological disposal is a viable alternative for the disposal of Canada's nuclear fuel waste using currently available or readily achievable technology. The choice of plutonic rock of the Canadian Shield as the preferred disposal medium, made in the late 1970s, was appropriate, and plutonic rock should remain the preferred disposal medium for Canada.
5. The disposal technology does not rely on institutional controls as a necessary safety feature; it is adaptable to a wide range of physical conditions and to potential changes in criteria, guidelines and standards, and it includes monitoring and retrievability.
6. The methodology to evaluate safety of a disposal system against established safety criteria, guidelines, and standards has been developed and demonstrated to the extent reasonably achievable in a generic research program.
7. Technically suitable sites are likely to exist in Canada.

We are confident that implementation of our disposal concept represents a means by which Canada can safely disposal of its nuclear fuel waste.

We are continuing research and development work to ensure the public and the industry have as much confidence as possible in the safety of the concept and in the feasibility of implementing it.

The process for a federal environmental review of the concept is well under way. The review of a concept as opposed to a site- and design-specific project requires focusing on whether it is appropriate to proceed with the first phase of implementation. We believe that we have reached the stage in the NFWMP where the greatest benefit will result if activities proceed on a site-specific basis. Therefore, we have made the following recommendations in the EIS:

1. We recommend that the strategy for long-term management of Canada's nuclear fuel waste be based on the concept of disposal in plutonic rock of the Canadian Shield.
2. We recommend that those who have responsibility for the safe management of used fuel - the federal government and owners of the used fuel - also have responsibility for implementing the concept. In addition to addressing their requirements, the plan for implementation should address the requirements of any provincial government that could be affected by implementation, and those resulting from the present environmental review.
3. We recommend that those responsible for implementing the disposal concept be committed to the principles of safety and environmental protection, voluntarism, shared decision making, openness, and fairness.
4. We recommend that Canada progress toward disposal of its nuclear fuel waste by undertaking the first stage of concept implementation - siting.

We are in a very public process. Our experience has shown us that such processes are not easy for the nuclear industry. We believe that the information included in the EIS should lead the Panel to recommend that we proceed to the next phase of the process leading toward disposal. Our confidence is founded on the strength and depth of our technical program and on a well-founded public consultation program.

CHAPTER 6

DEEP GEOLOGICAL DISPOSAL OF HIGH LEVEL RADIOACTIVE WASTE IN CHINA

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6.1 INTRODUCTION

China is now facing the challenge of how to safely dispose of nuclear waste. China's nuclear industry was first established in 1955, five years after the birth of the People's Republic of China. With the development of nuclear facilities, more and more radioactive waste has been generated. Much of liquid high level waste has been accumulated to date (Luo et al, 1995), and it is stored in stainless steel casks, waiting for vitrification.

On the Chinese mainland, there are two nuclear power plants (NPPs) in operation: the Qinshan NPP in east China's Zhejiang province and the Daya Bay NPP in south China's Guangdong province (Fig. 6.1). In the next five years, four more NPPs (8 units) will be built (Fig. 6.1); their estimated capacities are listed in Table 6.1. China has plans for a significant nuclear energy development. The total electrical capacity produced by the NPPs is planned to reach 20,000 MW by 2010. In the near future, China is also facing the problem of disposing of the spent nuclear fuel from the NPPS. At present, the national policy is that the spent fuel should be reprocessed before final disposal.

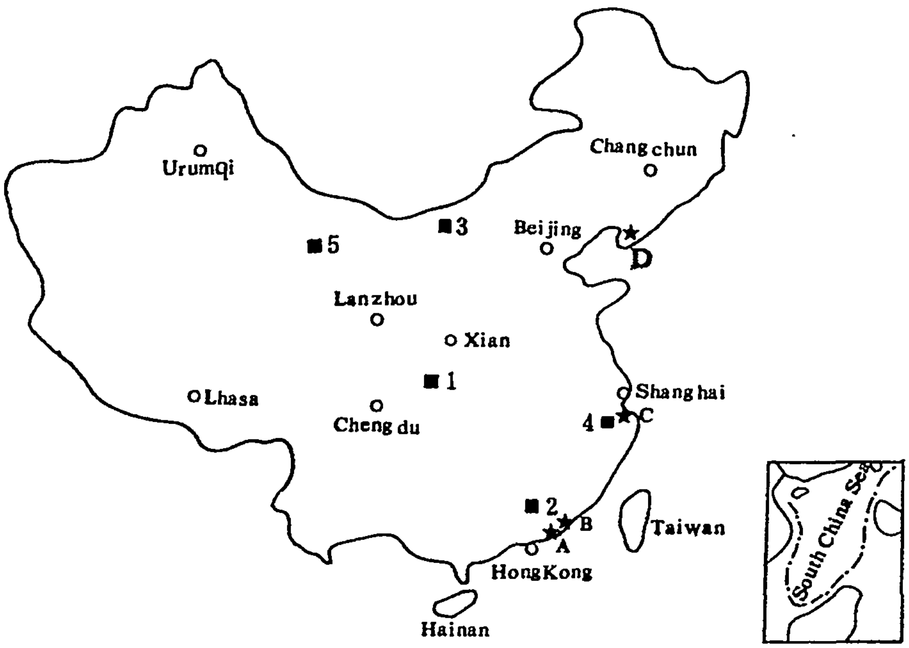

Figure 6.1. Map showing distribution of nuclear power plants (NPPs) on Chinese mainland. Legend: A - Daya Bay NPP; B - Lin'ao NPP; C - Qinshan NPPs (Phases 1, 2, and 3); and D - Liaoning NPP. Preselected regions for development of HLW repository: 1 - southwest China (granite, shale): 2 - Guandong area (granite); 3 - Inner Mongolia (granite); 4 - east China (granite, tuff); and 5 - northwest China (mudstone, shale, granite).

Table 6.1. Nuclear power plants of China.

	Name of NPP	Province	Capacity (MW)	Remarks
1	Qinshan	Zhejiang	300 x 2	in operation
2	Qinshan (2nd phase)	Zhejiang	600 x 2	under construction
3	Qinshan (3rd phase)	Zhejiang	700 x 2	to be built
4	Daya Bay	Guangdong	900 x 2	in operation
5	Lin'ao	Guangdong	1,000 x 2	under construction
6	Liaoning	Liaoning	1,000 x 2	to be built

6.2 ORGANIZATION

In China, work related to radioactive waste disposal is managed by the China National Nuclear Corporation (CNNC). The organizational structure for the disposal of high level radioactive waste is shown in Figure 6.2.

The China Environmental Protection Agencyis responsible for the issuance of related regulations, the final review of environmental impact statements, and the licensing for the construction and operation of a repository. The China National Nuclear Safety Administration is responsible for the safety issues related to the dispos-

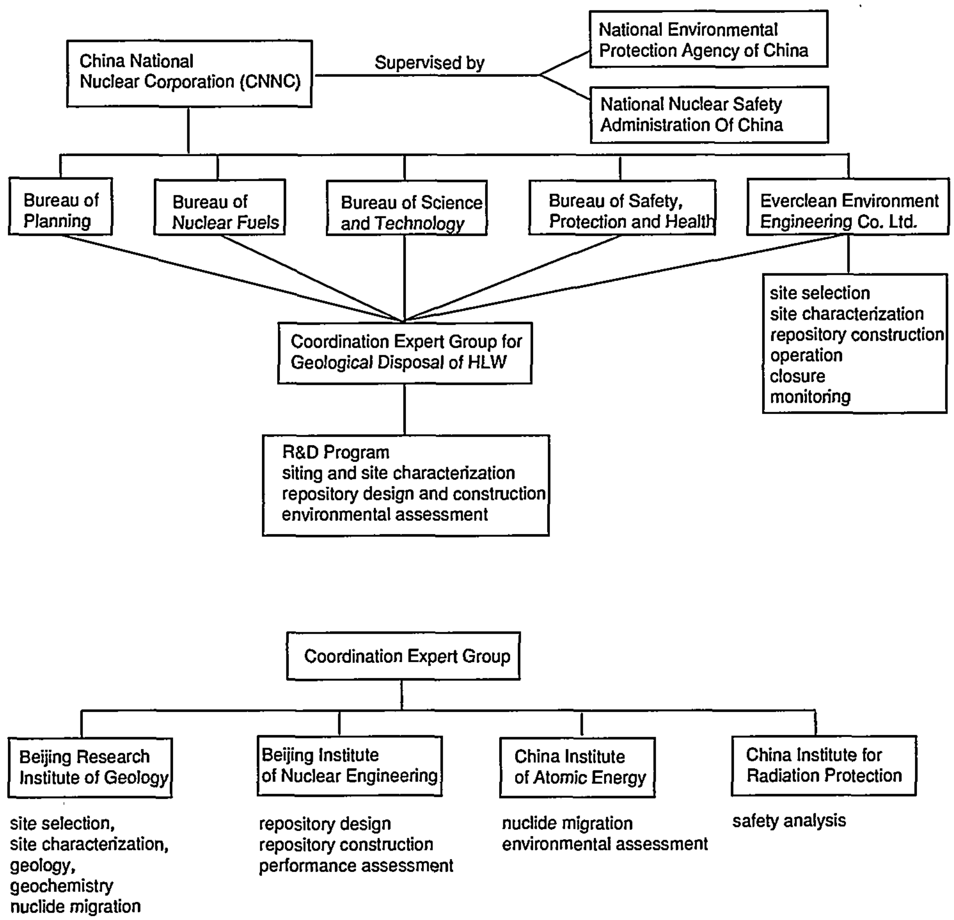

Figure 6.2. China's organizational structure for the disposal of high level radioactive waste.

al of high level waste.

The CNNC is responsible for the transportation of HLW and spent fuels, reprocessing of spent fuels, vitrification of liquid HLW, and final geological disposal of HLW. Under CNNC, four bureaus are involved in the disposal process: Bureau of Planning, Bureau of Nuclear Fuels, Bureau of Science and Technology and Bureau of Safety, Protection and Health.

The Everclean Environment Engineering Corporation, attached to CNNC, is responsible for site selection, site charactererisation, construction, operation, closure and monitoring of repositories of low and intermediate level, and high level radioactive waste.

A Coordination Expert Group was organized for the geological disposal of HLW in 1986. The group is composed of experts from the Beijing Research Institute of Geology, Beijing Institute of Nuclear Engineering China Institute of Atomic Energy, and China Institute for Radiation Protection. The group is responsible for R & D programs, site selection, research work related to site characterization, repository design, environmental assessment, safety analysis and performance assessment.

6.3 DEEP GEOLOGICAL DISPOSAL PROGRAM

In 1985, CNNC worked out an R & D program which has been called the Deep Geological Disposal (DGD) of HLW (Yang, 1992). The program is divided into 4 phases:

Phase 1: 1985-2025 Site Selection and Site Characterization

1. 1985-1986, Nationwide screening;
2. 1986-1988, Regional screening;
3. 1989-2010, District screening and preliminary site characterization;
4. 2011-2015, Pre-feasibility study of the pre-selected district;
5. 2015-2017, Licensing for site characterization;
6. 2017-2023, Site characterization and suitability study; and
7. 2023-2025, Licensing for the site.

Phase 2: 2025 - 2029, Repository Design

Phase 3: 2041-2050, Repository Construction

Phase 4: 2051- , Repository Operation.

Between Phase 2 and Phase 3, there is a 10-Year interval, during which an underground research laboratory will be built near the selected site between 2028 and 2033. A full-scale *insitu* test and disposal demonstration will be conducted in the laboratory between 2034 and 2040. The final repository design will be revised after the work in the underground research laboratory is finished.

According to China's policy, the final form of the disposed waste will be vitrified radiowastes after reprocessing. The conceptual design for the HLW repository will be a shaft-tunnel model, which will be located in the saturated zone in granite.

6.4 PROGRESS IN SITE SELECTION

According to the DGD Program, the site selection for China's HLW repository is in progress, and granite is considered as the candidate host rock for the repository. The site selection process, started in 1985, is composed of 4 stages: nationwide screening, regional screening, district screening and site screening. At present, the third stage has been reached, and the screening efforts are focused on the Beishan area, Gansu province in northwest China.

6.4.1 Siting Criteria

Because of the high radiotoxicity and long half-life of radionuclides in high level waste, sites for HLW repositories should be selected very carefully. Siting guidelines, or siting criteria, have been issued by many countries or international organizations, such as USA, Sweden, EC, IAEA, etc. Taking China's situation into account, as well as experience in other countries, we proposed preliminary siting guidelines that have been followed during our work (Xu, 1992).

The general siting principle is that, under the effects of natural and human activities, the long term (100,000 year) safety of the repository can be obtained, and the disposed radioactive wastes can be prevented from entering the biosphere and harming human beings. Furthermore, the following factors are considered in the siting process. However, detailed and exact technical requirements and limits for these natural factors have not been worked out.

Social Factors

- the distribution of nuclear industry in China;
- the animal and plant resources, the potential mineral

resources;
- the attitude of the public and the local government;
- the requirement of national environmental protection laws; and
- the feasibility for construction and operation of the repository.

Natural Factors

- natural geography, including topography, climate, hydrology etc.;
- geology, including crustal stability (earthquakes, active faults, etc.); and
- crustal stress, crustal thermal flow, host rock type, hydrogeology and engineering geology.

6.4.2 Progress in Site Selection

Since 1985, the site selection for China's HLW repository has progressed through three stages: (1) nationwide screening (1985-1986); (2) regional screening (1986-1988); and (3) district screening (1989-present).

Nationwide Screening (1985-1986)

During this stage, the first work was the investigation of the site selection process conducted in other countries. Then, based on these experiences, and considering the distribution of China's nuclear industry, the problems of crustal stability and social economic conditions, the following five regions were selected for the repository (Fig. 6.1): southwest China, Guangdong Area, Inner Mongolia, east China, and northwest China. In this stage, granite, tuff, mudstone and shale were considered as candidate host rocks.

Regional Screening (1986-1988)

Based on the work of Stage 1, twenty-one districts were selected for further investigation within the five above mentioned areas:

1. *Southwest China.* Three districts were selected: Hanwangshan district, Zhongba district, and Hannan district. The potential host rocks are shale, biotite granite, and plagioclase granite, respectively.
2. *Guangdong Area.* Since the Daya Bay NPP is located in this area and more NPPs are planned to be built, it was considered a potential area for the HLW repository. The candidate geological formations are the Fuogang granite body and the Jiufeng granite body. However, because of the rapid development in the Guanadong province, this area may not be considered for further investigation.
3. *Inner Mongolia.* The selected districts, including Parjiang Haizhi and Dabaolitu districts, are located in central Inner Mongolia. The host rock is Hercynian granite.
4. *East China.* In east China's Zhejiang and Anhui provinces, six districts have been selected: Lin'an, Gaoyu, Chenshi, Jiangshan, Guangde and Yixian. The host rocks include granite and tuff. The Chenshi district is located on a small island composed of granite.
5. *Northwest China.* The selected area is located in northwest China's Gansu Province It is the most promising area for the construction of a HLW repository. It is an arid gobi area with very low population density and without an economic potential. A total of five districts have been selected for investigation:
   - Toudaohe-Xiatianjinwei, biotite monzonitic granite;
   - Kuangqu, mudstone;
   - Baiyuantoushan, quartz diorite;
   - Qianhongquan, K-feldspar granite and plagioclase granite; and
   - Jiujin, plagioclase granite.

District Screening (1989-present)

Since 1989, our work has been focused on northwest China, with granite considered as the candidate host rock. The following aspects of the area have been studied: earthquakes, structural framework, active faults, crustal stability, lithology, hydrogeology, and engineering geology. According to the crustal stability, we consider the Qianhongquan and Jiujin districts, located in the southern part of the Beishan Folded Belt, as the most promising districts, and further work will be conducted in those areas.

6.5 BEISHAN AREA, GANSU PROVINCE, NORTHWEST CHINA

6.5.1 Regional Geology

Regional Tectonics

The Beishan area in Gansu Province, is the preselected area for a HLW repository in China. Geographically, the Beishan area is located north of the town of Yumen in northwest Gansu province, China. Tectonically, it is located in the Erdaojin-Hongqishan compound anticline

of the Tianshan-Beishan folded belt (Fig. 6.3). The anticline strikes nearly east-west, with its core composed of schist, laminated migmatite of the Laojunmiao Group and Dakouzhi Group of pre-Changchengian age, and its limbs are composed of schist, marble, quartzite and migmatite of the Yujishan and Huayaoshan Groups of pre-Changchengian age. In addition, along the limbs, there are Jiujin Caledonian magmatite (plagioclase granite), Baiyuantoushan Caledonian magmatite (quartz diorite) and Qianhongquan Hercynian magmatite (monzonitic and orthoclase granites). These magmatite rocks are the main candidate host rocks for the repository.

The border between the Beishan area and the Hexi Corridor Transitional zone is the blind Sulehe fault. Within the Beishan area there are three east-west-striking large scale ductile shear zones (DSZ), namely: South Erduanjin, Zhongqiujin-Jinmiaogou, and Erdojin-

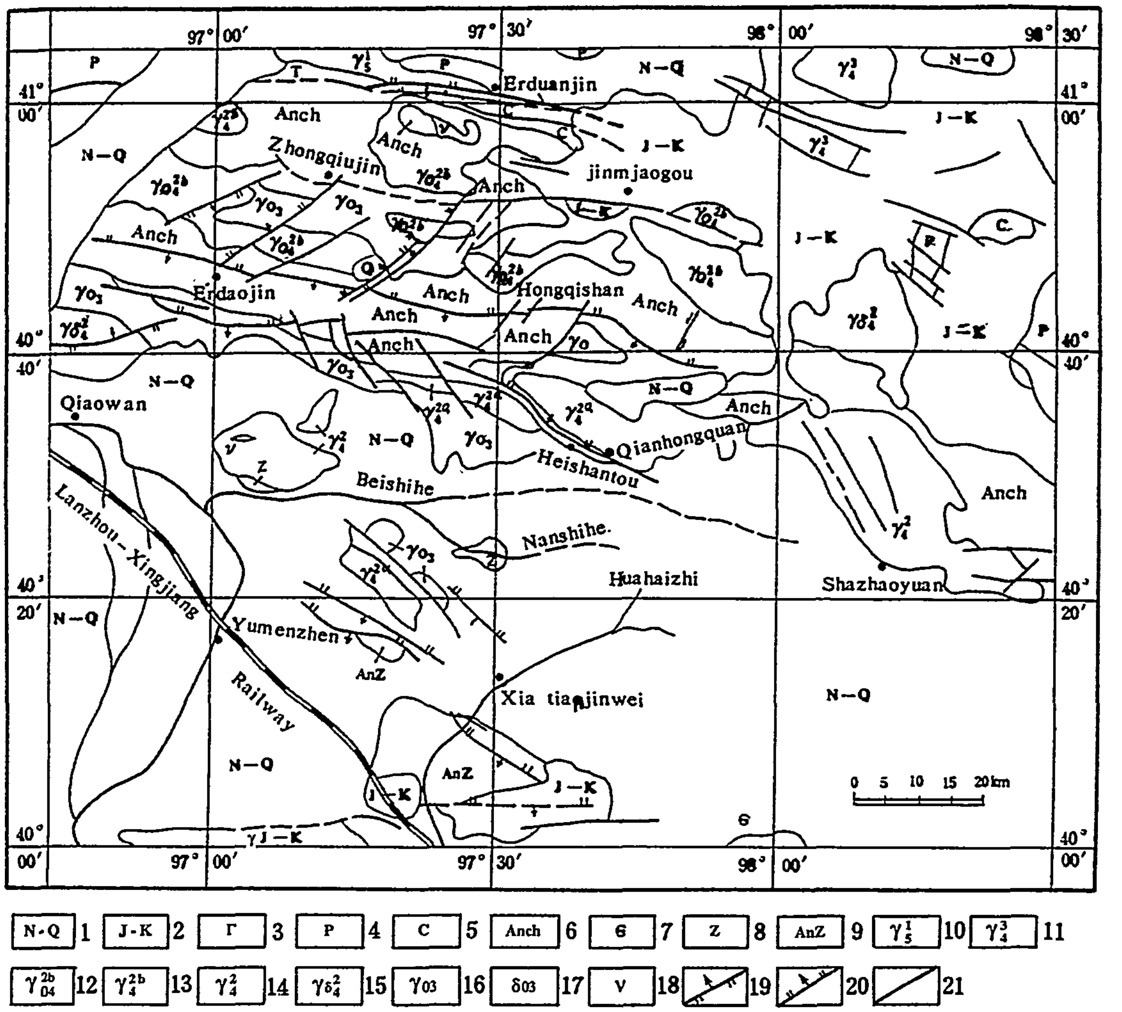

Figure 6.3. Geological sketch map of the Beishan area, Gansu Province, northwest China containing the pre-selected area for China's high level radioactive waste repository. Legend: 1 - Tertiary and Quaternary sediment; 2 - Cretaceous and Jurassic sandstone, shale and sandstone; 3 - Triassic conglomerate and pebbly sandstone; 4 - Permian System; 5 - Carboniferous system; 6 - Pre-Changchengian schist, gneiss, marble and migmatite; 7 - Cambrian System; 8 - Sinian System; 9 - Pre-Sinian System; 10 - Yanshanian granite; 11 - Heicynian granite; 12 - Heicynian plagioclse granite; 13 - Heicynian orthoclase granite; 14 - Heicynian Plagiolcase granite and two-mica granite; 15 - Heicynian granite diorite; 16 - Caledonian plagioclase granite; 17 - Caledonian quartz diorite; 18 - gabbro vein; 19 - normal fault; 20 - reverse fault; and 21 - fault.

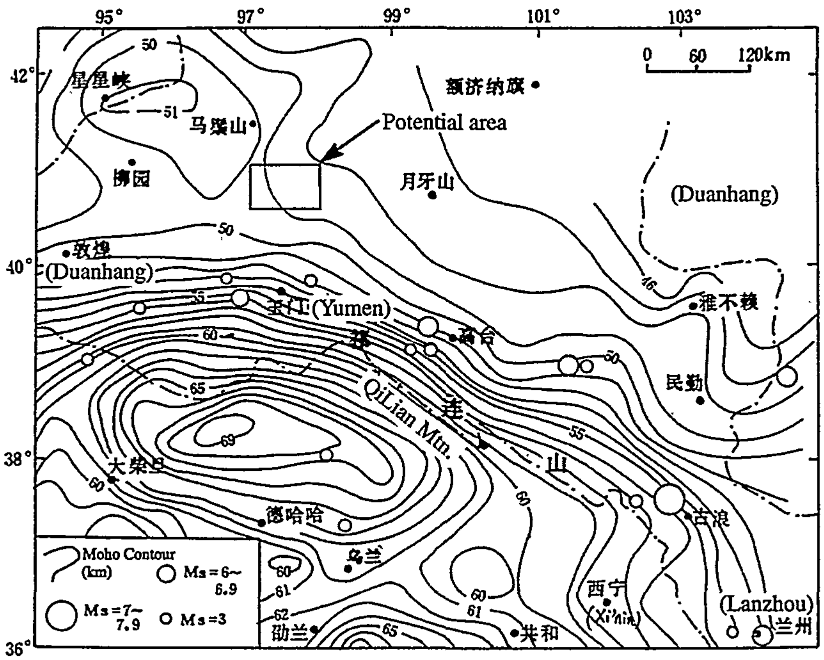

Figure 6.4. Moho discontinuity iso-depth contour map of northwest Gansu, China.

Hongqishan. The activity of brittle faults developed within the shear zones is a key factor affecting the crustal stability of the area.

The Zhongqiujin-Jinmiaogou DSZ is east-west striking with a length of 90 km and a width ranging between 50 and 2000 m. It was developed in monolithic gneiss, granite and sericite slate during the Hercynian period. The maximum depth of the shear zone is about 10 km. Several east-west striking brittle faults are developed in the shear zone. However, these faults do not cut the Quaternary sediments, and no earthquakes with Ms > $4^3/4$ have been recorded along them, which indicates that they are not active faults.

The Erdaojin-Hongqishan DSZ is about 130 km long and has a maximum width of 7 km. The southern and northern borders of the shear zone are two east-west striking brittle faults which are not active faults, either.

Crustal Structure

According to the main tectonic units, deep geophysical features and the crustal thickness, the crustal structure of the western Gansu province can be divided into the areas of Beishan-Alashan, North Qilian Mt., Hexi corridor, and Qilian. The selected area for the HLW repository is located within the Beishan-Alashan area. The depth contour of the crust in the area strikes NWW-EW. The thickness of the crust is 47-50 km (Fig. 6.4), and it gradually increases from north to south with very little variation. The gravity anomaly is -150 to -225x10^{-5} m/s^2 (Fig. 6.5) with a gradient that is less than 0.6 mGal/km. On the gravity anomaly map, the contours are very sparse without obvious step zones, indicating that there are no large faults extending to the depth of the crust. Based on these characteristics, the crust in the Beishan area possesses a block structure with good integrity.

Earthquake Activity

Earthquakes are a demonstration of modern crustal movement, and have a close relationship with tectonic movement, especially with the intensive activityof large deep-rooted faults. The Beishan area is located north of the Hexi Corridor earthquake zone. In this area, there is a lack of large active deep-rooted faults and strong earthquakes. According to data provided by the National Seismological Bureau of China, no earthquakes with Ms > $4^3/4$ have been recorded in the Beishan area. On the "Seismic Intensity Regionalization Map of China (1:4,000,000)", the Beishan area is within a VI seismic intensity region (Fig. 6.6).

In sharp contrast to the Beishan area, the Hexi Corridor earthquake zone has several active NWW deep-rooted faults. Along the North Hexi Corridor Great Fault, several intensive earthquakes took place, e.g., the M=7 1/4

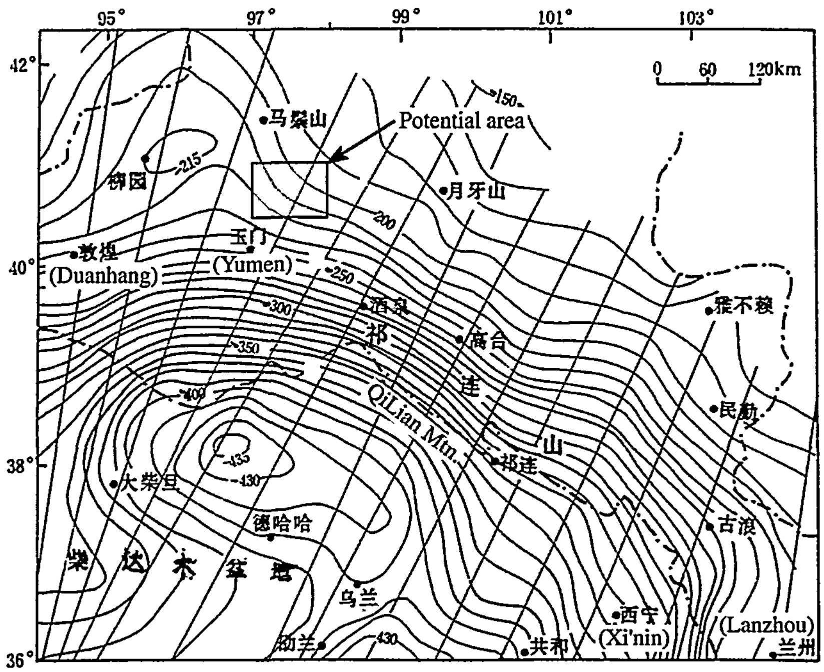

Figure 6.5. Regional magnetic anomaly map of northwest Gansu, China.

earthquake in Mingle County in 1790; the M=7 1/2 earthquake in Shandan County in 1954; and the M=6 earthquake in Gaotai County in 756.

Neotectonism and Tectonic Stress Field

Neotectonism refers to the crust tectonic movements since the Tertiary, including the horizontal and vertical movement of the crust, volcanism, earthquakes and slides, etc. According to the neotectonic characteristics, the western Gansu Province can be divided into three parts, namely: the Qilianshan blocking and intensive uplifting region; the Corridor depression region; and the Beishan weakly uplifting region (Fig. 6.7). The selected area for the HLW repository is located within the Beishan weakly uplifting region. The land form of the area is characterized by a flatter gobi and small hills with elevations above sea level, ranging between 1000 and 2000 m. The height deviation is usually several tens of meters. Since Tertiary time, it is a slowly uplifting area without obvious differential movements. The rate of uplift for the crust in the area is about 0.6 - 0.8 mm/a, much lower than that of the Qilian region (1.5-1.8 mm/a).

Comprehensive analysis of the structural deformation of the Cenozic faults and folds indicates that the area is undergoing horizontal compression at present, and the principal compressive stress is between 30 and 60 degrees. The data provided by the mechanism at the source of earthquakes show that the direction of the principal compressive stress is about 40 degrees.

The strike of the main faults in the Beishan area is between 95 and 120 degrees. The angle between the direction of the principal compressive stress and the strike of the main faults, which is also called the superimposed fault angle, ranges between 55 and 80 degrees. These are within the range of stable superimposed fault angles, suggesting that the main faults are stable and will not undergo strike-slip displacement.

6.5.2 Regionalization and Evaluation ofCrustal Stability

There are nine indices to evaluate the crustal stability of an area, namely: (1) crustal structure and deep-rooted faults; (2) Cenozoic crustal deformation; (3) Quaternary block and faults; (4) Quaternary and modern tectonic stress field; (5) Quaternary volcanoes and geothermal fields; (6) ground deformation and displacement; (7) gravity field; (8) earthquake strain energy; and (9) earthquakes. According to these indices, crustal stability can be divided into four classes: (1) stable region; (2) basically stable region; (3) sub-stable region; and (4) unstable region.

The western Gansu Province can be divided into three

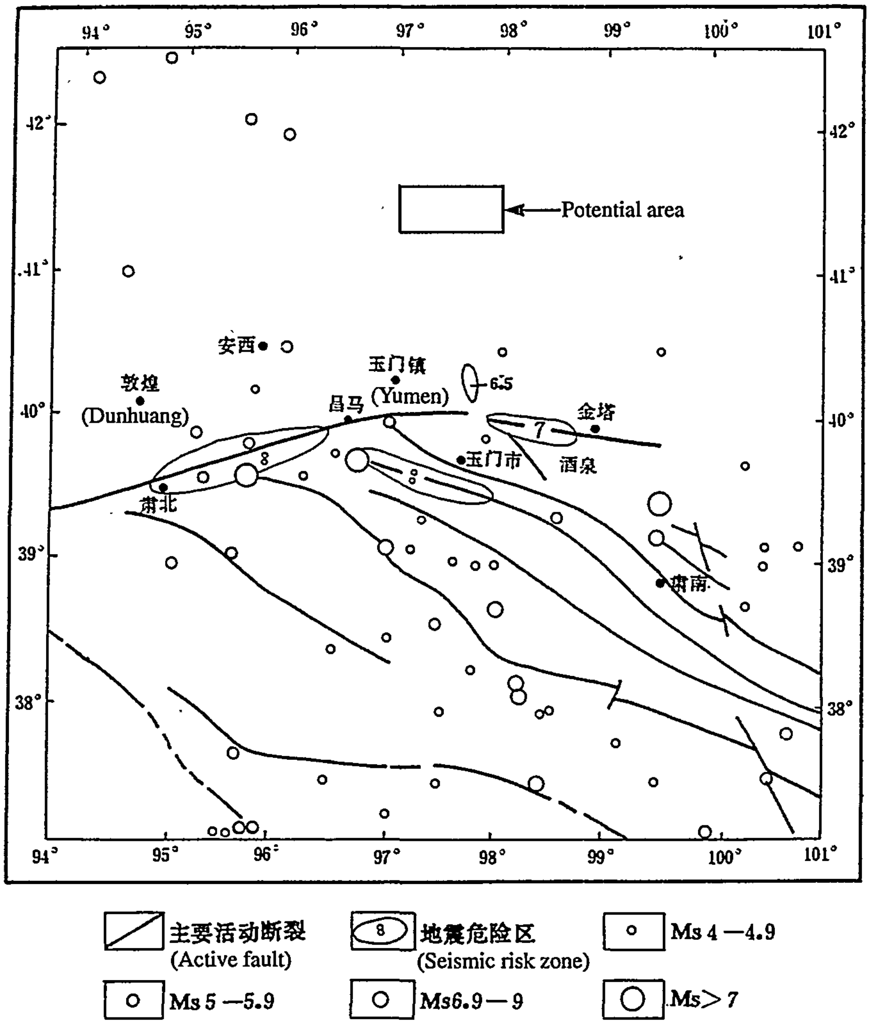

Figure 6.6. Seismic risk zoning of northwest Gansu, China.

regions (Fig.6.8): (1) Beishan stable region; (2) Yumenzhen-Huahai sub-stable region; and (3) Hexi Corridor unstable region. The characteristics of these regions are summarized in Table 6.2. From the table we can see that the Beishan region has a block crustal structure with good integrity and without regional active faults, and it is a slowly uplifting region. The direction of the principal compressive stress in the area is about 40 degrees, and the superimposed fault angles indicate that the main faults will not have strike-slipe displacement. There are no gravity steps in the area, and no records of intensive earthquakes either. The earthquake intensity of the area is less than 6. These features show that the crust in Beishan area is more stable than in the southern regions, and is a suitable candidate region for a HLW repository. On this basis, smaller districts can be further selected for detailed site characterization.

Geological work in the Beishan area is at the very beginning stage, and the above data are only preliminary results. In the future, systematic site characterization work will be conducted year-by-year, to determine the suitability of the Beishan area and to select the final site for a HLW repository.

6.6 OTHER STUDIES

The geological disposal of HLW is a long-term comprehensive task. In addition to site selection and site characterization, other studies are also being conducted on:

- Site preselection for an underground research laboratory;
- Experiments on radionuclide migration (laboratory and field);

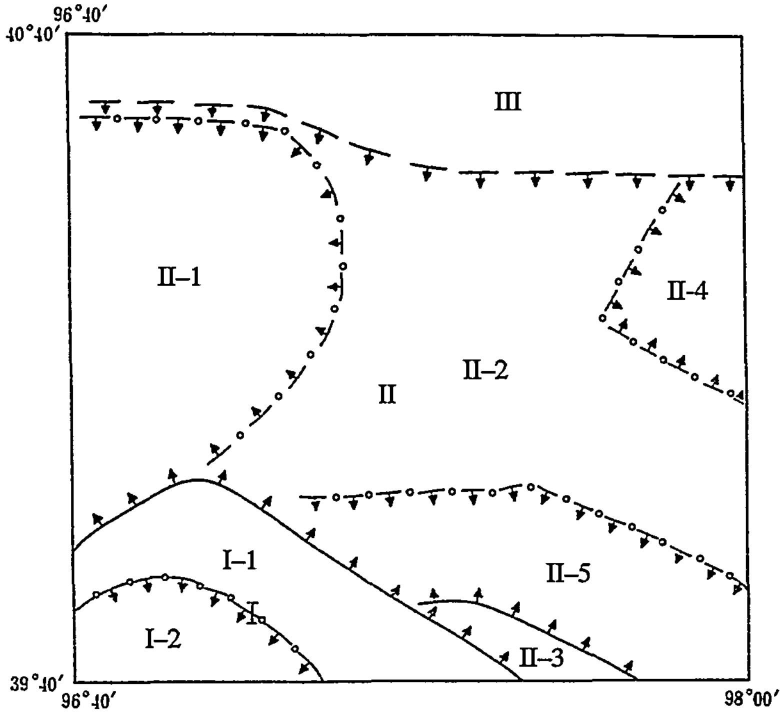

Figure 6.7. Map showing the regions of neotectonics. 1 - Qilian Mt. intensive uplifting region; II - Corridor depression region, and III - Beishan slight uplifting region.

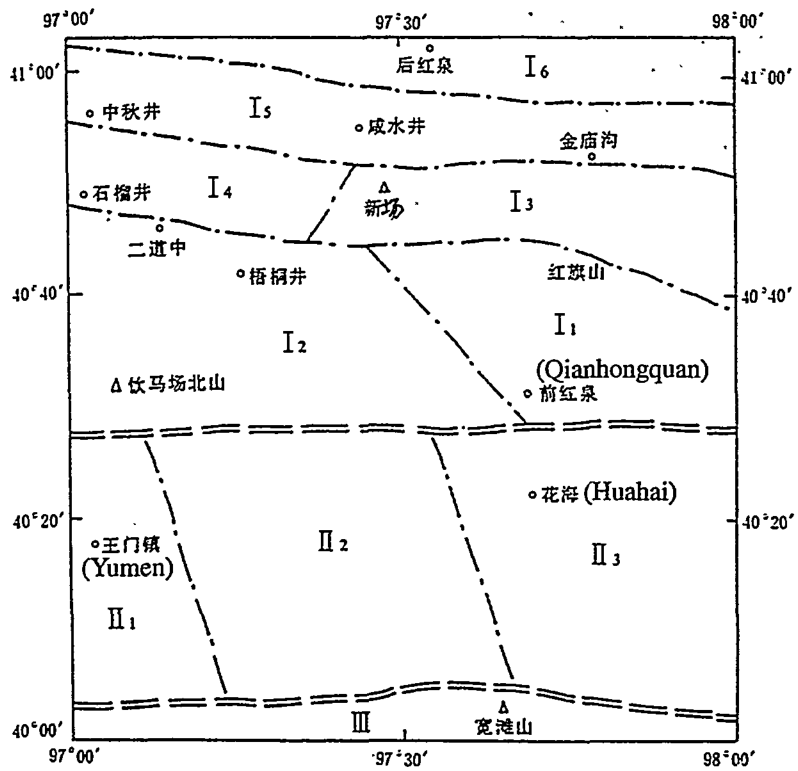

Figure 6.8. Map showing the classification of crustal stability of northwest Gansu Province. I - Beishan stable region; II - Yumenzhen-Huahai sub-stable region; and III - Hexi Corridor unstable region.

Table 6.2 Crustal characteristics in western Gansu Province, China.

Characteristics		Beishan Stable Region	Yumenzhen-Huahai Sub-stable Region	Hexi Corridor Unstable Region
Crust structure and deep fault		Crust is block structure with good integrity, NW-striking basement faults are distributed.	Crust is mosaic structure, with NE- and NW-striking basement faults.	Crust is crushed structure controlled by the north Qilian deep fault and Aljin active fault.
Active fault and Quaternary crustal movement		With few active faults, crustal movements of slowly uplifting region with less than 0.1 mm/a.	With Sanweishan and Aljin active faults, crustal movement is between 0.1 - 3.5 mm/a.	With Aljin active fault, largest crustal movement is 4.7 mm/a.
Superimposed fault angle		55° - 80°	45° - 70°	80°
Gravity field		Gravity anomaly is smooth, gradient is less than 0.5 mGal/km	With regional positive negative gravity anomalies, the gradient is 1 to 2 mGal/km	Located in gravity anomaly step zones with negative gravity anomaly
Seismicity	Largest earthquake (Ms)	3.0	5	6.0
	Energy $\sqrt{E}$ ($10\sqrt{erg}$)	0.14	10.59	25.54
	Frequency (Ms >3.0)	1	4	2
Possibility for construction of HLW repository		Possible	Not possible	Not possible

- Natural analogues;
- Buffer/backfill materials and their geotechnics;
- Speciation of transuranic elements in solutions;
- Heater test; and
- Models for safety and environmental assessments.

The safe disposal of high level radioactive waste is a worldwide challenging task. Although China has made much progress in this field, there is still a long way to go. For example, a policy act related to nuclear waste disposal should be established, a more effective organization should be formed to promote the required work, and a way should be explored to raise enough money for the safe disposal of nuclear waste.

Information exchange is very important for the disposal of radwaste. China is willing to learn of the successful experiences in other countries and to strengthen international cooperation. China is also willing to share its own experiences and achievements with other countries, for the purpose of protecting the living environment of human beings and protecting the Earth

REFERENCES

Yang, L. Progress in DGD program for high-level waste deep geological disposal in China (1986-1991), in: Selected Works on the Study of Deep Geological Disposal of HLW in China, ed. by Sci & Tech. Bureau, China National Nuclear Corporation, pp.1-5, Dec. 1992. (in Chinese)

Xu, G. Investigation on site preselection of HLW repository in China, in: Selected Works on the Study of

Deep Geological Disposal of ELW in China, ed. by Sci & Tech. Bureau, China National Nuclear Corporation, pp. 11-15.Dec. 1992. (in Chinese)

Luo, S., et al., Radioactive Waste Manacement, Atomic Energy Press, Beijing, pp.1-9, 1995. (in Chinese)

Li, X., et al., Radioactive Waste Management, Atomic Energy Press, Beijing, pp. 62-66, 1995. (in Chinese)

Wu, X., The future development for China's nuclear power plants, China Nuclear Industry Newspaper, Nov. 1, 1995.

Li, X. et al., Research theory and method of regional crust stability, Geological Press, Beijing, 1987.

China National Seismological Bureau, The geophysical exploration achievement of the crust and upper mantle in China, Seismological Press, Beijing, 1986.

China National Seismological Bureau, Atlas of seismic activity in China, China Science and Technology Press, Beijing, 1991.

Geological Bureau of Gansu Province, The geological map and its explanation of Yumenzhen Quadrangle (1:200,000), 1972.

Geological Bureau of Gansu Province, The geological map and its explanation of Houhongquan Quadrangle (1:200,000), 1969.

CHAPTER 7

SITE SELECTION OF LOW AND INTERMEDIATE LEVEL RADIOACTIVE WASTE REPOSITORY IN THE REPUBLIC OF CROATIA

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Abstract. The radioactive waste repository site-selection program in Croatia consists of two stages. The first, the site survey stage, consists of: an exclusionary screening of the national territory, an internal evaluation of selected potential areas, and an identification of potential sites, as well as the selection of a few preferred (candidate) sites which would be suitable for further detailed investigations. The second, the site evaluation stage, consists of all necessary field investigations and additional site-characterisation tasks at previously selected preferred sites. Eight potential areas containing 42 potential sites have been selected in Croatia so far. Selection of 3-4 preferred sites is under way. Site selection methodology, including a description of applied exclusionary and comparative criteria, is also presented in the paper. The report is completed with basic information on current Croatian legislation and organisation of a regulatory body in the field, as well as with a brief characterisation of a few more perspective potential sites.

7.1 INTRODUCTION: SOURCES OF RADIOACTIVE WASTE

The necessity of constructing a low and intermediate level radioactive waste (LLW/ILW) repository in Croatia has developed from the fact that there are two main sources of radioactive waste materials. The first group of sources is waste generated in the Republic of Croatia itself, and the second, includes radioactive waste originating from the operation of the Krsko Nuclear Power Plant (NPP). Although the plant is situated in the neighbouring Republic of Slovenia, it represents a joint venture facility of both Slovenia and Croatia. Therefore, Croatia is obliged to find an appropriate solution to dispose of half of all radioactive waste generated during the lifetime of the NPP.

Radioactive waste in Croatia is derived mostly from various nuclear applications: medicine, industry, agriculture and scientific research. In addition to this, there are about 50,000 ionising smoke detectors distributed in 950 buildings and some 500 ionising lightning arresters (protectors) installed on 320 buildings in the country. However, the total amount of radioactive waste which has been generated in Croatia so far is not more than 80 m^3, and its estimated gross activity is 2.3×10^{12} Bq. The waste is composed of radionuclides like ^{152,154}Eu, used in ionising lightning arresters; ^{241}Am, installed in ionising smoke detectors; ^{192}Ir, ^{90}Sr, ^{85}Kr and some others, used in measurement and processing techniques in industry; ^{137}Cs and ^{60}Co, used in numerous diagnostic and therapeutic methods in medicine, etc.

Basically, there are three radioactive waste disposal approaches being practised in Croatia. The first, which refers to hospitals and research laboratories, is an adequate storage of waste until its activity falls to the background level, and after that it is treated as common waste. The second approach is related to waste types containing long-lived radionuclides, which are being properly stored in two temporary storage facilities at scientific institutes "Ruder Boskovic" and the Institute for Medical Research and Occupational Health (both are situated in Zagreb, the capital of Croatia). According to available data, there are some 500 institutions including more than 5,000 persons that are professionally in direct or indirect contact with radiation sources in Croatia.

The Krsko NPP is obviously the greatest producer of all radioactive waste expected to be disposed of in Croatia. Since the plant started operation in 1982, some 2,000 m^3 of LLW/ILW with a total activity of about 3.6×10^{13} Bq have been generated so far. However, it is realistic to expect some 8,500 m^3 of LLW/ILW to be generated in the lifetime of the NPP. The total activity generated during the plant lifetime, could reach an estimated 1.5×10^{14} Bq. In addition, roughly 11,000-12,000 m^3 of decommissioned waste is expected to be produced in the

plant by the end of its operation. This type of waste is supposed to be composed of 53% LLW, 36% ILW and 11% high-level waste. The prevailing radionuclide is ^{60}Co, which is responsible for about 90% of the total activity of the waste. It is expected that Croatia will be responsible for final disposal of some 10,000 m^3 of LLW/ILW, i.e. one half of the total LLW/ILW generated during the lifetime of Krsko NPP.

7.2 ESSENTIAL STRATEGIC ISSUES IN RADWASTE MANAGEMENT IN CROATIA

The strategy on radioactive waste management in Croatia was originally drafted by the APO - Hazardous Waste Management Agency in 1992. It was recognised then that only a systematic and well-organised program, which would be based on responsible treatment with radioactive materials and tightly related to regulatory bodies and current legislation, can provide an acceptable approach to safe operation of nuclear facilities and radiation sources in different nuclear applications.

Basically, the strategy as developed in Croatia, includes the following topics, i.e. tasks to be regularly and continuously practised:

- identify all sources and precise quantities of radioactive waste in Croatia, as well as create and maintain an inventory of radiation sources and radioactive waste materials;
- define a legal framework, i.e. the system of responsibilities;
- establish the financing of a national radioactive waste program;
- introduce and, if necessary, improve existing legislation and regulations in the field of radioactive waste disposal;
- develop the LLW/ILW repository, including all necessary activities like site selection, technical design, safety analysis etc.;
- foster public relations; and
- give support to other radiation safety related actions and programs.

The strategy has been rearranged according to existing IAEA documents, and is expected to be modified in the future according to forthcoming requirements and recommendations of this international agency.

7.3 LEGISLATION AND REGULATORY FRAMEWORK

Regulations concerning radioactive waste management, which are temporarilyy implemented in the Republic of Croatia, have been partly taken over from the ex-Yugoslav legislation. In general, these regulations have been established according to world-wide practice and support other regulations concerned with environmental protection or the management of other wastes. The basic regulation is the, "Law on Ionising Radiation Protection and Special Safety Actions in Nuclear Energy Implementation", issued in 1984. From this law, 17 regulations and codes of practice have been subsequently derived. The new Croatian law on radiation protection and nuclear safety is expected to be approved by the end of 1996.

Documents to be emphasized here, due to their particular importance in the field of site selection and radioactive waste repository construction and operation, are "Code of Practice on Conditions of Locating, Construction, Start-up and Operation of Nuclear Facilities", "Code of Practice on Standard Format of Safety Report and Other Documentation Needed for Safety of Nuclear Facilities", as well as "Code of Practice on Methods of Collecting, Account, Processing, Storing, Final Disposal and Release of Radioactive Waste Substances in the Environment. (all of them were issued in the period 1986-88). These regulations went into effect according to the "Law on Taking Over the Federal Laws in the Field of Health Protection, Applied in the Republic of Croatia as Republic Laws".

A national regulatory body organising and controlling radioactive waste management, as well as radiation protection issues, has not been established in Croatia as a single institution covering all related issues. It consists of sections of three separated governmental entities: the Ministry of Health, the Ministry of Economy, and the State Directorate for Environmental Protection. Unfortunately, there is no permanent body in Croatia which would co-ordinate activities of these ministries in the field. The Sanitary Inspectorate, as a section of the Ministry of Health, is the competent national authority for radiation protection. The Ministry of Economy, i.e. Department of Nuclear Safety, is the competent national authority for siting, construction, start-up, operation and closure of nuclear facilities. The competence of the State Directorate for Environmental Protection is directed to issues related to environmental clean-up actions of contaminated sites, hazardous waste management, etc. It should be added that a few other ministries (Transport, Finances, Interior, etc.) are responsible for licensing particular activities, which are incorporated in

radioactive waste management and cover the transportation, import-export, release of effluents and some other issues.

7.4 SITE SELECTION METHODOLOGY: STRUCTURE OF SITE SELECTION PROGRAM

7.4.1 Concept Description

The global concept of the Radioactive Waste Repository Project in Croatia consists of several interrelated main task groups. Besides site selection, it also comprises licensing, technology & design development, safety assessment, economic evaluation, transportation analysis and waste characterisation. The whole project is supported by particular activities related to development of legislation and regulatory body organisation. Since there is an urgent economic need to ensure additional energy sources in Croatia, the site selection of thermoelectric- and nuclear power plants is encompassed by the same program as well.

The site selection includes two stages: the first, site survey stage (Fig. 7.1), terminating with the inclusion of candidate sites into the Physical Plan of Croatia; and the second, site evaluation stage, aimed at defining the final repository site through field investigations and other necessary actions.

The first part of site selection, i.e. the site survey stage, is currently under way. It includes the actions that are extremely sensitive since defining site selection methodology and criteria, as well as achieving political and public acceptance for repository siting, have to be completed before preferred (i.e. candidate sites) will be identified and included in the Physical Plan of Croatia and, thus, become available for further field investigations. The planned activities are projected to be performed in two phases: (1) regional analysis and selection of potential areas; and (2) selection of preferred sites (Fig. 7.1).

In other words, the philosophy of site selection is, at first, to define exclusionary criteria for the global reconnaissance of Croatia in order to find potential areas. After comparative and additional exclusionary criteria are defined, potential areas will be subjected to more detailed evaluation designed to identify a number of potential sites. Through a comparison of potential sites and their internal characterisation, a few preferred sites will be evident. These, preferred (or candidate) sites are finally supposed to be included in the Physical Plan of

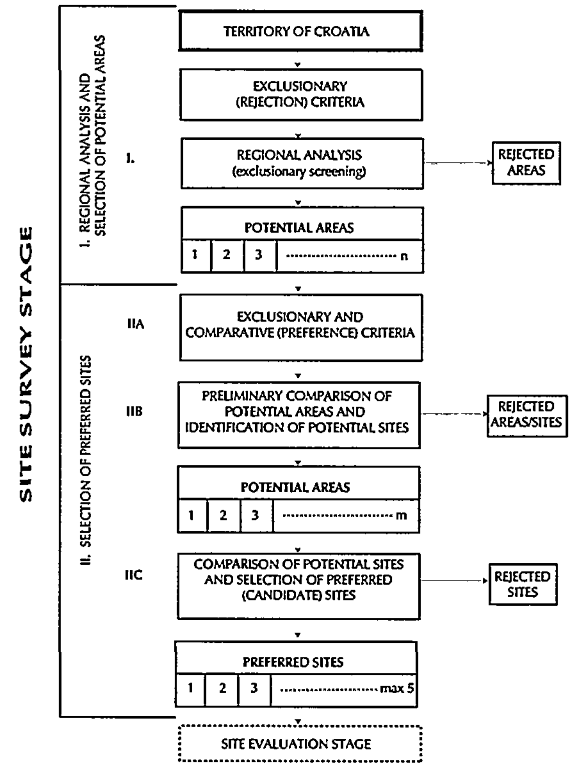

Figure 7.1. Site selection procedure as applied in site survey stage.

Croatia. In the second stage, detailed site investigations will be worked out at two or three preferred sites, which will result with the identification of the final repository site.

In accordance with the above mentioned, it is necessary to clarify terminology involved in referring to selected areas i.e. sites: *potential areas* represent larger areas (100-600 km^2), characterised by acceptable isolation properties; *potential sites* are smaller homogeneous areas (5-20 km^2), derived from potential areas, which are favourable for siting a radioactive waste repository, and finally, *preferred sites* are small areas (2-20 km^2), highly acceptable for the repository siting and convenient for inclusion in the Physical Plan of Croatia[1].

As shown schematically in Figure 7.1, the site survey stage consists of two phases. It could be described in more detail as follows:

- *Regional analysis*, terminating with selection of potential sites, represents the first phase of the site survey stage. It is based on an assessment of the whole territory of Croatia using selected exclusion-

ary criteria ("exclusionary screening"). Areas which do not meet the requirements, defined by at least one of these criteria, are eliminated. Remaining areas are designed as potential areas for further analyses. Regional analysis is based on appropriate information derived from topographic and other thematic maps of Croatia on the scale 1:300,000.

- In the second phase - *selection of preferred sites* - all previously selected potential areas are checked by the exclusionary criteria, and then evaluated on the basis of more detailed data and large scale maps (1:100,000). Comparative criteria are applied to both potential areas and potential sites in order to identify potential sites and preferred sites, respectively. This task is divided into three sub-phases as follows.

7.4.1.1 Selection of Parameters and Criteria.

Proposed parameters and criteria, applied in the identification, selection and comparison of potential sites, are defined by a special team of experts, composed of specialists from different fields of interest (from geology to sociology). Both exclusionary and comparative criteria were officially published in the Croatian official gazette "Narodne novine", No. 78/92. Comparative criteria are generally presented as requirements for achieving a certain goal or a desired state. These goals can be defined as desirable conditions, and if a desirable condition can not be achieved, the criteria should describe the degree of acceptability (i.e. to what extent does the solution approach desirable conditions). The following four groups of comparative criteria have been formed in accordance with basic aspects or dominant characteristics of the criteria involved:

A. *Engineering aspects.* Criteria comprised by this group show whether the engineering requirements for acceptable radioactive waste disposal are met. Emphasis is placed on the economic issues of facility construction and operation. Preference is given to site characteristics which enable simpler and, thus, economically more acceptable solutions. Characteristics of the site can also affect, to a lesser extent, certain safety aspects of the facility, but these can be successfully compensated by engineering interventions. Regarding this comparative criteria group, sites requiring simple and less expensive engineering solutions are preferred.

B. *Safety-related aspects.* In this group are collected criteria needed to determine whether the safety requirements for construction and/or operation of the facility are met. Concordance with safety requirements is checked during all phases of the site selection program, and is very much evident in the licensing procedure of the facility. Emphasis is placed on evaluating the physical properties of an area that could have negative effects on facility safety. However, some site characteristics could have an impact on the choice of additional needed engineering solutions that can influence on cost-benefit issues of construction and operation. Regarding this group of comparative criteria, preferred sites are those where facility safety is derived from more convenient physical (natural) characteristics of the site, requiring a minimum of engineering intervention.

C. *Environmental impact and acceptability in the immediate site area.* This group includes comparative criteria showing whether the safety requirements, concerning the impact of the facility on the immediate environment, are met. Emphasis is placed on the environmental impact of the facility during its regular operation as well as in cases of possible accidents. Site characteristics can also have an impact on the degree of social acceptance. Concordance with safety requirements must be confirmed throughout all phases of site selection, and it will be evident in corresponding documentation (e.g. study on environmental impact assessment, safety reports, site permits etc.). Preference is given to those physical (natural) site characteristics which provide less impact on the environment, as well as to existing and planned land use types which could act positively on the safety and acceptability of the site.

D. *Acceptability of the facility site in the broader area.* This group consists of comparative criteria that assess the possible impact of the radioactive waste disposal facility on the broader area. Emphasis is placed on an analysis of present and planned land use types, as well as on the degree of social acceptance regarding possible changes in the value of the area. Preference is given to present and planned land use types causing less conflict and therefore having a greater social acceptability, as well as to physical characteristics that provide milder environmental effects resulting from the facility operation.

Comparison of potential sites: method of weighted criteria

In this phase, it is necessary to choose the most appropriate method of site evaluation and comparison. There

are several techniques and methods in world practice. On the basis of experience acquired in Croatia and Slovenia in the field during the last few years, a multiple criteria analysis has been chosen as the most effective. In particular, the method PROMETHEE (i.e. "Preference Ranking Organisation Method for Enrichment Evaluations"), created by J.P. Brans and P. Vincke, has been applied in our case. The method represents a computerised analysis of a multiple criteria technique and decision-making methods. Thus, objectivity in site selection and assessment has been successfully achieved. The method is based on the application of numerous criteria in order to express interrelations among alternatives, indicating a group of "better" solutions.

The relative significance of each criterion is expressed by assigning to it a corresponding weighting factor. The values of the weighting factors are defined by applying the rating method, and are based on decisions of the experts. After discussion, members of the expert team propose weighting factors for all comparative criteria. As result, a special co-ordinating group adopts the final list of weighting factors (as is presented in Table 7.1; see the section "Comparative Criteria"). It is obvious from the Table that the expert team concluded that criteria group C ("Environmental impact and acceptability in the immediate site area") is the most important with a total weighting factor for this group of 52.5% . This is followed by the group B ("Safety related aspects") with a 30.0%. share. The acceptability of broader site area (group D) is expressed by a total weighting factor of 9.5%, since it is assumed that LLW/ILW repository will have almost no impact on the broader site area. Finally, the importance of engineering aspects is estimated to be only 8% because it is expected that the repository would not require considerable civil-engineering interventions.

7.4.1.2 Assessment and Comparison of Potential Areas and Identification of Potential Sites

In order to get more precise information, the first task in this sub-phase is to check on the exclusionary criteria application which would be based on more detailed maps (on the scale of 1:100,000). Geologic characteristics, i.e. engineering/geologic properties - including lithological and geomorphologic, tectonic and seismic, as well as hydrogeological characteristics of the selected sites, are of special interest.

7.4.1.3 Assessment and Comparison of PotentialSites Followed by Selection of Preferred (Candidate) Sites.

In this sub-phase, more detailed data on sites, previously chosen in sub-phase 7.4.1.2, are collected in order to perform a comparative assessment of their acceptability. The on-site investigation of all potential sites should be carried out, and the exclusionary criteria are supposed to be checked once again. In the field of potential sites, the previously determined reference points should be re-evaluated. With regard to basic safety site characteristics (lithology, hydrogeology, risk of flooding), a certain number of potential sites is expected to be excluded due to their lower quality, by comparison with the others. It has already been decided that not more than 3-5 preferred (candidate) sites will finally be proposed for further detailed on-site investigations.

The final comparison of potential sites, as well as the final selection and proposal of preferred sites, will be performed after validation of both criteria and weighting factors by a competent governmental/parliamentary body. Comments on the involved criteria that would probably result from public debate and discussions with experts will be thoroughly re-examined afterwards by members of the expert team. If agreement is reached, these comments will be accepted. In that case, the main sections of the site survey procedure will be reanalysed.

7.5 SITE SELECTION AS A MULTIPLE CRITERIA ANALYSIS

7.5.1 Some Basic Facts about Croatia

The Republic of Croatia (56,538 km^2) is situated in the southern part of central Europe. It includes three major physical-geographical regions: Pannonian basin, Dinaric Alps and the Adriatic coast. As part of former Yugoslavia, it proclaimed its independence in June 1991. There are about five million inhabitants living in Croatia, with an average population density of 80 inhabitants per square kilometre. Due to its unusual shape, the total length of the Croatian border is very long - 2,100 km. The neighbouring countries to Croatia are Slovenia, Hungary, Serbia, Bosnia-Herzegovina and Montenegro, as well as Italy (the border with Italy runs along the Adriatic sea). The major Croatian cities are Zagreb, the capital (900,000 inhabitants), Split (250,000), Rijeka (200,000), Osijek (120,000), Zadar (80,000) and Pula (70,000) (Fig. 7.2). Farming, forestry, fishery, industry, shipbuilding and tourism are the main economic activities. Industry, including mining and

Table 7.1. List of comparative criteria and their weighting factors.

Group: Percent:	A 8.0%	B 30.0%	C 52.5%	D 9.5%	TOTAL 100%
1. Transportation					3.7%
1.1 *Transportation or radioactive waste*			C.1.1. - 3.7		
2. Meteorology and hydrology					14.3 %
2.1 *Hydrological issues*		B.2.1 - 7.5	C.2.1 - 3.2		
2.2 *Meteorological issues*		B.2.2 - 2.8	C.2.2 - 0.8		
3. Geology and seismology					35.8%
3.1 *Seismotectonics and seismics*	A.3.1 - 4.0	B.3.1- 5.9			
3.2 *Soil mechanics and slope stability*	A.3.2 - 4.0	B.3.2 - 9.8			
3.3 *Hydrogeology*			C.3.3 - 12.1		
4. Demography					5.4%
4.1 *Population density*			C.4.0. - 5.4		
5. Present and planned land use types					22.2%
5.1 *Settlements*			C.5.1 - 4.4	D.5.1 - 3.5.	
5.2 *Tourism*			C.5.2 - 3.3	D.5.2 - 2.5	
5.3 *Agriculture*			C.5.5 - 2.6		
5.4 *Forestry*			C.5.4 - 2.2		
5.5 *Industry and mining*			C.5.5 - 1.1		
5.6 *Infrastructure*			C.5.6 - 1.5		
5.7 *Special purposes*			C.5.7 - 1.1		
6. Environmental protection					18.6%
6.1 *Protection of natural heritage*			C.6.1 - 3.0	D.6.1 - 2.5	
6.2 *Protection of cultural heritage*			C.6.2 - 1.6	D.6.2 - 1.0	
6.3 *Soil properties*		B.6.3 - 4.0	C.6.3 - 2.4		
6.4 *Bio-ecological values of the site*			C.6.4 - 2.5		
6.5 *Radiological aspects*			C.6.5 - 1.6		

energy production, produces 44% of the national gross income.

7.5.2 Geological Background

Sedimentary rocks prevail in the geologic structure of Croatia, occupying 95% of its surface lithology. Moreover, it should be emphasized that the distribution of Mesozoic carbonate rocks, like limestone and dolomite, is the leading rock type throughout the southern part of the country (i.e. south from Karlovac.) The morphology of this area is characterised by karst (Fig. 7.3), occurring in two basic varieties: shallow karst, covered by soil in the interior parts, and thick, exposed karst stretching along the Adriatic coast. An irregular hydrogeology and very sensitive mechanisms have a remarkable impact on the quality of surface- and ground-water and control some specific bio-lithological processes (e.g. formation of travertine). Consequently, a lot of human activities being practised in the area eventually leads to considerable environmental pollution.

In the northern part of the country in the Pannonian basin, fluvial, erosional and aeolian clastic sediments are dominant, particularly in the lowlands along the rivers Sava and Drava, as well as in the east Croatian plain. The central Slavonic massifs (Papuk, Krndija, Psunj, Pozeska gora), in the heart of northeastern Croatia (known as Slavonia), are mainly composed of old Paleozoic igneous (granite, basalt, etc.) and meta-

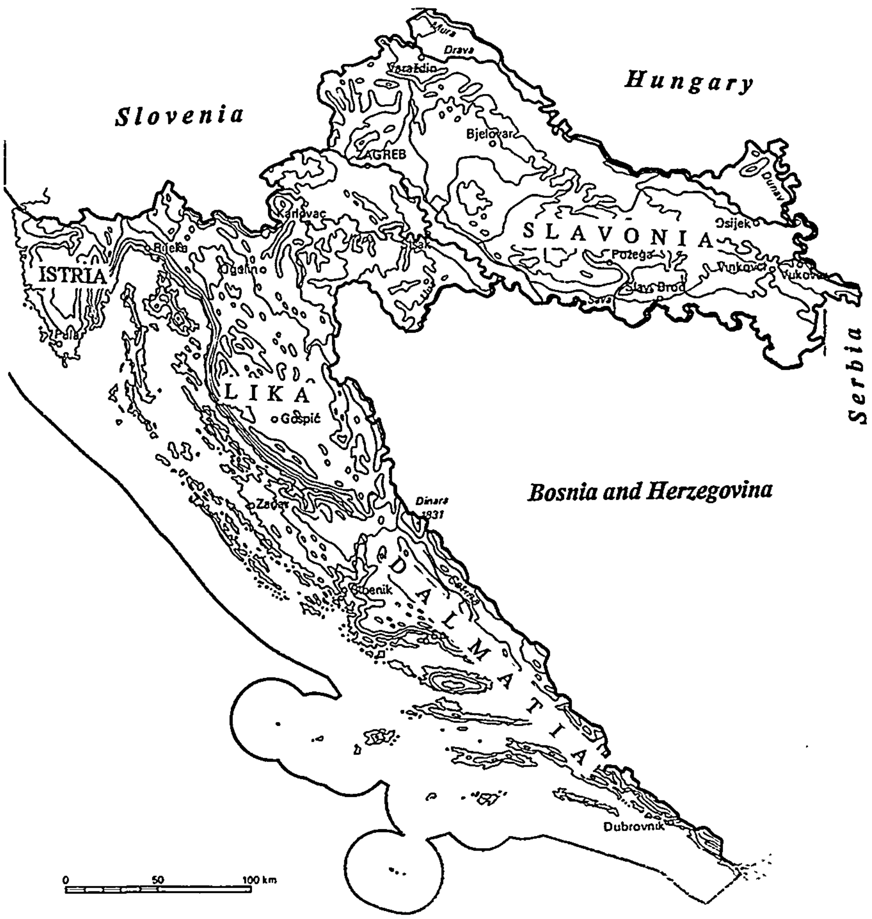

Figure 7.2. Croatia with its major cities.

morphic rocks (crystalline schists, gneiss, etc.) (Fig. 7.3).

The territory of Croatia is an unstable area from the standpoints of both tectonics and seismicity. Several major longitudinal faults run in a NW-SE direction, in both the northern and southern parts of the country. Where they intersect, transverse faults are in many cases characterised by geothermal springs. Earthquakes as intensive as VIII-X MCS can be expected in most of the Adriatic coast region (southern Dalmatia, in particular), but they can also occur in some areas of northwestern Croatia.

During site selection for a radioactive waste repository, attention should be paid to both repository design options and available rock types. The leading principle in both cases is to ensure environmental safety, and not to save money. Almost half of Croatia is not convenient for a repository siting due to the prevailing carbonate lithology, karst morphology and the consequent irregular circulation of groundwater. Since the remaining flatlands are generally unsuitable because of major infiltration with high water-tables (along the rivers Sava, Drava and Danube), the areas of interest are the previously mentioned mountains of the country's interior, Papuk, Krndija, Psunj and part of Bilogora as well as Moslavacka gora, that is situated somewhat to the west. All these mountains, except Bilogora, are horst-structures composed to a great extent of granite, gneiss and schist[2].

7.5.3 Site Acceptability Assessment

In the site selection process, it would be insufficient to locate only a host-rock capable of providing waste containment, i.e., to keep water away from the waste. According to the accepted approach, it is recognized that the entire system, i.e. waste form and package, host-rock, geological formations of the broader area, and environment of the region, should provide waste containment. In keeping with this philosophy, the applied criteria should address all site characteristics that contribute to waste containment and insulation. These characteristics

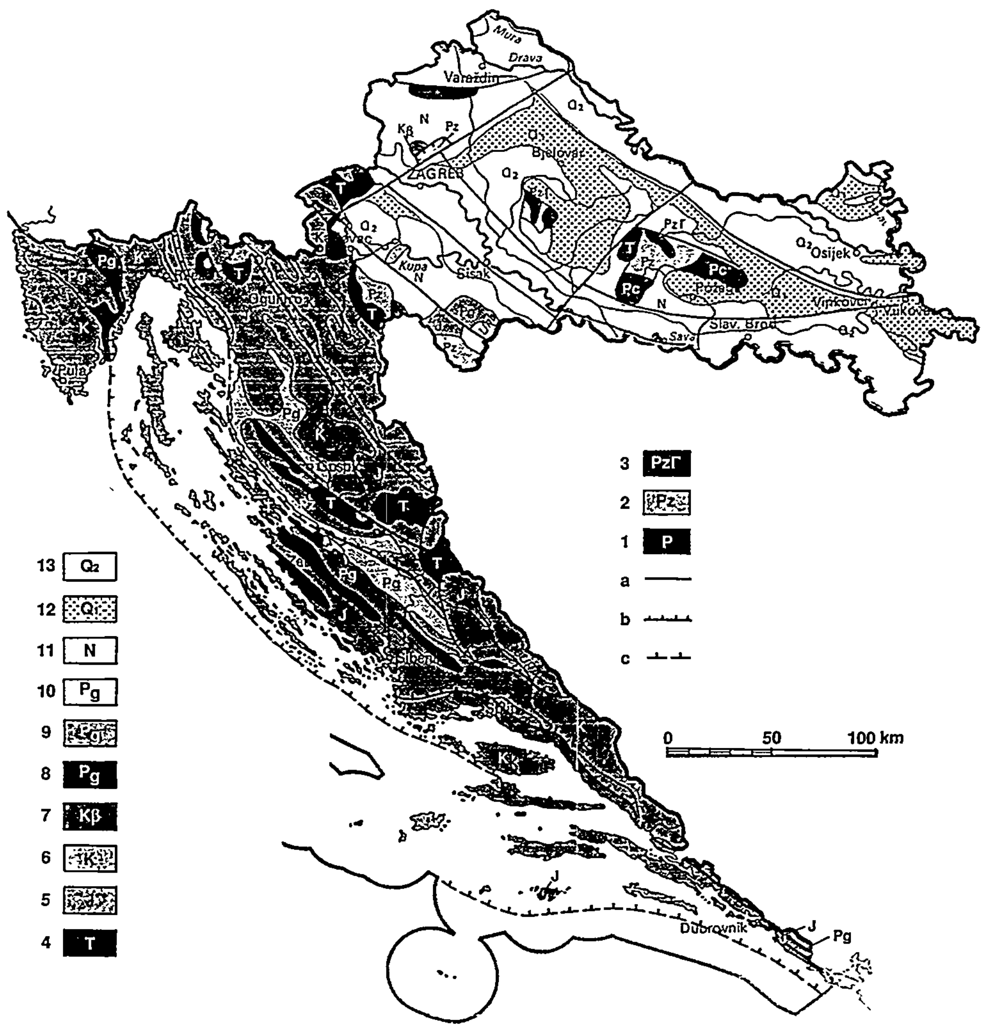

Figure 7.3. Geologic map of Croatia:. 1- Precambrian (metamorphic rocks); 2 - Paleozoic (clastites, additionally carbonates); 3 - Paleozoic (igneous rocks: granite); 4 - Triassic (carbonates, add. clastites); 5 - Jurassic (carbonates, add. igneous clastites); 6 - Cretaceous (carbonates, add. clastites/flysch); 7 - Cretaceous (igneous rocks: basalt); 8 - Paleogene (limestones); 9 - Paleogene (clastites, flysch) 10 - Paleogene (limestone clastites, limestones); 11 - Neogene (clastites, limestones); 12 - Pleistocene (clastites/mostly non-cemented); 13 - Holocene (clastites/mostly non-cemented.) Tectonics: a - normal faults; b - reverse faults and thrusts; c - reverse faults and thrusts (submarine).

include geotechnical and socio-economic concerns. In the application of these criteria, greater consideration should be given to those factors that directly influence the safety of the repository than to those criteria which affect only repository cost and timeliness.

In the final analysis, repository licensing and operation can not be performed until it is demonstrated that the total waste disposal system, including the waste form and its packaging, design of engineered facilities and barriers, as well as the multiple natural barriers, will adequately protect public health and safety and preserve the quality of the environment[3].

7.5.4 Exclusionary Criteria

There are ten exclusionary criteria to be applied in the first phase of the site selection process, i.e., in a regional analysis of the whole territory of Croatia, where the aim is to reject all areas that do not meet requirements of these criteria. The exclusionary criteria are divided, similarly to the previously discussed classification of comparative criteria, into four groups: A - Engineering aspects; B - Site safety related aspects; C - Site accept-

ability; and D - Facility safety. These four aspects involve three basic principles for repository siting: cost-benefit ratio, acceptability and safety. The following is a brief description of each criteria with examples of the excluded areas that result when the territory of Croatia is analyzed.

1. Meteorological and hydrological criteria: Flooding (Group D)

All natural floodplain areas excluded (assuming repeat periods at 1,000 year intervals), regardless of whether they are protected or not.

Comment: The LLW/ILW repository must be situated beyond the reach of floods. Hence, all natural floodplains, as well as those areas where accumulations of water are planned, are excluded. Areas, where it is not likely to prove a low risk of flooding by additional investigations, are also excluded. The risk of flooding is supposed to be studied throughout the site selection program.

Excluded areas: Large lowland zones along the major Croatian rivers of Sava and Drava, as well as the Danube in Baranja region; and numerous sinks in karst regions, which are characterised by seasonal flooding.

2. Geology and seismology: Seismotectonics (Group A)

Areas with maximum potential earthquake intensity equal to or higher than IX MCS, are excluded.

Comment: Elimination of areas that are potentially affected by strong earthquakes is also safety related and should be thoroughly evaluated in further investigations. From technical and economic viewpoints, the cost of facility construction increases exponentially as the expected earthquake intensity increases; therefore, the expected maximum intensity IX MCS has been taken as the highest acceptable value.

Excluded areas: Most of southern Croatia (Dalmatia), including the coastal zone, Kvarner bay in the north Adriatic, as well as the so-called "Balaton morpho-lineament", i.e., the mountains of Medvednica and Kalnik in northwest Croatia. Some locations in northeast Croatia (Slavonia) are also excluded by this criterion.

3. Geology and seismology: Neotectonics (Group A)

Zones characterised by known active faults are excluded.

Comment: It is known that in neotectonically active areas, as well as in the vicinity of regional and even local active faults, dislocations and rock cracking are possible. These events could cause damage to a waste disposal facility. This safety related criterion is also supposed to be applied (or re-checked) in subsequent phases of the site selection program. In addition it should be mentioned that the term "active fault", as used here, is related to a zone that is a few kilometres in width and includes two or more parallel faults. Earthquakes with magnitudes 5.5 - 7.0 can occur along these faults.

Excluded areas: Southern Croatia, including the coastal zone, Kvarner bay, parts of northwest Croatia, as well as some spots in northeast Croatia - Slavonia.

4. Geology and seismology: Lithology and geomorphology (Group A)

Option 1: Shallow ground disposal: *Areas characterised by an increased erosion rate due to lithology (prevailing rock types) and/or relief dynamics, as well as areas composed of rocks which are unstable under natural conditions (e.g. which are not resistant to weathering etc.) after the completion of civil-engineering works, are excluded.*

Option 2: Tunnel-type disposal: *Areas characterised by land slides and rockfalls are excluded if it is thought that these processes could pose a hazard to repository structures.*

Comment: Lithologic, geomorphologic and geotechnic characteristics, along with hydrogeological properties of an area, are considered the most important criteria related to the siting of an LLW/ILW repository. The upper Pliocene and Quaternary clays and Neogene marls (lower Pontic "Abichi" sediments, in particular) are considered the most suitable for a shallow-ground facility siting in Croatia. For the tunnel-type repository, areas composed of granites and some other igneous rocks are most suitable. Some metamorphic rocks like gneiss and some types of schists could also be used for this purpose. Areas which do not meet these requirements are excluded.

Excluded areas: Lowlands along the Sava and Drava rivers in the interior of Croatia, due to possible liquefaction caused by earthquakes; hills which surround the mountains of Zumberacka gora, Medvednica, Kalnik and Ivanscica in northwest Croatia and are composed of Neogene and Quaternary sands, clays and marls; flysch belts in the Istrian peninsula (west Croatia), parts of allu-

vial zone of the Sava river and its tributaries, and some sections of river valleys in Istria and Dalmatia and few other spots in the karst regions.

5. Hydrogeology: Protection of water-bearing layers (Group B)

Areas containing protected sources of drinking water are excluded. In order to preserve groundwater from possible radioactive pollution, a disposal facility can not be situated in an area containing significant water-bearing layers of any type.

Comment: Hydrogeologic properties of any potential site significantly influence the selection of repository design, and also represent specific mechanisms for the assessment of possible environmental pollution. Therefore, areas of high risk for groundwater pollution are excluded at the very beginning of the site selection program. In the case of a tunnel-type repository, suitable areas are those which are composed of solid, primarily impermeable rocks having no secondary porosity. For shallow-ground disposal, the more preferable areas are composed of thick clays and marls without major aquifers. The possible impact on springs and wells, in addition to the necessary protection measures, are examined in all further site specific investigations.

Excluded areas: Parts of the country characterised by major aquifers where groundwater is not protected by near-surface layers or by formations of high porosity. This means almost all of the lowlands along the rivers of Mura, Drava and Danube, the extreme upstream and downstream sectors of the Sava river zone, as well as practically the entire karst area, which represents in fact the southern half of the country.

6. Demography: Population density (Group C)

Areas characterised by a population density of 80 inhabitants per square kilometre (i.e., the average value for Croatia) or more, within a 20 km radius around the facility site, are excluded.

Comment: One of the preferred factors in lowering the risks of nuclear facilities to human health and environmental preservation is to have as low a population density as possible. For that reason the more densely populated areas of Croatia are excluded. The population density in the vicinity of a facility will be re-evaluated in subsequent phases of the site selection program.

Excluded area: Broad areas around major Croatian cities and the towns of Zagreb, Split, Rijeka, Osijek, Zadar, Pula, Dubrovnik, Karlovac, Sisak, Varazdin and Sibenik, which are all characterised by population densities higher than 80 inhabitants/km^2.

7. Present and planned land use types: Special purposes (Group B)

Areas designated for special purposes, including their protected zones, are excluded.
Comment: Certain areas are of special interest for national defence. All areas which are, or could be, in conflict with this criterion are excluded.

8. Present and planned land use: Mining exploration of ores and minerals (Group B)

Areas in zones of present or planned mining including exploration of minerals, gas, oil, coal etc., are excluded.

Comment: Exploration of ores and minerals could have an impact on the LLW/ILW disposal facility safety, and in this way make the potential site economically unfavourable. Some limited areas along the Sava and Drava rivers have been excluded by this criterion due to existing or perspective exploration of oil and gas.

9. Environmental protection: Protection of natural heritage (Group B)

National parks, natural parks and other specific natural areas of common interest, are excluded.

Comment: In early phases of the site selection program, it is necessary to identify areas that are protected as a natural heritage and are therefore not suitable for siting a LLW/ILW disposal facility. This criterion is closely related to the requirements defined in the Natural Protection Law, which mentions specific, ecologically sensitive areas that should be continuously protected.

Excluded area: National parks (Brijuni islands, Risnjak mountain, Plitvicka jezera lakes, Velebit mountain, Kornati islands, Krka river and part of Mljet island), natural parks (parts of some mountains, rivers, swamps, islands, etc.) and other significant natural reservations (landscapes, forests, etc.)

10. Environmental protection: Protection of cultural heritage (Group B)

Areas containing monuments of cultural heritage which are registered in the List of World Cultural and Natural

Heritage, as well as those that are of an extraordinary national importance, are excluded.

Comment: It is necessary at the very beginning of the site selection process to define areas which are protected as monuments of cultural heritage, since they have to be avoided as potential sites for the LLW/ILW disposal facility. The term "cultural heritage", as used here, is applied not only to an object of interest itself, but also to the surrounding area.

Excluded area: Monuments of cultural heritage presented on the List of World Cultural and Natural Heritage (e.g. Diocletian's Palace in Split and the old town of Dubrovnik), larger areas known by their specific landscape, as well as culturally, historically and aesthetically valuable locations.

7.5.5 Comparative Criteria

After potential sites are identified by the process of exclusionary screening (including re-checking of preliminary chosen sites by the same exclusionary criteria, but applied now on more detailed maps), they have to be subjected to a comparative analysis, i.e. an evaluation based on an application of comparative criteria. As already mentioned, these criteria were defined by the expert team members and divided into four aspect groups: A - Engineering, B - Safety, C - Environmental impact and acceptability in the immediate site area, and D - Acceptability of the site, i.e. the facility in the broader area. The criteria were then classified into six thematic groups, and, according to their importance, were given corresponding weighting factors, also defined by the expert team. A total of 28 comparative criteria (see Table 7.1) were applied in the site selection program. Table 7.2 provides a list of topics to which the comparative criteria in Table 7.1 were applied.

Table 7.2. Application of comparative criteria.

Identification Number	Comparative Criterion to be applied to:
A.3.1.	Seismic Activity
A.3.2.	Soil mechanics
B.2.1.	Risk of flooding
B.2.2.	Extreme meteorological phenomena
B.3.1.	Neotectonic activity
B.3.2	Lithology and geomorphology
B.6.3.	Chemical aggressiveness of soils
C.1.1.	Transport of radwaste
C.2.1.	Distance to surface waterways
C.2.2.	Dispersion in atmosphere
C.3.3.	Presence of infiltration
C.4.0.	Population density
C.5.1.	Settlements
C.5.2.	Tourism
C.5.3.	Agriculture
C.5.4.	Forestry
C.5.5.	Industry and mining
C.5.6	Infrastructure
C.5.7.	Special purposes
C.6.1.	Natural heritage protection
C.6.2.	Cultural heritage protection
C.6.3.	Plant production (soils)
C.6.4.	Bioecological issues
C.6.5.	Radiological issues
D.5.1.	Settlements
D.5.2.	Tourism
D.6.1.	Natural heritage protection
D.6.2.	Cultural heritage protection

7.6 DESCRIPTION OF COMPLETED ACTIVITIES

The activities on radwaste repository site selection in Croatia began in 1988. Until 1991, Slovenia and Croatia had not been proclaimed as independent states, and the preliminary activities on repository site selection in both republics (i.e. states) were managed by the Inter-republic Coordination Commission in order to harmonise the site selection procedures in both countries. In Croatia, the Ministries of Energy and Physical Planning, through the Croatian Electricity Management Board, were committed to the Institute for Urban Planning of Croatia to perform the study "Site Screening, Investigation and Assessment of Site Suitability of Fossil Fuel Power Plants and Nuclear Facilities in the Territory of the Republic of Croatia". The goal of the study was the identification of preferred sites for thermal and nuclear power plants, as well as a radioactive waste repository, which might later be nominated for this purpose in the Physical Plan of Croatia. The Institute for Urban Planning created a special team of experts, to define and assess selection criteria, as well as to act decisively in identifying potential areas, potential sites and preferred (candidate) sites. The set of exclusionary criteria was defined by July 1989, and validation of criteria was carried out by the Croatian Government.

Using the method of exclusionary screening for the whole territory of Croatia (Fig. 7.1), the resulting map showing areas which "survived" exclusion of inconvenient regions was obtained, and eight potential areas

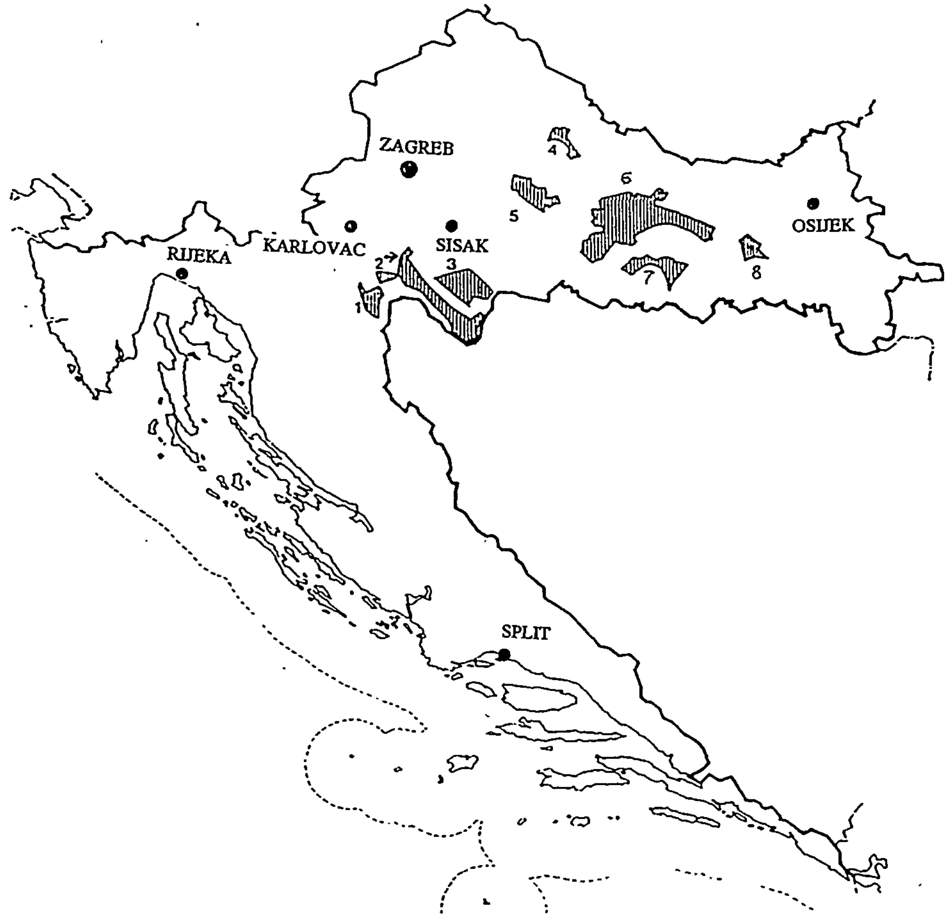

Figure 7.4. Locations of eight potential site areas for a LLW/ILW repository.

(Petrova gora, Trgovska gora, Zrinska gora, Bilogora, Moslavcka gora, Papuk-Krndija-Psunj, Pozeska gora, Dilj) were finally identified in autumn 1990 (Fig. 7.4). Collecting additional data on potential areas and defining comparative criteria were the main project activities being worked out during the rest of 1990 up to August 1991. Site selection methodology and criteria were examined by the IAEA Radioactive Waste Management Advisory Programme (WAMAP) in the spring of 1991, and were assessed as high-grade. Since autumn 1991, i.e. after the fall of former Yugoslavia, the newly established APO - Hazardous Waste Management Agency started to manage the remaining part of site selection program in Croatia. On the basis of previously performed actions, it was possible in autumn 1994 to identify 42 potential sites, which are dispersed within the above mentioned eight potential areas (Fig. 7.4). Consequently, the only task remaining in the second phase of the site survey stage is to select 3-5 preferred sites, which will be suitable for further detailed on-site investigations. This task should be done by the end of 1996.

7.7 CONCLUSIONS

The siting of a radioactive waste repository in Croatia has been adjusted to the requirements derived from physical (regional) planning documentation. The site selection program is composed of two stages: (1) a site survey, terminating with the inclusion of preferred (candidate) sites in the Physical Plan of Croatia; and (2), site evaluation to define the final repository site through field investigations and other necessary activities. There has been very slow progress in carrying out this program due to certain circumstances in the country and the entire region, and the preferred sites will not be defined

before the end of 1996.

The system approach includes a simultaneous preparation of the repository technical design, performance assessment and some additional activities. Furthermore, it would be preferable to find a site that would be convenient for a combined disposal facility, not only for radioactive waste, but also for other types of hazardous wastes. An interdisciplinary approach is applied in the site selection, where all relevant topics from geology to sociology are involved in the program. A standard site screening technique has been applied in this process. The stepwise approach to the site selection is based on validation of every single step (starting from the entire state territory), and is supposed to be terminated with the selection of few preferred sites. The program is based on the implementation of both exclusionary and comparative criteria that have been selected by a special team of experts.

The selection of a radioactive waste repository site is considered as a process leading to an optimisation of land use policy of the country in the light of attaining an optimum method of exploitation of the national resources. In addition, the site selection is conceived as an action supposed to be internationally verified, aiming to strengthen our proclamation that, "We are doing just the same as other countries". It is also worth mentioning that the radioactive waste repository site selection is part of a useful project directed to site selection of nuclear-, coal- and gas-fired power plants. Anyhow, a regularly performed site selection process could be assessed as a prerequisite for possible future nuclear energy program in Croatia.

Considerable effort is being given to a program of providing full, complete, continuous and honest information to the public. Necessary preparations for involvement of local communities into the site selection process have already been completed. Hence, democratisation in the siting process of a radioactive waste repository, as a controversial facility, could be achieved in the best possible way. It also includes the determination of incentives needed by communities living in the vicinity of the repository. In addition, being concerned of their possible NIMTOO ("Not in my Term of Office") behaviour, we are also very careful, and are doing our best to prevent "premature" exposure of politicians to the consequences of the expected NIMBY effect.

Finally, we are facing the specific problem of some malicious attempts, done mostly by certain politicians, or "experts," to explain the radioactive repository site selection program in Croatia as an action based, above all, on an ethnic criterion. It must be emphasised that in spite of arguments derived from an extremely irrational approach, these accusations could attain the end, which is not, in fact, to stop or aggravate the site selection process itself, but to deteriorate the very sensitive political and military situation in the broader region.

REFERENCES

1. Site Survey Stage of the Siting Process for Low- and Intermediate Level Radioactive Waste Disposal Facilities, The Institute for Urban Planning of Croatia, Zagreb, 1991.

2. Saler A., General approach and site selection criteria for low- and intermediate-level radioactive waste repository in Yugoslavia, *in* Geological Problems in Radioactive Waste Isolation: A World Wide Review, Lawrence Berkeley Laboratory Report LBL-29703, Ed. P.A. Witherspoon, pp. 173-179, 1991.

3. DOE (U.S. Department of Energy,) NWTS Criteria for the Geological Disposal of Nuclear Wastes: Site-qualification Criteria, ONWI Staff, 1980.

4. IAEA (International Atomic Energy Agency). The Mission Report Radioactive Waste Management Advisory Program (WAMAP), Vienna, 1991.

Chapter 8

Geological Disposal of Radioactive Waste in the Czech Republic

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8.1 Introduction

The Czech Republic has appeared on the map of Central Europe since 1/1/1993 after the split of Czechoslovakia in two parts; the other is the Slovak Republic. With its 10.3 million inhabitants and a territory of 78,864 km^2, the country is comparable with e.g. Belgium, Switzerland or Austria. The Czech Republic belongs to the more successful transformed states of the former Soviet block: A steady decrease of inflation (9% in 1995), increase of GNP (4.6% in 1995), non-deficit budget and low unemployment (3.5% in 1995) are signs of a stabilized economic situation.

The intensive industrialization of the country requires among other things, a high production of energy: the installed capacity is 13,793 MWe, of which 1,760 MWe is nuclear. Consumption of electric power, some 58 TWh in 1995, is provided from internal resources for 94%, of which nuclear plants provide approximately 21%. This power is produced in NPP Dukovany with four VVER 440 MWe blocks. Another nuclear facility is under construction: the Temelin plant shall operate two VVER 1000 MWe units. After their connection to the grid in 1997-8, the share of electric power produced by NPP should reach almost 45% of the national electric power production, and some coal burning plants will be decommissioned.

8.2 Organizational Structure of Waste Management

Radioactive wastes are produced and managed in the Czech Republic in three independent fields: uranium mining and milling, nuclear power plant, and other institutions (medicine, research, industry). The government controls all activities through the Ministry of Trade and Industry. The owner of uranium production facilities, DIAMO, is a state enterprise. The state also owns 67.46% shares of CEZ, which is the operator of Dukovany and Temelin NPPs. The company legally responsible for collection and disposal of institutional waste, NYCOM, is controlled by a "golden" share of 5% by the Czech state as well. Independent supervision of all activities is performed by the State Office for Nuclear Safety which regulates and licenses both from the point of nuclear and radiological safety. The Ministry of Environment indirectly influences these activities, as it is responsible for decisions in the Environmental Impact Assessment procedure.

The waste producers are obliged to treat, condition and dispose all radioactive waste at their own expense. However, especially for disposal activities, the responsibility may be transferred to another licensed organization. This procedure is obligatory for the disposal of institutional waste, as NYCOM is the only licensed operator of repositories for these wastes.

8.3 Waste Production and Management Strategy

The amount and characteristics of radioactive wastes produced in the Czech Republic vary according to the source. While uranium mining and milling produces huge volumes in the tens of millions of cubic meters of low contaminated material, other users of radioactive materials treat hundreds or thousands of tons of medium and highly radioactive wastes.

In the Czech Republic, uranium has been mined in seven regions and in some isolated shafts as well. Since the early fifties, about 100,000 MT of U have been produced. The ore was milled in at least four facilities. Wastes that have been produced during excavation are stored in surface piles. They contain more than 46 million MT of tailings contaminated with 5 - 300 ppm of U and 0.06 - 2.4 Bq/g of ^{226}Ra. Nearly 57 million m^3 of milling waste are stored in tailings ponds; the solid phase is contaminated with 0.3 - 5 mg/l of U, 5 - 30 Bq/g

^{226}Ra, and other chemical and radioactive pollutants. Furthermore, there are some 270 million m^3 of contaminated underground water that must also be treated. Rehabilitation of sites with surface deposits from uranium facilities is in progress; mine tailings are used for closure of abandoned mines while sludge ponds are dewatered and overcovered with low permeable soil layers. This is the basic and quite successful method of environmental protection against an influence of natural radionuclides.

The operation of NPP Dukovany produces annually 400 - 500 m^3 of conditioned low and intermediate level radioactive wastes. The prevailing radionuclides are typical fission products, such as ^{137}Cs, ^{90}Sr, and activation products (^{54}Mn, ^{60}Co). Liquid effluents are evaporated and bituminized while solid wastes are segregated, compacted and, when suitable, supercompacted. Sludges and ion exchange resins are dried and disposed in polyethylene High Integrity Containers. All these wastes are buried in the surface repository at the Dukovany site. It is designed so that it can also accept operational wastes from NPP Temelin; their amount and form after conditioning are basically the same as from NPP Dukovany. As the decommissioning of both facilities generates nearly 50,000 m^3 of wastes acceptable for disposal in this repository, its capacity will have to be enlarged, unless some more volume saving treatment methods are applied. Nuclear power plants, however, also produce wastes that shall be disposed of in a deep geological repository (DGR). The significant part of those wastes are in the form of spent fuel (nearly 3,000 MT) and decommissioning high level waste (HLW, some 20,000 m^3).

Institutional wastes were already produced in the Czech Republic in the 1920's; nevertheless, they have been treated, conditioned and disposed of at a central location since the late 1950's. During this period nearly 3,400 m^3 of wastes with gross activity exceeding 10^{16} Bq were accepted for disposal, currently with an annual increment of several tens of m^3. The wastes were contaminated mostly by ^{60}Co (54%), ^{137}Cs (37%), ^3H (7%), ^{241}Am (1%), etc. Sealed sources are kept at the storage site, or, when acceptable, grouted in drums with concrete. Liquid wastes are solidified with concrete (in seventies by bituminization) and solid wastes are low pressure compacted.

8.4 NEAR SURFACE REPOSITORIES

Intensive utilisation of radioisotopes and nuclear research facilities required commissioning of repositories able to accept institutional wastes. For this purpose the Hostim repository was the first in former Czechoslovakia which was put into operation. From 1953 until 1965 about 400 m^3 of institutional wastes were placed in two galleries in an abandoned limestone mine several tens of meters below the surface. The predominant disposed radionuclides were ^3H, ^{14}C, ^{60}Co, ^{90}Sr, ^{137}Cs with a total activity 0.1 TBq. Packages with higher activity and long lived nuclides were transferred to other facilities prior to the repository shut down. Currently, backfilling of void spaces in galleries is proposed for stabilization and ultimate closure.

In 1964, a new repository went into operation. It was also situated in an abandoned limestone mine 40 - 60 m below the surface: the Richard II complex was used as an underground military factory during World War II. This facility is designated for institutional wastes with the exception of those contaminated by natural radionuclides. The total activity of 2,700 m^3 of disposed wastes has reached 10^{16} Bq mostly formed by ^{90}Sr, ^{241}Am (both 31%), and ^{60}Co (30%). Approximately 95% of the total activity is in the form of sealed sources, while the main unsealed nuclide is ^3H. The overall available capacity is estimated to be 3,800 m^3 of wastes, although the expansion of disposal spaces for a further 2,800 m^3 has been considered. This reconstruction is strictly opposed by local authorities.

The third repository, Bratrství, is used for wastes contaminated only by natural radionuclides (^{226}Ra, ^{210}Po, ^{210}Pb, uranium and thorium isotopes). The facility was built in an abandoned uranium mine (granitic host rock) in Jáchymov, well-known town where M. Curie-Sklodowska took tailings for her experiments. During its operation, that started in 1974, about 250 m^3 of conditioned wastes were put there. The remaining capacity of 40 m^3 of wastes will be filled within 3-5 years. The major nuclides contained in these wastes are as follows: ^{226}Ra (10^{12} Bq), ^{232}Th (10^9 Bq), and others (10^9 Bq). Some wastes (mainly alfa bearing) are to be transferred to a deep geological repository after it is available.

A surface repository for low and intermediate level wastes has been in operation since 1993 at the Dukovany site. The repository spaces are formed by two double-rows of vaults, the dimensions of each are approximately 6x6x18 m. The existing volume, in 112 vaults of approximately 60,000 m^3, can be extended by construction of 8 new double rows. The radioactive inventory shall not exceed 10^{16} Bq. The facility has a fully engi-

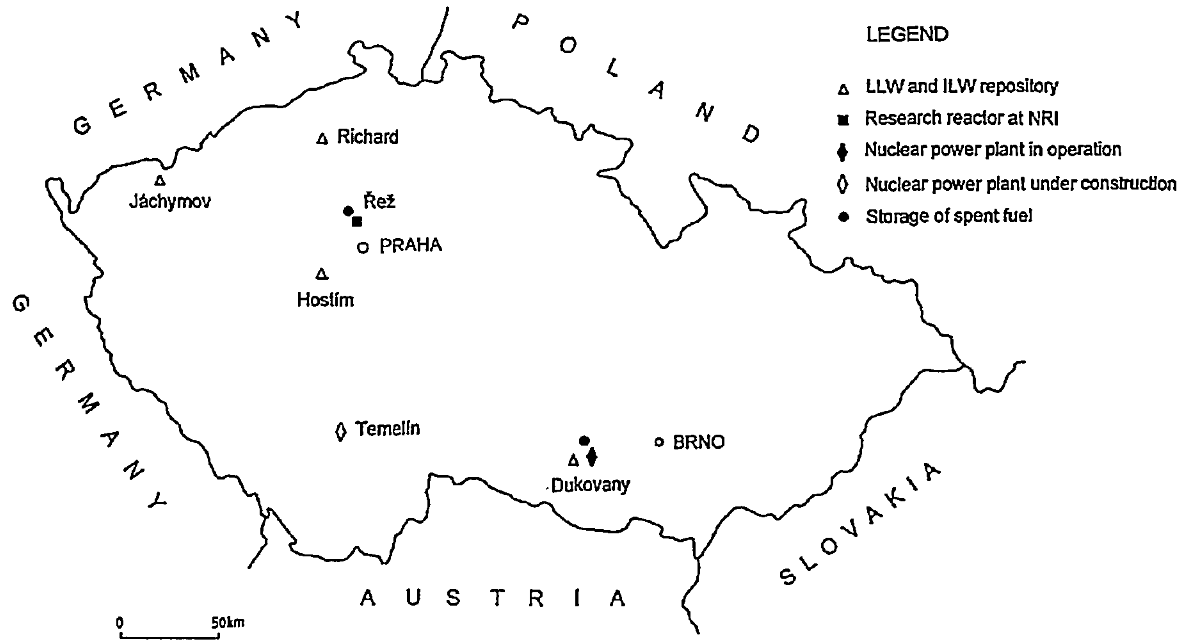

Figure 8.1. Location of nuclear facilities in the Czech Republic.

neered structure; reinforced concrete walls are isolated by asphalt-propylene microconcrete layer. The water movement on site is controlled by three independent drainage systems. A final multilayer capping will be installed after the doublerows are filled with waste. The void spaces in vaults are to be grouted with a concrete mortar.

Geographical locations of all the above mentioned repositories are shown in Figure 8.1.

8.5 SPENT FUEL MANAGEMENT

Spent fuel is not considered to be a radioactive waste in the Czech Republic. However, the producers must deal with it as a high level waste. Production of the spent fuel depends on the time of operation of the nuclear reactors. Provided that each unit will be in operation for 30 years, the Dukovany plant will produce fuel assemblies containing 1,504 MT of uranium metal, while the Temelín facility production will be 1,342 tons. The third producer, the Nuclear Research Institute in Rez, operates two research reactors of which one is used for irradiation (LVR-15 MW) while the other is only a critical assembly for physical studies. At the present time, there are more then 300 bundles of different types in storage at NRI.

The current scheme of the NPP spent fuel management is based on the assumption that it will be directly disposed of after some 50 years in storage, . However, no final decision has been made. Therefore, a dry in-cask storage facility was put in operation at Dukovany in 1995. It uses iron casks CASTOR VVER 440 developed by GNS (Germany) and produced in cooperation with SKODA WORKS Plzen. Each cask can contain 84 assemblies. The capacity of this facility was limited by a governmental decision that respected the requests of local authorities to 600 MT of heavy metal. This practically means that by 2005, a new storage facility must be operational. Its capacity shall be at least 2,200 MT of uranium, with enough reserve for a considerable period of NPP operation.

Spent fuel from the research reactors is kept in dry storage and a water pool within the NRI. Due to the high enrichment (80, resp. 40 %), it will probably be reprocessed so that the fissile material is under international supervision.

8.6 DEEP GEOLOGICAL DISPOSAL PROGRAM

Some activities aimed at the research aspects of a deep geological repository in former Czechoslovakia started in the mid eighties. Nevertheless, problems concerning disposal of HLW and spent fuel were studied separately in different organizations and institutions until 1992. In 1993, the Nuclear Research Institute Rez plc. (NRI) was

entitled to coordinate the activities which would lead to the development of the Czech geological repository. This year, a general geological disposal program was released and successfully reviewed by a WATRP mission (organized by IAEA) with recommendations for further direction of the program. Since 1994, the Program has been supervised by a Steering Committee of Six Institutions (C6).

The essential technical parts of the deep geological disposal program are:

- siting (and related geological activities such as an underground research laboratory and natural analogue study);
- repository design and disposal engineering;
- development of engineered barriers;
- safety analyses and performance assessment; and
- alternative disposal options (transmutation).

QA/QC, program/projects management and integration, program information systems, and external relations are additional activities of the Program. The Program has been focused mainly on preliminary and generic studies so far, and it considers hard host rock as a most probable option.

8.6.1 Repository design and disposal engineering

Preliminary conceptual disposal system design including infrastructure has been worked out. The present reference design considers two access shafts (one for personnel, one for waste), one ventilation shaft and a system of main and horizontal parallel disposal drifts. Other alternative designs are proposed that differ mainly in access tunnel design (for example spiral and inclined access to the underground). Particular alternative designs were evaluated preliminarily on the basis of safety and feasibility.

An optimization study of spent fuel conditioning prior to disposal indicates a preference for minimum restructurisation of fuel assemblies. An encapsulation unit is designed as an integral part of the disposal system at the site. Several repository transport and handling system options have been proposed.

The preferable excavation method for underground spaces has not been selected yet; the importance of the extent of disturbed zones is questionable. The problem is analogous to an underground cavern built for storage of gas; this was mined in granitoid host rock more than 1000 m below the surface in the Pribram uranium mining region. The alternative use of backfill and sealing materials is also studied.

Calculations of radionuclide contents and heat output (ORIGEN code) for spent fuel with different burnups have been performed. According to preliminary heat transfer calculations an optimal configuration for disposal rooms and disposal boreholes was designed.

A cost analysis of the disposal system was also carried out. The objective was to propose an algorithm for determining the waste producers fee for disposal of radioactive wastes (0.05 CZK/kWh). This study estimates an overall disposal system cost of approximately 100 billion CZK (i.e. approx. 3.5 billion US$).

8.6.2 Development of engineered barriers

Bentonite (and bentonite-based mixtures) was selected as a suitable buffer and backfill material. A bentonite availability study identified many deposits in the Czech Republic; the main criteria considered were material quality and planned volume of buffer and backfill. A preliminary design of a disposal container was proposed by the main Czech steel product manufactures; alternative materials, such as titanium and stainless steel, were also considered.

Planning of experiments, such as spent fuel leaching, engineered barriers degradation, corrosion of disposal container, has started. The main objective of these experiments is validation of models. Both laboratory and field experiments are to be used.

8.6.3 Performance assessment

Performance assessment is considered to be of key importance in the Program because of the necessity to prove long-term safety of the disposal concept. In a preliminary stage, all near-field and far-field features, events and processes (FEPs) have been continually identified, databased and evaluated. Suitability of codes shall also be evaluated. A safety analysis methodology was studied, including methods of probabilistic safety analysis (PSA). In 1995, a study using the Czech format for an Environmental Impact Assessment was performed.

8.6.4 Alternative disposal options

Transmutation technology has been studied as a complementary but not fully replaceable option to the disposal of spent fuel. Preferably, accelerator driven sys-

tems seem to be the most promising. When considering the practical aspects, many unsolved problems have been identified, such as continual hot chemical separations, complicated physical systems, non-verified technology, unresolved benefits and costs. Although mainly generic and system studies were performed, long-term experience of NRI staff with fluorine chemistry and reactor physics as well as the theoretical background of the Nuclear Physics Institute, skills of employees of the SKODA machinery works and promotion of reactor designers speak for reasonability and good starting points for initiation of the respective research and development experiments. Transmutation is a typical problem requiring wide international involvement, and thus, Czech specialists are linked with several US, European and Japanese institutions.

8.6.5 Public/external relations

Taking into account the high population density in the Czech Republic and the deep environmental feeling coming from a damaged natural environment, the strong NIMBY syndrome and general disapproval with the disposal concept are expected. As a consequence, the societal and political factors in a deep geological repository program will be very important.

The communication program is aimed not only to inform the general public but it shall also initiate some direct public participation in decisions. The communication policy is based on open, full and understandable information for the public. Current activities include publication of a series of informative articles in newspapers and magazines, editing of purposeful brochures, preparation of video presentations, presentation of the Program on the conferences and lecturing at universities. RACE methodology (research, analysis, communication, evaluation) is being applied; the final step is carried out in the form of a public opinion poll. Poor knowledge of radioactive waste management rather than dissent from any disposal solution and, surprisingly, general agreement with nuclear power generation are the main findings of the first public opinion polls.

An inseparable part of external relations is cooperation with foreign organizations involved in radioactive waste management, such as ANDRA, GRS, AECL, and ENRESA.

8.6.6 QA/QC

A quality assurance system is being created along with the quality and management guides/manuals; especially ISO 9,000 and 14,000 are taken into account. The system follows two general directions: QA of the Program management and QA binding technical activities within performance of the Program.

8.6.7 Scope of work in 1996

In 1996, the Program activities are focused on the revision of the Deep Geological Disposal Program. The objectives are to formulate general as well as detailed projects and plans based on evaluation of acquired experience and on information gained since the start of the Program. System re-engineering principles will be explored; they involve the complex, multidisciplinary and time/cost consuming character of the problem to be solved. License, technical and system requirements are being identified along with critical points of the Program. The year 2035 is the main time constraint, when the deep geological repository shall be operational.

8.7 GEOLOGICAL ASPECTS OF DEEP GEOLOGICAL REPOSITORY

8.7.1 Brief Geological History of the Czech Massif

The selection of an optimal geological unit for DGR is, among others, always limited by the geological structure of the territory. Moreover, in the case of the Czech Republic, this fact is underlined by a small areal extent and by a significantly complicated geological structure.

According to the classical geological Stille's division, the Bohemian Massif, belongs to the Meso - Europe, e.g. the vast zone, which was consolidated in the late Paleozoic period during the Hercynian orogeny. Within the European Hercynides, the Bohemian Massif ranks among the Variscides which create partial branches of the Hercynides. The Bohemian Massif, which represents the easternmost known part of the Variscan branch, forms the absolute majority of the territory of the Czech Republic. Brunovistulicum, a block consolidated during the Cadomian orogeny or even earlier, is situated in the eastern part of the territory of the Czech Republic. In the west, this block forms the underlying unit of the eastern part of the Bohemian Massif; in the east, it forms the underlying unit of the West Carpathians, which belong to the Alpine-Carpathian orogenetic belt (Neo - Europe).

The Hercynian history was closed by folding associated with emplacement of numerous plutons and extensive

regional metamorphism. This folding had reworked the whole region and often obscured the earlier, Cadomian and probably also some Caledonian structural elements. From this point of view, the Bohemian Massif has a tripartite structure consisting of the following vertical groups:

- Precambrian (Cadomian) basement;
- Crystalline and Paleozoic affected by Hercynian orogeny; and
- Post-Hercynian platform cover.

8.7.2 Basic geological requirements for host rock

The basic requirements for the geological unit that should become a host structure for DGR can be defined as follows:

- sufficient area (tens of square km) and thickness (at least 1000 m);
- lithological homogeneity of the unit with minimal hydrothermal alterations and rock veins;
- tectonic fractures and displacement as low as possible;
- no deposits and indications of mineral resources;
- permeability as low as possible; and
- seismic and geodynamic stability of the region.

8.7.3 Geological units of the Bohemian Massif from standpoint of basic requirements for host rock

Individual geological units can be evaluated from the point of view of suitability for DGR siting as follows:

Post-Hercynian platform cover. Post-Hercynian platform cover on the territory of the Czech Republic can be divided according to age into:

- Permo-Carboniferous sediments;
- Mesozoic sediments; and
- Tertiary sediments.

Permo-Carboniferous sediments occur in the territory of the Czech Republic in two different forms. In the northeast part of the Czech Republic, only a basin is situated where the parallel Namurian developed gradually from the marine Lower Carboniferous. In other parts of the Czech Republic, there are numerous limnic Permo-Carboniferous sediments placed discordantly upon various older units. We regard Permo-Carboniferous sediments as unsuitable for DGR because of their lithological variety and relatively extensive tectonic displacement.

The absolute majority of Mesozoic sediments in the territory of the Czech Republic are of Upper Cretaceous age. Continental sediments of Lower Triassic and epicontinental sediments of Upper Jurassic have no significance because of their small range. Lacustrine sediments of the Upper Cretaceous are known in the southern part of the Czech Republic, where they filled two separate basins. As far as their lithological nature is concerned, they are composed of sandstones, siltstones and claystones with rapid vertical as well as horizontal changes. In the north of the Czech Republic the Bohemian Cretaceous Basin is widespread. Its area is approximately 14,000 km^2. The largest supply of high quality drinking water in the Czech Republic is found in sandstones. Mesozoic sediments are regarded as unsuitable for DGR because of their lithological nature.

The majority of Tertiary sediments is mostly of lacustrine origin. They occur, on the one hand, in the northwest of the Czech Republic (accompanied by alkaline volcanism of Tertiary and Quaternary age), where they form the filling of the vast, tectonically predisposed, system of basins, and, on the other hand, in the south of the Czech Republic, forming the overlying beds of Upper Cretaceous sediments. As far as their lithology is concerned, they are characterized by rapid horizontal, as well as vertical, changing of clays and sands. In the northwest there are important and intensively exploited lignite seams, which are part of these sediments.

As far as the DGR is concerned, the Tertiary sediments do not have desirable features, especially because of their lithological variety and significant tectonic fractures.

Crystalline and Paleozoic affected by Hercynian orogeny. This group includes metamorphosed pre-Cambrian and Paleozoic rocks and non-metamorphosed rocks of Lower Paleozoic (Cambrian to Middle Devonian). Cadomian and Hercynian plutonic rocks are numerous.

Non-metamorphosed Cambrian to Middle Devonian formations occur in several separated basins. The formations in question are predominantly shales, siltstones, carbonates, sandstones and conglomerates, often accompanied by volcanism. In our opinion these formations are not suitable for DGR siting, because of their intensive folding, tectonic displacement and in view of

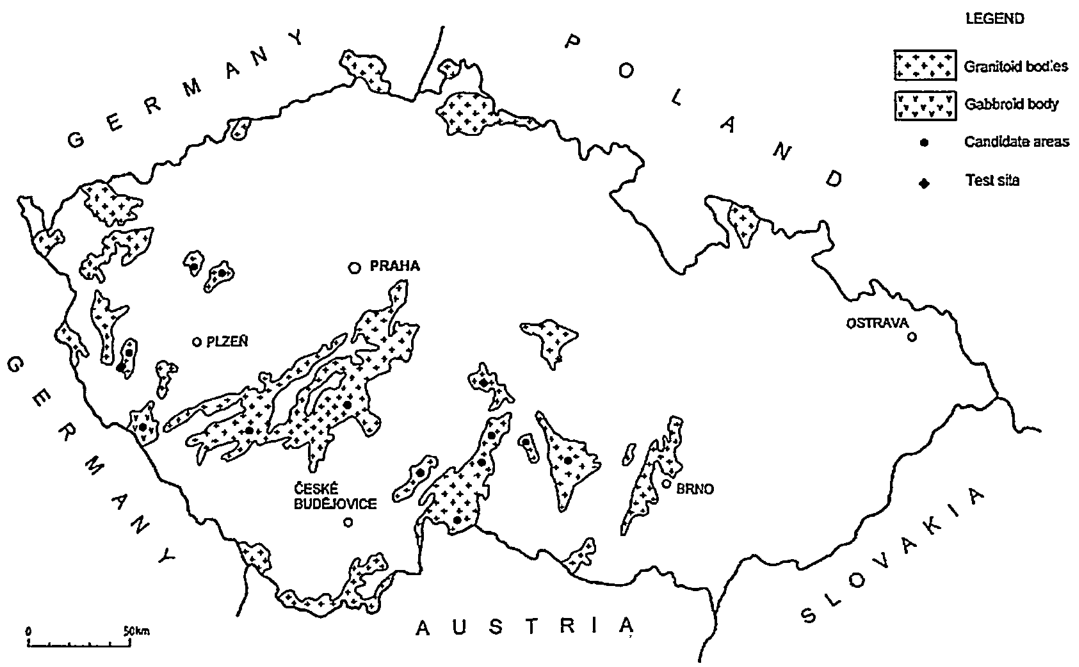

Figure 8.2. Location of candidate areas and test site in the Czech Republic.

varied lithological composition and karstification of limestones.

Units of metamorphosed rocks prevail in this group. Elements of Cadomian metamorphism appear in most of these units. According to geological and geochronological data, the Cadomian metamorphic activity can be observed from 650 to 550 Ma. Presence and role of the Caledonian metamorphism is still discussed by experts. The Hercynian regional metamorphism is most significant in the territory of the Bohemian Massif. This metamorphism started in the Ordovician, culminated in the Upper Devonian and ended in the Carboniferous.

In view of the very complicated tectonic structure, often and rapidly changing intensity of metamorphosis and lithological variety of metamorphosed units, where moreover numerous intercalations of crystalline limestones, graphite layers or quartz veins occur, we do not regard the metamorphosed units as suitable host rocks for DGR.

The occurrence of extensive plutonism is a characteristic feature of the above mentioned group (Fig. 8.2). The following plutonic complexes have been distinguished:

- metaplutonites of uncertain age, which occur predominantly in the form of orthogenesis in marginal parts of the Bohemian Massif;
- plutonites associated with Cadomian orogeny, which form several granitoid and gabbroid bodies in the west and north parts of the Czech Republic; and
- plutonites associated with Hercynian orogeny, which are the most frequent on the territory of the Bohemian Massif.

The Hercynian plutonites are exclusively granitoids. The two largest bodies - the Moldanubian Pluton with area of 8,000 square km approximately, and the Central Bohemian Pluton with the area of 3,200 square km approximately, as well as many smaller bodies belong among them. According to geochronological data, the age of the majority of the above mentioned granitoids is between 350 - 300 Ma, and their development is connected to the late tectonic phase of Hercynian metamorphic stage.

We consider granitoid bodies connected with the Cadomian as well as Hercynian orogeny as suitable for DGR .

Precambrian (Cadomian) basement. Precambrian basement untouched by the earlier Hercynian orogeny occurs in the territory of the Czech Republic just in Brunovistulicum. This unit as a whole is not suitable for DGR location especially because of its significant geo-

logical variety.

8.7.4 Description of General Project Objectives

The present state of activities on DGR siting in the Czech Republic results from work which was performed during the last 5 years. An area selection of 27 suitable regions was performed on the basis of existing regional geological information by the Czech Geological Survey.

The siting process continues with the approval of the General Project of Geological Activities for Deep Geological Repository Siting , which was elaborated by NRI staff. The project cut the number of potential areas from 27 to 13. From this number, 12 areas are located in granitoid bodies and one in an ultrabasic body. The area of relevant sites ranges from 120 to 20 square km.

The General Project became a principal document, upon which all geological activities concerning development of the deep geological repository are performed. Principal geological activities are planned in accordance with the General Project for approximately the following 15 years. It is assumed that the deep geological repository will be put into operation in 2035.

The activities described in the project are divided into the following stages:

- research activities;
- seismicity, geodynamics and neotectonic monitoring;
- critical evaluation of pre-existing geological information;
- site characterization ; and
- site confirmation.

The project does not deal in details, neither with an underground laboratory nor natural analogues. These activities are supposed to be solved by independent projects, whose time schedule and logical structure will be co-ordinated with the General Project, especially with the stage of research activities.

Stage of research activities. The main task of this stage is development, practical verification and elaboration of obligatory methodological procedures relevant to the deep geological repository siting. These activities will be performed at the test site selected in geological conditions similar to those, where the construction of the repository is proposed. The site in question is Moldanubian Pluton apophyse. DGR will not be built at the testing site, but we will consider the siting of an underground laboratory there. Currently, characterization of the site is being performed. The project of research activities will be completed in 1997, and the activities are supposed to start in the same year.

Stage of seismicity, geodynamics and neotectonics monitoring. In view of the above mentioned geological structure of the Bohemian Massif and its incorporation in the European Hercynides, no significant motions can be expected, but on the other hand insignificant motions cannot be excluded with sufficient certainty. However insignificant the motions might be, they could negatively influence the DGR safety. That is why we pay the necessary attention to monitoring the stability of the relevant part of the Bohemian Massif.

These activities will be started at the end of 1996 or in the beginning of 1997. In a preparatory phase, all available information on hardware and software concerning scanning data and their transmission to the processing center as well as their processing and evaluation will be gathered. Simultaneously, selection of suitable posts for seismometers, determination of sites suitable for neotectonic studies, densification of the network of existing observing points will be realized. The remote sensing evaluation will also be realized within this stage.

Stages of critical evaluation of pre-existing geological information, site characterization and site confirmation. These stages represented activities, which will result in selection of the optimal site for DGR in the territory of the Czech Republic. The activities began in 1995 and at present, a critical evaluation of available geological information is in progress. Starts of the next two stages are scheduled for 1998 and 2003. Contents of the single above mentioned stages are as follows:

The stage of **critical evaluation** of pre-existing geological information includes collecting of geological data from archives. The principal evaluating criterion is the nature and quality of each information and its utilization for siting. The data being evaluated concern geology, geophysics, hydrogeology, hydrology, climatology, geochemistry, seismology, but also demography, environmental protection and the most significant conflicts of interests. Approximately 7 sites are supposed to be selected from existing 13 at the end of this stage and recommended for the next stages.

Within the **site characterization** stage, a non-destructive (not disturbing rocks in the depth, where DGR might be situated) geological survey will be performed at all sites. This stage will include:

- geological mapping at a scale of 1 : 10,000 and structural geology studies;
- aerial and surface geophysics, especially gravity measurement, magnetometry, geoelectric methods, logging measurements in all bore holes and other necessary methods;
- boring works down to a depth of 200 - 300 m;
- detailed petrologic and mineralogic studies;
- geochemical mapping;
- hydrological and hydrogeological characterization of the site; and
- engineering-geological mapping and geotechnical characterization of the site.

These activities are supposed to be performed on sites with an area of 10 - 50 square km each. In addition, the granitoid body surrounding metamorphosed complexes will be examined to the necessary extent in justified cases.

A detailed descriptive characterization of the whole territory will result from this phase. On the basis of this characterization, a final decision on the suitability of the examined site, or its part for DGR siting, will be made. This stage results in a selection of two or three sites for further investigation.

The **site confirmation** stage. Destructive phase of activities will be started in one of the candidate sites, the others will serve as spare ones. Boring and special geophysical methods will be used predominantly in this stage. According to the assumptions of the General Project, the details of the verified site for DGR construction in the Czech Republic should be known by 2010.

8.8 CONCLUSIONS

Disposal of radioactive wastes has been practiced in the Czech Republic for more than 40 years. Different geologic environments were selected as host rocks for those subsurface facilities. This experience allows us to carry out successful research and development of a deep geological repository able to accept both decommissioning long lived and/or high level wastes and spent fuel from nuclear reactors. When considering the available geological systems and its variability in the Czech Republic, hard rocks, namely granitoid bodies, seem to be the most hopeful rocks for construction of a deep geological repository. The relevant national program was initiated in the early nineties aimed at development of this underground facility.

Technical abilities, however, are not fully supported by existing legal tools and the organizational system. Parallel management of radioactive wastes arising from different sources will probably survive till the newly prepared Atomic Law passes through the Czech parliament. In dealing with radioactive waste management, the governmental proposal contains several new principles, namely:

- state guarantees safe disposal of radioactive wastes, although the waste generators must bear the respective costs;
- import of radioactive wastes for their disposal on the Czech territory is forbidden;
- Nuclear Account, established out of the state budget, collects fees from waste producers to ascertain enough financial sources for final disposal of all radioactive wastes; the Account is controlled by a committee consisting of representatives of waste producers, state institutions and public; and it is under the supervision of the Ministry of Finance;
- new Office of Repository Management is created; it provides disposal services in existing facilities and is responsible for research and development in this field, including development of a deep geological repository; and
- a position of the State Office for Nuclear Safety as an independent supervisor in nuclear and radiological issues is stressed.

The Atomic Law shall be completed with a set of nearly 20 decrees providing the concrete regulations for users of radionuclides.

Even if the existing legislation is updated, siting of nuclear facilities, such as storage and disposal facilities, will have to face the high population density in the Czech Republic (130 inhabitants per square km) and all the negative consequences that this fact may imply. Nevertheless, it is believed that all kinds of radioactive wastes and spent nuclear fuel will be safely conditioned and disposed of in the Czech Republic.

Chapter 9

Disposal of High-Level Radioactive Waste in Finland

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Abstract. Detailed site investigations are under way on three candidate sites with the aim of selecting the site for a final repository for spent fuel in 2000. In May 1995, the Finnish nuclear power companies, IVO and TVO, made an agreement on the establishment of a joint company for final disposal of spent fuel in Finland, and the new company, Posiva, has taken over the work so far conducted solely by TVO for the disposal of spent fuel. After site selection, the first exploratory shaft will be constructed on the nominated site, and the actual construction of the repository will be started in the 2010's. According to the time schedule set by the Finnish Government in 1983, the repository must be available in 2020.

9.1 Introduction

About one third of the electricity produced in Finland is currently generated by nuclear power plants. There are two pressurized water reactors (PWR) with a combined capacity of 890 MWe at Loviisa and two boiling water reactors (BWR) with a combined capacity of 1420 MWe at Olkiluoto. The PWR power plant is owned and operated by Imatran Voima Oy (IVO), the BWR plant, by Teollisuuden Voima Oy (TVO).

According to Finnish legislation the nuclear power producer is solely responsible for the management and disposal of all radioactive wastes, including the spent fuel. Funds for future waste handling and disposal are included in the price of electricity and are set aside in a special fund under the authority of the Ministry of Trade and Industry (KTM). The Government has set the basic guidelines for waste management in a resolution of 1983 concerning the goals for research and planning in the area. These guidelines have been further focused and updated by decisions of KTM. In these guidelines and decisions, time schedules are set for construction of the final disposal facilities, and, in addition, some general principles are given for the research and development (R&D) programme.

The Loviisa power plant units are based on the Russian VVER-440 design and, in connection with the purchase of the reactors, the utility also made contractual arrangements for the entire fuel cycle service from the former USSR. This included both the supply of fresh fuel and the return of spent fuel back to the supplier. After the break-up of the USSR, the contract was endorsed by the Russian Government and the return shipments of spent fuel continued on a regular basis.

In late 1994, however, the Finnish Parliament passed an amendment to the Nuclear Energy Act which prohibits all exports and imports of nuclear waste, including spent fuel. A two-year transition period was granted for IVO's shipments of spent fuel to Russia, but from 1997 on, the spent fuel from Loviisa nuclear power station will be retained for final disposal in Finland.

In view of this new situation, the two nuclear power companies, IVO and TVO, started discussions on cooperation for a common disposal facility for their spent fuel. The companies already coordinated their nuclear waste research through the Nuclear Waste Commission of the Finnish Power Companies (YJT), and the question was how this cooperation could be enhanced in a joint undertaking.

These discussions were concluded in May 1995 in an agreement on the establishment of a joint company for the final disposal of spent fuel in Finland. From the beginning of 1996, the new company, Posiva, takes care of all the planning, research and development work needed and will later construct and operate the encapsulation facility and the deep underground repository. The mission neither includes interim storage of spent fuel nor the disposal of low- and intermediate-level operating wastes, for which the two power companies will

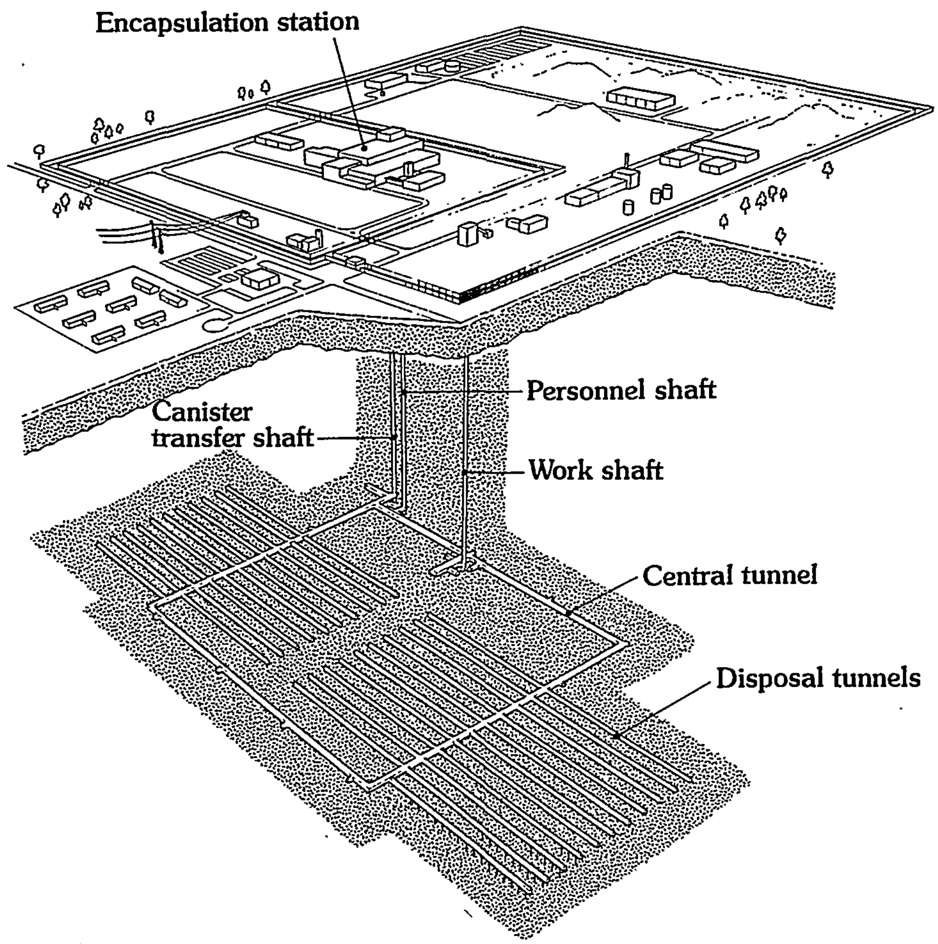

Figure 9.1. Final disposal facility for spent fuel.

have separate disposal facilities; TVO has had a repository in operation since 1992, and IVO has a repository under construction.

The programme for the final disposal of spent fuel will proceed along the time schedules originally set by the Government for the disposal of spent fuel from the TVO power plant. The plan is to dispose of the spent fuel directly, without reprocessing. According to the time schedule, the final repository for spent fuel must be available in 2020, and the site for the repository has to be selected by the end of the year 2000. Accordingly, the construction of the repository will be started in the 2010's. The total amount of spent fuel to be disposed of in the repository will be about 2500 tHM.

9.2 PROGRAMME STATUS

The technical plan for final disposal is based on the KBS-3 concept[1], but a new design for canisters has been introduced by TVO. Complete technical plans for disposal, together with a safety assessment, were submitted to the authorities at the end of 1992[2].

The concept consists of emplacing the packaged spent fuel in vertical holes in tunnels that have been excavated deep in the bedrock (Fig. 9.1). The repository depth will be somewhere between 300 m and 800 m depending on the geology and other characteristics of the site and also on various safety-related considerations. Fig. 9.1 depicts an idealized situation; in practice the tunnel network will be adapted to local bedrock features and may be split in several parts to avoid crossing major fracture zones. The packaging of the waste will take place at an encapsulation plant which will be constructed at the repository site at the surface. An elevator can then be used to transfer the waste packages directly to the repository.

In the new canister concept, the "Advanced Cold Pro-

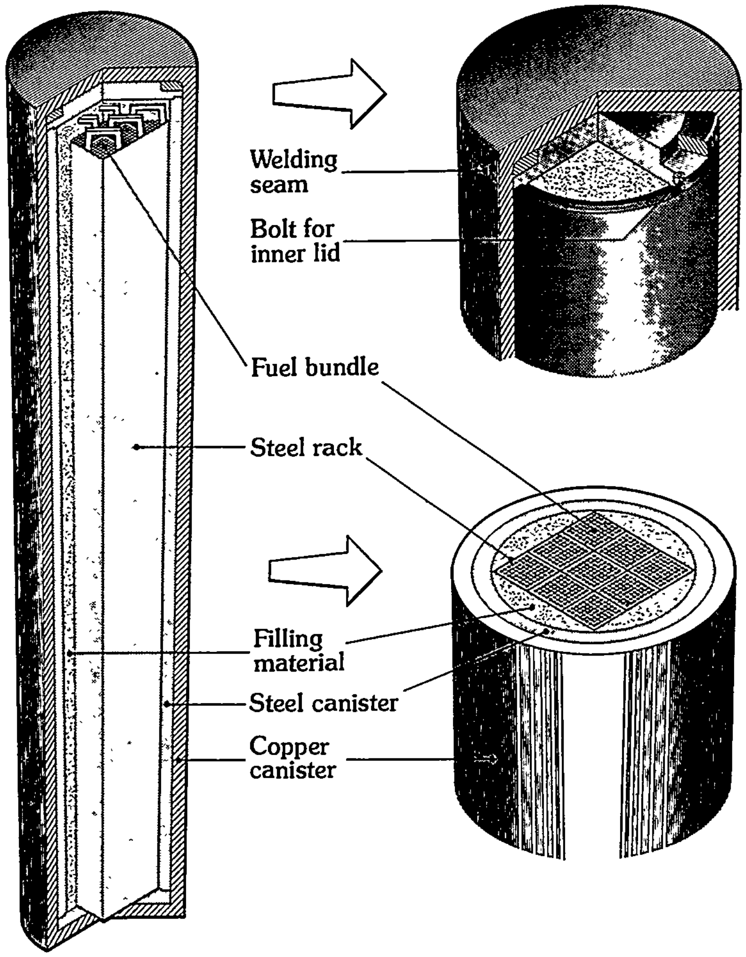

Figure 9.2. Advanced cold process (ACP) canister for direct disposal of spent fuel.

cess" (ACP) canister is a composite canister consisting of an inner container of steel as a load-bearing element and an outer container of oxygen-free copper to provide a shield against corrosion (Fig. 9.2). The wall thicknesses are 60 mm for the copper canister and 55 mm for the steel canister. The original canister design was only for 9 BWR fuel assemblies from the TVO power plant, but it has been modified for the hexagonal fuel elements from the Loviisa PWRs assemblies as well. The empty space in the canister can be filled with a granular material, such as lead shot, glass beads or quartz sand, but other alternatives are also being considered. Depending on the filling material used, the total weight of the full canister will be 14 to 20 tonnes.

The name of the canister design (Advanced Cold Process Canister) is due to the principal driving force behind the development work. The original KBS copper canister design assumes a lead casting to fill the voids between the fuel assemblies and canister walls. Since the lead casting would probably be the technically most difficult phase in the encapsulation process, the question was raised whether the use of molten lead could be avoided by replacing it with some "cold" material. However, the copper canister without the cast lead might not meet the mechanical requirements, mainly because of the creep properties of copper. In the ACP canister, the use of molten lead has been avoided because the mechanical strength is now provided by the inner steel canister.

For the selection of the repository site, the strategy is to find a "good enough" site, i.e., a site that offers sufficient characteristics to ensure the long-term safety of disposal. It is considered both unnecessary and unrealistic to try searching for the "best" site.

The site selection process for a repository started in 1983 with a country-wide survey of the principal geological features in Finland. The approach used for local-

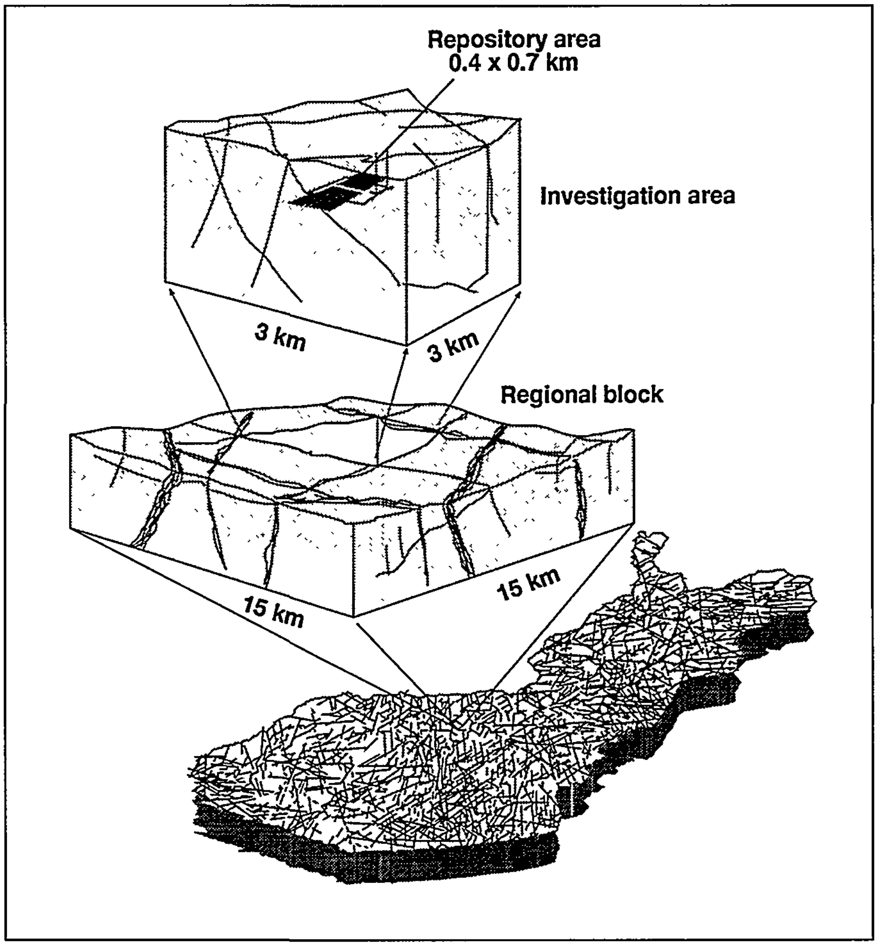

Figure 9.3. Principle site selection for the spent fuel repository.

ization of candidate sites was based on the crush-tectonic block structure of the Finnish bedrock. All the bedrock in Finland is broken down into blocks separated by regional crush (or shear) zones whose length may be dozens of kilometres (Fig. 9.3). These, in turn, are divided into smaller sectors according to the presence of smaller fracture zones. What is important for repository siting is that the rock response to future changes in stress conditions be mainly confined to these crush zones, while the rock outside the fracture zones remains basically unaffected. This will ensure sufficient stability for the host rock of the repository.

A total of 327 large regional blocks were identified in the first country-wide screening of the bedrock features. Mainly for land-use restrictions and other geographic reasons, the number of possible regions was later reduced to 61, among which 134 smaller areas were selected as candidates for investigation sites. In 1987, five areas were selected for preliminary site investigations. These are situated in different parts of the country and represent somewhat different geologic histories (Fig. 9.4). In addition to geologic factors, geographic factors, land ownership and transport and other infrastructure considerations were taken into account in the selection of the sites for preliminary site investigations. One of the areas chosen for investigations is the Olkiluoto island in Eurajoki where the TVO power plant is also situated.

The principal goal of the preliminary site investigations was to characterise the candidate sites to the extent needed to judge whether the preconception on their suitability for hosting a repository could be confirmed. The investigations at each site comprised drillings and samplings, various geophysical, geohydraulic, rock-mechanical, chemical and mineralogical studies, and modelling of the bedrock structure and groundwater

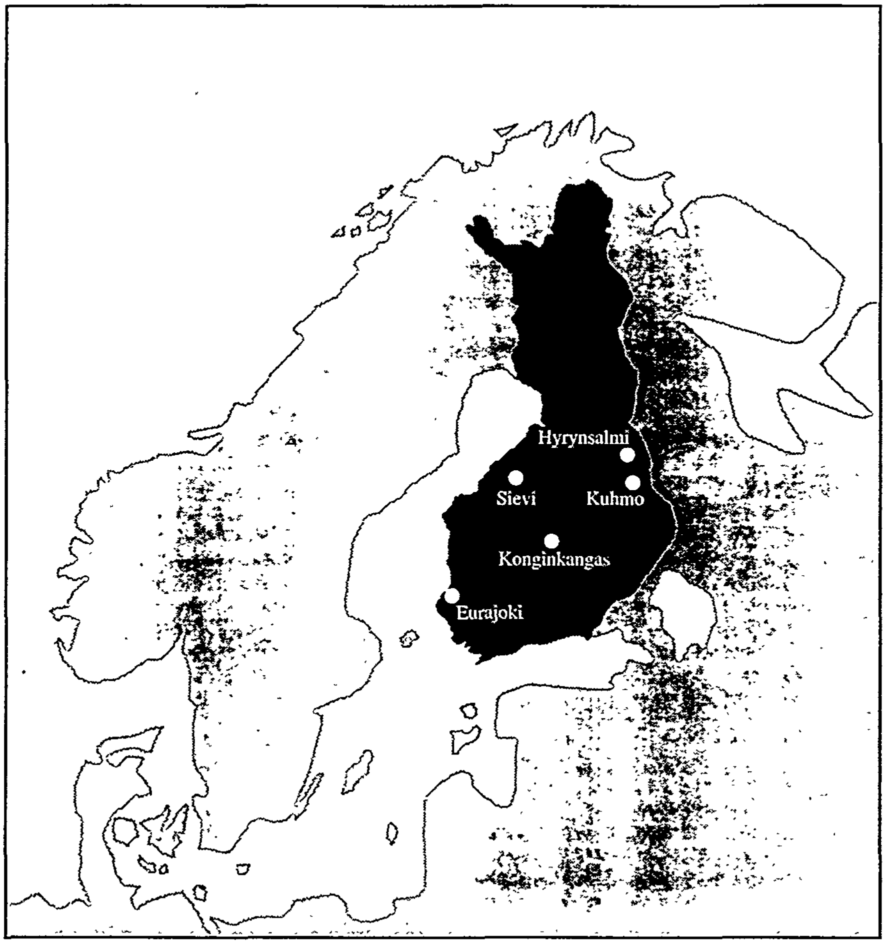

Figure 9.4. Preliminary site investigations areas in 1987-1992. Kuhmo, Konginkangas (since 1993 a part of Aanekoski town) and Eurajoki were selected in 1992 for detailed site investigations.

flow in the area. At least five deep (500 to 1000 m) cored boreholes were drilled in addition to a number of shallow boreholes. The results of the preliminary site investigations were submitted to the authorities at the end of 1992[3]. These results did not reveal any surprises in relation to expectations; no exceptional features were detected that would compromise the siting of the repository within the area studied.

In parallel with these site investigations, a safety assessment ("TVO-92") was carried out[4]. The analysis used the data from the preliminary site investigations but was not specific as to any of the sites studied. The principle has been to select the parameters and assumptions in such a way that a location could be found at each of the sites with certainly more favourable conditions than were assumed in the safety assessment. For example, very conservative assumptions were made with regard to groundwater flow at the repository site. It was assumed in the reference scenario of the analysis that the groundwater transit time from the repository to the biosphere is only five years.

The safety analysis includes a detailed groundwater flow analysis for a hypothetical repository located at one of the investigation sites. Figure 9.5 depicts how the repository could be adapted to the local fracture zones around the deep drillholes. It is seen that two fracture zones, R11 and R15, cross the repository area. If the transmissivity of the repository (together with the excavation damage zones) were high in relation to the surrounding rock, then in principle, a U-tube flow condition could form. Although, in reality, this would be very unlikely, such a situation has been taken as the basis in the reference scenario of the safety assessment.

The number of scenarios was kept small by restricting the analysis to a set of bounding cases that, nevertheless,

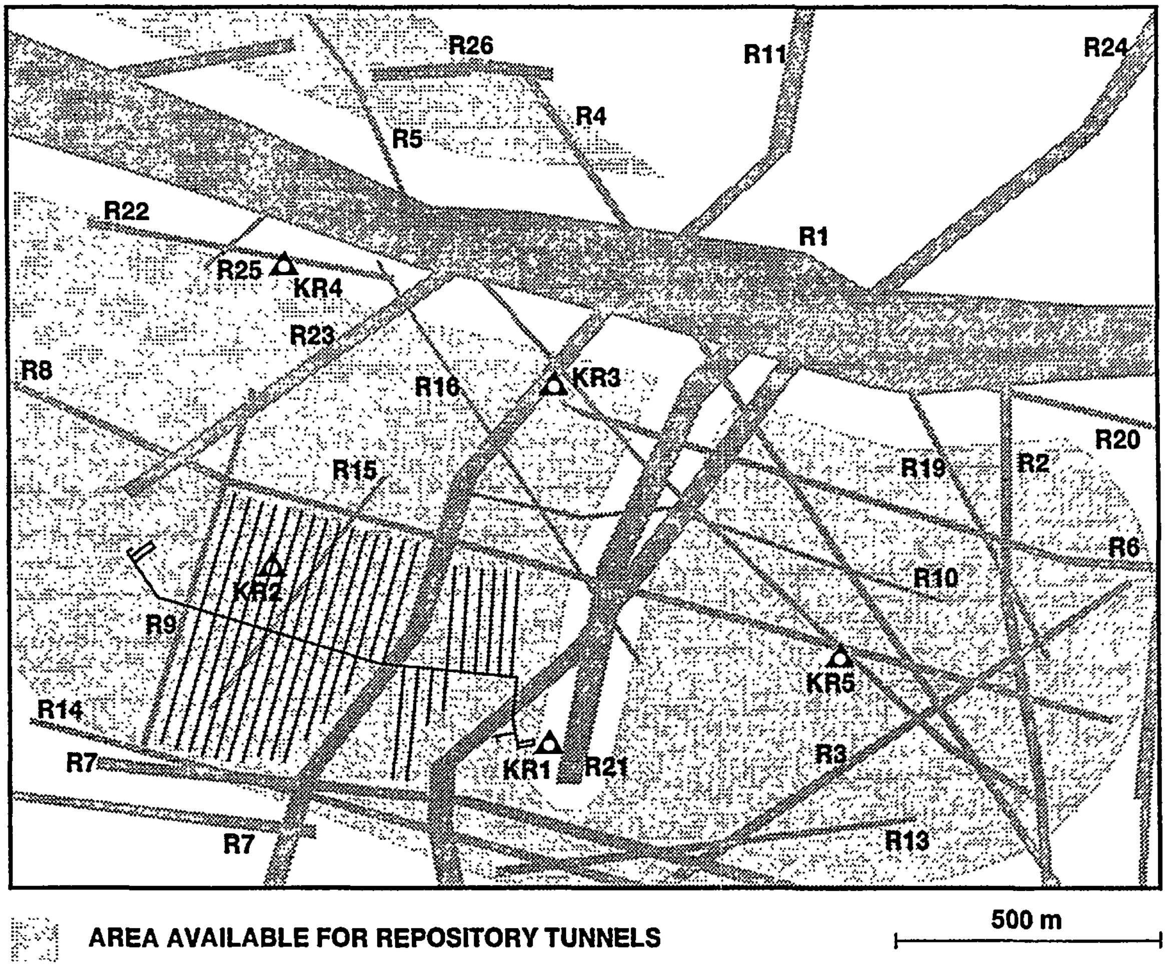

Figure 9.5. Adaptation of the repository to local bedrock features (example from Hyrynsalmi investigation site).

cover all reasonably conceivable situations. Aside from the base case (the conditions in the vicinity of the repository remain basically as they are now), the analysis has been centred on the reference case, which simply assumes a total loss of canisters after 10,000 years from the time of deposition. This simplification has helped to greatly reduce the number of different scenarios in the analysis.

Despite the conservative data and concepts adopted, the safety analysis shows that the criteria proposed by the Nordic safety authorities in 1993 can easily be met[5]. At the same time, the analysis shows that safety is ensured no matter which one of the candidate sites is chosen.

In addition to geology and long-term safety, various other aspects of the candidate sites were assessed, such as local infrastructure and available means and routes of transportation and constructability. It turned out that there were no decisive differences between the sites studied. However, for focusing the field work further, the decision was made to restrict the detailed investigations to three candidate sites: Eurajoki, Kuhmo and Äänekoski (formerly Konginkangas). The choice was mainly based on the praciticality of investigations; it was evident that at the two discarded sites (Sievi, Hyrynsalmi), investigations would have to be extended over a larger area than was well covered by the drillings during the preliminary site investigations. At the three selected sites, the future investigations could most efficiently make use of the already existing boreholes and other data.

9.3 FUTURE INVESTIGATIONS

Detailed site investigations are now under way on the three candidate sites with the objective of establishing the database needed for site selection in the year 2000.

In 1994, TVO also conducted a prefeasibility study in Kannonkoski, a neighbouring municipality to Äänekoski, as a possible site for the repository. On the basis of a geologic survey and other considerations, it was concluded that Kannonkoski could be an alternative to Äänekoski, in case there was need to search for new candidate sites; but for the moment, the three candidate sites are considered sufficient. Now, after the decision by IVO and TVO to work toward a joint repository, a similar prefeasibility study is being conducted at Loviisa, the site of the IVO power plant. The need for drilling and other field work at Kannonkoski or Loviisa will be considered at the end of 1996, when an interim report of all studies and investigations conducted since 1992 will be submitted to authorities.

For site selection, a site-specific safety analysis will be carried out for the nominated site and the technical concept by the end of 2000. Updated technical plans will be presented, including preliminary layouts adapted to the local geologic features. Later, in the early 2000's the first exploratory shaft will be constructed at the site, and the construction work for the repository itself can be started after 2010, assuming a favorable development with the licensing.

For drilling and other surface-based site investigations, the only license required so far is permission from the land-owner, who, in the case of the candidate sites, is the state or the power companies themselves. However, construction of the repository is possible only after a positive decision by the Parliament, and such decision can only be made if the local municipality approves the siting proposal. Since the local veto cannot be overridden, public acceptance will be crucial for the successful implementation of the plans. According to the present time schedule, the municipality will have to define its position on the proposed siting in about ten years. To gain local support for the plan will be a major challenge in providing for the ultimate disposal of nuclear wastes in Finland.

REFERENCES

1. Final storage of spent nuclear fuel, KBS-3, Swedish Nuclear Fuel Supply Co., 1983.

2. Final disposal of spent fuel in the Finnish bedrock: technical plans and safety assessment, Teollisuuden Voima Oy. Nuclear Waste Commission of Finnish Power Companies, Report YJT-92-31E, Helsinki, 1992.

3. Final disposal of spent fuel in the Finnish bedrock: preliminary site investigations, Teollisuuden Voima Oy. Nuclear Waste Commission of Finnish Power Companies, Report YJT-92-32E, Helsinki, 1992.

4. Vieno, T., A. Hautojärvi, L. Koskinen, and H. Nordman, TVO-92 safety analysis of spent fuel disposal, Nuclear Waste Commission of Finnish Power Companies, Report YJT-92-33E, Helsinki, 1992.

5. Disposal of high level radioactive waste: Consideration of some basic criteria, The Radiation Protection and Nuclear Safety Authorities in Denmark, Finland, Iceland, Norway and Sweden, 1993.

CHAPTER 10

STATUS OF RESEARCH ON GEOLOGICAL DISPOSAL FOR HIGH LEVEL RADIOACTIVE WASTE IN FRANCE

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10.1 INTRODUCTION

Research programs in the field of high-level, long-lived radioactive waste management are defined and regulated by Law 91-1381 of December 30, 1991 ("the Waste Act") and its implementing decrees, particularly Decree 92-1391 of December 30, 1992 pertaining to Andra and Decree 93-940 of July 16, 1993 on underground laboratories.

The Waste Act relative to research on radioactive waste management mandates three research programs on high-level and long-lived waste management:

- separation and transmutation of long-lived radioactive elements;
- solidification processes and long-term surface storage; and
- examination of options for retrievable and non-retrievable disposal in deep geologic formations,particularly through the construction of underground laboratories.

The first two programs are relegated to the French Atomic Energy Commission (CEA) and will be conducted in the latter's laboratories. The Act confers responsibility for the third research program on Andra. In this capacity, Andra is charged with conducting a program of research and testing to examine the potential for construction of a retrievable or non-retrievable repository in deep geologic formations. Andra is also called on to participate in the other two research programs in association with the CEA.

Article 13 of the Waste Act charges Andra with long-term management of radioactive waste and "to participate in defining and conducting research and development programs on long-term radioactive waste management, in association with the French Atomic Energy Commission in particular. "

Annual progress reports on the three research programs are required (Article 4), as is an overall assessment report, *"no later than 15 years after the promulgation of this law, accompanied by proposed legislation authorizing the creation of a repository for long-lived radioactive waste, as appropriate and establishing the conditions for essential and seats relating to this repository."*

These reports, to be prepared by the National Assessment Commission, are to be submitted by the government to Parliament, which refers them to the Parliamentary Office of Science and Technology Assessment. All reports will be made public.

The decree of December 30, 1992 relative to Andra additionally stipulates that the latter shall provide the following to its oversight ministries:

- an annual report on work performed to date or to be performed in the underground laboratories to determine the suitability of deep geologic formations for radioactive waste disposal; and
- a summary report on all findings no later than December 31, 2005, accompanied as appropriate by a design for an underground repository for high-level and long-lived radioactive waste.

This Technical Activity Report by Andra reviews studies performed to date and serves as a baseline for Andra research programs to be conducted over the coming

*Currently with International Atomic Energy Agency in Vienna.

years.

10.2 What is a Deep Geologic Repository?

The purpose of geologic disposal is to contain radioactivity with a system of devices and measures that prevents radioactive materials transport to the biosphere, or at least limits it to a specified level.

Containment is based on the concept that a series of natural and artificial barriers can be placed between the source term, i.e., the waste itself, and the biosphere, listed below:

- the waste package (waste, solidification material, container and possibly overpack);
- the engineered barrier (repository structures and backfill material around the waste packages); and
- the geologic barrier (natural geologic media and access closure system).

The geologic medium plays a dual role in this scheme of things. It provides protection with respect to the source term and it protects the biosphere:

- by protecting the artificial barriers from human intrusion and the effects of weather;
- by providing a physical and chemical environment for the artificial barriers that is stable over long periods of geologic time; and
- by helping to retard and restrict radionuclide transport to the biosphere (retention, dilution before reaching the biosphere).

The three barriers in the containment system reinforce each other, with the geologic barrier playing a critical role in long-term containment. The overall integrity of the three-barrier system must be demonstrated.

Reference conditions for the multi-barrier system must be taken into consideration in two areas:

- the waste, expressed in terms of radioactive content, physical characteristics and solidification materials; and
- the geologic medium, represented by the formation and its surrounding geologic environment.

10.2.1 Waste

Generators have been generating and storing long-lived waste for several decades pending the availability of a geologic repository. Most of this waste has already been solidified in a variety of materials in accordance with regulatory requirements: glass for fission products from reprocessing, and concrete or bitumen for dry active waste. Although Andra played only a small part in the preliminary selection of materials for this type of waste in the past, henceforth it will have a larger role, as spelled out in Article 13 of the law- "Andra is charged with defining, in accordance with safety regulations, radioactive waste solidification and disposal specifications." Andra could, therefore, require the use of additional solidification materials, for example, even for waste that has already been solidified waste, or could change the cooling requirements for exothermic waste.

For purposes of long-term management, radioactive waste is differentiated by type, by activity level and by the half-life of the contaminating radionuclides it contains, based on the duration of their potential hazardousness.

Low- and medium-level, short-lived waste, called Category A waste, contains primarily beta- and gamma-emitting radioelements whose radioactive half-lives are approximately thirty years or less (such as cesium 137, cobalt 60, strontium 90, etc.). This waste, generated by routine nuclear facility operations, represents approximately 90% of the total volume of radioactive waste generated annually in France, but accounts for only 1% of its total activity. The radioactivity of this type of waste decays to natural levels in less than 300 years due to the relatively rapid decay rate of the radionuclides it contains, and it may, therefore, be suitable for near-surface monitored disposal during this time frame.

The other type of waste, which is long-lived waste and high-level waste, is divided into two categories, each of which includes long-lived waste:

- low- and medium-level Category B waste contains significant amounts of long-lived radionuclides, especially alpha-emitting transuranics;
- Category C waste contains high concentrations of both short- or medium-lived fission products and long-lived, alpha emitting transuranics.

Category C waste is highly radioactive and heat-emitting in the beginning, primarily because of its fission product contents, but this quickly declines due to the rapid decay rate of these short-lived elements. In the end, Category C waste will contain mostly long-lived elements, and represents the same types of hazards asso-

ciated with Category B waste. For the most part, Categories B and C waste come from spent power reactor fuel reprocessing and from civilian and defense research and production sites of the CEA. Category B waste will also be generated during nuclear facility dismantling and must be included in waste volume estimates for disposal.

The waste volumes to be generated in the coming decades are not yet finalized due to several uncertainties, particularly:

- the amount of fuel to be used in power generation, fuel burnups, and the back-end option selected (reprocessing or direct disposal);
- potential technical advances in reprocessing, particularly in the types and volumes of waste generated; and
- method selected to solidify existing waste,

Taking a simplistic scenario — light water fuel reprocessing of 1100 metric tons of initial uranium metal per year at a burnup of 33,000 MWd/t beginning in the year 2000 — the total volume of solidified waste through the year 2020 can be estimated at 100,000 m^3 for Category B waste and 6400 m^3 for Category C waste.

Data from tests conducted as part of the other two research programs mandated by law could result in major technical shifts in solidification processes affecting the types and volumes of waste generated by reprocessing in addition to their solidification. Conversely, as data becomes available at the underground laboratory sites, Andra may identify new priorities in the other two research programs. For example, a hierarchy of radionuclides may be established in light of the retention properties of the multi-barrier system, which could establish separation and transmutation priorities for certain radionuclides. Or, Andra may develop preliminary specifications for solidification materials based on disposal conditions in deep geologic formations.

All aspects of the final waste packages must be carefully characterized to facilitate geologic repository design, Characterization must not be limited to waste volumes or to the physical, chemical and radioactive characteristics of waste packages that have been or will be fabricated; they should also include long-term behavior, failure probabilities, and their interaction with other barriers. In the case of exothermic waste, duration of storage is an essential parameter for repository design insofar as it is a factor in the waste's temperature; Andra's calculations must therefore include the date that the spent fuel that produced the exothermic waste was discharged from the reactor.

Studies of the long-term behavior of waste packages will make it possible to assess containment capacity — and possibly to find ways of improving containment — and to make sure that the waste packages are compatible with the repository concept (receiving capacity, engineered barriers ' handling, retrievability, etc.). This will result in waste package specifications that integrate both safety-related requirements and mechanical and physical requirements for waste package handling and disposal.

10.2.2 Geologic Barrier

Why dispose of radioactive waste in a deep geologic medium?

Surface disposal was rejected because the service life of any repository structures that could be built is much shorter than the half-lives of long-lived radionuclides. Moreover, it would merely postpone the problems of repository monitoring and maintenance to the future, with all the risks that that implies in the deliberately conservative scenario in which future civilizations are presumed to have neither the material means nor the technical capability to manage these problems.

Since the goal is to contain radionuclides for long periods of time, a medium with suitable radionuclide containment characteristics that evolves slowly is sought. Deep geologic formations may meet both of these criteria. There are vast areas of deep rock suitable for containing radioactive products for long periods of time.

At first glance, the low permeability and retention capacity of some of these rocks make them suitable for containing radioactive products. Enormous areas are available at depths of 200 to 1000 meters, a range that provides enough protection from surface intrusions yet does not compromise the technical feasibility of repository construction.

These available areas thus constitute physical barriers as much for the radioactive waste as for intrusions. They are also chemical barriers: water seeping through them is "buffered," acquiring geochemical properties that usually are not very corrosive through interaction with the rocks. This phenomenon is what explains how uranium deposits could subsist for hundreds of millions of years,

for example. It also explains why the fossil reactors discovered in the Oklo uranium deposit in Gabon are an extraordinary illustration of the influence of the geologic medium in the repository scenario; fission products produced during "operation" of these natural reactors have remained trapped in place, contained and protected for nearly 2 billion years in an envelope of clay.

In addition, inasmuch as the excavated rock volume for a geologic repository will be a small percentage of the host rock volume (around 1%), the repository should not significantly modify the overall containment capabilities of the host rock, especially since the volume of waste to be disposed of will be even lower (around 1/10%).

Geologic evolution is slow enough to ensure containment integrity for the time necessary for the waste to undergo radioactive decay, as long as certain areas are avoided, such as those with recent volcanic activity or strong tectonic activity. Potential geologic events are limited to those that are possible in the geodynamic context of the area under considerations for example, the creation of a mountain chain during the time scales considered (around one hundred thousand years) is not plausible in a stable area such as the Parisian Basin.

To illustrate the slowness of geologic phenomena, one could also take the example of an area with a great deal of tectonic activity, such as the Messine and Calabre regions, which have risen by 1,500 to 2,000 meters in 5 million years at a geologically rapid average rate of 0.4 mm/yr.. For a stable area such as the Parisian Basin, the average estimated rates are 20 times lower, discounting the appearance of significant new discontinuities (fault fractures, etc.) in the reference time scale.

In recognizing the potential for disposal in deep geologic formations, can one therefore consider that all geologic media are valid? A consensus was quickly reached among international experts on three suitable media: granite, salt and clay. In France, the Second Castaing Commission (1984) had this to say about rejecting certain formations: *"A number of formations, known to be permeable, were rejected outright (sand, sandstone, limestone, basalt, etc.")*. However, the Commission indicated its support for an extension of the initial selection list: *"In the first phase, three types of formations were selected at the European level -granite, salt and clay - perhaps without adequately emphasizing the variety of rocks that enter into these categories. The group considers that the list of rocks resulting from clay evolution (such as shale) or even their more highly metamorphosed forms (schist and some gneiss), which are widespread rocks, should not be rejected outright."*

Nonetheless, the fundamental role of the site itself was not forgotten in this expanded list: *"It is both a series of diverse formations and the interfaces between them that make a site attractive; the same formation at another site could be determined to be completely unsuitable."* The Commission came to this conclusion: *"Potential variation in the properties of the formations, ranked by category, are so great that research must focus on specific rocks and sites as soon as preliminary screening is completed."*

However, even the cited media have different theoretical advantages and disadvantages for disposal. In addition, one medium may be more suitable than another for exothermic waste disposal. Lastly, the geologic barrier will have to be reestablished by backfilling and closing the access shaft after the repository has been constructed, and the method of doing this differs for different media.

10.2.3 Engineered Barrier

Engineered structures and backfill material for the disposal pits and the surrounding galleries separate the waste packages from the geologic medium and constitute an additional barrier which, like the natural medium, plays a dual role:

- it protects the waste packages by minimizing water contact or by creating a chemical environment that is conducive to the long-term integrity of the packages; and
- it retains any radionuclides that may have been released from the waste packages.

The design of the engineered barrier should therefore take the geologic medium and the initial radioactive content of the waste into account, in addition to the exothermic properties of certain waste packages. This applies to C waste, unless the length of its storage is to be extended, which would lead to treating C waste separately from B waste.

10.2.4 International Consensus

International organizations agree that geologic disposal is the reference solution for waste management. Other possibilities for long-lived waste management were considered, but have been rejected. Sending waste into orbit in space has been rejected as it is technically infeasible at the present time, given the volumes involved and reli-

ability requirements. Waste burial in the earth's crust in the subduction zones of continental plates or in volcanos is not very realistic. Dilution by submersion in the sea is prohibited for long-lived radioelements by international regulations due to the risk of their return to members of the public. Burial in an ice cap was also considered; the heat of the C waste would serve melt the ice beneath each waste package, allowing it to gradually penetrate the center of the ice cap under its own weight, but implementation of this "solution" not only raises technical challenges, it also had the drawback of targeting areas that are increasingly being protected from industrial use.

France has participated in in-depth research, and an international test program has been conducted on waste burial in ocean bed sediments. Only simulated waste packages without any radioactive products were used during the program. On a technical level, and given the present state of the art, sub-oceanic burial is attractive from a safety point of view, but international legal considerations block its implementation.

Performance objectives established by the French regulatory authorities have been derived from recommendations of the International Commission on Radiological Protection (ICRP), an independent organization that analyzes the health effects of radiation exposure and develops recommendations for measures to be taken for protection from peaceful uses of artificial radioactivity. ICRP uses medical and epidemiological statistics and operating experience from a century of worldwide use of radioactivity.

Natural radioactivity comes from a certain number of radioactive isotopes that exist in nature. Natural radioactivity, and therefore exposure, varies significantly from one location on the earth to another; it varies considerably in France. Generally, the maximum allowable exposures to artificial radiation for members of the public must be less than the average exposure to natural radiation. In this very conservative approach, it is further assumed that an individual may be exposed to several different sources, and therefore, only a fraction of the allowable exposure may come from each source. The only exception to this rule is for radiation exposure for medical purposes, such as x-rays and radiation therapy, which may be much greater because the health benefits may be much higher than the related risks. ICRP recommendations are incorporated into the regulations of the International Atomic Energy Agency (IAEA) as well as into French regulations.

International organizations have been engaged in research on geologic waste disposal for some time. This was set forth in the December 14, 1990 report on high-level nuclear waste management by the Parliamentary Office on Science and Technology Assessment and is summarized below.

10.3 CHARACTERIZATION OF GEOLOGIC MEDIA

In the preceding section, deep geologic disposal is recognized as a potential solution for long-lived and high-level waste. To go one step further, specific selection criteria must be identified for the characteristics of a suitable geologic medium and site in terms of the specific use for which it is intended.

A major phase in the deep repository program involves the definition of: selection criteria, criteria ranking, determination of objectives to be achieved for each criterion, and identification of the means of verifying that the objectives are achieved in a geologic medium selected to host a geologic repository. Considerable work has already been performed in these areas in France, culminating in the June 1991 issuance of Fundamental Safety Rule (FSR) III.2.f by the Ministry of Industry's Division of Nuclear Facility Safety, whose subject matter is described as follows: "Definition of objectives to be achieved during the design and construction phase of the deep geologic repository for radioactive waste to ensure safety after the end of the repository operating period."

FSR III.2.f is the culmination of deliberations and recommendations of several advisory groups that have been working since 1983 to develop a safety-related approach to radioactive waste repositories in deep geologic formations. These include:

- working group on research and development in the field of radioactive waste management, chaired by Professor Castaing;
- working group on technical selection criteria for a geologic disposal site for radioactive waste, chaired by Professor Goguel; and
- working group on scenarios to be used in safety analysis of a geologic repository.

In addition, FSR 111.2.f takes the recommendations of cognizant international organizations into account, including those of the ICRP, IAEA, and OECD/NEA.

Two essential criteria are highlighted in the FSR: site stability and site hydrogeology. These are followed by important criteria, such as the mechanical and thermal characteristics of the medium, which determine the fea-

sibility of repository construction, operation and closure-, and the geochemical characteristics of the media that could alter man-made barriers and determine radionuclide retention.

A minimum depth must be maintained to prevent the containment performance of the geologic barrier from being affected significantly by erosion (especially after glaciation), by a seismic event, or by the consequences of *"direct or indirect human intrusion (drilling, milling, wells, surface or subsurface construction)."* This last criteria translates into the need for geologic formations with suitable characteristics (mechanical characteristics, hydrogeology, etc.) in terms of repository safety but which also must be at sufficient depth. Lastly, there is an obvious advantage of using areas without natural resources that might attract mining activities at a later date.

10.3.1 FSR 111.2.f Criteria

The principal characteristics required by FSR 111.2.f for a disposal site for waste containing long-lived and high-level radionuclides are summarized in the following paragraphs.

Stability

"Site stability should be such that any modification to reference conditions by geologic occurrences (glaciation, seismicity, neo-tectonic shifts) must be acceptable from a repository safety perspective. In particular, stability must be demonstrated for a period of at least 10,000 years, which encompasses limited and predictable evolution. For each selected site and based on current conditions, these occurrences are to be assessed in qualitative and quantitative terms for the recent past (historical) and especially for the more distant past (Quaternary and possibly end of the Tertiary) so that the parameters characterizing these factors as well as their variations can be quantified and their influence determined. To accomplish this, it will generally be necessary to investigate the regional geologic environment of each site."

Hydrogeology

"The hydrogeology of the site must be characterized by very low permeability in the host formation and a low hydraulic gradient. Moreover, preference will be given to a low regional hydraulic gradient in the formations surrounding the host formation. Hydrogeologic measurements are to be performed in a much larger area than the repository site so as to construct flow models that factor in flows from the source to the discharge areas. The intensity and direction of underground flows can be simulated using these regional data. Discontinuities or heterogeneities which could significantly lessen the efficiency of the geologic barrier due to their type and geometry must be taken into account, and must be mapped and characterized with the greatest care so as to avoid them at the site, if necessary."

The FSR also specifies the following four criteria as being important for site assessment.

Mechanical and Thermal Characteristics

"Repository feasibility is conditioned on [mechanical and thermal characteristics], i.e., the ability to design a repository that does not significantly alter the geologic harrier. The selected repository medium must also allow for design of disposal pits that do not require access to adjust tolerances during filling operations. Research is to be performed, especially with coupled modeling of thermal and mechanical phenomena, on the influence of waste placement modes and sequences on mechanical effects in the repository, and particularly the amount of preliminary cooling and the density of waste disposal containers. This research will make it possible to determine the corresponding physical parameters and to identify the influence of these phenomena."

Geochemical Characteristics

"[Geochemical characteristics] play an important part in the long-term safety of a radioactive waste repository because they can have an effect on the alterability of man-made barriers, and they govern retention retention phenomena for radionuclides that may have been released. A quantitative description of the geochemical characteristics of the system is to be established to provide for an analysis of radionuclide transport conditions. Mineralogical analyses of the materials of the host formation are to be performed, and their geochemical evolution modeled as a function of temperature and irradiation. The role of clay minerals in particular will be studied."

Minimum Depth

"The selected site must be such that the projected repository depth guarantees that the containment performance of the geologic barrier is not significantly affect-

ed by erosion (particularly after glaciation), by a seismic event, or by 'normal' intrusion. The surface area that could be disturbed in this manner is to be assumed to be approximately 150 to 200 meters."

Depletion of Underground Resources

"With regard to underground resource management, the site is to be selected in a manner that avoids areas with a high value, whether known or suspected."

Obviously, requirements for selection of the geologic medium for the repository site are not unrelated to initial site suitability characteristics. Accordingly, the FSR specifies the following:

"The location of the repository site in the geologic formation must be:

- *in a host block devoid of large faults likely to constitute preferential sectors for hydraulic flows in a the case of crystalline media, with disposal modules to be built away from typical fracturing, although access structures could penetrate the latter; and*
- *in a medium devoid of large heterogeneities and at an adequate distance form surrounding aquifers in the case of sedimentary rock."*

10.3.2 Characterization Methodology

To supplement these rather general considerations, the FSR provides the equivalent of a scope of work for the type of investigations to be conducted, and sometimes for the methods to be employed, to characterize a site in terms of the criteria identified above. The impacts of media-specific particularities on the generic workscope are specified, as in the case of hydrogeological studies.

Crystalline Site

"For deep hydrogeology, and particularly for water transport times and discharge identification, studies are to be performed on fracturing on a variety of scales (low fracturing, hectometric fracturing, large faults bordering the host block) and on all other elements necessary for modeling."

Salt-formation Site

"For surface and lateral hydrogeology, detailed analysis of the hydrologic balance of each catchment basin is to be performed to estimate surface aquifer supply. For all aquifers, a regional hydrogeologic diagram is to be prepared showing supply areas, discontinuities, discharge areas, and interactions between aquifers as well as a hydrogeologic balance. A local hydrogeologic study is to be performed showing the geometric characteristics of the aquifers (lithostratigraphic type, morphology, continuity, etc.) and of the impermeable layers and their hydrodynamic characteristics (permeability, transmissiveness, porosity, etc.), taking into account the influence of host rock fracturing in particular and any other element necessary to quantify flows, such as local pumping. These hydrogeologic assessments are to make it possible to predict the probabilities of dissolution."

Clay-formation Site

"Surface hydrogeology is to be described at the local level to estimate surface aquifer supply. The following elements are to be determined as precisely as possible for all formations:

- *a regional hydrogeologic diagram showing source/depletion areas and the interaction between aquifers as well as a preliminary hydrogeologic balance;*
- *a local hydrogeologic diagram showing:*
  - *the geometric characteristics of the aquifers (lithostratigraphic type morphology, continuity, etc.) and of the semipermeable and impermeable levels;*
  - *their vertical and horizontal hydrodynamic characteristics (porosity, permeability, [transmissiveness, etc.), taking into account host rock fracturing in particular and any other element necessary to quantify flows;*
  - *their geochemical characteristics, particularly salinity; and*
  - *their hydrodynamic parameters and the geometry of any vertical discontinuities which could result in interactions among different stratigraphic levels."*

10.4 REPOSITORY DESIGN

The fundamental objective of the deep geologic waste repository is to protect members of the public and the environment now and in the future (FSR III.2.f).

Decree 92-1391 of December 30, 1992, concerning Andra requires that Andra submit a summary report to its oversight ministries no late than December 31, 2005 on the results of research, accompanied as appropriate

by a repository design. The law clearly stipulates that the decision to create a repository is subject to numerous conditions. Foremost among these is the review of the various waste management research programs by the National Assessment Commission.

10.4.1 Performance Assessment

What tools will be used for performance assessment and how will Andra apply them to the development of its design concept?

Direct assessment of the effectiveness of the various barriers in a repository for Category B waste isn't possible due to the length of time required for containment. Long-term performance assessment depends on several scientific disciplines and follows an approach that is naturalistic, experimental and model-based to understand phenomena brought into play by the repository.

The naturalistic approach is quantitative and historical; it compares the various geologic situations observed to a historical and experimental understanding of the medium. In particular, this approach provides information on the past evolution of sites (climate, neotectonics) over time scales consistent with the radioactive decay periods of the radionuclides in the repository in order to predict future behavior.

The experimental approach provides access to host formation behavior at various locations so that: properties can be measured or at least assessed in actual conditions, disturbances to the host rock caused by the repository can be identified and ranked, behavior models can be validated and the repository concept can be adapted to the reference medium.

Modeling is a means of summarizing data from a variety of fields to understand the effects of thermo-hydromechanical and chemical coupling; it is used to perform sensitivity analyses and simulations.

All of these approaches are used simultaneously rather than sequentially, and all results are factored into the performance assessment.

As the last step in the process, FSR 111.2.f also requires that changes in the behavior of the repository be monitored over time: *"Given the period of time involved in repository operations and the disturbances caused during that period, specialized instrumentation is necessary to monitor changes in site and repository structural parameters. Said instrumentation is to be set up as soon as possible to ensure that the repository structure and the site are monitored not just during repository operations, but before them as well. In particular, the following should be monitored:*

- *site piezometry;*
- *deformations and more generally behavior over time of the walls of the repository that are to remain open for very long periods of time (certain reconnaissance bore holes, access shafts, service galleries);*
- *seismic movements; and*
- *thermal behavior of the medium and its effects (constraints, displacements, fracturing, etc.)."*

10.4.2 Design Concept

Having presented the regulatory and legislative context in which Andra performs research on high-level and long-lived waste disposal, the repository concept and Andra's approach to the feasibility study of the repository will be explained. Andra must first:

- identify the principal functions of the repository, which must contain the radionuclides in the waste packages, i.e., minimize and retard their potential release and migration to protect the environment and members of the public now and in the future; and
- examine the potential for waste package recovery during the period of retrievability.

To meet these requirements, the repository concept includes several elements that respond to specific objectives, particularly:

- an underground facility layout that can be adapted to the conditions likely to be encountered in deep geologic formations;
- waste placement systems; and
- radionuclide containment systems including solidified waste and geologic formations with complementary artificial barriers.

The technical feasibility of these functions requires assessment to verify that it will be possible to construct, operate and close the facilities in accordance with the requirements identified earlier while responding to the following questions:

- What constraints must be placed on the facilities due to their depth and the high temperatures generated by certain high-level wastes?

- Are there technological answers that are readily available to industry?
- Given the current limitations in knowledge and technology, what developments are necessary in the fields of mining engineering, underground handling of radioactive materials, and construction of man-made barriers?

The constraints applicable to Category C waste are different from those of B waste, which would appear to translate into specialized features for the repository. In addition to justifying the proposed technical solutions, feasibility studies are needed to identify potential requirements for additional research and development for the solutions under investigation.

10.5 UNDERGROUND RESEARCH LABORATORIES

The Waste Act designates underground research laboratories as one means of investigating the potential for retrievable or non-retrievable disposal in deep geologic formations. Andra's research objectives for these laboratories are as follows:

- perform in situ rock or fluid measurements while disturbing these materials as little as possible to understand the parameters already partially assessed during the surface reconnaissance program;
- conduct more general experiments to determine the behavior of the various rocks and fluids, taking into consideration natural phenomena and modifications caused by the construction of a potential repository as well as by the presence of waste packages;
- investigate the medium, particularly its spatial variability, to assess site suitability and the possible location of galleries and future repository excavations; and
- determine the data needed to design excavation, backfilling and closure of the disposal sites.

It should be noted that a large number of lithologic, structural, petrographic, hydrogeologic, thermomechanic and tectonic characteristics are already available at the surface, which make it possible to analyze them in a regional context and conduct a preliminary assessment of the suitability of the site to host a repository. This preliminary assessment will be expanded and supplement ed by investigations in the underground laboratories. Surface and underground work can be conducted in parallel rather than sequentially.

The Waste Act and its implementing Decree 9340 of July 16, 1993, specify the conditions under which laboratory construction and operation will be licensed. The sheer size of the laboratories makes them true industrial projects. There will be complete openness in the methodology used, as set forth in FSR III.2.f, which identifies essential and important criteria for site characterization and specifies general requirements for site investigations in Appendix 1.

The primary purpose of the measurements and tests to be conducted on site and in the laboratory is to confirm the initial assessment of the site's qualities and drawbacks and the overall adequacy of the selected location. In addition to initial measurements and tests, phenomenological studies and tests in the underground laboratories, along with research of a more fundamental nature in conventional laboratories, will help Andra to assess the site behavior in more depth and detail that results from the disturbances to which it has been exposed during construction and disposal.

It should be noted that validation of the complex approach described above involves an experimental period that cannot be cut short, followed by interpretation of test results, which translates into a rather tight schedule.

10.6 ANDRA RESEARCH PROGRAMS

10.6.1 Research Budget and Participants

Andra has embarked on a vast research program in furtherance of its missions, as reflected in this report and in the large budget and numerous contractors reporting to Andra. The 1993 deep disposal research budget was FF 250 million francs for planning activities alone.

The size of the subcontractors varies widely; some research is so specialized that sometimes only a university laboratory or a single engineering company can respond. On the other hand, it is sometimes necessary to turn to large groups such as BRGM [French Geological Survey], CEA [Atomic Energy Commission], EDF, MDPA, Bertin, Cogema, and others. Andra controls the research objectives, of course, but may also select the laboratory within these organizations that is best suited for the work requested.

The Agency's research and development programs are scrutinized by its Scientific Council, who:

- issues, opinions and recommendations on scientific

and technical objectives and on costs,
- is kept informed on progress, and
- assesses the results of these programs.

The opinions, recommendations and report of the Scientific Counsel are submitted to Andra's Board of Directors. Andra's research results for the year are set forth in its annual report on research and development, presentations by Andra and its suppliers at international conferences, and articles by Andra and its suppliers in the trade press. Whenever work funded by Andra needs to be protected legally for commercial use, patents are registered and other forms of protection are adopted.

Chapter 11

Geoscientific and Rock Mechanical Activities for the Radioactive Waste Repositories in Germany: Key Issues, Status and Future Plans

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Abstract. The engineering geological and rock mechanics activities for radioactive waste repositories are strongly linked to the safety concepts for the repository projects. Key issues are: proper site characterization, proof of geotechnical safety and quantitative description of geochemical processes and of geological scenarios for long-term safety assessment calculations. A status report of the German repository projects Gorleben and Konrad based on this concept will be presented. While these projects are still being pursued the former Morsleben salt mine in Sachsen-Anhalt, which was used as a repository for short-lived low and intermediate level waste from 1981 until 1991 has resumed its operation in 1994. In the paper some issues are demonstrated by an example: site characterisation work for a salt dome, geotechnical measurements and calculations to proof the stability of a mine, and considerations of geomechmical natural analog for calibration of constitutive laws. For the future the role and contributions of geoscientific and rock mechanics work within the safety assessment issues (e.g. geomechanical safety indicators) must be identified in greater detail.

11.1 Introduction

Since the early sixties, the radioactive waste disposal policy in the Federal Republic of Germany has been based on the decision that all kinds of radioactive waste are to be disposed of in deep geological formations. The basic aspects which must be taken into account to achieve this objective of disposal are compiled in the "RSK-Sicherheitskriterien für die Endlagerung radioaktiver Abfälle" (Safety Criteria for the Disposal of Radioactive Waste in a Mine). The following criteria are considered to be the most important ones:

The required safety of a repository constructed in a geological formation 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 intended to be disposed of.

In the post-closure phase, the radionuclides which might reach the biosphere via the water path as a result of transport processes not completely excludable must not lead to individual dose rates which exceed the limiting values specified in section 45 of the German Radiation Protection Ordinance (0.3 mSv/a concept).

11.2 Geotechnical Safety

Natural geological and geotechnical barriers are an important part of a multiple-barrier system. Thus, the loadbearing capacity of the rock (expressed, for example, through subsidence or cavern stability), its geological and tectonic stability (e.g. mass movement or earthquakes), and its geochemical and hydrogeological development (e.g. groundwater movement and the potential for dissolution of the rock) are important aspects of the safety analysis. Therefore safety cannot be assessed from a purely engineering point of view, but must include geological factors. A site-specific modelling of geotechnical and geomechanical features and processes is needed.

The safety analysis must be based on a safety concept that includes the possibilities of failure (failure scenarios) that could occur during the excavation, operation and post operation phases, as well as measures to avoid such failures. Monitoring is also a part of the safety con-

cept.

11.2.1 Geotechnical safety plan

The site specific geotechnical safety plan has to include the individual dose/risk scenarios and possible contingencies for which again measures and/or verifications are required. The safety plan has to be updated as new experience becomes available (e.g. during the construction or during the operation of the plant) The safety plan for an underground disposal plant (disposal mine) should, inter alia, describe the following (Langer et at 1993):

- measures to avoid or reduce risks;
- possible actions to enhance stability of the plant-based on monitoring systems; and
- acceptable residual risks.

The potential risks define the limiting situations which are to be avoided. These could be for example:

- local fractures in mine openings;
- failure of pillars and roofs;
- rock burst;
- rock mass loosening due to large cavity or shaft convergence, leading to a loss of integrity of the rock mass; and
- loss of functionality of seal structures(e.g. dams).

The safety plan should assess the following:

- system failure of load bearing structures;
- long term integrity of geological barriers; and
- seal functions of seal structures.

To this end the following measurements and/or calculations are available:

- the short term and long term convergence of cavities;
- large scale deformations in rock mass and neighbouring rock and overburden; and
- stress states in rock mass.

11.2.2 Numerical proof of geotechnical safety

The potential risks as described in section 11.2.1 which generally represents states of a hypothetical character and are hence not covered by the in situ measurements are to be checked mathematically. Relevant models have to be developed for each situation, and are to comprise the following component sections:

- presentation of the dose/risk scenario under investigation;
- effects such as primary state of rock mass, temperature, cavity convergence, effects of waste, earthquake, etc;
- calculation model which must cover rock mass formations, cavities and their changes as realistically as possible;
- material models for rock mass and overburden, and possibly also waste and backfill;
- calculation of safety relevant status variables such as deformations, stresses, and possibly permeability of liquids (and gases);
- checking and assessing calculation results; and
- safety concept which provides statements as to threshold values for risks.

The informative value of the theoretical proofs is decisively influenced by the expressions introduced into the calculation model. For this reason it is necessary to be aware of these influences when assessing safety matters. These include, inter alia:

- show sources of errors e.g. in the structural description of the disposal facility, in the material laws, in the numerical calculations methods; and
- show the sensitivity of the results to changes in the input parameters by calculating with parameters variations (e.g. also to identify natural scatter).

In the practical performance of the proofs it may be useful to initially work with assumptions which are unequivocally conservative which produce simple verification methods and then move on to more complex expressions in places where the conservative calculations indicate possible critical states. When using this procedure of simplified verification the conservativeness of the expressions must be thoroughly evaluated.

Uncertainties are for example:

- variation of material properties with respect to space and time;
- uncertainties in the determination of the load;
- inexactness of the model (simulation of the physical and geological conditions); and
- efforts of unexpected and/or possibly omitted events.

Geological and geotechnical uncertainties can be mastered by a method of calculated risk ("Geoengineering

Confidence Building", see Fig. 11.1). The main part of this method is the handling and validation of models (Langer, 1994).

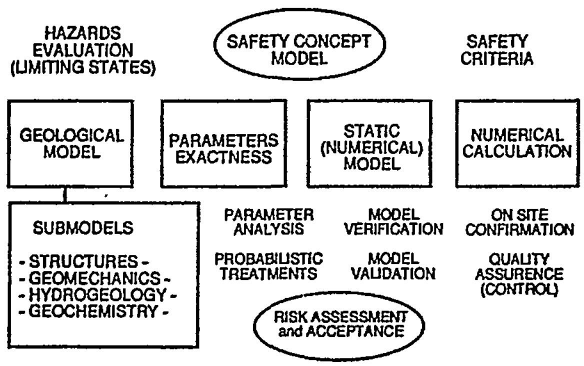

Figure 11.1. Geo-engineering safety confidence building.

11.3 GORLEBEN REPOSITORY PROJECT

11.3.1 Site investigation and planning

The Gorleben salt dome has been under scrutiny since 1979 to host a repository at depths between 840 and 1200 m, mainly for high level and/or alpha bearing wastes and spent fuel. Site investigations and planning are at present based on a nuclear capacity of 2500 GWa, leading to a total activity of about 10^{21} Bq and an alpha activity of about 10^{19} Bq (Brennecke et al., 1994). The following work from above ground was carried out between 1979 and 1985 to investigate the geology and hydrogeology of the Gorleben site:

- 4 boreholes, each about 2000 m deep, for investigation of the salt dome;
- 44 boreholes for investigation of the cap rock and the underlying salt beds;
- 2 preliminary boreholes for the shafts Gorleben I and Gorleben 2;
- 156 km of seismic profiles;
- 145 investigation drillings into the Cenozoic cover;
- 326 drillings for the installation of piezometers;
- 4 long time pumping tests (pumping time about 3 weeks for each test); and
- 1 borehole for investigations of the Palaeogene in the rim syncline.

Other investigations which were carried out include geoelectrical and geothermal studies, gravimetry, seismology, geochemistry, isotope geochemistry, and micropalaeontology.

The objective of the exploration from underground is to acquire all information needed to evaluate the operational and long-term safety of the planned repository. This has to be achieved while keeping disadvantages of potential damage to the geological barriers as low as reasonably achievable. Two shafts have been erected. Subsequently two pairs of exploratory drifts, connected by eight cross-cuts, will be driven to the northeast and southwest in a depths of 840 m. From there, numerous exploratory wells (in total about 60 km) will be drilled horizontally and vertically up to about the outer boundary of the prospective repository fields.

The stratigraphy and structure of the salt dome will be investigated by geological, geophysical and petrographic methods in such detail that it will be possible to identify rock salt sufficiently large and otherwise suitable for the different types of radioactive waste. Also, the positions of the more problematic layers, such as the main anhydrite and the Stassfurt potash seam, as well as brine pockets and gas-bearing salt bodies, will have to be determined exactly for proper risk assessment.

11.3.2 Results

In agreement with the international approach (IAEA, 1994), remarkable results were recently achieved in the use of safety indicators for the barrier performance of rock salt in general, and of the Gorleben salt dome in particular:

- An empirically tested exponential mass change model predicts isolation potential of repositories in rock salt of millions of years. The results from the site investigations in Gorleben are in agreement with these predictions (Röthemeyer, 1991).
- This evidence is supported by site-specific natural analogues. The analysis of fluid inclusions gave evidence of the depth down to which the glacially influenced subrosion processes effected the salt dome, and proved a past isolation period of 2.5 x 10^8 years at the depths of 860 to 1360 m envisaged for waste disposal.
- Regarding the halokinetic process of the Gorleben salt dome, the average rate of uprise of salt in the top part of the salt diapir has been calculated. The diagram (Fig. 11.2) shows a rapid increase in the rate of uprise from the beginning of piercement to a culmi-

nation in the Cretaceous, and a gradual decrease in the rate up to the present day. Jaritz (1991) plotted curves of the variation in the rate of uprise with time using Zirngast's figures. He obtained values of 0.07 to 0.08 mm/a (Fig. 11.2) for the maximum rate of uprise of the salt, using different time scales. This is slightly lower than the other salt diapirs in NW Germany, for which Jaritz gives values between 0.1 and 0.5 mm/a. Both curves agree as to the present rate of uprise (0.01 mm/a). In the light of the more or less constant dynamic conditions, on which the supply of fresh salt from below depends, it is possible to predict that in the future the rate of salt uprise will not change significantly for at least one million years. Thus, assuming a rate of uprise of about 0.01 mm/a, uprise over the next 10,000 years will amount to 10 cm and over the next 1 million years, about 10 m.

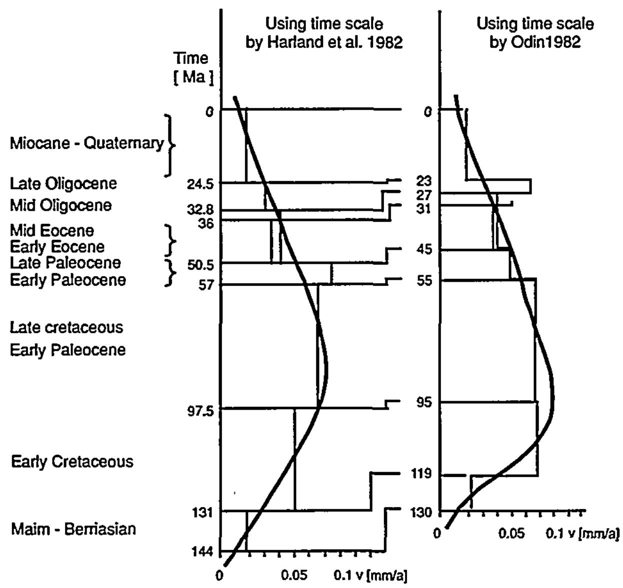

Figure 11.2. Mean rate (v) of uprise of the salt at the top of the salt diapir, after ZIRNGAST in JARITZ, 1991.

- The effect of future geological processes have been considered with probalistic modelling of structural response. As an example, impact of a temperature decrease during glacial age on the integrity of the salt barrier above a repository has been analysed. A sensitivity study showed that besides the depth of the salt dome, also the shape of the dome and the stiffness of the adjacent rock have a significant influence on the development of tensile stress at the top of the salt dome. However, creep capacity of rock salt enables the salt formation to significantly reduce the total amount of tensile stress. Even under very conservative assumptions, thermally induced fractures will only develop as far as about 100 metres depth, and therefore will not endanger the barrier function. With respect to the computational procedure, it has to be mentioned that the linear approach to the system response surface is valid for the case in question but can be insufficient for other evaluations. A nonlinear approach would be recommended. Respective development is under progress (Wallner and Eickemeier, 1994).

- Regarding the rock mechanical properties of the rock salt from Gorleben dome, the creep behaviour has especially been investigated. The application of improved experimental techniques, as well as examination of our extensive data base, show that the specific textural characteristics of the various types of rock salt cause significant differences in creep behaviour. In spite of the same experimental stress and temperature conditions, samples yield steady state creep rates that vary from each other by a factor of more than 10. Differences of up to a factor of 100 have even been recorded from rather pure rock salt (greater than 95% halite). Figure 11.3 illustrates a summary of the experimental results for steady state creep rates at room temperature, which were obtained on rock salt samples from the Gorleben salt dome. These suggest that there is a large range of creep characteristics. The "locally" large differences (factor of 100) are rarely taken into consideration during modelling calculations, and consequently the results are not particulary reliable. The results shown in Figure 11.3 also confirm the "miners law" in that the "older halite" (z2) on average creeps faster than the "younger halite" (z3). Two formulae for steady state creep (BGR2 and BGRb) have been determined. In Figure 11.3 the BGRa law is located in the upper domain for the creep rates, and the BGRb law in the lower domain. Both formulae yield a stress exponent n= 5 for the stress sensitivity. Miroscopic impurities are the main reason for the large differences in creep. It is known from material science that it is not the overall mass-proportion of impurities but the number, distribution and type of particles that affects the ductility. This is easy to comprehend, since every defect in the crystal lattice is an obstacle to the moving dislocations (Hunsche and Schulze 1994).

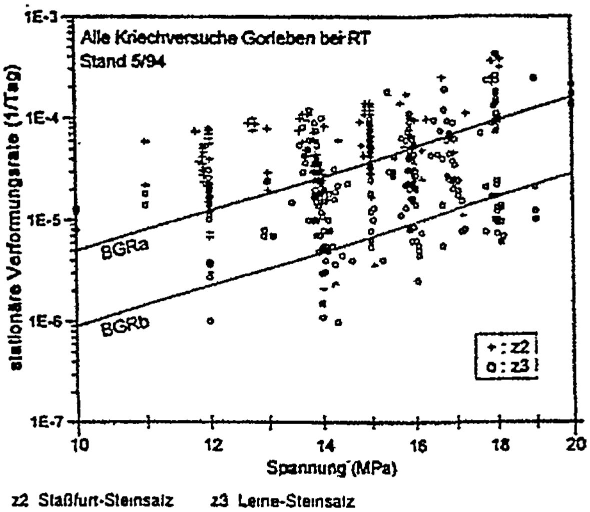

Figure 11.3. Stationary creep rates of salt cores from Gorleben.

11.4 KONRAD REPOSITORY PROJECT

11.4.1 Safety assessment and planning

It is intended to emplace into the Konrad repository - an abandoned iron ore mine in the southeast of Niedersachsen - radioactive wastes which exert a negligible thermal influence on the host rock (total activity 5 x 10^{18} Bq, alpha activity 1.5 x 10^{17} Bq). The application for the initiation of a licensing procedure for the Konrad repository project was filed in 1982. A decision on the project is expected for the near future.

New cavities of about 10^6 m^3 volume will be driven at depths between 800 and 1300 m for the emplacement of the waste. A time of operation of at least about 40 years is expected, depending on the amount of waste to be disposed of annually.

Because of the licensing autbority's opinion on long-term safety, limitations will arise to 7.0 x 10^{11} Bq for ^{129}I and 1.9 x 10^{12} Bq for ^{238}U. Due to the long travel times from the repository to the biosphere, a potential radiation exposure of the biosphere will only result in the case of long-lived radionuclides like ^{129}I and ^{238}U with its decay products, and only after hundreds of thousands of years and several million years, respectively. These results are taken from model calculations for radionuclide migration in the repository's near and far field. Since the potential radiation exposure will occur far beyond a time limit for which it may be assumed with sufficient reliability that the marginal conditions like the present geological and hydrogeological situation at the site underlaying the calculations are still valid, the results obtained for individual doses only have the purpose to judge the isolation potential of the site. This opinion of the applicant was discussed controversially within the licensing process. More details were presented by Röthemeyer (1993).

11.4.2 Rock mechanics work

Among others, the following types of measurements have been carried out:

- measurement of subsidence at ground level using leveling;
- measurement of the drop in roof level in the main levels;
- measurement of the convergence in the main levels and exploratory drifts;
- measurement of rock mass deformation above a former working field (flushing field); and
- measurement of rock stresses.

Subsidence was measured using precision levelling on a fixed network of 390 points distributed over approximately 40 km^2. The results of these measurements may be summarized as follows:

- The first subsidence occurred approximately one year after the start of mining.
- The maximum subsidence of the trough is in the area above the flushing field. In May 1985, 20 years after the start of mining, the subsidence here totaled 264 mm. The subsidence trough had a limiting angle of between 35 and 39°.
- During ore extraction the maximum rate of subsidence over the southern field was 2.8 mm/month; in 1985 this had reduced to only 3 mm/year.

The development over time of the subsidence of levelling points does not indicate any irregulatrities. The trough is subsiding uniformly.

Overall, the subsidence may be considered as small. This is confirmed by the fact that no damage has been observed at the surface. The measurement results allow the assumption to be made that subsidence is close to its final value. A large number of convergence measure-

ment stations were set up on the main levels and exploratory drifts. It proved in part possible to carry out initial measurements very soon after cavity excavation.

The evaluation of the convergence measurements has provided the following results:

1. The convergence processes observed in the Konrad mine are not based on creep processes, as in rock salt, for example. They are rather related to microfissuring processes in the nearby surrounding rock, which lead to deconsolidation of the rock mass.
2. The convergence of the drifts over time may be described using a logarithmic function (Fig. 11.4).
3. The convergence behaviour of the drifts is dependent upon the degree of excavation; the higher the degree of excavation the higher the convergence.
4. The floors of the drifts frequently show greater convergence than the roofs.
5. In drifts which pass through old mine sections the effects of stress redistribution in the area of the old workings are continuing.

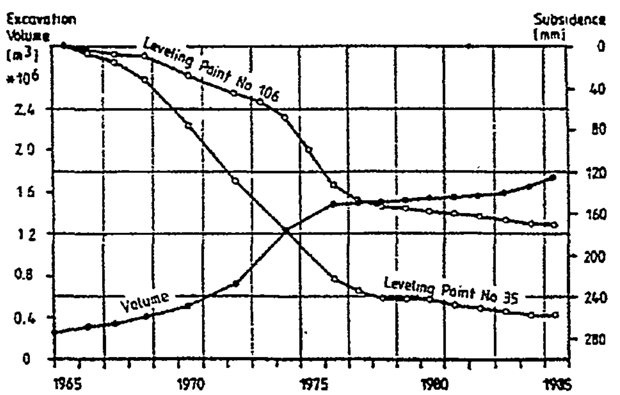

Figure 11.4. Subsidence and excavation volume versus time, Konrad mine.

In addition, numerical calculations were executed to assess the rock mechanical processes in the overburden and the stability of the repository. The calculations were carried out using the Finite Flement Program System ADINA.

Primary stress measurements were carried out in two boreholes at borehole depths of approx. 15 m to 25 m. The results are plotted in Figure 11.5. The maximum and minimum stresses are shown as functions of depth of the borehole under study. The measurements allow maximum values of between approx. -18.0 and -24.5 MPa to be read, and minimum values of between approx. -12.0 and -17.6 MPa. The overall averages are -21.0 MPa (maximum values) and -14.8 MPa (minimum values). The maximum stresses are generally subvertical (±20° to ±30°) in orientation.

Typical results measured for secondary stresses normal to plane of bedding me given in Figure 11.6 for measurement section MQ4 in the pillar between chamber 231S and chamber 241S. Here the measured stress changes were raised by the amount of the theoretical primary stress. The measured values show a marked drop as distance from wall increases. At approx. 10 m distance stress changes have fallen to minor levels.

As a comparison the results of the numerical model calculations are shown. These model calculations are reported in detail in Dickmmn et al. (1991). As Figure 11.6 shows, there is good agreement between measured and theoretical secondary stresses.

Generally, the rock mechanics tests undertaken at the Konrad drift system in field 5/1 have as their most important result that the construction of storage chambers in middle Jurassic coal oolite for radioactive wastes generating low levels of heat with cross sectional areas of 40 m^2 and chamber separations of approx. 35 m can be safely carried out from a rock mechanical point of view. The results of the geotechnical measurements performed in test field 5/1 do not indicate my risk to stability of the hard rock and the drifts.

With respect to the test techniques under the difficult *in-situ* conditions in the underground, the instrumentation proved itself for the most part very well and produced reliable results. The test concept of overlapping and complementary test methods has proved itself to be constructive and expeditious.

11.5 MORSLEBEN REPOSITORY

Short-lived low and intermediate level radioactive waste from the operation of nuclear power plants and the application of radionuclides in research, medicine and industry in the former Geman Democratic Republic was disposed of in the Endlagerung für radioaktiver Abfälle Morsleben (ERAM, Morsleben repository for radioactive waste) an abandoned salt mine located near the village of Morsleben (now in the state of Sachsen-Anhalt).

Until 1969, this mine had produced potash and rock salt. The salt was excavated in room and pillar mining down to a depth of about 500 m so that mine openings result-

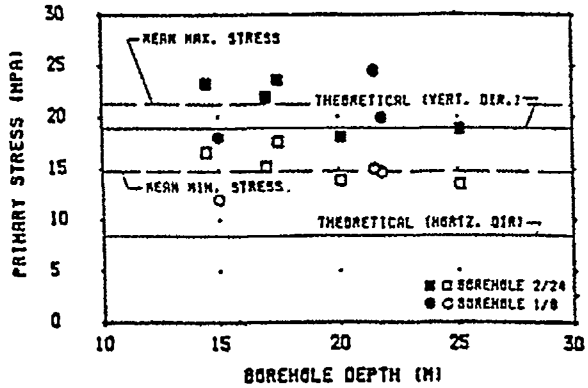

Figure 11.5. Primary stresses in test field 5/1.

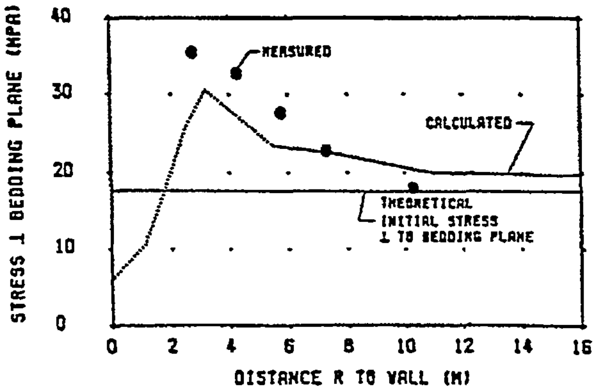

Figure 11.6. Secondary stresses normal to plane of bedding.

ed with a maximum length of about 150 m and a maximum width and height of about 30 m, respectively. In total, a volume of about 7.6 x 10^6 m^3 was excavated and partly backfilled (2.1 x 10^6 m^3,). Since the first operational license was granted in 1981, the repository has been used for the emplacement of short-lived low and intermediate level waste without the intention of retrieval. Spent radiation sources were also disposed of in this facility.

Until 1991, radioactive waste with a total emplacement volume of approximately 14,500 m^3 and about 6,200 radiation sources were disposed of. In total, an activity of 1.8 x 10^{14} Bq was emplaced. The activity of alpha emitters amount to 1.6 x 10^{11} Bq, and the activity of beta/gamma emitters, to 1.8 x 10^{14} Bq. The waste was mainly delivered by combined rail-and-road transports using standardized freight containers. About 2,900 such containers were delivered until 1991.

Since the German Unity in October 1990, the Morsleben facility has the status of a federal repository. The operating license is limited by law until June 30, 2000. Emplacement of waste has been resumed in January 1994. Until the end of the year 1995, 5,691 m^3 of waste with the total activity of 2.6 x 10^{13} Bq [(α: 1.4 x 10^{10} Bq] has been disposed of.

11.6 Future Research Work

The European Commission has recently performed a review study on the status of understanding of thermal, mechanical, hydrogeological and geochemical properties of host rock formations (Balz, 1995). In general, it was concluded that the large number of heating experiments which have been performed, both in laboratory and in the field, have provided sufficient knowledge about the nature of the purely thermal aspects and existing models seem to be adequate for predicting the resulting temperature field and its evolution with time.

It becomes more difficult when one has to consider the influence of temperature increase on the mechanical, hydrogeological and geochemical properties and in particular to understand and perform predictive modelling of coupled effects.

Therfore, it has also been recognised that the couplings

Table 11.1. Recommendations for research

Priority 1
• Further development of investigation and interpretation methods for coupled geochemistry/geomechanical/brine migration models.
• Further studies of geomechanical properties with gas and brine related parameters as variables.
• Development of consistent thermomechanical models and relevant codes for backfill material.
• Scientifically based engineering studies to develop performance criteria for backfilling and sealing techniques.
• *In-situ* studies of the gas problem.
Priority 2
• Comparison of stress measurements methods and interpretation procedures, taking salt creep into account.
• Development of a freshwater/brine groundwater flow model, and validation under *in-situ* conditions.
• Development of models for generation and transport of gases and brines.
• Optimization of "geoprospective" natural analogue, scenarios and palaeohydrological methods.
• Testing of backfilling techniques in a URL.

of chemistry with thermo-hydromechanical (THM) processes of the various host rocks is a crucial issue which has to be further investigated.

Another subject which has gained attention in recent years is the issue of gas generation and transport and possible impact on operational and long term safety. In any repository for radioactive wastes, gases will be formed due to corrosion of metals, microbial degradation of organic matter and to a lesser extent from radiolytic decomposition of water and organic compounds. As a consequence, gas pressure will build up in the near field until it is released through the system of engineered barriers into the geosphere at a rate equivalent to the production rate. Research efforts have been undertaken to assess the rate of production for various waste forms, disposed of in various host rocks. Moreover, models have been developed for describing gas transport and/or two phase flow through the geologic formations. However, more research is needed for a better understanding of basic mechanisms of gas and/or two phase flow through the host rock or along preferential pathways, fractures and faults. Site specific data are necessary for the assessment of the possible impact of gas generation on repository safety. Technical measures are available or further being developed to cope with these issues by appropriate repository design.

Reflecting these results and considering the following key questions:

- can a candidate respository site be adequately characterised (availability of site characterisation techniques and methodology)?
- are thermal, hydrogeological and mechanical properties and processes of salt formations well enough understood (long-term efficiency of the geological barrier)?
- will it be possible to build, operate, backfill and seal a repository in salt rock in a safe and economic way (repository design aspects)?

Conclusions for further scientific work have been given (Langer, 1995). Regarding rock salt as host rock the recommendations listed in Table 11.1 can be given. The aim of this research work is to assure that appropriate use of safety assessment methods, coupled with sufficient information from proposed disposal sites, can provide the technical basis to decide whether specific disposal systems would offer to society a satisfactory level of safety for both current and future generations.

REFERENCES

Langer, M., et al., Empfehlungen des Arbeitskreises Salzmechanik zur Geotechnik der Untertage-deponierung, Bautechnik 70: 734-744, 1993.

Langer, M., Geoscientific evaluation of geological and geotechnical barriers with respect to waste disposal projects, Proc. 7th Int. IAEG Congr. 2827-2935, Rotterdam, Balkema, 1994.

Brennecke, P., et al., Final disposal in Germany, Kerntechnik 59: 23-27, 1994.

IAEA, Safety Indicators on different time frames for the safety assessment Tec. Doc. 767, Vienna, 1994.

Röthemeyer, H., ed., Endlagerung radioaktiver Abfälle, VCH Verlag Weinheim, 1991.

Jaritz, W., Die Entwicklungsgeschichte des Salzstocks Gorleben, Geol. Jb A 132: 3-31, 1991.

Wallner, M. and R. Eickemeier, Modelling of consequences of future geological processes by means of probabilistic methods, NEA-Workshop on Site Evaluation (SEDE), Paris, 1994.

Hunsche, U. and O. Schulze, Das Kriechverhalten von Steinsalz, Kali u. Steinsalz 11: 238-255, 1994.

Röthemeyer, H., Aspects of a confidence building process in long term safety evaluation for HWL-repositories, Proc. Int. Conf. Safe Waste, Avignon, p. 290, 1993.

Diekmann, et al., Geotechnical measurements during excavation of parallel drifts, 3rd Int. Symp. Field Measurements, Oslo: 561-570, 1991.

Balz, W., Research and development on radioactive waste management, In: IAEA Tec. Doc. 853, 72-87, Vienna, 1995.

Langer, M., Review of R&D requirement on THM properties of salt, In: CEC Proc. Workshop, Testing and Modelling of thermal, mechanical and hydrogeological properties of host rocks, 223-250, Brussels, 1995.

CHAPTER 12

HUNGARIAN APPROACH FOR FINAL DISPOSAL OF HIGH LEVEL RADIOACTIVE WASTE

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12.1 INTRODUCTION

In Hungary, nuclear power provides a substantial portion of the total electricity produced in the country. The Paks Nuclear Power Plant (NPP) contains four VVER-440 nuclear reactors, each capable of producing 460 MW of electricity. This plant regularly produces more than 40% of the electricity consumed in Hungary. There is currently no plan to increase the electrical generating capacity of Hungary, but if future additions to nuclear-based electrical generation were to be made, they would significantly increase nuclear power's share of electricity consumed.

Fuel for the Hungarian NPP, as is the case for all other East European VVERs, has been supplied in the past by the ex-Soviet Union. As part of the fuel agreement, the Soviet Union was obliged to take back and dispose of all spent fuel (using another supplier would leave Hungary with the problem of high level waste disposal). In the past, Hungary was glad to have the problem of disposal dealt with in this way. Now, although contracts for fuel supply and return are still in force, Hungary is less certain how long this "comfortable arrangement" will last.

The shipment of spent fuel for 1992 was ready for dispatch but has been delayed due to problems with transport through the Ukraine. It is understood that Russia, while they may still be prepared to take and process spent fuel, will now send the high level waste back to Hungary for disposal. The cost of such disposal will be charged to the Paks NPP operation and, therefore, future costs are likely to increase.

This likely interruption of the current spent fuel disposal route may lead to a fairly immediate problem in Hungary. The spent fuel ponds will be totally full by the end of refueling in 1995. It is therefore necessary for a new program for removal of fuel from the ponds to be implemented before the 1996 refueling to avoid the necessity to shutdown the reactors.

The current intention is to build an interim storage facility to hold spent fuel after it has completed five years of cooling in the ponds. In 1993, a decision was made to construct a Modular Vault Dry Store at Paks based on the GEC Alsthom design, which was first used at Fort St. Vrain (USA). At some point in the future, the fuel would either be packaged for direct disposal, or reprocessed outside the country and the high level vitrified waste returned to Hungary for disposal.

As insurance against the waste remaining in Hungary or being returned after reprocessing, it is highly desirable to proceed by planning for possible disposal of spent fuel some 50 years or more in the future.

Normally, the siting of any HLW repository should be started by screening the entire country for suitable locations. However, as a result of preliminary investigations, Hungary has a geological formation that is considered at least as a national "treasure." This is a siltstone, an Upper Permian red-colored formation, covering some 150 sq. km. Its thickness from borehole drilling is about 800-900 m.

Based on preliminary assessments and technical considerations, the use of the Permian Boda claystone formation in the Mecsek Mountain area is being considered for HLW disposal. To evaluate the suitability of this for-

mation as a location for a waste repository, investigations have started with the technical assistance of the Atomic Energy of Canada, Ltd.

12.2 Spent Fuel Characteristics

The radioactivity and heat output of a spent Paks NPP fuel assembly is similar to spent fuel assemblies from other pressurized water reactors for the same burnup. However, the geometry and components of these fuel assemblies are different from those of fuel assemblies from other pressurized water reactors. Two types of fuel assemblies are used in the Paks NPP, the operational assembly and the follower part of the absorbers. Both types have a hexagonal cross section with a key dimension across the flat side of the hexagon of 144.2 mm.

Each assembly contains 126 fuel rods in a hexagonal array and surrounded by a hexagonally shaped tube. This tube is fabricated from an alloy containing 97.5% zirconium and 2.5% niobium. The fuel rods consist of UO_2 pellets contained in a tube made from a 99% zirconium and 1% niobium alloy. The inside and outside diameters of the cladding tube are 7.72 mm and 9.15 mm, respectively. The assembly top head, nozzle units and spacers (10 per assembly) are made from stainless steel. The overall length of the operational fuel assembly is 3,217 mm and the active length is 2,470 mm. The total weight of the operational assembly is 215 kg, which includes 120.2 kg of fissile material. The differences between the Paks NPP fuel assembly and a typical pressurized water reactor fuel assembly would not preclude direct disposal of the spent Paks assemblies in a repository.

Assuming a 30-year operational life for each reactor unit and no change in the fuel cycle, 15,316 spent fuel assemblies will be discharged from the Paks NPP.

12.3 Options for Spent Fuel Management

Paks NPP Ltd. had two contracts with the Soviet Union (now Russia): one, on the governmental level, addressing the fresh fuel supply; and the other, a private, low-level contract covering the shipment of spent fuel back to Russia. Furthermore, in April 1994, a new protocol was signed between the governments of Russia and Hungary complementing the earlier governmental contract. The protocol confirmed that the fresh fuel for Paks NPP will be supplied by Russia, and Russia is ready to receive Hungarian spent reactor fuel. In spite of the new protocol, the Hungarian government was not able to ship spent fuel to Russia in 1994. Because of current difficulties and the fact that the protocol does not contain a guarantee that Hungarian spent fuel will be returned to Russia, there is a very low probability that long-term shipment of Paks NPP spent fuel to Russia will be realized. Therefore, it is assumed that in the long term, the return of spent fuel to Russia will not be possible.

The Hungarian government and Paks NPP Ltd. had to consider developing a program to manage the spent fuel assemblies from the existing Paks NPP and from any future nuclear power placed in operation in Hungary. This program would be similar to spent fuel strategies of other small VVER operators, such as the Czech and Slovak Republics and Finland. It is also consistent with the current international tendency of each country operating nuclear reactors to develop an independent high-level waste management strategy for storage and disposal.

The options available to Hungary for the management of spent fuel include: reprocessing, direct disposal or a deferred (wait and see) decision between these two options. In the latter case, the spent fuel would be stored in Intermediate Spent Fuel Store (ISFS) for approximately 50 years. Each of these options, for the backend of the fuel cycle, requires a geological repository for disposal of HLW. For the reprocessing option, the waste forms would be vitrified fission products, plutonium waste (if mixed oxide fuel is not acceptable) and eventually fuel assemblies (when the fissile material can no longer be recycled). For the direct disposal option, the waste form would be one or more types of spent fuel assemblies in containers. For the wait and see option, the waste form will depend on whether reprocessing or direct disposal is selected as the management strategy at the end of the storage period.

12.4 Geological Formations to be Evaluated

The long-term management of high-level radioactive waste has only recently become an issue in Hungary. Concurrent with the attempts to return spent fuel to Russia, a decision was made to erect an ISFS at Paks NPP, and this facility is now being constructed. The decision concerning the spent fuel management option at the end of the interim storage period has not been made.

Because of the geology of Hungary, only a limited number of potentially suitable disposal sites for high level

waste are available within the country. One of these is a Permian claystone deposit called the Boda Claystone Formation near the city of Pécs in Southwestern Hungary. The uranium mine is located in a Permian sandstone formation close to part of this claystone formation. The formation has also been investigated as a potential host for low-level waste disposal. When high-level disposal became a concern for Hungary, the Mecsek Ore Mining Company proposed that the Boda Claystone Formation, of Permian age, be investigated for its suitability as host rock for a Hungarian nuclear waste repository. Information about the lithology and structure of the overlying sandstone has been collected during uranitun mining over the past 40 years.

Useful information about, the groundwater flow conditions of the sandstone has also been collected from the mining operations. About 50 boreholes were drilled from surface to investigate the uranium deposits in the sandstone. Four boreholes have penetrated the underlying Boda Claystone Formation to considerable depths (a few hundred meters). Two of these boreholes were drilled in 1991 to 1200 m depth to obtain information about the vertical characteristics of the claystone over most of its thickness. In 1993, a specific study program was started within the framework of the National Waste Disposal Program to further examine the Boda Claystone Formation. The following activities have been carried out under this program:

- several underground core drilled boreholes;
- an access tunnel (280 m length), about 80 m into the Boda Claystone);
- systematic sampling in the tunnel;
- geological mapping (and documentation) of the tunnel walls including the subsequent tunnel faces;
- hydrogeological boreholes, corresponding measurements, and collection of water samples;
- rock mechanics measurements during the excavation of the tunnel, and after completion;
- geophysical measurements;
- geochemical, mineralogical and geomechanical laboratory investigations in affiliated institutions; and
- laboratory testing of water samples.

So far, the investigations from this exploratory tunnel and two previous deep surface boreholes, BAT-4 and BAT-5, indicate that the Boda Claystone Formation is a highly compact rock unit of very low overall permeability ($<10^{-10}$ m/sec). Based on an understanding of the faults, fractures and their relative movements from studies of the overlying uranitun bearing sandstone, it is believed that most of the faults in the claystone are sealed and filled with calcite, barite, gypsum and clay minerals. The accumulated data suggests there are only a few widely spaced fault zones in the Boda Claystone Formation. Methods to determine the location, orientation, extent and hydrogeologic characteristics of these faults are currently being assessed.

The objective of all the above studies was a preliminary characterization of the Boda Claystone Formation as a potential host rock for a high-level radioactive waste repository. The results obtained so far confirm the geochemical and hydrogeologic suitability of the claystone for HLW disposal and favor further investigation of this claystone formation. This is particularly the case in view of the fact that there is not a wide choice in Hungary of suitable geological formations (size and quality) for a HLW repository.

12.5 Geological Structure of the Area

The Western Mecsek Mountain Permo-Triassic inlier is bounded to the north, south and east by three left-lateral, strike-slip faults. These enclose gently eastward-plunging (~10°) folds. A broad anticline, flanked by smaller synclines, contains the detailed area of interest, and is bounded to the north by a 30° south-dipping reverse fault, and to the south by a 50° north-dipping reverse fault. Both of these faults strike east-northeast and are paralleled by lesser reverse, normal and oblique-slip faults with high intermediate to subvertical dips that are connected by splays in some cases. A further set of east-dipping normal faults strike north-northwest, perpendicular to the anticlinal axis.

The stratum of interest, the Boda Claystone Formation, is a silty claystone at least 800 m thick. This is a lacustrine deposit set within a sequence of fluvial sandstones all of which were deposited in a semi-arid climatic environment. The claystone is bounded beneath by a conglomeratic sandstone (Cserdi Formation) concordantly gradational over 100 m. It is bounded above by another conglomeratic sandstone (Kővágószölösi Formation), which contains the uranium deposits mined by Mecsekurán Ltd. This upper contact is gradational over 40 m but is discordant in places.

The Boda Claystone Formation is exposed at the surface over an area of about 20 km^2 near the village of Boda. In order to evaluate this formation, it will be necessary to study the entire unit, down dip from the surface outcrop towards Shaft IV (transportation and ventilation)

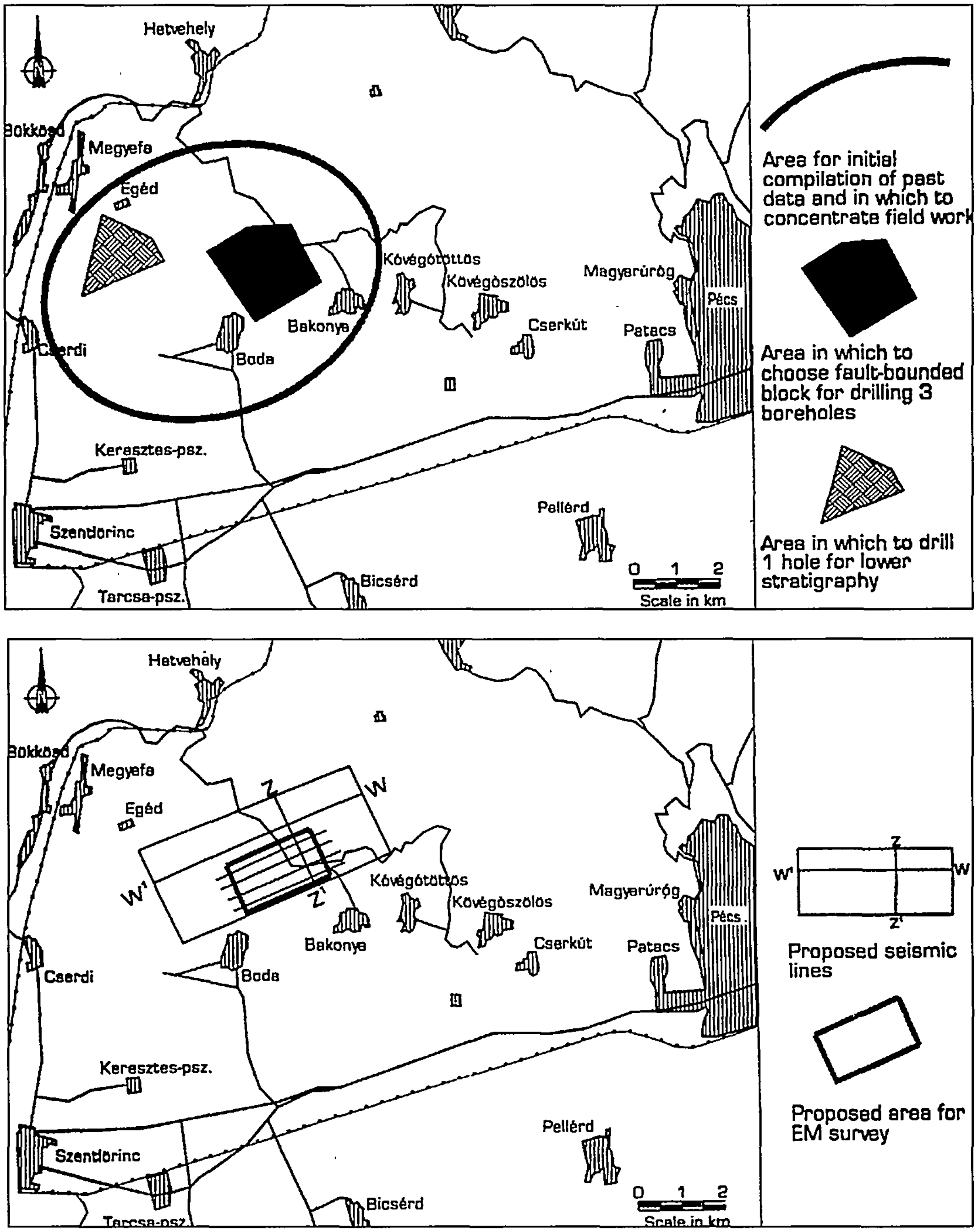

Figure 12.1. Proposed surface geological and geophysical investigations.

and Shaft V (transportation). This will define a candidate area of approximately 40 to 50 square kilometers, as shown in Figure 12.1. The tentative regional study area is marked on Figure 12.2 by boundary IJKL. It is estimated to contain about 40 cubic kilometers of Boda Claystone Formation with an average thickness of 800 m, in which exploration can be carried out for the selection of possible disposal sites. The northern boundary could be extended further to I'J', if necessary.

Being a lacustrine deposit, the Boda Claystone Formation, probably varies lithologically over much larger distances laterally than vertically. It contains several stratigraphic members in the vertical, stratigraphic

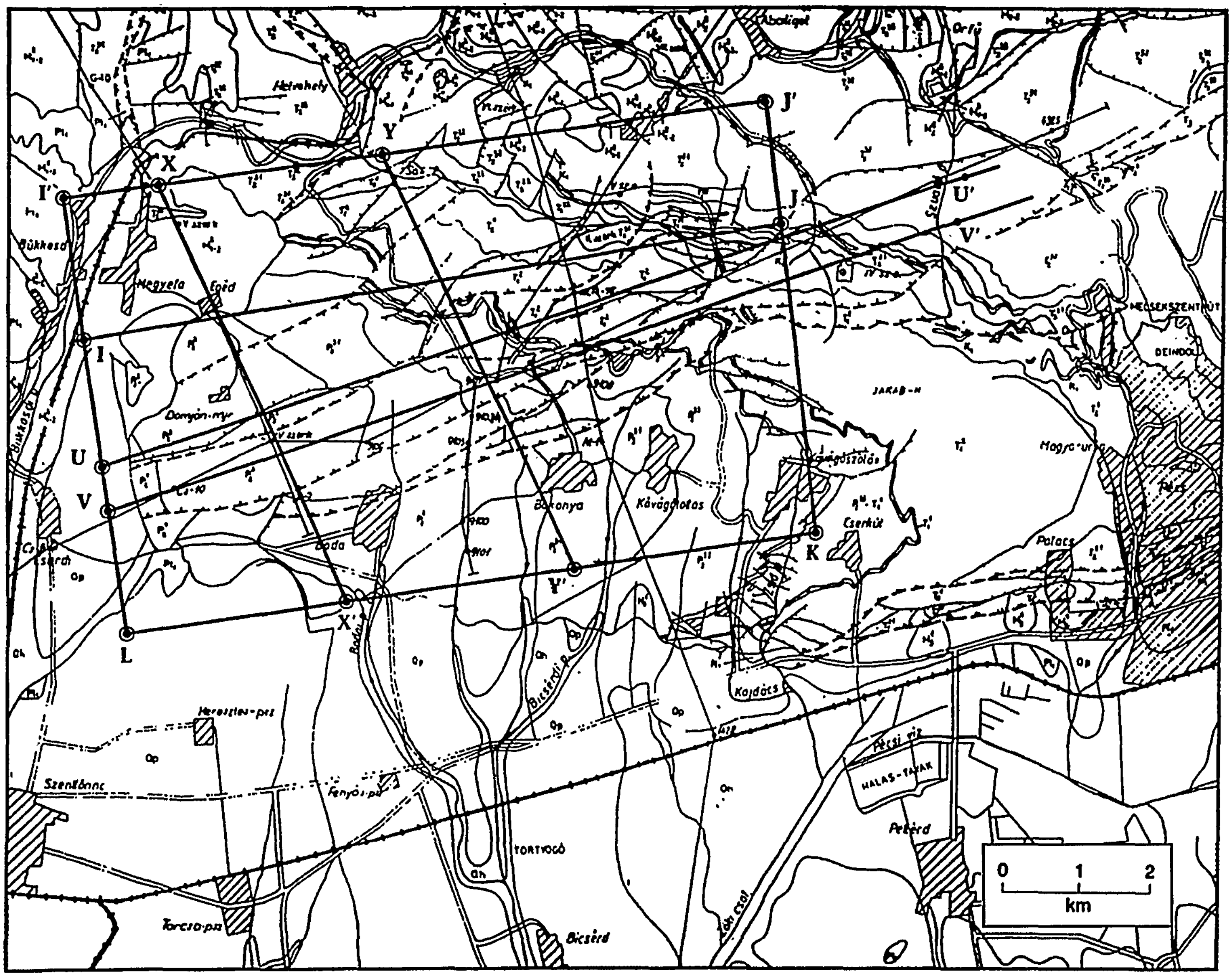

Figure 12.2. Geological map showing proposed study area.

section, without obvious boundaries but with slightly different properties. In general, the chemical composition is quite uniform throughout stratigraphically; it is composed of quartz and feldspar with a high clay content and a significant amount of fine grained hematite. The claystone has a higher quartz content in the upper part of the formation, and the matrix contains up to 10% primary dolomite in layers 1-10 cm thick. The claystone is generally oxidized but a few unoxidized layers are evident in the geologic logs of some boreholes drilled from the surface. These layers can be at least up to 20 m thick.

Knowledge of the location, extent and hydrogeologic properties of fault zones and fractures within the Boda Claystone Formation is of critical importance in selecting a suitable location for a high-level waste repository. Based on geologic mapping of the overlying sandstone in the uranium mine, it is clear from a WSW-ENE cross-section that the 700 to 900 m thick claystone stratum may be offset by a number of WSW-ENE trending fracture zones. This claystone may also be affected by north-south trending subvertical faults. From recent mapping in the "Alpha" exploratory tunnel at the 1100 m depth in the mine, two dominant sets of mesoscopic fractures have been observed in the Boda Claystone Formation. These fractures are filled with calcite and clay minerals, and are apparently sealed to groundwater flow. It will be essential to develop a much more detailed understanding of the lithological, hydrogeological, geophysical, geochemical and geomechanical conditions of this claystone and adjacent formations and crosscutting faults to develop performance assessment models in order to evaluate different areas of the Boda Claystone Formation for a high-level waste repository.

12.6 CHARACTERIZATION PROGRAM OUTLINE

The region of the Boda Claystone Formation that has been identified as a potential siting area will require a comprehensive characterization program to select the preferred disposal site. The existence of an access tunnel from the uranimn mine into the claystone at 1100 m, provides an opportunity to characterize some potentially important vertical structures that could act as groundwater pathways through the claystone. Thus, the characterization program for this particular area in the Boda Claystone differs from the program that would be developed for a site or formation where there is no existing underground access.

The characterization activities are part of a long-term program that could take 7 to 15 years to complete before a final disposal site in the Boda Claystone is approved. However, there is a need to determine whether or not a long-term program should be initiated in the Boda Claystone. The objective of this short-term program is to consolidate the information that has been gathered during 40 years of mining related activities in this region, collect new information from selected target areas and develop a comprehensive geotechnical model for the region.

Very slow groundwater movement or diffusion within pores and fractures in the Boda Claystone Formation, surrounding the site eventually chosen for the repository, is expected to provide an effective barrier to the release and migration of repository contaminants. Adsorption on the clays in the rock matrix and fracture fillings is also expected to be a major factor in retarding the contaminants. Thus, a thorough knowledge of the groundwater flow and diffusion paths through the claystone and adjacent rock formations is required to determine the likely pathways for radionuclide migration from potentially suitable repository locations to the human environment.

The main problem in characterizing the candidate area is to choose a location for the repository such that the hydrogeologic and geochemical conditions will delay and impede the release and migration of repository contaminants to the accessible human environment. This will involve both surface and underground evaluations of the Boda Claystone Formation and the overlying and underlying sandstone formations in the candidate area.

A major emphasis in the site characterization program would be on developing an understanding of the physical and chemical conditions of the groundwater pathways in the claystone and determining how these conditions interact with the fluid flow conditions in the overlying and underlying rock formations. Another emphasis would be on determining any future change or disruptions that could affect these flow conditions. Still another emphasis would be on determining how any future changes or disruptions, during the construction or operational stages of the repository or during the time frame of concern for long term assessments of performance, could affect groundwater movement in the Boda Claystone Formation.

Because this claystone has a very low primary permeability, yet is situated in a geologic setting that has been faulted and fractured, it will be necessary to determine

the geometry and hydraulic properties of the fracturing at different scales within the formation to develop an understanding of the groundwater flow conditions. The groundwater regime in the overlying Permian sandstone has been dramatically affected by the uranium mining activity, and therefore, it will differ from original conditions. The disturbed hydrogeological regime in the overlying Permian sandstone may have altered the groundwater conditions in the Boda Claystone, and these effects will also need to be evaluated.

The data gathered from these activities and data already collected by Mecsek Ore Mining Company, and other Hungarian institutions can be used to select a preferred site for a HLW repository within the potential siting area. There are two possible alternatives that would affect the characterization program. They are:

- the preferred repository site is close enough to the uranium mine so that the site can be accessed from the mine; or
- the preferred site is located such that a new access from the surface would be necessary or preferred.

If the preferred repository site is chosen near the location of the uranium mine in the overlying sandstone so as to be accessible from the mine, the local characterization program would include some surface boreholes to provide information immediately around the repository. However, the majority of the characterization program would probably be conducted underground from currently existing and new access tunnels.

If the preferred repository site is located away from the uranium mine where new access from the surface is required or preferred, the initial characterization will be mainly from the surface into the undisturbed portion of the Boda Claystone Formation and will progress logically to underground characterization activities as new access to the underground is constructed.

Under normal circumstances, a characterization program to select a preferred site for a HLW repository would initially be surface based, followed by an underground evaluation program at the preferred site using exploratory shafts and tunnels. This is the approach recommended by Davison et al., for the Canadian nuclear fuel waste disposal concept. In the case of the Boda Claystone Formation, the underground environment has already been accessed by an underground tunnel at 1100 m depth from the adjacent uranium mine, thus providing the opportunity to conduct further underground characterization activities, if there are both cost and technical advantages to doing so.

Because the area presently under consideration as a nuclear fuel waste disposal site is relatively large (40 km^2), both surface-based and underground characterization programs are recommended. One of the normal advantages of a surface-based characterization program is that the underground environment is relatively undisturbed prior to excavation. This allows background measurements of the undisturbed conditions at the site to be obtained from a network of surface-based characterization boreholes, and allows the construction of a detailed model of the groundwater regime at the site. Predictions of drawdown resulting from shaft sinking and the excavation of exploratory tunnels can be made and compared to actual measurements to assess the accuracy of the hydrogeological model.

For the characterization of the Boda Claystone Formation, the fact that many of the potential faults extending down from the overlying sandstone formation are sub-vertical means that they are better accessed and instrumented using sub-horizontal boreholes that are easily drilled from the underground. However, there are significant operational problems associated with conducting characterization activities from the underground. Firstly, there may be limitations on the drilling of long exploratory boreholes underground due to technical factors. The drills presently used underground at the uranium mine are apparently limited to drilling boreholes to a maximum length of 300 m. Suitable equipment to drill longer, and perhaps larger diameter boreholes would need to be acquired. In addition, the ambient rock temperature at the depth of the present access tunnel into the Boda Claystone Formation is about 45° C. If this exploratory tunnel is advanced further into the claystone, the distance to the closest shaft will increase, and with it the problem of cooling and ventilating the working areas.

The present laboratory tests and measurements from the exploratory access tunnel were finished by the end of 1994. Based on the encouraging results of these characterization activities, further investigation of the Boda Claystone Formation as a host formation for HLW disposal was recommended; there is not an abundant choice of suitable geological formations (size and quality) in Hungary for siting and for economic reasons.

The activities recommended for characterizing the Boda Claystone Formation, both surface-based and under-

ground, have been organized according to discipline and/or activity. For both approaches these include: geological mapping, geophysical surveys, exploration drilling, hydrogeological monitoring and testing, geochemical and hydrogeochemical studies, *in situ* stress determinations, site modeling and model validation, laboratory testing and analyses, and seismic risk analyses.

In addition, it is recommended that full-scale, *in situ* tests be conducted underground in representative geological conditions. Such tests could include: radioactive migration testing, excavation response studies, thermal studies and tests, thermal studies on rock behavior, testing of mining methods related to minimizing the disturbed zone, and ground support methods in the Boda Claystone Formation.

12.7 Short-term Characterization Program

A short-term characterization program is proposed as part of the activities being carried out to investigate the suitability of the Boda Claystone Formation as a host rock for a high-level nuclear waste repository. The objective of the short-term study is to gather, analyze and interpret existing and new information on this formation in order to decide whether to begin a long-term investigation of the claystone as a suitable high-level, radioactive waste disposal medium.

The current situation regarding geoscience investigations in the Boda Claystone Formation is as follows. Data from previous investigations carried out during the development of the mine, some going back as far as 40 years, are available. There are also several investigations in progress at the present, and the necessity of initiating some entirely new investigations or repeating earlier surveys, where the results were not satisfactory in the present context, has been recognized. Analysis and interpretation of all of the above described data will be required in order to make a decision on a long-term characterization program. To assist in the compilation of these data, a computerized database will have to be established, so that data are available, on a systematic basis, for compilation, interpretation and modeling.

The field component of the proposed short-term program will consist of surface as well as underground activities. The latter are divided into two phases; the first phase of the underground activities will not include further excavation, whereas the second phase will be based on additional excavation.

There are certain time constraints on the short-term program, due to the governmental decision to close the uranium mine by 1997. All field activities, both surface and underground, must be completed in a two-year period, commencing in May 1995. Analysis and interpretation of field data must be completed within a reasonable time period after March 1997.

The short-term characterization program will concentrate on confirming the important processes controlling groundwater flow and possible radionuclide, transport in the Boda Claystone Formation. Particular attention will be paid to developing an understanding of the occurrence and geotechnical properties of tectonic zones, or fracture zones, within the claystone. Knowledge of the location, extent, hydrogeological properties, geochemical properties and geomechanical properties of tectonic zones and fracture zones within the Boda Claystone in the candidate siting area is needed to develop models of the groundwater flow pathways within the Boda Claystone Formation in order to conduct assessments of the long-term safety and performance of a repository in the claystone.

Investigations carried out over the past 40 years during the development of the uranium mine, indicate that the Permian sandstone overlying the Boda Claystone Formation is cut by a series of inclined fracture zones. Based on geological mapping, these fracture zones may be spaced as close as 50 to 200 m. Evidence from the "Alpha" tunnel, at the 1100 m level of the mine indicates that these fracture zones are water bearing but that the blocks of Boda claystone bounded by these fracture zones are relatively impermeable. There is no information on the occurrence, distribution or hydrogeological and geochemical characteristics of these fracture zones within the Boda Claystone Formation. The short term characterization program involves performing some surface-based characterization studies (including deep borehole drilling) near the outcrop area of the claystone and by performing some underground characterization studies in the existing exploratory tunnel into this formation from the 1100 m level of the uranium mine.

At present, little is known of the faults and fracture network in the Boda Claystone Formation at the depths proposed for the repository. This information can best be obtained by borehole drilling, combined with geophysical surveys and with some detailed geological mapping of fault subcrops if access is available. Also, hydrogeological investigations and hydrogeochemical sampling in the boreholes will be very important components of subsurface characterization. Analysis and collation of existing data through a database will also be

very useful.

Because of the time constraint, only a largely conceptual model of the controls on groundwater flow through the Boda Claystone, as opposed to a partly idealized deterministic model, can be developed. Therefore, the surface investigations are to be concentrated in a few areas (at least two) from which a good understanding of the principal factors controlling the flow of groundwater can be obtained.

It is suggested that the surface geological investigations (surface mapping) be confined to the same area as the proposed geophysical surveys. The two principal objectives of surface mapping are to:

1. determine the location and conduct fault-rock characterization for the subsurface projection of faults in the Boda Claystone, especially those that are potential drilling targets. This can be achieved by detailed mapping, from outcrops or trenches, of topography, mesoscopic fractures, rock alterations and fracture fillings. This work will be concentrated in the area in which a fault-bounded block is likely to be chosen as a drilling target.
2. determine the general character and history of ductile, brittle deformation of the Boda Claystone, and decide whether lithologic variation is likely to be important. For example, the bedding attitude data in the underground tunnel into the Boda Claystone from the 1100 m level of the mine suggest crossfolding in the claystone with a NW to NNW plunge. Such zones of folding may be centers for more concentrated mesoscopic fracturing.

Drilling and geoscientific and geotechnical characterization of four deep boreholes from the surface is proposed. Three boreholes are to be drilled to investigate conditions and to characterize the bounding faults within two adjacent fault-bounded blocks within a 3-4 km radius to the northeast of the village of Boda. Borehole depths would depend on the actual location chosen, but two of them should be 500-700 m deep. The third should be at least 1300 m deep and penetrate the sandstone footwall. The fourth borehole is to be drilled approximately two km northwest of Boda within the outcrop of the Boda Claystone Formation. This hole is also expected to be about 1300 m in depth to obtain hydrogeologic information in an area expected to have sparse faulting. This borehole should penetrate well into the footwall and be used to characterize the lower members of the Boda Claystone Formation, geoscientifically and geotechnically.

12.8 CONCLUSIONS AND PROSPECTS

In 1993, the decision was taken to develop a relatively inexpensive access from the existing mine to study the adjacent formations. This involved developing and servicing an exploratory access tunnel from the uranium mine into the Boda Claystone Formation. This exploratory access was completed in 1994. The tunnel has been used to conduct a preliminary research program to study the *in situ* characteristics and properties of the claystone. These data have contributed to assessing the potential suitability of the Boda Claystone for high-level waste disposal in terms of safety assessment and repository design. The results from these preliminary studies have continued to confirm the geochemical, geomechanical and hydrogeologic suitability of the Boda Claystone as a host media for high-level waste disposal in Hungary.

As the Boda Claystone Formation has been identified as a potential high-level waste disposal site, experts are now developing a thorough program to assess its suitability. This will require that sufficient data be gathered on the characteristics of the claystone to select appropriate location(s) for a repository and to complete a safety assessment for a repository at each location. An extensive database of geologic formations exists for the area of the uranium mine, both from the previous surface-based exploration programs as well as from the underground characterization program. Much of information is relevant to understanding conditions in the Boda Claystone beneath the uranium bearing Permian sandstone. The structural information on jointing and fracturing is particularly relevant to the identification of suitable locations for a high level waste disposal repository.

The most probable way for radionuclides, that are sealed in a deep geological repository to reach humans and the environment is by transport in groundwaters through the pores and cracks in the rock surrounding the repository. Therefore, one major requirement in characterizing possible repository sites in the Boda Claystone Formation is to quantify the important processes governing the transport and absorption of radionuclides within the groundwater systems in the rock mass as well as the engineered barriers used in the repository. Groundwater movement in rocks such as the Boda Claystone is mainly controlled by the presence or absence of fractures, the permeability and interconnectivity of these fractures, the permeability and porosity of the unfractured rock, and ground-

water pressure gradients. The chemistry of the groundwater, the rock matrix, and minerals within the fractures also govern the potential transport of radionuclides within groundwater. Therefore, the presence of open fractures (joints and bedding planes), fracture zones and processes that can change their hydraulic or chemical properties, must be understood for repository siting and design.

Two underground facilities are likely to be necessary for the characterization of a site and the design of a repository in the Boda Claystone Formation. The first facility is the Underground Characterization and Test Facility (UCTF) that is being started with the construction of the Access Tunnel from the Mecsekurán uranium mine into the claystone. It will be used to provide information on the suitability of the claystone as a repository medium as well as preliminary repository design information, and to train staff, develop methods and study design and safety issues. Because it is located near the uranium mine and the contact between the sandstone and the claystone, it is likely to be in a disturbed environment.

The second facility is a repository characterization facility (RCF), to be located at the preferred repository site, which would be constructed late in the site characterization stage. It would provide access to the repository rock volume for final assessment of the site suitability, and for commissioning and longer-term coupled interaction tests on repository systems and components. It would be incorporated into the repository design so that its excavations would become part of the repository and would not interfere with the repository operation.

The UCTF will be located in a section of claystone that likely has been influenced by the mine construction and operation. In particular, the groundwater pressures, flows and chemistry in the sandstone, and perhaps the claystone, are likely to be affected by the presence of the mine. As there has not been a long-term hydrogeological monitoring program in the sandstone and claystone near the mine, the original groundwater conditions and the magnitude and areal extent of the disturbance caused by mining activities over time are not known.

Without this information, studies and demonstrations that require regional groundwater conditions as a step in planning or in performance and safety related analyses may not be possible with the limited data gathered from the UCTF. These studies will have to planned as part of RCF activities at a preferred repository site where these data can be collected. However, significant groundwater studies can still be done in the UCTF to:

- develop and demonstrate the equipment, instrumentation and procedures for drilling and monitoring groundwater conditions in boreholes;
- install monitoring systems in new and existing boreholes to study the hydraulic connections in fractured rock; and
- observe the pressure and chemistry changes that occur when new boreholes and excavations are done.

These are valuable and necessary studies in preparing for the characterization of a potential repository site, and for the environmental and safety assessments. These studies will also be used to select the methodologies that should be applied in the characterization and assessment studies that will be conducted at the preferred repository site.

Also, it is our understanding that the mine operations have been confined to the sandstone formation. This may make it possible to assess the impact of a large "drain" (i.e.,. the mine in the sandstone and the UCTF in the claystone) on the pore water pressures within the Boda Claystone Formation. If it can be demonstrated that the pore pressures in this claystone have been only slightly affected by the presence of the mine and the development of the UCTF, this information can be used to demonstrate the large-scale low permeability of the claystone, an attractive characteristic for waste disposal. This type of *in situ* information can only be obtained by monitoring the response of the groundwater system to construction of underground excavations such as the access tunnel and the UCTF.

The effects of the mine excavations on the mechanical conditions in the claystone around the UCTF will probably not significantly influence the studies and demonstrations that may be planned for the facility. However, the proximity of the UCTF to the contact between the sandstone and the claystone may have to be considered in analyzing the data from UCTF studies and in extending the results to a potential repository site elsewhere in the claystone.

The second underground facility, the repository characterization facility (RCF), would be constructed at the preferred repository site during the later stages of repository site characterization to conduct tests: (a) to confirm the suitability of the site for disposal; (b) complete the repository design; and (c) refine and test repository systems and components. The RCF design would be inte-

grated into the repository. As the repository site will likely be several kilometers away from the uranium mine and away from any other underground excavations, it should have a relatively undisturbed groundwater environment. In the site characterization program at the potential repository site, the regional characterization and the long-term hydrogeological monitoring would be done prior to any underground excavation so the baseline hydrogeological and chemical conditions would be known. The changes in these conditions caused by additional borehole drilling, and by the excavation and testing in the RCF would be known. The changes in these conditions caused by additional borehole drilling, and by the excavation and testing in the RCF would be measured and could be used to develop and test computer models that will be used to predict the performance of the groundwater system.

REFERENCES

Buday, G., Recent status of spent fuel storage in Hungary and outlook to the future, IAEA AGM, Vienna, April 15-18, 1992.

Buday, G., and G. Ferenczi, Status of the nuclear fuel cycle back-end strategies, IAEA TGM, Vienna, June 1-4, 1993.

Ormai, P., L. Maróthy, and K. Bérci, Hungarian policy on the back-end of the fuel cycle, VVER Nuclear Waste Management Workshop 92, Toronto, Canada,1992.

Vigassy, J., I. Czoch, and P. Ormai, Hungarian experience with the National Project for Radwaste Management, Symposium on Waste Management, Tucson, USA, 1994.

A recommended program for the characterization of the Boda Claystone Formation as a potential site for the safe disposal of high-level radioactive wastes, Report by AECL Research, 1995.

Design requirements for the construction and operation of an underground characterization and test facility in the Boda Claystone Formation, Report by AECL Research, 1993.

CHAPTER 13

STATUS OF SITING AND HOST ROCK CHARACTERIZATION PROGRAMME FOR A GEOLOGICAL REPOSITORY IN INDIA

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13.1 INTRODUCTION

The Indian programme in search of suitable sites for location of a deep geological repository for disposal of high level vitrified waste has been in progress for the last few years. The whole country has been screened, based on well identified criteria, for suitable host rock formations in tectonically stable areas. The programme is being pursued in a phased manner in stages, to narrow down the choice from larger areas to a few candidate sites of specific size. In parallel, a study is underway to directly evaluate and characterize the host rock lying within the exclusion zone of a nuclear power plant for locating a possible waste repository.

The main host rock under consideration is a plutonic granitic formation available in tectonically stable areas and having homogeneous and uniform geological, structural, physico-chemical and hydrogeological characteristics with a favorable socioeconomic environment.

Out of thousands of square kilometers of granitic areas screened, the choice has been narrowed down to a few granitic zones of 100 - 150 sq. km. Two such zones have been investigated in detail to further demarcate the most suitable sub-zones. Micro level investigations are to be taken up in the sub-zones to assess the possibility of finding the repository candidate sites.

In one of the sites, geological, hydrogeological, geophysical (preliminary), environmental and socioeconomic surveys have been completed, whereas in the other, deep drilling down to a depth of 620 metres is in progress. Simultaneously, laboratory investigations to evaluate mineralogical, petrographical, micro-structural, thermal, mechanical, physical and chemical properties of the rockmass are also in progress. Modeling studies for joint fracture characterizations, ground water flow and radionuclide migration, and stress behavior in a conceptual repository have been taken up. Geophysical surveys, bore hole logging and *in-situ* stress measurement are underway.

13.2 REPOSITORY SITE SELECTION PROGRAMME

The programme is aimed at selecting one or more geological repository sites and characterizing them for final disposal of immobilized high level radioactive waste, through various stages of investigations comprising field and laboratory studies. To carry out the multi-disciplinary investigations, a number of national expert organizations/agencies have been involved to address specific issues.

13.2.1 Major Site Selection Criteria

The major criteria on which the selection of a candidate repository site depends are:

- Tectonic stability of the area;
- Three dimensional homogeneity, large extent and massiveness of the host rock mass;
- Suitable hydrological and hydrogeological environment;
- Favorable thermal, thermomechanical and geochemical properties of host rock; and
- Favorable socioeconomic factors.

13.2.2 Stages of Site Selection

The approach followed for site selection is to narrow down the choice of the area with an increasing level of confidence. This is planned to be achieved in different stages, reducing the size of the area to smaller entities at every stage. The following main stages are considered:

Stage I. Collection of all available data from various sources and their interpolation to evaluate specific

attributes for delineating promising zones. Such data have already been generated and a few zones have been identified for further investigations.

Stage II. Mainly semi-detailed studies including data generation through field surveys, geological mapping, hydrological investigations, collection of soil/rock/ water samples and their analysis. Based on the data generated during these investigations, two zones have been identified for further studies.

Stage III. Extensive field surveys, detailed geological and structural mapping, subsurface investigations and intensive analysis and interpretation of all parameters. Detailed geological and structural mapping of the potential zones has already been initiated. Subsurface investigations have also been planned in this stage.

Stage IV. Micro level studies, subsurface characterization with the help of geophysical investigations and ground water flow studies.

Stage V. Finalization of site for pilot repository from one of the selected sites. Further detailed investigations by conducting *in-situ* experiments.

Stage VI. Final stage involving actual design and construction of a full fledged repository.

13.2.3 Evaluation of Zones and Methodology

Based on the criteria developed, a number of attributes have been identified for data acquisition, collation and interpretation. An attribute is a factor or a parameter having varying degrees of influence on the geological, mechanical, hydrological, thermal and radiological integrity of the proposed repository. Each attribute has been allotted a weighting factor depending upon its relative importance at each stage of repository siting. In Stage I, 20 attributes as listed in Table 13.1 were thoroughly examined. Each attribute was allotted a maximum score point of 10 and a minimum of 0. These attributes have been organized and grouped based upon their relative merits in different stages of the programme.

Initially geological, structural, tectonic, geomorphological and socioeconomic data were collected and represented on 1:25,000 scale for further evaluation. All the granitic areas were then divided into grids of 20 x 20 km or 10 x 10 km. Each unit was taken as a candidate unit for matrix analysis. The units scoring maximum points were considered as promising zones for further evaluation. Based on the product of score points for each attribute and the corresponding weighting, a relative grading was determined using the following equation:

Table 13.1. List of attributes considered in Stage 1.

No.	Attribute
1.	Lithological formation
2.	Seismicity
3.	Distance form stuctural discontinuities
4.	Distance from surface water bodies, viz. dams/rivers/lakes, etc.
5.	Ground water level
6.	Rainfall
7.	Surface water runoff
8.	Population
9.	Distance from economic mineral occurrences and mining activities
10.	Distance from industrial/archeological/tourist/ religious spots
11.	Floods
12.	Soil cover and weathering pattern
13.	Vegetation cover
14.	Accessibility of the area
15.	Intensity of intrusives, veinlets, etc.
16.	Joint patterns, fracture pattern, etc.
17.	Topography
18.	Dip of formation/foliation
19.	Homogeneity of rock mass
20.	Political awareness

$$GI = \Sigma \ Wi \ x \ Pi$$

where: Wi = weighting factor of attribute, Pi = Score points against an attribute, and GI = grade index for suitability.

Until now, this methodology has been adopted in Stage I and has enabled an identification of two favorable zones of 100 to 150 sq. km for detailed investigations in Stage II.

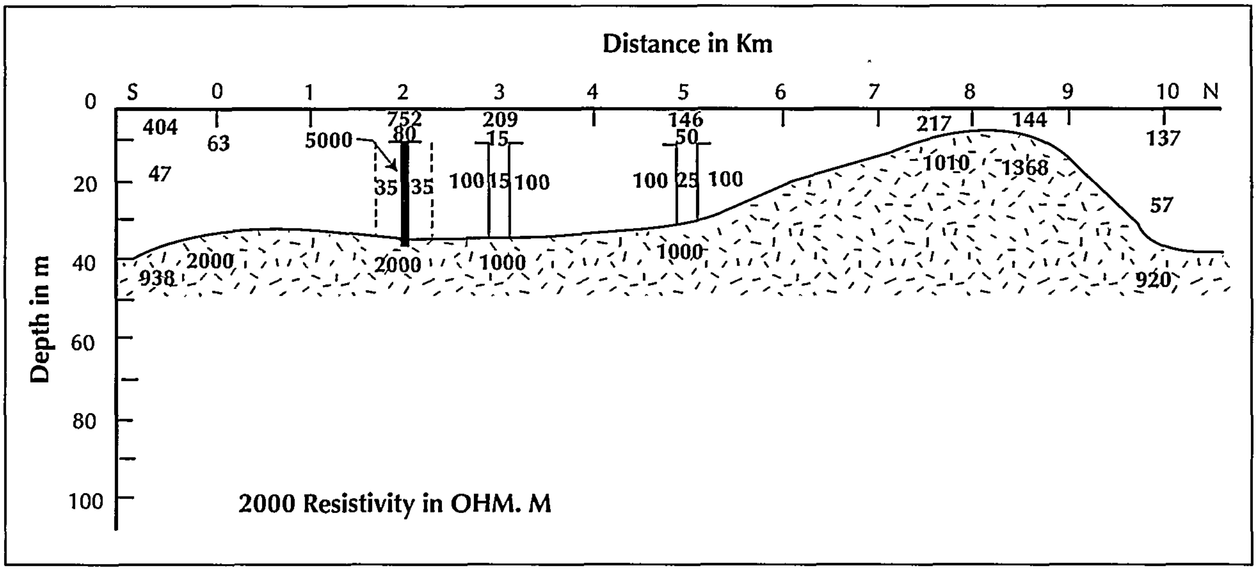

Figure 13.1. A resistivity section along the main N-S traverse P1, based on joint DCR and HLEM 1-D inversion and limited 3-D modeling.

In the second stage, demarcation of two zones of 25 to 30 km^2 has been achieved. In the third phase, further narrowing down of these zones to those of four sq. km each will be considered. They will then be treated as test or candidate sites.

The methodology adopted in this programme has been found to be most suitable. Subsurface investigations have now been planned for further characterization of the granitic rockmass. The zones so far identified largely satisfy the conditions of compositional and structural homogeneity, massiveness and a suitable hydrogeological set-up. The rainfall in these zones is scanty, population density very low, and vegetation cover almost negligible.

13.3 Repository Site Characterization Programme

While characterizing a particular rock formation for its structural and compositional homogeneity, a thorough assessment of its geological, structural, hydrogeological, geophysical and petrographic characteristics is essential. In addition to the above investigations, it is also required to study various thermal and mechanical properties of the rockmass so as to understand its behavior at elevated pressures and temperatures.

13.3.1 Geophysical Characterization

Geophysical surveys involving electrical methods were carried out at one of the promising zones to test the suitability of various methods to evaluate the homogeneity of the granitic rock mass under investigation and detection of fracture zones, intrusive rocks etc. The observations were recorded over a path length of 12 km at an interval of 100 m. The line was so selected as to almost bisect the area. The following methods were employed to achieve the objective:

1. Direct Current Resistivity (DCR) method to obtain information at greater depths;
2. Horizontal Loop Electro-Magnetic (HILEM) method to probe shallow depths; and
3. Very Low Frequency (VLF) method to get near surface information.

Based on the data obtained, it has been interpreted that beneath a sandy top cover, weathered to semi-weathered granite exists between varying depths of 7 to 60 m below the surface with unweathered fresh granite below these depths. A dyke was also demarcated along the line of a survey. Figure 13.1 shows the vertical section along this line.

Besides these surveys, a seismic sounding was also carried out over selected outcrops along two short orthogonal lines using a hammer source. This experiment was conducted for studying seismic propagation characteristics of the area when there is little or no sand cover.

From the above investigations, the following possible

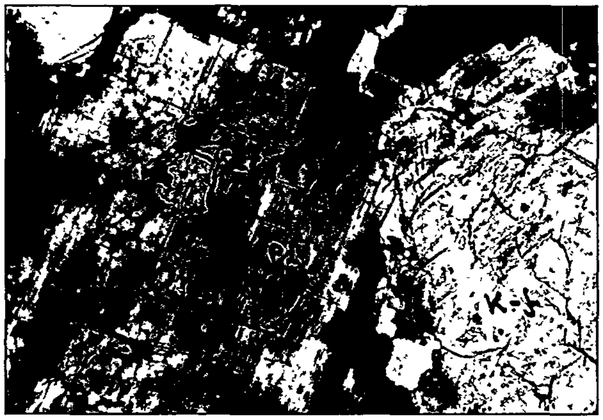

Figure 13.2. Plagioclase feldspar (P1) and K-feldspar (k-f), filled with sericite in a thin section of granite. Network of microcracks are also observed. Under crossed polars (x16).

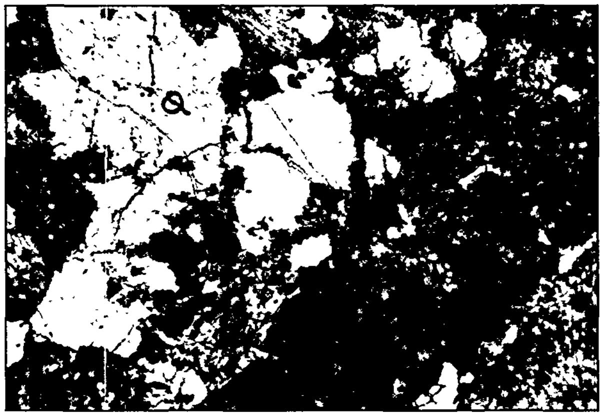

Figure 13.3. Microcracks in quartz and plagioclase (P1) in a granite thin section. Under crossed polars (x16).

subsurface conditions have been brought to light:

- Extensive cover of loose sand;
- Low resistivity near the surface indicating higher degree of weathering of the top zone;
- Highly variable surface and near surface electrical resistivity and seismic velocity; and
- Possible presence of shallow anomalous objects within deep weak zones in granite.

It may, however, be noted that these interpretations are based on a single test survey. More detailed investigations are planned in the near future, using advanced techniques like a Multi-channel Digital Data Acquisition System, which will retrieve very weak signals from greater depths.

A resistivity survey on a grid interval of 5 x 5 km or 10 x 10 km will be conducted shortly to assess the major elements of inhomogeneity in the rock mass down to a depth of 1 km.

13.3.2 Petrographic Characterization

Petrographic examination of selected samples of granite in the area under investigation was carried out to understand the microscopic features of the rocks. The following parameters were emphasized:

- Deformation and consequent stress effects on mineral grains, especially quartz;
- Presence of microfractures in the rock;
- Nature of alteration of constituent minerals; and
- Radioactivity of constituent minerals.

The salient petrographic features of granite in one of the zones are as follows:

- The rocks can be classified as porphyritic micro granites, micro diorites and porphyritic granophyres, with major quartz (30% to 45%), potash feldspar (22% to 54%) and plagioclase feldspar (12% to 28%). Accessory minerals include biotite, chlorite, epidote, apatite and opaques, mainly oxides of iron and titanium;
- Feldspars have undergone a considerable degree of weathering and have been sericitised. Biotites have been chloritised;
- In general, the overall degree of alteration of granites is moderate, as supported by weathering index data;
- Radioactive assay data for these samples indicate that the granite contains 15 to 20 ppm eU_3O_8. Solid State Nuclear Track Detection (SSNTD) studies indicate that no discrete radioactive phase is present in the samples; and
- In most samples, quartz is unstained and microcracks are not widespread.

The above observations suggest that the granite rock mass under investigation could form a suitable host rock from the petrographic point of view. Figures 13.2 and 13.3 show mineral assemblages and micro fractures in

Table 13.2. Compressive strength of heat treated granite.

No. of Samples	Thermal Treatment Temperature (°C)	Unaxial Compressive Strength (MPa)	Average Strength (MPa)	Standard Deviation
14	unheated	205, 194, 193 210, 205, 206 216, 196, 207 184, 192, 220 204, 209	202.93	9.96
08	100	226, 241, 208 215, 215, 225 209, 201	217.50	12.69
09	200	230, 231, 223 218, 206, 233 246, 224, 216	225.22	13.52
12	400	195, 201, 204 207, 215, 224 221, 198, 196 197, 195, 201	204.50	10.20
02	600	125, 142	133.50	

typical pink granite of the area.

13.3.3 Thermomechanical Characterization of Rock

The objective of carrying out mechanical studies on the rock samples is to understand the behavior of the rock mass at normal and elevated pressures and temperatures. This would help in modeling fracture/joint systems and to evaluate the stability of the repository.

The study pertains to the development of fractures and extension and enlargement of existing fractures, when subjected to heat induced stresses. These investigations were carried out with an Acoustic Emission Monitoring System and a High Temperature-High Pressure Triaxial Cell.

Salient features of the laboratory studies carried out are as follows:

- Uniaxial compressive strength of granite, heated to 200°- 245° C is higher than unheated and heated to 100°, 400° and 600° C samples (Table 13.2);
- Heat treated granite and charnockite undergo more axial strain and less lateral strain compared to unheated samples;
- Young's Modulus decreases with increasing temperature (Table 13.3);
- Higher stress is required to generate acoustic emission events in case of granite heated to 200° C;
- Triaxial compression experiment carried out at a confining pressure of 30 MPa indicates that the average strength of granite at 200° C is higher than that at 100° C and 150° C, but lower than the unheated samples (Fig. 13.4). Sample heated to 200° C undergoes more axial deformation;
- Uniaxial compressive strength decreases with increase of water content in the samples; and
- Preliminary analysis of results of uniaxial compressive tests and acoustic emission experiments on heat treated rocks (heated up to 245° C) appears to indicate that microcracks strengthen the rock material like dislocations in metals.

In-situ thermomechanical experiments were carried out

Table 13.3. Effect of temperature on Young's modulus.

Heat Treatment Temperature (°C)	Average Young's Modulus (GPa)
Unheated	74
100	72
200	71
300	67
400	62

in two underground chambers at a depth of 1000 m in a mine facility to study the behavior of host rock subjected to decay heat generated by high level vitrified radioactive wastes. The waste containers were simulated by electrical heaters of the same dimensions. Thermocouples, vibrating wire stress meters and extensometers were installed in the bore holes within and around the array of heaters to measure temperatures, stresses and expansions in the rockmass. The main results of single and multi-heater experiments have shown that:

- The observed temperature profiles match with those of predicted ones in the majority of monitoring points. A typical profile is shown in Figure 13.5;
- The observed heat induced stresses are about half the values of the predicted stresses, the maximum value being 45 MPa; and
- The extension of the rockmass was found to be negligible.

13.4 CAPTIVE SITE EVALUATION PROGRAMME

Under this programme, a charnockite rock formation has been identified for designing an underground facility. The proposed site is presently being evaluated to assess its suitability with respect to geology, structural features, ground water conditions, and the thermal and mechanical integrity of the rockmass.

In this connection, the proposed site has been studied by geological and structural mapping, resistivity surveys, soil profiling and physico-chemical analysis. Currently, deep drilling operations at the site are in progress. Three bore holes of more than 600 m depth have already been drilled and a few more are planned. Core samples from them are being studied for mineralogy, structural homogeneity, geochemistry, and thermomechanical properties. Further evaluation will be carried out by geophysical logging techniques in the bore holes to have a better understanding of the subsurface features. The measurements of *in-situ* stresses and permeability in the bore holes are to be undertaken shortly. Cross-hole seismic tomography is also planned. The study of cores indicates that the rock is massive without many open joints and fractures. Long, intact core lengths up to 6.0 m (Fig. 13.6) have been recovered from the bore holes. The ground water aquifer is confined to shallow depths in soil and weathered zones. The water is saline to some

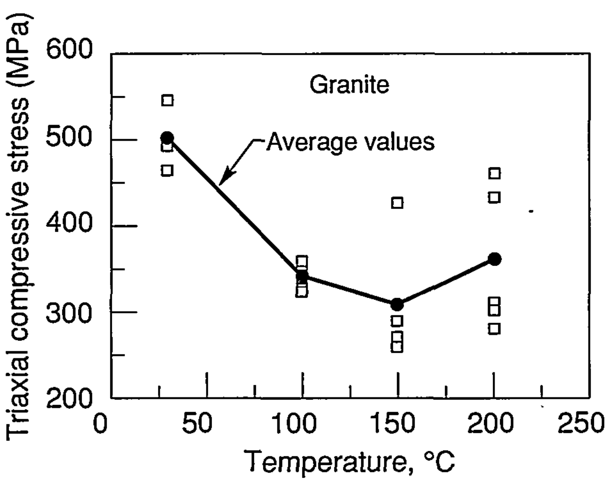

Figure 13.4. Triaxial compression test showing temperature-strength relationship.

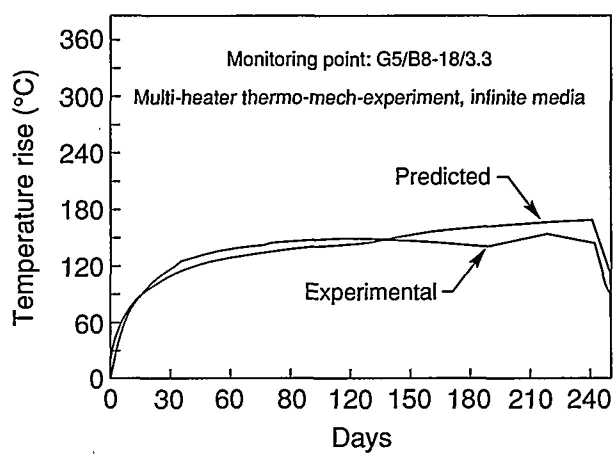

Figure 13.5. Typical profiles from *in-situ* thermomechanical experiment.

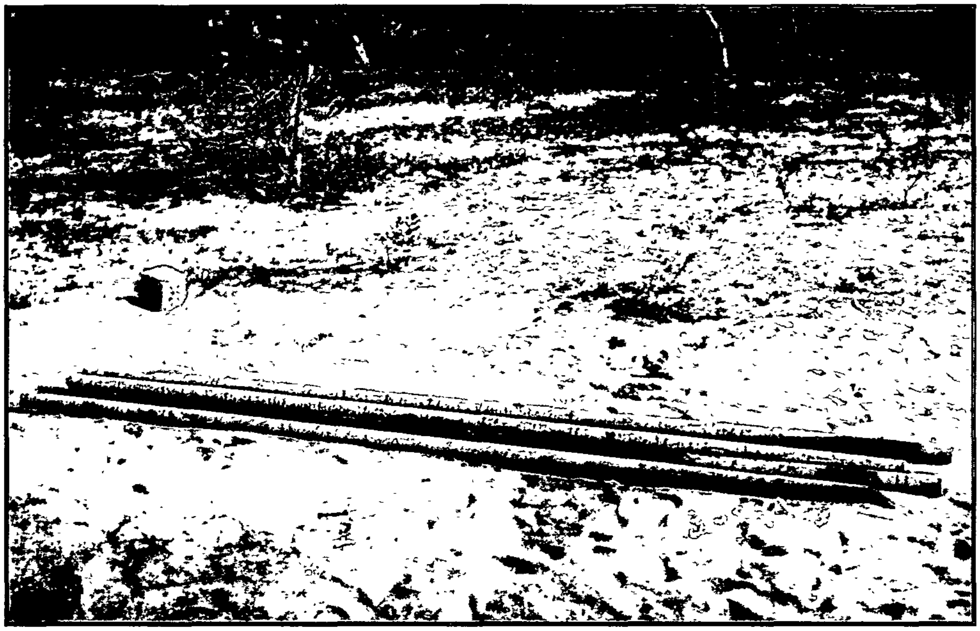

Figure 13.6. Intact long borehole core samples from charnockite formation.

extent, and the rock has good sorption properties.

13.5 CONCLUSION

The Indian programme of siting and host rock characterization for a geological repository in granite comprises selection of a site by the method of narrowing down the choice from larger areas to a specific site, on the one hand, and directly characterizing a known potential host rock formation within the captive area of a nuclear site, on the other.

CHAPTER 14

CRITICAL DATA REQUIRED TO POTENTIALLY INVESTIGATE GENTING ISLAND AS HIGH-LEVEL RADIOACTIVE WASTE REPOSITORY SITE FACILITY IN INDONESIA

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14.1 INTRODUCTION

The Indonesian archipelago is one of the regions in the world that has active volcanisms. There are 129 active volcanoes in the region. The Indonesian archipelago is developing in response to the complex interaction between the southward moving Eurasian plate, the northward moving Indian-Australian plate and the westward-moving Pacific plate (Fig.14.1). The Java trench and Timor Trough represent the major area of collision between Eurasian and Indian-Australian plates. The Sorong fault indicates the area of interaction between the Pacific and Indian-Australian plate[1].

Indonesia has been actively planning to build several NPPs in the near future, despite the concerns about the existing volcanism. The operation of these NPPs will generate waste, namely high-level radioactive wastes (HLRW). Some islands in the country have been investigated and selected as potential HLRW repository sites.

Genting island is one of these islands with a potential site for a HLRW repository site facility. A wet environment repository concept must be developed since the proposed island is an ocean-island, and the groundwater table is shallow such that the wastes will have to be emplaced in the saturated zone.

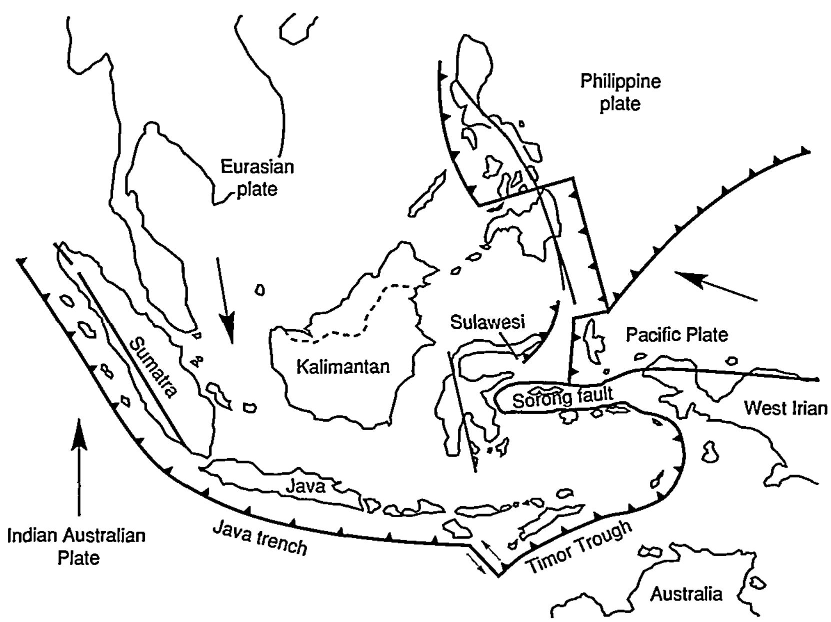

Figure 14.1. Plate boundaries in the Indonesian Archipelago.

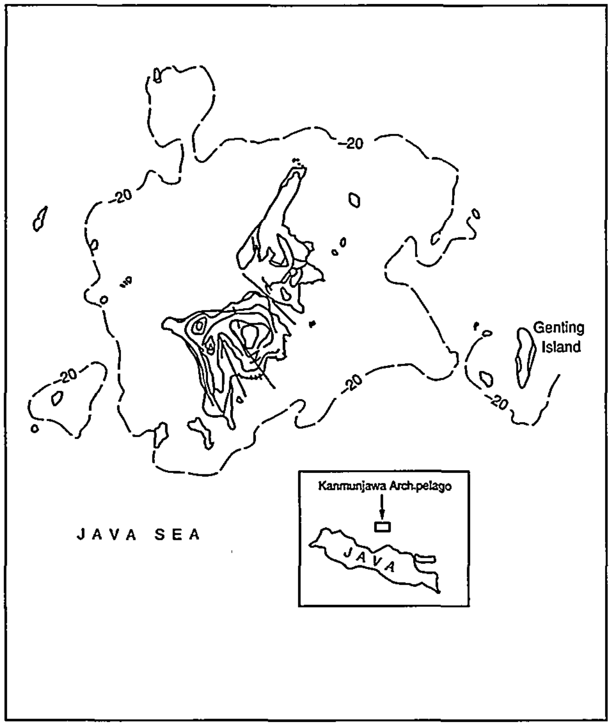

Figure 14.2. Map of Karimunjawa Archipelago showing location of Genting Island. The long-dashed line shows the location where the water depth is 20 m.

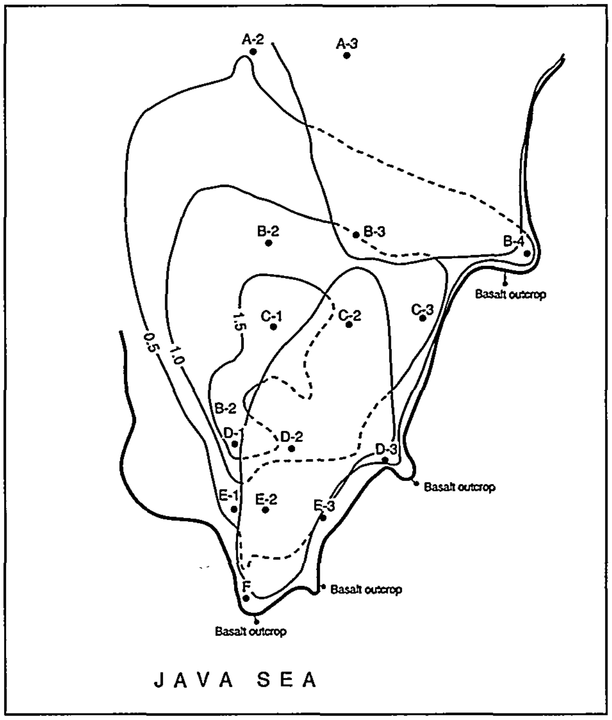

Figure 14.3. Map of groundwater levels for southern tip of Genting Island in meters above sea level. The solid lines indicate water levels that are above the land surface, and the dashed lines indicate levels that are below the land surface.

14.2 General Description of the Island

Genting island is a small island situated on the north side of Java island on the Eurasian plate (Fig. 14.2)[2]. The distance from the Java trench is approximately 400 km. This island is nearly uninhabited and there is no potential economic activity. The groundwater level at Genting Island is considered shallow, and it is influenced by changes in sea level and the rate of rainfall (Fig. 14.3)[2].

14.2.1 The Lithology of the Area

The top layer (thickness ±1.5 - 3.5 m) of soil is mainly alluvium consisting of pebble, gravel, clay, coral limestone and coarse grained rocks. Below this layer is basalt (thickness ±24 - 35 m) consisting of basaltic lava or alkaline basalt, classified as a strong rock (approximate strength is 1550.36 kg/m^2 in compression)[2]. Its strength and the interlocking of fracture blocks can limit displacement along fractures. The diffusion time of radionuclides along the rock fractures can be delayed, so that it will eventually take longer for the radionuclides to reach the accessible environment (AE). The depth to groundwater is approximately 103 m. The permeability measurements of Genting Island are shown in Table 14.1.[2]

Table 14.1. Permeability measurements for Genting Island.

Coefficient of Permeability (m/s)	Type of Rocks
1.82×10^{-6}	Soil and
1.75×10^{-6}	Basalt
1.26×10^{-7}	Basalt
5.55×10^{-6}	Basalt
4.79×10^{-6}	Basalt

The water chemistry of Genting Island is such that the concentrations of Mg^{2+}, Na^{+}, and Cl^{-} are rather high in the coastal area. The content of HCO_3^- tends to increase on the southern side of the Island. The existence of

HCO_3^- in the groundwater is due to the influence of decomposed plants and swamp materials.

The southern side of the Genting Island is primarily a volcanic cone region. The highest point of this area is 40.5 m above mean sea level, and the lowest point is about 5.0 m above mean sea level.

From geotechnical investigations, the unconfined compression strength is 360.86 kg/cm^2 . The mean value of Poissons ratio is 0.31. The mean value of the rock density is 2.797 g/cm^3, and the mean value of cohesiveness is 57.87 kg/cm^2.

14.2.2 Near-Field Conditions

In the model, pH, water contact mode, and temperature are included, where pH (at the 50 m depth in an experimental borehole), varies from 7.7 to 6.8. The average pH is 7.25, which can be considered neutral. The water contact mode, in the saturated zone can be thought of as zero velocity, which indicates diffusive transport, because Genting Island is an ocean-island[3,4,5].

14.3 Design of the Repository Site

The proposed design for the HLRW repository site at the southern tip of Genting Island is divided into two areas or rings. The inner ring (called the high-temperature ring) that contains a group of waste packages withan areal power density of approximately 100 kw/acre, which includes 75% of the waste packages. The outer ring contains a group of waste packages with an areal power density of approximately 30 kw/acre, which includes 25% of the waste packages, and represents the ambient-temperature ring[3].

14.4 Pathway Parameters

For an ocean-island repository that is sited below the groundwater table, the most important pathway for radionuclide releases is through the groundwater. The radionuclides released through the rock will eventually reach the groundwater. In the groundwater, the radionuclides will travel in a dffusive manner to the AE. Therefore, the groundwater is expected to be the primary agent affecting the performance of the Genting Island repository. In addition to being the transport mechanism for the radionuclides, the groundwater will also corrode the waste containers when it comes in contact with the containers[6]. The shallow groundwater is assumed to exist under reducing conditions. Consequently, it is also necessary to consider solubility under reducing conditions.

In regard to gaseous flow and transport, this study suggests that C-14 will not be released in any significant quantity.[3,4,5] However a more accurate analysis should be conducted. The gas flows are driven by heat and in turn affect waste package temperatures. A coupled transient model of heat transfer and gas flow employing a relatively fine grid will be required.

The ocean dilution factor plays a very important role in reducing the concentration of radionuclides released to the AE. Their concentrations become essentially negligible when this factor is incorporated in the analyses. Therefore, it is necessary to employ a more accurate model to calculate the dilution factor[7]. However, the ocean will not be considered as the source of drinking water; only the biota in the ocean should be analyzed as the potential transport pathway in future studies[5].

14.5 Summary and Conclusions

In addition to the problem of pathways for radionuclide releases, earthquakes play an important role in the repository integrity. When the rate of occurrence was selected to be 1×10^{-7}, no effects are shown in the results[3]. But, when the rate of occurrence was taken to be 1×10^{-2}, some significant effects were seen. Therefore, further study regarding seismic analyses of the site must be undertaken[8].

Understanding groundwater flow characteristics is essential when attempting to predict repository performance more accurately[9]. These characteristics are required, especially parameters of the directions of flow and the flow rate. Furthemore, if there are releases of radiation in the near-field environment, the possible degradation of the rocks in the buffer zone should also be further investigated[10].

References

1. Katili J.A., Magmatic affinities of volcanic rocks of Ungaran, Central Java, Journal Geologi Indonesia, The Journal of the Indonesian Association of Geologists, Volume 60, Jakarta, Indonesia, 1989.

2. Faculty of Mineral Technology, Institute of Technology - Bandung, Further Geological and Hydrogeological Investigation at Genting Island as Proposed Location for Permanent Radioactive Waste

Repository, National Atomic Energy Agency (BATAN), Jakarta, Indonesia, 1990.

3. Imardjoko Y.U., Total system performance assessment of the proposed high level radioactive waste repository at Genting Island, Karimunjawa, Indonesia, Ph.D. Dissertation, Iowa State University, Ames, IA, July, 1995.

4. Imardjoko Y.U., D.B. Bullen, and S. Yatim, Performance assessment modeling of the proposed high-level radioactive waste disposal facility at Genting Island, Karimunjawa, Indonesia, Journal Manusia dan Lingkungan, PPLH-UGM, Indonesia, December 1995.

5. Imardjoko Y.U., D.B. Bullen, and S. Yatim, Performance assessment modeling of the proposed Genting Island repository facility, Proceedings, International High Level Radioactive Waste Management Conference, ANS-ASCE, Las Vegas, pp. 172-175, NV, April, 1996.

6. EPRI, Geological disposal of nuclear waste, Electrical Power Research Institute Journal, Palo Alto, CA, May, 1982.

7. Forsberg C.W., An ocean-island geologic repository - A second - generation option for disposal of spent fuel and high - level waste, Nuclear Technology, Vol. 101, January, 1993.

8. Wallmann P.C., I. Miller, and R. Kossik, Assessment of volcanic and tectonic hazards to high level radioactive waste repositories, High Level Radioactive Waste Management - Proceedings of the Fourth Annual International Conference, Las Vegas, Nevada, Vol. 1, pp.188-195, April 26 - 30,1993, ANS, La Grange Park, IL, 1993.

9. Ahoka H., and F. Ky, Roles of fracture zones in controlling hydraulic head and groundwater flow - experience from site characterization program in Finland, High Level Radioactive Waste Management - Proceedings of the Fourth Annual International Conference, Las Vegas, Nevada, Vol. 1, pp. 431-436, April 26 - 30,1993, ANS, La Grange Park, IL, 1993.

10. Dershowitz, W.S., P.C. Walimann, T.W. Doe, and J. Geier, Discrete feature modeling at the Stripa mine in Sweden: Significance for hydrologic modeling of fractured rock masses, High Level Radioactive Waste Management - Proceedings of the Fourth Annual International Conference, Las Vegas, Nevada, Vol. 1, pp. 443-450, April 26 -30,1993, ANS, La Grange Park, IL, 1993.

CHAPTER 15

GEOLOGICAL DISPOSAL DEEP UNDERGROUND A STUDY OF THE JAPANESE GEOLOGICAL ENVIRONMENT AND ITS STABILITY

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15.1 INTRODUCTION

In order to demonstrate the scientific and technological safety of geological disposal in Japan, it is important to make an accurate assessment of the geological environment and to incorporate this knowledge into the performance assessment and R & D of disposal technology based on a realistic model of this environment. At present, surveys and studies are being conducted without specification of regions or rock types.

Geoscientific knowledge to date indicates that the geological environment deep underground is characterized by generic qualities found in all regions and other qualities, which are peculiar to a specific region. For an effective assessment of the Japanese geological environment, it is important to make a systematic collection of data on specific and generic regional characteristics that are relevant to geological disposal.

Based on their origin, rocks can be divided roughly into three groups, igneous, metamorphic, and sedimentary. Within these groups, various subdivisions are categorized by mineral/chemical components, grain size and texture, etc. However, from the viewpoint of groundwater flow and mass transport, which are pertinent to the performance assessment of geological disposal, the present characteristics of rocks have a greater significance than those at the time of their formation. These characteristics include physical and chemical qualities, the hydraulic structure of rocks deep underground, and the chemical properties of rock groundwater systems, such as the extent and rate of nuclide sorption. Based on this approach, it is possible to classify the rock formations in Japan into two groups: crystalline rocks (fractured media) and sedimentary rocks (porous media). Granitic and sedimentary rocks of the Neogene are widely distributed throughout Japan and can be taken as being representative of the two groups defined above.

The Power Reactor and Nuclear Fuel Development Corporation (PNC) is promoting geological research with emphasis on present-day conditions. A new research division has been established to perform surveys and studies of the deep geological environment as a basis for research and development (R & D) programs on geological disposal.

This report outlines the R & D activities conducted so far, as well as future plans.

15.2 PURPOSE AND PROCEDURES OF RESEARCH AND DEVELOPMENT PROGRAMS

The purpose of geological research is to characterize the geological environment of Japan from the point of view of geological disposal, to construct models of geological structures and groundwater flow, to systematize available data, to make accurate and efficient assessments of the geological environment, and to develop practical technologies for analysis and evaluation purposes. The models and data obtained through such studies are applied in performance assessment and R & D programs.

Data on the flow and geochemical characteristics of groundwater, as well as on mass transport, are important in assessing the performance of the near-field and far-field in a geological disposal system. In near-field studies, it is essential to obtain accurate data on the hydraulic and geochemical characteristics of the bedrock adjacent to the engineered barrier system, including the excavation disturbed zone. For the far-field, on the other hand, data are required on the groundwater flow and mass transport characteristics of fracture and alteration zones over a wide region and on hydraulic

and chemical properties at the boundary between seawater and freshwater in coastal regions.

Since data on the deep geological environment must be as precise and reliable as possible from the viewpoint of geological disposal, it is therefore necessary to develop and improve the technologies used in surveys and measurements to ensure more efficient and detailed data acquisition. Equipment is being developed that will allow surveys of bedrock with very low permeability or under the high pressure and temperature conditions prevailing deep underground in order to advance studies of hydraulic and geochemical characteristics. At the same time, attention is being given to methods which are disturbance-free, i.e. they do not cause damage or upset the natural condition of the rock. For this purpose, efforts are being directed toward improving techniques of physical and drilling surveys as well as integrating various survey techniques.

A given geological environment is considered stable for the purpose of geological disposal if the formation, selected as being most appropriate, can maintain its required role for safe disposal despite the potential for changes in the environment. In order to assess the stability of a geological formation, it is necessary to ascertain potential changes in the environment brought about by various phenomena, as well as the extent of these changes. Predictive studies require analysis of data related to the occurrence and regularity of natural phenomena (extent, region, regularity, mechanism of occurrence, etc.) and identification of a pattern of regularity, if any.

Studies on the geological environment include compiling accumulated data and the most recent findings in related fields of science and engineering and applying the results of geological R & D at the Tono and Kamaishi mines. Cooperative studies with other countries advanced in the field of geological disposal also contribute to establishing investigation techniques and procedures for performance assessment.

15.3 Characteristics of Geological Environment and Current State of Knowledge

Deep geological formations are generally characterized by extremely slow groundwater flow and a reducing chemical environment of neutral to slightly alkaline nature. As a result, it is unlikely that radionuclides within the waste will be significantly leached out; even if the waste matrix does dissolve, the likelihood is that the escaping nuclides will either be sorbed onto clay minerals in the adjacent bedrock and fractures or will be precipitated. It is expected, therefore, that the migration of radionuclides will be even slower than the rate of groundwater flow.

Tests conducted on sedimentary rocks in the Tono region of Gifu Prefecture indicated the hydraulic conductivity to be approximately 10^{-8} m/s to 10^{-10} m/s and the oxidation-reduction potential to be low at -300 mV for groundwater at 160 m depth. In the case of the Tono uranium deposit, one characteristic of the geological environment which is evident is that no major migration of uranium has occurred in the past 10^5 years.

The deep geosphere also has the characteristic, compared to the ground surface, of not being readily affected by natural phenomena such as earthquakes, glaciation, weathering and erosion, or by human activities.

Earthquake observation in the gallery of the Kamaishi Mine indicated that the ground acceleration rate measured several hundred meters underground was approximately half that measured at the ground surface. Long-term observation of the pore pressure and chemical properties of groundwater has revealed no major change so far. Temporary changes are recorded at the time of an earthquake, but these are within the range of seasonal fluctuations (Shimizu, et. al., in press).

One of the natural phenomena considered to have a potential influence on the stability of the geological environment is the fault activity assumed to have occurred during the Quaternary. A survey of fault distributions indicates that there is a wide region without any faulting and, even within regions where numerous faults are found, there are blocks of rock where no faults exist. Igneous activity occurring in the Japanese archipelago is likely to be related to the location and depth of plate subduction in the vicinity, but there has been no major change in the past 12 million years. Periods of glaciation and sea-level changes are global phenomena which have been repeating in cycles of approximately 10^5 years for the past 7×10^5 years. An accurate understanding of regional characteristics, regularities and cycles in natural phenomena, allows the stability of the geological environment to be assessed reliably.

15.4 Present Research and Development

Of the geological research and development currently

underway, a report is presented here on the present studies of geological formations and on R & D procedures relating to assessing long-term safety.

15.4.1 Investigations Conducted at and Around the Tono Mine

The Tono Mine is a uranium mine situated in the Tono region of Gifu Prefecture. The region around the Tono Mine is formed basically of granite, covered with Neogene sedimentary rock (Fig. 15.1). In this region, an enormous amount of geological information is being accumulated through uranium prospecting and academic research. The shaft and gallery leading to the mine allow access to sedimentary rocks, including uranium deposits, over one hundred meters underground. Surveys conducted in this region include the hydrology and geochemistry of groundwater, mass transport by groundwater and the effect of excavating galleries on the geological environment (Yusa, et. al., 1992).

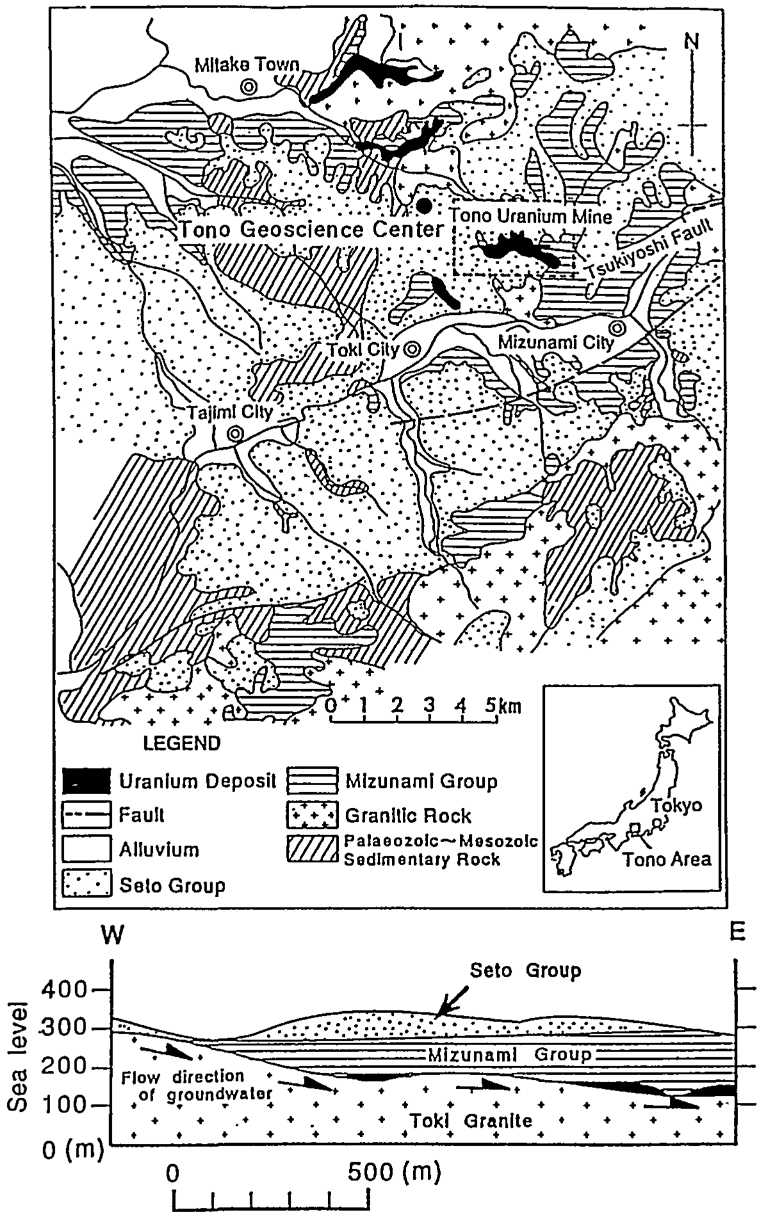

Figure 15.1. Geological and Location Maps of the Tono Mine Region.

The hydrological research being conducted at present includes accumulation of reliable hydraulic data down to a depth of 500 m below ground surface, construction of a hydrogeological model based on survey results, consideration of methods for predicting groundwater flow and evaluation and verification of the groundwater flow model (Fig. 15.2). The results of permeability investigations conducted on Neogene sedimentary rocks and granites in the Tono region indicated that the hydraulic conductivity of sedimentary rocks is higher for coarse-grained formations. The value was approximately on the order of 10^{-8} m/s for medium to coarse sandstones. The hydraulic conductivity of granites is classified into two categories of high permeability (10^{-5} to 10^{-6} m/s) measured in the vicinity of open fractures and low permeability (10^{-8} to 10^{-11} m/s) measured in the unfractured rock mass and in fracture zones filled with clay minerals.

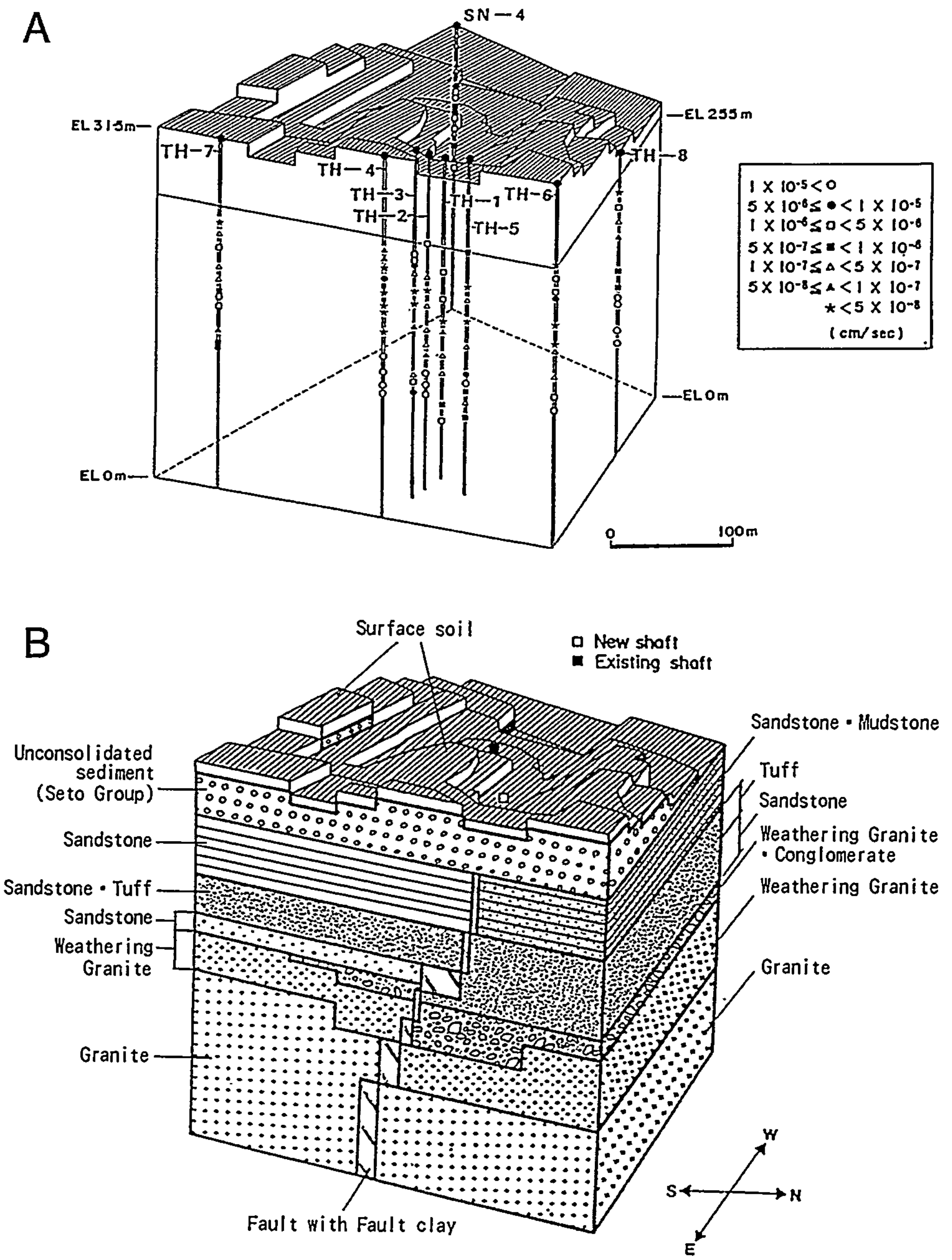

Figure 15.2 A - Hydraulic conductivity distribution, and B - hydrogeological model for the Tono region.

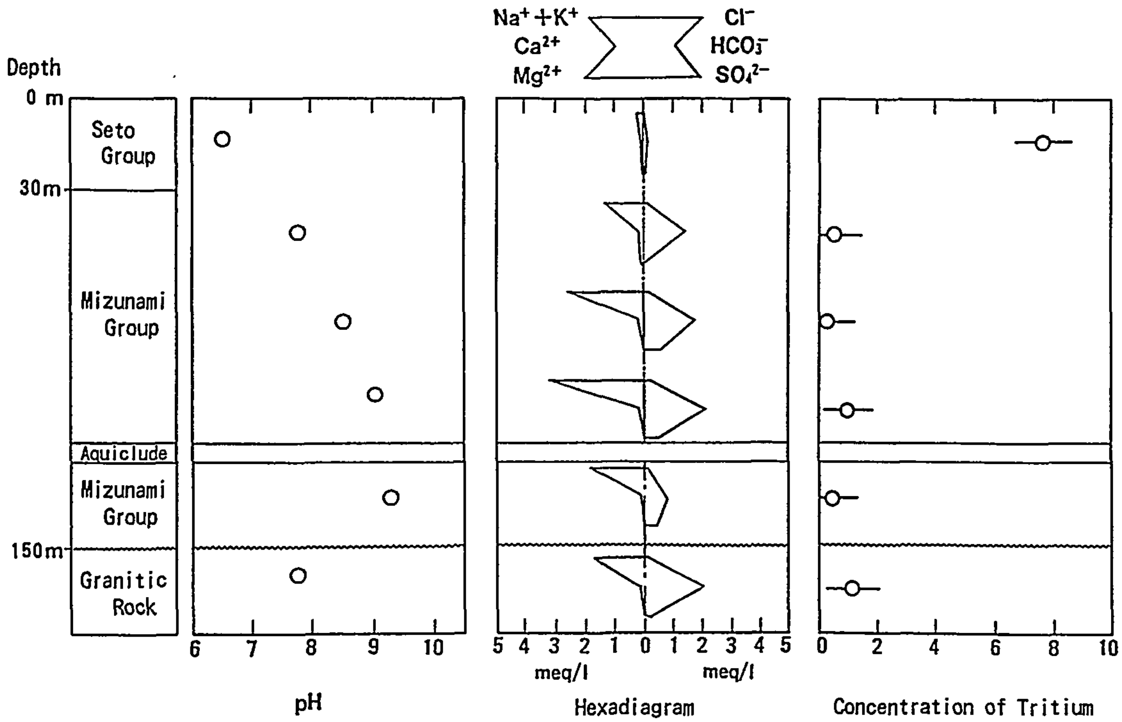

Figure 15.3. Groundwater pH, chemistry and tritium concentration for the Tono region (Major solution components are presented on a hexadiagram.

For a three-dimensional analysis of groundwater flow taking the example of the Tono region, piezometric heads are almost hydrostatic except near the surface.The hydraulic gradient is below 0.04 for areas deeper than 500 m underground. The procedure for future research on groundwater flow will be to obtain hydraulic data for depths down to 1,000 m underground, to improve methods for analyzing groundwater flow and to develop suitable investigation methods for a given region.

Geochemical studies aim to determine the age and origin of groundwater and the distribution of geochemical characteristics and to confirm the relevance of the model of the geochemical evolution of groundwater. For this purpose, sampling and analysis of groundwaters from boreholes is being carried out. Groundwaters in sedimentary rocks are mostly the Na-HCO_3 type, and the pH value moves from neutral to alkaline as depth increases (Fig 15.3). A comparative analysis of oxygen and hydrogen stable isotopes confirms the theory of precipitation as the origin of the groundwater.

Radiocarbon dating reveals the groundwater at the base of the sedimentary formations to be at least in excess of 104 years old. The figures for pH, redox potential and concentration of chemical components obtained from a geochemical equilibrium model and from actual measurements in the region agree well with one another. Future tasks are to accumulate geochemical data down to a depth of 1,000 m and to study the geochemical evolution of deep groundwaters, also for granitic rocks (Yamakawa, 1991; Yoshida, et. al., 1994).

To examine solute migration in the deep geological environment, the Tono uranium deposit and its surrounding area are being studied as a natural analogue for migration and immobilization of material within a geological formation. Studies are underway on the environmental conditions for generation and preservation of uranium deposits and on the migration and retardation of natural series nuclides by groundwater flow. Studies so far show that uranium is immobilized in a reducing environment, that the migration and concentration of uranium depends not only on the mineralogy and chemistry of the rocks but also on groundwater flow, and that the uranium found in the ore deposit has not migrated significantly over the past 10^5 years. Future studies should clarify the route of groundwater flow and its relationship with migration of natural series nuclides, and also quantify migration and advance modeling of sorption phenomena.

To ascertain the influence of excavation on the rock formations surrounding a gallery, a series of studies are

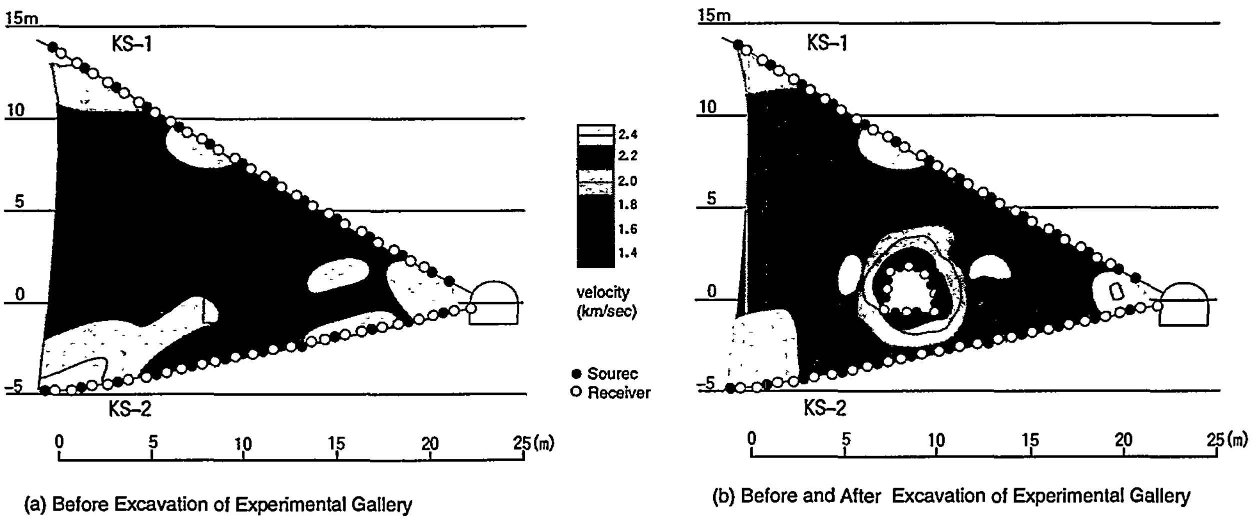

Figure 15.4. Experiment on excavation effects at the Tono Mine.

being conducted in which the original condition of the bedrock is assessed before excavation, followed by prediction of the effect of excavation and, finally, by actual measurement of the effects of excavation (Fig. 15.4). In a study of excavation effects, a shaft with a 6 m inner diameter and 150 m deep has been observed for approximately the past four years. Observations indicate that the "excavation disturbed zone" created by blasting the bedrock extended about 1 m from the shaft wall, while changes in pore pressure extended to a radius of approximately 100 m from the shaft. The next step is to ascertain the effects on different rock types and of varying methods of excavation, as well as to improve the techniques employed in measurement and analysis. It is also necessary to continue monitoring over an extensive period in order to understand the long-term impact on bedrock deformation and hydrology, etc (Sato, et. al., 1995).

15.4.2 Surveys Conducted at Kamaishi Mine

The Kamaishi Mine is an iron and copper mine with granite as the parent rock; it is situated near Kamaishi City in Iwate Prefecture. PNC has been conducting investigations in the Kurihashi granitic diorite since 1988, in a gallery located 550 m above sea level (approximately 300 m below ground surface). In 1993, a second phase was launched for a five-year period in a gallery 250 m above sea level (approximately 700 m below ground surface). The research includes assessment of geological characteristics deep underground, appraisal of the extent of the excavation disturbed zone in the bedrock, hydraulic and migration tests in crystalline rock, testing of engineered barriers and seismic surveys (Fig. 15.5) (Takeda and Osawa, 1993).

In the first five years, the aim was to collect data on the distribution of various characteristics and phenomena occurring in the geological environment. At the same time, the appropriateness of investigation techniques was also tested. Surveys conducted included investigations of fractures in gallery walls and boreholes, physical surveys to ascertain the distribution of fractures, hydrogeological research to assess the permeability of the bedrock, groundwater flow and its modeling, geochemical research to determine the origin, age, chemical evolution, etc., of groundwater, bedrock dynamics to test the effects of gallery excavation, seismic research, and research on engineered barriers.

These studies contribute to improving the understanding of physical properties such as the strength of the Kurihashi granitic diorite, initial stress conditions, the permeability of the bedrock, channeling phenomena whereby groundwater tends to flow along distinct pathways within a fracture, geochemical properties of groundwater deep underground and *in-situ* swelling of bentonite clay as a backfill in bedrock. Studies conducted on the effect of gallery excavation on the geological environment showed that any change in the permeability and deformation of the bedrock occurred within a radius of 1 m from the gallery wall.

Through these studies, it is possible to ascertain the effectiveness of techniques for modeling groundwater flow near the gallery and the deformation behavior of

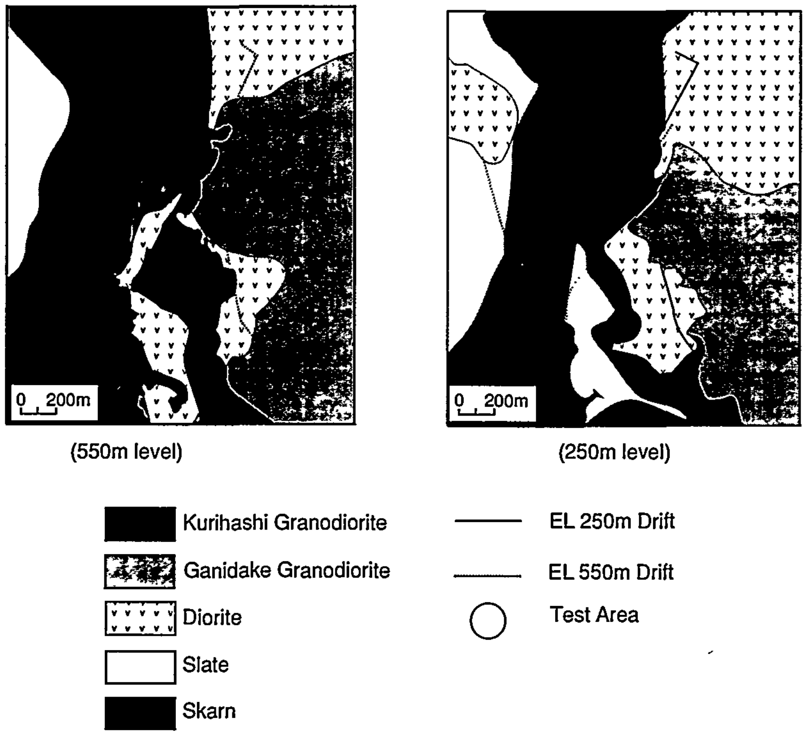

Figure 15.5. Geological map of the *in-situ* experimental site at the Kamaishi Mine.

adjacent bedrock due to the gallery excavation. They also allow validation of radar and resistivity tomography methodology, the derived model of fracture distribution and the technique used for permeability determination.

Seismometers were placed in galleries at different depths to observe seismic-related characteristics; a continuous analysis of pore pressure and the chemical properties of groundwater was also carried out. The results indicate that, for the majority of earthquakes, the rate of acceleration several hundred meters below surface is about half that at the ground surface (Fig. 15.6). Long-term observation of pore pressure and chemical properties of groundwater showed a temporary change at the time of earthquakes, but this was within the range of seasonal fluctuations.

In 1993, a new five-year project was launched to investigate the different characteristics of the deep geological environment and to obtain a more detailed understanding of the range of effects caused by gallery excavation. The work covers geochemical changes occurring within nearby bedrock, the distribution of fractures in the bedrock and galleries, solute migration and water flow, thermal, hydraulic and dynamic interactions between bentonite clay and surrounding bedrock and groundwater, the impact of earthquakes on groundwater flow, and the decrease of earthquake motion with depth.

So far, the studies have shed light on the initial stress conditions at various depths (galleries at 250 m level and 550 m level), the distribution of fractures and microcracks from the viewpoint of mass transport and the redox conditions of groundwater in the vicinity of galleries.

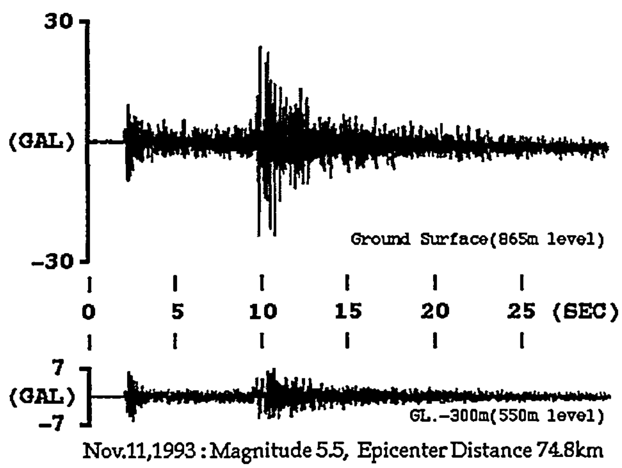

Figure 15.6. Seismic observations at ground surface and underground at the Kamaishi Mine.

15.4.3 Development of Techniques for Investigating Deep Geological Environment

In order to make quantitative measurements of hydraulic parameters deep underground, a hydraulic measurement device designed for conditions of slow water pressure build-up has been developed for application in low permeability bedrock. An *in-situ* measurement device has already been constructed for use at depths down to 500 m and in the order of 10^{-11} m/s for the hydraulic conductivity. Other equipment developed allows a continuous *in-situ* measurement of parameters such as redox potential, pH and dissolved oxygen concentration. With the development of a packer-system in a pressured groundwater sampling device, which is gas-tight, it is now possible to obtain groundwater samples without disturbing their original condition within the rock formation. A new device was completed in 1994; it can be used down to a depth of 1,000 m and is in the process of being tested. Efforts have also been devoted to developing an evapotranspiration device for measuring the distribution of influx volumes from gallery walls and a sinusoidal, crosshole experimental device for use in a single fracture near the gallery wall.

When gathering data in the deep geological environment, it is desirable to do so without disturbing the natural conditions, so as to fully preserve the fractures within the bedrock, the geological structure and hydraulic characteristics. As a consequence, the development and testing of resistivity and seismic tomography are presently underway. Furthermore, a radar technique is being developed, which uses multiple boreholes in order to determine precise fracture zone locations and their extent.

It is also important to improve surface-based methods for predicting conditions in the deep geological environment; the results can then be confirmed using data obtained after excavation of galleries. Research in this area is being carried out jointly by SKB and PNC.

15.4.4 Research on the Long-Term Stability of the Geological Environment

In the first technological report (PNC's Heisei 3 Report), natural phenomena with a potential influence on the geological environment were summarized, suggesting the existence of regionality and regularity in these phenomena. They include earthquake and fault activity, volcanic activity, uplift and denudation, climatic fluctuation and sea-level change. In view of the importance of understanding and predicting the influence of these phenomena on the geological environment, surveys are being conducted of relevant literature published in Japan and abroad (Shimizu, et. al., 1992).

Literature surveys and analyses are performed to first clarify the history of individual natural phenomena. The next step is to conduct research on the nature and scale of the impact of such phenomena on the geological environment and to establish a method for defining a realistic range of fluctuation for the predicted influence of these phenomena.

Regarding fault activity, attention is focused on the possibility of the formation of new faults, by observing the fluctuations in time and space and the locations of fault activity. A method for ascertaining the hydraulic and geochemical fluctuations in groundwater is being considered in order to provide an understanding of the range of influence of fault activity on the nearby geological environment.

To predict the range of variation in uplift and subsidence, it is necessary to have accurate knowledge of past data and an understanding of crustal movements in Japan; a systematic compilation is therefore being made of relevant geomorphological data. To estimate topographic and decreases in the thickness of geological formations, which accompany uplift and subsidence, modeling of denudation and future topographic changes is also necessary.

As for volcanic activity, the characteristics and regionality of Quaternary volcanic activity should be ascertained; the locations of activity as well as the relationship between geological structure, stress fields and plate distribution should also be clarified in order to substantiate the assumption that the region of volcanic activity will not change in the future. Studies should be able to determine the range and scale of thermal impact and its influence on the geochemical characteristics and flow of groundwater.

Studies should be carried out to assess the scale and regularity of climatic and sea-level changes on a global scale.

15.5 CONCLUSIONS

Any discussion of the geological environment of Japan within the context of geological disposal programs requires the development of transparent and logical ana-

lytical methods and a systematic accumulation of reliable data. It is also important to make maximum use of existing information. Advances in a new field of research and development, as in the case of this report, should be pursued with the support of public understanding and participation, from the beginning, of experts from all related areas of study; this ensures transparency and steady progress. By way of recognizing the importance of obtaining the consensus of experts from various fields on the overall progress and direction of R & D programs, as well as on individual studies, projects are pursued with constant referral to various commissions.

Regarding sites for research on the deep geological environment, the Atomic Energy Commission of Japan has indicated the desirability of having multiple facilities in view of the wide range of geological features in Japan; this is stated in the "Long-Term Program for Research, Development and Utilization of Nuclear Energy." There is also a clear distinction between plans for an underground research facility and plans for geological disposal. PNC is also aware of the importance of carrying out research, not only in existing galleries, but also in geological environments with undisturbed conditions. It is also important to verify predictive surveys made from the ground surface, following actual excavation of galleries. A research facility which would allow such activities to be pursued is of utmost importance and PNC is devoting considerable efforts toward construction of such a facility as soon as possible.

REFERENCES

Sato, T., K.Sugihara, and H. Matsui, Geoscientific studies at the Tono mine and the Kamaishi mine in Japan, Proc. of 8th International Congress on Rock mechanics, Vol.1, pp. 47-51, 1995.

Shimizu, H., K. Ishimaru, K. Furuya, and Y. Yusa, Natural processes and events relevant to long-term stability of geological environment in Japan, Proc. of the Workshop WC-1 "Waste Disposal and Geology Scientific Perspectives", 29th Internatioal Geological Congress, pp.381-394, 1992.

Shimizu, I., H. Osawa, T. Seo, S. Yasuike, and S. Sasaki, Earthquake related ground motion and groundwater pressure change at the Kamaishi Mine, Engineering Geology Special Publication (in press, 1996).

Takeda, S. and H. Osawa, Current status and future program of in-situ experiments of Kamaishi, Japan, Proc. International Symposium on In-Situ Experiments of Kamaishi, Japan, Nov.11-12, 1993.

Yamakawa, M., Geochemical behavior of natural uranium-series nuclides in geological formation, Proc. Third International Symposium on Advanced Nuclear Energy, pp.150-158, 1991.

Yoshida, K. et al., Data Compilation of Geoscientific Studies of Tono Uranium Deposits, Central Japan, PNC TN 7410 94-15, 1994.

Yusa, Y., K. Ishimaru, K. Ota, and K. Umeda, Geological and geochemical indicators of paleohydrogeology in Tono uranium deposits, Japan, Proc. Workshop in Paleohydrogeological Methods and Their Applications for Radioactive Waste Disposal, OECD/NEA, Paris, 9-10 Nov., 1992.

CHAPTER 16

GENERIC PERFORMANCE AND ENVIRONMENTAL ASSESSMENT OF A RADIOACTIVE WASTE REPOSITORY IN KOREA

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Abstract. The Nuclear Environment Management Center (NEMAC), a subsidiary of the Korea Atomic Energy Research Institute (KAERI), has conducted a performance and environmental assessment of the proposed Korean radioactive waste repository. In this paper, a description is provided of the performance and environmental assessment of a generic Korean radioactive waste repository. A description is given of the various data needed in post-closure performance assessment calculations. The data are divided into four major categories: inventory and design data, chemical data, hydrogeological data, and biosphere data. The results suggest that a generic performance assessment of the conceptual repository may be acceptable when measured against the regulatory criteria. The results indicate, however, that the performance will be strongly dependent on the geology and hydrogeology of the repository location, and it should be noted that the assessment was carried out on the basis of a limited understanding of the site-specific characteristics. It will be necessary to obtain site-specific data at the intended repository location in order to enable a more detailed assessment to be undertaken. This preliminary assessment of the conceptual repository provides a firm foundation for future site-specific assessment activities.

16.1 INTRODUCTION

The radioactive waste management program in Korea dates back to mid-eighties when KAERI perceived, from prior studies, the necessity for a national program for a comprehensive and systematic management of radioactive wastes including spent fuel arising from this country's ambitious nuclear power plant program. The recommendation therefrom was taken in consideration by the Korea Atomic Energy Commission which made a decision at its 221st meeting in 1988 to establish a relevant program to be implemented. Institutional arrangement for this program was that the required waste fund be levied on waste generators by the "polluter pays" principle. Initial works of the program had been implemented, including conceptual design of a low-level radioactive waste repository and the development of transport casks, until the whole program has been seriously hindered by the difficulty of site acquisition.

The major difficulty in site acquisition came from opposition of local communities at potential sites. The bad image of waste burial, combined with fear of potential nuclear danger, seems to have made the local communities abhorrent to any attempt to access the sites as culminated by the Anmyon Island incident in late 1990, where a series of demonstrations by local residents influenced the government to cancel the nomination of the site. All the efforts of NEMAC and the government to convince the local communities of potential sites have failed in a social mood overwhelmed by the NIMBY syndrome and an anti-nuclear movement. At the end of 1994, the government nominated Guleop Island, with only nine residents, as the candidate site for the repository and later finalized it as the official site. However, even after the governmental announcement, a series of demonstrations by the residents of Dukjuk Island, mother island of the smaller Guleop Island, acted as big obstacle against the government and NEMAC. The final blow against the Guleop Island Project was made at the end of 1995 by the confirmation of active fault zones, near and on the island, so that the government had to announce the cancellation. After a decade of unsuccessful efforts in search of a site, the government decided to amend the institutional approach to the problem.

In this paper, the groundwater pathway results of a performance and environmental assessment study[1] of the conceptual Korean radioactive waste repository, carried out jointly by NEMAC and AEA Technology from 1994

to 1995, are summarized.

In Section 16.2, a description is given of various data used in the assessment calculations. In Section 16.3, a description is given of the assessment calculations undertaken to address radionuclide transport along the groundwater pathway; this was the major component of the post-closure performance assessment. However, the results of the assessment carried out to address the effects of gas generation and migration on the performance of the proposed Korean repository and the calculations carried out to address the radiological risks arising from inadvertent human intrusion into the proposed repository are not included in this paper.

16.2 PREPARATION OF ASSESSMENT DATA

16.2.1 Design and Inventory Data

Radioactive wastes requiring treatment and disposal in Korea are dominantly produced by operation of the civil nuclear power reactors. These are of the PWR and CANDU types, and produce a range of wastes in the low- and intermediate-level categories. Spent fuel is not considered at this time.

The levels of radioactive contamination vary widely, and a proportion of each of the first three waste types appears in both the LLW and ILW categories. The choice of waste treatment technologies reflects this variation.

The liquid concentrates come from the treatment of water discharged from the primary reactor circuit, the secondary heat transfer circuit, and the spent fuel storage pools. The water is treated so that dissolved contaminants are precipitated, and it is then evaporated to reduce the volume. Using current technology, the resulting 'sludge' is mixed directly with cement and poured into 200-litre disposal drums. In the near future, material may be evaporated to dryness and disposed in drums containing a paraffin wax encapsulation matrix.

The ion exchange resins arise from in-service clean-up of the water circuits and spent fuel pool. When they have served their useful life, they are discharged and treated for disposal. Using the current technology, resins are cemented directly into 200-litre drums. In the future, resins will be dried and loaded into high-integrity containers (HIC), composed of stainless steel, which will be welded shut.

The spent filters arise from various air and water filtration uses around the power plants. Some of them are of the HEPA type and consist largely of paper. All filters are treated in the same manner, by direct cementation into 200 litre drums.

General contaminated trash arises from day-to-day operations at the power plants, and consists of contact clothing, wipes, sample containers, redundant equipment and tools, etc. Some of this material may not be contaminated at all, but the general policy is to prevent accidental spread of contamination by treating all waste from the active area as if it were radioactive. These wastes will be loaded into 200 litre drums, and subjected to high-force compaction, reducing their volume by a factor of three or more. The resulting 'pucks' will then be loaded into larger overpack drums (of about 300 litre capacity) for disposal.

Korean repository design

The conceptual design of Korean repository[2] (for the initial phase of the repository construction) consists of the following five caverns, as illustrated in Figure 16.1:

a. LLW (Type I) cavern;
b. LLW (Type II) caverns (two caverns);
c. LLW (Type III) cavern; and
d. ILW cavern.

The cross-sectional designs of the LLW and ILW caverns are illustrated in Figures 16.2 and 16.3, respectively. Physical parameters defining the cavern and disposed waste volumes are given in Table 16.1.

It should be noted that no decision has yet been made concerning the backfilling strategy to be adopted in the ILW cavern. The implications for repository performance of various backfilling strategies for the ILW cavern were addressed in this performance assessment. These strategies consist of:

a. cementitious backfill;
b. 10% bentonite / 90% crushed rock mixture; and
c. no backfill.

Waste inventory

Summary information for each of the radionuclides appears in Table 16.2. A summary of the number of drums in each cavern associated with each waste type

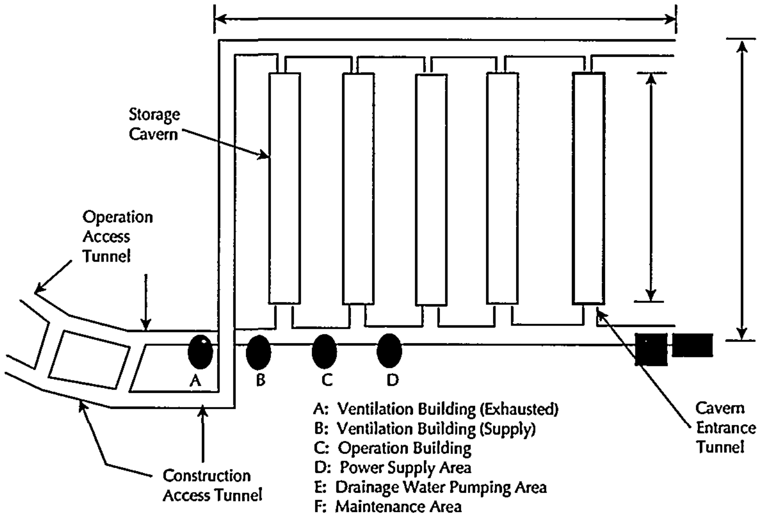

Figure 16.1. Proposed layout of Korean repository (Initial Phase).

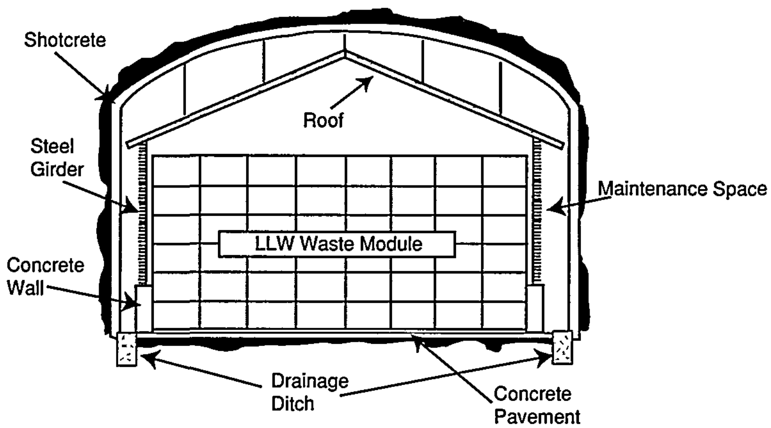

Figure 16.2. Cross section of LLW cavern.

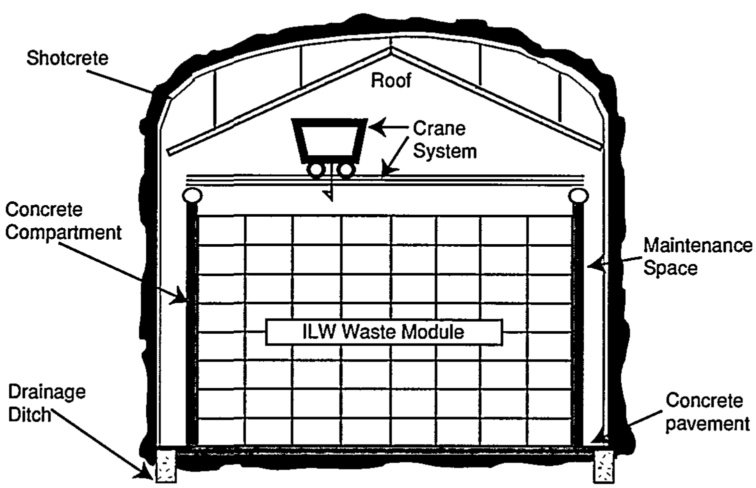

Figure 16.3. Cross section of ILW cavern.

Table 16.1. Cavern and disposed waste volumes (m^3).

Parameter	LLW (Type I)	LLW (Type II) (total)	LLW (Type III)	ILW
V_{wst} (a)	$3.85\ 10^3$	$7.26\ 10^3$	$3.37\ 10^3$	$5.52\ 10^3$
V_{dmod} (b)	$1.08\ 10^4$	$2.15\ 10^4$	$1.08\ 10^4$	$1.49\ 10^4$
$V_{b/f}$ (c)	-	-	-	$3.05\ 10^4$
V_{exc} (d)	$2.73\ 10^4$	$6.06\ 10^4$	$3.03\ 10^4$	$5.62\ 10^4$

Notes:
(a) Volume of packaged waste in 200 litre containers.
(b) Volume of disposal module (6-pack LLW) or 8-pack (ILW).
(c) Volume of backfill (ILW only).
(d) Total volume of caverns.

and of the total volume occupied by the wasteform is provided in Table 16.3.

16.2.2 Chemical Data

Calculations have been carried out to investigate the chemistry of the JWS-2 groundwater, collected at a depth of 330 m, which was considered to have a chemistry most analogous to that in the vicinity of the study area. The aqueous speciation of this groundwater and the likelihood of certain minerals precipitating or dissolving has been modelled using the HARPHRQ program[3] with the HATCHES database[4]. Further calculations have been performed to predict the effect of groundwater equilibration with a high-calcium cement. These calculations have been used as the basis for determining the solubility of a number of key radionuclides in two groundwaters, the JWS-2 water at pH 5.7 and a cement-equilibrated water at pH 12.9.

Table 16.2. Radionuclide Inventory.

Radionuclide	Half-life (years)	Inventory (Bq)
^{3}H	$1.24\ 10^1$	$4.31\ 10^{12}$
^{14}C	$5.73\ 10^3$	$4.11\ 10^{12}$
^{60}Co	$5.27\ 10^0$	$6.16\ 10^{14}$
^{59}Ni	$7.50\ 10^4$	$1.02\ 10^{13}$
^{63}Ni	$9.60\ 10^1$	$3.12\ 10^{14}$
^{90}Sr	$2.91\ 10^1$	$7.95\ 10^{12}$
^{94}Nb	$2.03\ 10^4$	$2.46\ 10^{11}$
^{99}Tc	$2.13\ 10^5$	$3.34\ 10^{11}$
^{129}I	$1.57\ 10^7$	$1.93\ 10^{10}$
^{137}Cs	$3.00\ 10^1$	$3.41\ 10^{14}$
^{235}U	$7.04\ 10^8$	$1.11\ 10^7$
^{238}U	$4.47\ 10^9$	$3.91\ 10^8$
^{238}Pu	$8.77\ 10^1$	$1.59\ 10^{11}$
^{239}Pu	$2.41\ 10^4$	$3.28\ 10^{11}$

The PDFs for elemental solubility limits that were used in this assessment were centred on the results of the HARPHRQ speciation calculations, with log-triangular PDFs covering whole orders of magnitude. The main materials of interest to the conceptual repository are the in-drum cement, concrete and bentonite (candidate backfill materials for the ILW caverns), and the mineral andesite, which is the host rock considered for this study.

A literature survey has been carried out to determine the likely Rd values for sorption of radionuclides onto the

Table 16.3. Waste inventory.

Waste type	Number of drums	Volume occupied (m^3)			
		LLW (Type I)	LLW (Type II)	LLW (Type III)	ILW
Liquid conc. in cement	10,425	-	4,996	-	$3.08\ 10^3$
Liquid conc. in paraffin	-	-	-	10,273	$2.05\ 10^3$
Ion exchange resin in cement	4,516	-	-	-	$9.03\ 10^2$
Ion exchange resin in HIC	-	-	-	16,445	$3.29 10^3$
Spent filters in cement	4,307	-	-	882	$1.04 10^3$
General trash	-	36,288	11,868	-	$9.63\ 10^3$

bentonite backfill and onto the host rock formation.

16.2.3 Hydrogeological Data

The first step in the development of a numerical model of the groundwater flow in the region around the study area is to develop a conceptual model of the geological and hydrogeological structure of the region. 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 most appropriate type of model to construct for this illustrative assessment was therefore a two-dimensional vertical cross-section model along a line roughly perpendicular to the coast. It was noted that the model would have to extend some distance offshore so that the saline transition zone near the coast could be modelled satisfactorily. The geological structure along the main line of section selected for the groundwater flow model is shown in Figure 16.4.

In order to carry out this preliminary assessment, input PDFs were required for all of the hydrogeological properties used in the groundwater flow and transport models. The materials identified in the geological structure were:

1. upper andesite;
2. lower andesite;
3. fractured lower andesite;
4. granite;
5. shale;
6. sandstone/mudstone/siltstone sediments; and
7. gneiss.

In addition, the hydrogeological properties of the repository itself, and of the access tunnels, had to be considered. A PDF was required for the permeability and porosity of the material in three cases: normal rock, rock within a fault core, and rock within a fault halo. The notation T(a,b,c) is used to indicate a triangular PDF with upper and lower limits c and a, respectively, and peak value b. For convenience, logarithms to base 10 were used. The hydraulic conductivity for some of the rock types are summarized in Table 16.4, as an example.

16.2.4 Biosphere Data

The approach adopted is to identify critical groups, and hence to evaluate potential maximum individual risks. This evaluation is based on the definition of hypothetical subsistence communities, which are assumed to make maximum reasonable use of local food resources and to be located in the area of the environment assessed to be the most contaminated as a result of possible future discharges from the repository.

In this assessment, the screening process was carried out using the generic biosphere program BIOS_3A[5], which considers a relatively wide range of potentially important pathways. On the basis of the screening calculations, a simple process model was developed taking advantage of the Compartment Biosphere submodel available within the MASCOT program[6].

The terrestrial biosphere considered comprises the catchment of a small river discharging into a coastal sea. The section of river that may be contaminated by

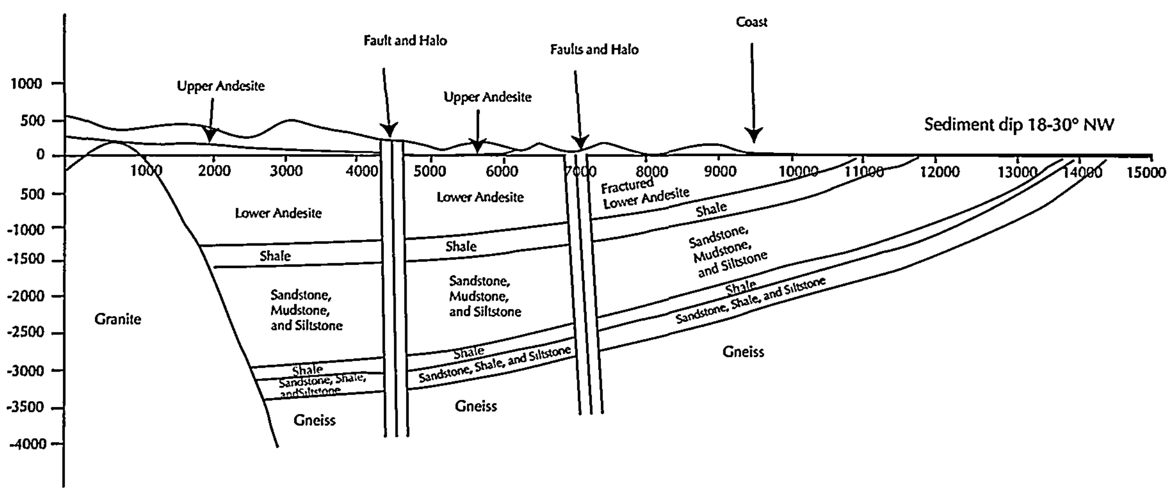

Figure 16.4. Geological structure along the line of the model cross section. Distances and elevations in metres.

Table 16.4. Summary of log-hydraulic conductivity distributions.

Rock type	Log-hydraulic conductivity[a] Lower bound	Peak	Upper bound	Comment
Andesite	-12	-8	-6	Isotropic
Fractured Andessite	-11	-7	-5	Isotropic
Granite	-11	-8	-6	Isotropic
Gneiss	-13	-10	-7	Isotropic
Shale	-12	-9	-7	Isotropic
Sandstones	-10	-7.6	-5	Isotropic

Note: (a) Values given are for log-hydraulic conductivity (permeabilities in m s^{-1}, log to base 10).

groundwater from the repository, either directly or indirectly (through drainage from adjacent farmland), is estimated to be 5 km in length, and the area of its associated catchment, 10 km^2.

The critical group at risk from radionuclides entering the biosphere is assumed to be a community of subsistence farmers who live and work in the land. For the preliminary assessment, their diet was assumed to be similar to the Korean average.

The endpoint of the calculations corresponds to the maximum dose received resulting from a steady 1 Bq yr^{-1} release rate from the geosphere, which occurs as concentrations in the environment approach, steady-state equilibrium values. The calculations demonstrated that a wide range of terrestrial pathways should be considered. Marine pathways, however, were found to be less important, because of much higher levels of dilution in coastal waters. The results of the analysis using BIOS_3A provided a basis for selecting pathways for the second phase of the assessment using MASCOT's biosphere compartment submodel facility. Terrestrial pathways considered in the second stage of the assessment using MASCOT included ingestion of rice and vegetables, drinking water, consumption of meat products (beef, milk, liver, and poultry), external gamma exposure from soil, and dust inhalation. Although less important, ingestion of marine foodstuffs was also considered.

Figure 16.5 shows the structure of the biosphere model developed for the preliminary assessment.

16.3 GROUNDWATER PATHWAY

16.3.1 Groundwater Flow Modelling

After the closure and resaturation of a repository for radioactive waste, it can be anticipated that radionuclides will dissolve in groundwater flowing through the wasteform and will be transported with the groundwater through the geosphere to the biosphere. This is the natural pathway by which radionuclides, disposed in a repository, may return to the human environment. It is almost certain that radionuclides will return to the biosphere by this route, and so an analysis of the risk arising from the groundwater pathway forms an important part

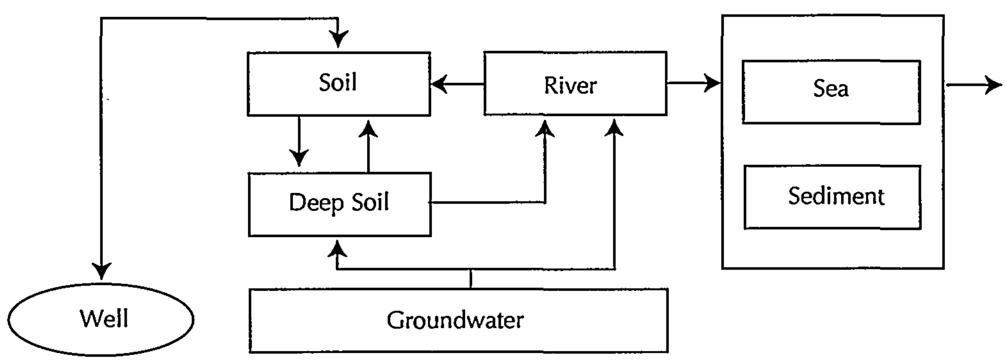

Figure 16.5. Structure of biosphere model.

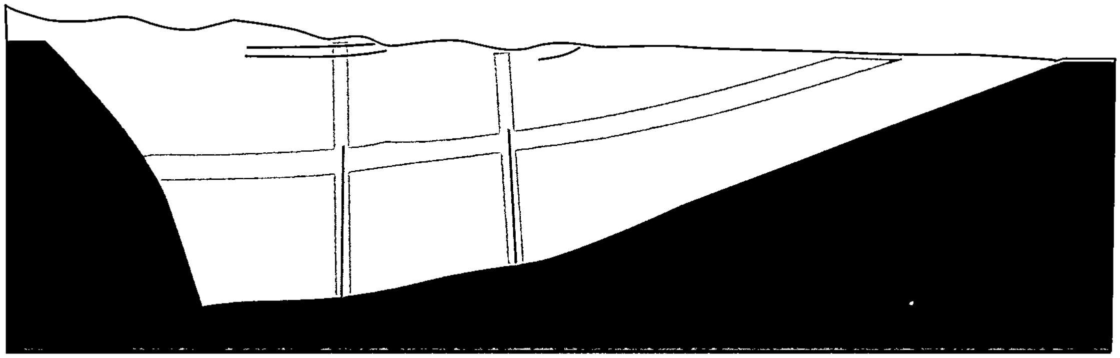

Figure 16.6. Shade plot of the NAMMU patch grid showing assignment of rock types. The lines from the repository locations represent the access tunnels.

of an assessment of the performance of a repository for radioactive waste.

In order to carry out this analysis, it is necessary to be able 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).

The first stage in the development of a NAMMU groundwater flow model[7] of the site is the construction of a finite-element mesh that represents the important features of the geology and topography of the site (see Fig. 16.4) and extends to regions where well-defined boundary conditions can be identified. Careful design of the finite-element mesh is necessary both to ensure that the numerical solution is as accurate as possible and to allow for subsequent changes to the model to be made with ease.

The top surface of the mesh was a smoothed representation of the topography, treated as a number of straight-line segments that preserved the major features. The left-hand boundary of the model was taken to be an assumed (vertical) groundwater divide under the high ground inland.

The gneiss has a very low permeability and so, in principle, the base of the sediments could have been taken to form the base of the model. However, a satisfactory treatment of the region of saline groundwater offshore would require that the distribution of salt concentration on the base of the sediments be known a priori, which is not the case. It was therefore preferable that the model extended far enough offshore to include the offshore surface outcrop of the gneiss. It is then reasonable to assign a constant salt concentration along the right-hand boundary of the model. Figure 16.6 shows the assignment of rock types in the base-case model. The aim was to produce a grid in which the largest elements were about 200-250 m across. In practice most of the elements are considerably smaller. The final grid used for the calculations is shown in Figure 16.7.

The overall (large-scale) flow is driven by the high heads produced by the high ground on the left-hand side of the model and moves downwards towards the right-hand side of the model. This flow continues at depth until it reaches the salt front, which is intruding from the right-hand side of the model. The saline transition zone is at the point where the driving head from the topography exactly balances the head due to the increased density of the seawater. The freshwater therefore flows up the salt front and discharges near the coast. Seawater intrudes from the right-hand boundary of the model and flows into the model until it reaches the saline transition zone, which it flows up before discharging through the sea bed. In the near-surface part of the model onshore, small flow cells driven by local topographic variations are superimposed on the large-scale flow. Local discharge occurs in all of the small valleys represented in the model. There is very little flow in the gneiss layer.

A series of pathline calculations were performed to investigate the general nature of the flow system. The pathlines that were used for the MASCOT analysis were started from the following two sets of points: (3600,55), (3700,55), and (3800,55) to represent paths starting from the 'upper repository' (i.e. a repository located 50 m above OD, about 250 m below the ground surface), and (3600, - 50), (3700, - 50), (3800, - 50) to represent pathlines starting from the 'lower repository' (i.e. a

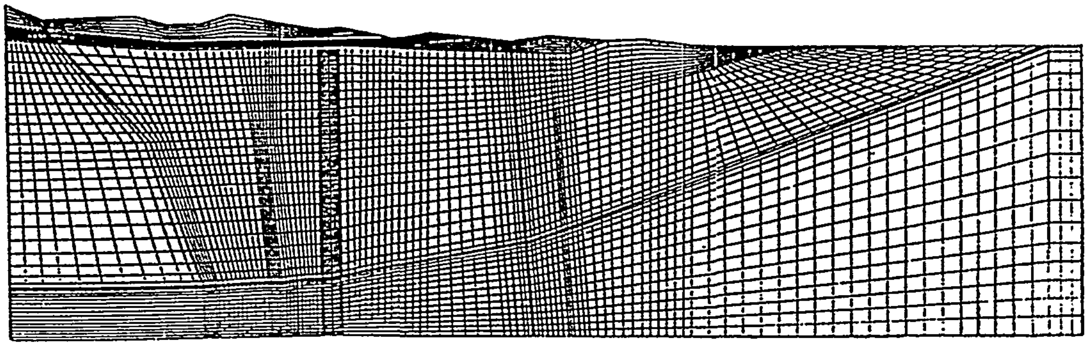

Figure 16.7. Plot of finite-element mesh.

repository located 50 m below OD, about 350 m below the ground surface). The travel times and path lengths for these paths, and the specific discharges at the repository locations, are summarized in Table 16.5.

One of the pathlines that was started from the upper repository position crossed the fault zone near the repository and emerged near the tunnel entrance after a travel time of approximately 1200 years. The other two pathlines from the upper repository position emerged up the fault zone with travel times of approximately 500 years after being caught in the flow cell that discharges in the small valley near the fault (Fig. 16.8a). All three pathlines that started from the lower repository crossed the fault and emerged near the tunnel entrance with travel times of about 1100 years (Fig. 16.8b). Three pathlines were also started from a location near the coast, solely in order to illustrate the general nature of the flow in the vicinity of a hypothetical repository sited there. Two of the pathlines emerged a short distance offshore, but the third was caught in a local flow cell and emerged in the valley closest to the coast (see Fig.16.8c).

Finally, a calculation was performed to estimate the resaturation time of the repository. The base-case model was modified to apply a boundary condition of atmospheric pressure at the repository location. The total flow of groundwater into the repository induced by this condition was then estimated by integrating the specific discharge along the four sides of a box closely matching the repository location, and then multiplying the result by the width of the repository perpendicular to the cross section. The resaturation time is then estimated by dividing the empty volume of the repository by the volume of flow towards the repository.

The total flow into the repository in the two-dimensional model was found to be 4.67 10^{-6} m^2s^{-1}. The repository region was taken to have a length of 440 m perpendicular to the section, giving a total flow into the repos-

Table 16.5. Results of the base-case pathline calculation.

(a) Upper Repository

Path	Travel time (years)	Pathlength (m)	Repository specific discharge (x component, m s^{-1})	Repository specific discharge (y component, m s^{-1})
1	1176.2	1828.6	1.2265 10^{-9}	-8.684 10^{-10}
2	565.93	971.61	1.3302 10^{-9}	-6.9320 10^{-10}
3	432.12	777.98	1.3916 10^{-9}	-5.1647 10^{-10}

(b) Lower Repository

Path	Travel time (years)	Pathlength (m)	Repository specific discharge (x component, m s^{-1})	Repository specific discharge (y component, m s^{-1})
1	1243.1	1882.9	1.0702 10^{-9}	-7.4150 10^{-10}
2	1114.8	1722.3	1.1650 10^{-9}	-6.0711 10^{-10}
3	1049.1	1602.4	1.2276 10^{-9}	-4.684510^{-10}

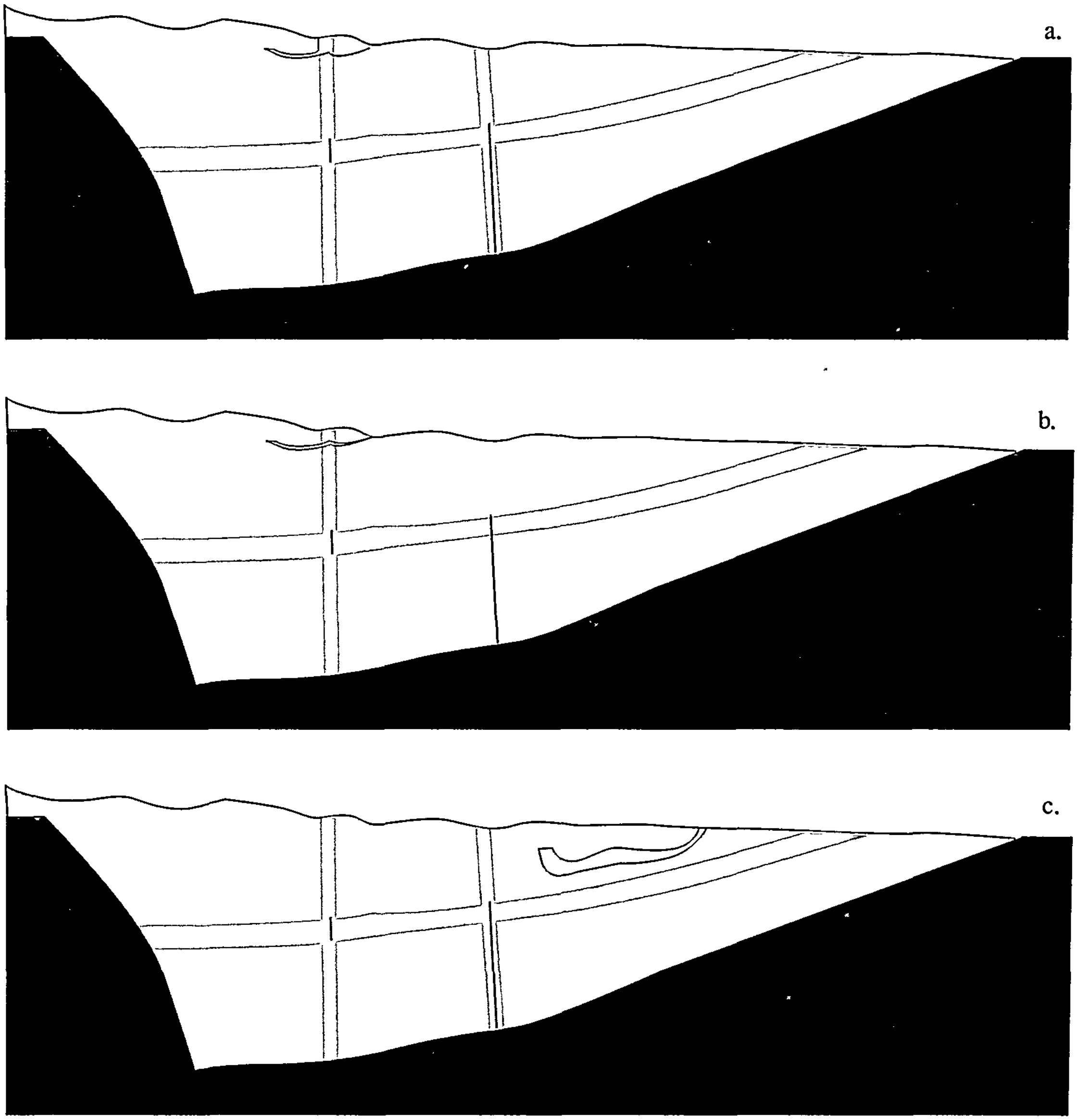

Figure 16.8. Results for some key pathlines: a - From the upper repository position; b - From the lower repository position, and c - From the coastal repository position.

itory region of 2.05 10^{-3} m^3s^{-1}. The total empty volume of the caverns after repository closure is 1.54 10^5 m^3, which gives a resaturation time of 2.4 years. This short time is a result of the high heads above the repository location and the relatively high permeability of the andesite. It should be noted that as the repository resaturates, its pressure will rise and the inflow will be correspondingly reduced. Thus, the value given above is likely to be an underestimate of the resaturation time.

16.3.2 System Model for Groundwater Pathway

In this subsection, the construction of an overall model of the system for groundwater-mediated return of radionuclides to the biosphere is explained.

Source-term submodels

The repository under consideration is planned to contain wastes of several different types, and there will be some degree of segregation into different parts of the repository. The different waste types can therefore be expected to contribute differently to the source term. In general, the key factors determining source-term behaviour are:

a. degradation of the physical containment;
b. chemical control on the solution concentration of the

Table 16.6. Calculated buffering times for caverns.

Cavern	Amount of CaO (mol m^{-3})	Buffering time at pH12.2 (yr)
LLW (Type I)	975	$1.5\ 10^3$
LLW II	0	0.0
LLW III	218	$3.3\ 10^2$
ILW (no backfill)	20	$3.2\ 10^1$
ILW (bentonite backfill)	20	$3.8\ 10^4$
ILW (concrete backfill	2,400	$4.6\ 10^6$

radionuclides; and

c. the rate of groundwater flow through the repository.

Although physical containment of the wastes is important during the operational phase of the repository, the long-term behaviour after closure will be affected significantly by physical containment only in the case of containers specially engineered for exceptional performance. It is recommended that for all wastes in 200 - litre mild steel drums, physical containment be ignored, since all the other time scales involved in release and transport of radionuclides will be much longer than the expected corrosion lifetime of these drums. For the ion exchange resins in stainless steel HIC containers, however, a period of absolute containment should be modelled.

Some of the source term models selected for analysis of the release of radionuclides take account of the conditioning of the pore water to high pH by dissolution of free calcium hydroxide in the cements. It is appropriate to analyse each disposal vault, to determine whether there is sufficient $Ca(OH)_2$ to buffer the pH during the relevant assessment period.

The results of a simple analysis are given in Table 16.6 for each type of disposal vault; note that the buffering times are averaged across the entire vault, and do not represent the smaller-scale buffering within packages. This analysis indicates that the current plans would result in a wide range of buffering times for the various vaults. The LLW I and backfilled ILW vaults would clearly be buffered for a substantial length of time, whereas the LLW II and non-backfilled ILW vaults would not. The LLW III vaults represent an intermediate case. It is clear that extended buffering of pore water to high pH would require either increased amounts of free $Ca(OH)_2$, or reduced groundwater flow through the vaults (most easily achieved by backfilling).

Because the six types of waste are distributed in different caverns, it was necessary to adopt an approach in which nine different source-term submodels were used, as shown in Figure 16.9.

The submodels LLW1A, LLW1B, etc. are MASCOT Containment submodels, which are used to specify the initial inventory of each radionuclide, and any time of absolute containment of the waste. These feed into source-term submodels, ST1A, ST1B, etc., which calculate the release of the radionuclides. The purpose of each of these submodels is summarized in Table 16.7.

In Figure 16.9, the submodels named DIST1 and DIST2 are Distributor submodels used in MASCOT for combining and/or redistributing fluxes between submodels. In this case, DIST1 adds the outputs of the three ILW source-term submodels to provide the input to submodel BARR, which represents the barrier provided by the backfill in the ILW cavern. Finally, DIST2 adds the output of BARR and those of all the LLW caverns to make the total source-term output to be fed into the Geosphere section of the system model.

The BARR submodel is added to represent the time delay for release of radionuclides from the ILW cavern into the geosphere, arising because of the backfill in that cavern. For initial calculations, it was decided to treat the effect by using a porous geosphere submodel, with nominal water transit time equal to the approximate time for water molecules to diffuse through the barrier. The different radionuclides would then be transported through this barrier submodel in times longer than this, according to their retardation coefficients, calculated from sorption distribution coefficients appropriate to the backfill material.

Geosphere transport submodels

From the pathline calculations, two possible pathways from the repository to the surface emerged; the first (Path A) spending time in the upper/lower andesite, followed by a time crossing the fault zone, followed by a further time in the upper/lower andesite, the second

Table 16.7. Source-term submodels within MASCOT system model.

Name	Submodel type	Representing
ST1A	SLST[(a)]	Cemented liquid concentrates (LLW cavern Type I)
ST1B	SLST[(a)]	Cemented ion-exchange resins (LLW cavern Type I)
ST1C	SLST[(a)]	Cemented filters (LLW cavern Type I)
ST2	SLST[(a)]	Compacted trash (LLW cavern Type II)
ST3A	SLST[(a)]	Cemented liquid concentrates (LLW cavern Type III)
ST3B	SLST[(a)]	Compacted trash (LLW cavern Type III)
ST4A	Leaching	Paraffin-encapsulated liquid concentrates (ILW cavern)
ST4B	Leaching	Ion- exchange resins in HIC (ILW cavern)
ST4C	SLST[(a)]	Cemented filters (ILW cavern)

Notes: (a) Solubility-limited source term.

(Path B) spending time in the upper/lower andesite followed by a time travelling vertically up the fault zone to the surface.

The network of geosphere submodels is shown in Figure 16.9. The input to the geosphere network is a combined flux from all the source terms (from submodel DIST2. The distributor submodel DIST3, divides the flux leaving the AND1AB submodel (representing the first section of andesite that occurs in both paths) into two parts, one which takes Path A (to submodel FAULTA), and one which takes Path B (to submodel FAULTB). The fluxes from the two paths are recombined in submodel DIST4 to provide a total flux to the biosphere. The geosphere submodels represent the following:

AND1AB The first section of upper/lower andesite in which both paths A and B spend time;

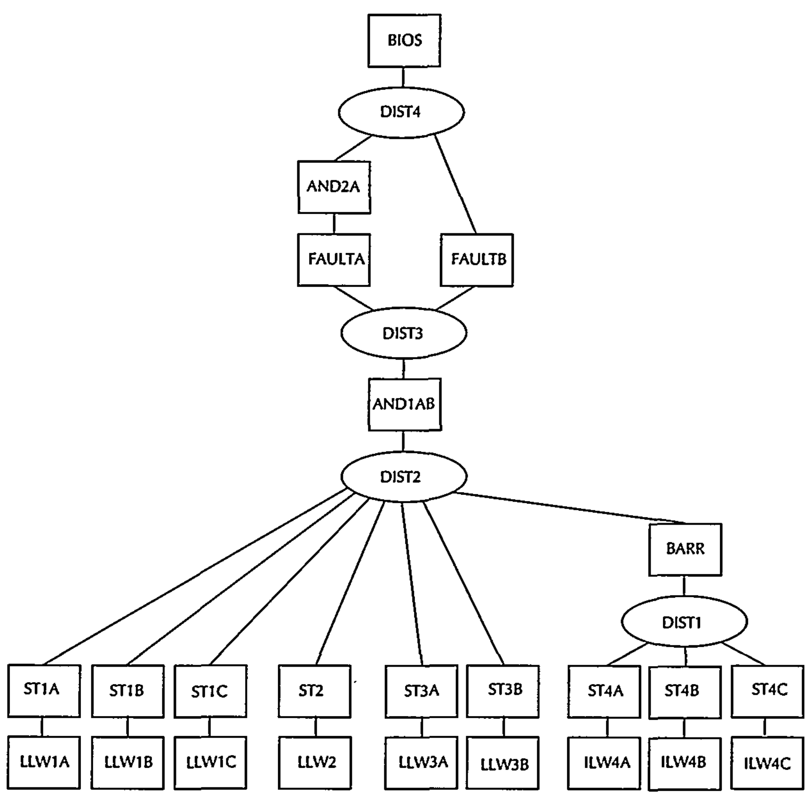

Figure 16.9. System model made up out of source term, geosphere, and biosphere submodels.

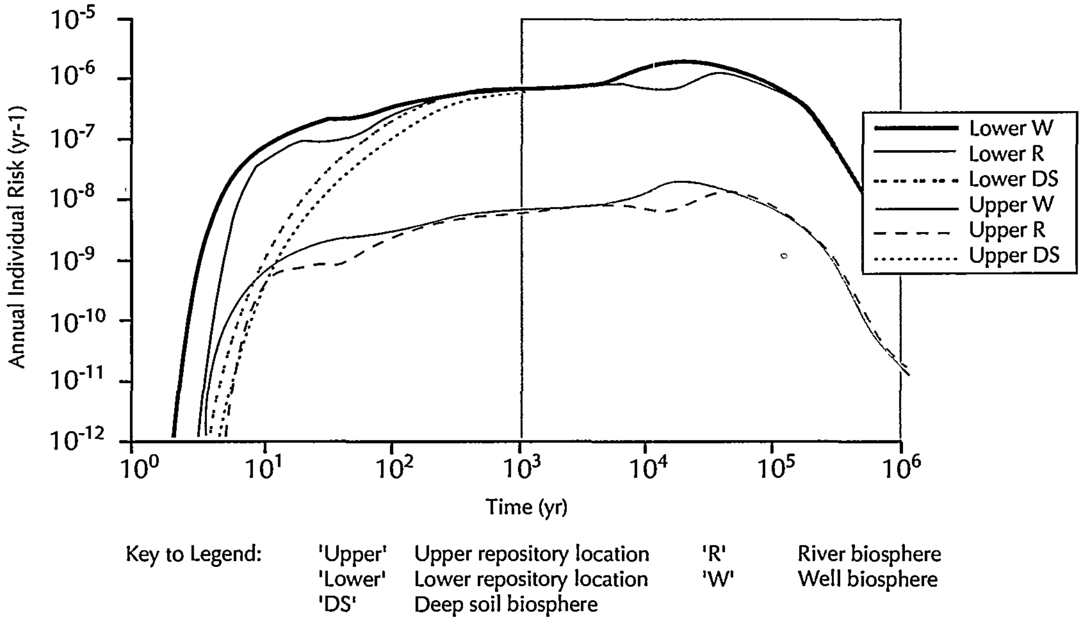

Figure 16.10. Total annual individual risk as a function of time for the upper and lower repository locations for each biosphere.

FAULTA Path A in the fault zone;
FAULTB Path B in the fault zone to the surface;
AND2A Path A in the second section of upper/lower andesite to the surface.

All these submodels are MASCOT Porous Geosphere submodels. This type of submodel assumes that the rock matrix is fully accessed and hence the porosity is the full matrix porosity.

Biosphere submodels

Three exposure pathways were considered, in which the flux from the geosphere is released to the deep soil, a river and a well. Three identical compartment biosphere submodels (DEEPSOIL, RIVER and WELL) 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 was to different compartments appropriate to the three exposure pathways. The compartment structure of the biosphere is shown in Figure 16.5. The outlet from the biosphere was in each case a total dose from concentrations in soil, water (river or irrigation) and seawater.

16.3.3 Results

The total annual individual risks calculated for each biosphere model for the upper and lower repository locations are shown as functions of time in Figure 16.10. It can be seen that for both locations, the deep soil and well biospheres give very similar peak risks (marginally higher in the case of the well), whereas the river biosphere gives risks which are about two orders of magnitude lower.

The risk values, being obtained from mean doses that are estimated from a finite number of realizations (1000 for these runs), are subject to statistical error. From the variance of the observed values, this error can be estimated, and typical 95% confidence intervals are shown in Figure 16.11 (for the lower repository location). For most of the time-range, the upper 95% confidence limit exceeds the best estimate by a factor of about 1.5 or less. This is considered perfectly adequate for a preliminary assessment, with many uncertainties in the data and in the model choices that may well cause biases of rather greater magnitude. The statistical estimation errors could be reduced by including more realizations; to halve the error, four times the number of realizations would be necessary.

16.3.4 Discussion

Risk levels

Using the models and data adopted for these assessment

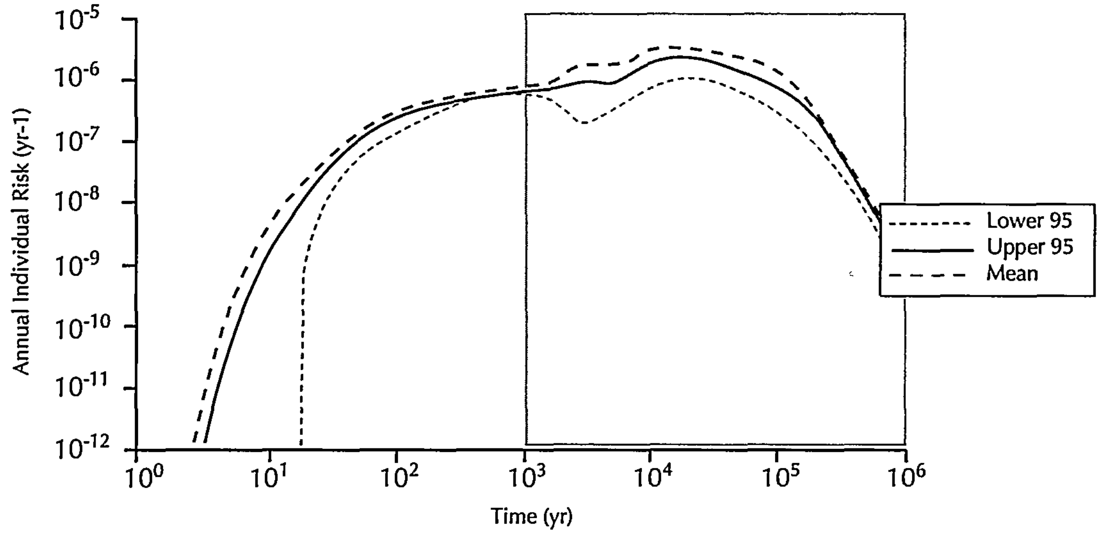

Figure 16.11. Total annual individual risk for release to the deep soil biosphere for the lower repository location plotted against time (with 95% confidence limits).

calculations, it was found, for both choices of repository location and for all the alternative assumptions about the release point to the biosphere, that the predicted risk remained below the target of 10^{-6} yr^{-1} throughout the period of detailed quantitative assessment, up to 1000 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.

Because of the parameter-value uncertainties, which the probabilistic approach is designed to treat, different sampled realizations of the system gave conditional risks either higher or lower than the mean. From the calculated distributions, it was found that there was a probability of 0.25 to 0.30 of exceeding a risk of 10^{-6} for times up to 10^3 years, and a probability of about 0.50 of exceeding a risk of 10^{-6} for times up to 10^6 years.

Important radionuclides

Up to 1000 years after closure, only ^{129}I makes a significant contribution to risk, other radionuclides either being effectively contained, or else delayed in their passage through the geosphere. Beyond the 1000-year period, the only significant additional contributor to risk was found to be ^{59}Ni.

Effects of the alternative biosphere models

The risk curve for ^{129}I when using the biosphere model in which discharge to the deep soil is assumed, shows a substantially greater initial delay relative to that found using the river or well biosphere models. This suggests that the transport process (when present) from the deep soil to the other compartments provides a non-negligible supplement to the geosphere travel times. Apart from this difference, the shapes of the risk versus time curves for the three biosphere models are very similar.

Differences between the upper and lower repository locations

The peak risk for the upper repository location was found to be 9.8 10^{-7} for the deep soil biosphere and is due to ^{59}Ni. The ^{59}Ni curve clearly shows the difference between paths A and B as two separate peaks. For the lower repository location, the peak risk is higher, at 1.4 10^{-6} for the deep soil biosphere. In fact, the travel times are in general longer from the lower repository location. However, in this case the two peaks arising from paths A and B are not so obviously separated. For the lower repository location, the travel times for the total Path A and the total Path B are more comparable than they are for the upper repository location, and the reason that the risk is lower in the case of the upper location is that the larger difference between paths A and B causes more spreading to occur in the geosphere and hence reduces the risk. This interesting result shows that the spreading of arrival of radionuclides at the surface is not just a matter of hydrodynamic dispersion, but can be contributed to by division of the transport between major alternative paths. However, it should be noted that these two calculations provide insufficient evidence to draw

general conclusions about the relationship between repository placement and the extent of this type of dispersion.

Effect of parameter uncertainties

Whereas the expected contribution to risk from ^{59}Ni arises from just a small number of realizations that lead to relatively high conditional risks (those in which both the geosphere travel time and retardation of Ni are low), the contribution to risk from ^{129}I arises from a moderate risk in almost all realizations. This shows that the potential contribution to risk from ^{59}Ni could be subject to significant revision as future reductions are achieved in the present uncertainty about hydrogeological conditions and the retardation of Ni in the geosphere.

16.4 SUMMARY OF PROBABILISTIC SAFETY ASSESSMENT

A number of deterministic variant calculations were carried out, to explore the sensitivity of the system performance to several hypothetical changes. The changes addressed were:

a. no backfill in the ILW caverns;
b. concrete backfill in the ILW caverns;
c. high specific discharge at the source;
d. reduced geosphere travel times;
e. combination of (c) and (d); and
f. low near-field sorption coefficients.

These variations were considered relative to a base case defined by giving to each uncertain parameter in the probabilistic calculations its best-estimate value (the mode or median of the PDF).

The main conclusions are summarized here.

a. The overall peak total risk is similar in the deterministic base case to that given by the probabilistic calculations.
b. In the time range 10^2 to 10^4 years, ^{129}I is the dominant contributor to risk, as in the probabilistic calculations.
c. ^{59}Ni hardly contributes at all to risk in the deterministic base case, indicating that its significant contribution to the mean risks arises from realizations with less than average return time of this radionuclide to the biosphere.
d. Removal of the barrier provided by the ILW backfill was found to affect risk significantly; in the case of the ^{129}I contribution, this is mainly due to its effect on the spreading in time of the release to the geosphere, whereas for ^{59}Ni it is associated with the change in travel time affecting the amount of decay.
e. Decreasing the barrier thickness would increase risks, to an extent intermediate between the base case and the variant with zero thickness; increasing the barrier thickness, however, is unlikely to reduce risks significantly relative to the base case.
f. Decreased groundwater travel times in the geosphere leads to a significant increase in risk; in the case of the ^{129}I contribution, this is mainly due to the associated reduction in the spreading in time of the arrival in the biosphere, whereas for ^{59}Ni, the effect is because reduced travel times allow less decay to occur.
g. The variants involving either an increased flux of groundwater through the repository, or changes to near-field chemistry, do not lead to very great increases of risk relative to the base case.

REFERENCES

1. Agg, P. J., et al., P-KAERI 94 Performance and Environmental Assessment of a Radioactive Waste Repository in Korea, NEMAC/AEA Technology, 1995.

2. Park, H. S., et al., A Study on the Requirements for Basic Design of a Repository for Low-Level Radioactive Waste, KAERI-NEMAC/PR-32/93, 1994.

3. Brown, P. L., et al., HARPHRQ: A Geochemical Speciation Program Based on PHREEQE, NSS/R188, UK Nirex, 1991.

4. Cross, J. E., et al., HATCHES - A Thermodynamic Database and Management System, NSS/R212, UK Nirex, 1990.

5. Martin, J. S., et al., User Guide for BIOS-3A, NRPB-M285, NRPB, 1991.

6. Sinclair, J. E., et al., MASCOT and MOP Programs for Probabilistic Safety Assessment, NSS/R336, UK Nirex, 1994.

7 Hartley, L. J., and C. P. Jackson, NAMMU (Release 6.1) User's Guide, AEA-D&R-0472 (Release 6.1), AEA Technology, 1993.

Chapter 17

Research on Radioactive Waste Disposal in the Netherlands with Special Reference to Earth Scientific Studies

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17.1 Introduction

Investigations for radioactive waste disposal in the Netherlands were carried out within the framework of the OPLA research programme (OPLA is a Dutch acronym for 'Disposal on Land'). The programme intended to explore the disposal options for nuclear waste in the deep subsurface below the Netherlands. Initially, the research focused on rock salt as a host rock.

Massive bodies of rock salt, with a thickness of 200 to 3,000 m, are present in the subsurface of the northern Netherlands. This potential host rock for radioactive waste disposal was deposited in layers about 220 to 250 Ma ago (the Zechstein period) by evaporation of a closed or semi-closed marine basin. After the evaporites were covered by fresh sediments, the layered salt started to flow at several places in the subsurface of the Netherlands. At many locations this resulted in the formation of salt pillows and sometimes, successively in salt diapirs. The tops of the shallowest salt diapirs rose to depths of about 100 m below the surface. The non-disturbed layered salt is situated generally at a depth of several kilometres below the earth's surface.

The OPLA research programme was divided into three phases for purposes of the interim parliamentary decision-taking procedure. At the end of each phase, government and parliament decide on the question of whether a subsequent phase of the programme is to be performed and, if so, what content and scope it should then have.

Execution of the first phase commenced in 1984. This first phase comprised a feasibility study, investigating various disposal methods and salt formation types with reference to safety. For that purpose, desk studies and laboratory investigations were performed on the basis of existing data in the public domain. In many cases, these investigations were carried out in international co-operation, for example in the area of *in situ* investigations. No decisions have been taken as yet regarding the execution of any further phases of the research programme, which will be more site-specific.

The OPLA Committee reported on the first phase of the research programme in mid 1989. A review of the first phase was given by Van Montfrans in the preceding issue (1991) of the present world-wide review. The conclusions of the studies undertaken in that programme amount, in brief, to a statement that safe disposal of radioactive waste in rock salt formations of a nature and scale as very probably occur in the Dutch subsurface is, in principle, feasible. The OPLA Committee also concluded that certain assumptions on which the conclusions were based required further verification to establish their reliability and accuracy. It was furthermore found that the possibility of identifying in concrete terms exactly which salt formations would be most eligible for further investigation on technical grounds was limited by a lack of site-specific data.

At the same time, according to the ILONA Committee (advising on the policy for radioactive waste management in the Netherlands) it was possible and necessary to perform supplementary research in order to reduce the limitations and margins of uncertainty established in a number of respects by the 1989 findings. This view was supported by the results of an international review performed by the European Community (EC) and the Nuclear Energy Agency (NEA) of the Organisation for Economic Co-operation and Development (OECD). In addition, it was considered important that attention should be given to new developments, such as the

retrievability of disposed-of waste.

17.2 CONTENT AND STRUCTURE OF THE SUPPLEMENTARY RESEARCH PROGRAMME

Based on the conclusions and recommendations of the first phase and the results of the EC/NEA review, the OPLA Committee formulated the main themes of a limited programme of supplementary research, "Phase 1A" (OPLA Committee, 1993). The most important themes are:

- Analysis and reduction of bandwidths in the results of Phase 1, particularly in the results of the safety analysis. This task invoked an improvement of the methodology of the Phase 1 safety approach. This approach was expanded into a methodology enabling systematic treatment of disposal risks. This improves the inter-comparability of risks, and makes it possible to allow for probabilities of processes and data.
- In addition, careful analysis and further enlargement of the database on the subsurface was required. At the same time, the extension and further validation of the generic computer models applied in Phase 1 were dealt with. New models, describing the long-term geological evolution of the subsurface, were developed as well. The approach outlined above provides a better confirmatory basis for the findings of the safety study, and they may therefore be considered more accurate.
- Classification of salt formations for further investigation. Grouping of the salt formations was carried out on the basis of expected long-term stability and accommodation space for repository designs. This theme will not be discussed further in the present report.

At the same time, the Committee was requested to consider new developments; these are classed as a separate main theme. Two topics were dealt with in this theme, viz. the retrievability of disposed-of radioactive waste and the direct disposal of spent fuel elements.
Further, the Committee strongly advocated participation in international research programmes.

The supplementary programme was performed in the period from mid 1990 to mid 1993. Within the programme about 20 studies were performed in the areas of study of the OPLA research programme. These areas of study had already been considered in Phase 1: safety (the central theme), technical feasibility (mining engineering), geology and geohydrology, rock mechanics and radiation effects. This paper will deal with results of the supplementary programme and in particular will focus on the outcome of the various earth scientific background studies.

17.3 SAFETY ANALYSIS

Progress was made on the following matters:

- risk assessment method; and
- data and models needed for the risk assessment.

17.3.1 *Health risk assessment method*

In Phase 1, the health risks of disposal were calculated on a deterministic and conservative basis, i.e. choosing circumstances unfavourable for safety and thereby over-estimating risks. This approach hampered the inter-comparability of risks for the various disposal concepts in Phase 1. In addition, this method did not allow uncertainties and sensitivities to be treated in an adequate way.

Within the supplementary programme, a method was developed known as PROSA (PRobabilistisch Onderzoek aan de veiligheid van in Steenzout opgeborgen radioactief Afval = Probabilistic study into the safety of radioactive waste disposed of in rock salt), which accounts systematically for the probability of processes and subsurface data (Prij et al., 1993a). As a result, the risks as calculated are better confirmed and are more readily inter-comparable. The systematic treatment of uncertainties was incorporated within PROSA, thereby improving the significance of the sensitivity analysis as compared to Phase 1.

The PROSA method is essentially based on a system developed for the determination and selection of scenarios (possible situations in which radionuclides are released from a repository and can potentially reach the biosphere). The method is based on the internationally accepted multi-barrier system, which comprises three barriers:

- engineered barriers of the repository;
- rock salt around the repository; and
- overburden with aquifers and impermeable strata.

The degree in which this system acts as a barrier is affected by a complex of Features, Events and Processes (FEPs). These FEPs were used to construct scenarios relevant to conditions in the Netherlands. These scenarios can be subdivided into three classes:

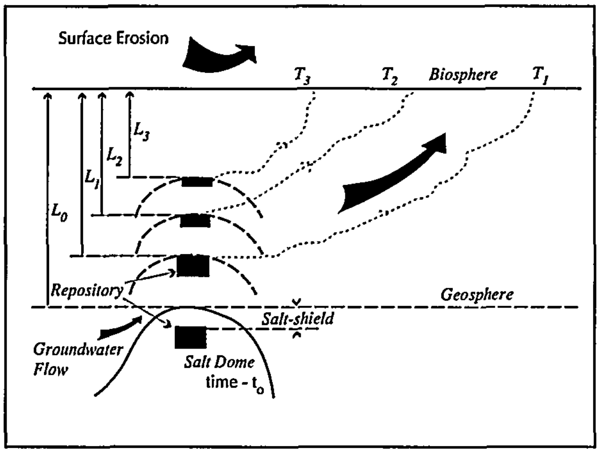


Figure 17.1. Schematic representation of the dissolution/diapirism scenario (Prij et al., 1993a).

- Subrosion/diapirism scenarios belonging to the category of normal evolution scenarios (Fig. 17.1). Gradual ascent of a salt dome (diapirism) brings it into contact with groundwater, salt dissolves (subrosion), radionuclides from the repository enter the groundwater and from there travel into the biosphere.
- Flooding scenarios belonging to the category of altered evolution scenarios. Groundwater penetrates into the repository, for example, as a result of fracturing or a water-permeable inclusion in the rock salt. Radionuclides and salt dissolve in this water. Convergence of the disposal chambers in the salt next forces the brine contaminated with radionuclides out of the salt into the groundwater. The radionuclides then enter the biosphere via the groundwater.
- Human intrusion scenarios belonging to the category of disruptive event scenarios. Various forms of mining engineering activities (exploratory drilling, construction of a mine) may bring future generations into involuntary contact with the waste (see also Prij, 1993).

Due to the lack of sufficient knowledge, two phenomena considered potentially important could not yet be fully calculated with the PROSA method: glaciation (the consequences of an ice age for the repository) and gas formation (consequences for the repository of gas formation near the waste resulting from thermal and radiation effects). Further, only those human intrusion scenarios were recalculated where improved insight relative to Phase 1 made it meaningful to do so.

In the supplementary programme, the risks were calculated for each of the above-mentioned scenario classes and for a number of schematised disposal situations in Dutch rock salt. The assumptions applied are based on the currently available data on the nature and size of rock salt formations as these very probably occur in the Dutch subsurface.

17.3.2 Data and Models Needed for Risk Calculations

Calculating the risk of subsurface disposal requires among other things a comprehensive database relating to the structure and the groundwater system of the subsurface. This database yields the necessary description of the multi-barrier system. For the purpose of the risk calculations, the processes involved in transporting radionuclides through this multi-barrier system are described by means of models.

The database built up in Phase 1 with published data on the subsurface was found to be limited in scale and usefulness. The supplementary desk studies therefore incorporated thorough analysis and enlargement of that geological and geohydrological database. This led to the determination of a number of bandwidths, for example, for spatial geological data on salt formations. In other respects, too, the precision of the data has been improved, for example, on the size of salt formations, rate of diapirism (Fig. 17.2, lower than had to be assumed in 1989), speed of dissolution processes of rock salt in ground water (Fig. 17.3, subrosion), structure and permeability of the rock surrounding the salt, and the mechanical behaviour of salt under the action of temperature effects. In addition, the database was sup-

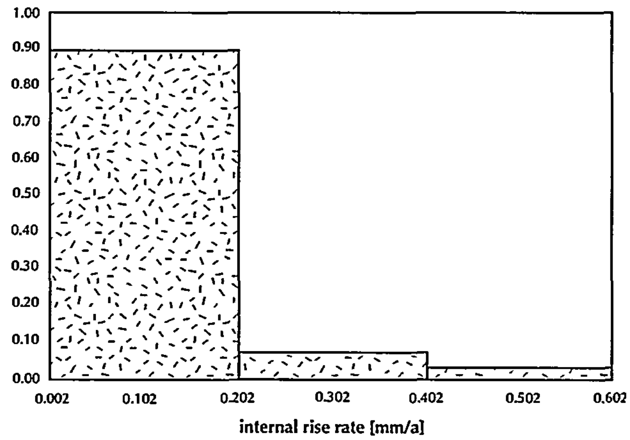


Figure 17.2. Probability density of the average internal rise rate for salt domes in the Netherlands and Germany (Prij et al., 1993a).

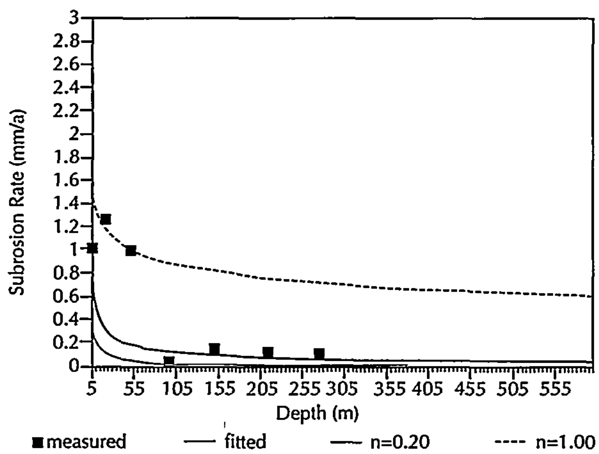


Figure 17.3. Input parameter of the depth-dependent subrosion rate. The measured values have been calculated for salt domes in Germany and the Netherlands with predominantly Quaternary subrosion activity. The upper line corresponds to a power-fit with exponent -0.2 and for the lower line this value is -1.0. The middle line with exponent = 0.64 has been used in the calculations (Prij et al., 1993a).

plemented with data on the subsurface which have become available by the geological mapping of the Netherlands and from geological exploration of the North Sea. This material also made a contribution towards improving the precision of the data.

In Phase 1, three important models were developed and applied for making risk calculations:

- release and transport model for radionuclides from the rock salt;
- transport model for radionuclides in the subsurface; and
- transport and exposure model for the biosphere.

These models underwent limited practical validation in Phase 1. Supplementary research proved the possibility of developing these - essentially generic - models further and validating them against a number of actual practical situations and data. For instance, the model describing the transport of radionuclides in rock salt was further confirmed with results from *in situ* experiments (the Asse Mine in Germany), i.e. under the action of such temperature and pressure as occur in an actual disposal situation. These results indicate that during the initial period of disposal, subsurface spaces in rock salt are closing by convergence more rapidly than was previously assumed. Further validation of this convergence process with reference to more practical data is needed in view of its great impact on safety.

In recent years, the model for radionuclides transport through the subsurface, named METROPOL, has successfully undergone comprehensive international validation in the INTRAVAL project (INternational TRAnsport model VALidation; Leijnse & Hassanizadeh, 1994; Van de Weerd et al., 1994; Van Weert et al., 1994; Leijnse et al., 1996 submitted). Comparison with other models within this project confirmed the reliability of this model. Its applicability to an actual situation was also demonstrated. For conditions prevailing in the Netherlands, this is supported by a case study on the Zuidwending salt dome (Oostrom et al., 1993).

The above-mentioned studies on processes and models and in particular the practical validation of these models have yielded greater clarity on the question of which data are needed for an adequate description of processes such as convergence, diapirism, subrosion and erosion.

17.3.3 Risks of Considered Scenarios

As stated above, by comparison with the approach in Phase 1 the present safety approach has yielded an improved insight into the uncertainties associated with the risks of geological disposal. The risks as now calculated are therefore to be regarded as more precise. Using the PROSA method to calculate the risks of the subrosion/diapirism scenarios yields values smaller than 10^{-9}/year. Similar risk values were found in Phase 1.

For the flooding scenarios, improved insights as compared to Phase 1 allowed modification of the rate of the process whereby subsurface spaces in rock salt are closing (convergence). The resultant radiation doses following dispersal of the radionuclides in the biosphere are found to be very low, and the associated risks much lower than as calculated in Phase 1, namely smaller than 10^{-10}/year.

The above-mentioned model modification for the convergence process in rock salt also has an important influence on the human intrusion scenario further considered (a leaking storage cavern near the repository). The risks now calculated are significantly lower than

Phase 1. For the human intrusion scenarios not analysed in PROSA, the risk appearing from the Phase 1 figures is smaller than 10^{-6}/year. Because large variations can occur in the transport of nuclides in the salt formation, in the overburden and in the biosphere, there is a large spread in the resulting risks calculated. Nevertheless, for all the scenarios considered the disposal risk remains smaller than 10^{-6}/year, and for "natural" scenarios (subrosion, diapirism, flooding) even smaller than 10^{-9}/year.

17.3.4 Risk Determining Features

Certain features of a repository and the geological barrier system around it may be determining for the risk of this particular facility. These features can be identified by determining the influence of all possible FEPs on the risk (sensitivity analysis). Phase 1 showed that the approach then used did not possess sufficient discriminating power to identify risk-determining features of a repository and the multi-barrier system. The principal drawback lay in the conservative approach and the uncertainty as to the extent thereof for each of the disposal situations considered.

Development of the PROSA method made it possible to eliminate this drawback by allowing a systematic approach and introducing the probability aspect. This makes it meaningful to perform a sensitivity analysis to identify risk-determining features of the multi-barrier system. From this sensitivity analysis, it follows that on the basis of the currently available limited database there are two features which predominantly determine the risk of the "natural" scenarios, namely:

- depth of the repository; and
- rate at which this facility is uplifted by possible diapirism.

To summarise, it may be stated that application of the currently available version of the PROSA method to a number of important scenarios and disposal concepts has proved extremely useful. In the first place, this system, as indicated above, provides an improved insight into the risks and uncertainties. Secondly, it helps, for example by means of the risk-determining features, to guide the direction of research in the various sub-areas of the OPLA programme.

17.4 TECHNICAL FEASIBILITY

In Phase 1, two different disposal techniques were studied:

- conventional (dry) mine, and
- deep boreholes drilled from the surface, in combination with caverns.

After study, it was concluded that both techniques are, in principle, technically feasible for salt formations in the Netherlands.

For the mine concept, the supplementary research programme provided further confirmation for that conclusion with the following findings:

- In co-operation with Germany, the practical feasibility has been demonstrated of a mine provided with deep boreholes, dry-drilled from a gallery to a depth of several hundred metres. These boreholes, intended for disposal of canisters with fission waste, constitute an important part of the mine concept.
- Within the same co-operative project, the practical development of a system for lowering, transporting and emplacing radioactive waste canisters in a disposal mine according to the German design has been successfully completed. In addition, the availability of a test facility for *in situ* investigation into rock salt has made a major contribution towards the acquisition of mining engineering experience (the HAW project in the Asse Mine; Prij & Hamilton, 1992; see also section 17.6.1).
- Important advances have been made with regard to technical feasibility, especially relating to the convergence process of rock salt (see section 17.6.1 for further information).
- As a result of the effects of radiation and heat, gas formation and radiation damage can occur in the vicinity of the radioactive waste disposed of in the rock salt. Conclusions regarding the significance of the two phenomena for mine design and disposal safety are still of a preliminary nature (see section 17.6.2 for further information).

With regard to the "deep boreholes from the surface combined with caverns" disposal technique, Phase 1 found that inspection of the reliability of the borehole and cavern seals was an item demanding special attention. In spite of supplementary literature research, no essential progress has been made in this area (Technische Universiteit Delft, 1993b). On the other hand, it has been found that the depth range of this disposal technique exceeds the range found in Phase 1. Depending on the diameter, depths down to around 3000 m are technically feasible with the deep borehole

technique.

17.5 EARTH SCIENTIFIC BACKGROUND STUDIES

The major tasks of earth scientific research for radwaste disposal are the description of the present state of the geological barriers (site characterisation) and the prediction of the future state of the geological barriers in support of the scenario analysis.

The supplementary research in the areas of geology and geohydrology consists of two parts:

- improving data and models to support the risk calculations in the safety study (see also section 17.3); and
- increasing the basic knowledge which is needed for proper understanding of important processes in the subsurface. This item includes a number of subjects which emerged as recommendations from Phase 1, and which are not yet immediately applicable in the safety study at this time. Results will be discussed in this section.

17.5.1 Data Quality and Methods of Data Collection

Uncertainty Analysis of Available Geological Data

This research topic deals with the reliability of spatial geological data in general and more specifically with the depth values of geological horizons on contour maps and vertical profiles which are available for the research in the OPLA programme (Wildenborg et al., 1993). An uncertainty analysis of geological data was not carried out during Phase 1 of the OPLA programme.

For this analysis, the following approach was used; various sources of error are responsible for the differences in depth values between observed and real values. Those error sources are introduced by the acquisition of basic data (borehole data and seismic reflection data), by processing and interpretation and also during the construction of derived spatial data like contour maps and vertical profiles. After analysis of the various error sources involved, the magnitudes of the error sources are quantified in standard deviations of the difference between real and observed depth. Assuming mutual independence, all standard deviations of the sources of observation errors could be summed up to obtain insight in the total standard deviation of the observation error :

$$s^2 = \frac{1}{n-1}\sum_n (z_w - z_e)^2 \qquad (1)$$

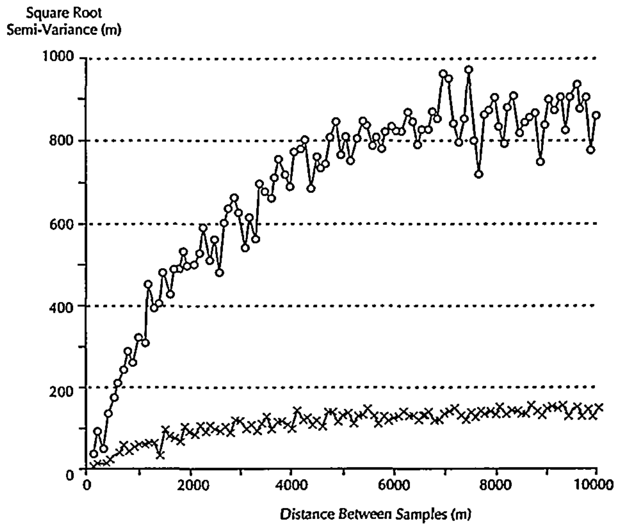


Figure 17.4. Semi-variogram of depth data derived from seismic observations.

where: s = standard deviation, z_w= observed depth, z_e= expected depth, n= number of observations

This analysis was performed on derived spatial data, namely contour maps and vertical profiles (Fig. 17.4).

Borehole data and seismic reflection data are both used as original data for the construction of the spatial data mentioned. Borehole data are highly reliable. The standard deviation of the observation error of the depths to the base of the Tertiary and the top of the Zechstein are estimated to 2 m. Seismic reflection data are less reliable; the standard deviations of the observation errors could be as high as 125 m.

To construct a contour map, interpolation of the original data is necessary. Interpolation by itself is another source of error which increases with increasing distance from data points. The interpolation error also depends on the structural complexity (average dip) of the geological horizon). Using the estimations of the standard deviations of the observation error of borehole and seismic reflection data and the standard deviations of the interpolation error, it was possible to construct maps showing the place-dependent standard deviation of the top of the Zechstein around salt structures, and the base of the Tertiary above the salt.

Methods of Data Collection

Data on the shallow subsurface are available to a reasonable extent when it comes to investigating and modelling important geological and geohydrological

processes for use in the safety study. When it comes to data on deeper formations, for example the composition and structure of rock salt and caprock, the rate at which salt dissolves in groundwater, salt concentrations in groundwater, and the diapirism rate of salt formations, they are inadequate.

The OPLA Committee therefore advised that two studies should be commissioned regarding the possibilities for collecting such data. One study concerned the development of equipment for the acquisition of reliable hydrological data at greater depth. An initial practical test has demonstrated the practicality of the technique chosen, and that it is suitable for further development (Fig. 17.5).

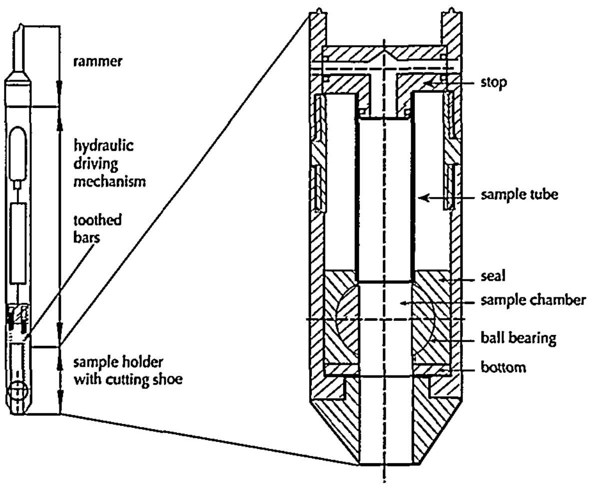


Figure 17.5. Apparatus designed for collecting geochemically undisturbed sediment samples in boreholes (Delft Ground Mechanics, 1993).

Another study was focused among other things on the possibilities of advanced seismic methods in particular for the purpose of high-resolution determination of the depths and size of salt formations and the structure of the overburden (Table 17.1). Here, too, practical tests indicate that progress has been made.

Caprock Properties

Caprock refers to the rock which may be formed at the top of salt formations by the dissolution and removal of rock salt, while poorly soluble constituents remain and newly formed constituents are deposited. Knowledge concerning this rock is important because of the possible effects on mining engineering activities in or near caprock.

A literature study was performed in order to gain further insight into the processes which determine the formation and properties of caprock. That study made it clear that uncertainties still exist about the formation of caprock and the course of the processes relevant to it, and that little is known about the caprock of rock salt formations in the Netherlands. In a general way, it may be stated that it is important to allow for the properties of local caprock when constructing shafts.

A 6 m core section from the lowest part of the caprock was available from the Zuidwending salt dome. Detailed petrographic, geochemical and isotopic analyses were made of this core. The composition of the lowermost part of the caprock evidences a low salinity subroding brine. From the texture of the caprock it could be concluded that fluid flow was predominantly sub-horizontal. The lack of a vertical chemical gradient in the gypsum part of the caprock argues convincingly against any salinity gradient in the water in the caprock. Diffusion processes must therefore be regarded as insignificant.

17.5.2 Prediction of Future State of Geological Barriers

Long-term future predictions of the geological system are of great importance for the risk assessment of radioactive waste disposal. A better understanding of the

Table 17.1. Resolution of various geophysical reconnaissance methods with depth in meters (Meekes et al., 1993).

Depth interval (in m)	Gravity Method	Geo-electrical method	Electromagnetic method	Seismic refraction method	Seismic reflection method
1-50	1-25	1-50	1-50	1-10	no signal
50-100	25-70	15-100	15-100	10-20	3-15
100-300	70-150	30-300	30-300	20-60	4-20
300-500	150-200	100-500	100-500	60-100	5-25

physical and chemical mechanisms of the processes that affect the geological barriers, contributes significantly to a more reliable prognosis of the future state of the subsurface. For rock salt, three processes have been identified that may directly influence the geometry of the rock. These are salt movement, salt dissolution and erosion. Modelling studies have been performed to investigate their relation with the major climatic and geodynamic forces, and their effect on the geological barriers (Wildenborg et al., 1993).

The possible future states of the natural barriers are in general introduced as discrete scenarios in the safety studies. A drawback of this approach is that the total effect of all natural release scenarios is difficult to assess. Geological processes are in varying degree coupled to one another, which makes it difficult or even impossible to split the system in several autonomous compartments. Therefore, it is advisable to pursue an integrated approach parallel or complementary to the scenario method, in which the whole natural barrier system, the processes inclusive, is considered as one entity.

An example of a coupled system is the relation between climatic processes and surface or near-surface processes. Our knowledge of the impact on geologic disposal of recurrent ice ages, induced by climatic changes, has been substantially expanded (Wildenborg et al., 1990). The geological cycles concerned are relatively short, with a length of about 100,000 years, and have almost completely dominated the surface or near-surface processes in Northern Europe for the past one million years. This knowledge, based on comprehensive observations of deposits formed during a number of past cycles, makes it possible to model a large part of these processes in their relevance to the disposal issue. This holds especially for the effects of salt dissolution and erosion on the salt barrier.

Barrier Model

A geological barrier model ideally should simulate the future evolution of the barriers over a period of 10^5 to 10^6 years, should comprise physical and chemical formulations of all relevant processes that directly or indirectly affect the future barrier state and is driven by time-dependent climatic and tectonic forcing mechanisms. Since the beginning of the eighties, various models have been developed to simulate the future behaviour of the geological system in connection with radioactive waste disposal, e.g. GSM and FFSM in the United States, CASTOR in France and TIME4 in Great Britain.
Central in the barrier model to be developed is the local submodel describing the geometry and the properties of the various geological barriers around the disposal facility. The locally operating processes like salt dissolution, salt movement and erosion are incorporated in this part of the model. The climatic and geodynamic boundary conditions are generated in supraregional submodels on the scale of NW Europe or on an even larger scale. Geological observational data have to be gathered in a systematical way to provide input data for the model and data for extensive testing of the model output (Wildenborg et al., 1995).

The mechanisms of the processes that may directly affect the salt barrier (Fig. 17.6), have been studied in detail and will be discussed briefly here.

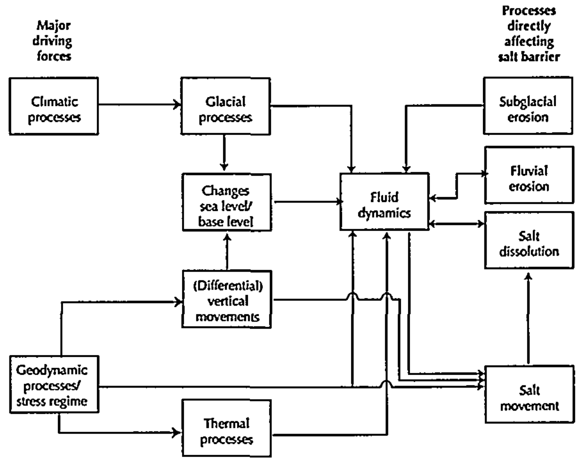


Figure 17.6. Relation diagram of the major processes to be incorporated in the geological barrier model. Note that only part of the pertinent relations between the processes have been marked in this diagram.

Process of Salt Movement

Observations in the Netherlands on- and offshore region show that salt structures are geographically related to faults in the salt basement (Remmelts, 1996; see also Fig. 17.7). The timing of salt movement is related to tectonic phases. For example, the average rise rates of salt structures during the early part of the Oligocene, a period of tensile intra-plate stress, were three to five times higher than during the Late Tertiary and Quaternary, which period was governed by compression.

The role and timing of changes in intra-plate stress also appears on a basin-wide scale. An increased level of compressive stress during the Late Neogene in the North

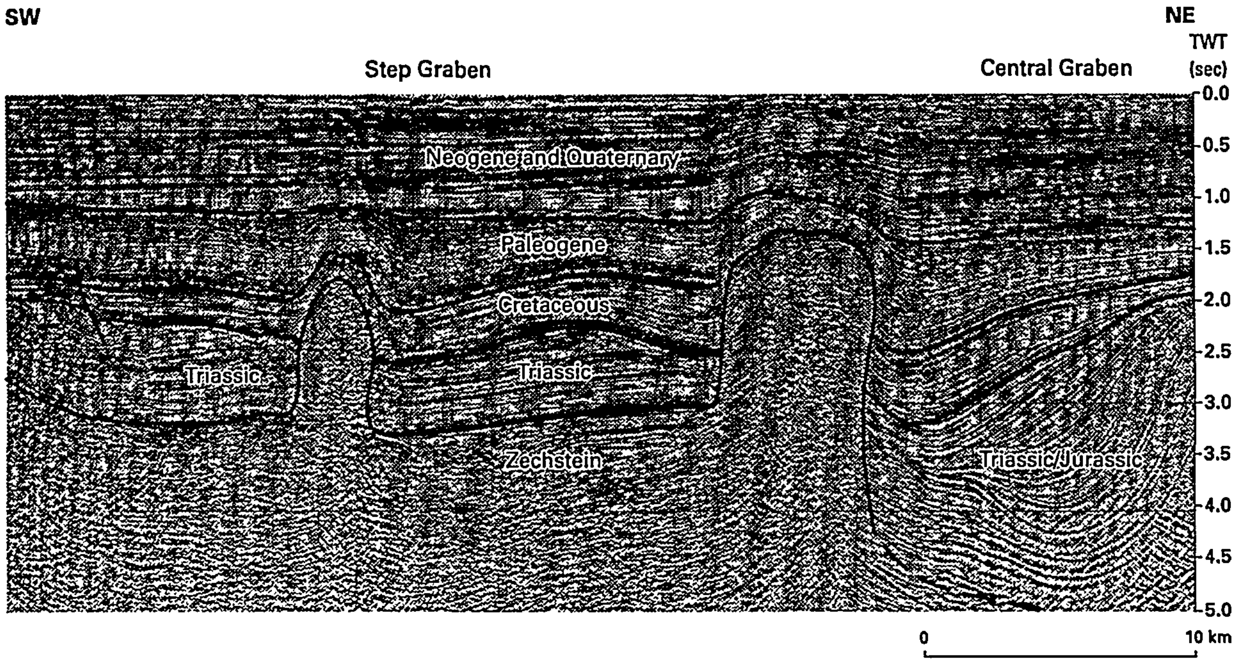


Figure 17.7. Seismic section across salt structures in the offshore region of the Netherlands.

Sea basin is reflected in high subsidence rates. These high rates can not be explained only by the classical lithospheric stretching-cooling model of McKenzie (Cloetingh & Kooi, 1992). Increased rate of uplift during the Late Neogene has been reconstructed for peripheral parts of the North Sea Basin (Van den Berg et al., 1996 submitted). The change in stress level probably is related to important plate-boundary reorganisations. The compressive stress regime is still dominant at present in Northwest Europe (Müller et al., 1992). It is marked by a NW-SE direction, related to the Africa/Europe collision and to ridge-push forces operating in the northern Atlantic.

The effect of intraplate stress on salt movement has been investigated in numerical model studies (Daudré & Cloetingh, 1994). The numerical experiments have been carried out with a two-dimensional planar model of a visco-plastic salt layer and a brittle overburden. Three tectonic scenarios have been investigated: one in which buoyant forces are the only driving mechanism, one in a tensile stress regime and the last with a compressive stress regime (see also Fig. 17.8).

The first modelling experiment has revealed that buoyant forces are too weak to overcome the yield strength of the overburden. Diapiric movement of the salt occurs in a tensional regime with the simultaneous development of shear bands in the overburden. The modelling for a regime of compressional intraplate stress has shown a more modest salt movement and the development of various systems of shear bands.

These results strongly suggest that the classical theory of a gravitational drive for the initiation of salt movement probably does not hold. It appears that regional tectonic forces are a prerequisite for the start of salt movement. The most favourable situation for the initiation of salt movement is a tensional regime; re-activation of diapirism under compression appears to be quite important.

A better understanding of the rheological properties of the overburden is necessary for a more reliable simulation of diapiric deformation. Better understanding of the dynamics of salt diapirism obviously depends on better constraints on the interplay of fluid flow and the initiation of diapirism on a basin-wide scale. Especially, the experiments with compressive stress need further attention, since the fluid pressure in the sedimentary rocks increases significantly in a compressive regime (Van Balen & Cloetingh, 1993). Fluid pressure in rocks is a major factor in determining the rheological properties of the rock. More attention should also be paid to the effect of sedimentation and erosion on the development of salt diapirs (Poliakov et al., 1993).

Process of Salt Dissolution and Climate-Induced Hydrological Changes

The dissolution of salt diapirs is primarily controlled by the amount of available groundwater around the salt

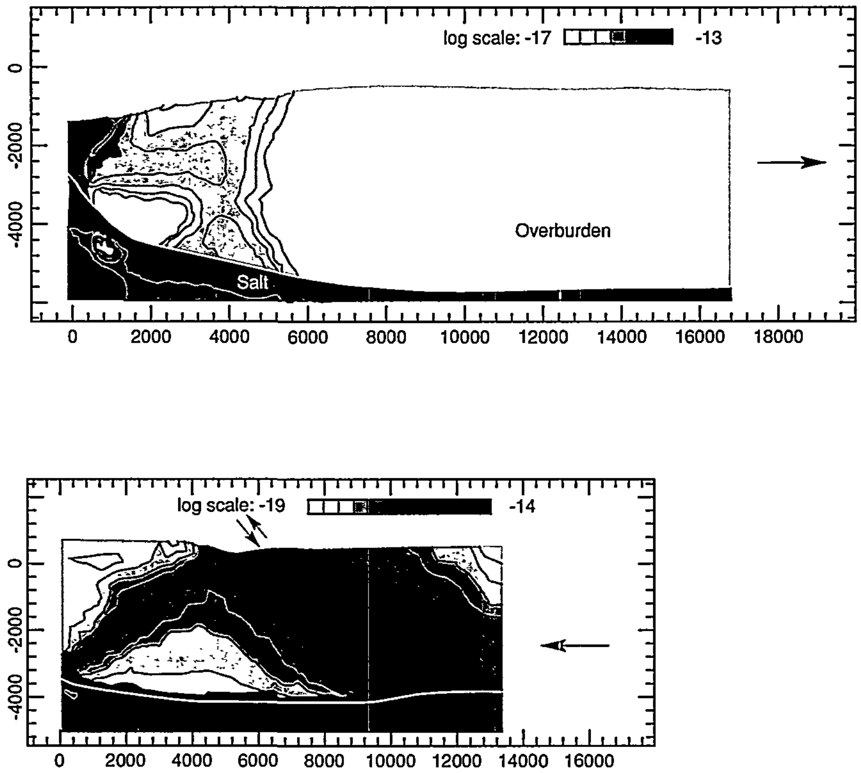


Figure 17.8. Results of mechanical modelling of diapirism. Above: Strain rates after 30 Ma in a tensile regime; strain rates in the range of log(10^{-17} - 10^{-13}/s). Below: Strain rates after 25 Ma in a compressive regime; strain rates in the range of log(10^{-19} - 10^{-14}/s) (Daudré and Cloetingh, 1994).

structure and the solubility and dissolution rates of the evaporite minerals present in the salt dome. In turn, these parameters are controlled by a complex set of other, frequently interrelated parameters such as temperature, pressure, fluid flow composition (i.e. the undersaturation of the groundwater with respect to NaCl), and the permeability of the surrounding sediments. Effects of temperature and pressure on dissolution rates and solubilities are well known from the literature. Subrosion rates as well as solubilities increase with fluid flow velocity and temperature. Since fluid flow velocity is the dominant effect and flow velocities in aquifers in the northeastern Netherlands rapidly decrease with depth, subrosion rates of salt diapirs are expected to decrease rapidly with depth.

The effects of climatic change on groundwater flow and consequently on subrosion at the top of diapiric structures have been investigated and modelled on different scales (Edinburgh University et al., 1996 in prep; Oostrom et al., 1993). Hydrogeological and (palaeo-)environmental data sets have been prepared for application in groundwater flow and subrosion models for:

- the salt-tectonically disturbed subsurface of the north eastern Netherlands for which a regional model and a local model have been developed in a joint project called SESAM in which the Geological Survey of the Netherlands (RGD), the Dutch National Institute of Public Health and Environmental Protection (RIVM) and the TNO Institute of Applied Geoscience participated (see also Fig. 17.9. Van Gijssel, 1995).
- the Cainozoic and Mesozoic subsurface of the Northwest European lowlands, in the scope of an EC-funded, joint project of the University of Edinburgh, RGD, RIVM and the University of Paris-Sud and the Catholique University of Louvain-la-Neuve (Boulton et al., 1993).

Glacial cycles associated with ice sheet expansion and permafrost conditions beyond are believed to have had a major impact on the groundwater flow patterns and hence subrosion rates. In order to approximate this impact, a time-dependent, thermo-mechanically coupled flow line model has been developed in the EC-funded project and applied to a supra-regional transect from South Sweden to northern France, so as to match the

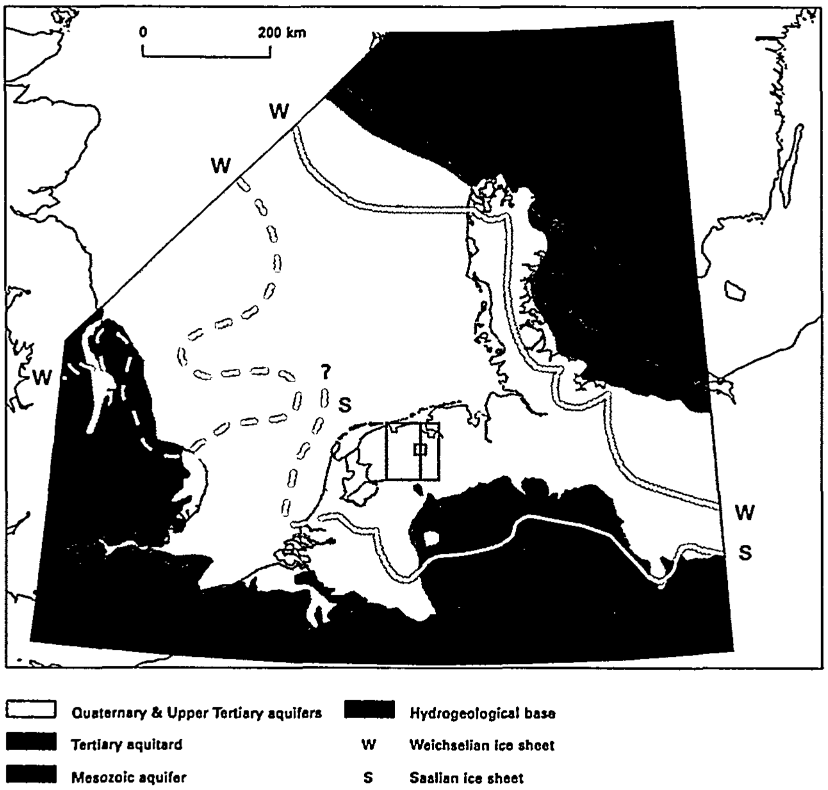


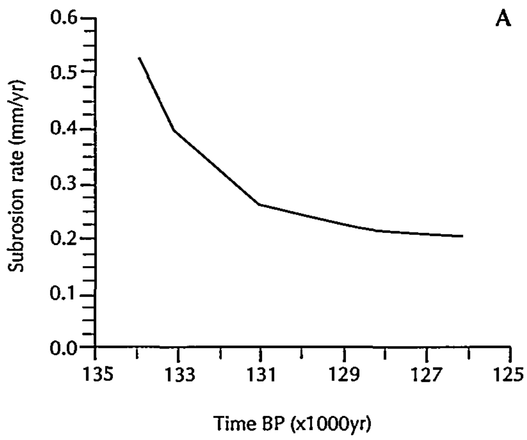
Figure 17.9. Location of the local and regional model areas in the northeastern Netherlands used for the simulation of salt dissolution and the boundaries of the major ice advances in the Late Quaternary. The major part of the Zechstein salt structures in the Netherlands are present within the regional model area.

inferred Weichselian and Saalian glaciations (Boulton & Van Gijssel, 1996). A general subglacial groundwater flow model for NW Europe, consisting of several components and interfaces (an ice-sheet model, a sea-level model, a large-scale groundwater flow model and a subrosion model for rock salt) has been applied to a large number of potential geological situations (Van Weert et al., 1996). The simulations corroborate the importance of the regional hydrogeological conditions and the permafrost distribution beyond the ice sheet margin in controlling ice and groundwater flow.

Drainage and topography have been dramatically changed by the subsequent ice sheet advances. The northward flowing rivers in northwest and central Europe responded by changing their courses in a westerly direction. Lakes that were impounded by glacier ice often drained suddenly, in particular during the deglaciation of the ice sheets and melting of the permafrost. Palaeo-channels formed by abnormally high discharge events are frequently found in the subsurface of the marginal zones of the Pleistocene ice sheets. Deep and penetrative flushing of aquifers by glacial meltwater may have lead to increased subrosion rates of salt diapirs in contact with these aquifers.

The large-scale flow boundary conditions provided by the EC-funded project have been used in the regional and local model simulations for the northeastern Netherlands, carried out in the scope of the SESAM-project, which was focused on one salt diapir only. On the basis of the schematic geometrical frameworks and averaged values of the geohydrological parameters, numerical hydrological models have been reconstructed to evaluate the groundwater flow systems and subrosion rates in the vicinity of selected salt structures in the northeastern Netherlands. The simulations have been concentrated in the first instance on present-day boundary conditions.

In order to investigate the effects of climate change on the groundwater flow patterns, subsequent modelling experiments have been performed using estimated hydrological boundary conditions for relevant time intervals during the Late Quaternary. Regional 3-D groundwater flow simulations on the basis of recon-



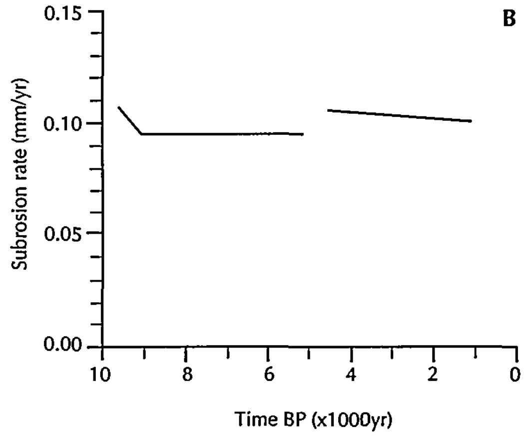


Figure 17.10. Simulated subrosion rates for different time spans. A. Late Saalian ice-free periglacial conditions; B. Interglacial conditions; 2-D simulations, heterogeneous flow field, permeability $k_{65}=10^{-14}$ m^2 (Oostrom et al., 1993).

structed isohypse maps for different post-Saalian, steady-state time intervals in the ice-free area of the northeastern Netherlands clearly indicate the effects of changing climatological conditions on the upper groundwater system (Oostrom et al., 1993). Groundwater flow velocities were a factor two higher in the period following the deglaciation of the Saalian ice sheet as a consequence of large differences in relief and low sea level stands. Subsequent relief levelling and sea level rise during the Eemian interglacial led to low potential gradients and hence resulted in low groundwater flow velocities. Groundwater flow velocities during the Weichselian cold stage with periglacial conditions, were similar to those during the interglacial periods.

Using these regional boundary conditions for a local model, simulating the subrosion rates at the top of a selected salt diapir by the METROPOL-3 code (finite element mesh: Sauter et al., 1990), showed the influence of the magnitude of the groundwater flow velocities near the salt diapir on the subrosion rate. The subrosion rates simulated for the Late Saalian period are found to be substantially higher than for the other periods (Fig. 17.10). Significant differences in subrosion rates between the interglacial periods and the Weichselian time intervals with permafrost conditions, could not be assessed. The input parameters for the sensitivity analyses, such as permeability and anisotropy ratio, show that the resulting range in the simulated groundwater flow velocities and subrosion rates exceed the values calculated for the past.

The subrosion simulations in many ways are simple representations of natural conditions, particularly the 2-D simulations. Since testing of the models used is not possible, because adequate field data and predictions covering thousands of years are lacking, emphasis has been put on calibration for present-day conditions, detailed sensitivity analysis, groundwater flow simulations of past time intervals (which are not necessarily similar to future observations) and comparison of calculated subrosion rates with values known from the literature.

Process of Fluvial Erosion

Fluvial erosion and sedimentation are the results of complex process interactions. Quaternary fluvial dynamics in the Netherlands are mainly governed by the combined effects of sea-level fluctuations, tectonics and climatic change. In addition, and dependent on the space and time scales used, a fluvial system can be controlled by other more local factors and processes (Wildenborg et al., 1993). These complex interactions of processes and factors within the fluvial system were integrated within the model FLUVER (Veldkamp & van Dijke, 1994). FLUVER is a finite-state model using quantitative and qualitative relationships of fluvial systems acting over a long time span (500 ka). Model construction, organisation, operation and testing are extensively described in previous papers (Veldkamp, 1991; 1992). FLUVER simulates the evolution of a fluvial landscape by both sedimentological and erosional processes as controlled by relief, climate, tectonics and fluctuations in base-level of erosion (sea-level changes). The model processes are long-term and large-scale analogies of real erosion and sedimentation processes which react to

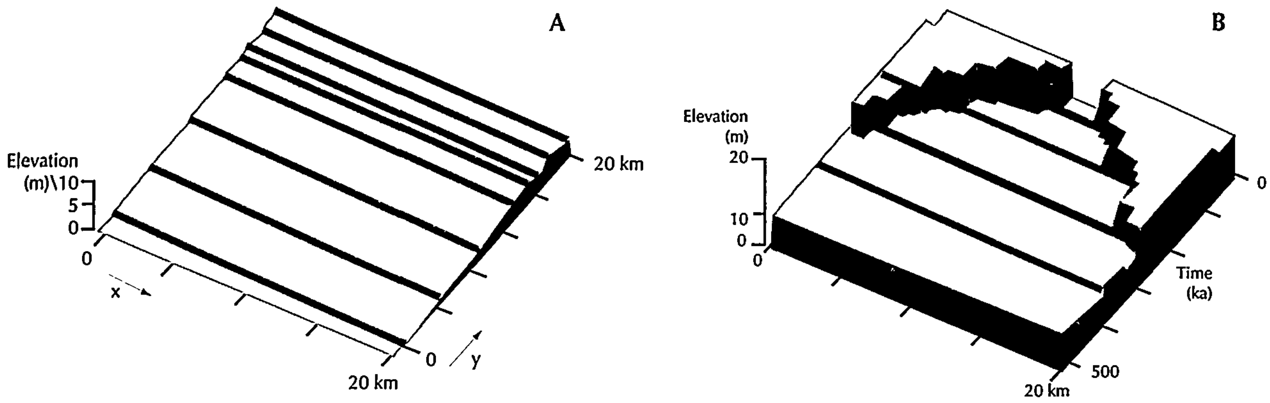


Figure 17.11. Simulation of fluvial landscape in a tectonic subsidence scenario. A - Dynamic evolution of a cross-section through time; B - Landscape after 500 ka.

changes in the climate controlled discharge-sediment load equilibrium which is calculated for time steps of 2 ka each. Both two ka discharge and two ka sediment load are assumed to be a function of climate change as described by the astronomical parameters of Milankovitch theory. Furthermore, both relationships are supposed to be out of phase (Veldkamp, 1991; 1992).

When sediment-input load would exceed the sediment transport capacity, the excess volume will be deposited, and in case the transport capacity exceeds the input load the difference is eroded in the simulated system. The zone with active erosion migrates headward along the longitudinal profile, while the sedimentation zone migrates in a downward direction. Except for changes in climate the simulated fluvial system also reacts to changes in base-level of erosion and vertical crustal movements. In case of uplift or lowering of the base-level of erosion, the fluvial system will strive to compensate by erosion, which is not always possible since climatic conditions may be such that the simulated systems tends to deposition. A similar relationship applies for crustal subsidence and rise of the base-level of erosion which will stimulate sedimentation in the simulated system.

Several simulation runs were made, demonstrating that vertical fluvial erosion in the Netherlands is strongly controlled by sea-level fluctuations and vertical tectonic movements. An example of the simulations with tectonic subsidence is illustrated in Figure 17.11. The tentative conclusion is that during the next few hundred thousand years, fluvial erosion will probably not affect buried salt structures, assuming that climate and change in the base-level of erosion can be satisfactorily described with the Milankovitch theory, tectonic activity in the Netherlands will not change considerably and that no larger fluvial systems than the Rhine or Meuse will enter the Netherlands.

Although reliable long-term input for future conditions is lacking and FLUVER simulations have conceptual validity only, FLUVER can demonstrate potential long-term fluvial erosion effects. To arrive at a more quantitative reliable model, it will be necessary to collect more quantitative data on site-specific fluvial records. Since this model study was mainly based on general data, a comparison and evaluation with actual field data could also improve model performance. Extended and well dated fluvial records of subsiding and uplifting regions like fluvial basin infills and fluvial terraces may serve as a primary data set to which FLUVER or similar models could be tuned. A first tuning exercise was done for the Meuse terraces at Maastricht (Veldkamp & Van den Berg, 1993). As long as extensive testing exercises have not been carried out FLUVER has no reliable quantitative forecasting status.

Process of Subglacial Erosion

From the variety of erosional features related to glacial activity, the formation of very deep tunnel valleys is undoubtedly the most important with respect to the stability of the geological barrier (Wildenborg et al., 1993). Very deep tunnel valleys in northwestern Europe of Elsterian age where analysed on their geometry and possible genesis in order to define tunnel-valley process conditions. Mathematical translation of the formation processes were subsequently used for the construction

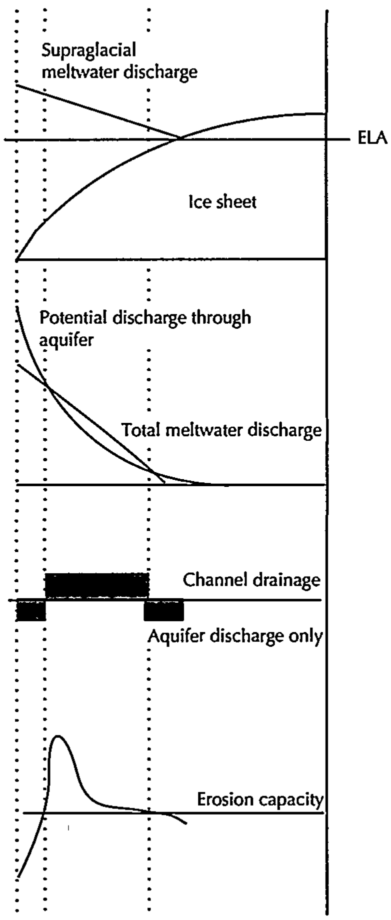


Figure 17.12. Graphical presentation of the conceptual model SUBGLER. ELA = equilibrium line altitude. Below ELA supraglacial meltwater is produced which is assumed to be added to the subglacial meltwater. Drainage through subglacial channels and subsequent erosion occurs if the production of subglacial meltwater is larger than the discharge capacity of the underlying aquifer.

of a computer simulation model (SUBGLER). The aim of the model construction is the prediction of the erosional effects of a future glaciation in the Netherlands.

In accordance with generally accepted theories concerning the development of tunnel valleys, SUBGLER simulates the longitudinal one-dimensional flow of meltwater in subglacial channels under glaciostatic pressure (Nye, 1976; Shoemaker, 1986; Boulton & Hindmarsh, 1987; see also Fig. 17.12). Furthermore, sediment transport functions were developed to relate erosion to the amount of subglacial melt water.

Model construction demonstrated that subglacial channels under a large continental ice sheet overlying an area consisting of a thick sandy aquifer, comparable to the pre-Elsterian conditions in the Netherlands, are only to be expected during the last phase of a glaciation. During this phase, large volumes of supraglacial melt water are assumed to be added to the subglacial system (Jeffery, 1991). The incorporated influence of supraglacial meltwater makes the model sensitive to climatic changes during deglaciation.

Preliminary modelling conclusions after several test runs are based on the individual contribution of various selected input parameters (see also Fig. 17.13). The relative significance of the input parameters are (in decreasing order of importance): mean summer air temperature at the ice sheet margin, longitudinal ice sheet profile, rate of ice sheet margin retreat, and hydraulic conductivity of the aquifer.

The simulations suggest that deep glacial erosion can be expected during any future glaciation, but no reliable estimations can be given on their maximum theoretical depths. In order to allow more precise simulation of future behaviour of subglacial systems, considerable geological field data on Quaternary tunnel valley formation is required. More information is needed on water pressures in subglacial aquifers, sediment loads in subglacial channels and the contribution of supraglacial meltwater to the subglacial system. Current research on the subglacial hydrologic system shows that modelling of groundwater flow within the vicinity of subglacial channels enables a more precise assessment of maximum depths of tunnel valleys (Van Dijke & Veldkamp, 1995).

17.6 Interaction Between Waste and Host Rock

17.6.1 Effects of Changes in Temperature and Stress

The behaviour of rock salt as host rock under the action of temperature and pressure was comprehensively studied in Phase 1, both *in situ* (the Asse Mine) and in the laboratory (Prij et al., 1995). In the supplementary research programme, the study into rock mechanics was concentrated on a continuation of the *in situ* investigations. Results obtained in that context concerning the convergence process of rock salt, which are important for the safety studies, have yielded an increased understanding of this time-dependent process in geological

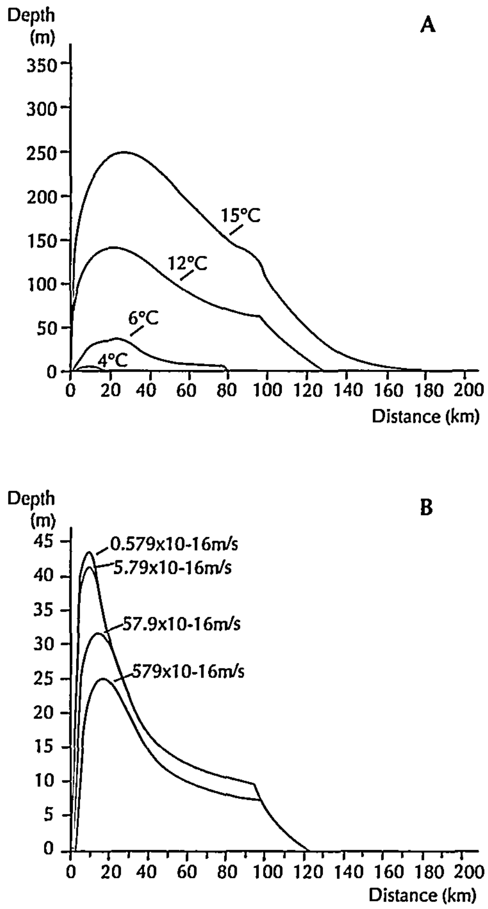


Figure 17.13. Results of the simulations with SUBGLER. A. Effect of changes in air temperature; B. Effect of changes in hydraulic conductivity.

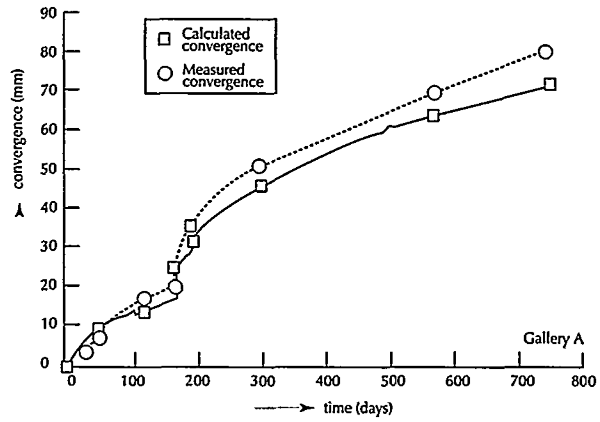


Figure 17.14. Convergence in the mining gallery in the HAW-experiment (Prij et al., 1993b).

spaces. The convergence process initially proceeds faster than was previously assumed, but after some time it proceeds much more slowly (Fig. 17.14; Prij et al., 1993).

Further modelling of the thermomechanical behaviour of rock salt and practical validation of the models against measurement results have improved the reliability of these models. This made it possible to improve the quality of predictions on the behaviour of a subsurface disposal space under the influence of temperature and pressure. This is important for any modifications and optimisation of repository designs.

The Netherlands Energy Research Centre (ECN) is cooperating with German, French and Spanish research institutes on a number of joint *in situ* projects in the Asse Mine (ECN, 1993b; Prij & Hamilton, 1992; Heijdra & Prij, 1996, in prep). One of those projects, the High Active Waste (HAW) project is a demonstration project for the disposal of high-level radioactive waste. In particular, the interaction between salt and thermogenic waste is being studied here in a practical situation.

17.6.2 Radiation Effects

Disposed-of waste produces radiation. This radiation can cause abnormalities (damage) in the crystalline structure of the rock salt. As a result, energy builds up in the damaged salt, and later it can potentially be released. The results of the Phase 1 research showed that the build-up of radiation damage in salt depends strongly on the presence of certain natural contaminants, such as potassium, bromine, lithium, etc. Uncertainty as to the maximum level of radiation damage under disposal conditions meant that the Phase 1 calculation of the effects of energy release allowed for a damage rate of several tens of percentage points. On the basis of preliminary calculations and assuming a spontaneous and total energy release, these consequences were found to be very limited in scale and did not lead to the release of radionuclides from the salt.

Supplementary laboratory investigation has yielded a number of fresh results (Groote et al., 1991; Seinen et al., 1992; 1995; Garcia Celma et al., 1993). The complexity of this subject matter was such that new questions arose which have not yet been answered. These questions concern the initial stage of the damage, the form and dimensions of the damage nuclei (the colloids) and the influence of such factors as temperature, radiation dose, dose rate, pressure, deformations and natural contaminants (Prij, 1996a, in prep a; Garcia Celma et al., 1995). Insight into these is needed to improve the calculation model that predicts radiation damage under disposal conditions. The foundation for the calculation

model was laid by Jain and Lidiard, and in that form it was applied in Phase 1. Progress has been made in the matter of model development, but uncertainties have continued to exist as well, for instance with regard to the causes of the action of contaminants on the damage level.

Laboratory investigations into the properties of strongly irradiated salt (up to radiation doses 6 times higher than under disposal conditions) has shown that, under certain conditions, the damage can rise to around 10%. It has furthermore been found that higher doses are capable of causing even more damage (Rijksuniversiteit Groningen, 1993; Weerkamp et al., 1994).

In some cases, a very rapid reaction was observed, taking an explosive course. Further investigation into the nature of the release process is underway. The significance of this process under disposal conditions merits special attention.

Since Phase 1, good progress has been made with modelling the effects on the surrounding rock salt of the above-mentioned release, capable of allowing for a spontaneous, very rapid and total release of all the energy built-up with radiation damage. The approach taken, which is still subject to scientific debate concerning the release rate and the accompanying temperature rise, broadly confirms the conclusion as drawn in Phase 1 that the effects of sudden energy release would be very limited (ECN, 1993a; Rijksuniversiteit Groningen, 1993; Prij, 1996; Prij, in prep b).

In view of the above, it appears meaningful to continue the investigation into the consequences of radiation damage for disposal safety. For that purpose, the chemical stability of salt and glass, as well as design engineering aspects relating to the repository, can play an important role.

17.7 ALTERNATIVE DISPOSAL CONCEPTS

Current developments in the public interest sphere demanded attention for the following subjects:

- retrievability of disposed-of radioactive waste;
- direct disposal of spent fuel elements.

In an exploratory way, it was examined whether these developments can be integrated in the Phase 1 disposal concepts at the technical level. Their effects, if any, on disposal safety have been considered as far as possible.

17.7.1 Retrievability of Disposed-of Radioactive Waste

Discussions held in wider context and, among other things, an OPLA preliminary study into retrievability disclose a central question: what is envisaged with retrievability? A number of considerations differing widely in nature may be relevant. At any rate, it must be borne in mind that accessibility of disposed-of waste is time-related. Although mining engineering measures make it possible to extend the period of this accessibility to, possibly, some hundreds of years, nevertheless it is impossible to extend it indefinitely. This is because a geological facility is designed for isolation over very long periods, of an order of magnitude of 100,000 years. In that term, of course, retrievability cannot be guaranteed. The question then arises, how is retrievability to be viewed in this context?

All in all, these considerations must be carefully dealt with in a generic study. To that end, an initial step has been taken in carrying out a general EC study, on Dutch initiative, dealing with economics, mining engineering and various host rocks. This study, which is not part of the OPLA programme, is still in progress (Prij & Heijdra, 1995; Heijdra et al., 1996, in prep.).

Quite apart from the above, it is worthwhile considering what mining engineering options are currently available to accomplish a certain measure of retrievability (Technische Universiteit Delft, 1993a; Van den Broek et al., 1994). The investigative OPLA research programme indicates that, in a general sense, retrievability is technically feasible but that this requires the Phase 1 mine concept to be modified in a number of respects. In particular, these modifications include a reduction of the length of the boreholes drilled from the galleries, and a reduction of the thermal effect, in order to keep the waste accessible as easily as possible and for as long as possible. The study also reports that certain concepts for "direct disposal" of spent fuel elements as studied for example in Germany can be readily combined with possible retrievability of the waste. For this purpose, use is made of highly robust "Pollux" containers, which are disposed of in galleries in a mine.

Although the OPLA study is not intended to assess the measure of accessibility for predetermined periods, based on preliminary findings the period refers to a retrievability term of several hundred years, the period during which full information will be present concerning the location of the waste. In the longer term, a situ-

ation is envisaged in which the repository has been closed by polydimensional rock pressure and the waste can only be recovered by means of "remining". Underground geophysical survey methods will be able to provide guidance in this case.

The investigative OPLA programme made no statements on the safety aspects of retrievability.

There is no clarity as to the objectives of retrievability. There are a number of possible replies to the question of what is primarily envisaged with retrievability. Different objectives can lead to different mining engineering consequences, each with their own consequences for the "terms of guarantee" and longer-term safety.

17.7.2 Direct Disposal of Spent Fuel Elements

In Phase 1 of the OPLA research programme, it was assumed that, in accordance with the present situation in the Netherlands, spent fuel is reprocessed into re-usable fissile material. The reprocessing waste produced in that process (including the fission waste) form an essential component of the total radioactive waste flow to be disposed of. Developments in the sphere of public interest are calling attention to a fuel cycle without reprocessing. A number of countries (the United States, Sweden) have in the past for various reasons opted for the non-reprocessing route. In such a situation, therefore, it is not processing waste but spent fuel material that has to be disposed of.

The consequences of this for safety and for repository design have for some time been the subject of studies in other countries. For instance, the German Projekt Alternatieve Entsorgung (PAE) investigates two techniques for direct disposal. One technique, which is regarded as the most important, is based on large, thick-wall steel containers in which the entire spent fuel elements are disposed of. These "Pollux" containers are emplaced in galleries in the salt formation, after which the interjacent spaces are backfilled with crushed salt. This method differs essentially from the mine concept with vertical boreholes in the galleries, as studied in OPLA Phase 1. There is greater similarity to the Phase 1 concept in the second method, in which containers filled with short sections of fuel elements are emplaced in boreholes in the galleries.

This study, due for completion in 1995, also considers various design options for the combined disposal of reprocessing waste and spent fuel. This "combined concept" has been subject to exploratory study during the supplementary OPLA research programme (ECN, 1993c). In the disposal concept investigated, the spent fuel originates from a possible enlargement of the nuclear energy potential by 3000 MWe. The reprocessing waste to be disposed of originates from the operations of the existing Dodewaard and Borssele nuclear power stations. As regards technical feasibility, the OPLA study indicates that the chosen disposal technique with boreholes in galleries will not require any drastic modifications to the Phase 1 design. As regards safety, modelling was used to calculate the risk associated with flooding in a disposal mine. It is found that the risk for the technique and waste volume studied here is smaller than 10-6/year.

The exploratory study relating to direct disposal reveals that several feasible disposal techniques are available. The findings of this study match those of the abovementioned PAE study among others.

17.8 CONCLUSIONS

General

On the basis of the results of the supplementary research programme presented above, the OPLA Committee reached the following statements:

- The supplementary research programme disclosed no phenomena or combinations of phenomena which would a priori lead to rejection of the "disposal of radioactive waste in salt formations" option.
- Insight into the parameters affecting the uncertainties in the risks of disposal has increased strongly; as a result, the safety approach is now based on better confirmed data. It has been possible to further validate, analyse and extend the models and data required for the safety calculations. In this way, substantial advances have been made in determining the bandwidths in the results of Phase 1. Accordingly, the objective of the first theme of the supplementary research programme has been achieved.
- Even after completion of the supplementary programme, a number of major uncertainties remain with regard to definition of the risk-determining features of a repository and the natural barriers, practical validation of the models used in the safety study, and the consequences of gas formation and radiation damage in rock salt.
- Development of the PROSA safety method has yielded an extremely useful instrument for scientifically

based risk assessment of geological disposal in rock salt formations. In addition, the PROSA approach appears to have a broader application potential.
- Exploratory study of several new aspects, retrievability of disposed-of radioactive waste and direct disposal of spent fuel in rock salt, has demonstrated that integration of these variants will entail engineering design consequences. The associated design modifications appear to be technically feasible; no prohibitive factors were identified in the study.

Earth Scientific Background Studies

From recent research on model development of geological processes, the following conclusions can be drawn:

- Intraplate stress is the major driving force for the initiation of salt diapirism in the North Sea, which is in contrast with the classical theory of a purely gravitational drive of diapirism.
- Salt dissolution rates in Dutch diapirs are positively correlated with the performance of glacial and subglacial processes.
- Results from the FLUVER model suggest that geological barriers comprising salt diapirs and their cover beds in the Netherlands, have not been, and will not be, affected significantly by fluvial erosion in the past 200,000 years and in the next 100,000 years.
- Although the process of subglacial erosion, having created deep tunnel valleys in the North Sea basin, is still not yet fully understood, the results from SUBGLER model studies suggest that these phenomena were formed during the last phase of deglaciation, as a result of overpressurizing Quaternary aquifers by subglacial meltwater.

Geological studies on the underground disposal of long-lasting hazardous waste in many countries have demonstrated the need for a time-dependent geological simulation model for the geological barriers enveloping the disposal site, in which relevant geological processes are integrated and coupled. Promising results of climatic and tectonophysic research indicate that the development of such a barrier model would be feasible in the next decade. The development of an integrated geological barrier model is only feasible if a large concerted multidisciplinary effort, preferably in an international framework, is carried out.

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REFERENCES

Boulton, G.S. & R.C.A. Hindmarsh, Sediment deformation beneath glaciers: Rheology and geological consequences, J. Geophys. Res., 92 p., 9059-9082, 1987.

Boulton, G.S., A.F.M. Slot, K. Blessing, P. Glasbergen, T. Leijnse and K. van Gijssel, Deep circulation of groundwater in overpressured subglacial aquifers and its geological consequences, Quat. Sc. Rev., vol. 12, pp. 739-745, 1993.

Boulton, G.S. and K. van Gijssel, Synthesis of the Saalian and post-Saalian stratigraphy in Northwest and Central Europe: A data-model comparison, Quat. Sci. Rev., (1996, submitted).

Cloetingh, S. & H. Kooi, Intraplate stress and dynamical aspects of rift basins, Tectonophysics, vol. 5, pp. 167-185, 1992.

Daudré, B. & S. Cloetingh. Mechanical modelling of salt diapirism: Influences of the tectonic regime. Tectonophysics, vol. 240, p.59-79, 1994.

Delft Ground Mechanics, Development of a sampling apparatus for the collection of geochemically undisturbed sediment samples at great depth (in Dutch), Report nr. CO-327954/02, 1993.

Den Hartog, H.W., J.C. Groote & J.R.W. Weerkamp, Stored energy in irradiated NaCl - Radiation Effects and Defects in Solids, pp. 133-134, 1995.

ECN, Development and anneal of radiation damage in salt - Phase 1A of the OPLA Programme, Report no. ECN-C-93-086, 1993a.

ECN, In situ experiments in the Asse salt mine, Phase 1A of the OPLA Programme, Report no. ECN-C-93-

069, 1993b.

ECN, Direct disposal of spent fuel, Phase 1A of the OPLA Programme, Report no. ECN-C-93-080; annex: no. ECN-C-93-083 and no. ECN-C-93-099, 1993c

Edinburgh University, RIVM, RGD, Université de Paris Sud und Université de Louvain-la-Neuve, Simulation of the effects of long-term climatic change on groundwater flow and the safety of geological disposal sites, EC Programme on Management and Storage of Radioactive Waste, project no. F12W/0046, (1996, in preparation).

Garcia Celma, A., C. De Las Cuevas, P. Teixidor, L. Miralles and H. Donker, On the possible continuous operation of an intergranular process of radiation damage annealed in rock salt repositories, In: Proceedings of a Symposium on the Geological Disposal of Spent Fuel and High Level and Alpha Bearing Wastes, Antwerp, 19-23 October 1992, IAEA, Proceedings Series: 133-144, 1993.

Garcia Celma, A. and H. Donker, Radiation-induced creep of confined NaCl, In: Radiation Effects and Defects in Solids, vol. 132, p. 223-247, 1994.

Garcia Celma, H. Donker , W.J. Soppe and L. M. Miralles, Development and anneal of radiation damage in salt. CEC, Nuclear Science and Technology Series, EUR 15941 EN, 1995.

Groote, J.C., J.R.W. Weerkamp and H.W. den Hartog, An irradiation facility for radiation damage investigations, Meas. Sci. Technol, vol. 2, pp. 1187-1191, 1991.

Heijdra, J.J., J. Bekkering, J.v.d. Gaag, P.H. v.d Kleyn, and J. Prij, Retrievability of radioactive waste from a deep underground disposal facility, Report EUR 16197 (1996, in preparation).

Heijdra, J.J. and J. Prij, Geomechanical measurements in a deep dry-drilled borehole in Asse II rock salt, EUR 16485 (1996, in preparation).

Jeffery, D.H., Comment on the origin of major incisions, Marine Geology 96, pp. 125-126, 1991.

Leijnse, A. and S.M. Hassanizadeh, Short communication: Model definition and model validation, Advances in Water Resources, vol. 17, pp. 197-200, 1994.

Leijnse, A., H. van de Weerd and S.M. Hassanizadeh, modelling of madionuclide transport in Koongarra, Australia; the effect of a moving weathering zone, Water Resources Research, (1996, submitted).

Meekes, J.A.C., E. Bakker, P.F.M. de Groot, J.M.J. Kuijk, M.H. Mulder and A.F.B. Wildenborg, Geophysical methods for the hydrogeological reconnaissance of salt domes and the adjacent sediments, Phase 1A of the OPLA Programme, Report of IGG-TNO & RGD, no OS 93-15-B, 1993.

Müller, B., M.L. Zoback, K. Fuchs, L. Mastin, S. Gregersen, N. Pavoni, O. Stephansson and C. Lunggren, Regional patterns of stress in Europe, J. Geophys. Res., vol. 97, p. 11783-11803, 1992..

Nye, J.F., Water flow in glaciers: Jökulhlaups, tunnels and veins, J. Glaciology, Vol. 17, No. 76, pp. 181-207, 1976.

Oostrom, M., K. van Gijssel and W. Zijl, Modelling subrosion and groundwater flow in the vici-nity of the Zuidwending diapir on the basis of geometrical and (palaeo-)hydrological boundary conditions for the north eastern Netherlands, The SESAM-project in the OPLA Programme, Phase 1A, RIVM-Report 715205004, 1993.

OPLA Committee, Research into geological disposal of radioactive waste in the Netherlands, Final Report on Phase 1 of the OPLA Programme, Ministry of Economic Affairs, 1989.

OPLA Committee, Research programme on geological disposal of radioactive waste in the Netherlands, Final report Supplementary Research Programme, Phase 1A, 1993.

Poliakov, A., R.T. van Balen, Y. Podlachikov, B. Daudré, S. Cloetingh and C. Talbot. Erosion and diapirism: A comparative analysis of salt diapirism for different cases of erosion, Tectonophysics, vol. 226, pp. 199-216, 1993.

Prij, J.. The role of human intrusion in a waste repository in rock salt, International Conference Safewaste 93, organised by SFEN, French Nuclear Energy Society, Avignon 13-18/06/1993, Proceedings, vol. 2, pp. 490-501, 1993.

Prij, J., Myth or nightmare: Safety consequences of the release of radiation induced stored energy, Nuclear

Technology, January, 1996.

Prij, J., Radioactive waste repository relevant parameters, In: Radiation damage in salt, EUR, (1996a in preparation,).

Prij, J., Evaluation of the safety consequences of radiation induced stored energy in a repository in rock salt, In: Radiation damage in salt, EUR, (1996b in preparation).

Prij, J., B.M. Blok, G.M.H. Laheij, W. van Rheenen, W. Slagter, G.J.M. Uffink, P. Uijt de Haag, A.F.B. Wildenborg and D.A. Zanstra, PRObabilistic Safety Assessment, Final report, of ECN, RIVM and RGD in Phase 1A of the OPLA Programme, 1993a.

Prij, J., V. Graefe, T. Rothfuchs, and L.H. Vons. Modelling of the thermo-mechanical behaviour of solid rock salt and crushed salt, CEC Workshop on THM properties, report EUR 16219 en. 1995.

Prij, J. and L.F.M. Hamilton. Thermo-mechanical analyses of the HAW testfield, Nuclear Engineering and Design vol. 138, pp. 203-215, 1992.

Prij, J., and J.J. Heijdra, Feasibility and economic consequences of retrievable storage of radioactive waste in the deep underground, Global, vol. 2, pp. 1770-1777, 1995.

Prij, J., J.J. Heijdra, and B. van den Horn. Convergence and compaction of backfilled openings in rock salt, 3rd Conference on the Mechanical Behaviour of Salt, September 1993b.

Remmelts, G., Salt tectonics in the southern North Sea, the Netherlands, KNGMG Special Publication: Geology of gas and oil under the Netherlands, 1996.

Rijksuniversteit Groningen, Radiation damage in NaCl - Report in Phase 1A of the OPLA Programme, 1993.

Sauter, F.J., S.M. Hassanizadeh, A. Leijnse, P. Glasbergen and A.F.M. Slot. METROPOL, A computer code for the simulation of transport of contaminants with groundwater, RIVM-report 725205002, 1990.

Seinen, J., J.C. Groote, J.R.W. Weerkamp and H.W. den Hartog, Radiation damage in NaCl: General model of nucleation and aggregation processes in doped NaC, Radiation Effects and Defects in Solids, vol. 124, pp. 325-339, 1992.

Seinen,J., D.I. Vainshtein, H.C. Datema, and H.W. den Hartog, Radiation Damage in NaCl: the annealing behaviour of heavily damaged KBF4 doped crystal, J. Phys.: Condensed Matter, vol. 7, pp. 705-716, 1995.

Shoemaker, E.M., Subglacial hydrology for an ice sheet resting on a deformable aquifer. J. Glaciology, vol. 32, no. 110, pp. 20-29, 1986.

Technische Universiteit Delft, Retrievability of radioactive waste, Phase 1A of the OPLA Programme, report no. 54595, 1993a.

Technische Universiteit Delft, Aspects of the deep boreholes/cavity option, Phase 1A of the OPLA Programme, report no. 52610, 1993b.

Van Balen, R.T. and S. Cloetingh, Intraplate stresses and fluid flow in extensional basins, In: A. Horbury & A. Robinson (eds). Diagenesis and basin development. AAPG Studies in Geology, vol. 34, p. 87-98, 1993.

Van den Berg, M.W., W. Groenewoud, G.K. Lorenz, P.J. Lubbers, D.J. Brus and S.B. Kroonenberg, Patterns and velocities of recent crustal movements in the Dutch part of the Roer Valley Rift System, Geologie en Mijnbouw, Special Issue Roermond earthquake, (1996, submitted).

Van den Broek, W.M.G.T., M.J.V. Menken and H.C. Heilbron, Preliminary study on retrieval of radioactive waste from a salt-mine repository, Proc. Symp Waste Man., Tucson (Arizona), March 1994, vol. 1, pp. 593-596,1994.

Van de Weerd, H., A. Leijnse, S.M. Hassanizadeh and M.A. Richardson-Van der Poel, INTRAVAL Phase 2, Alligator Rivers Natural Analogue: Modelling of uranium transport in the weathered zone at Koongarra (Australia), Report no. 715206005 RIVM, 1994.

Van Dijke, J.J. and A. Veldkamp, Climate-controlled glacial erosion in the unconsolidated sediments of north western Europe, based on a genetic model for tunnel valley formation, Earth Surface Processes and Landforms, vol. 20, 1995.

Van Gijssel, K., Hydrogeological and palaeoenvironmental data input for hydrological model simulations

in the north eastern Netherlands, Meded. RGD, vol. 52, 1995.

Van Montfrans, H.M.. Research Program on geological disposal of radioactive waste in The Netherlands -In: Geological Problems in Radioactive Waste Isolation, Proc. Workshop W3B, 28th IGC, Washington, D.C., July 1989, pp. 103-114, 1991.

Van Weert, F.H.A., A. Leijnse and S.M. Hassanizadeh. , INTRAVAL Phase 2, Gorleben Test Case, Report no. 715206004 RIVM, 1994.

Van Weert, F.H.A., K. van Gijssel, A. Leijnse and G.S. Boulton, The effects of Pleistocene glaciations on the geohydrological system of Northwest Europe, J. Hydrology, (1996, submitted).

Veldkamp, A., Quaternary river terrace formation in the Allier basin, France, A reconstruction based on sand bulk geochemistry and 3-D modelling, Doctoral thesis Agricultural University Wageningen, 172 pp, 1991.

Veldkamp. A., A 3-D model of fluvial terrace development in the Allier basin (Limagne. France), Earth Surface Processes and Landforms, vol. 17, pp. 487-500, 1992.

Veldkamp, A. and M.W. van den Berg, Three-dimensional modelling of Quaternary fluvial dynamics in a climo-tectonic dependent system, A case study of the Maas record (Maastricht, The Netherlands) Global and Planetary Change, 8: pp. 203-218, 1993.

Veldkamp, A and J.J. van Dijke, Modelling of potential effects of long term fluvial dynamics on possible geological storage facilities of nuclear waste in the Netherlands, Geologie en Mijnbouw, vol. 72, pp. 237-249, 1994.

Weerkamp, J.R.W., J.C. Groote, J. Seinen, and H.W. den Hartog. Radiation Damage in NaCl, Part I to IV, Phys. Rev., vol. B50, pp. 9781-9802, 1994.

Wildenborg, A.F.B., J.H.A. Bosch, E.F.J. de Mulder, R. Hillen, F. Schokking and K. van Gijssel, A review: effects of (peri-)glacial processes on the stability of rock salt - Proc. 6th Int. Congr. Int. Ass. Eng. Geol., 5, pp. 2763-2770, Amsterdam, Balkema, 1990.

Wildenborg, A.F.B., M.C. Geluk, Th.A.M. de Groot, G. Remmelts, G.Th. Klaver, A.N.M. Obdam, A. Ruizendaal and P.J.T. Steins, Evaluation of salt bodies and their overburden in the Netherlands for the disposal of radioactive waste, Rijks Geologische Dienst, vol. a. Spatial data, vol. b. Salt movement, vol. c. Caprock formation and subrosion vol. d. Fluvial and subglacial erosion, vol. e Geological barrier model, Project GEO-1A in the OPLA Programme, Phase 1A, report 30012/ER, 1993.

Wildenborg, A.F.B., S.A.P.L. Cloetingh, E.F.J. de Mulder, R.T. van Balen, J.J. van Dijke, K. van Gijssel, A. Veldkamp, B. Daudré and G. Remmelts. Towards a predictive barrier model for Zechstein rock salt in the Netherlands, Colloque Géoprospective, April 1994, Paris, 1995.

CHAPTER 18

RADIOACTIVE WASTE MANAGEMENT IN POLAND: CURRENT STATUS OF INVESTIGATIONS FOR RADIOACTIVE WASTE REPOSITORY AREAS

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Abstract: General information about the regulations and limitations concerning radioactive waste in Poland is given in the paper. Radwaste, 95% of which is low level at present, comes from one research reactor and over 2000 smaller producers. The responsibility for collecting, handling and disposal of all radwaste is delegated to one organization partially supported by the state. The status of new repository site investigations is reviewed.

18.1 INTRODUCTION

The suitable management of radioactive waste and spent fuel from research reactors appears to be one of the most important problems in Poland, because it has an impact on the safety and public acceptance of nuclear energy and the further development of this technology.

It is estimated that ionizing radiation sources in this country with an activity of about 200,000 Ci are being used in medicine, industry and scientific research activities. As of January 1, 1993, spent fuel from research reactors with an activity of 800,000 Ci (including 6000 Ci from transuranic elements) were in storage at the Institute of Atomic Energy (IEA) at Swierk. Decommissioning of research reactors is also a very important problem that should be taken into account in waste management programs.

18.2 LEGISLATION FOR RADIOACTIVE WASTE MANAGEMENT

Radioactive waste management in Poland is regulated within the framework of:

1. The Atomic Law, laid down April 10, 1986, by an Act of Parliament, in which the utilization of atomic energy for the public, as well as the economic needs of the country, is defined; and
2. A regulation from the President of the National Atomic Energy Agency (NAEA), announced May 19, 1989, on the Principles of Defining Waste as Radioactive, Classifying Them and Keeping Records, and the Immobilization, Storage and Disposal of Wastes.

To ensure safe transport of radioactive material, the IAEA regulations (Safety Series No. 6, 1985), and modal ADR, RID, IATA and IMO regulations are applied as appropriate. In practice, the transport of radioactive waste is only by road.

Radioactive waste is separated as follows: (1) beta and gamma emitters - high level (HLW); (2) alpha emitters; and (3) spent sealed radioactive sources. According to the regulation mentioned above from the NAEA's President, waste classifications are based on ALI principles as shown in Table 18.1.

18.3 SOURCES OF RADIOACTIVE WASTE IN POLAND

In Poland, radioactive waste comes from research reactors, scientific and educational institutions, industrial organizations and hospitals. Only low and intermediate level wastes are produced. The high activity gamma emitters in spent sources should be transported back to the supplier, but a number of them are still stored at different places in the country.

For many years, Poland has operated two research reactors and one critical assembly; and at present, a few thousand spent fuel elements from these reactors are stored at the site of the Institute of Atomic Energy. The storage facilities were originally planned only as tempo-

Table 18.1. Waste classifications.

Waste Form	Radiation	LLW	ILW	HLW
Solid	beta, gamma (ALI/m^3)	10^2-10^6	10^6-10^9	$>10^9$
	alpha (ALI/m^3)	$>10^2$	-	-
Liquid	beta, gamma (ALI/m^3)	10^{-2}-10^2	10^2-10^5	$>10^5$
	alpha (ALI/m^3)	$>10^{-2}$	-	-
Gaseous	beta, gamma (DAC)	0.1-10	10-10^6	$>10^6$
	alpha	>0.1	-	-

Note: ALI denotes a derived factor being the annual limitof radioactive intake through the alimentary canal (ALIp) or respiratory system (ALIo) for people employed in conditions of radiation exposure, stated in separate provisions.
DAC denotes a derived factor being the concentrations of radionuclides in the atmosphere for people employed in conditions of ionizing radiation, stated in separate provisions. DAC = ALIo/2400 m^3 where value of 0.1 DAC corresponds to the ventilation outlet.
In the case of unidentified isotopes, a more restrictive limit expressed in Bq/kg or Bq/m^3 may be used.

rary storage on the assumption that the spent fuel would be taken back by the Soviet supplier. The spent fuel is kept in a wet storage facility close to the reactor, and the age of the oldest irradiated fuel elements is 35 years. Conditions at the storage facilities are controlled by the user and by the National Inspectorate for Radiation and Nuclear Safety. The question of how and where this spent fuel is to be transported, stored or reprocessed appears to be one of the most important questions to be considered by the Government. Establishing a clear policy regarding the management of spent fuel seems to be one of the major elements having an impact on public acceptance of nuclear energy.

18.4 Organizations Responsible for Waste Management and Scope of Their Duties

According to the above mentioned Atomic Law:

- The issues falling within the scope of the Agency's activity is radioactive management;
- The head of the organizational unit within which the radioactive wastes arise is responsible for their handling in full conformance with the nuclear safety and radiation protection requirements and their preparation for transport and storage; and
- The head of the organizational unit that has been licensed to operate a radioactive waste disposal facility is responsible for keeping the radioactive waste in full conformance with the nuclear safety and radiation protection requirements.

The President of the National Atomic Energy Agency is designated, and can be recalled, by the Prime Minister and reports directly to him. The Management Committee, under the supervision of the President, acts within the Agency. The Committee adopts resolutions on matters related to the scope of the Agency's activities.

The responsibility for LLW/ILW over the entire country is delegated to the Institute of Atomic Energy. At present, practically all radwastes are collected, treated and conditioned at the IEA and disposed of at the Central Repository (CR) located at Rozan.

18.5 Treatment and Conditioning of LLW/ILW

The radioactive waste treatment and the conditioning methods at the IAE are aimed at reducing volumes and

preparing for safe transportation and storage to fulfill the requirements for final disposal at the CR.

Low level liquid waste is chemically treated in a clarifier resulting in a volume reduction of about 100 times, and the sludge is transferred to a bitumization plant for further treatment. On the other hand, ILW is concentrated by evaporation, and the distillate is further purified by ion exchange before being released. The concentrates (evaporator sludge) are conditioned by cementation, and the radioactivity and chemistry of the decontaminated liquid effluents are controlled before being released. Solid LLW is compacted into 0.2 m^3 drums using a 12-ton press, and the biological wastes, after urea-formaldehyde conditioning, are stored in 0.05 m^3 drums.

18.6 STORAGE OF RADIOACTIVE WASTE

The Central Repository for radioactive waste is a near-surface type located 90 km from Warsaw on the grounds of a former military fort built in 1905. The CR was put in operation in 1961. The geology of the site is characterized by boulder and sandy clay. No historical records regarding seismic activities in the area are available.

Most of the repository is characterized by a concrete structure of military design with roof and wall thicknesses of 1.2 to 1.5 m, and a floor thickness of about 30 cm. Within a protection trench of the fort, a moat is used for final disposal with a concrete cover about 20 cm thick. Only solid and solidified low and intermediate level wastes are stored at the CR. After 33 years of operation, about 5400 m^3 of wastes have been disposed of in this repository. The cumulative activity of these wastes is 250,000 GBq (without decay), or 40,000 GBq with decay.

18.6.1 Storage in Concrete Bunkers

Concrete bunkers are used for temporary storage of alpha waste and contaminated installations and devices, which will be reused. Solid alpha wastes are placed in a chamber that is sealed off, after being filled, with a brick wall. Sealed sources of waste, with activities that do not exceed 4 GBq, are disposed of in one of the underground concrete bunkers. The access hole to this bunker is sealed with a lead cover lid 200 mm in thickness.

Except for the alpha waste categories, the LLW is disposed of in the moat of the CR where the bed and walls are made of concrete. The containers of the conditioned waste are placed in layers that are separated by layers of concrete. This procedure is repeated until the moat is filled to capacity, and the top layer is protected by asphalt.

18.6.2 Environmental Radiation Monitoring

The on- and off-site radiation monitoring system at the CR includes two basic groups of measurements:

- Radioactivity levels in environmental samples, and
- On-site and off-site gamma radiation levels.

Records of measurements are made by the IAE and presented annually to the National Inspectorate for Radiation and Nuclear Safety as well as to the appropriate local administration.

18.7 NEW REPOSITORY SITE INVESTIGATIONS

A study was initiated in Poland in the late seventies aimed at selecting areas suitable for radioactive waste repositories. Initially, the main attention was concentrated on selecting areas characterized by rock systems suitable for the permanent isolation of wastes. Salt beds, crystalline rocks and clay formations of considerable thickness were considered the most appropriate rocks for an underground repository.

The study was conducted upon the request, and was coordinated, by the National Atomic Energy Agency. Many specialists from various scientific institutions participated in the elaboration of specific issues. The investigations were designed to determine potentially useful repository sites in three categories: (1) superficial; (2) shallow underground; and (3) deep underground (see Nos. 1-10, Fig. 18.1).

18.7.1 Deep Underground Waste Repositories

An examination of geological formations that initially appeared suitable for the construction of deep waste repositories and could satisfy nuclear safety requirements led to the following selection of sites shown on Figure 18.1:

- Silurian shales in northern Poland (No. 5);
- Granites, bastard granites, crystalline Pre-Cambrian shales in eastern Poland (No. 6); and
- Triassic mudstone (No. 4).

The location of a repository within these formations

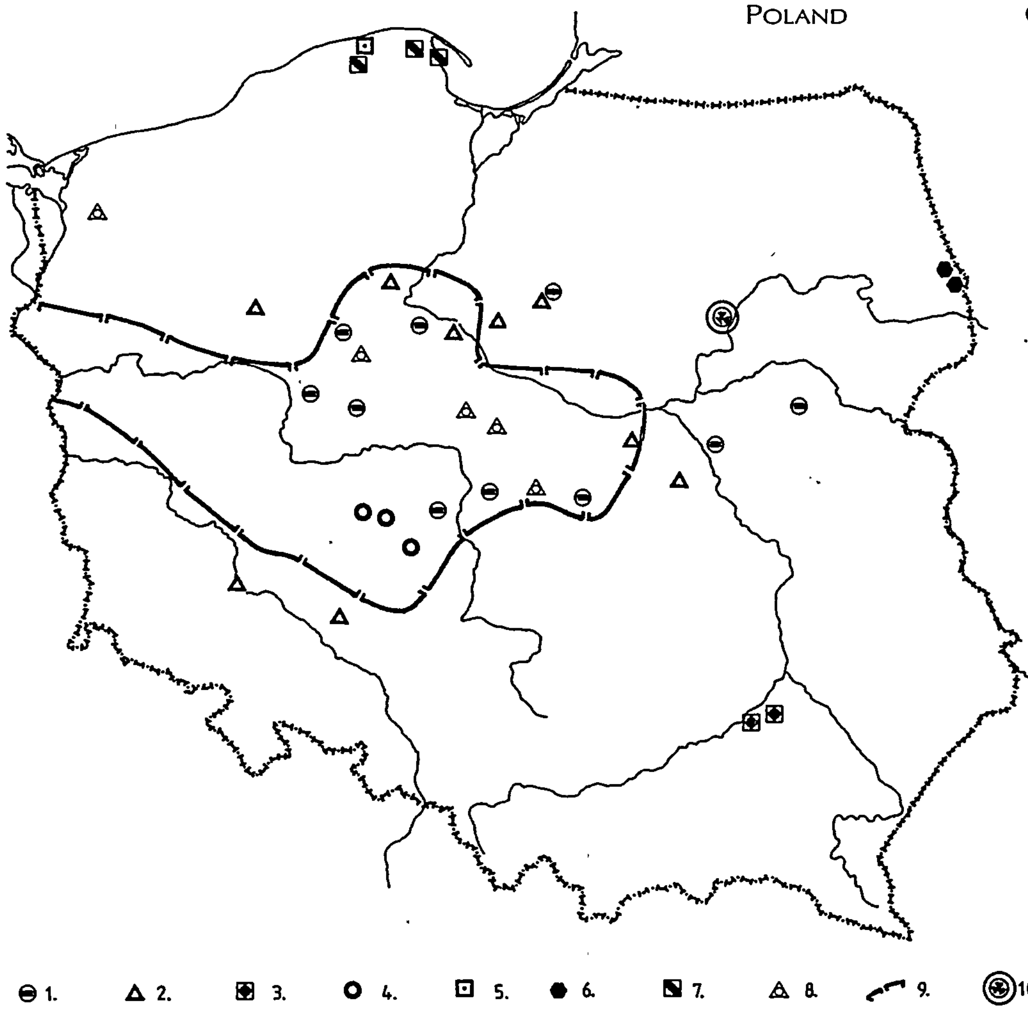

Figure 18.1. Areas selected in Poland for the possible location of a radioactive waste repository: 1 - superficial (Quaternary); 2 - shallow underground (Tertiary-Pliocene; 3 - shallow underground (Tertiary-Sarmatian); 4 - deep underground (Triassic)' 5 - deep underground (Silurian); 6 - deep underground (Pre-Cambrian); 7 - deep underground (Permian); 8 - deep underground(Permian); 9 - limits of area with favorable climatic conditions for superficial radioactive waste repositories; and 10 - active repository at Rozan.

would require the construction of deep underground works.

In discussions concerning waste storage in other rock types, the concept of storage in salt formations was also considered. The following selections were made from an analysis of formations of Permian and Miocene age:

- Rock salt in the Baltic region (No. 7); and
- Salt diapirs in central Poland (No. 8).

Information on rock salt has been revealed in varying degrees by wells and geophysical investigations. These earlier studies were conducted in terms of assessing the salt resources for the chemical industry. Solution mining (lixiviation) of storage cells in one of the rock-salt diapirs was considered to be the most appropriate solution.

During the period of planning for the development of nuclear energy, a study was carried out to determine the best waste repository policy to adopt concerning rock-salt beds in northern Poland. A method was presented for the prognosis of thermal effects in an underground repository for highly radioactive waste and the problems of optimizing the underground works and the storage technique. The repository was assumed to be located in Permian salt beds at a depth of 740 m beneath the surface. The average bed thickness in the area of the repository is about 200 m, and the overlying formations are

anhydrides and Permian dolomite limestones. Above these beds are Mesozoic and Cenozoic formations with possible water-bearing strata.

To develop the prognosis of the temperature distribution, a method was proposed that is based on an analytical solution to the thermal conduction equation for the individual source (waste container). With regard to the design stages of the repository, this method allows one to more rapidly assess an optimal scheme for the distribution of repositories. The results of this model study showed that the thermal impact of such a repository becomes apparent only after some one hundred years and that the impact is practically negligible.

18.7.2 Shallow Underground Waste Repositories

Further considerations were therefore limited to the construction of a shallow repository in clay formations. The radioactive waste would be stored in shallow pits or large-diameter wells some 50-70 m under the surface. The following formations were selected:

- Krakowiec clays (Tertiary period - Sarmatian) in southeast Poland (No. 3); and
- Spotted clays (Tertiary-Pliocene) in central Poland (No. 2).

An analysis of the population conditions and physical management was carried out at these sites.

18.7.3 Superficial Waste Repositories

The existence of outwash sands that are located in boulder clays were selected as a favorable location for superficial waste repositories. The most favorable location on the watersheds of rivers is also essential. A study was carried out at ten locations using the methodology recommended by the International Atomic Energy Agency for the pre-selection and selection stages. The prospective areas were separated from major regions for further detailed analysis. An evaluation of the usefulness of the areas selected was carried out, in relation to an analysis of geographic conditions, on the basis of the existence of conditions that would exclude or limit the area itself. The characteristics of the areas were analyzed in terms of the following issues:

- Geological setting and hydrogeological conditions;
- Geodynamical processes;
- Potential of raw materials;
- Hydrology;
- Meterology and climate; and
- Environmental management and protection.

Most locations were concentrated in central Poland (No. 1 on Fig. 18.1) within an area where favorable climatic conditions for a superficial repository prevail. The following conditions were considered to be preclusive factors:

- Presence of legally protected areas (reservations, national and landscape parks);
- Areas with planned regional limitations on physical management;
- Areas under the influence of a concentrated groundwater exploitation;
- Areas with imminent 100- and 500-year floods;
- Areas where the subsurface waters are highly mineralized; and
- Areas with shallow groundwater.

An analysis of the social and economic conditions were also carried out in some of the locations. At the same time, attempts were made to obtain social acceptance for locations selected for a superficial repository.

18.8 SUMMARY

The nuclear energy development program in Poland is still not precisely defined, but it is clear that any further advancement in the field of nuclear technology cannot be pursued without solving the waste isolation problem. For the future of the national economy, there is a need for serious consideration of this source of energy. An increase in environmental protection studies is now quite evident and results in an enforced continuation of previous research concerning the the location and documentation of new repository sites.

The results presented in this paper are concerned primarily with the preselection stage. Only in the case of superficial radioactive waste repositories are some elements of the selection stage being investigated. Within a period of about ten years, the next studies will concentrate on the choice of possible locations for underground and superficial radioactive waste repositories. It is evident that, after potential sites have been selected, quite different programs of time-consuming research will be required for each of these types of repositories.

CHAPTER 19

PROGRAM OF GEOLOGICAL DISPOSAL OF SPENT FUEL AND RADIOACTIVE WASTES IN SLOVAK REPUBLIC

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19.1 INTRODUCTION

The present stage of spent fuel and high level waste management is a development from the nuclear cycle strategy of the former Czech and Slovak Federal Republic (CSFR). This concept was based on intergovernmental agreements between former CSFR and the Soviet union (USSR) on cooperation and assistance during construction and operation of Czechoslovak nuclear power plants (NPP). The cooperation program between CSFR and USSR provided for confirmed unpaid transport of spent fuel to USSR during the whole time period of operation of Czechoslovak NPP's. Up to 1988, about 700 assemblies of spent fuel were transported to the USSR.

Fulfillment of the above agreements was not fixed in the USSR approach. First, there was a prolonged period of spent fuel storage on CSFR territory, and after the political changes, new transport conditions, abandonment of unpaid transport, and a decrease of assembly pieces dedicated for transport. As a result, the Slovak Power Plants company decided to construct long period spent fuel storage. Wet storage is used in this operation and its capacity is 600 tHM (5,040 pieces of fuel assemblies).

The main reason for changing the original strategy of high level waste and spent fuel management was the abandonment of unpaid spent fuel transport after the political changes in the Russia Federation. With respect to the present economic situation in the Slovak Republic, it is impossible to decide on a new strategy in this field. Therefore, the Slovak Power Plants company decided to solve the problem with interim storage of spent fuel for approximately 50 years and start preparation of deep geological disposal of high level wastes and spent fuel under the conditions of the Slovak Republic.

19.2 HISTORY OF PROJECT DEVELOPMENT

The history of repository development designed to accept high level and long lived radioactive wastes and spent fuel started two decades ago. At that time the Nuclear Research Institute Rez (Czech Republic) issued some basic policies concerning the conceptual and safety aspects of a deep geological repository.

The year 1984 became a turning point for the intensification of activities towards underground facilities. Initiated by Energoproject with participation of Gasproject, and Construction Geology, the Nuclear Research Institute and other companies examined the possibility of constructing a deep silo (over 500 m) for reactor waste from NPP Dukovany and later NPP Temelin. Even though the project was abandoned, it provided valuable experience from the fields of geology, safety studies and contacts with the public.

Further activities, stimulated mainly by the need for processing waste arising from the decommissioning of NPP A-1 in Jaslovske Bohunice, were more precisely defined. They sought tools usable for evaluating the acceptability of sites, the solicitation of proper regions on the basis of archival data, and the extension of possibilities to provide safety analysis. The leading companies performing these activities were Geoindustry, Central Institute of Geology, Institute of Geophysics, Nuclear Research Institute, Nuclear Power Plant Research Institute, Dionyz Stur Geological Institute and others. One of the achievements reached was a basic evaluation of the Slovak territory for purposes of siting a deep geological repository.

It is reasonable that works aimed at the geological repository were performed by a number of institutions,

and their costs were covered from different sources. This led to a certain heterogeneity of activities and as a consequence an incongruity and incoherence in the results obtained. Therefore, urged by a need to initiate a program of spent fuel management, which arose after the political changes in Europe, the Federal Ministry of Economy of CSFR started the preparation of a technical program initiating the most urgent works in 1992. The purpose was to concentrate available agencies in a consistent, centrally coordinated contract. In the course of preparing this task, practically all central organizations involving contractors, as well as most research bodies active in waste management as suppliers, were involved. Firstly, the technical content of the project was defined and then, as a result of competition, the Nuclear Research Institute was appointed as coordinator. Unfortunately, due to the division of Czechoslovakia, funding of the project collapsed. To promote the primary purpose of the project, in the spring of 1993, the Czech Power Company and the Slovak Power Company decided to order a study on, "The plan for development of a deep geological repository" and to share its expenses equally.

In view of the importance of the report for further progress in repository development in the future, it was decided that the contractor should ensure an international review of the prepared document. For this purpose, a contact was established with the administration of The Nuclear Cycle Division of the International Atomic Energy Agency, which advised that the government ask for an evaluation within the Waste Management Assessment and Technical Review Program (WATRP). The results of the mission were attached to this study as a separate document in December 1993.

After the division of CSFR, work on development of deep disposal of high level wastes and spent fuel in Slovak Republic started only at the end of 1995. Because of the developments since 1993 in world-wide experience with deep disposal, and because the original project was elaborated for conditions in the former CSFR, the first step in continuing the work was the need to revise the original project for deep disposal development and adapt it to conditions in Slovakia.

19.3 PURPOSE OF PROJECT

The main goal of the revision was to specify the main bounds among the particular tasks of deep disposal development. This means taking into account all requirements for site selection, near-field and far-field interactions, quality assurance, safety analyses, the role of the public as well as design studies and licensing steps under the conditions of the Slovak Republic. The requirement of the customer was to compile a basic overview of activities aimed at the construction of a deep repository designed for spent fuel and for long lived and high level wastes. Especially, the research, development and design phases were to be thoroughly elaborated so that connections within each topic as well as between different problems are respected.

The technical characteristics of solutions for each topic had to be completed with consideration for time and economy; however, there are some doubts about the reliability of both tasks. The reason is simple. Development of the deep repository needs a very long time (tens of years), and thus, it is possible to make only extremely rough estimates of the time demands, and as a consequence, the economical needs of particular tasks. Furthermore, the time consumption is often dependent on non-technical issues, such as: the influence of public acceptability of particular solutions, licensing period, changes in development procedures due to political decisions, probability of receiving unacceptable results following the repetition of a certain volume of work, consequences of changes in legislation, etc.

The next requirement of the customer was to postulate in detail a 5-year program of work respecting the programs in progress, underlining within each topic the main activities that could, when postponed, retard solution of other development problems, and initiating new key tasks that have not yet been started.

The document, which is the result of the deep disposal project revision, will serve as a guide for all necessary research and development procedures that must be coordinated, and completely and equably answered in all aspects.

19.4 RESULTS OF PROJECT REVISION

To reach all claims during a very short period, the following procedure has been adopted:

- a philosophy of the study was selected that consists in parallel solutions of particular tasks by working groups supervised by an institution and a specialist experienced in waste disposal;
- some comprehensive problems were defined and

supervisors for each of them were nominated;
- nearly every three weeks, coordination meetings were organized in which the progress and results obtained were evaluated and critiqued, and goals for the next period were set up; and
- the final versions of the respective parts were compiled in this document and completed by introductory chapters and annexes.

The above mentioned procedure resulted in a structure of the document which describes a gradual improvement in each problem in connection with new developments in the field of deep disposal as well as their adaption to conditions in the Slovak republic. It is clear mainly when inspecting the diagrams. The diagrams are complemented by commentary that should explain the content of each of the activities. Its other role is to mention problems that could not be easily read from a diagram. The description of each particular task contains discussion about the economical and time aspects for the solution of the topic.

The detailed 5-year plan of activities describes the solution for each particular task in the development of spent fuel and deep disposal of high level waste until the year 2000. The plan contains concrete data on time and economy that are necessary for the deep disposal project realization. The main aim of the short-term plan is to show waste producers, as well as contractors of revisions, an extent, a content and a cost of anticipated works for the preparation of the deep repository construction. In this way, the plan can serve as a background document for planning purposes.

The basic result of the project revision, in which the solution of the whole project is shown, is a "Diagram of deep repository development" (Fig.19.1). This diagram is vertically divided into the main particular tasks, following the text of this report. Horizontal divisions indicate a time succession of activities in each topic and, to a certain level, it also parallels the implementation of the main works. It should be mentioned that the diagram is significantly simplified and that the time axis is not linear, which means that there is no correspondence with real timing of the topics being considered .

The diagram shows one interesting point. Practically every particular task, such as research and development, consists of several phases. They include construction, operation and closure of a facility, and they even include some data on the course of a final evaluation of safety and reliability of the system. When considering that this concerns a time period of more than one hundred years, then the role of a coordinator of all activities, credible working groups, and uncompromising control, supervision and licensing bodies must be stressed.

The main branch of the diagram is connected to a topic "Design activities and construction". This line comprises all principal decision processes. It also includes the key outcomes, design studies and designs, and construction procedures. It can be stated that the other main particular tasks provide the necessary data for decisions of a designer concerning the final solution of an underground repository and auxiliary facilities.

Geological works are connected to an evaluation of the mobility of contaminants in the geological system (far field interactions). Activities contained in this double-topic are focused on selection and verification of the suitability of a site and corresponding geological system for construction of a repository. They also summarize input data for safety analyses of long term behavior of the system.

Studies of the behavior of engineered barriers, final waste forms, packages, overpacks, sealing and filling materials, and underground constructions (near field interactions) are aimed at a choice of an optimum composition of barriers and to evaluate the effectiveness of their retention ability for radiocontaminants. Other species of disposed materials are to be evaluated as well (source term). The results of these studies provide proposals for material composition of repository construction addressed to designers, and data dealing with velocity, mechanisms, and probability of release of immobilized species.

The task of safety analyses is to summarize and evaluate all data received in the course of geological investigation and research, during construction and operation of a facility, at waste treatment processes. In addition, they lay down limits and conditions that any construction structure shall fulfill. Elaboration of safety analysis is the necessary condition for any licensing procedure, and thus, it can be considered to be the absolute key topic of repository achievement.

Quality assurance belongs to a group of activities designed to secure the maximum level of biosphere preservation. Its aim is to work out programs of control and supervision of all activities connected with reposi-

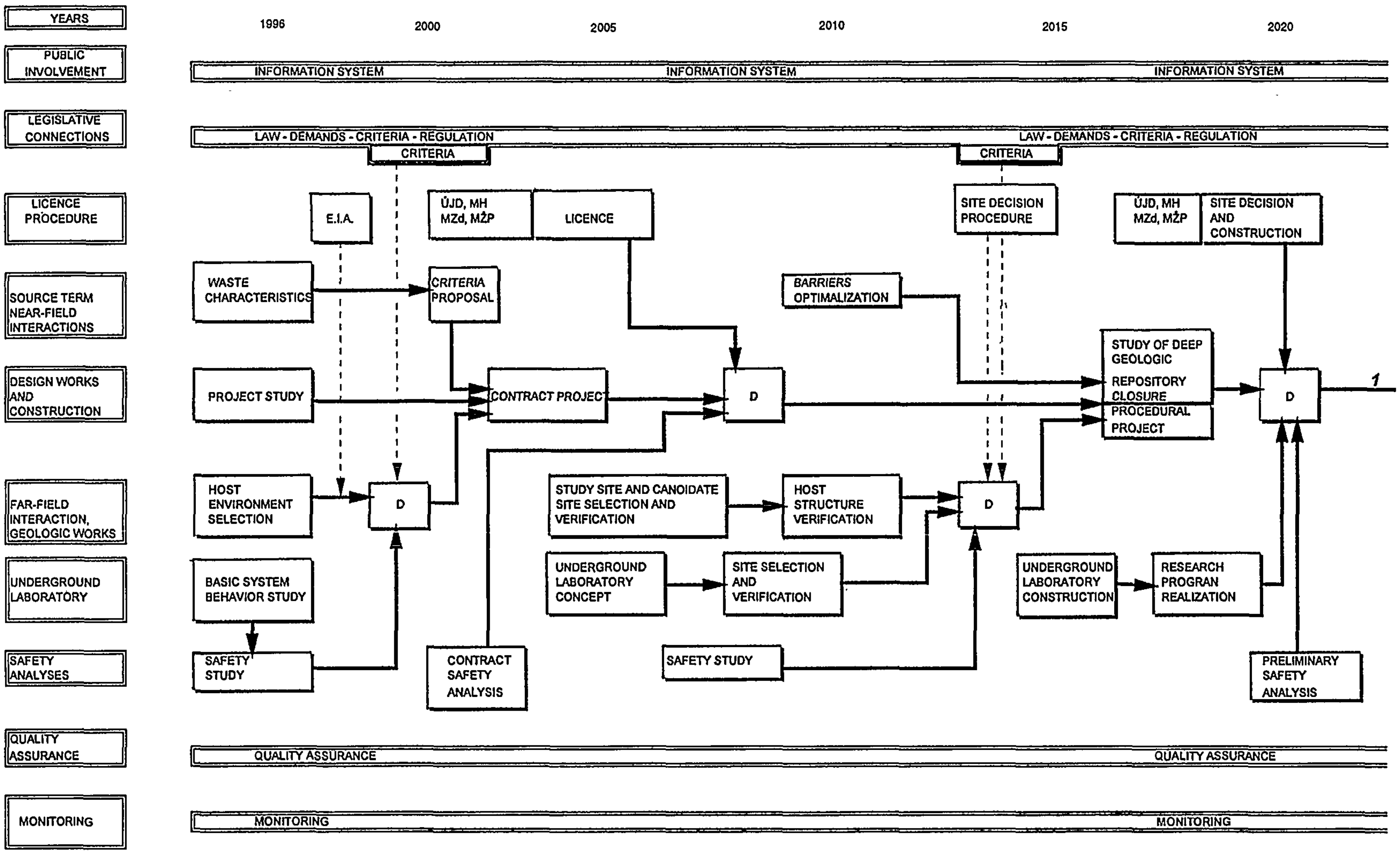

Figure 19.1a. Diagram of repository development.

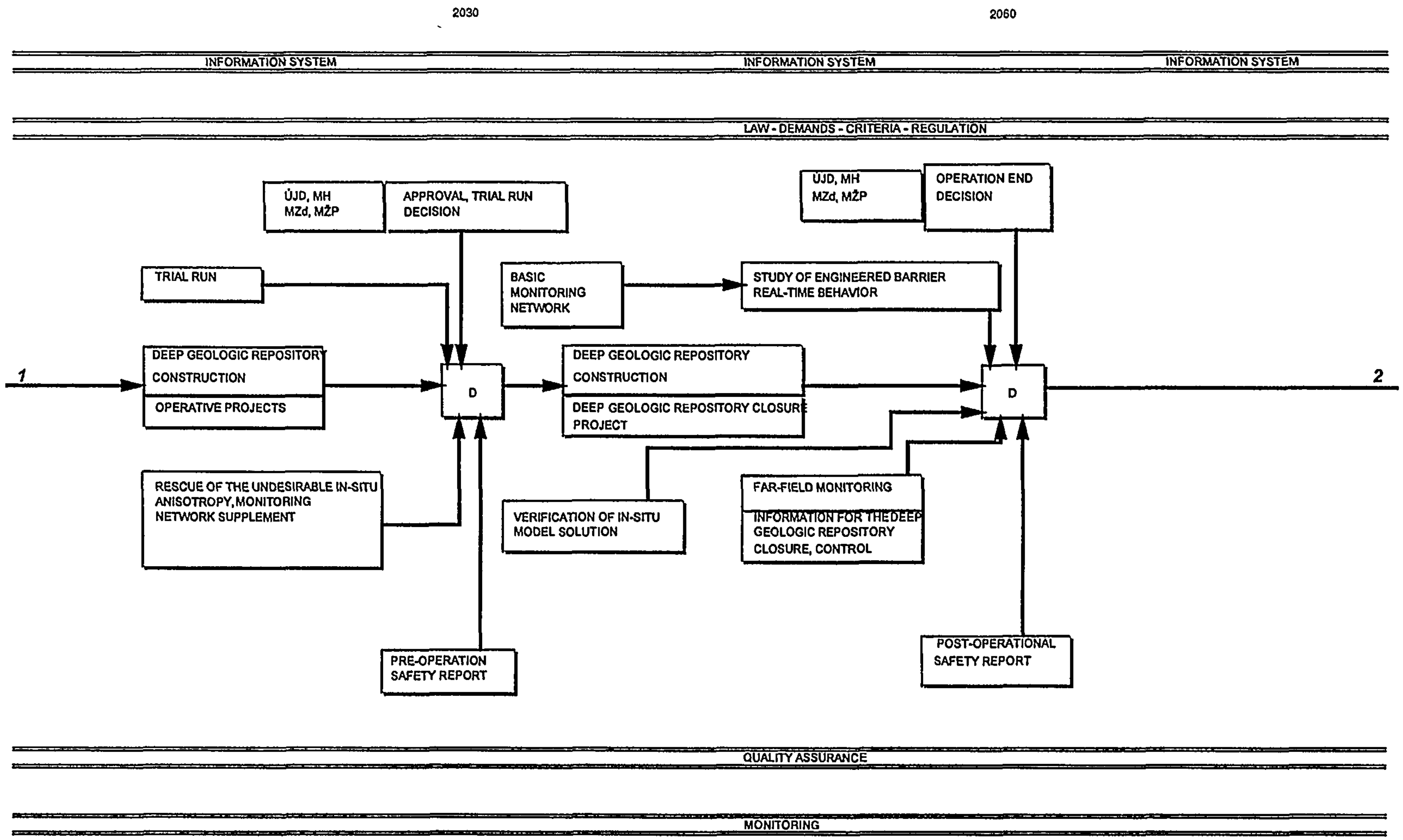

Figure 19.1b. Diagram of repository development (continued).

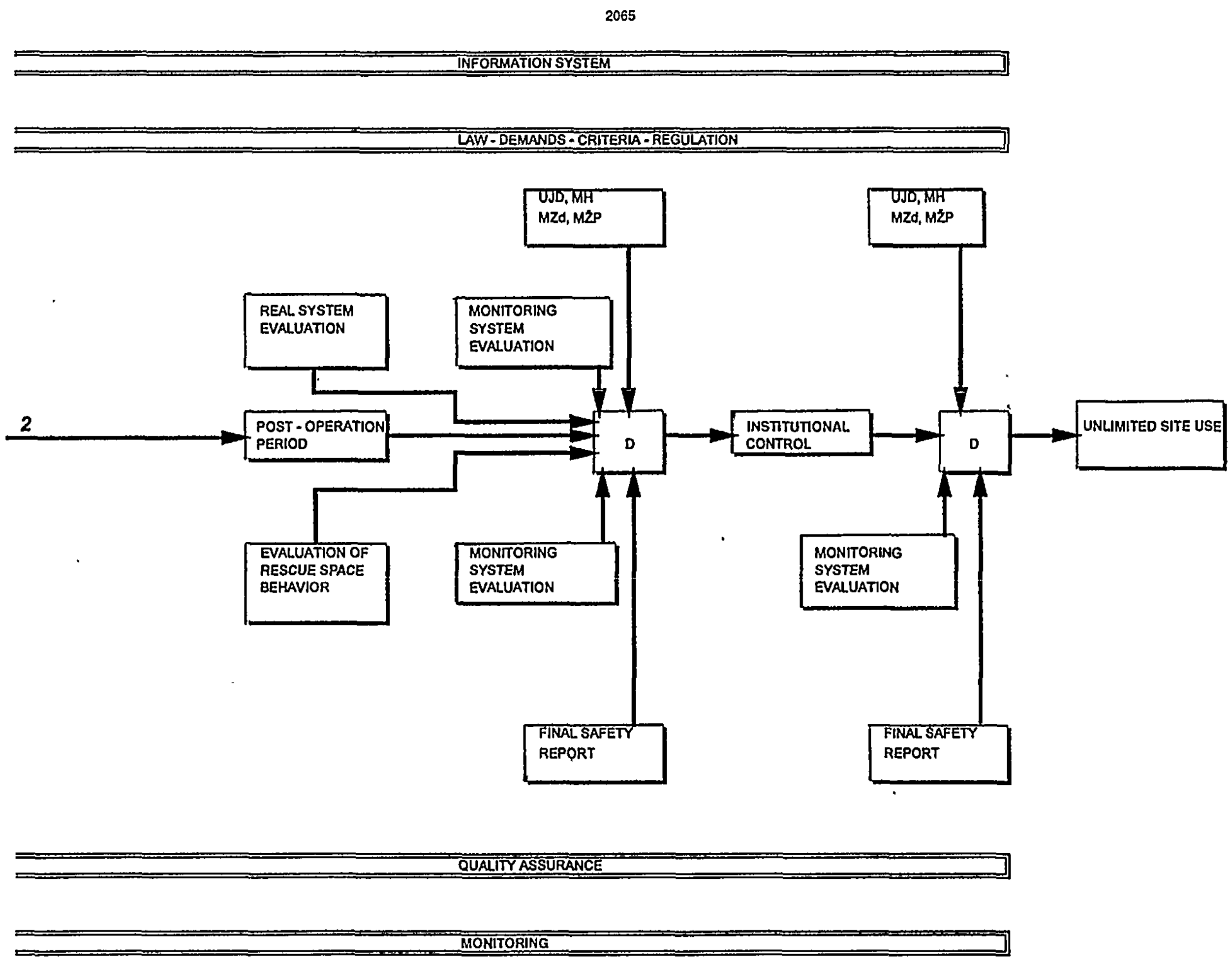

Figure 19.1c. Diagram of repository development (continued).

tory development and construction. The goal of these procedures is to eliminate risks mainly connected with the so called "human factor".

The licensing processes are included in the particular task "legislation connections" because these processes influence significantly all other steps of deep disposal development. They permit each subsequent activity; they may on the basis of some independent evaluation prevent the implementation of such steps. Actually, they are connected to the formulation of claims, requirements, and methods. Sometimes they even set the conditions, and the licensing decisions are issued using an interpretation of these limits and recommendations for a concrete system.

Monitoring in the diagram (Fig.19.1) is considered to be an independent, permanently valid particular task, although in the study, it is described in other particular tasks. Deep disposal monitoring includes a number of steps: radiochemical, hydrological and hydrogeological monitoring of sites from the beginning of research; geological monitoring of a rock structure; measuring the radiological impacts to personnel and the public; recording the changes of state and behavior of barrier materials; meteorological records; observation of destructive and corrosion phenomena, etc. Interpretation of the monitored data is one of the basic conditions required for closure and release of the facility.

The results of particular tasks of the public may by its consequences not only influence but completely change any technical decision. Public involvement in the process of developing a deep repository has at least two aspects. First, passive, which is interpreted as an information campaign about the repository system, design, construction, safety, risks and advantages of its realization. Second, active, which is an effect of public meaning and also the design of the repository by independent opinions and evaluations. A first-rate program on communication with the public may simplify the achievement of the repository; on the contrary ignoring the necessity of reaching a concensus with the general public can completely eliminate the project.

The revised document for a project of deep disposal development is seen first of all as a methodological document. Particular problems of the project may differ in their content, but the approach to their solution has some common features, such as long-term considerations and the principle of conservatism.

The general feature of systems of radioactive waste management is the long-term consideration of all processes, activities and applicable phenomena. This is shown by the fact that partial issues are converted into material outputs after a long period, often reaching tens of years. Time factors bring a number of questions to an evaluation of the behavior of repository elements. To answer them requires finding unusual and substitute ways, e.g. mathematical modeling, studies of natural or man-made analogues.

An indirect interpretation used for waste form behavior during disposal is rather lengthy, and it involves some uncertainty. To reach a desired level of safety and functionality of the system, a principle of conservatism is applied in any evaluations. This means that those phenomena, or some combination, are considered that bring less favorable results.

The important fact in the methodology of deep disposal development is to have an objective approach to any step, activity or decision. That kind of solution is provided by preparation and realization of quality assurance programs. The main part of those programs is multiple opinions and evaluations of solutions. There is a tendency to leave out this demand. This is dangerous from two points of view. Any incorrect decision may cause irreparable damages even resulting in canceling the previous results, or the process of repository development may be suspended by a qualified opposition because of the inability technically to defend chosen solutions.

19.5 CONCLUSION

The revision of this project is the result of the work of a group of employees of the following institutions: DECOM Slovakia Ltd., Nuclear Power Plant Research Institute, Geologic office of Slovak Republic and EPG Invest Ltd. It has been worked out under the technical and editorial coordination of the DECOM Slovakia Ltd., however, separate authors are responsible for their own sections. All of those involved have attempted to develop an approach in the most objective way.

The goal of the activities was to perform a revision of the project from the year 1993, in which must be included the deep disposal development of the former Czechoslovakia, to incorporate the new world results of this field development, to modify it for conditions of Slovakia, and to add the economic and time considerations

to the technical solution. The determination of concrete activities through the year 2000 was an inseparable part of this methodological document. Each particular task of the project contains ideas about concrete activities, financial costs and solutions as well.

A summary of the above mentioned data in one document creates the material, which defines concrete activities and the needs of finance to assure the required output from a solution for a deep repository development under the conditions of Slovakia.

CHAPTER 20

GEOLOGICAL ASPECTS OF SITE SELECTION FOR LOW AND INTERMEDIATE LEVEL RADWASTE REPOSITORY IN SLOVENIA

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Abstract. According to the guidelines for a low and intermediate level radwaste repository site selection in Slovenia, the siting process has been divided into four steps. The first three steps of the surface site selection were completed in 1993. A set of the most exclusionary geological criteria to be applied in selecting the surface site is described. Some reasons for the failure of this process are also described. Since the fourth step was stopped due to strong public opposition, an alternative of underground disposal is now being considered. In 1994, the Agency for Radwaste Management started the preparation of basic guidelines for underground repository site selection. Joint recommendations, that consider both surface and underground site selection parameters, are now being developed in the Slovenian Nuclear Safety Administration.

20.1 INTRODUCTION

The guidelines for the selection of a low and intermediate level radwaste (LILW) disposal site were setup in 1991. The guidelines that were announced included rules according to which, under the given urban and social conditions, the most suitable site for the shallow ground disposal of LILW in Slovenia would be selected. Both the existing world-wide experience and the national regulatory conditions in Slovenia were considered in creating these guidelines.

In selecting disposal sites, it was necessary to have a detailed knowledge of the process of migration of contaminants into the biosphere. Slovenia has a very sophisticated geological and tectonic setting dominated by various combinations of geological structural elements such as: faults of different type and age, overthrusts, folds, naps and lateral transformations of different lithologic units. In most cases, it was very difficult to determine the migration of radionuclides in underground water. Thick layers of impermeable rocks are the only reliable natural barrier in such geological and hydrogeological conditions.

However, the requirements given by the guidelines are that a shallow disposal site is to have rocks of low permeability in the basement, and a distance to the underground water table that is as large as possible. Sites with these geological conditions, such as saturated clay marls, were the only ones selected as being acceptable. These rocks, regardless of fracturing in neighboring layers, provide a sufficient natural barrier to prevent migration of radionuclides.

20.2 SITE SELECTION PROCESS FOR LILW

The procedure used in selecting disposal sites was divided into three steps containing 43 criteria. In a final fourth step, the technical confirmation was based on a detailed field examination of the geology, hydrogeology, and seismology of the site. Each step was terminated by a presentation to the public of the results.

In the first step, unsuitable areas were excluded by taking into consideration certain exclusion criteria, such as: national parks, urban zones, ground water resources, presence and location of active faults, geothermal areas, flood areas, presence of ores, minerals, oil, gas, hydraulic conductivity, soil composition, thickness and extent of geologic units.

In the second step, the remaining acceptable areas were evaluated according to land use, water resources, seismic and geological criteria, so they could be further reduced to so called potential sites.

In the third step, several of the most suitable of the potential sites were chosen by comparing their locations on the basis of the following criteria: population, eco-

nomic feasibility, transport, ecological value, and public acceptance.

In the final fourth step, a comprehensive analysis of the most suitable sites from the third step was carried out by applying the criteria of the previous steps and additional criteria concerning the corrosion of waste containers (biological processes, chemical properties of the soil and groundwater), and then a detailed field investigation was carried out to confirm the suitability of the sites. The results of the fourth step produced one or two of the most suitable sites that were considered to be technically confirmed. A schematic diagram of this process is shown in Figure 20.1

20.2.1 Step One of Site Selection

In carrying out Step One, a series of overlaying maps were used which contained areas that are defined by seven exclusionary criteria as described in Table 20.1. This process eliminated the unsuitable areas of the Republic of Slovenia from further consideration.

After considering the exclusionary criteria of the first step, the acceptable areas for an LILW repository site in

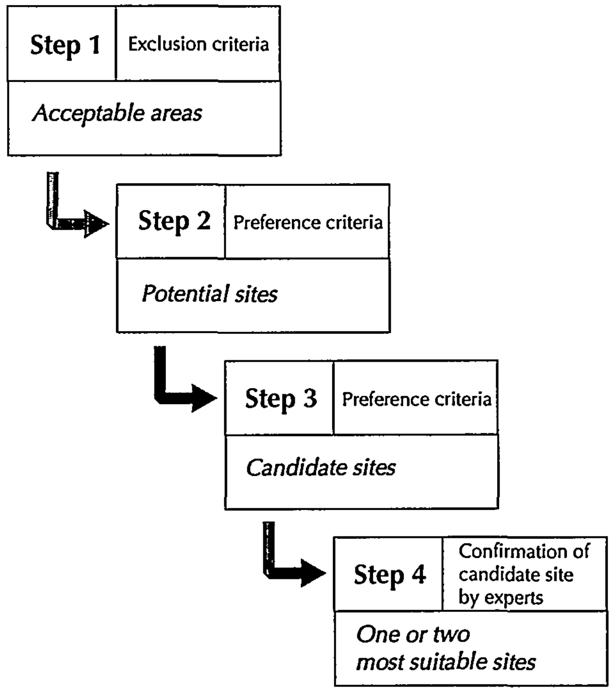

Figure 20.1. Schematic diagram of the site selection process for LILW in Republic of Slovenia.

Table 20.1. Exclusionary criteria of Step One.

Exclusion Criteria	Explanation
National Parks	The areas defined as national parks are excluded.
Urban Zones and Settlements	Excluded are all areas defined as settlements with more than 5000 inhabitants.
Drinking Water Resources–Aquifers	Excluded are all areas defined as drinking water resources.
Known Active Geological Faults, Geothermal Areas and Seismicity	Excluded are all areas located on a known active fault at a distance up to 3 km and the areas where the expected earthquake acceleration exceeds 0.3 g.
Flood Areas	Excluded are areas which are located in an area of 500 year floods.
Presence of Ores, Minerals, Oil and Gas	Excluded are areas with proven resources of ores, mineral, oil and gas.
Geological and Lithological Soil Composition	Excluded are the areas where surface homogeneity of layers is smaller than 300x300 m and the quotient between the thickness and hydraulic conductivity of layers is smaller than 5×10^9 s. Excluded are lithological layers having a hydraulic conductivity greater than 1×10^{-8} ms^{-1} and a thickness of layers smaller than 20 m.

Slovenia were identified. The potentially acceptable areas were those that had not been excluded according to any criterion of Step One. All of these areas were considered to be equivalent, i.e., the acceptable areas had not been assessed and evaluated. Figure 20.2 shows the locations of the acceptable areas after the application of the first step.

20.2.2 Step Two of Site Selection

In carrying out Step Two, the preference criteria were divided into four groups: geological, seismic, land use, and potential water management. These criteria were then applied to the acceptable areas selected in the first step.

The following geological preference criteria were applied:

- Presence of groundwater;
- Site seismicity;
- Presence and vicinity of active faults;
- Exploitation of ores/minerals, oil and gas;
- Areal extent of host rock;
- Thickness of rock mass;
- Soil instability;
- Erodibility;
- Rock composition and hydraulic conductivity;
- Angle of slopes; and
- Radionuclide paths to the biosphere.

The result was the selection of 36 potential sites occupying a total area of approximately nine km^2.

The examination of the potential locations was performed at the end of the theoretical studies to verify the procedure, and to determine discrepencies in the results obtained. This examination resulted in an expert conclusion that: (a) five locations are not suitable for the construction of a repository, and (b) another five locations are only suitable for a tunnel type repository and not for a surface type as previously envisioned. One potential site, suitable for both types of repository, was also identified and considered in further analysis.

The results of the second step of surface repository site selection were reviewed by a group of experts that confirmed the accordance of the procedure with the guidelines.

20.2.3 Step Three of Site Selection.

In the the third step, five candidate sites were selected among 36 potential sites from the second step. The

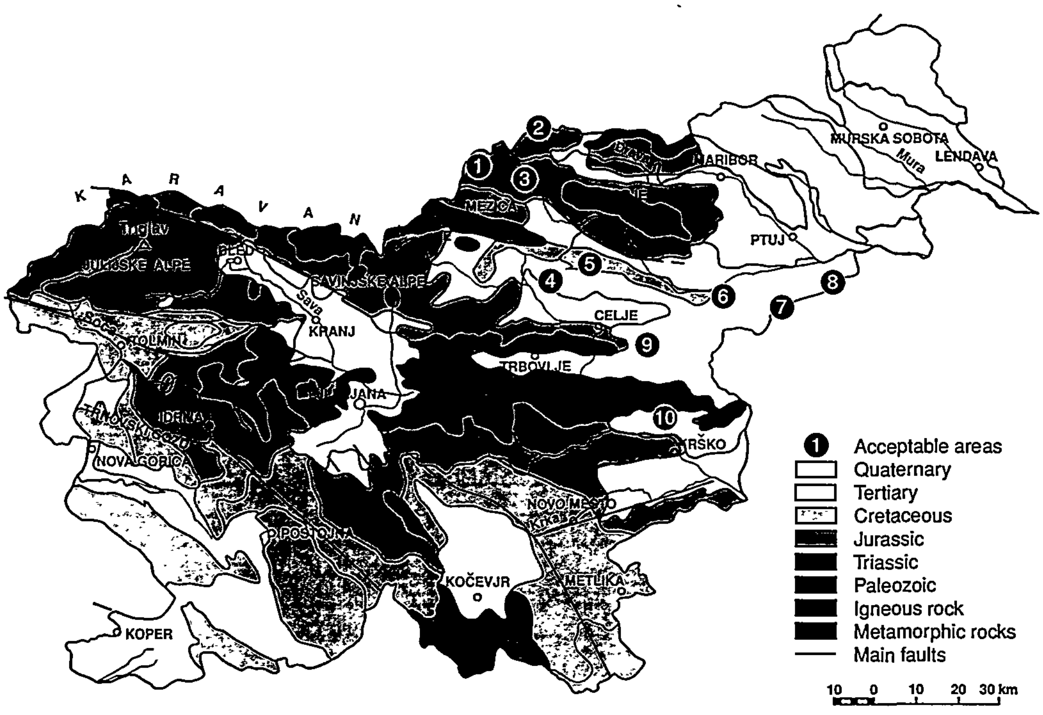

Figure 20.2. A generalized geological map of Slovenia with acceptable areas after the application of Step One.

method of assessment was based on the repeated use of the criteria from the first two steps. In addition, preference criteria concerned with economic and technical feasibility, transport, and social acceptability were considered as well.

One of the selected sites was found to be suitable for a surface repository, two, for a tunnel repository, and the remaining two sites were appropriate for either type of repository, surface or tunnel. The main geological characteristics of the sites are summarized in Table 20.2.

In accordance with the practice from previous steps in this procedure, the results were presented to the public. The presentation was not successful and has provoked strong disapproval within the local communities. Their representatives declared that waste disposal in the vicinity of their communities was not acceptable. It was evident that public acceptance of the candidate locations could not be achieved. Therefore, it was impossible to proceed to the fourth step in which the most suitable locations could be verified and approved by the experts. The project was stopped.

20.3 Did Application of Geological Criteria Influence an Unsuccessful Surface Repository Site Selection?

It is clear the natural site characteristics play an important role in the selection process for a radioactive waste

Table 20.2. The main gelogical characteristics of the sites.

	Main Geological Characteristics	Site 1 Surface Repository	Site 2 Tunnel Repository	Site 3 Tunnel Repository	Site 4 Surface or Tunnel Repository	Site 5 Surface or Tunnel Repository
HOST ROCK	Rock Type	Sandy marl	Sandy marl	Sandy marl	Marl	Marl
	Permeability (m/s)	10^{-9}-10^{-11}	10^{-9}-10^{-11}	10^{-9}-10^{-11}	$10^{-11} < K < 10^{-9}$	$10^{-11} < K < 10^{-9}$
	Relative Porosity (%)	22	22	22	20-33	20-33
	Thickness of Layer (m)	300-400	300-400	300-400	50-400	50-400
	Areal extent of Rock Mass (ha)	11	332	104	19	28
	Angle of Slope (°)	10-20	5-20	5-20	10-15	10-20
	Erodibility (mm/300 yrs.)	4.2-5.4	3.2-7.5	6.4-8.6	5.4-6.4	4.2
	Point Load Index I_s (50) MN/m^2	0.87-1.94	0.87-1.94	0.87-1.94	0.09-0.31	0.09-0.31
	Unconfined Compressive Strength (MPa) Q_u=22 I_s	17.17-42.71	17.17-42.71	17.17-42.71	1.99-6.73	1.99-6.73
	Natural Volume Weight (KN/m^3)	20.95	20.95	20.95	18.22-19.43	18.22-19.43
Distance from Active Faults (km)		3-4	3-4	3-4	3-4.6	3-4.6
Max. Expected Horizontal Acceleration in a Time Period of 1000 years (cm/s^2)		190	190	190	250	250
Max. Expected Local Intensivity in a Time Period of 1000 years (MCS)		8	7-8	7-8	8	8

repository site, and that a site within an appropriate geological environment is, to a great extent, based on geological conditions.

But geological criteria, applied to the process of selecting a surface repository site in Slovenia, used only properties of the geological barrier and took no account of the other two barriers, i.e., conditioned waste and engineered barriers. In other words, the objective of finding a site for a surface repository was *to find a location with geological properties (natural geological barriers), where engineered barriers would not necessarily be used to achieve the safety standards.* This was certainly the most economic way for repository construction, but on the other hand, the site selection was exceptionally difficult.

The siting process applied highly quantitative exclusionary or preference geological criteria, i.e., geological criteria were stated with numerical values of the geological parameters, that made the whole siting process very inflexible. Some geological criteria, according to their "importance" (according to the size of the excluded areas in surface repository site selection) are presented and described in the following.

The criterion of "active faults" as a single tectonic exclusionary criterion has eliminated 97% of Slovenia for the purpose of siting a repository. To meet this criterion, areas located in the vicinity of a known active fault at a distance up to three km were unsuitable.

Considering the same criterion in the second step, i.e., presence and vicinity of active faults, acceptable areas from the first step ranged over the following distances:

- Unsuitable sites, where the site was to be located on, or near a fault, at a distance up to 3 km;
- Less suitable sites, where the distance from the fault is 3 to 8 km; and
- Suitable sites, where the distance is greater than 8 km.

It should be noted that up to the third step, the work included office work only, and no site investigations were performed to confirm activity of the faults.

Although Slovenia lies in a seismic territory, and tectonic causes of seismic activity, i.e. surface faults, are distributed all over Slovenia, there is a basic question (that could be discussed) whether the application of a uniform step-off distance is a matter of policy rather than being grounded on technical principles. Without detailed site investigations, it is difficult to select suitable locations, and the geological properties of specific sites must first be confirmed through field investigations.

According to the criterion for "Active faults", the WAMAP mission[5] recommended that at an early stage, it is important to decide on the definition for an active fault and the significance that rock structure could have on the integrity of a repository over its 300-year assessment period. It is necessary to have a single representative data base, or set of maps, supporting the interpretation and application of this criterion.

A similar situation occurred in the first step in connection with the "Lithology" criterion, where rocks with a hydraulic conductivity greater than 10^{-8} m/s, a thickness of layers less than 20 m, and a seismicity where earthquake accelerations greater than 0.3 g would be expected, were recognized as unsuitable. In further analysis, the Lithology of acceptable areas was compared considering the preference criteria "Areal extent of host rock" and "Thickness of rock mass", where the area suitability increased with extent (greater than 300 x 300 m, i.e., 9 ha) and layer thickness (more than 20 m). Again, this site analysis only involved office work. No field investigations, to confirm exclusive parameters for the rocks, was made.

In considering the preference criteria "Site seismicity", acceptable areas ranged between unsuitable (where the expected earthquake accelerations a_{max} exceeds 0.3 g for a period of 1,000 years), less suitable (a_{max} = 0.15 to 0.3 g) and suitable (a_{max} is less than 0.15 g). The maximum horizontal ground acceleration for the territory of Slovenia was evaluated with a probabilistic seismic hazard analysis.

The mission report[5] suggested that in general, an application of highly quantitative exclusion or site preference criteria, especially at an early stage of the selection process, was not recommended.

Quantitative criteria, as applied in the siting process and as described above, can only be used where quantitative data are available to justify their use, i.e., data confirmed by site investigations. Much of existing technical data is regional (non-site specific) and qualitative in nature as well. Some criteria, used in the first (exclusionary) step simply assume certain site specific data, which would only be available in the necessary detail after a careful

site investigation. Such criteria should therefore be left to the appropriate later steps in repository siting[5].

It is very important to recognize the uncertainties in understanding the geology during the siting process, when data are based only on office studies. The actual site conditions may be significantly different from those envisaged, and as a result, the site selection process must remain flexible enough in order to accomodate unexpected features. More confidence can be placed in sites selected in a location where the geological structure is non- or less complex.

20.4 NEW APPROACHES

According to the fact that the necessity for the final disposal of low and intermediate level radioactive wastes is growing, the final location of the disposal site should be selected within the next five to ten years. The existing wastes are temporarily stored in interim storage facilities located at the Krsko nuclear power plant.

It is obvious that problems concerning final disposal of LILW should be solved in a satisfactory manner in the near future. Solutions for this problem are being searched for in the following directions:

1. In verification, new estimates and corrections of the three most exclusionary geological criteria (active faults, seismicity, and hydrogeological parameters), but the most important features have been revealed in the application to sites for final waste disposal.
2. In considering the newest techniques and technologies that have been developed in disposing of, and protecting, radwastes in the developed countries, and in reconsidering geological criteria in this new light.
3. In taking into consideration the possibility of underground waste disposal of LILW in geological structures, and in this way minimizing the risks arising from the seismicity and activity of fault zones.

In the analysis of the reasons for failure in the first campaign, it appeared that there was a bad coordination between experts in the different fields of science. For example, geologists considered "impervious" rock as the only suitable rock for a disposal site, regardless of the possibility of using engineered barriers (such as, canisters, filling materials, etc.). It is well known that over a period of 300 years, which is the time necessary for the radioactivity to decay to normal levels, it is possible to produce effective engineered barriers. Therefore, the requirement for an "impermeable" basement beneath the deposit is no longer necessary. On the contrary, in some repositories, such as Centre de l'Aube, a permeable basement is part of the design of the facility.

The Agency for radioactive waste disposal in Slovenia has also noted this deficiency from the first campaign. The new approach was therefore to provide some basic technology to the experts who don't have much experience in dealing with problems of packing and deposition. In this way, the Agency expected to ameliorate the cooperation of these experts with that of others.

In accordance with this new policy, the Agency has redirected geological experts to review the new technologies in the field of radwaste disposal in the developed countries. A series of such reviews have been carried out in which the first aim has been achieved; the geological experts of today are well acquainted with the technological possibilities and requests for construction of surface, or underground, disposal facilities. In addition, the Agency has made it possible for some of the experts to visit existing sites, and to meet other geologists and experts in other fields of science at international conferences. In this way, our geologists not only gathered new data, but also established contacts with colleagues from different European countries, exchanged opinions and learned new ways of thinking. It was especially useful for us to learn of unpublished experiences (both good and bad) that led to the solution of problems on multinational projects (such as the underground laboratories at Mol in Belgium, Grimsel in Switzerland, etc.).

Based on this new knowledge, a set of six possible types of disposal facilities has been defined for Slovenia, which include geological and rough technological conditions. They provide a basis for new considerations and estimations in carrying out campaigns of field investigations.

The existing criteria from the first campaign have been thoroughly reexamined. The result is a new approach in the evaluation of the exclusionary criteria. The philosophy has changed; the elimination of a site or a region on the basis of a certain criterion should be based on direct or indirect evidence.

There is another novelty in our way of thinking; we no longer look for the geologically best location, but for all acceptable locations. In this way, these locations are also available for analysis using other necessary criteria. We no longer have criteria for site selections or the elim-

ination of territories, but guidelines that can give us an indication of possible problems. This approach does not limit our decisions in advance, and thus enables a more flexible treatment of the site selection process.

Since the site selection process for a surface repository has been stopped, this new approach is more likely to gain public acceptance by disposing of radioactive wastes in an underground facility.

The expansion of the site selection program to include an underground disposal facility gives us another possibility, and is the result of the new approach over the last few years. New geological guidelines for underground low and intermediate level waste disposal have been made and revised, and on this basis, new geological guidelines for surface disposal of LILW have also been remade.

Since some geological criteria are more important and can be more applicable to underground than to surface repository site selection (or vice versa), the proposed criteria differ, in many respects, from those for surface site selection. With regard to seismicity for example, underground structures are less susceptible to seismic disturbances than surface structures due to the fact that effects from earthquakes diminish with depth. Different transport pathways for radionuclide migration through groundwater to the biosphere should be considered in both site selection processes as well.

By placing the disposal system underground in rock means, on the one hand, having the possibility to minimize the influence of the most selective criteria used for a surface repository; and on the other hand, providing an underground disposal facility that, hopefully, would be more acceptable to the public.

The new proposed guidelines for underground LILW disposal consist of the following main parameters. (We are presenting them here to show the differences with the first criteria used in the selection process for a surface site.)

- Geological rock structure
  - volume
  - simplicity
- Lithology
- Hydrogeological conditions
  - permeability
  - hydraulic gradient
- Migration
  - geochemical properties of rock and soil
  - geochemical properties of groundwater
- Active endogenetic processes
  - seismicity
  - recent fault movements
  - volcanoes
- Rock disturbance
  - human reasons
  - natural reasons
- Potential resources
  - value
  - genesis
  - technology
- Geomorphologic stability
  - surface stability
  - water degradation processes
  - extreme climates
- Geomechanical conditions

The Agency for radwaste disposal, being responsible for the site selection process in Slovenia, will have to use this new approach and also help it to find its way to the public. Reports of all studies made are available in the Central Technical Library, and summaries of these studies are translated into English. This enables all concerned to be kept informed about the dangers, scientific approaches, and other work done on prevention and on site location for a disposal facility.

REFERENCES

1. Faculty for Architecture, Civil Engineering and Geodesy, Department of Civil Engineering and Geodesy, Site Selection Repository Low and Intermediate Level Radwaste in Slovenia Step 2 research, Seismicity, March 1991.

2. Geological Survey Ljubljana, Institute of Geology, Geotechnics and Geophysics, Site Selection Repository Low and Intermediate Level Radwaste in Slovenia Step 2 research, Conclusions, June 1991.

3. Geological Survey Ljubljana, Institute of Geology, Geotechnics and Geophysics, Site Selection Repository Low and Intermediate Level Radwaste in Slovenia Step 3 research, Geological Estimation sites, June 1992.

4. Geological Survey Ljubljana, Institute of Geology, Geotechnics and Geophysics, Geological Guidelines for Site Selection of an Underground Disposal of LILW, Ljubljana, 1995.

5. IAEA, WAMAP Mission to the Socialist Federal Republic of Yugoslavia, Travel Report, April 1991.

6. IBE, Low and Intermediate Level Radwaste Repository Site Selection in the Republic of Slovenia, Steps 1, 2, 3, Ljubljana, 1990, 1992, 1993.

7. Jeran, M., Radioactive Waste Repository Site Selection in the Republic of Slovenia, 1. Meeting of the Nuclear Society of Slovenia, Bovec, June 1992.

8. Republic Administration for Nuclear Safety, Guidelines for the Low and Intermediate Level Radwaste Repository Site Selection in Slovenia, Revision 0, February 1990.

9. Republic Administration for Nuclear Safety, Guidelines for the Low and Intermediate Level Radwaste Repository Site Selection in Slovenia, Revision 1, September 1991.

CHAPTER 21

RADIOACTIVE WASTE MANAGEMENT IN SPAIN MAIN ACTIVITIES UP TO THE YEAR 2000

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21.1 INTRODUCTION

In 1995 there were nine nuclear power stations in operation in Spain with a total capacity of 7.4 GW, supplying about 36% of the Spanish electrical energy. At present, spent fuel from the nuclear power plants is stored on site in pools constructed for this purpose. As of the end of 1993, there were 1457 tU of spent fuel.

The estimate of the total volume of high level wastes, spent fuel, that will have to be managed in Spain is up to 11,700 m^3 (40 years of expected life). Total volume of LLW is expected to be about 200,000 m^3 of which about 137,000 m^3 will be from the dismantling of power plants.

The strategy and main activities for the definitive disposal of high level, long lived wastes are given in the General Radioactive Waste Plans (GRWPs). Following a period of intermediate storage of these wastes, their transport and encapsulation, they will be disposed of in a deep geological formation. Transport of the spent fuel will be carried out by ENRESA as the responsible authority, either using the company's own resources or through specialized firms.

Waste conditioning or encapsulation will be carried out at a plant that is planned to be constructed on the same site as the disposal facility. The technique to be used for disposal will be based on a programme of study, research and international co-operation.

Disposal of LLW, will continue in the "El Cabril" facility, which has been in operation since 1992.

The legal framework governing radioactive waste storage facilities and radioactive or nuclear installations in Spain is established by the Nuclear Energy Act 25 of 1964 and the Regulations on Nuclear and Radioactive Installations of 1972.

The "Consejo de Seguridad Nuclear" (CSN), the Spanish Nuclear Safety Council, was constituted in 1980 as an organization existing under Common Law, independent from the Central State Administration. CSN has its own legal standing and corporate assets independent from those of the State, and is the only body in Spain with responsibility in the fields of nuclear safety and radiological protection.

The management of radioactive wastes in Spain is undertaken by "Empresa Nacional de Residuos Radioactivos, S.A." (ENRESA), the Spanish national radioactive waste company, constituted in 1984. Eighty percent of the company is held by the Spanish Centre for Energy, Environmental and Technological Research (CIEMAT), previously known as the "Junta de Energia Nuclear" (Nuclear Energy Council).

21.2 LOW AND INTERMEDIATE LEVEL WASTES

The strategy applied to low and intermediate level wastes continues to be based on a one-to-one relationship between the disposal facility and the wastes themselves. Two major courses of action have been established. The first includes the conditioning, transport and characterization of radioactive wastes and corresponding acceptance criteria, as well as the inspection criteria and procedures required to guarantee compliance. The second includes the design, construction and operation of the disposal facilities.

ENRESA was awarded a Provisional Operating Permit for the Extension to the Nuclear Installation for the Disposal of Solid Radioactive Wastes located in Sierra Albarrana by a Ministerial Order issued on 9th October 1992. As a result of this award, the installation in ques-

tion, known as El Cabril, will be used over some 20 years for many of the stages involved in managing the LILW generated in Spain, such as conditioning, characterization and disposal. This waste constitutes a new operating stage of special relevance in our country.

21.2.1 Waste Conditioning, Transport, Characterization and Acceptance.

Except in the case of the minor producers, the previous treatment and conditioning of low and intermediate level wastes is the responsibility of the producer, who is obliged to generate packages satisfying the acceptance criteria defined by ENRESA for subsequent conditioning and disposal at the El Cabril facility. In the case of the minor producers, waste treatment and conditioning is carried out at the aforementioned facility.

Transport of the wastes is carried out by ENRESA as the responsible operator, either using its own resources for the removal of wastes generated by the minor producers, or the services provided by specialist companies in the case of conditioned wastes.

The contracts signed between ENRESA and the waste producers include the criteria and technical specifications to be considered in relation to the characterization and acceptance of wastes for subsequent disposal at El Cabril.

A key component in the process of waste quality verification, which to date has been mainly performed abroad, has been the construction in Spain of a Low and Intermediate Waste Quality Verification Laboratory for performance of the corresponding tests (destructive testing, verification, characterization, etc.). This laboratory is part of the El Cabril installations, along with the Disposal Structure Conditioning Plant and other Services.

21.2.2 Disposal of Low and Intermediate Level Wastes

With a view to ensuring the disposal of the low and intermediate wastes produced in Spain, ENRESA operates the El Cabril centre, located in the province of Córdoba; an extension of the works at this facility was completed in 1992.

El Cabril incorporates the most advanced technologies used for this type of installation. Technically, the facility is based on a system of shallow disposal with engineered barriers, similar to the French model. This system guarantees compliance with the necessary safety objectives and criteria, such that there will be no significant radiological impact during the period required for the activity of the wastes to decay to harmless levels. The facility is made up of the following buildings and structures as shown on Figure 21.1:

1. Low and intermediate level waste Conditioning Building, which houses the necessary treatment and conditioning systems (compaction, incineration, manufacturing of hydraulic conglomerant, etc.) for the liquid and solid wastes arising from the application of radioisotopes in medicine, industry, agriculture and research; the solid wastes from CIEMAT, Juzbado Uranium Concentration Plant and the nuclear power plants, and the wastes generated at El Cabril itself as a result of operations.
2. Disposal Structures for the duly conditioned low and intermediate level wastes from the Spanish nuclear and radioactive installations. These structures consist of cells aligned in two rows along two esplanades; it is estimated that their capacity will cover Spain's needs until the end of the first decade of next century (see Fig. 21.2 for layout of disposal platforms).
3. Quality Verification Laboratory where the processes of characterization, testing and control of the characteristics of radioactive packages received or conditioned at the facility are carried out, and for research activities aimed at enhancing the processes of low and intermediate level waste conditioning and characterization.
4. Services and Control Building where industrial safety, reception, technical services, general services, maintenance workshop, concrete container manufacturing and administration are carried out.

The El Cabril facility has been operational since October 1992, when the buildings and structures described above were constructed and the necessary assembly operations and tests were performed.

Up to that date, ENRESA had stored the conditioned low and intermediate level wastes from CIEMAT and the minor producers in the surface modules of the old El Cabril installations. In recent years, these modules have also been used for packages from the José Cabrera, Santa María de Garoña and Ascó nuclear power plants. The other (conditioned) low and intermediate level wastes generated in Spain are temporarily stored at the producers' authorized on-site installations awaiting transfer to El Cabril.

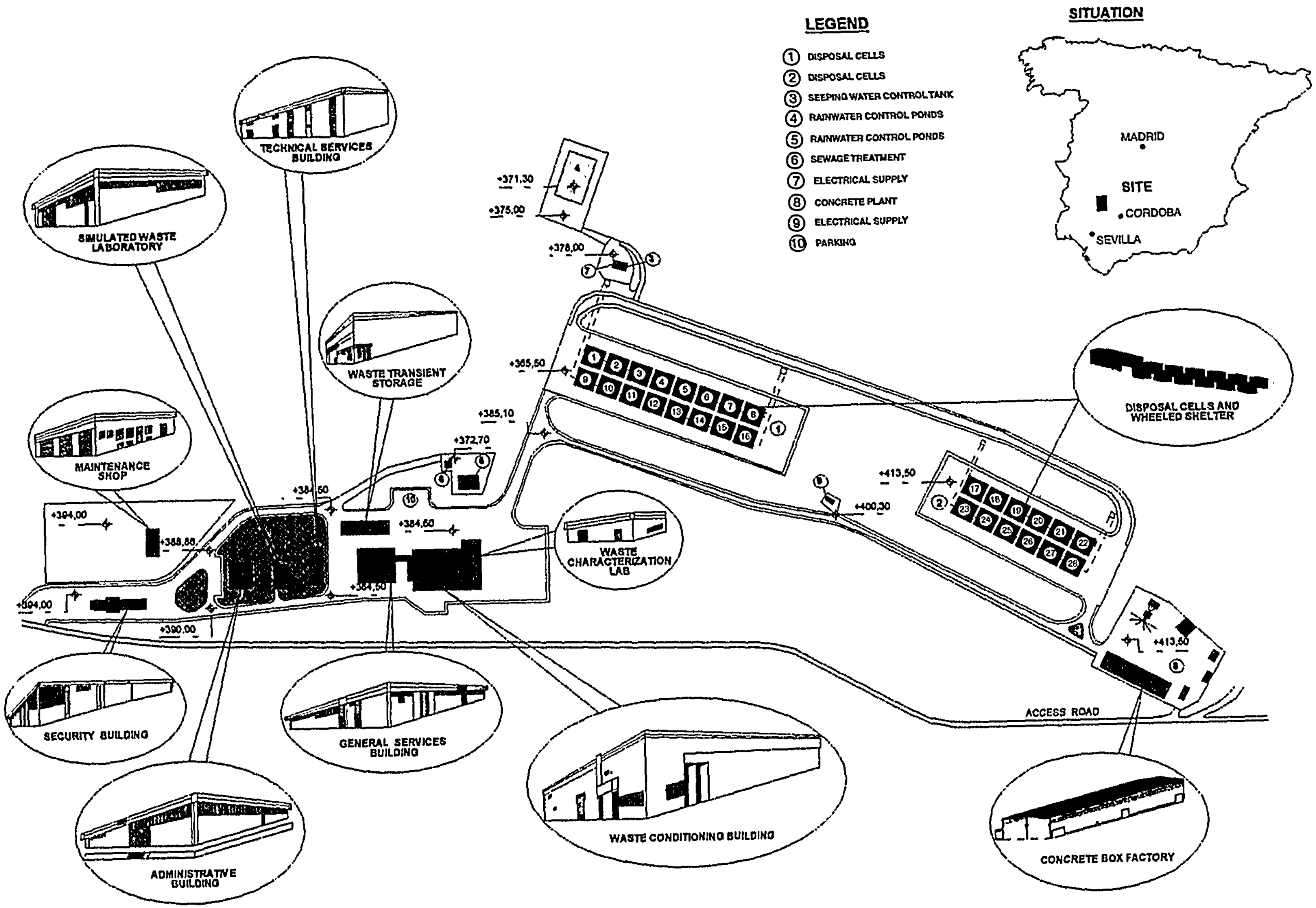

Figure 21.1. General layout of the "El Cabril" LLW disposal facility.

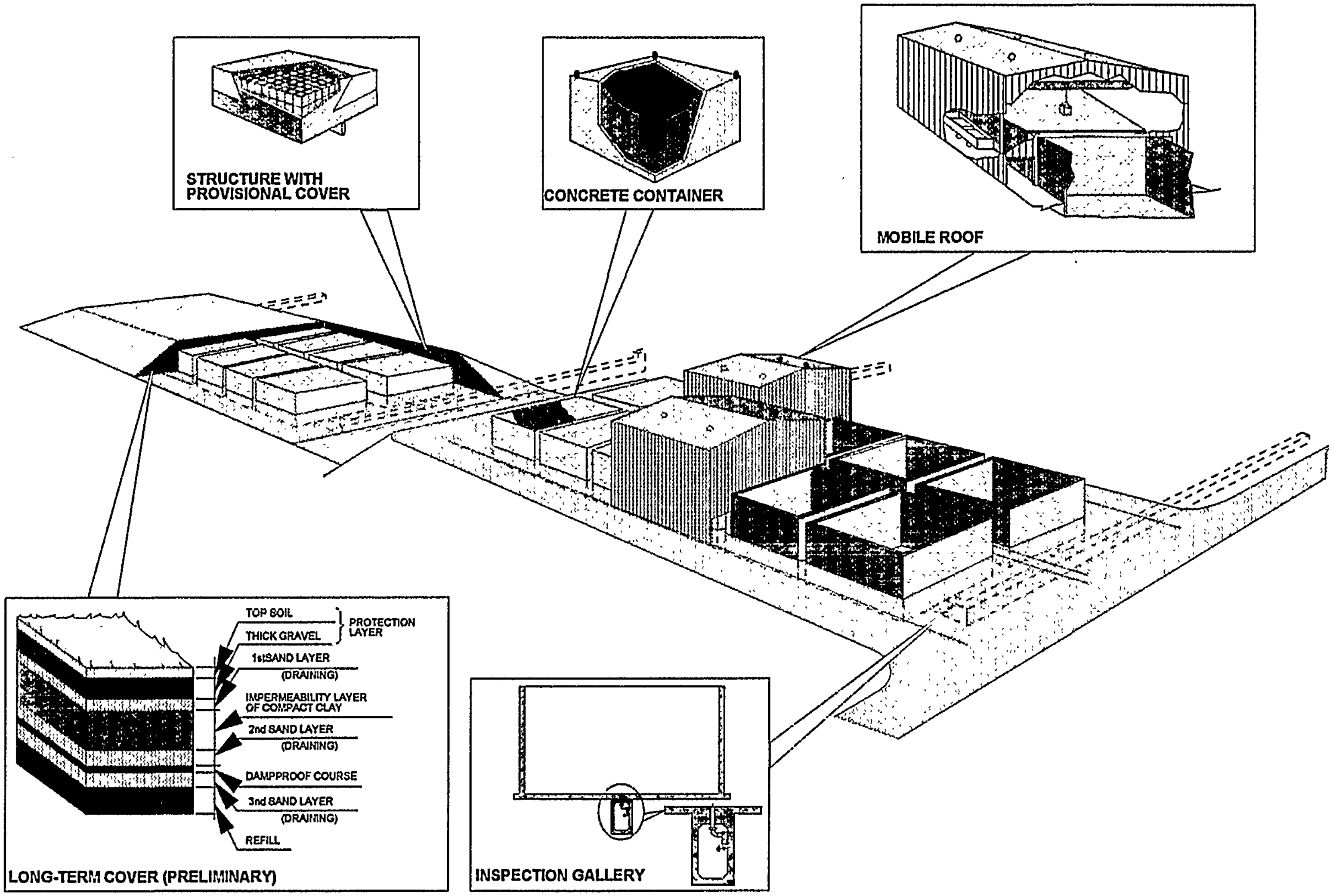

Figure 21.2. Layout of the disposal platforms.

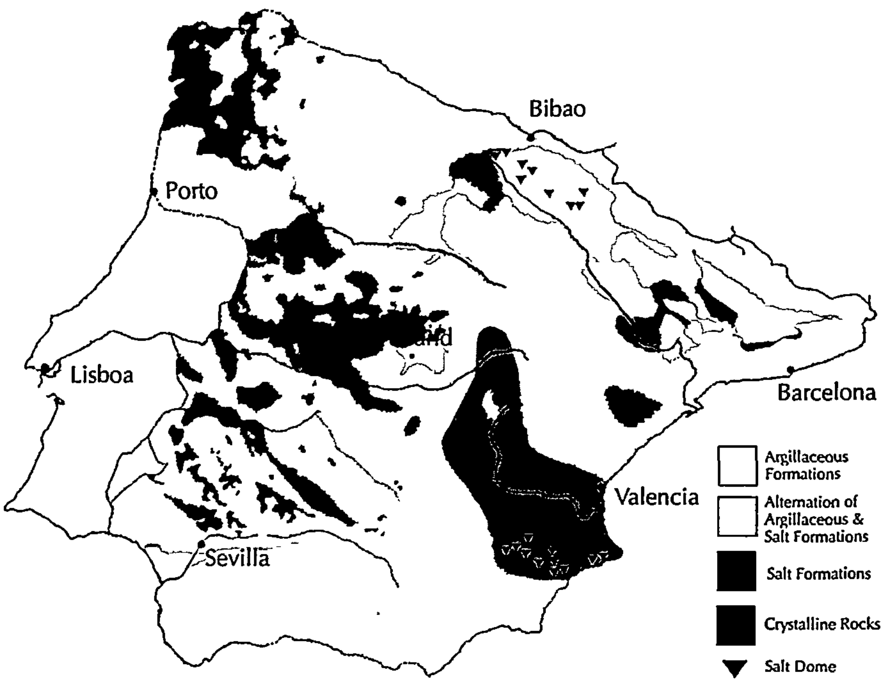

Figure 21.3. European catalogue of geological formations in Spain having favourable characteristics for disposal of HLW.

All this makes Spain one of the few countries, along with France, United States, Sweden, United Kingdom and Japan, to possess an overall capacity for management of low and intermediate level wastes produced by the country's nuclear power plants and some one thousand smaller installations (hospitals, industry, etc.) from a modern environmental point of view.

21.3 HIGH LEVEL WASTES

In Spain, high level wastes are understood to be the non-reprocessed spent fuel from nuclear reactors, the exception being the fuel from Vandellós I NPP, which has been sent to France for reprocessing.

In view of the overall open cycle strategy applied to this fuel in Spain, there are basically two types of high level wastes which will have to be managed: (a) the spent fuel generated by the country's light water reactor nuclear power plants, by far the larger volume, and (b) the vitrified wastes arising from the reprocessing in France of the Vandellós I spent fuel.

Before the definitive disposal of these wastes is accomplished, it is necessary for them to be kept for a period of interim storage in order for prolonged cool down and decay of their isotopic activity to occur.

According to the economic studies carried out, the most reasonable solution for interim storage of the spent fuel, while the nuclear power plants are in operation and taking into account the plant lifetime considered, will be for them to be stored on site at the plants where such spent fuel is generated. This storage will be accomplished either in the plant fuel pool or using dry storage techniques on site. Consequently, the date by which a centralized temporary storage facility for this fuel should be available will depend, fundamentally for the time being, on the time at which the first nuclear power plant dismantling process is undertaken, in other words, on the service lifetime considered for these installations.

The date on which the deep geological disposal facility enters operation will not, however, undergo any variation, regardless of whatever hypothesis is adopted regarding service lifetime.

21.3.1 Search for a Site for Construction of Facilities.

The process of designating a site for disposal of HLW started in 1986 and continues today. The geological media contemplated are granites, salts and clays. A National Inventory of Favourable Formations (IFA Project), has been developed during 1986-7 which confirms the Spanish contribution to the European Catalogue (Fig. 21.3), and a selection process known as the "High Level Regional Studies" (ERA Project) has

been drawn up. The site screening in the ERA project has been based on geological, hydrogeological, seismic, environmental and societal data.

The second phase of the process known as the "Study of Favourable High Level Areas" (AFA Project), which covered the period 1990-1995, identifies more than one thousand municipalities with potential capabilities to construct a HLW repository.

In the 4th General Radioactive Waste Plan (December 94) a project of law to arbitrate the procedures to designate the site of surface and underground installations is under consideration by the government. The bill will dictate the means of participation in the ultimate decision of State and other concerned institutions, as well as the general public.

21.3.2 Development of Basic Design for Deep Geological Disposal Facility.

Progress is being made in defining the conceptual design for future surface and underground installations. The aim of this project, which was initiated in July 1990 and is being performed by Spanish engineering firms, in collaboration with Swedish and German organizations, is to perform systems analyses, and to evaluate various detailed disposal concepts and alternatives.

This entire process serves as an important central activity of the R&D programme and is needed to design and construct the disposal facility, regardless of whatever geological medium is chosen.

The first important milestone was the development in 1992, of a preliminary conceptual design for salt and granite formations.

Once ENRESA analyzed the results obtained, a second three-year phase was addressed. Its aim was to get the disposal concept for the three formations ready by the end of 1995, supported by a preliminary safety assessment. Successive later phases are to be undertaken until the selection of the final disposal system project is made.

21.3.3 Acquisition of Technology and Training of teams Required for Characterization of Chosen Site and Construction of Disposal Facility.

Site characterization and disposal system design and verification require a scientific technological support system provided by associated R&D activities. Once human and technical equipment have been selected by previous R&D investigations, a further verification of different methodologies using field studies on different scales will be needed. These studies will provide the necessary information for a long-term performance assessment of the disposal system.

After the final candidate site is selected, the three above mentioned areas of work will be directed towards the same objective which can be summarized as follows:

1. Detailed site characterization by R&D developed and perfected techniques. These will include surface workings, drilling and an underground research laboratory.
2. The previous disposal system design will be adapted by preparing a detailed project to assist in its final construction.
3. The R&D Plan will be aimed at completing the remaining activities, particularly the safety assessment of the chosen site, including specific works to be performed on the site.

All the planned works are to be completed by year 2015 and the final disposal system construction is foreseen to start and finish during the decade of 2020.

21.4 DECOMMISSIONING OF INSTALLATIONS

From the technological and waste production point of view, the most significant aspect of this important management issue in Spain is decommissioning the country's nuclear power plants. In this respect the Vandellós I NPP is of particular importance at this moment in time, with decommissioning of other plants currently in operation constituting a longer term activity.

In spite of the importance of these plants, there are other installations, such as uranium mines, the Andújar uranium mill and the La Haba concentrates manufacturing facility in Badajoz, whose decommissioning will have to be addressed and which are currently in different phases of performance, as described below.
The spent fuel from the Argos and Arbi experimental reactors was transported to the United Kingdom in 1992 for storage and reprocessing; it is foreseen that the waste generated in the process will be returned to Spain. As regards the JEN-1 reactor, dismantling is currently being addressed by CIEMAT; this organization is carrying out a research and development programme in rela-

tion to this issue, with participation by Spanish and overseas institutions and financing by the EU. There is also an agreement with the UKAEA for storage and eventual reprocessing of the fuel from this reactor, which was transported to the United Kingdom in 1992.

With regard to the issue of dismantling, special mention should be made of the particularly important question of the declassification of materials as radioactive wastes, since this implies total or partial exemption from the control systems applied to such materials, thus allowing them to be managed by means of methods similar to those used for conventional wastes. Work is currently advancing rapidly at national and international levels with a view to completing detailed development of specific criteria and methodologies for the application of such exemption practices in Spain.

21.4.1 Closed Uranium Mines

As was pointed out in the Third GRWP, ENRESA has carried out a study of the conditions at closed-down mining facilities belonging to the then Nuclear Energy Board, now CIEMAT. As a result of this study, the decision was taken to perform projects at certain of these facilities with a view to restoring the terrain altered by the operations, eliminating rubble tips, refilling quarries, shafts, etc. and in general carrying out whatever corrective measures might be required for the sites to be integrated into their natural surroundings.

The so-called Action Plan for the restoration of closed uranium mines was finalized at the beginning of 1994. Following the corresponding evaluations, work will begin on detailed development of the project at the mines considered to be of interest, as a preliminary step to performance of the field work. According to current forecasts, these tasks will imply specific actions at 2-3 mines; such actions are to be initiated in the last quarter of 1994 and completed during 1995, including the corresponding control procedures.

21.4.2 Andújar Uranium Mill

Authorization for the decommissioning of the Andújar Uranium facility was awarded by Ministerial Order on 1st February 1991, and performance of planned activities began immediately. For performance of the Decommissioning Plan, ENRESA analyzed the technology used in other countries for this type of project and defined the activities to be performed using the USA UMTRAP (Uranium Mill Tailing Remedial Actions Project) programme, which covers 24 installations of this type, as a point of reference.

The proposal includes dismantling of the installations, demolition of buildings and incorporation of the resulting rubble into the mass of tailings, and stabilization of the whole through reduction of banks and construction of a cover providing protection against erosion, diffusion of radon, and infiltration of water.

The design criteria and objectives contemplated relate to the control of dispersion, long-term radiological protection, durability, the cleaning of contaminated soils, the control of radon diffusion, protection for groundwaters and the minimization of long-term maintenance.

The works were completed in May 1994, in accordance with the existing schedule; what now remains to be accomplished is establishment of the corresponding monitoring programme, to be performed over the next 10 years and prior to declaration of the decommissioning of the facility.

21.4.3 Decommissioning of La Haba

The main activities to be accomplished at LaHaba include restoration of the terrain affected by the mining works, by transferring the rubble tips to the mine openings and subsequent replanting, and closure of the Lobo-G plant and the associated tailings dike. This task consists of dismantling the installations and the stabilization and covering of the dike.

21.4.4 Decommissioning of Nuclear Power Plants

Following the final removal from service of the Vandellós I NPP, it was necessary to adjust development of the strategies and technical activities contemplated for this area of management in the first waste plans. These emphasized the fact that this particular problem was a long-term issue and contemplated initiation of total decommissioning (Level 3) of all the Spanish nuclear power plants five years after final shutdown of the reactor.

Based on experience acquired in other countries, especially in France where the technology originated, ENRESA has performed studies aimed at defining the most feasible strategy from a technical and economic point of view, taking into account the specific circumstances of Vandellós I NPP.

The following three possible alternatives were consid-

ered:

1. Maintenance of the plant in the final definitive shutdown for an indefinite period of time. (Level 1 dismantling).

2. Dismantling the conventional parts of the plant and active parts other than the reactor and its internals (Level 2 dismantling).

3. Total dismantling, leaving the site in conditions allowing it to be used without any type of restriction (Level 3 dismantling).

To date, no Level 3 dismantling process has been carried out at any commercial plant. Consequently, this alternative may be ruled out for Vandellós I in the short term, owing to the technological, methodological and licensing risks involved. After following a process of study and assessment of various parameters (technological, radiological impact, regulatory, economic, logistics and the volume of wastes to be managed), it was considered that the most feasible strategy for decommissioning the Vandellós I NPP would be immediate dismantling in accordance with alternative 2, followed by a period of waiting, estimated to last 25 years, for completion of total dismantling the remaining parts of the plant in accordance with alternative 3.

The alternative chosen not only represents the most feasible approach from the point of view of both performance and impact on general waste management, but is also backed by French experience in relation to the dismantling of the two units of the Saint Laurent des Eaux plant (SLA 1 and 2). This led to the decision to undertake a Level 2 dismantling followed by total dismantling (Level 3) following a suitable waiting period, estimated at between 25 and 30 years.

At Vandellós I NPP, the programme of activities to be performed prior to dismantling is being coordinated with the fuel removal activities carried out by HIFRENSA, such that during 1995, the detailed engineering project and licensing process will have been completed. In this respect, ENRESA submitted a dismantling and decommissioning project to the Ministry of Industry and Energy in May 1994, for its approval. This project contemplates partial dismantling of the facility (Level 2), which will make it possible to determine the most suitable period of waiting prior to initiation of the total dismantling process. It is estimated that four years will be required for completion of the partial dismantling process contemplated in this project, as from the date of initiation.

As regards the other light water reactor nuclear plants currently in operation, consideration is currently still being given, from the point of view of calculation and planning, to the alternative of undertaking complete dismantling (Level 3). This process is to be initiated between four and eight years after final shutdown of the plant.

REFERENCES

Ministerio de Industria, Comercio y Turismo, Third General Radioactive Waste Plan, July 1991.

ENRESA, Second Research and Development Plan 1991-1995, Vol. I, Approach and Summary 1992.

ENRESA, Segundo Plan de Investigación y Desarrollo 1991-1995 Vol. II, Desarrollo del Plan de I&D, 1992.

CEC, El Berrocal Project, Characterization and Validation of Natural Radionuclide Migration Processes under Real Conditions on a Fissured Granitic Environment, Reports 1 and 2, 1991.

Codelli, N. and G. Cottone, Actas de la Jornada de Información PAGIS, CEC. EUR. 12676, 1989.

Ministerio De Industria y Energia, Cuarto Plan General de Residuos Radiactivos, Diciembre, 1994.

CHAPTER 22

PROGRESS TOWARDS A SWEDISH REPOSITORY FOR SPENT FUEL

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Abstract. Nuclear power is producing electricity for the benefit of society but is also leaving radioactive residues behind. It is our responsibility to handle these residues in a safe and proper manner. The development of a system for handling spent fuel from nuclear power plants has proceeded in steps. The same is true for the actual construction of facilities and will continue to be the case for the final repository, for spent fuel, and other types of long-lived wastes. The primary objective in constructing the repository will be to isolate and contain the radioactive waste. In case the isolation should fail for some reason, the multibarrier system shall retain and retard the radionuclides that might get in contact with ground water. A repository is now planned to be built in two steps, where the first step would include deposition of about 400 canisters with spent fuel. This first step should be finished about 20 years from now and be followed by an extensive evaluation of the results from not only this particular step but also from the development of alternative routes before deciding on how to proceed. A special facility to encapsulate the spent fuel is also required. Such an encapsulation plant is proposed to be constructed as an extension of the existing interim storage CLAB. Finding a site for the repository is a critical issue in the implementation of any repository. The siting process was started a few years ago and made some progress but is by no means yet completed. It will go on at least into the early part of the next decade. When the present nuclear power plants are about to be shutdown, there should also be facilities in place to take permanent care of the long-lived radioactive residues. Progress in siting will be a prerequisite to success in our responsibility to make progress toward a safe permanent solution of the waste issue.

22.1 INTRODUCTION

Sweden has twelve nuclear power reactors with 10,000 MWe electric capacity. These reactors are producing some 70 TWh per year and from that production arises about 250 tonnes of spent nuclear fuel. Up to 2010, it is estimated that the cumulative amount of fuel will be some 8,000 tonnes (uranium weight). The responsibility for taking care of this fuel rests with the owners of the nuclear power plants in accordance with the principle that "the producer is responsible". The owners have given the mission to the Swedish Nuclear Fuel and Waste Management Co. (SKB) to execute this responsibility for them.

The progress of the Swedish waste management programme is closely monitored by the government and by the pertinent authorities in Sweden. According to the law, SKB has to submit a programme for research, development and all other necessary measures in order to be able to handle and finally dispose of the spent fuel and other radioactive wastes arising from the operation of the nuclear power plants. Such a programme must be submitted at three year intervals and is then scrutinized by the authorities and a broad set of reviewers before the government decides on the programme. In the past, SKB has submitted several such programmes (SKB 1986, SKB 1989, SKB 1992, SKB 1995a). The latest programme is still under review by the government.

As the implementation of the programme proceeds and reaches the siting and construction phases, the same authorities will be responsible for evaluation of the safety and giving stepwise permission to proceed towards completion of the system.

22.2 STEPWISE DEVELOPMENT

The development of a system for handling the spent fuel has proceeded in steps. The first steps were taken during the 1970s when a parliamentary committee proposed the construction of a central interim storage facility for spent nuclear fuel and to initiate research for disposal of high level radioactive wastes deep in the crystalline bedrock in Sweden (Aka 1976). The research took a great step forward with the KBS-project which was

established in late 1976 in response to a new law. This was the stipulation law, which required that the nuclear power plant owners work out a plan for handling and final disposal of high level waste from spent fuel before the last six reactors of the power programme were allowed to start operation. These plans became known under their abbreviated names KBS-1, KBS-2 and KBS-3 (SKBF 1977, SKBF 1978, SKBF 1983, respectively).

KBS-1 addressed the disposal of vitrified waste from reprocessing, whereas KBS-2 and KBS-3 described the disposal of unreprocessed spent nuclear fuel. All three studies included a period of interim storage before the final disposal; a period of 30 - 40 years was considered appropriate in order to decrease the thermal load on the repository. After about 40 years, the residual decay heat in the fuel will have decreased by about 90 % in comparison with one year old spent fuel. A further equally important consideration in favour of interim storage is the creation of flexibility and buffer capacity in the management system for the spent fuel.

In the 1980s, the Swedish programme was focused on final disposal of the spent fuel without reprocessing. There were several reasons. The nuclear power programme was limited to twelve reactors. The prices for natural uranium and enrichment services stabilized or even decreased, whereas the prices for reprocessing services and Pu-recycle continued to increase. Thus, direct disposal got an economic advantage. The concern for nuclear proliferation, as a consequence of increased handling of plutonium, created a political resistance against reprocessing and Pu-recycle. The development of the breeder slowed down.

Several alternative methods for final disposal of spent fuel in the Swedish crystalline bedrock were studied and evaluated as a part of the broad R&D-programme, which continued based on the KBS-reports. In 1992, SKB concluded that a method similar to the one described in the KBS-3-report would be best suited for use, at least for the first step, in implementation of a deep repository. In general this conclusion was accepted by the authorities although some reviewers considered the choice to be premature.

22.3 STEPWISE CONSTRUCTION

An interim storage facility, CLAB, was constructed in the early 1980s at the Oskarshamn nuclear power plant. It was put in operation in 1985. CLAB has at present a capacity of 5,000 tonnes and can easily be expanded to meet future demands. About 2,300 tonnes of spent fuel was in storage at the beginning of 1996.

Following the evaluation of alternative methods in 1992, SKB decided to start the implementation process for the first steps in building a deep repository for spent nuclear fuel. This comprises the siting and basic design of an encapsulation plant for the fuel and of a deep repository. The first stage of the repository is planned for about 400 canisters or 800 tonnes of fuel and should start operation in 2008 (Fig. 22.1). The encapsulation plant is planned for filling one canister per day.

After the first stage has been completed, a thorough evaluation will be made both of the experience gained from the first stage and from development of other, alternative treatment and disposal methods that have been studied and/or applied in Sweden or elsewhere. The opportunity to change the route or even to retrieve the canisters that have been deposited will be available. This strategy thus provides an approach where irrevocable decisions must not be made until all aspects of the repository implementation have been fully demonstrated.

The implementation of the first stage will also proceed in steps with siting, basic design and supplementary R&D during the 1990s, with construction during the main part of the next decade and the first stage operation and evaluation during the 2010s. The stepwise approach is thus a key element in the planning and implementation for a repository.

22.4 SAFETY APPROACH FOR A DEEP REPOSITORY

The safety of a deep repository is dependent on the radiotoxicity and on the accessibility of the waste. Both these properties are time functions. Thus, the safety of the repository has to be assessed as a function of time. There will always be a fundamental uncertainty in the prediction of the future behavior of any system and the uncertainty may increase with time. The Swedish Radiation Protection Institute has discussed the influence of the time horizon on the safety assessment and radiation protection (SSI 1995). They conclude that:

- Particularly great attention should be given to describing protection for the period up to closure of the repository and during the first thousand years thereafter, with a special focus on nearby residents.
- The individual dose up to the next ice age, i.e. up to about 10,000 years, should be reported as a best esti-

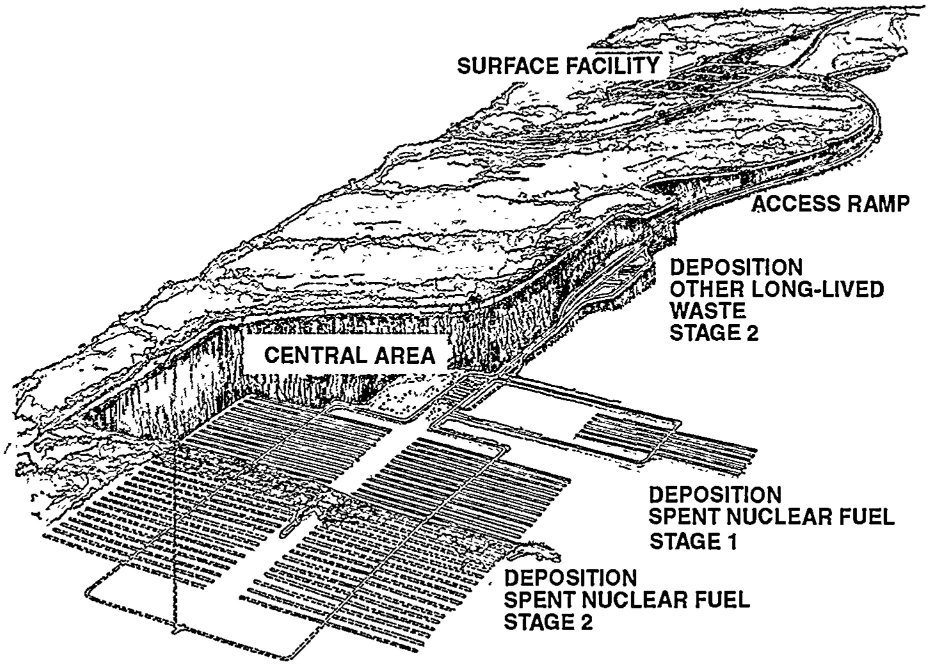

Figure 22.1. Deep repository showing schematic layout of stage 1 and stage 2.

mate with an estimated margin of error. Environmental protection should be described for the same period of time.

- For the period from the next ice age onward, qualitative assessments should be made of what might happen with the repository, including scenarios taking into account the risk of increased releases.

The safety of a repository is achieved by the application of three principles:

- Level 1 - Isolation. Isolation enables the radionuclides to decay without coming into contact with man and his environment.
- Level 2 - Retardation and retention. If the isolation is broken, the quantity of radionuclides that can be leached and reach the biosphere is limited by:
  - very slow dissolution of the spent fuel;
  - sorption and very slow transport of radionuclides in the near field; and
  - sorption and slow transport of radionuclides in the bedrock.
- Level 3 - Recipient conditions. The transport pathways along which any released radionuclides can reach man are controlled to a great extent by the conditions where the deep groundwater first reaches the biosphere (dilution, water use, land use and other exploitation of natural resources). A favourable recipient means that the radiation dose to man and the environment is limited. The recipient and the transport pathways are, however, influenced by natural changes in the biosphere.

The safety functions at levels 1 and 2 are the most important and the next-most important. They are achieved by means of requirements on the properties and performance of both engineered and natural barriers and on the design of the deep repository. Within the frames otherwise defined, a good safety function at level 3 is also striven for by a suitable placement and configuration of the deep repository.

22.5 DEEP REPOSITORY

The isolation of the spent nuclear fuel from the biosphere is achieved by encapsulating the fuel in a canister with good mechanical strength and very longlived resistance against corrosion. The conceptual design adopted is a copper canister with a steel insert. The copper provides a very good corrosion resistance in the geochemical environment foreseen in a deep repository in Sweden. The steel insert provides the mechanical protection needed. Each canister contains about 2 tonnes of spent fuel. The canisters are placed in deposition holes drilled from the floors of tunnels at about 500 m depth in the crystalline, granitic bedrock (Fig. 22.2). Each can-

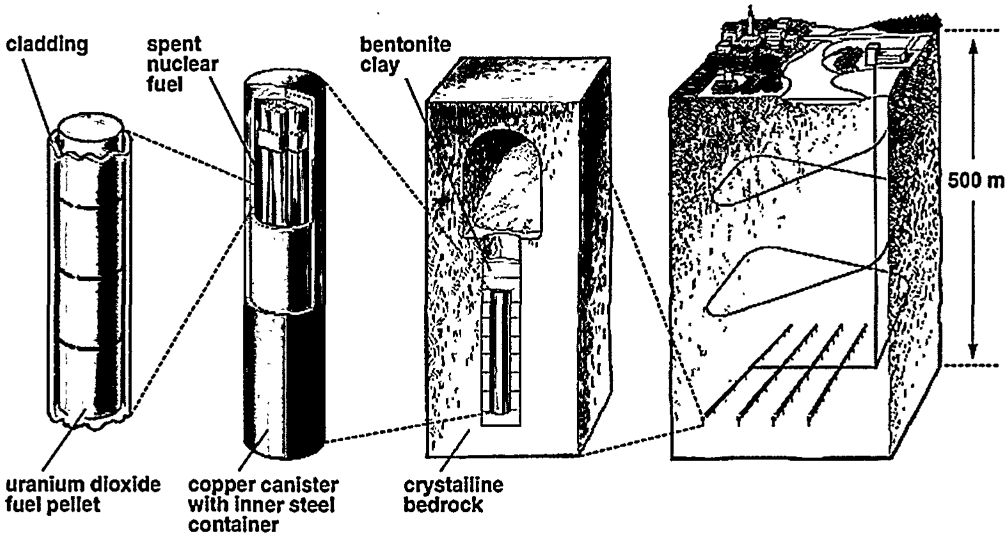

Figure 22.2. Deep repository in accordance with the KBS-3 concept.

ister is surrounded by blocks of compressed bentonite. When the bentonite absorbs water from the surrounding bedrock it will exert an intense swelling pressure and completely fill all void space in the near vicinity of the canister with bentonite clay. The clay barrier will contribute to the isolation by preventing or delaying dissolved corrosive species that may exist in minor amounts in the ground water to reach the canister. The clay will also provide some mechanical protection for the canister. The tunnels will eventually be backfilled by some material like a mixture of crushed rock and bentonite.

A repository for all spent fuel from the present Swedish programme should have a capacity of about 8,000 tonnes or 4,000 canisters. In addition it should be able to deposit other types of long lived wastes at the same site. This means that the underground facilities will need some 30 - 40 km of tunnels and cover an area of about one km^2. The surface facilities at the repository site will require an area of about 0.2 km^2.

SKB has started the planning work for a deep repository by preparing plant descriptions. These provide examples of possible ways to design the repository with its buildings, land areas, rock caverns, tunnels and shafts. They also contain requirements on, and principles for, the various functions of the repository. To a large extent, the construction and operation of the facility can be based upon experience and proven technology from nuclear installations and underground rock facilities. Special attention is given to: the impact of the excavation work on surrounding rock, methods for preparation and installation of the buffer bentonite blocks, and the technology for backfilling and sealing.

22.6 ENCAPSULATION OF SPENT NUCLEAR FUEL

Another necessary facility where the planning work has started is a plant for encapsulating the spent nuclear fuel. The intention is to expand the existing interim storage facility, CLAB, at Oskarshamn with such a plant. The plant will take fuel assemblies from the underground storage pools, dry them, transfer them to canisters made of copper with a steel insert, change the atmosphere to inert gas, put lids on the canisters and seal the lids by electron beam welding. The quality of the filled and sealed canisters will be inspected by non-destructive examination (NDE) methods - ultrasonic and radiographic - before shipping to the repository.

Each canister will contain 12 BWR fuel assemblies or 4 PWR assemblies. The copper thickness will be about 50 mm and the steel thickness also about 50 mm (Fig. 22.3). The copper thickness shall be enough to prevent corrosion from penetrating the canister during the time when the spent fuel radiotoxicity substantially exceeds what one would find in a rich uranium ore. The combined thickness of steel and copper should be enough to prevent any significant radiolysis of water outside the canister after deposition in wet bentonite clay. The steel insert is designed to withstand the normal mechanical loads that will prevail on the canister in the repository such as hydrostatic pressure and the bentonite swelling

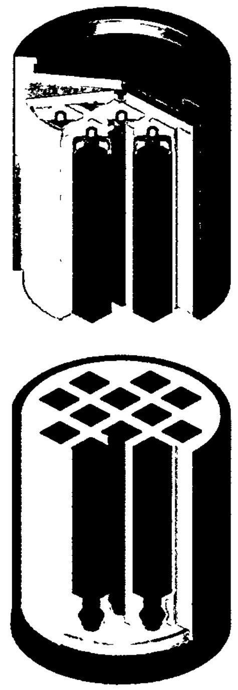

Figure 22.3. Copper canister with steel insert.

pressure. The total weight of a canister with fuel will be about 25 tonnes. In total some 4,000 canisters will be required for the spent fuel arising from the Swedish reactors up to 2010.

The design of the steel insert is still under evaluation. The present reference concept is a cast insert with thick steel walls between each fuel assembly. This gives a good mechanical stability as well as providing adequate protection against criticality in the unlikely case that the canister, at some unspecified future time, should be filled with water.

The fabrication of copper canisters of the size needed is by no means an industrially available technology. The seal welding technology has recently been demonstrated on a laboratory scale in work sponsored by SKB at The Welding Institute in UK. Full scale canisters have also been fabricated on the laboratory scale. In order to make key technology more mature, SKB has decided to create a laboratory for encapsulation technology at the former shipyard in Oskarshamn. This laboratory will be ready sometime during 1997 and will primarily be devoted to further development of the seal-welding process and the NDE-methods.

The design of the encapsulation plant is in progress. The main contractors for the design are BNFL for the key hot cell parts and ABB Atom for the other systems. The work is at present directed towards final specifications for the above mentioned laboratory as well as the development of an Environmental Impact Statement and a Preliminary Safety Report forming a basis for an application for a siting and construction permit. The application is planned for submission in early 1998.

22.7 REPOSITORY SITING

The most crucial part of the development of a deep repository is the siting process. SKB started geological site investigations at an early stage in the programme. Throughout the 1970s and 1980s, so called study sites were investigated at some 10 locations scattered from the southern to the northern part of the country. The investigations included measurements in boreholes as deep as 1000 m, as well as geophysical measurements from the surface. The main conclusion from these site investigations was that there are many places in the Swedish bedrock that provide conditions which are suitable for siting a repository at a depth of about 500 m. This implies that the safety requirements for a site can be met at many places and that factors other than safety can also be decisive in siting.

One such factor of importance is the acceptance by authorities and local residents. After the presentation of the RD&D-programme 92, SKB got in contact with several municipalities in various parts of Sweden. These contacts led to the proposal to start so called feasibility studies for the municipality in order to more clearly define the possibilities and consequences of siting a repository in the municipality. The intent was that the municipality as well as SKB should get a comprehensive set of documentation on which to base any decision of further, more detailed studies. A prerequesite was that the feasibility study should be based on existing geological data; no drillings and new measurements would be included. The discussions resulted in feasibility studies in Storuman and Malå in northern Sweden, Lappland (Fig.22.4).

The study for Storuman was published in February of 1995, (SKB 1995b) and that for Malå in March 1996 (SKB 1996). In both municipalities, two fairly large areas - 50-100 km^2 - were identified to be of interest for further investigations as potential host formations for a repository. However, in September of 1995, Storuman had a referendum on whether to permit further investigations for a repository site in the municipality, and the outcome was more than 70 % no-votes. This means that

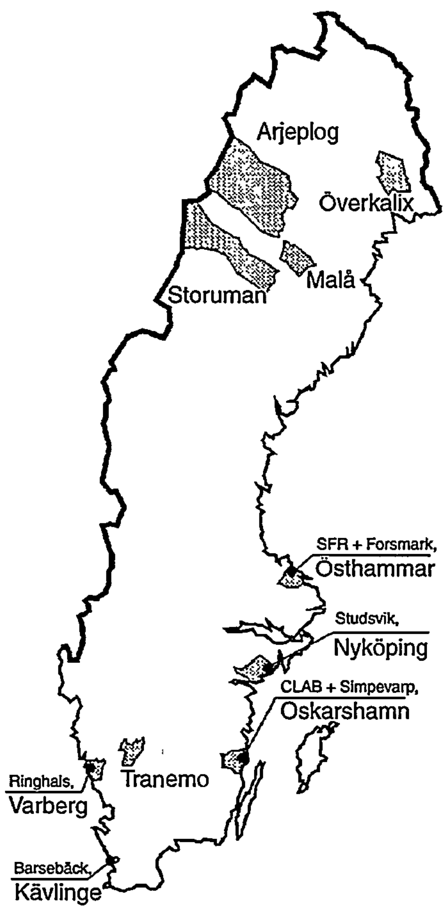

Figure 22.4. Location of some municipalities in Sweden.

SKB's work at Storuman has ceased. The study in Malå is now being reviewed by the municipality with the help of an independent group of experts.

SKB also performed a general study of five municipalities: Östhammar, Nyköping, Oskarshamn, Kävlinge and Varberg, where nuclear facilities are already located. The conclusion from this study was that there was a clear interest to continue with specific feasibility studies for the three first mentioned municipalities. There was also some interest for a study in Varberg, although the existing geological data for that area are rather meager and must be supplemented. As a result, feasibility studies are now going on in Östhammar and Nyköping whereas the proposal is still being evaluated by Oskarshamn. Varberg has declined a feasibility study.

SKB has also had fairly extensive discussions with three other municipalities: Överkalix, Arjeplog and Tranemo. These have finally resulted in negative answers from the municipalities mainly due to strong local opposition. In a few other cases, the negative answers were given at an early stage.

In 1995 SKB published a General Study '95 (SKB 1995c) which provides background material and gives an overview of some important siting factors on a scale covering the whole of Sweden. One main conclusion from this study is a confirmation of previous observations that it is not feasible to identify interesting areas on such a coarse scale. It is, however, possible to identify some major areas like Gotland, Skåne and the mountain range at the border with Norway where the geological and other conditions are such that it is of less interest to look for a site.

Based on the General Study '95, SKB will continue to study some parts of Sweden on a regional scale in order to identify areas of possible interest. The ambition of SKB is to make feasibility studies of some 5 - 10 municipalities as a basis for selecting at least two sites for investigation including extensive drilling as well as geophysical, geohydrological and geochemical measurements. These investigations will give the necessary data base to seek permits to enable detailed site characterization including construction of tunnels and/or shafts down to repository depths. The government has concluded that such a detailed investigation means that part of the construction of the future repository facility will actually start. This means that a siting permit according to the Act on Management of Natural Resources must also be accompanied by a permit according to the Act on Nuclear Activities.

Critics of SKB claim that the siting process followed by SKB is non-scientific and unsystematic; some even claim that it violates the Swedish environmental protection act. The siting process was, however, outlined in detail in the supplement to RD&D-programme 92, which after a rather extensive review was approved by the authorities and by the government. The fourth paragraph of the environmental protection act says: *For an environmentally hazardous activitity, one shall select such a site that the purpose can be achieved at a minimum of impact and inconvenience and without unreasonable costs.* The process followed by SKB for finding a site for the deep repository is fully in line with this paragraph; the strategy is to find a site where the purpose to construct a safe deep repository can be achieved and then at first hand look at sites where there is some local interest to consider receiving the facility. Thus, one should minimize the inconvenience to areas where there

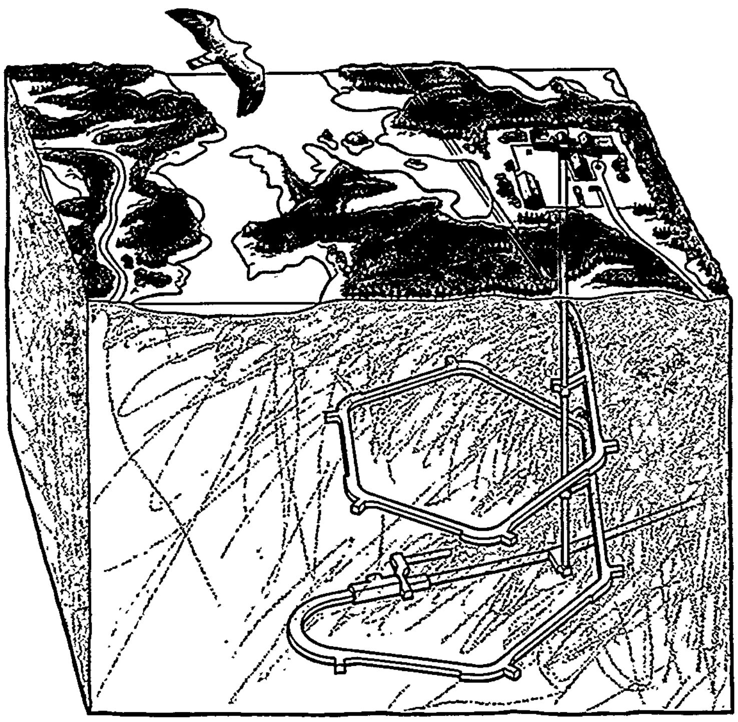

Figure 22.5. General layout of the Äspö hard rock laboratory.

is no such interest. At the same time, the authorities want to have a reasonably broad base for judging that there are no other sites which would obviously be better choices than the one finally selected for an application to construct the repository.

22.8 ÄSPÖ HARD ROCK LABORATORY

In order to prepare for the siting and construction of a deep repository, SKB has built the Äspö Hard Rock Laboratory. The planning of this facility started some 10 years ago in 1986. The work at the laboratory has proceeded in three stages: planning and site investigations, construction, and operations. The first two stages have now been completed and the operational stage has started. A basic objective in the planning of the laboratory was to create a facility for research and development in a realistic and unperturbed environment at a depth planned for the future repository.

The Äspö HRL is designed to meet the requirements on R&D. The underground construction starts with a tunnel from the site of the Oskarshamn nuclear power plant heading north down to about 220 m depth under the island of Äspö. The tunnel then goes down in a spiral with a radius of some 150 m down to 450 m depth. The total length of the tunnel is about 3,600 m. The last 400 m were excavated by a Tunnel Boring Machine (TBM) as opposed to the first part that was drilled and blasted. The cross section of the tunnel is about 25 m^2 (Fig. 22.5).

Overall objectives for the research conducted at Äspö are to:

- increase the scientific understanding of the safety features and function of a repository;
- develop and test technology that will simplify the disposal concept and decrease costs while retaining quality and safety; and
- demonstrate the technology that will be used for disposal of spent fuel and other long-lived wastes.

When the Äspö-project started in the late 1980s, the work program was formulated in several stages. The goal of the first stage was to verify that investigations on the surface and through boreholes from the surface will provide enough data on the important properties related to safety of the bedrock at the repository level. The comprehensive site investigation programme, conducted before the start of construction, was the basis for predictions on geological, geohydrological and geochemi-

cal data and behaviour of the rock at depth. These predictions were then compared with observations and measurements performed during the tunnel construction stage. A general conclusion concerning the first stage goal is that the methods available for site investigations are well suited to obtain data and information on the bedrock at a given site. These data can be used to select the proper volume of rock needed for a repository and to make the safety assessment needed for obtaining a permit to construct tunnels and/or shafts for starting detailed site characterization.

The goal of the second stage was to develop the methodology for such detailed site characterization. During the construction of the laboratory, considerable experience has been gained in how the detailed studies of the host rock can be conducted, and a good basis has been laid for the actual work to be made at a repository site in the future.

The other stages were related to improved understanding of the natural barrier - the bedrock - and to a demonstration of the technology to be used in the repository. Some of the tasks addressing these goals have already been performed, whereas others are part of the ongoing experimental and investigation programme at Äspö HRL. A series of Tracer Retention Understanding Experiments (TRUE) is aimed at increasing the knowledge on the capacity of the bedrock to retain and retard the transport of radionuclides in fractured rock. Experimental studies are being carried out to determine how and at what rate the oxygen, present in the repository at closure, is consumed by reactions with the rock. A special borehole laboratory - CHEMLAB - has been developed. It permits chemical experiments to be conducted under repository-like conditions with respect to groundwater composition and pressure. Such experiments will give data to verify models *in-situ* and check the data used to assess the dissolution of radionuclides, the fuel corrosion, the sorption on rock surfaces, the diffusion in buffer materials etc. A full-scale (in-active) prototype repository is planned to be built at Äspö. It will provide the opportunity to simulate all stages in the deposition sequence in a realistic environment. It will also be possible to observe the simulated repository several years in advance of depositing the first (active) canister in the final deep repository.

The work at Äspö has attracted a large international interest, and at present eight foreign organizations from seven countries are participating in the Äspö programme. Several of the experiments are conducted with very active participation by scientists from the foreign participants. This provides a good mechanism for further strengthening the scientific quality of the work and gives all participants access to a broad international forum for discussing the planning, execution, evaluation and interpretation of the experiments.

22.9 CONCLUDING REMARKS

The implementation of a deep repository for spent nuclear fuel is a very lengthy and tedious process in today's society. Considering the time scale intended for the isolation of spent fuel, it is of course still a short time period. In comparison with many other more common projects, however, it is unusual and demanding for all those involved. It extends over several decades and must proceed in steps, where each step requires careful planning. This stepwise progress is a key element. It must be stressed that no step should really be irrevocable; it should always be possible to step back and reconsider and even take another route.

The siting of a deep repository in Sweden is now in progress. To complete this process, efforts will be needed not only from the responsible implementing organization but from all parties involved such as the safety authorities, affected local authorities and political bodies. Building of confidence and trust is a key element in the process. Nuclear power is producing electricity for the benefit of society, but it is also leaving radioactive residues behind. It is our responsibility to handle these residues in a safe and proper manner. When the present nuclear power plants are about to be shut down, there should also be facilities in place to take permanent care of the long-lived radioactive residues. This is the responsibility of our generation which has benefited from the electricity produced. It will be up to the following generations to decide on how to use, extend or change the system we have provided. In this way we can take care of our responsibility without depriving future generations of their possibilities to take their own actions. Considering the existence of long-lived radioactive wastes, a deep repository for such residues will, however, be an asset for society.

REFERENCESR

Aka 1976, Spent Fuel and Radioactive Waste, Report of the Aka Committee, SOU 1976:30 Part I, SOU 1976:31 Part II, SOU 1976:41 Appendix, 1976.

SKBF 1977, Handling of Spent Nuclear Fuel and Final

Storage of Vitrified High Level Reprocessing Waste, Parts I-V, Projekt Kärnbränslesäkerhet, Stockholm, November 1976.

SKBF 1978, Handling and Final Storage of Unreprocessed Nuclear Fuel, Parts I-II, Projekt Kärnbränslesäkerhet, Stockholm, September 1978.

SKBF 1983, Final Storage of Spent Nuclear Fuel - KBS-3, Parts I-IV, Stockholm, May 1983.

SKB 1986, R&D-Programme 86, Parts I-III, Stockholm, September 1986.

SKB 1989, R&D-Programme 89, Parts I-II, Stockholm, September 1989.

SKB 1992, RD&D-Programme 92, Main report plus 3 background reports, Stockholm, September 1992.

SKB 1995a, RD&D-Programme 95, Stockholm, September 1995.

SKB 1995b, Feasibility study for siting of a deep repository within the Storuman municipality, SKB Technical report 95-08, January 1995.

SKB 1995c, General Siting Study 95, Siting of a deep repository for spent nuclear fuel, SKB Technical report 95-34, October 1995.

SKB 1996, Feasibility study for siting of a deep repository within the Malå municipality, SKB Technical report 96-xx, March 1996, (English version in print).

SSI 1995, Statens Strålskyddsinstitut skyddskriterier för omhändertagande av använt kärnbränsle, (SSI protection criteria for handling of spent nuclear fuel), SSI-report 95-02 in Swedish, 1995.

CHAPTER 23

HIGH LEVEL RADIOACTIVE WASTE MANAGEMENT IN SWITZERLAND: BACKGROUND AND STATUS 1995

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23.1 BACKGROUND

Switzerland is a small country, with limited natural resources (other than hydro power) and must import about 80 % of its primary energy needs (predominantly petroleum products). Electricity covers about 20 % of energy demand; about 40 % of the electrical energy is supplied from nuclear plants, with almost all the rest being generated by hydropower.

Nuclear power production is the main source of Swiss radioactive wastes, although wastes arise also in medicine, industry and research. Switzerland currently has 5 nuclear power plants (pressurized water reactors and boiling water reactors) with a total capacity of 3 GW(e). The spent fuel, containing most of the waste radionuclides produced by fission, may be prepared for direct disposal or reprocessed to recover useable uranium and plutonium, with the resulting wastes being immobilized in glass blocks. To date, Swiss disposal planning has focused on waste returned from foreign reprocessing plants but, currently, the preferred strategy of the utilities is to keep open both options (reprocessing or direct disposal). The prime reason for originally choosing the reprocessing route was to optimize use of resources; the current arguments against reprocessing are primarily economic.

Nuclear power and its future role in the nation's energy mix is a controversial issue. The initial widespread acceptance has been replaced to a significant extent by uncertainty or even opposition; this led in 1990 to the adoption by popular referendum of legislation placing a 10 year moratorium on expansion of nuclear power. The waste disposal issue, as will be detailed in the following section, plays a prominent part in the debate on nuclear energy. There is a strong incentive for those responsible for waste management to ensure that continuing progress is made towards development and implementation of an integrated waste management strategy.

23.2 LEGAL, REGULATORY AND ADMINISTRATIVE ISSUES

The Atomic Law of 1959 clearly placed the responsibility for nuclear waste disposal with the producer of the waste. The Ruling of 1978 further stipulates that *"The general license for nuclear reactors will be granted only when the permanent, safe management and final disposal of radioactive waste is guaranteed."* This Ruling was extended by the Government to existing reactor operating licenses and this led to preparation of a special Project Gewähr (PG85) which, as described below, was submitted to the Government for review in 1985[1].

The safety conditions which the final repositories must satisfy are defined in the Guideline R-21 (1980, revised 1993) of the Nuclear Regulatory Authorities. Three protection objectives are defined:

- The repositories must ensure the safety of human beings and the environment from any harmful effects of ionizing radiation. Accordingly, the central point is Objective 1, which states: "Radionuclides which can be released into the biosphere from a sealed repository as a consequence of realistically assumable processes and events may not at any time lead to individual doses which exceed 10 mrem (0.1 mSv) per year." This objective is ambitious not only because of the absence of any time limit for demonstration of compliance but also because of the comparatively low levels of radiation doses permitted. For comparison, the natural radiation exposure of the Swiss population results in an average radiation dose of about 140 mrem per year, with a range of approximately 100 to 300 mrem per year.

- Objective 2 provides a quantitative risk level for judging the consequences of low-probability scenarios: "The individual radiological risk of fatality from a sealed repository subsequent upon unlikely processes and events not taken into consideration in Protection Objective 1 shall, at no time, exceed one in a million per year". The direct radiological risk of fatality from a scenario is thus multiplied by the estimated probability of the scenario occurring and this product should not exceed one in a million per year when summed over all such scenarios. For comparison, the dose limit of 0.1 mSv per year corresponds to a nominal risk of fatality of 5 in a million per year.

- Besides the safety aspects, the Guideline R-21 reflects the understanding that the responsibility for disposing of radioactive waste lies with today's beneficiaries of nuclear power and should not be passed on to future generations. This is expressed in Objective 3: "A repository must be designed in such a way that it can at any time be sealed within a few years. After a repository has been sealed, it must be possible to dispense with safety and surveillance measures." Once the repository has been sealed, it must thus be possible to "forget" the radioactive waste in the sense that it should not be necessary for future generations to concern themselves with it. There is thus no requirement for monitoring or retrievability of the waste.

In addition to the requirements formulated in these three protection objectives, non-nuclear regulations must also be taken into consideration; these include international law, district planning, environmental protection; and nature conservation.

As noted above, 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 the disposal and, if required, pre-emplacement conditioning of wastes. The responsibility for spent fuel reprocessing and transport, for the 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 L/ILW repository at a site selected by Nagra, the Genossenschaft für die Nukleare Entsorgung Wellenberg (GNW).

23.3 CHARACTERISTICS AND EVOLUTION OF THE SWISS NUCLEAR WASTE DISPOSAL PROGRAMME

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 principle of the multi-barrier concept, the requirements for packaging, engineered structures and geological isolation were derived for different types of waste. Two separate geological repositories are planned[2]; one for low-level radioactive wastes and shorter-lived intermediate-level wastes (L/ILW) and another for high-level wastes (HLW) and intermediate-level wastes containing higher concentrations of long-lived alpha- emitters ("TRU").

Highest priority at present is allocated to the L/ILW repository which is intended to be implemented in horizontally accessed rock caverns with some hundreds of metres of overburden. An extensive site-selection procedure resulted in 1993 in the nomination of Wellenberg in Central Switzerland as the preferred repository location. More detailed site-characterization work, to form the basis of the application for construction and operation permits, is now ongoing. The principal for the development of Wellenberg as a repository site has been accepted in public Referenda at the community level. It is also supported by the federal authorities. At the cantonal level, however, there has been opoposition, leading in 1995 to a popular vote which produced a narrow majority against the currently proposed project. Nevertheless, current planning assumes that, following appropriate amendments to the project, the L/ILW repository should be operational early next century.

For the present limited nuclear programme in Switzerland, operation of all plants for a 40 year lifetime will result in around 3000 t of spent fuel. If all of this were reprocessed abroad, the resulting volume of vitrified waste returned would only be around 500 m^3, although several thousand cubic metres of additional L/ILW could also be returned (depending on the contract with the reprocessor). It is planned to store HLW for at least 40 years in order 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 of vitrified HLW and for other reprocessing wastes will be constructed before the turn of the century by the ZWILAG organization, a daughter company of the utilities.

Implementation of a HLW repository will not take place in Switzerland before the year 2020, and there are sound economic arguments for delaying this date even further. Nevertheless, there is strong pressure from the public and the government - and a strong will on the part of the waste producers - to move the project ahead as quickly as possible, at least up to the level of demonstrating the feasibility of construction of a safe repository at a potential site.

Site selection is very much constrained by the small size of Switzerland and by its relatively active tectonic 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 (which is equivalent to 1-2 km in the million year timescale which is considered for HLW safety analysis!). Excluding alpine areas and other complex geological structures associated with the Jura mountains and the Rhine Graben leaves only a limited area in Northern Switzerland which 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.

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 safety of waste disposal as a condition of extending reactor operating licenses. These factors together led to designs which are certainly robust (or even overdesigned) and are not optimized in an economic sense. Accordingly, although estimates of absolute costs for the small size repository required are comparable to those from other countries, the costs per unit waste volume (or per kWh of nuclear electricity) are relatively high. Optimization of designs would, therefore, clearly be an important objective before moving to an implementation phase. The concept, illustrated in Figure 23.1 for the crystalline host rock option[3], has the following features:

1. extremely deep disposal (about 1 km below surface) in a carefully-constructed facility;
2. in-tunnel emplacement of HLW waste packages in a geologic medium whose principal roles are to limit water flows and to ensure favorable groundwater chemistry;
3. very massive engineered barriers; in addition to the vitrified waste in its steel fabrication canister, a 25 cm thick steel overpack is envisaged which is surrounded by more than one metre of highly compacted bentonite clay (Fig. 23.2); and
4. co-disposal of TRU in silos or in caverns in a separate part of the repository.

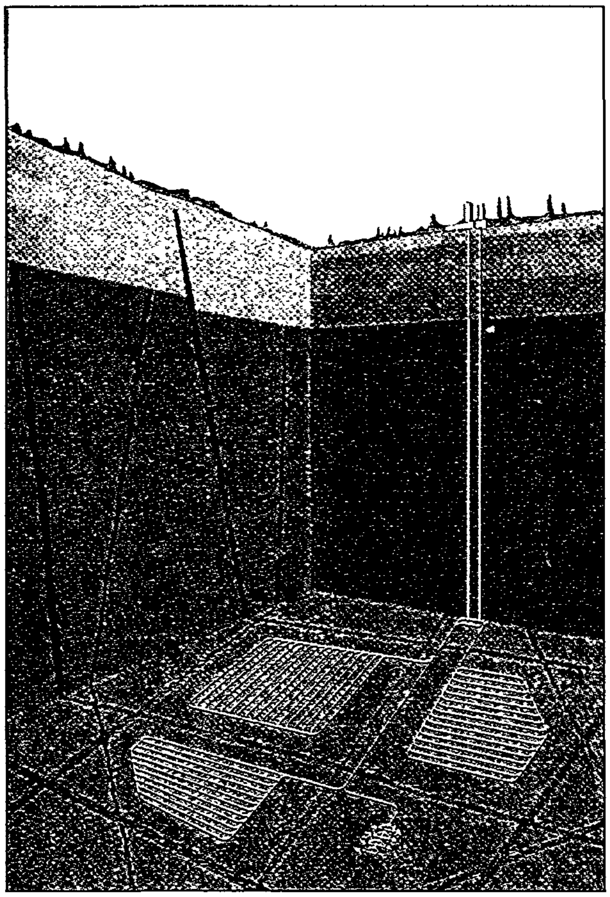

Figure 23.1. Sketch of possible repository layout. The case of 3 HLW emplacement panels on a single level and TRU silos in separate blocks of low-permeability basement between major (layout-determining) water-conducting faults.

Analysis of this concept in the Project Gewähr 1985 study (mentioned above), 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 having the properties postulated by Nagra. However, only limited data from isolated boreholes were available in 1985, and the 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

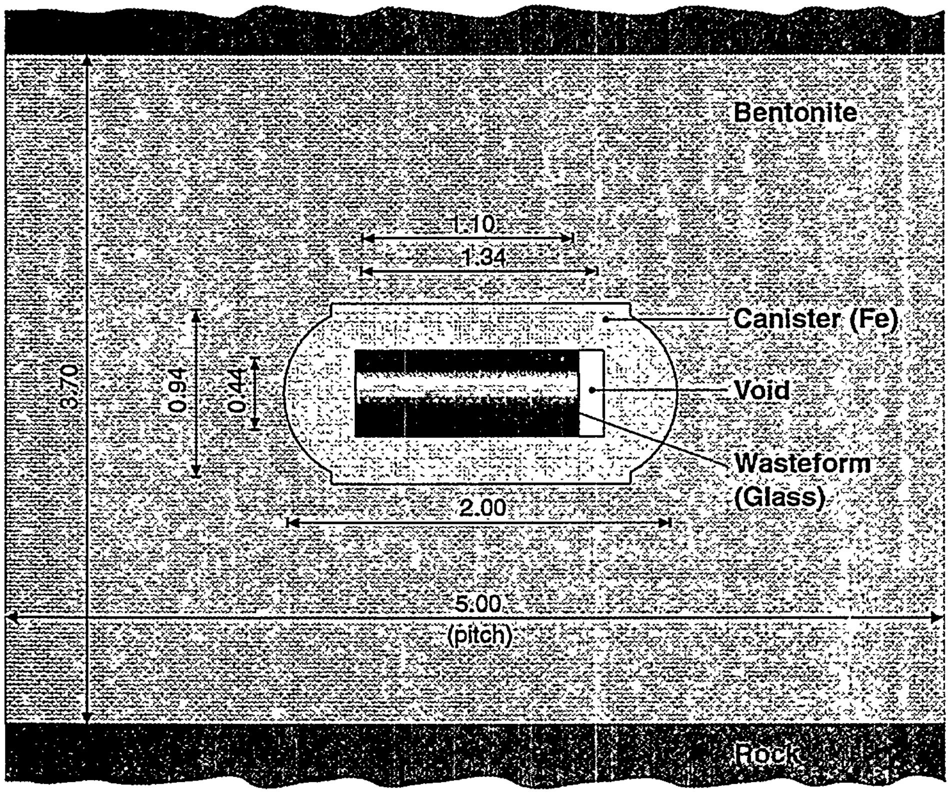

Figure 23.2. Waste emplacement geometry (dimensions in metres).

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, although it is understood that more evidence of siting feasibility would be required before any applications for new reactor licenses could be sought.

Since 1985, the regional investigation of the crystalline basement has been completed and documented[4]. The geological studies have clearly shown that the extent of the accessible crystalline basement is much less than originally thought due to the presence of a previously unknown, extensive Permocarboniferous trough which cuts the region. Only two restricted areas remain for selection of a possible site, each covering about 50 km^2. Nevertheless, it does seem feasible to find a suitable repository for the required low volume of waste and a strategy for site characterization has been developed. 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[5]. For Opalinus Clay, which was identified as the higher priority option for Nagra, a total potential siting area of some 100 km^2 has been identified. Programmes of site-specific studies are now running in parallel in the crystalline basement and the Opalinus Clay.

The next major milestone in the HLW programme will be the demonstration of repository siting feasibility (Project Entsorgungsnachweis) scheduled for 2000. This may include one or both of the potential siting areas studied.

23.4 Ongoing Geological Studies Associated with the HLW Programme

Geological characterization prior to repository construction is planned to progress in three phases. Phase I is a regional study of potential host rocks from the surface, involving seismic studies, investigation from deep boreholes, etc. This phase is followed by more detailed investigations of a potential siting area from the surface (Phase II), and then Phase III underground studies involving construction of an access shaft to the potential repository depth and an underground laboratory.

Phase I has been completed and documented for both the crystalline basement and Opalinus Clay[4,5]. For the former, the most relevant open question to be addressed in Phase II is the distribution of major shear zones within the crystalline basement. A detailed performance assessment (see below) has demonstrated that blocks of low permeability crystalline basement found in the area north of the North-Swiss Permocarboniferous Trough (Fig. 23.3) would be a very suitable host rock. Statistical analysis of major faults identified during Phase I (by geophysics, borehole mapping, mapping surface exposures in the neighboring Black Forest etc.), indicates that the probability of finding sufficiently large blocks for repository construction in this area is high. This statistical model will be tested by drilling a "star" of four inclined boreholes at a site to be chosen in Northern Switzerland (cf. Fig. 23.4). Location of "layout-determining" faults will be on the basis of core-logging, complemented by cross-hole hydro-testing and cross-hole seismic tomography. In addition, a surface campaign of seismics will be carried out. For the crystalline basement, this technique is somewhat limited as it identifies only faults which cause significant displacement in overlying sedimentary formations; no clear determination of structure within the basement is possible using this method.

In contrast, however, the Opalinus Clay formation is a clearly defined seismic reflector which can be well mapped by the planned 3D seismic survey. It is relatively homogeneous in composition and, in the area of interest, shows little evidence of tectonic disturbances. The planned borehole at a site near the village of Benken is aimed primarily to calibrate the seismic studies. Somewhat more problematic in the Opalinus Clay case, however, are the mechanical properties of this rock. Studies to date indicate that emplacement tunnels for HLW and caverns for TRU could be constructed at depths of up to ~800 m, but the extent of tunnel linings required needs to be established based on site-specific, rock mechanics data.

The demonstration of siting feasibility also requires geological input which cannot be obtained from these two sites in any easy way. Some generic data for crystalline and sedimentary rocks can be obtained from other national programmes, but underground laboratories in Switzerland provide a key testing ground for methodology development and use of destructive characterization methods.

Nagra's main underground test site is situated at Grimsel Pass in the Swiss Alps[6]. This facility is situated in granite/granodiorite below ~500 m of overburden. Although tectonically unsuitable for a HLW repository, this site has played an important rôle in the development of the technology required for site characterization. In the present phase of work (Phase IV, 1994-1996), HLW-relevant studies include testing of the limitations of current methods of seismic tomography, examination of the properties of the excavation-disturbed zone around tunnels through crystalline rock, validation of models of radionuclide migration through fractures, and demonstration of the methodology for emplacement of HLW packages. A final Phase V of work at Grimsel, lasting until 2002, is currently being planned.

A further underground test site has been initiated within the scope of an international project, utilizing a road tunnel through the Opalinus Clay at Mt. Terri in the Jura mountains. Studies at this site include examination of water flow paths through this formation and also characterization of the excavation-disturbed zone in this formation.

23.5 MAKING THE SAFETY CASE FOR HLW

Over the last decade or so, several countries have published comprehensive assessments of the safety of various disposal options for vitrified HLW and spent fuel. It has been found that these concepts differ quite significantly from each other and place the emphasis for a demonstration of safety on different parts of the multi-barrier system. At one extreme, the German concept of HLW disposal in a salt dome places emphasis on the role of the host rock in isolating the waste from advective water flow for very long time periods. Another concept with strong emphasis on geological barriers is illustrated by the Belgian concept of disposal in plastic clay. Canadian disposal in granite with very low hydraulic conductivity also emphasizes the geologic barrier, although long-lived container designs are also considered. At the other extreme, the Swedish concept for spent fuel encapsulation in thick copper canisters achieves long-term isolation by depending on the inertness of copper (which gives an expected canister life in the order of a million years and places very modest requirements on the geologic medium). The Swedish concept has also been adapted to Finnish conditions.

The Swiss concept places less emphasis on the retention capabilities of the geosphere or on the performance of individual engineered barriers; rather, it focuses on the behavior of the near field (all engineered barriers in their geological setting) as a whole[7,8]. Following waste emplacement, this near-field environment evolves in a

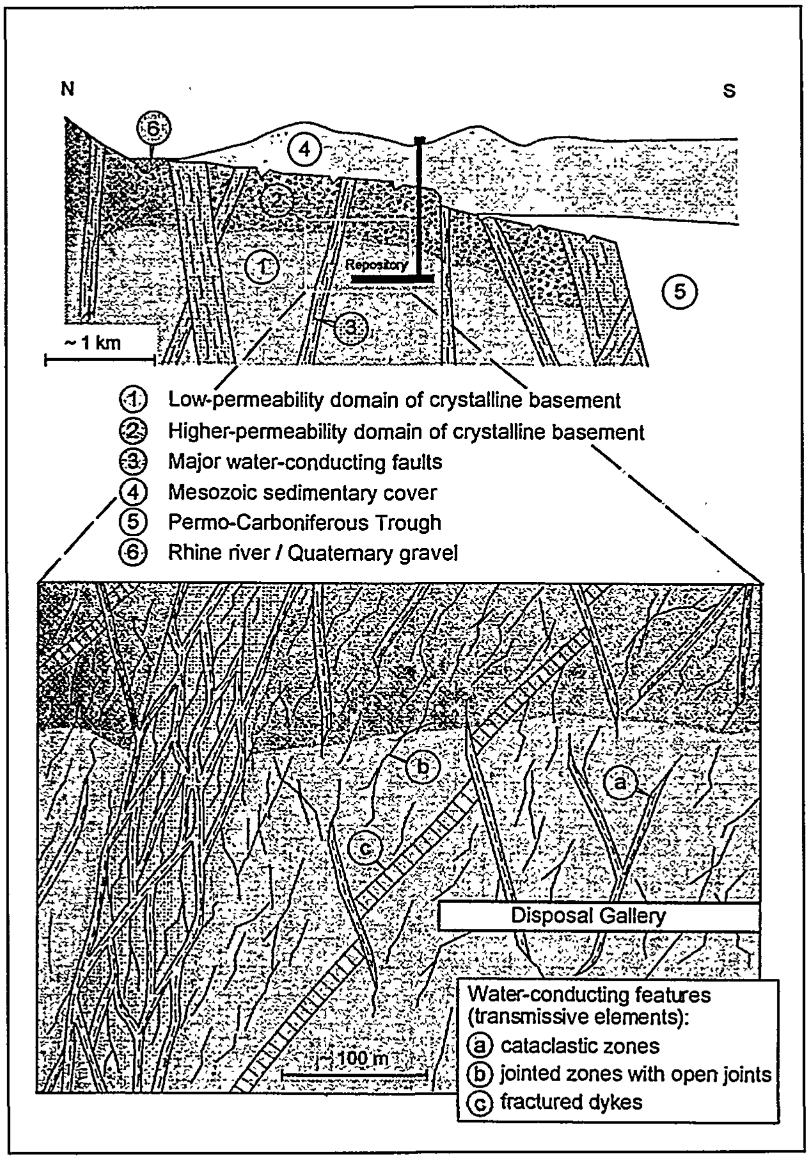

Figure 23.3. Diagrammatic representation of the conceptual model of the crystalline basement.

well defined manner.

After emplacement, water will resaturate the surrounding rock and then invade the bentonite which will swell to seal any gaps. Temperatures in the bentonite will increase due to heat from the canister, but storage of waste prior to emplacement ensures that the maximum temperature in the backfill does not exceed ~160°C. Calculations indicate that complete resaturation may take in the order of hundreds of years. The steel canister corrodes anaerobically at a very low rate (~50 m/year). Only after ~1000 years will mechanical failure occur due to pressure from the expanding bentonite. The water chemistry in the compacted bentonite will be determined by interactions of inflowing granitic groundwater with mineral surfaces in this microporous material. The bulk of the bentonite itself will not undergo any significant mineralogical alteration over relevant timescales (~1 million years).

After canister failure, corrosion of the glass will occur in an environment with effectively stagnant porewater. Corrosion of the glass is expected to gradually release the contained radionuclides over a period of the order of 10^5 years. The release of many radionuclides may, however, be further constrained by their very low solubili-

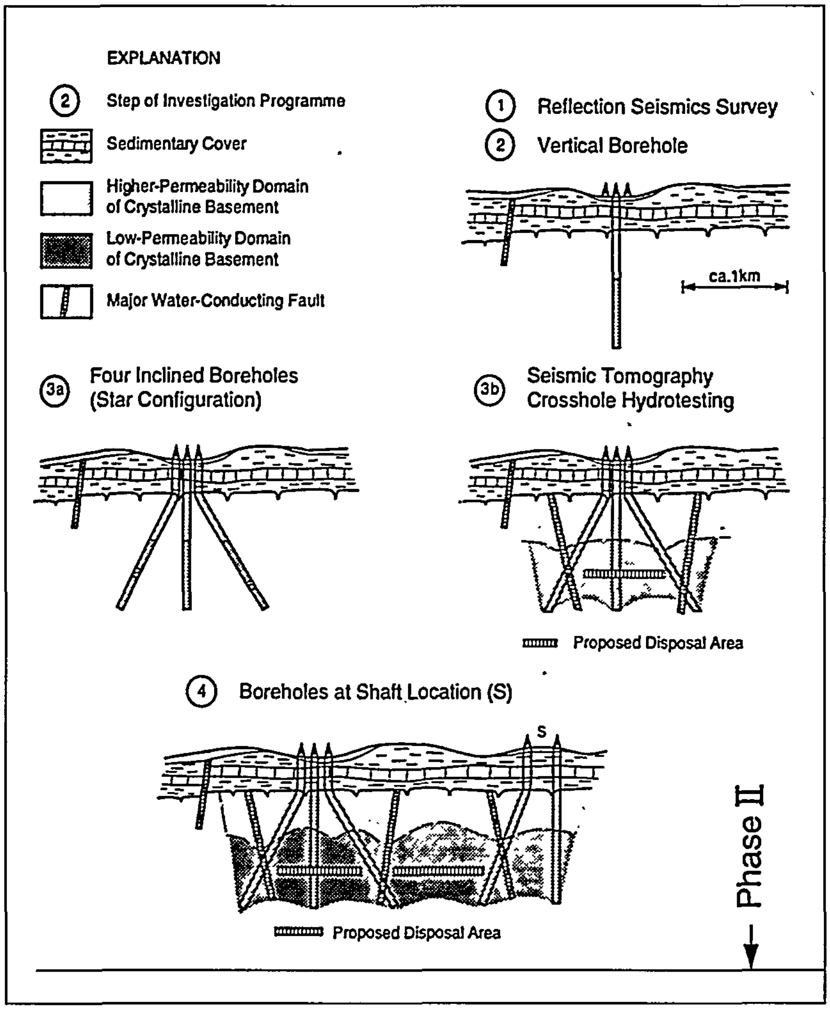

Figure 23.4. Illustration of the investigation concept for the crystalline basement.

ties. Even after radionuclides are released from the glass matrix, output to the geosphere is greatly limited by the transport resistance of the bentonite backfill. Due to its extremely low hydraulic conductivity, solute transport through the saturated bentonite will occur predominantly by diffusion. Sorption processes result in very low diffusivities for many radionuclides, so that their transport time through the backfill exceeds their half-lives, and thus, output is negligible. This analysis of the repository near field is believed to be robust[7], in that it does not take credit for all possible processes which would decrease release rates, and is relatively insensitive to the variation of uncertain parameters within reasonable ranges.

Although sufficient safety can be demonstrated with only minimal requirements on the geology, the geological barrier can also be extremely powerful, reducing the already low concentrations of radionuclides yet further. Under the expected geological conditions in Northern Switzerland, the geological barrier would ensure that, in effect, no releases to the human environment would occur for all timescales which are meaningful (up to one million years). The calculation of retardation of radionuclides in the geosphere is, however, more sensitive to parameter values which are difficult to determine in the field and hence a safety case based strongly on the barrier effect of a fractured geologic medium is less robust.

The demonstration of long-term safety for a HLW repository must be based upon predictive modeling, and it is important to realize the capabilities and the limitations of such models. The mathematical models used in performance assessment are supported by a range of laboratory and field experimental studies, but the extrapolation of such work to very long timescales must also

be justified. Enhanced confidence in our understanding of long-term performance of the near-field, in particular, may be illustrated by using natural analogues.

A natural analogue is a process that has occurred in the past and is similar to those that are predicted to occur in the future evolution of a repository. The proposed corrosion rates of the waste matrix and canister in the expected chemically reducing environment can be supported by observation of the preservation of archaeological glass and steel artifacts over millennia. Mineralogical stability of bentonite can be shown on an even longer timescale by observations of natural bentonites that have remained unaltered for millions of years in conditions comparable to those expected in the repository. Even the ability of clay to isolate radioactive substances can be illustrated by natural ore bodies in appropriate geologic settings.

A safety case made on the robustness of a system of engineered barriers appears to be appropriate to the Swiss geological environment, and it is interesting to note that a very similar concept has been adopted by Japan (a country with even more complex and tectonically active geology than Switzerland) in their H-3 performance assessment.

23.6 THE SWISS PROGRAMME IN AN INTERNATIONAL CONTEXT

The Swiss waste management programme, although relatively small in terms of budget and manpower, is very wide in scope, with one site currently being characterized in detail for a L/ILW repository and two types of host rocks under investigation for disposal of vitrified HLW and long-lived ILW. This programme is only feasible if priorities are set and adhered to, and if maximum advantage is taken of work performed elsewhere. Therefore, extensive use is made of international collaboration agreements in order to spread the work load. Individual information exchange agreements with other programmes have allowed effort to focus in specific areas. For example, Switzerland could deliberately concentrate on studies of steel canisters for HLW because Sweden has concentrated on copper, and a bilateral Nagra/SKB agreement provided for exchange of results. Such agreements also allowed direct cooperation/co-funding of larger studies, such as the Japan/Sweden/ Switzerland (JSS) study of the leaching of vitrified HLW. Cooperation with the US DOE has allowed results from the Swiss underground test site to be made available to modeling groups in the US who, in turn, make their interpretations available to Nagra.

Apart from active participation in the IAEA and the NEA, Nagra has formal agreements with the European Economic Community (EEC), United States (DOE, NRC), Sweden (SKB), Finland (Posiva), France (CEA and ANDRA), Belgium (ONDRAF, CEN/SCK), Germany (GSF/BRG), Japan (PNC), Spain (ENRESA), Taiwan (AEC) and the United Kingdom (NIREX). Informal collaborations extend the list further.

23.7 CONCLUSION

Despite its small size and limited nuclear power capacity, Switzerland has succeeded in establishing an internationally recognized programme for management of radioactive wastes. The restricted size makes lines of communication shorter and coordination of effort simpler. The relatively strong economy makes the financing of projects, without the benefits of scale, a feasible proposition, although the economic sense of establishing various small-scale projects through the world can be questioned. Sound technical projects can be developed and implemented with limited human resources, provided that care is taken to make polyvalent use of expertise and to profit from mutually beneficial collaboration with other national programs. Even front-line science and advanced engineering skills are, however, of little use if public opposition prevents their application. Hence, it is important that a waste management organization like Nagra devotes strong efforts to communication with the public at all levels.

REFERENCES

1. Nagra, "Projekt Gewähr 1985: Nuclear waste management in Switzerland: Feasibility studies and safety analysis"; Nagra Project Gewähr Report series, NGB 85-09, Nagra, Baden, Switzerland, 1985.

2. Nagra, Nukleare Entsorgung Schweiz - Konzept und Realisierungsplan; Nagra Technical Report series, NTB 92-02, Nagra, Wettingen, Switzerland, 1992.

3. Nagra, Kristallin-I: Conclusions from the regional investigation programme for siting a HLW repository in the crystalline basement of Northern Switzerland; Nagra Technical Report series, NTB 93-09E, Nagra Wettingen, Switzerland, 1994.

4. Thury, M., et.al., Geology and hydrogeology of the

crystalline basement of Northern Switzerland - Synthesis of regional investigations 1981-1993 within the Nagra radioactive waste disposal programme; Nagra Technical Report series, NTB 93-01, Nagra Wettingen, Switzerland, 1994.

5. Nagra, Sedimentstudie - Zwischenbericht 1993: Zusammenfassende Uebersicht der Arbeiten von 1990 bis 1994 und Konzept für weitere Untersuchungen; Nagra Technical Report series, NTB 94-10, Nagra Wettingen, Switzerland, 1994.

6. Nagra, Grimsel Test Site - Overview and Test Programs; Nagra Technical Report series, NTB 85-46, Nagra Wettingen, Switzerland, 1985.

7. Nagra, Kristallin-I - Performance assessment; Nagra Technical Report series, NTB 93-22, Nagra, Wettingen, Switzerland, 1995.

8. Neall, F.B., (ed), Kristallin-I - Results in perspective; Nagra Technical Report series, NTB 93-23, Nagra Wettingen, Switzerland (1996).

CHAPTER 24

LOW LEVEL RADIOACTIVE WASTE MANAGEMENT IN TAIWAN

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24.1 BACKGROUND

The commercial operation of Chinshan Nuclear Power Plant (NPP) Unit one marked the beginning of Taiwan's nuclear power program. There are now three NPPs each consisting of two units, in operation. With a generating capacity of 5,144 MWe, nuclear power produces some 30 percent of the electricity supplies in Taiwan. However, the nuclear power component is decreasing as the power demand increases. In order to meet the increased power demand, the Taiwan Power Company (TPC) , a state-run and sole electricity utility in Taiwan, decided to build one more nuclear power plant with two reactors at the Yenliao Site in addition to the other sources. Detailed information about Taiwan's nuclear power program is shown in Table 24.1.

As far as low level radwaste (LLRW) is concerned, TPC is the principal source contributing more than 90 percent of total volume of waste produced in Taiwan. Small producers in the medical and research institutes and universities are responsible for the remaining 10 percent.

24.2 RADWASTE MANAGEMENT POLICY AND ORGANIZATIONAL SCHEME

24.2.1 Policy

On the 16th of September 1988, the Executive Yuan (the Cabinet) promulgated the Radwaste Management Policy (RWMP) that set up the principal guidelines to enable the Taiwan nuclear industry to plan and manage its radwaste. Highlights of the RWMP concerning LLRW are summarized as follows:

- the radwaste producers should strive to minimize the waste generation rate and reduce the volume;
- the responsibility of safely treating, transporting, storing, and disposing of radwaste should rest with the producer. Therefore, the producer is responsible for the necessary expenses; and
- an LLRW disposal site should be located by 1996, and operational by 2002.

24.2.2 Organizational Scheme

The organizations related to radwaste management are shown in Figure 24.1. 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 orga-

Table 24.1. Information on nuclear power plants in Taiwan.

Unit	Reactor Type	Installed Capacity	Commercial Operation	Status
Chinshan 1 (C1)	BWR/4	636	1978	operating
Chinshan 2 (C2	BWR/4	636	1979	operating
Kuosheng 1 (K1)	BWR/6	985	1981	operating
Kuosheng 2 (K2)	BWR/6	985	1982	operating
Maanshan 1 (M1)	PWR	951	1984	operating
Maanshan 2 (M2)	PWR	951	1985	operating
Yenliao	ABWR	1300	2000 (scheduled)	bidding

nization to the AEC, assumes regulatory control over radwaste management matters. The Institute of Nuclear Energy Research (INER) was empowered by AEC to take responsibility for collecting radwaste generated by small producers and treat the waste as necessary. In TPC, the Nuclear Backend Management Department (NMBD) and the Nuclear Operation Department (NOD) take care of radwaste generated by the NPPs. NOD's major responsibility is to supervise treatment and storage of LLRW within the NPPs, whereas NMBD is responsible for radwaste transportation, the operations of both the Lan-yu storage site and the Volume Reduction Center, but more importantly, the final disposal of LLRW in Taiwan.

24.3 Technical aspects of LLRW management

Before introducing the detailed technical issues of LLRW management in Taiwan, it is better to review the LLRW management diagram (see Fig. 24.2).

24.3.1 Radwaste Generation

LLRW in Taiwan can be divided into two categories: wet waste and dry active waste. Wet wastes, namely: evaporation residues, filter sludges, and spent bead resins, are first solidified in carbon-steel drums and then stored in structurally safe warehouses. Dry active wastes, which are mainly waste paper, clothes, plastics, wood materials, metal, etc., are either segmented or shredded and also stored in warehouses. The cumulative amounts of radwaste generated through August 1994 are listed in Table 24.2. Cement is the most commonly used solidification agent for wet waste. However, bitumen is used in solidifying incinerator ash.

Thanks to waste reduction efforts implemented by industry, the annual radwaste generation rate at the three nuclear power plants has been decreased from more than 12,000 drums prior to 1990 to less than 8,000 drums afterwards. A particularly significant reduction has been achieved for solidified wastes. Together with radwaste generated by small producers, the present annual radwaste generation rate is approximately between 5500 and 6500 drums. Up to now, almost half of the radwaste drums have been shipped to the Lan-Yu National Storage Site for extended storage. However, the remainder of the radwaste is stored in warehouses on site. As the nuclear facilities are nearly running out of storage capacity, a computerized and improved and better equipped on-site warehouse at two of the three NPPs and at INER is either being constructed or is planned. These new facilities are scheduled to commence operation in the near future.

24.3.2 Waste Volume Reduction

Reducing the volume of both combustible and compactable wastes is justified as a good way of mitigating storage pressures given the ever-increasing quantities of

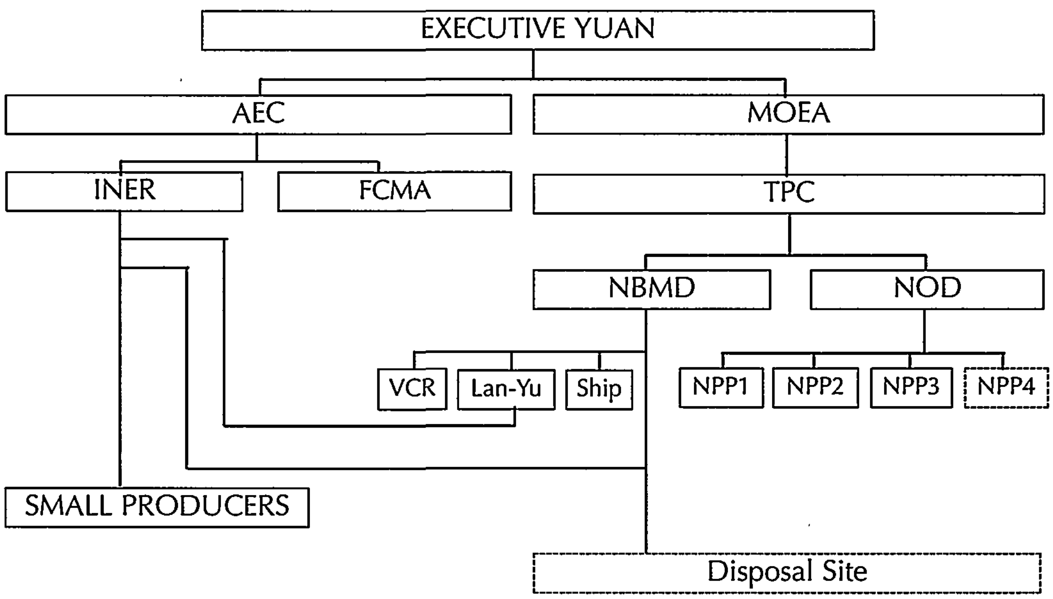

Figure 24.1. Organizations related to radwaste management in Taiwan.

Table 24.2. Total amount of LLRW in Taiwan (unit: 55 gal. drums).

Plant	Solidified Waste	Non-Solidified Waste	Totals
NPP I (C1&C2)	43,684	31,474	75,158
NPP 2 (KI&K2)	53,090	20,876	73,966
NPP 3 (MI&M2)	7,004	3,627	10,631
INER	13,774	328	14,102
Total	117,552	56,305	173,857

LLRW. Hence, TPC built a Volume Reduction Center at Kuosheng NPP site. The center comprises a controlled air incinerator and a supercompactor. With a capacity for burning 100 kg/hr of combustible waste and compressing five waste drums per hour of compactable waste, this center is able to eliminate about 3,500 waste drums annually. This has helped to relieve storage problems to a great extent. The important operating parameters of the Volume Reduction Center are shown in Table 24.3. INER has also constructed a controlled-air type of incinerator with a burning rate of 40 kg/hr to treat combustible wastes originating from small producers islandwide.

24.3.3 Lan-Yu Interim Storage

The National Lan-Yu Storage Site provides off-site interim storage for solidified radwaste. This site is located on the small island of Lan-Yu that has an area of about 45 km^2, and indeed, was originally designed as a port of departure for sea dumping that is no longer allowed. Twenty-three semi-underground engineering trenches were constructed on the site, providing a storage capacity of 98,000 drums with three drums being stacked vertically. As of August 1996, the site had received about 97,700 waste drums, and it is anticipated to be full by 1996. However, in a continual search for sufficient storage space and to allow ample time for proceeding with the LLRW disposal program, TPC, the site owner, plans to expand the storage capability by adding six better shielded trenches with a capacity of 59,000 drums. Simultaneously, an optimized waste drum loading pattern will be adopted, making better use of the land. The environmental impact report for this project is under review by the AEC's Environment Evaluation Committee.

24.3.4 LLRW Transportation

Due to the need to continually ship solidified LLRW to the Lan-Yu site before the disposal facility was commissioned, TPC built a modern LLRW transport ship, Teen-Kung No. 1, to replace an old ship in 1991. The new ship is 53 meters long and has a deadweight of 737 metric tons at the designed draft. It can reach a speed of 11.5 knots. Furthermore, it features a double-shell hull, auto-

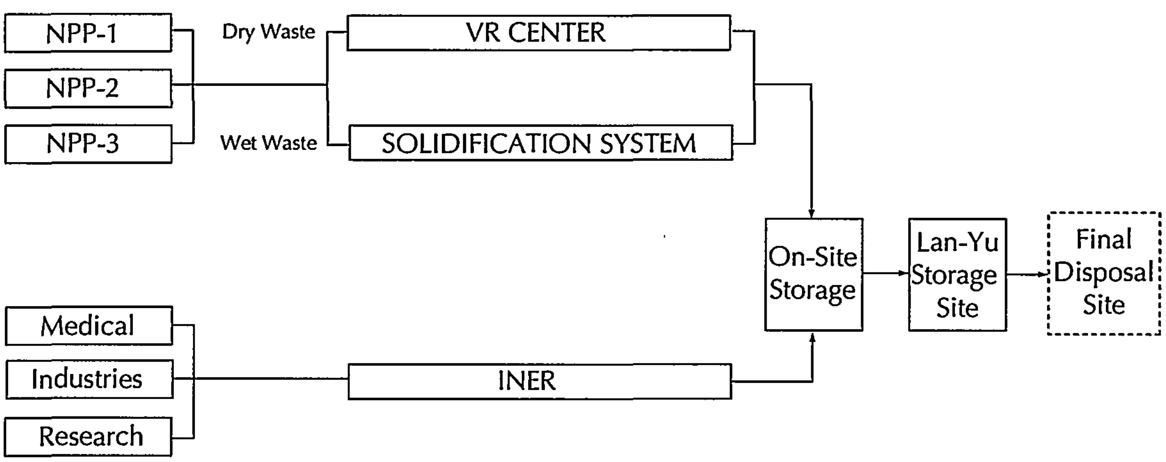

Figure 24.2. Diagram of low level radwaste management in Taiwan.

Table 24.3. Important operating parameters of the Volume Reduction Center.

Parameters	Value
Incinerator:	
Burning rate (kg/hr)	100
Operating temperature (°C)	
1st chamber	700 - 900
2nd chamber	1000 - 1200
Volume reduction ratio	30-100:1
Weight reduction ratio	30-40:1
Supercompactor:	
Compacting force (ton)	1500
Feeding rate (drums/hr)	5
Volume reduction ratio	3 - 5

matic navigation, satellite-relayed communication, and state-of-the-art radiological protection equipment. It can carry up to 576 waste drums per shipment.

24.4 LLRW FINAL DISPOSAL

Low level wastes presently stored on site or in the Lan-Yu site have to be permanently disposed of in a safe manner. Due to the RWMP direction and in light of the fact that TPC contributes 90 percent of the LLRW generated in Taiwan, TPC has been designated to assume this work.

24.4.1 Regulatory Requirements

According to the "Low Level Radwaste Land Disposal Licensing Regulations" issued by AEC-RWA, the annual dose to any member of the public resulting from release of radioactivity from a disposal site must not exceed 25 millirems (0.25 mSv). When the individual dose is less than 1 millirem (0.01 mSv) and the collective dose less than 100 man-rem (1 man-Sv), the disposal site can then be freed from institutional control. The regulations also point out a set of siting requirements for the final disposal program. They are that the site should:

- be situated in an area with low population density and low development;
- avoid an area in which tectonic activity, geological processes, hydrological and geohydrological conditions could endanger the safety of the disposal facility; and
- be kept away from an area where geological and hydrological data are too complicated to be adequately evaluated.

24.4.2 Geological conditions in Taiwan

Taiwan measures about 36,000 km^2 in area with a spindle shape for the island. There are 81 islets spreading out in the surrounding Pacific ocean, and 64 of them are known as the Penghu Island Group, or the Pescadores, in the Taiwan Strait (Fig. 24.3).

Located at the boundaries between the Eurasian plate and the Philippine sea plate, Taiwan island reaches a maximum elevation of about 4000 m as a result of the compression and shear forces. It is an arcuate island extending its shorter arm eastward to the Ryukyus and its longer arm southward to the Philippines. The backbone of this mountainous island is the Central Range which is mainly Tertiary in age. It is fringed on the west by the Foothill Zone and separated on the east from the Coastal plain with the very shallow Taiwan Strait farther west; east of the Coastal Range is the deep Pacific Ocean. The offshore islets of Taiwan include the Penghu Group in the Strait and Liitao and Lan-yu off the southeast coast. Kinmen and Matsu are two islands close to mainland China covered with Mesozoic granitic gneiss which may be a surface extension to Taiwan. In the less tightly compressed northeastern and southwestern parts of the mountain complex of the Central Range and Foothills of Taiwan, there is the Ilan plain and the Pintung Valley, each in the form of an intramundane trough intruding from the sea into the island.

24.4.3 LLRW Final Disposal Program

The TPC's program plan for LLRW disposal will be carried out in the following 6 phases:

Phase 1. Selection of Disposal Site and Method
The site selection criteria and process were developed taking into account Taiwan's local conditions and foreign experience. Based on the available geological and socio-environmental situation, a handful of candidate sites will be identified in accordance with siting criteria. Further investigations, including core drilling and laboratory testing on those candidate sites will then follow. Various land disposal methods will be assessed against each candidate site condition to determine those that are suitable. In this manner, the most favorable disposal site

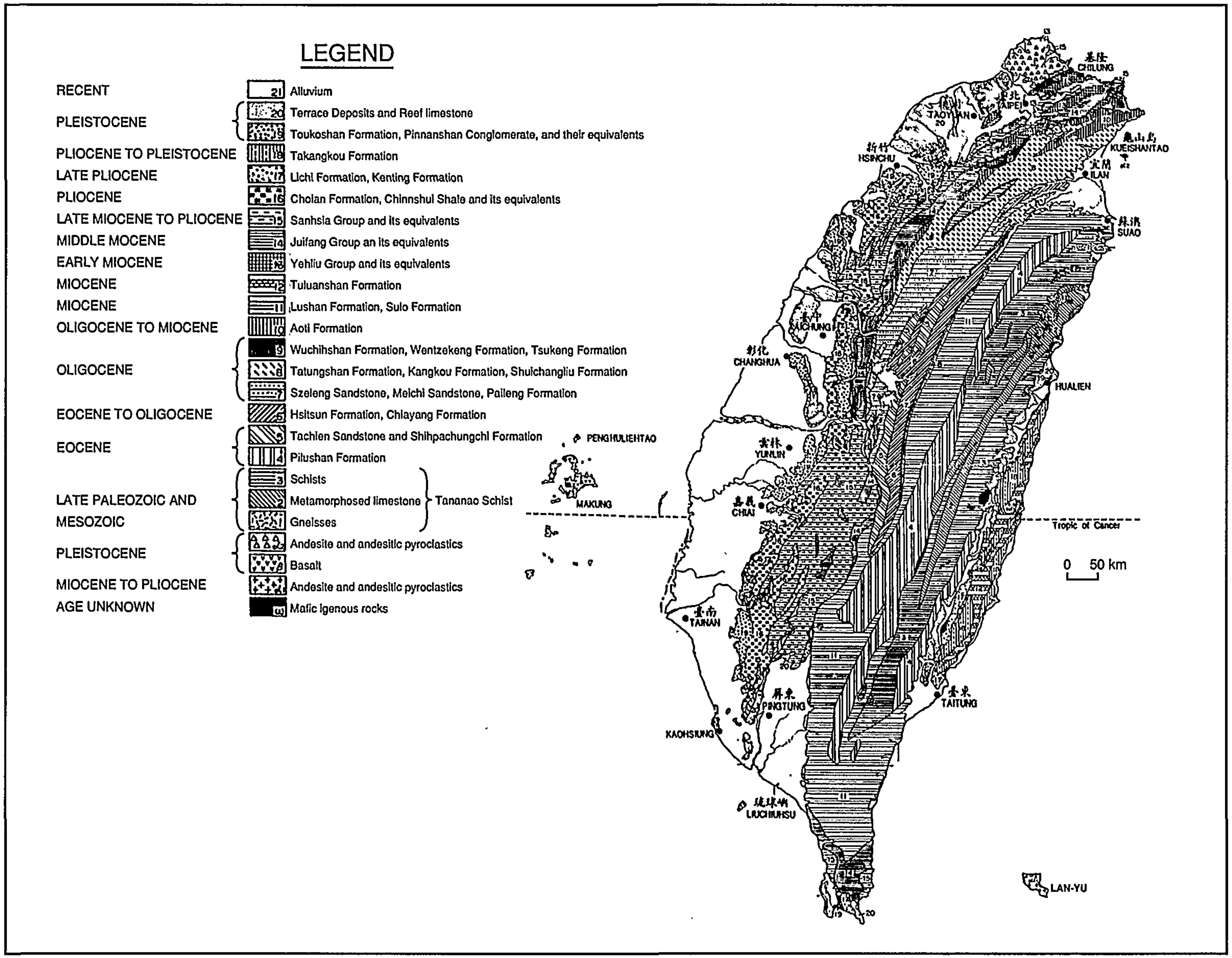

Figure 24.3. General geologic map of Taiwan.

and method can then be selected.

Phase 2. Environmental Survey and Assessment
The environmental data survey and documentation will be conducted in parallel with drilling investigations on the above mentioned candidate sites. Results of the environmental assessment will become part of the attributes used for evaluating and comparing candidate sites.

Phase 3. Site Characterization, Engineering Design and Licensing
It is expected to take at least two years to complete this phase. During the phase, the site characterization, engineering design and the detailed safety analyses will be undertaken to support the presentation of a construction license application.

Phase 4. Site Construction
Depending on the site condition and disposal method, it is expected to take three years to complete the initial phase in the construction of the disposal facility. An operating license application is scheduled to be submitted to the government in 2001 for review.

Phase 5. Operation
The disposal facility is programmed to be commissioned in early 2002 if everything goes as planned.

Phase 6. Post-Operation Monitoring
After the disposal facility ceases operation, it will be backfilled, stabilized, and covered with earth and vegetation. The disposal site and its vicinity will then be monitored until the radioactivity in the disposed waste has decayed substantially and no longer presents a risk to the environment.

The milestones for each phase of the disposal program are shown in Table 24.4. However, due to nontechnical factors, Phase 1 has been postponed to the end of 1996.

24.5 PUBLIC ACCEPTANCE

The importance of securing public acceptance in proceeding with the LLRW management program has long been recognized by the nuclear industry. The continuing receipt of protests against the storage of LLRW in the Lan-Yu Storage Site from native residents is one of the examples of this kind. Another example could be justified by the strong and violent protests from an opposition party in the parliament to freeze the budget for the construction of the fourth NPP. Currently, the opposition party has about one-third of the seats in the parliament. As elsewhere in the world, nuclear safety and radwaste management in Taiwan have become the two major issues of the anti-nuclear movement.

It is anticipated that, in the future, the establishment of a LLRW final disposal facility could receive many objections from the public since the disposal site will be situated at a given location for a few hundred years. The radwaste people in the nuclear industry are deliberating on how to get the public involved at an early stage in

Table 24.4. Taipower's overall program plan for LLW final disposal.

Plan Phase	1992	1993	1994	1995	1996	1997	1998	1999	2000	2001	2002
Phase I: Selection of Candidate Site and Disposal Technology (10/92-9/95)											
Phase II: Environmental Impact Assessment (5/93-9/95)											
Review by Government Authorities (10/94-9/95)											
Phase III: Site Characterization and Engineering Design (10/95-3/99)											
Review by Government Authorities (4/98-3/99)											
Phase IV: Site Construction (4/99-9/2002)											
Review by Government Authorities (4/2002-9/2002)											
Phase V: Facility Operation (10/2002-)											

proceeding with any of the LLRW management programs. To clearly separate the issue of radwaste from that of nuclear power plant development may be strategically important in resolving the radwaste issue. Nevertheless, both to ensure the safety of the final disposal site and to provide a satisfactory financial aid to offset local objections may be the first two essential tasks to be worked on among other things.

24.6 CONCLUSIONS AND RECOMMENDATION

Taiwan is a country of scare natural resources of energy, and, therefore, the use of nuclear energy becomes a necessity. The management of radwaste arising from the use of nuclear power has to be safely planned and implemented. To locate a site, as early as possible, to permanently accommodate LLRW in Taiwan is considered the top priority among other management activities. Since the country is heavily populated and small in area, it welcomes any form of regional cooperation in the disposal of radwaste. Indeed, international cooperation in radwaste disposal is believed to be of benefit to the whole world.

It is hoped that an active program of regional cooperation on the disposal of LLRW can be initiated by a competent organization, such as PBNC (Pacific Basin Nuclear Conference), in light of the potential benefits to this region.

REFERENCES

1. Huang, C. H., Status of radioactive waste management in Taiwan, in Proc. 1993 International Conference on Nuclear Waste Management & Environmental Remediation, Prague, Czechoslovakia, 1993.

2. Huang, C.C., Y.T. Shao, and C.M. Tsai, How a developing country is facing LLW disposal Problem, in the Proc. 1993 International Confer-ence on Nuclear Waste Management and Environ-mental Remediation, Prague, Czechoslo-vakia, 1993.

3. Liu, D.S. Tony, R.T. Lee, and C.M. Tsai, Nuclear waste management in Taiwan - The past three years, in Proc. 1991 Joint International Waste Conference, Seoul, Korea, 1991.

4. Liu, D.S., R.T. Lee, and C.M. Tsai, Efforts of low level waste reduction in Taiwan, in Proc. SPECTRUM'90, Knoxville, Tennessee, 1990.

5. IAEA, A handbook for guiding good communication practices at nuclear fuel cycle facilities, Vienna, 1994.

CHAPTER 25

PROGRAMME AND RESULTS OF INITIAL PHASE OF RADIOACTIVE WASTE ISOLATION IN GEOLOGICAL FORMATIONS IN UKRAINE

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Abstract: The concept and a programme for radioactive waste disposal in geological formations of Ukraine have been developed. On the basis of certain criteria, an evaluation of the territory of Ukraine has led to the selection of three geological regions and three types of formations that are favourable for RAW disposal. The programme of research and development includes three stages: preparatory (1993-95), preparatory/experimental (1995 -2004), and preparation for construction (2005-2010). The completion of the preparatory stage resulted in the selection of zones and a number of candidate sites that are favourable for RAW isolation.

25.1 INTRODUCTION

Ukraine has been forced to develop research and development (R&D) programmes on radioactive waste (RAW) management due to their accumulation in significant quantities. This is a result of the rapid development of nuclear power and other RAW-producing industries, as well as the consequences of the Chernobyl disaster. According to the generally accepted point of view, a realistic solution for the RAW disposal problem is isolation in geological formations. The importance of this problem resulted in a research program initiated by the State Committee on Nuclear Power Utilization (Goskomatom), National Academy of Sciences, State Committee on Geology, and some other organizations.

The preparatory stage of research on RAW isolation in geological formations has been completed. Ukrainian scientists have developed the concept and a programme of R&D (experimental and methodological studies on a pilot scale as applied to geological and mining activities). The territory of Ukraine has been assessed as to the conditions for RAW isolation, and geological regions and formations favourable for this purpose have been selected. Regional studies, to be discussed below, have resulted in the selection of a number of favourable zones (areas) within these regions, and candidate sites have been selected. Simultaneously, preliminary analyses of the main engineering and construction problems related to RAW isolation have been made.

However, Ukraine is still lagging considerably behind in the field of R&D as compared to the countries that have been developing their programmes over several decades. Ukraine has established scientific relations with specialists in the field of RAW management. As a result of an international conference on "Isolation of RAW in Geological Formations" that was held September 20-24, 1994 in Kiev, a basis for cooperation in this field has been initiated with eastern European countries.

Investigations on R&D have been sponsored by Goskomatom partly by finances from budgets of participating institutions, as well as by special funds from the State Committee on Science and Technology for individual projects. These investigations are carried out by specialized multidisciplinary research teams, from 23 R&D institutions (Institute of Geological Sciences, State Committee on Geology, Kiev University, Goskomatom, etc.).

25.2 DESCRIPTION OF THE WORK

25.2.1 General Concept

The general concept for RAW isolation in geological formations in Ukraine 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[1-5]. The principle of long-term (over 10,000 years) RAW isolation is based on the idea of disposal as a geological engineering system that must satisfy a range of conditions (final form of RAW, disposal in deep geological formations at an

appreciable depth, special engineering barriers, etc.).

In the preparatory phase of the R&D programme, certain criteria were adopted, and an evaluation of the territory of Ukraine was carried out[3]. As a result, three geological regions (Fig. 25.1) and three types of geological formations, favourable for RAW disposal, have been selected (see below).

The amount of waste to be isolated is (metric tons): spent fuel - 27,000, decommissioned waste - 12,000, Chernobyl zone - 20,500, for a total of about 60,000. The future accumulations of spent fuel are estimated to be (metric tons): year 2000 - 2020, year 2005 - 3725, year 2010 - 5460.

After an appropriate period of cooling, or reprocessing procedure, the spent fuel has to be encapsulated. Technological waste must be conditioned and solidified. As for the Chernobyl zone wastes, there are two variants: straight burial and separation (enrichment) for volume decrease. Two types of canisters have been considered: stainless steel and steel-copper. After being reprocessed in Russia, the spent fuel has to be returned in standard containers intended for burial.

In selecting procedures for repository construction, world experience in underground methods, and current projections were taken into consideration. Our approach is to develop procedures that are the most simple and least expensive. The repository will consist of a wide transport tunnel and system of galleries for disposal (Fig. 25.2). Disposition of the different types of RAW involves: spent fuel in short boreholes in the floor of gal-

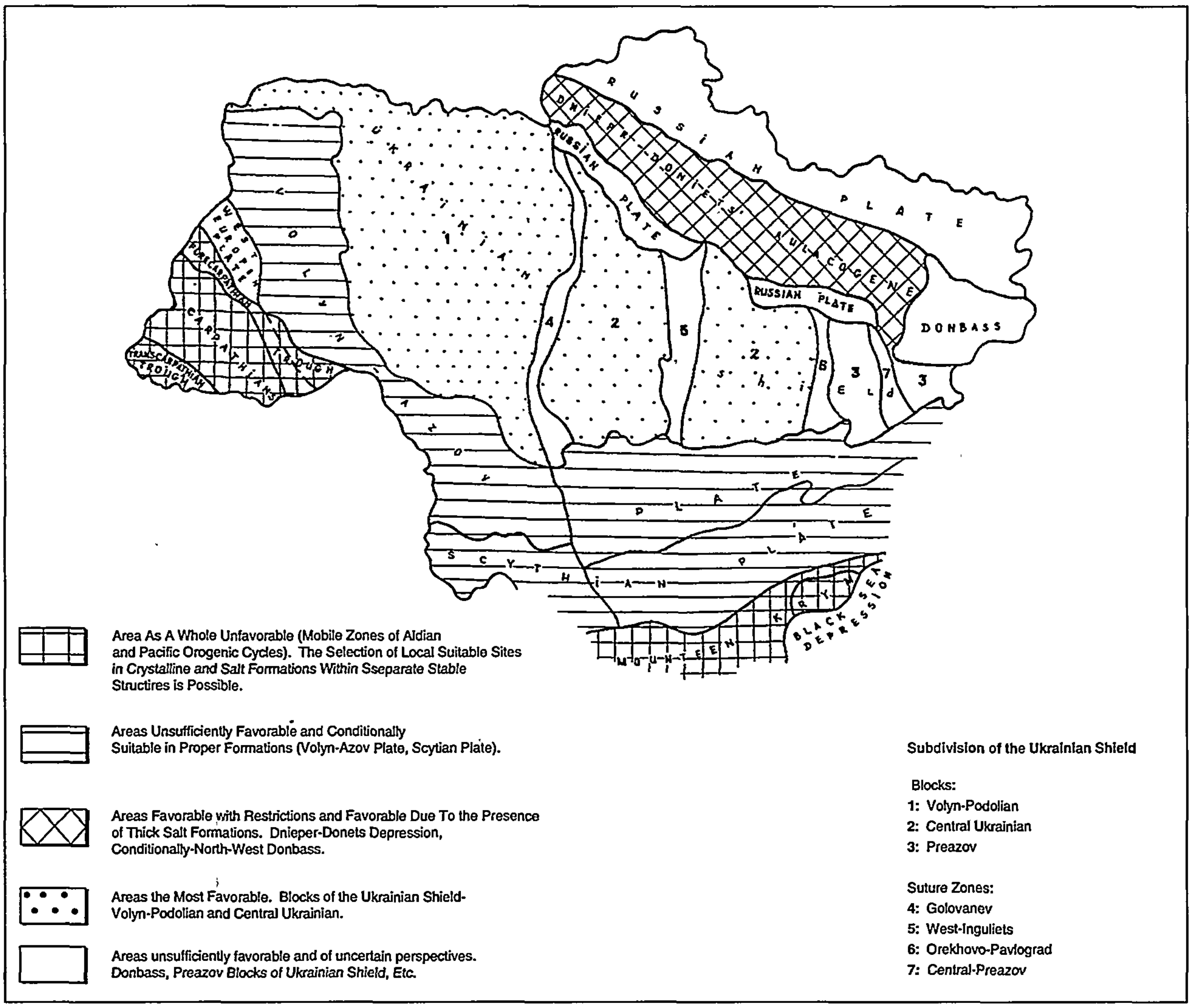

Figure 25.1. Subdivision of the Ukraine on conditions of RAW isolation in geological formations.

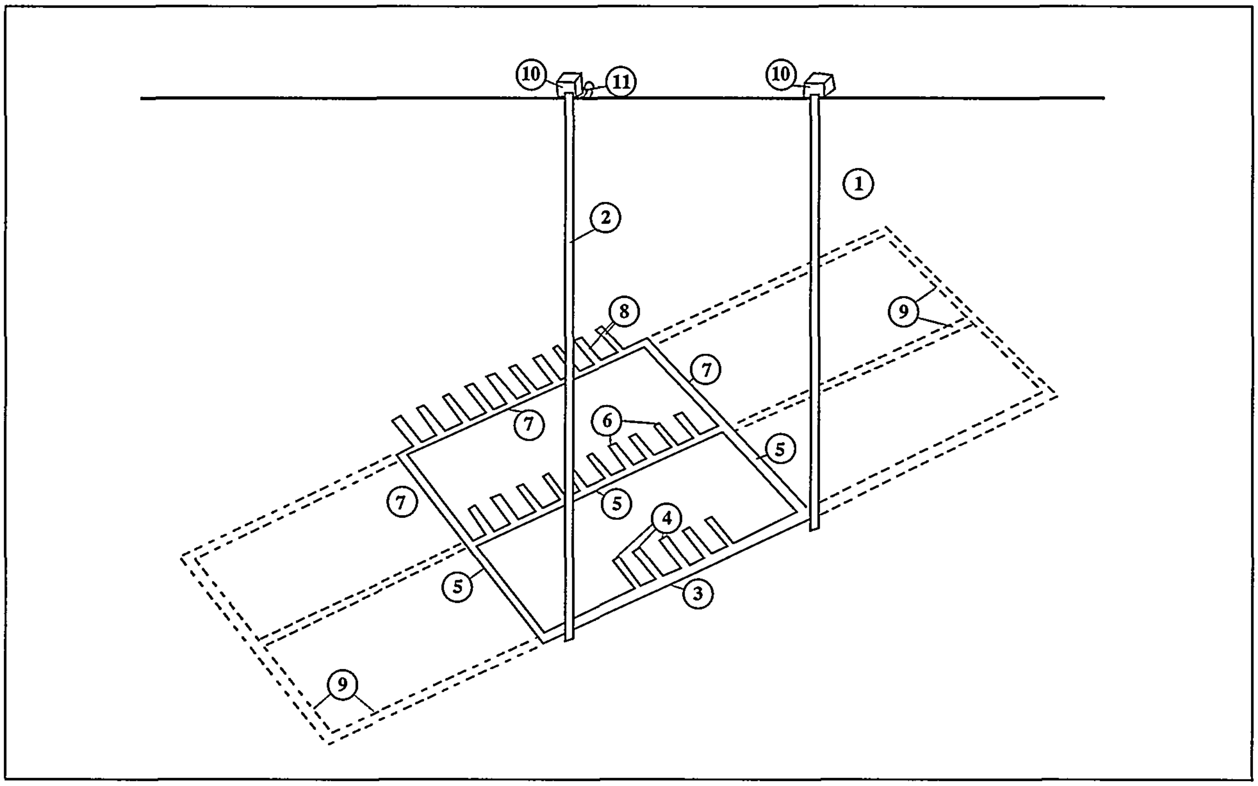

Figure 25.2. Concept of underground experimental laboratory and storage. Legend: 1 - Mine shaft; 2 - ventilation shaft; 3 - crosscut of laboratory; 4 - chambers of laboratory; 5 - crosscut of the first stage of storage; 6 - chambers of the first stage of storage; 7 - crosscut of the second stage of storage; 8 - chambers of the second stage of storage; 9 - crosscut of the following stages ; 10 - mine surface building; 11 - fan installation.

leries, and technological waste and Chernobyl waste in special cells in the galleries. The system of engineering barriers includes the matrix, a buffer, containers, compactors, and backfilling material of a bentonite composition. Sometimes, a protective covering is needed on the cavern walls. Special non-blasting 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. The investigations in the URL are of a traditional nature but the programme may be shortened using results from actual world experience.

25.2.2 Research and Development Programme

The purpose of the R&D programme is to develop the complex measures needed for RAW isolation (long-term storage and final burial) in geological formations. The programme is based on total safety for the population and environment using principles elaborated by IAEA. The overall programme for R&D includes: site selection and investigation, projections, exploration, construction, testing, exploitation and final closure of the isolation facility. There are seven topical areas: (1) mining/geology (including geological exploration); (2) technology; (3) social; (4) regulatory; (5) legal; (6) management and (7) construction.

The area of mining/geology is actually central and is relatively independent due to its long term duration and the essential value of the data being collected. The main tasks in the mining/geology area are as follows:

1. Conducting theoretical investigations of geological, geochemical, hydrogeological, geomechanical, mining, thermophysical and other problems connected with site selection, exploration, construction, exploitation and closure of the isolation facility, as well as safety and the development of a methodology of investigations.

2. Evaluation of the territory of Ukraine from the point of view of RAW disposal.
3. Regional studies to elaborate on selection criteria, structures and the selection and evaluation of sites.
4. Supervision of exploration on selected structures and sites.
5. Construction of URL to carry out selected experiments.
6. Perfection of construction and technological parameters for the RAW isolation facility based on the synthesis of exploration data, experiments in the URL etc.
7. Develop a prognosis for the functioning of the isolation facility under the influence of the effects of geological evolution and scenaria of possible catastrophic events (safety analysis).
8. Develop a monitoring system and system of control (management).
9. Provide a basis for controlling construction of the RAW isolation facility.
10. Provide a basis for supervising the exploitation and closure of the facilities.

The programme of R&D includes the following stages:

1. Preparatory - 1993-95
   Goal: elaboration of concepts, selection of sites, exploration;
2. Preparatory/experimental - 1996-2004
   Goal: exploration, construction of URL, collection of experimental data during construction and exploitation of the isolation facility, and
3. Preparation for construction - 2005-2010
   Goal: final preparation for construction of RAW isolation facility (eventually the beginning of the construction).

The durations of these stages are not yet firm and will actually depend on the financial situation.

25.2.3 Methodology of Scientific Investigations.

The generally accepted concept of the repository as a multibarrier, geological/engineering system takes into account that the rock formation, as a main barrier, is the leading factor in determining the safety of long term isolation. That is why a comprehensive investigation of the geological environment provides a foundation for investigations in the area of mining/geology.

The final goal in the preparatory stage of the R&D programme is the selection of site(s) for the isolation of RAW. This goal can be achieved by solving the following tasks in a proper hierarchical sequence that is correlated with the stages of programme investigations mentioned above:

1. Evaluation of the territory of Ukraine from the point of view of RAW isolation;
2. Selection of geological regions and formations potentially favourable for RAW isolation;
3. Regional analysis of potentially favourable formations in a hierarchical sequence. Region-Zone (group of structures)-Local Structure (site), (Fig. 25.3); and
4. Selection and evaluation of sites.

Tasks 1-3 and part of 4 have already been accomplished.

As a result of the evaluation and ranking of 12 geological regions in Ukraine, only three have been selected as favourable for RAW disposal: (1) Ukrainian shield; (2) Dnieper-Donets depression; and, (3) northwestern Donbass. Conditionally, the southwestern slope of the east European platform of the ancient Volyn-Azov plate is under investigation.

The selection of formations in these regions was made on the basis of an initial evaluation of parameters and by analogy with world experience. By mean of this approach, three types of potentially favourable formations have been selected: crystalline, salt and argilaceous. The next step was to carry out a regional analysis of formations.

The hierarchical sequence of investigations (Regional-Zonal-Local) is similar for all formations. The methodology of the selection process on these three levels is based on the usage of a set of mining/geology area models, categorized on different scales according to the level of investigation.

The results of this selection of a set of models (as well as data from social, economic, ecological and other studies) provide the basis for setting up criteria for the selection, comparison, estimation and ranking on zonal and local levels. This set includes three groups of practically equal importance:

I. Safety (technical-geological)
   a. tectonic
   b. neotectonic
   c. seismic
   d. hydrogeologic
   e. type of formation

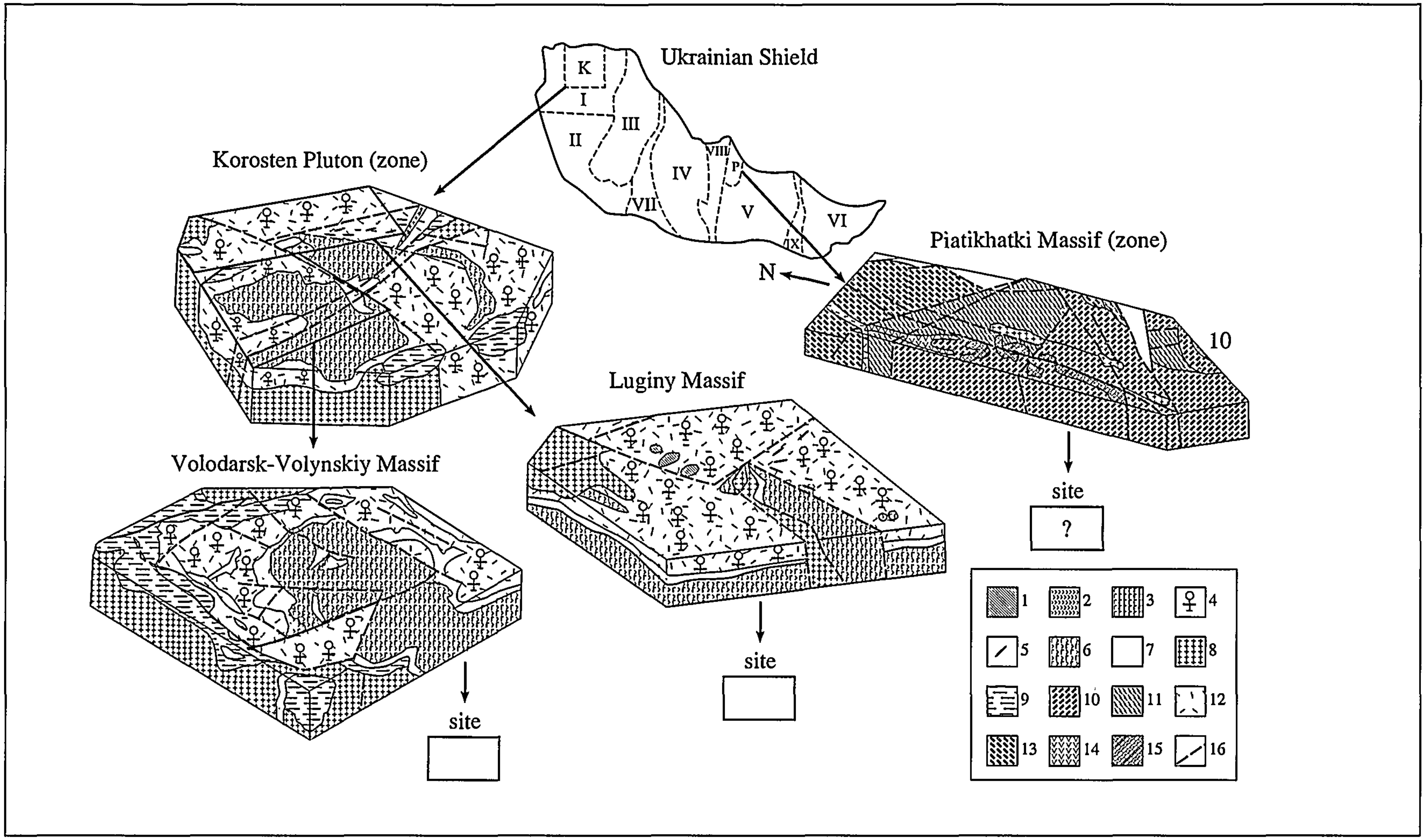

Figure 25.3. Principles of site selection in Ukrainian shield: I - Volyn, II - Podolia, III - Bielaya Cerkva, IV - Kirovograd, V - Near-Dnieper, VI - Near-Azov Pavlograd, VII - West Ingoulets, VIII - Golovanev, IX - Orechovo. Legend: 1 - alkaline syenites; 2 - monzonites, gabbro monzonites: 3 - olivine gobbros, rarely gabbro-peridotites, gabbro-pyroxenites; 4 - granite-porphyries, rapakivi and rapakivi-like granites; 5 - dikes of diabase and gabbro-diababase; 6 - anorthosites, gabbro-anorthosites; 7 - norites, gabbro-norites; 8 - granites (Zhitomir and Kirovograd types); 9 - biotite gneiss, quartzitic sandstones; 10 - plagiomigmatites, plagiogranites; 11 - granites, migmatites; 12 - biotite granites, aplitic and pegmatoid granites; 13 - tonalites, plagiogranites, plagiomigmatites; 14 - andesite and diabase metaporphyry amphibolites; 15 - graphitic and biotites shales, metasandstones; 16 - faults.

f. goemechanic
g. geochemistry (including waste composition)
h. geomorphologic
i. hydrologic
j. climatic
k. technogenic/ecologic conditions
l. mineral deposits

II. Social-political
a. demographic
b. psychologic
c. contamination from Chernobyl

III. Technologic complexity
a. construction cost
b. technologic complexity

On the zonal and local levels, the results of the social-economic and ecological investigations must be developed in different degrees of detail.

In the areas of mining/geology, the set of models that are being used in the hierarchical sequence of regional investigations includes both static and dynamic aspects. The dynamic aspect must consider two variants: evolutionary and revolutionary (catastrophic, or maximum project risk). Every model has its own tasks, objects and phenomena for investigation, but all models are integrated within the whole set. An understanding of the functioning of the disposal facility has to be developed from an appropriate synthesis of these models.

These models are developed using three kinds of data: theoretical, computational (mathematical, statistical, probabilistic) and experimental. The experimental data are obtained as a result of URL investigations. The dimensions of these models are generally known: near field and far field. The objectives of the investigations for this work are well described in the literature. The main task in analyzing the functioning of a disposal facility is the prognosis of its long term safety. This prognosis has to be developed within a framework that includes scenaria of evolutionary and catastrophic phenomena. One of the terminal tasks of modelling is the estimation of radionuclide behaviour in the biosphere (accumulations in surface waters, sorption by clays and organic matter etc.). The monitoring of an RAW disposal facility may be realized in the near field by means of direct observations *in* situ; in far field, by mean of special boreholes and surface observations.

25.3 RESULTS OF REGIONAL STUDIES

Regional studies have been carried out in the Ukrainian Shield, Dnieper-Donets Depression and Donbass.

In the Ukrainian shield (Fig. 25.3), two zones have been selected as favourable for RAW disposal: (1) Korosten pluton and a group of structures in the middle of the Near-Dnieper area, where the preferable type of rocks, granites and gabbro-anorthosites of Proterozoic age are found; and (2) in salt domes of the Dnieper-Donets depression in the northeastern, and southwestern marginal zones, and in the southeastern part of the Donbass depression, in bedded salt formations. Argillaceous formations of sufficient thickness are spread over the southwestern slope of the east-European platform (Volyn-Asov plates, Cambrian, Oligocene), and in the Donbass-Dnieper-Donets depression. In the latter locations, detailed geological investigations have not yet been carried out.

In the course of regional studies, several candidate sites have been chosen. In the northern part of the Korosten pluton, two favourable massifs (subzones) have been selected, i.e. Luginy and Volodarsk-Volyn massifs. The Luginy massif is composed primarily of granite-rapakivi, and two sites within its limits have been chosen. Eight salt domes that are potentially favourable for RAW disposal have been selected within the boundaries of the Dnieper-Donets depression.

On the basis of the mining/geology models and considering the criteria mentioned above, a ranking of selected sites has been made. Crystalline formations within the Korosten pluton have been ranked as follows: (1) first priority for the Pribytkov and Doroginby sites in the Luginy massif; and (2) second priority for the Novo-Borovaya and Zankovo sites in the Volodarsk-Volyn massif. Within the limits of the Dnieper-Donets depression, sites have been ranked as follows: (1) first priority for the Kaplintsy, Isachki and Yatsyno-Logoviki salt domes in the northeastern marginal zone; (2) second priority for the Dmitrievka, Siniovka, and Romny salt domes in the southwestern marginal zone; and (3) third priority for the Aleckseevka salt dome in the southeastern part of the depression.

Two zones of Permian bedded salt formation have been studied in northwestern Donbass.

25.4 CONCLUSIONS

As a result of completing the initial stage of the R&D programme, certain regional studies have been carried out. The regional studies resulted in the selection of favourable zones of crystalline formations in the Ukrainian shield and salt formations in the Dnieper Donets depression (as well as in northwestern Donbass, where technogenetic activities have to be considered.) Several candidate sites have been selected in favourable zones. The completion of this initial stage leads to the next stage of specialized exploratory geological/geophysical investigations. This stage is much more complicated and much more expensive.

The initial stage of investigations was completed during 1993-95. Such rapid advances were possible due to a thorough understanding of the geology of the territory of Ukraine, the excellent work of the scientific team, and the availability of results from world experience in the field of RAW in advanced countries (USA, France etc.).

Ukraine possesses scientific and technological capabilities sufficient for the effective completion of the necessary R&D related to exploration and URL construction. But the actual economy in the Ukraine provides no reason for optimism that financing sufficient for an effective realization of such an expensive program will be forthcoming. Thus, the possibilities for program support will depend upon a significant increase in national funds and the organization of international cooperation.

The Institutions involved in R&D programmes in waste isolation (as well as the Ministry of Environment Protection and Nuclear Safety) and the State Committee on Nuclear Energy Utilization, as a sponsoring institution, have initiated an annual international conference, "Isolation of RAW in Geological Formations." The first conference was held September 20-24, 1994 in Kiev. This conference has revealed the interest of eastern European countries (Poland, Slovakia, the Czech Republic, Russia, Hungary, Slovenia, Belarus, etc.) in a program of cooperation. A second conference is scheduled to be held in 1995.

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REFERENCES

1. Geological Problems in Radioactive Waste Isolation, A World Wide Review, ed. P.A. Witherspoon, Lawrence Berkeley Laboratory, Rept. LBL-29703, p. 233, 1991

2. Barr, G.E and E. Dunn, A working definition of scenaria and a method of scenario construction, Proceedings of the Fourth International Conference on High Level Radioactive Waste Management, Las Vegas, NV, pp. 1093-1098, April 26-30, 1993.

3. Khruschov D.P., V.I. Lialko, and O.M. Kharitonov, Isolation of radioactive wastes in geological formations of Ukraine, p.60, Preprint of Inst. of Geo. Sci., ed. Naukova Dumka, Kiev, 1993. (in Russian)

4. Radioactive Waste Management, an IAEA source book, p. 274, IAEA, Vienna, 1992.

5. Safety Principles and Technical Criteria for the Underground Disposal of High-Level Radioactive Waste, p. 33, IAEA Division of Nuclear Fuel Cycle, Vienna, 1989.

CHAPTER 26

THE INVESTIGATIONS OF THE GEOLOGY AND HYDROGEOLOGY AT SELLAFIELD IN THE UNITED KINGDOM

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26.1 INTRODUCTION

United Kingdom Nirex Limited (Nirex) is responsible for providing and managing a national disposal facility for solid intermediate-level (ILW) and low-level (LLW) radioactive waste. Such wastes have been produced in the UK for over 40 years and have come from the nuclear power industry, medical and defense establishments, as well as from other research and industrial sources. UK Government policy is to dispose of these wastes in a deep underground repository. Similar policies are followed by other countries which produce substantial quantities of long-lived radioactive waste.

Following an extensive site selection exercise, Nirex announced in 1989 that it would investigate, initially, sites at Dounreay in Caithness and Sellafield in Cumbria, to establish their suitability as safe locations for a deep disposal facility for ILW and LLW. Initial boreholes and other geological and geophysical surveys subsequently indicated that the geology at both sites had the potential to meet the demanding safety requirements for a deep repository. In July 1991, Nirex announced that it was to concentrate its further investigations at Sellafield. Given that there appeared to be little otherwise to distinguish between the overall suitability of the two sites, transport of waste and the associated costs were major considerations in this decision; an estimated 60 per cent by volume of the radioactive waste destined for the repository arises from British Nuclear Fuels' operations at Sellafield.

This paper presents a broad overview of the investigations carried out at Sellafield, up to approximately the end of 1994 and of the results obtained. A description is provided of the strategy being adopted for the continued investigation of the site. Descriptions of the geology, hydrogeology and geochemistry studies at Sellafield are provided by Michie (1996), Sutton (1996), and Bath, et al., (1996).

26.2 SCOPE OF INVESTIGATIONS

26.2.1 Scientific Approach

Nirex has adopted a systematic scientific approach to the design and implementation of the investigations. A wide range of technical specialists and techniques have been used in the conduct of the work and care has been taken to avoid undue reliance on any single technique in the interpretation of the ground conditions. The quality of the work being undertaken by Nirex has been recognized by independent reviewers, for example, by the Royal Society Study Group (1994) and RWMAC (1994).

Nirex makes information from the investigations widely available through publications and presentations. A series of papers on the geology of the Sellafield area were presented at a meeting of the Yorkshire Geological Society in late 1993) and subsequently published (Holliday and Rees, 1994). A second meeting on the hydrogeology was held at the Geological Society Apartments in May 1994, the papers presented at this meeting have been submitted for publication. Nirex also releases a significant number of detailed reports on the results of the investigations (For example: Nirex, 1992 Nirex, 1993a-i, Nirex, 1994a-b). An independent panel of university professors carries out review of the work undertaken by Nirex. The first Annual Report of this review panel was released in December 1994.

26.2.2 Areas of Study

The studies carried out in West Cumbria have been contained within three areas (Fig. 26.1):

1. An area onshore and offshore (A) of approximately 60 km by 65 km for which information has been gathered on geological features which might have relevance to a repository safety assessment, using exist-

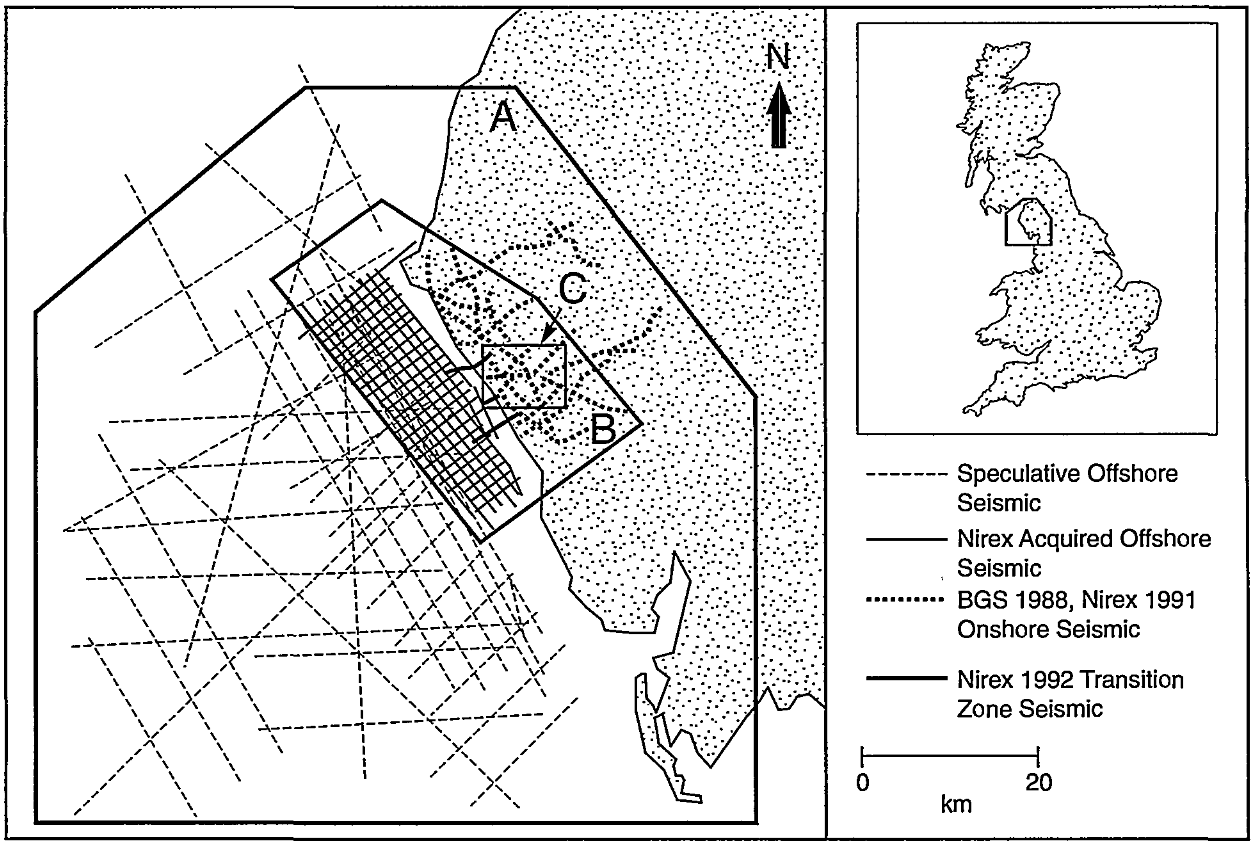

Figure 26.1. Investigation areas showing location of seismic surveys.

ing published sources of information and commercially available offshore seismic survey data. Additional data, including structural geological data relevant to seismic hazard studies, were collected from wider areas.

2. An area (B) of approximately 20 km by 30 km within which geological features have direct relevance to the repository. Within this area Nirex has commissioned new geological, geophysical and hydrogeological investigations. These investigations have been supplemented by study of data from past mining activities.
3. An area (C) immediately around the potential repository covering an area of approximately 50 km^2 and within which all the Nirex deep boreholes are located.

26.2.3 Regional Surveys

The extent of the regional geophysical surveys commissioned or acquired by Nirex is shown in Figure 26.1. These surveys have included some 1,950 line kilometres of seismic reflection, both onshore and offshore, and 8,500 km of airborne magnetic and radiometric surveys. Gravity data has been collected along many of the seismic lines. Geological mapping has been carried out by the British Geological Survey and regional surveys of near-surface hydrogeological features have been commissioned, as have remote seismic studies. Monitoring of springs, river gauging and meteorological observations are continuing. A programme of work to characterize the Quaternary deposits of the area has commenced.

26.2.4 Boreholes

By December 1994 Nirex had drilled twenty one deep boreholes (Fig. 26.2). Many of these were around 1,000 metres deep, with the deepest, Borehole 2, extending to 1,950 metres depth. Several phases of drilling have been completed, namely:

- An initial pattern of boreholes (Boreholes 1, 2, 3, 4, 5, 7, 10, 11, 12 and 14) to obtain an understanding of

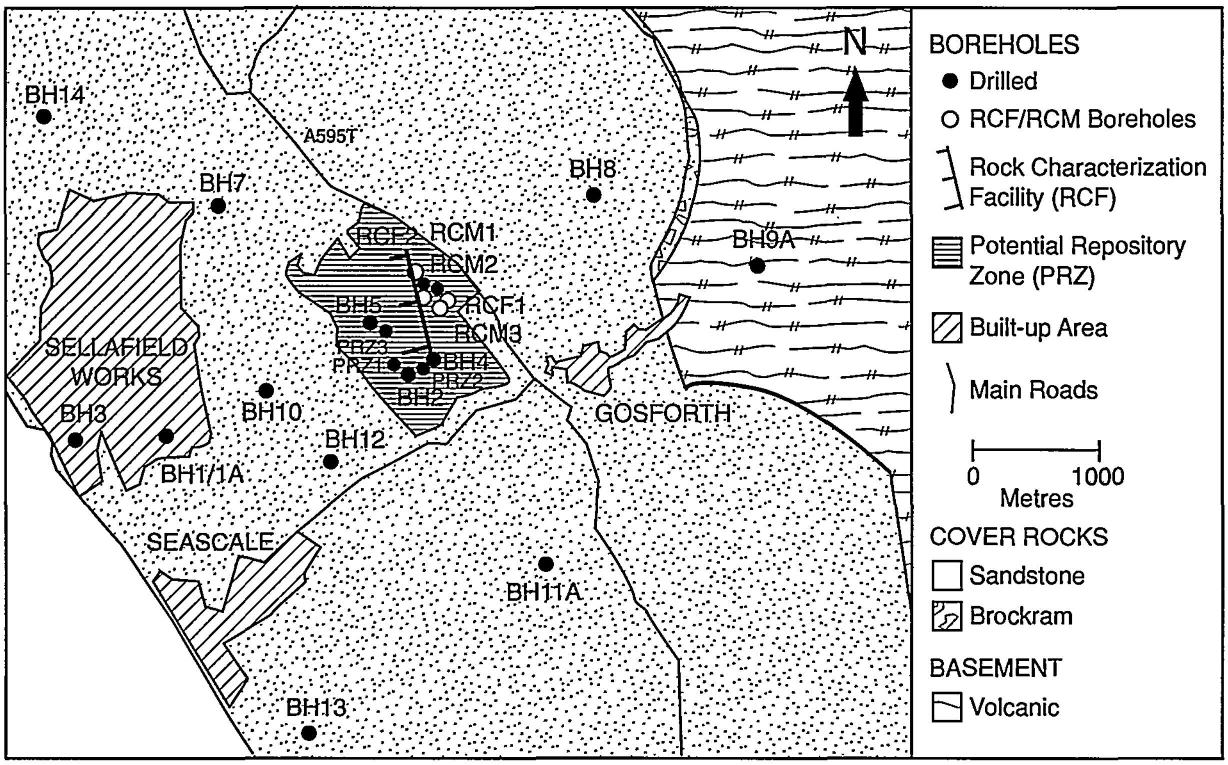

Figure 26.2. Schematic location of Nirex boreholes within the Sellafield site area, December 1994.

the regional geological and hydrogeological setting of the site.

- A subsequent series of boreholes to investigate specific aspects of the site. Boreholes 8 and 9 have been drilled in the upper part of the catchment to investigate groundwater recharge and Borehole 13 has investigated the area to the south where the thickness of the Permo-Triassic sequence increases markedly.
- A series of six boreholes (RCF1 to 3) and RCM1 to 3 have been drilled in the area of the proposed Rock Characterization Facility to characterize the ground in advance of underground excavation and to permit installation of a groundwater monitoring system close to the proposed shafts (Nirex, 1994b).
- Inclined boreholes are being drilled to characterize parts of the potential repository zone. Boreholes PRZ2 and 3 are completed; PRZ1 has still to be completed.

Two boreholes, drilled by others several decades ago for mineral exploration purposes, have been instrumented to supplement the groundwater monitoring system.
The majority of the drilling carried out by Nirex has been to obtain continuous core which is used for detailed characterization of the rock penetrated. Geophysical logging is also carried out to determine rock properties and particularly to provide information on the characteristics of the fractures which occur in the rocks.

Hydrogeological testing is carried out in the boreholes to determine groundwater pressures and the hydraulic conductivity of the rocks, that is, their ability to transmit water. Testing is carried out during breaks in the drilling and after completion of drilling to investigate the hydraulic properties of the rocks at a range of scales (Fig. 26.3).

Sampling of groundwater and analysis of samples is routinely undertaken during drilling and subsequently. Special measures are taken to reduce the levels of contamination of the groundwater by drilling fluids and to quantify the extent of any contamination to permit the determination of groundwater chemistry.

The completed boreholes are also used for undertaking specialist testing programmes. Examples include cross-hole seismic tomography and cross-hole hydraulic testing. A major programme of pump testing to measure the responses of the groundwater system over a wide area to

Field Activity	Scale	Hydraulic Characteristics	Hydraulic Connections to Overlying Units	Calibration/ Validation	Transport Processes
Standard Well Testing	10m	●	●		
Borehole 2/4 crosshole	100-500m	●	Partially		
Fracture Network Testing	10-100m	●			
Short Interval Testing	1-5m	●			
RCF3 Pump Test	Up to km		●	●	
PRZ Monitoring Network	All		●		
Tracer Tests	10-100m			●	●
RCF	All			●	●

Figure 26.3. Types of hydrogeological testing.

pumping from a central borehole is currently in progress.

26.2.5 Long-Term Monitoring

Most of the boreholes have now been converted for long-term monitoring of groundwater pressures by the installation of Westbay multi-level groundwater monitoring systems. These systems permit measurements of groundwater pressures within selected sections of the boreholes. Many sections are now equipped with automatic logging systems which provide measurements of pressures at two minute intervals to a high level of precision (Nirex, 1994a).

Nineteen boreholes have Westbay strings installed, and a further borehole is equipped with an alternative monitoring system. Some of the monitoring strings are amongst the most complex and deepest instrumentation systems of their type ever installed.

The monitoring network is designed to establish baseline groundwater conditions and to provide the means for monitoring the response of the groundwater system to induced perturbations, such as from cross-hole testing, pumping tests and RCF shaft excavation.

26.3 SUMMARY OF RESULTS

26.3.1 Geology

The proposed repository host rock at Sellafield comprises the volcanic rocks of the Ordovician Borrowdale Volcanic Group. Within the potential repository zone, the top surface of volcanic rocks is at a depth of 400 to 600 metres, occurring beneath the immediately overlying Permian breccia, the Brockram (Fig. 26.4). This is in turn, overlain by the Triassic Sandstones of the Sherwood Sandstone Group. The top of the volcanic rocks dips to the west such that at the coast they are some 1,600 metres below the surface. On approaching the margins of the East Irish Sea Basin, the Sherwood Sandstone Group is underlain by a thicker sequence of Permian rocks comprising the St. Bees Shale, the St Bees Evaporite and the Brockram. These are in turn underlain by the Carboniferous Limestone which rests unconformably on the Borrowdale Volcanic Group rocks (Michie, 1996; Holliday and Rees, 1994).

The rocks have been subjected to numerous periods of faulting and folding during their geological history. The distribution of the various formations at depth and the locations of the faults which cut them have been defined primarily by interpretation of the seismic reflection data, calibrated by the deep boreholes and utilizing mine plan data for the area to the north of Sellafield. Structure contour maps covering areas A and B have been generated for all the major geological boundaries within the sequence (Nirex, 1993a, b). Within the potential repository zone, additional detail is now being added based upon further boreholes, seismic reflection surveys, cross-hole tomography surveys between sets of co-planar boreholes and complex analysis of existing vertical seismic profiling (VSP) data.

26.3.2 Hydrogeology

Much of the work being undertaken by Nirex at

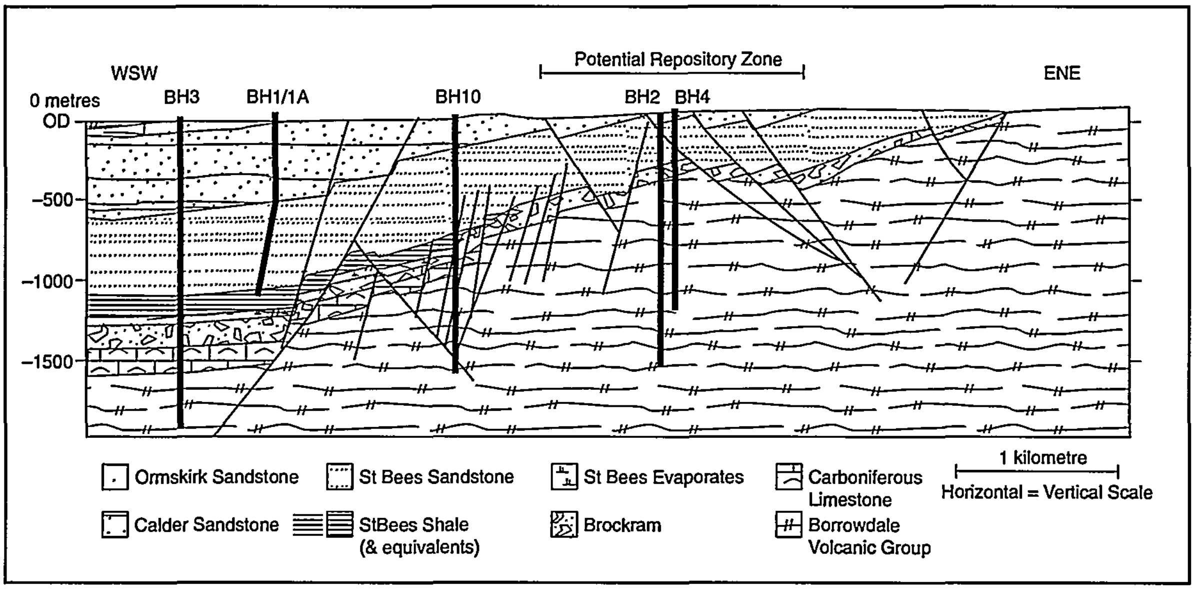

Figure 26.4. Schematic geological WSW-ENE cross-section through the Sellafield area.

Sellafield has been focused on determining the hydrogeology of the site, given that the flow of groundwater is recognized as the dominant mechanism for transport of radionuclides from a repository back to the surface (Nirex, 1993i; Black and Brightman, 1996).

Measurements have been made of hydraulic properties, especially heads and conductivities in boreholes. Having recognized that the flow of groundwater through the volcanic rocks is principally through fractures, effort has been directed towards characterizing those fractures which are both open and inter-connected such that they are hydrogeologically significant.

Geochemical studies of the groundwater have provided an independent record of past flow and mixing, and hence geochemical studies have featured prominently in the work undertaken (Bath, et al., 1996). Finally, numerical modeling has been extensively used to develop the understanding of the processes which are controlling-groundwater flow (Heathcote, et al., 1996).

The hydraulic conductivity of the rocks was initially determined in the boreholes using 50 metre long contiguous sections. Within the Borrowdale Volcanic Group the conductivity values are typically very low (Fig 26.5) with half the values measured over 50 metre lengths in the boreholes being less than $1x10^{-10}$ ms, including tests over faulted and fractured zones.

In order to identify the distribution of the hydrogeologically significant fractures in the parts of the sequence dominated by fracture flow, production tests have been carried out over the full lengths of boreholes, often in a series of stages. Inflow of water into the borehole is induced by imposing a drawdown in the order of 100 metres head and identifying flow zones by production logging. In many cases flows are so low as to preclude the effective use of spinner logging, zones only being identified from differential temperature and conductivity logging (Fig 26.6). Individual fractures, or groups of fractures, which carry flow are characterized by reference to the core logs and the borehole imaging geophysical logs.

Most of the fractures intersected by the boreholes have no detectable flow. Flowing fractures are therefore relatively widely spaced. Just over 150 have been identified in over 20,000 metres of drilling. Studies are currently being undertaken to characterise them.

Although fractures encountered in particular boreholes can make a major contribution to the conductivity of the rock mass in the immediate vicinity, it is the extent to which conducting fractures are connected which will determine groundwater flow in the Borrowdale Volcanic Group. Cross-hole seismic tomography has helped to define the geological structure between adjacent boreholes.

The extent of the connectivity is being examined using single borehole fracture network testing, cross-hole

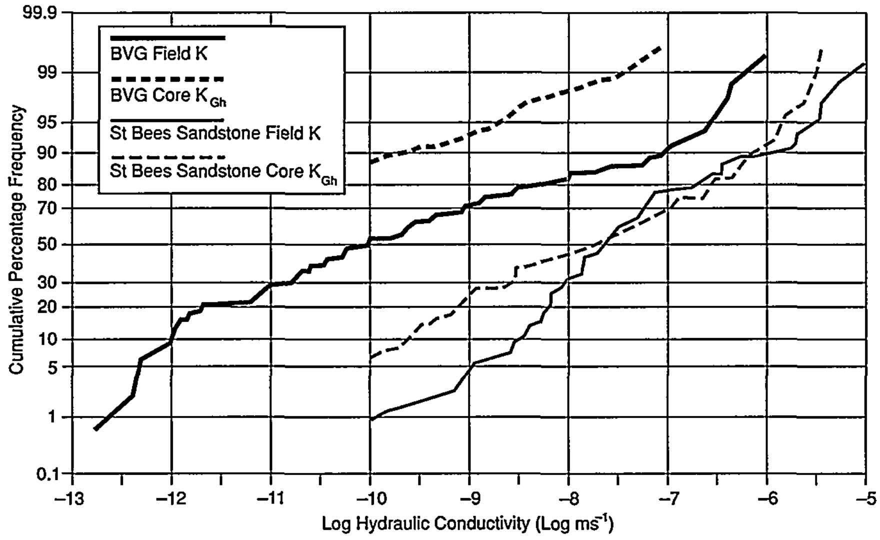

Figure 26.5. Summary of hydraulic conductivity values.

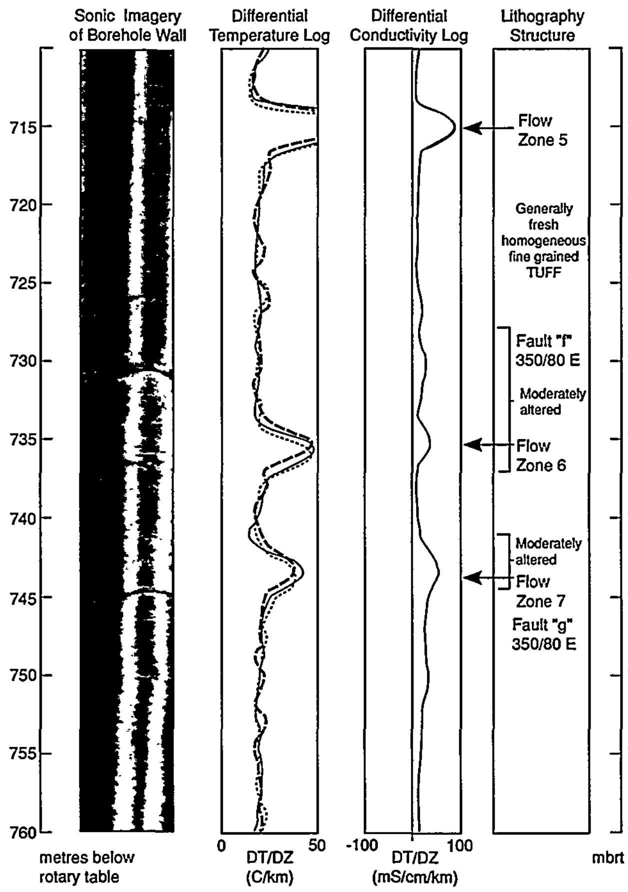

Figure 26.6. Identification of flow zones in Borehole 2.

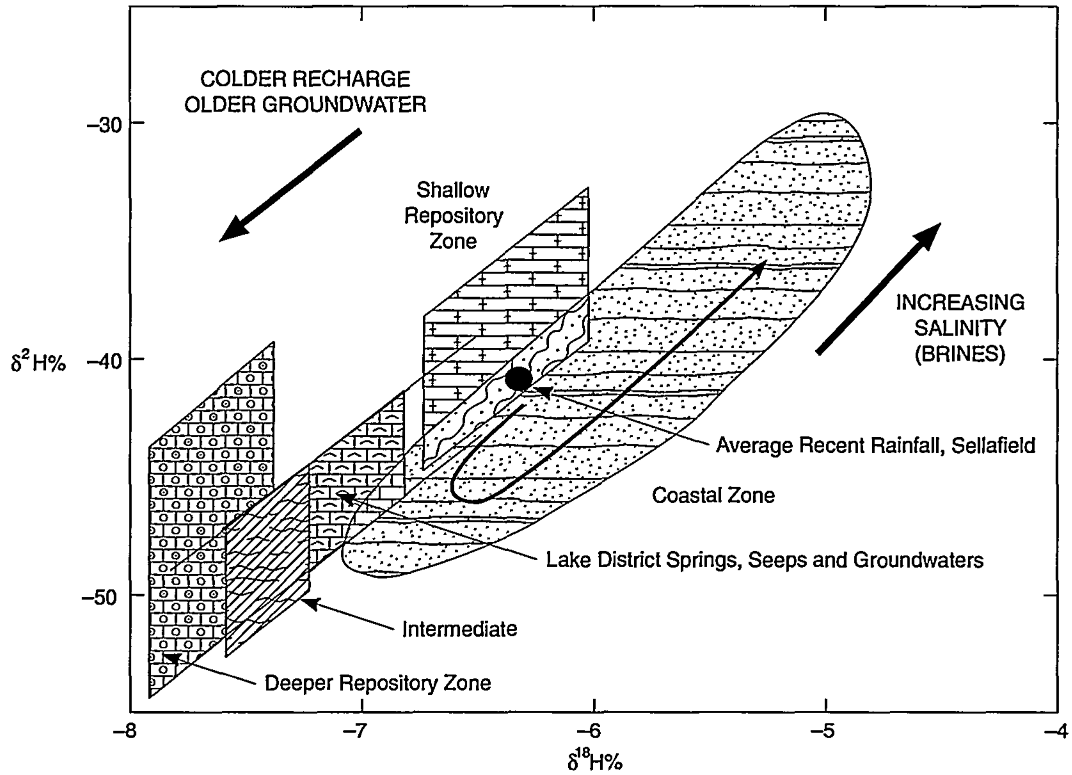

Figure 26.7. Stable isotopic discrimination between groundwaters with different origins.

hydraulic testing and by a series of large scale pump tests. Preliminary results from recent testing suggest that the fracture network may be less well connected than previously considered to be the case.

26.3.3 Geochemistry

Geochemical studies (Bath, et al., 1996) are carried out for three reasons: (a) to support the development of a conceptual model of the present-day hydrogeology, (b) to investigate how the groundwater system has evolved over time; and (c) to characterize the baseline hydrochemical conditions to support other studies.

Considerable progress has been made and the present dataset has contributed substantially to the construction of a conceptual model of the hydrogeological system, on which numerical modeling can be based. Some hydrochemical aspects of the conceptual model (particularly salinity sources and mixing zones) will provide specific tests of the adequacy of numerical modeling.

The palaeohydrogeology of the area is dominated by its location on the margin of the East Irish Sea Basin. The influence of basinal brines has been a feature of the deep sediments and the Borrowdale Volcanic Group basement in the west of the area for considerable geological time. Within the potential repository zone, a range of analyses including stable isotopic and noble gas temperature data for groundwaters in the Borrowdale Volcanic Group basement suggests that the waters at depth are clearly distinguishable from the shallow groundwaters and from modern rainfall (Fig. 26.7), and that the deeper waters probably have long residence times. This is a consistent pattern shown by several independent determinants and studies (Bath, et al., 1996).

26.3.4 Hydrogeological System

The current conceptual model of the hydrogeological system is illustrated in diagrammatic form in Figure 26.8. The three component parts of the system: the Irish Sea Basin Regime, the Hills and Basement Regime, and the Coastal Plain Regime are essentially as defined in mid 1992 (Nirex, 1992), although greater confidence in this model has been obtained with the availability of geochemical data.

The conceptual model and its evolution are supported, not only by geological, hydrogeological and geochemi-

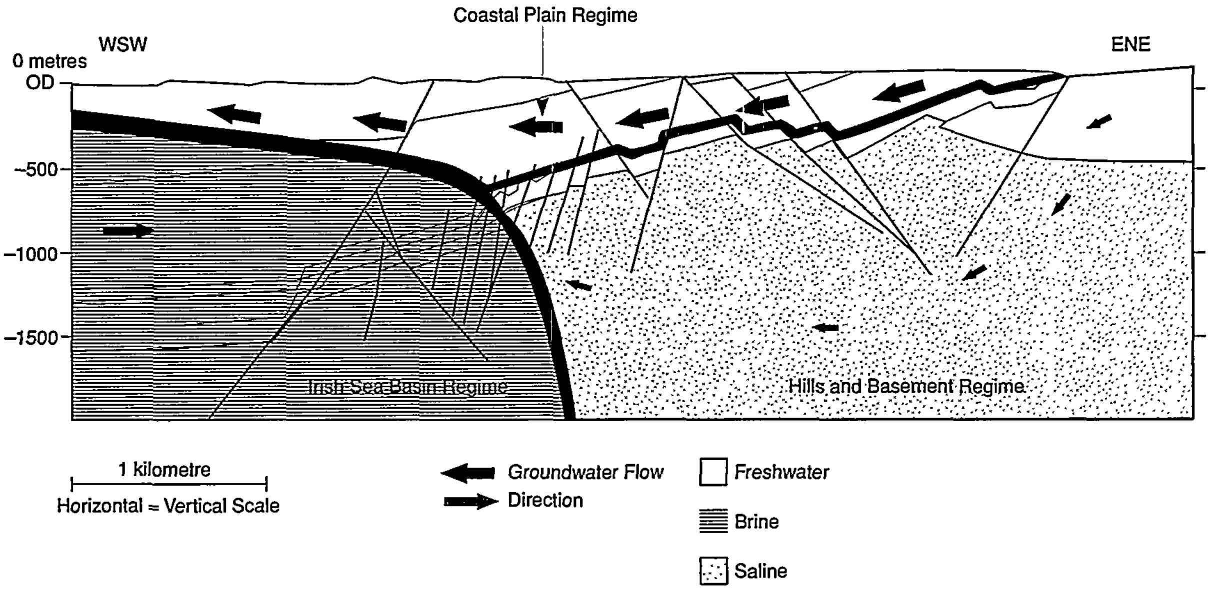

Figure 26.8. Current conceptual model of the groundwater system in the Sellafield area.

cal studies, but also by a range of numerical modeling studies which have served to examine and test a series of concepts concerning the behavior of individual components of the total system (Heathcote, et al., 1996).

26.4 FURTHER STUDIES

The investigations carried out to date from the surface are providing Nirex with a good understanding of the geological and hydrogeological conditions at Sellafield as they affect the decision on whether or not the site will be suitable for the construction of a repository to meet the stringent regulatory safety targets. Further investigations from the surface are in progress or are planned to further characterize specific features of the site. In particular, the pump test in Borehole RCF3 and the continued monitoring of groundwater conditions utilizing the installed instrumentation system are important components of this forward programme.

Investigations from the surface are, however, unable to resolve all the remaining uncertainties regarding the characteristics of the site. Nirex, in line with similar agencies in other countries, considers that a phase of investigations carried out underground from a suitably constructed experimental facility is a logical and essential extension to characterization from the surface. For this reason, Nirex has applied for planning permission to construct an underground Rock Characterization Facility (RCF) at the Longlands Farm site at Sellafield. This application is the subject of an appeal by Nirex against refusal, by Cumbria County Council, of planning permission.

26.5 CONCLUSIONS

The following conclusions are drawn from the investigations carried out at Sellafield:

1. An extensive programme of investigations has been carried out by Nirex. Various independent reviewers have commented on the high quality of the work undertaken.
2. The geological succession and structural geology of the site has been determined in significant detail.
3. Cross-hole seismic tomography has demonstrated that the geological structures can be mapped between boreholes. This observation is providing added confidence regarding the definition of the geological structure within the Borrowdale Volcanic Group and its influence on the hydraulic conductivity of the rock mass.
4. Preliminary quantitative assessments have been made of the distribution of hydraulic conductivity values in all the major hydrogeological units. Values measured in the Borrowdale Volcanic Group are typically low.
5. A limited number of individual fractures has been identified in the Borrowdale Volcanic Group along which groundwater flows. These fractures form a network which controls the flow of water through the rocks. Nirex is currently assuming that the fracture

network is well connected. Preliminary results from recent testing suggest that the fracture network may be less well connected than previously considered to be the case.

6. Geochemical and isotopic analysis of groundwater samples have assisted the development of the hydrogeological conceptual model and are helping to give some indication of the age and provenance of the groundwater within the Borrowdale Volcanic Group rocks. Evidence is suggesting that the deeper groundwater in the potential repository zone is old.
7. Good progress has been made with the investigations, and with the interpretation and modeling studies which follow on, to determine whether or not the site is suitable as a potential repository. However, much work remains to be done to resolve uncertainties and to develop confidence in the understanding of the site and the models which are constructed to represent its behavior. Construction of an underground Rock Characterization Facility forms a logical and essential continuation to the investigations carried out from the surface to progressively reduce uncertainty and to provide the information necessary to determine the suitability of the site for construction of a deep radioactive waste repository to meet the stringent regulatory safety requirements.

REFERENCES

Bath, A. H., R. A. McCartney, H. G. Richards, R. Metcalfe, and M. B. Crawford, Groundwater chemistry in the Sellafield area: a preliminarty interpretation, Quarterly Journal of Engineering Geology, 29, S39-S57, 1996.

Black, J. H., and M. A. Brightman, Conceptual model of the hydrogeology of Sellafield, Quarterly Journal of Engineering Geology, 29, S39-S57, 1996.

Heathcote, J. A., M. A. Jones, and A. W. Herbert, Modelling groundwater flow in the Sellafield area, Quarterly Journal of Engineering Geology 29, S59-S81, 1996.

Holliday, D. W. and J. G. Rees, Advances in West Cumbrian Geology resulting from the Nirex Investigations around Sellafield; Convenors' Report, Proceedings of the Yorkshire Geological Society, 50, Part 1, 1994.

Michie, U. McL., The geological framework of the Sellafield area and its relationship to hydrogeology, Quarterly Journal of Engineering Geology, 29, S13-S27, 1996.

Nirex, Sellafield Hydrogeology, Nirex Report 268, 1992.

Nirex, Lower Paleozoic rocks of the Sellafield Area, December 1993 Update, Nirex Report 515, 1993a

Nirex, The early Carboniferous (Dinantian) rocks of the Sellafield Area, December 1993 Update, Nirex Report 516, 1993b.

Nirex, The Permo-Triassic rocks of the Sellafield Area, December 1993 Update, Nirex Report 517, 1993c.

Nirex, Jurassic to Neogene structural evolution of the Sellafield Area, December 1993 Update, Nirex Report 518, 1993d.

Nirex, Quaternary Geology of the Sellafield Area, December 1993 Update, Nirex Report 519, 1993e.

Nirex,, The Geological Structure of the Sellafield Area, December 1993 Update, Nirex Report 520, 1993f

Nirex, A review of recent non-seismic geophysical studies in the Sellafield Area, December 1993 Update, Nirex Report 521, 1993g.

Nirex, The Geology and hydrogeology of the Sellafield area, Interim Assessment, December 1993, Nirex Report 524, 1993h.

Nirex, Nirex deep waste repository, Scientific Update 1993, Nirex Report 525, 1993i.

Nirex, Sellafield Geological Investigations, Interim assessment of hydrogeological data, December 1994, Nirex Report No. S/94/006, 1994a.

Nirex, Sellafield Geological Investigations, Summary geological logs for RCF and RCM Boreholes, December 1994, Nirex Report No. S/94/008, 1994b.

Nirex, Review Panel, Annual Report, 1994, Nirex, December 1994.

The Royal Society, Disposal of Radioactive Wastes in Deep Repositories, Report of Royal Society Study Group, 1994.

RWMAC, Fourteenth Annual Report of: The Radioactive Waste Management Advisory Committee, HMSO, London, June 1994.

Sutton, J. S., Hydrogeological testing in the Sellafield area, Quarterly Journal of Engineering Geology, 29, S29-S38, 1996.

CHAPTER 27

HIGH-LEVEL RADIOACTIVE WASTE MANAGEMENT IN THE UNITED STATES BACKGROUND AND STATUS: 1996

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27.1 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 deep mined geologic repository. At this time, the United States program is investigating a single site, although in the past, the program involved successive screening and comparison of alternate locations. The United States civilian reactor programs do not reprocess spent fuel; the high-level waste repository will be designed for the emplacement of spent fuel and a limited amount of vitrified high-level wastes from previous reprocessing in the United States. The legislation enabling the United States program[1] also contains provisions for a Monitored Retrievable Storage facility, which could provide temporary storage of spent fuel accepted for disposal, and improve the flexibility of the repository development schedule.

Yucca Mountain is a mountainous ridge located in the southwestern United States (Fig. 27.1) in the southern Great Basin, the largest subprovince of the Basin and Range physiographic province of the United States. The Basin and Range province is that area of southwestern North America that is characterized by more or less regularly spaced subparallel mountain ranges and intervening alluvial basins formed by extensional faulting. The regional climate of the southern Great Basin is typically hot and semi-arid. Generally, the geology of the province can be described as a late Precambrian and Paleozoic continental margin assemblage that has been complexly deformed by late Paleozoic and Mesozoic orogenies. Western portions of the province are broadly overlain by Cenozoic volcanic rocks; the distinctive physiography is largely a product of the most recent phase of extensional deformation. The alluvial basins are characterized by low rainfall, high evapotranspiration, ephemeral streams and closed hydrologic systems, evidenced by the absence of drainage external to the basins[2]. Characteristics such as these were important waste isolation considerations in the selection of Yucca Mountain for site characterization.

The repository design concept is a mined excavation at a depth of approximately 300 meters below the crest of Yucca Mountain and at a distance of approximately 300 meters above the regional groundwater table. The site is in silicic volcanic rocks, comprising alternating layers of welded and non-welded volcanic tuffs. The non-welded tuffs underlying the proposed repository horizon contain layers that are extensively zeolitized. The strategy for waste isolation relies on both engineered and natural barriers to provide defense in depth. The strategy for long term waste isolation places primary reliance on, and takes advantage of, the natural barriers, which include the aridity of the site, the unsaturated character of the host rock, and the deep regional water table. All indications are that these geologic conditions have been both spatially and temporally stable for many millions of years.

The United States high-level waste disposal program is managed separately from activities related to the management of transuranic waste from national security activities or commercially-generated low-level wastes. The transuranic waste program in the United States is also pursuing development of a mined geologic repository for disposal[3]. The Waste Isolation Pilot Plant, located near Carlsbad, New Mexico, is constructed in a salt formation.

27.2 LEGISLATIVE BACKGROUND

The high-level waste disposal program in the United States evolved through several different approaches between 1955 and 1982. In 1955, the National Academy of Sciences was asked to recommend a strate-

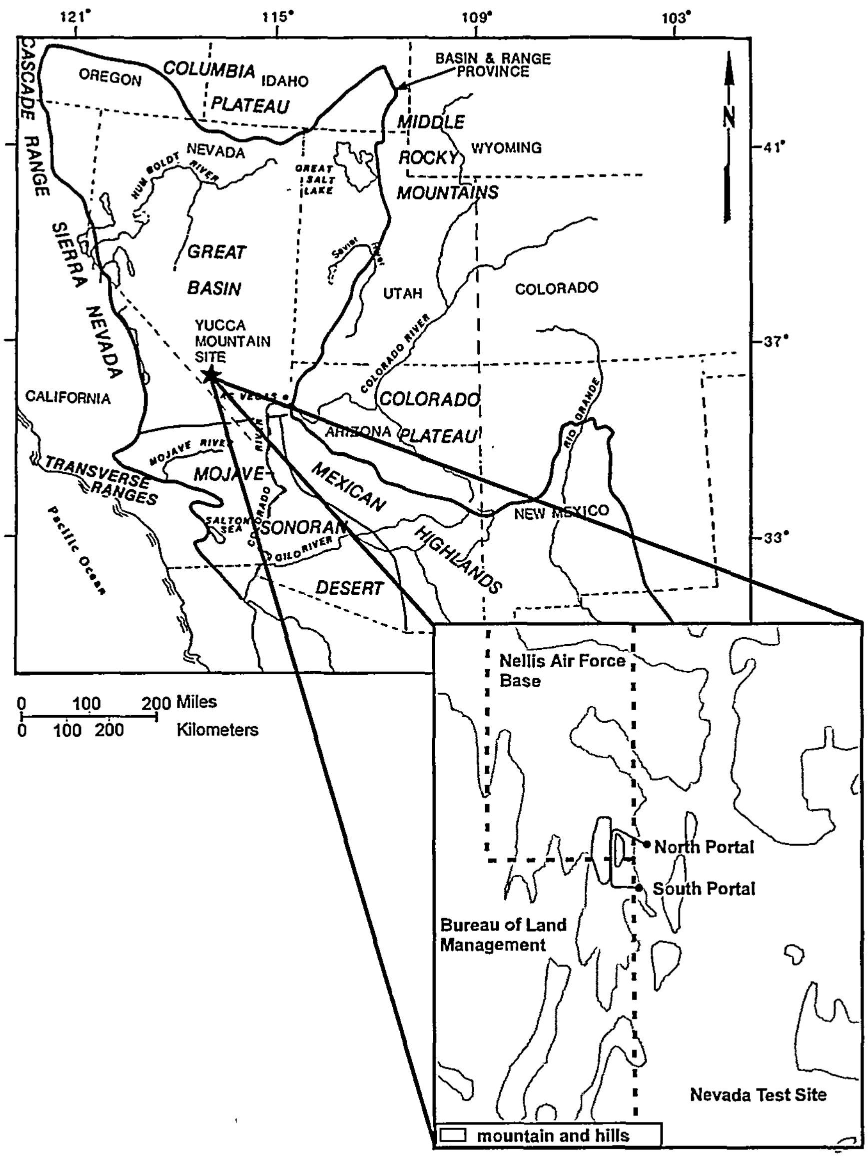


Figure 27.1. Location map for Great Basin physiographic province and Yucca Mountain Site.

gy for the disposal of liquid chemically hazardous radioactive wastes resulting from the reprocessing of spent nuclear fuel. The National Academy of Sciences recommended disposal in salt formations, to take advantage of the evident geologic stability and isolation from water. Subsequently, the United States program recognized the advantages of other rock characteristics, and began a program of national screening in the late 1970s. In 1982, the United States Congress passed the Nuclear Waste Policy Act (the Act) [4], setting forth an integrated plan for the disposal of commercially-generated spent fuel and high-level wastes from military reprocessing . The Act established responsibilities, schedules, and a funding mechanism whereby the users of nuclear power would pay for the disposal of the generated wastes. A portion of the wastes generated through various United States defense programs are included in the provisions of this law.

Under the Act, the United States Department of Energy was assigned responsibility for management of the civilian high-level waste program, the United States Nuclear Regulatory Commission was assigned responsibility for approving or disapproving a license to construct a repository and amendments to construct, operate, and close it, and the United States Environmental Protection Agency was assigned responsibility for promulgation of generally applicable standards for protection of the gen-

eral environment from offsite releases from radioactive material in repositories. The United States Environmental Protection Agency promulgated its standard[5] in 1985. This standard is not explicitly health based; the postclosure behavior of a repository licensed under this regulation would be judged against total cumulative releases at the end of a 10,000 year period. In 1987, a court challenge, based to some degree on inconsistency with other United States Environmental Protection Agency regulations, led to a remand of the standard. The United States Nuclear Regulatory Commission had already promulgated its procedural and technical requirements[6] by 1983. Those requirements, which implement rather than duplicate the United States Environmental Protection Agency release standards, address the concepts of multiple barriers and defense in depth, placing specific requirements for postclosure performance of the repository on components of both the engineered and natural subsystems.

Following completion of site characterization, the United States Department of Energy will submit initial documentation for a licensing hearing to authorize construction of the repository. Authorization of construction will be based in part on an understanding of the long term performance of the proposed repository. The United States Nuclear Regulatory Commission requirements embody a phased approach to construction and emplacement of high-level wastes in a repository. After sufficient construction to affirm that site conditions and underground excavation response are within the limits specified in the license to construct, the applicant submits documentation for a hearing for a license to receive and possess wastes. Amendment of the license to allow the repository to receive and possess wastes marks the first point in time that high-level wastes can be emplaced for disposal in the repository. With the approval of the United States Nuclear Regulatory Commission, up to ten metric tons of spent fuel could be emplaced for testing purposes during site characterization. Other than this limited amount of waste allowed for use by the Nuclear Waste Policy Act for purposes of site characterization (the provision was not invoked by the United States Department of Energy for Yucca Mountain), radioactive wastes are not permitted to be emplaced until this license is received. After operation of the repository and a defined period of monitoring, an application is submitted for a license amendment to decommission and then permanently close the repository.

The intention of the Program created by the 1982 Nuclear Waste Policy Act[4] was to characterize multiple sites and recommend sites for development as repositories. The Act envisioned the need for two repositories in the commercial waste program; accordingly, it directed the United States Department of Energy to undertake two repository characterization programs. The Act specified that the United States Department of Energy develop guidelines and prepare Environmental Assessments to be used as the basis for selecting each set of three sites to be characterized. The Act also explicitly required the preparation, and submittal for review by the United States Nuclear Regulatory Commission, of a Site Characterization Plan. It was the intent of the Act that following completion of the characterization of three candidate sites for each repository, Environmental Impact Statements would be prepared and serve as the basis for the recommendations of the sites for which the United States Department of Energy would apply for a license to construct a repository. In 1987, the United States Congress amended the Act. The Nuclear Waste Policy Amendments Act[1] selected Yucca Mountain, Nevada, as the single site to be characterized. The amendment also directed the United States Department of Energy to cease work on the second repository program.

The United States Congress recently passed a comprehensive Energy Policy Act[7] that contained provisions that probably will affect the regulations governing a repository at Yucca Mountain. That legislation required the United States Environmental Protection Agency, based upon, and consistent with the findings and recommendation of a study to be undertaken by the United States National Academy of Sciences, to promulgate public health and safety standards for protection of the public from releases from radioactive materials stored or disposed of in the repository at the Yucca Mountain site. These standards are to prescribe the maximum annual effective dose equivalent to individual members of the public. The issue of a dose-based standard for the United States high-level waste program dates back to the United States Environmental Protection Agency's own Science Advisory Board and the National Academy of Sciences[8] noting that such a standard would be appropriate for the United States Program.

The National Academy of Sciences study[9] provided recommendations as to whether a health based standard is reasonable, whether it is reasonable to assume that a system of postclosure oversight, based on active controls, will prevent a risk of breaching the repository, and whether it is possible to make scientifically sound predictions of the probability of human intrusion over 10,000 years. The United States Environmental

Protection Agency currently is in the process of developing a new compliance standard for the Yucca Mountain site that addresses these recommendations.

27.3 YUCCA MOUNTAIN SITE WASTE ISOLATION STRATEGY

The strategy for waste isolation for the Yucca Mountain site consists of reliance on a number of barriers, both natural and engineered, that either are attributes of the site or are engineered in a manner to complement the site attributes[2]. As water is the medium that can dissolve and transport solid wastes, the strategy takes advantage of the paucity of water at the site.

Yucca Mountain is a remote mountainous ridge located in the arid southwestern United States (Fig. 27.2), where rainfall averages approximately 15 centimeters per year. The water table in the vicinity of the Yucca Mountain site is deep, approximately 700 meters below the crest. Placing a repository at a depth of approximately 200 to 300 meters below the surface would leave a distance of several hundred meters between the repository and the water table. The repository would thus be in unsaturated rocks, with water held in place by capillary forces. The stratigraphy at Yucca Mountain consists of alternating layers of welded and non-welded volcanic tuffs. The welded tuff matrix is relatively impermeable; however, the rocks are fractured and will transmit water provided there is a sufficient source. The non-welded tuffs are porous and permeable; however, they tend to form capillary barriers at contacts with the welded tuff units and transmit significant quantities of water only when fully saturated. The repository would be located in a thick welded layer, overlain by a non-welded layer with a welded caprock (Fig. 27.3). Conceptually, this combination should be effective in limiting the amount of water that could eventually reach the emplaced wastes. The rock beneath the repository area includes layers that are conspicuously zeolitized, providing the potential for sorption to be effective in retarding the transport of some radionuclides.

It is intended that the engineered components of the repository complement the natural attributes of the site. The subsurface layout of the repository, as shown in Figure 27.4 and Figure 27.5, would comprise two inclined access ramps, two vertical ventilation shafts, and essentially flat-lying main and waste emplacement drifts.

The waste container is expected to function as the prin-

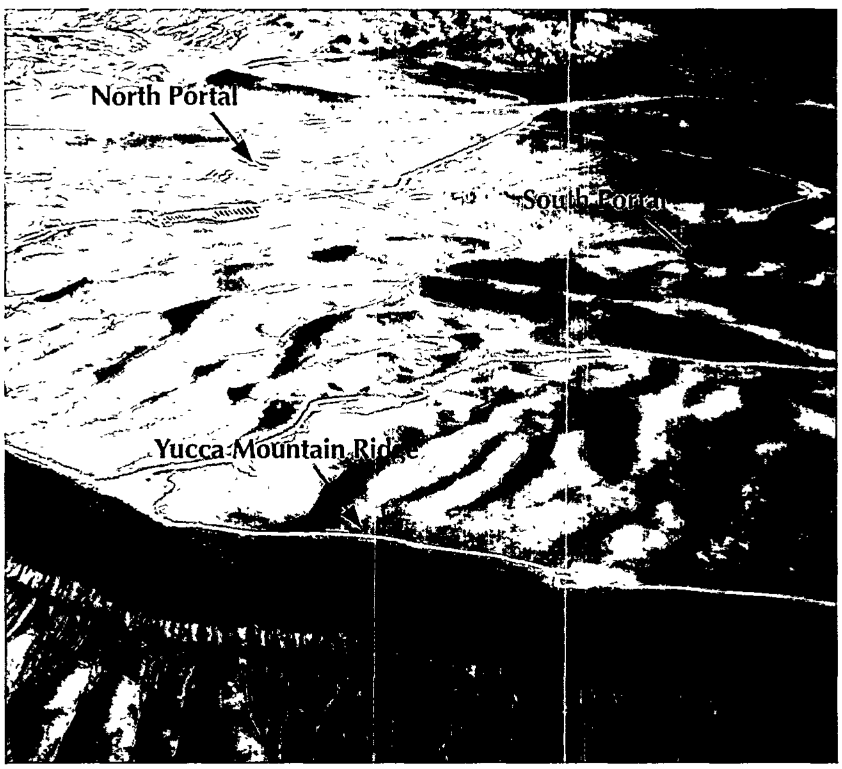


Figure 27.2. Aerial view of Yucca Mountain looking to the northeast.

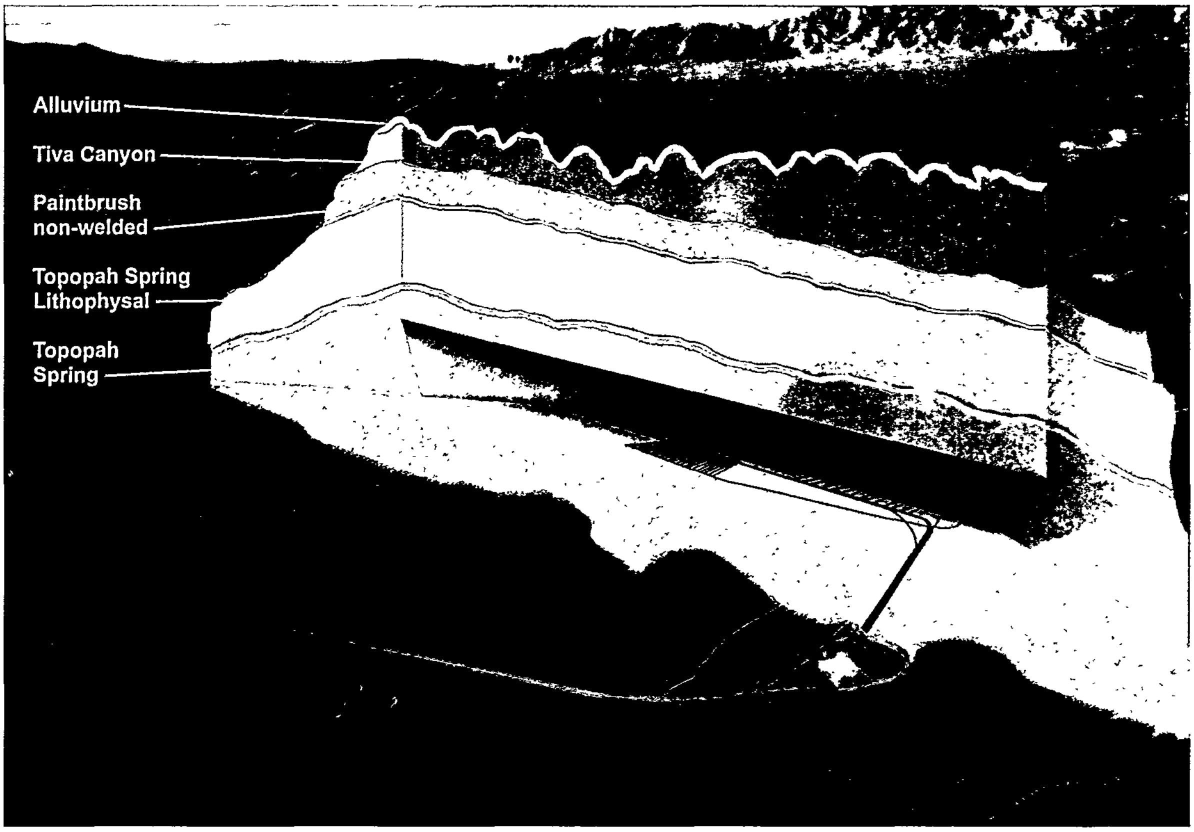


Figure 27.3. Artist's conception of a repository layout at Yucca Mountain illustrating stratigraphy of the mountain.

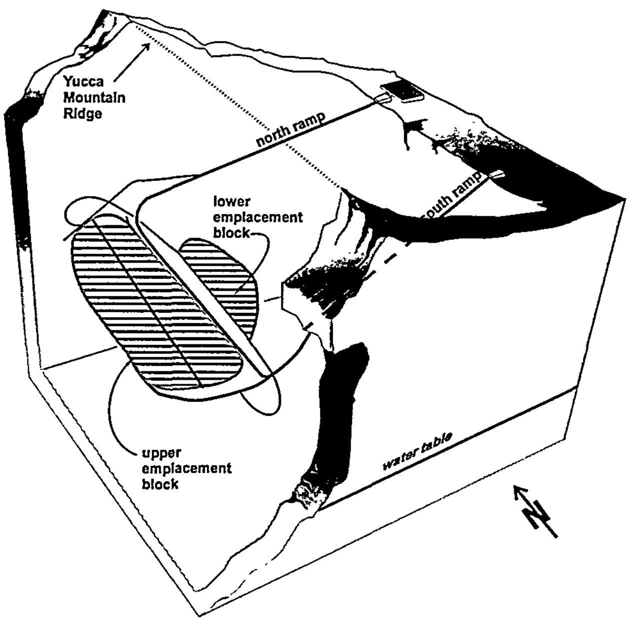


Figure 27.4. The subsurface of the repository will consist of about 250 kilometers (150 miles) of drifts, most of which will be used for emplacing the waste packages. The emplacement drifts will be divided into two areas. The upper block will be large and will lie to the west and slightly above the lower block.

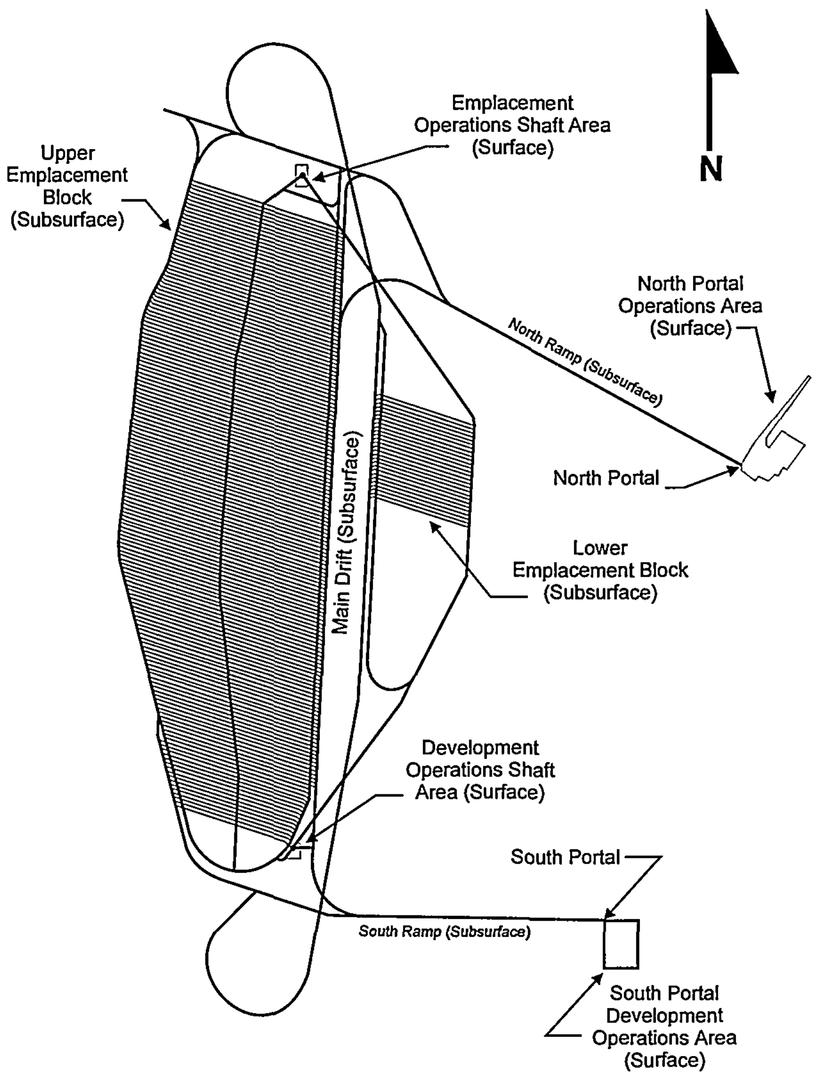


Figure 27.5. Overall repository site map.

cipal barrier to the release of radionuclides from the engineered barrier system. The barrier will be designed to provide substantially complete containment of the wastes for at least the time that heat and radiation emitted by the wastes are at their peak. This occurs early in the postclosure time frame, encompassing the period of approximately 300 to 1000 years following closure. The limited amount of water flowing through the unsaturated zone is expected to enhance the ability of the container to limit the release of radionuclides. Additionally, container materials will be chosen to be compatible with the geochemical properties of the water to prolong container life should water contact it. The waste form itself is also expected to be a barrier to the release of radionuclides. The low probability of early container failure and the small amount of water available are expected to limit the dissolution and leaching of radionuclides from the solid waste materials for at least several thousand years[10].

The components of the waste packages, in this relatively dry environment, are intended to confine the waste for thousands of years. The current container designs[11] are deliberately robust; the dual wall design uses a corrosion allowance material outer layer and a corrosion resistant material inner layer to form the walls of the waste packages. The heat from the waste packages is expected to keep the rock, immediately around the emplacement drift, relatively dry for hundreds of years, which should reduce the corrosion rate of the waste packages. The air gap between the container and the host rock is also expected to contribute to limiting the

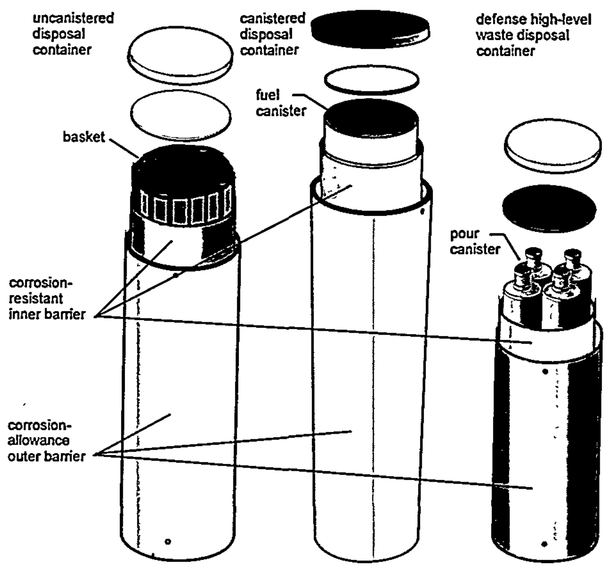
release of radionuclides. Because percolation rates are expected to be low, and because most water is expected to be tightly confined in the rock matrix, little water should cross this air gap. Limited water movement in fractures is possible; however, the amount of water that could potentially contact the containers is expected to be a small fraction of that contained in the rock. As the waste packages and emplacement drifts eventually cool, water could begin to seep back toward the waste packages, especially along fractures.

After the waste packages eventually corrode and deteriorate, and the engineered barrier function is degraded, the natural barriers will provide the primary means of isolation. The various rock layers at Yucca Mountain, due to low water content and movement, are expected to retard the movement of released radioactive material to the accessible environment. Finally, any radioactive material that eventually reached the water table beneath the repository would be diluted, further reducing the potential amounts that could reach the environment.

The current repository conceptual design assumes a relatively high emplacement density of 83 metric tons of initial heavy metal (MTU) per acre. The resulting emplacement scheme divides the subsurface facilities into two sets of waste emplacement drifts called blocks[11]. In this concept, the emplacement drifts would be spaced at 22.5 meters and emplacement drifts would range from about 250 meters to about 600 meters in length. The upper block would cover about 324 hectares, and could accommodate about 11,000 waste packages. The lower block would cover about 69 hectares, and could accommodate up to 2,400 additional waste packages. A total of about 12,000 waste packages are expected to be emplaced[11, 12]. Approximately 90 percent of the radioactive waste will be spent fuel in waste packages, each containing up to 9 metric tons of spent fuel. However, it is currently expected that more than one-fourth of the total number of waste packages will be high level waste, with each package containing approximately 2 metric tons of high level wastes.

The physical arrangement in which waste packages would be placed underground will have an impact on the environmental conditions in the emplacement drifts. After emplacement, heat will raise the temperature of the emplacement drift rock walls. The spacing between the emplacement drifts and the spacing between waste packages within the emplacement drifts will determine how hot the emplacement drift environment and surrounding rock will become. Conceptual designs have been developed for fuel assemblies that have been shipped to the repository in containers (canistered and defense high-level waste disposal containers) that can be placed into an overpack for subsequent emplacement for disposal in the repository. Conceptual designs also have been developed for fuel assemblies that arrive in a shipping container (uncanistered) and must be repackaged for subsequent emplacement for disposal in the repository (Fig. 27.6).

The waste packages would be mounted on rail cars that



Conceptual design characteristics for disposal containers.

	Uncanistered		Canistered		DHLW
	large	small	large	small	
Capacity (number of fuel assemblies or pour canisters)					
PWR	21	12	21	12	—
BWR	44	24	40	24	—
Canisters	—	—	—	—	4
Dimensions in millimeters					
Diameter	1629 (64")	1298/1265* (51" / 50")	1802 (71")	1531 (60")	1709 (67")
Length	5335 (210")	5335 (210")	5682 (224")	5647 (222")	3680 (145")
Weight in kilograms (approximate)					
Empty	31,000 (34 tons)	22,000 (24 tons)	31,000 (34 tons)	25,000 (28 tons)	13,000 (14 tons)
Loaded	47,000 (52 tons)	31,000 (34 tons)	65,000 (72 tons)	47,000 (52 tons)	22,000 (24 tons)

PWR - Pressurized Water Reactor BWR - Boiling Water Reactor
* larger diameter for PWR, smaller diameter for BWR

Figure 27.6. Conceptual designs address disposal containers for fuel assemblies not in canisters, fuel assemblies in canisters, and pour canisters.

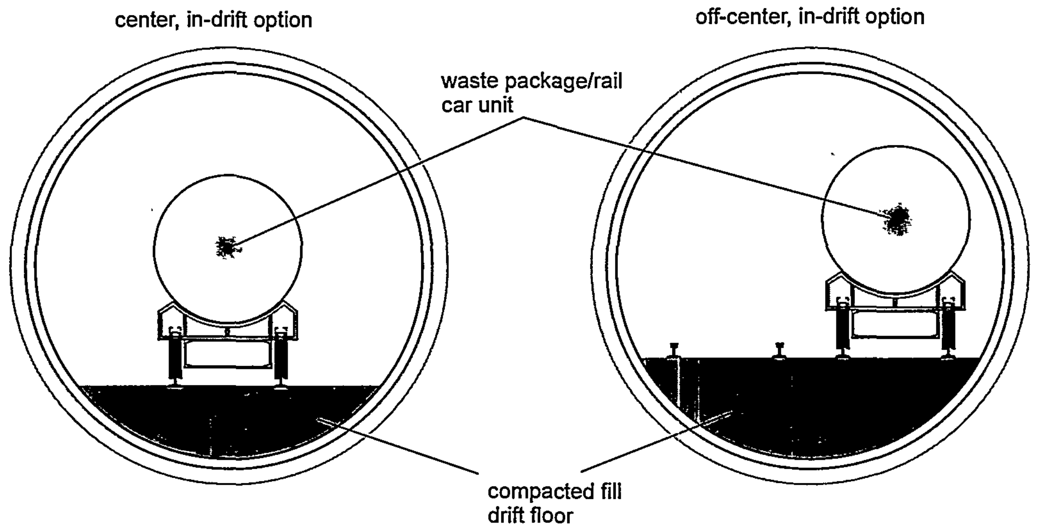


Figure 27.7. Conceptual design options for emplacement drifts: center in-drift, with a single set of rails for placing the waste package in the center of the drift; and off-center in-drift, with two sets of rails for placing the waste package to one side of the drift.

would be placed on rails within the emplacement drift. Once waste packages are placed in an emplacement drift, no human entry into that emplacement drift would be allowed under normal conditions. Two options, illustrated in Figure 27.7, are being studied for emplacement of waste packages within the emplacement drifts. One option, center in-drift, involves a single set of rails running down the center of the drift on which the waste package rail car unit would be placed. The diameter of this emplacement drift would be 5 meters. The other option, off center in-drift, would contain two parallel sets of rails. The rail car would be placed on one set of rails to the side of the drift, and the set on the other side of the drift would provide remote access along the drift for inspecting, monitoring and maintaining the drift, emplaced materials, and equipment. The diameter of this emplacement drift would be 5.5 meters.

Each combination of drift and package spacing, and waste package heat output, will result in a different overall thermal load. Corresponding temperatures will cause various changes in the repository rock and its water content, which can affect waste package performance. Potential changes in rock strength, water content and mineral composition depend primarily on the thermal loading. These changes are being investigated by both field and laboratory testing and by the use of computer modeling techniques. A repository design goal to not exceed wall rock temperatures of 200° C was established[11]. The current conceptual repository design uses a relatively high thermal loading that results in maximum emplacement drift wall temperatures of approximately 155° C at about 40 to 60 years after emplacement, if no ventilation is supplied to the drift after emplacement is completed[12].

As would be expected, this strategy is sensitive to disruptive processes and events, especially those that could modify the dry character of the site, which is central to the compliance arguments. Of particular importance to the strategy are climate changes and processes or events that could enhance infiltration. The effects of repository heating on the conditions of the rock mass are also of concern. A high thermal load may drive moisture away from the canisters for significant periods of time. Conversely, the temperature changes could lead to potentially irreversible changes in the hydrologic and geochemical properties of the zeolites in the rock mass.

Another issue of engineered barrier performance is related to the presence in the waste of gaseous radionuclides such as $^{14}CO_2$. The natural barriers at Yucca Mountain are not likely to be effective in containing gases that could be released through eventual failure of waste containers. Early waste package failure could lead to the release of quantities of $^{14}CO_2$ that could possibly violate the remanded standard for total curies released, even though there would be no significant associated health effects. Although the standard has been remanded, it is still used as a standard for comparison by the program until a new one is in place The waste packages likely could be designed to contain the

gases, but the lack of true health and safety protection makes it difficult to justify the costs.

27.4 RESULTS OF SITE CHARACTERIZATION COMPLETED TO DATE

The test program described in the Site Characterization Plan[2] included both surface based and underground test programs, the latter in a facility developed at the base of an exploratory shaft. Review comments on that plan from the Nuclear Regulatory Commission[13] and the United States Nuclear Waste Technical Review Board[14] were addressed[15] and, in so doing, the configuration was changed from shaft access to ramp access. The resulting facility design incorporated requirements to support a test program to address data needed to assess the role of both the engineered barriers and the natural barriers. Heater tests, ranging in scale up to room size, will be conducted over the next several years to investigate coupled thermal-mechanical-hydrological-geochemical processes, moisture movement, mechanical effects, and near field effects. Other underground tests underway or planned include construction tests, which examine excavation methods, measurements of the response of ground support systems, possible emplacement equipment tests, seal component tests, seismic tomography, and rock strength tests. Hydrology and transport will be studied through suites of tests such as large scale permeability tests, radial borehole permeability (gas and water) tests, hydrochemistry tests, mapping, diffusion tests, radionuclide transport tests (using non-radioactive tracers) and percolation tests, potentially above, at and below the repository horizon. The field experiments are complemented by laboratory tests, including, thermal, mechanical, hydrological and rock water interaction.

Observations of the natural system and site data collected since 1978 suggest that the natural system is robust, and that numerical models and calculations will be able to bound many of the uncertainties for radiological safety evaluations, leading to enhanced confidence in the performance of the site. No major unexpected conditions have been encountered; tunneling and testing are confirming the hypotheses on site conditions described in the 1986 Environmental Assessment and the 1988 Site Characterization Plan. Recent performance assessments[10] have led to increased confidence that a Yucca Mountain repository would contain and isolate radioactive waste and would meet a reasonable EPA standard. Disruptive events, such as volcanism or seismicity, are considered unlikely to adversely impact performance. The project recently completed an independent external expert elicitation on probabilistic volcanic hazards[16] that affirmed the project position on low (order of 10^{-8} per year) probability of a volcanic event penetrating the repository. An improved site and engineering data base supports performance assessment calculations that provide more realistic bounding conditions.

Site hydrologic models[17] indicate groundwater flux is likely to be limited and very low at the repository horizon, possibly with local exceptions. The results of modeling and field investigations support a conclusion that infiltration could be diverted laterally away from the repository horizon, owing to the distinctly different hydrologic properties at the contact between coarse-grained and permeable non-welded tuff, and underlying and overlying fine-grained and relatively impermeable welded tuff layers (Fig. 27.3). These hydrologic conditions have the potential to buffer the effects of increased infiltration that potentially could occur as a result of climatic changes. Exploratory studies facility tunneling progress has greatly increased the opportunity for underground observations and confirms constructability and geologic characteristics with no observable water seepage. Also, extensive underground drifting, which provides a greater opportunity for observation and sampling, has supplanted the originally proposed drilling program.

27.5 PLANNED FUTURE WORK

Program scientists and management today believe that a reduction in the scope of the characterization program from that originally proposed is justifiable and desirable. This reduced scope is supported both through increased understanding gained through progress in the characterization program and through realignment of licensing expectations with the information that can reasonably be obtained at different phases of the program[18]. The current understanding of the importance of the individual elements of the disposal system is better and more quantitative than at the time the Site Characterization Plan was written. Some of the uncertainties identified in the Site Characterization Plan are now known to be not as important as others. Performance assessments and modeling have identified the most important uncertainties and methods to bound resolution of other uncertainties without executing every activity of the extensive characterization program originally described in the Site Characterization Plan. What has been learned leaves fewer, but still technically important, questions to be answered about significant features and processes of the natural geologic, hydro-

logic, and engineered components that would be part of a potential Yucca Mountain repository.

In addition to new site characterization information and performance based analyses capable of evaluating total system performance, other developments such as updated repository and waste package conceptual designs, and considerations related to the change from a release standard to a dose or risk-based standard, with an as-yet unspecified regulatory time frame, have contributed to the need to refine the strategy for evaluating waste containment and isolation.

The updated strategy currently being developed maintains a number of fundamentals of the original strategy. The updated strategy continues to recognize the important role of the relatively dry conditions at Yucca Mountain, which contributed to the site originally being selected for characterization studies. The updated strategy also continues to recognize the geochemical setting provided by Yucca Mountain as important to determining the rate at which radionuclides may be released into the environment in the future, when containment by the waste packages is eventually lost.

The current program is structured around a series of major products leading to an eventual license application. One of the most visible major products in the near term is a Viability Assessment to be completed in 1998. The Viability Assessment has four components:

1. Design of critical parts of the repository, waste packages, and engineered barrier system;
2. Performance assessment that incorporates current knowledge of natural features, processes, and responses at Yucca Mountain, and that evaluates the long term performance of the total natural and engineered system;
3. Total System Life Cycle Cost estimate for the construction, operation, and closure of the repository; and
4. Licensing Plan to lay out tests, design activities, or other actions needed to complete an initial license application for submittal to the United States Nuclear Regulatory Commission.

The purpose of the Viability Assessment is to provide policy makers with an integrated view of a repository system, its estimated performance capabilities, and the associated cost and schedule. If policy makers accept and endorse that assessment, it is anticipated that resources will be committed to ensure continued progress toward the license application.

Current project planning also reflects the need to complete field studies and analyses to reduce uncertainty and enhance understanding of system performance to support the assessment of system safety needed for the license application. The larger technical questions identified in recent total system performance assessments[10] as key to evaluating repository and waste package performance are related to the following attributes of the system:

1. Rate of water seepage into the repository;
2. Integrity of waste packages (containment);
3. Rate of release of radionuclides from waste in the breached waste packages;
4. Radionuclide transport through engineered barriers and natural barriers; and
5. Dilution in the groundwater below the repository.

The refined waste isolation and containment strategy will also address what approach will be taken to gathering data and developing models to make better predictions of these attributes over time. As the repository generates heat and then gradually returns to ambient temperatures, it is expected that at least the first four of these attributes will be affected, changing their relative importance to system performance as a function of time.

In the absence of a definitive compliance standard for geologic disposal, the United States Department of Energy is defining waste containment and isolation for purposes of conducting the viability assessment in such a way as to be independent of the specific compliance measures that eventually will be promulgated[19]. Waste containment has been defined as: the near-complete containment of radionuclides by waste packages for several thousands of years. Isolation has been defined with a system-level safety goal as: an acceptable dose to a member of the public living near the site. Quantitative dose modeling results will be used to evaluate compliance with applicable standards; more attention will be paid to evaluating potential doses for the first ten thousand years. However, calculations will be carried out over longer times in order to provide qualitative insight into peak dose potential, and to support system enhancement studies.

The United States Department of Energy recognizes there are issues which cannot be completely resolved in the 1998 Viability Assessment. Scientific and engineering studies will continue to be conducted to guide con-

firmation of or revision to the basis for modeling performance of the repository system. There is an expectation that additional data and analyses will be required to support a license application. It also recognizes that, if a license is granted, confirmatory technical work will continue beyond the time of license application into the construction and operational phases of the repository.

27.6 CONCLUSIONS

Following amendment in 1987 of the legislation authorizing characterization of sites for a repository, the United States' high-level waste program focused on Yucca Mountain in the southwestern United States as the single site under consideration. The attributes of Yucca Mountain that made it technically attractive nearly twenty years ago continue to be the technical underpinnings of the strategy for long term waste containment and isolation. Significant progress has been made in the characterization of Yucca Mountain as a potential site for a mined geologic repository. Conditions encountered in the exploratory studies facility tunnel at repository depth are consistent with expectations of such a facility constructed in the unsaturated zone. Total system performance assessments of the long term behavior of a repository at Yucca Mountain continue to mature, and have provided significant guidance in helping define priorities in the test programs and design solutions for the engineered barriers.

The technical strengths of the Yucca Mountain site depend on limited water available to contact the wastes and a corresponding high potential for isolation of the wastes. Today, the United States regulatory approach to long term compliance is uncertain. While the United States Nuclear Regulatory Commission regulations are in place, the United States Environmental Protection Agency standards for disposal safety are remanded. Actions underway to develop a new standard for disposal safety are reopening issues fundamental to the structure of the regulatory approach.

The United States high-level waste program regulations were, in the past, based on a relatively long time frame of regulatory interest, 10,000 years, and assessed compliance against limits on total system releases at an accessible environment, located five kilometers from the repository. The National Academy of Sciences[9] recommendation that the United States adopt a dose based standard for postclosure compliance for a repository has raised issues relative to the regulatory time frame, dose and risk, the definition of the reference biosphere, human intrusion and the quantitative treatment of natural processes and events. Deliberation of these issues is expected to be intense and time consuming, and fundamental re-evaluation of the United States approach to long term compliance should not be unexpected.

Technically, the Yucca Mountain site remains attractive because of its great potential to isolate wastes. However, there are significant concerns about the ability to bring to closure a regulatory proceeding that could have to deal with what are unprecedented time frames in the context of regulation. The potential for a geologic disposal standard that could introduce a need to rely on dilution in a closed hydrologic basin to meet a dose based standard takes the United States high-level waste program full circle back to the promulgation of the United States Environmental Protection Agency standards for geologic disposal. In those proceedings, a dose based standard was considered to be an inappropriate policy that could increase overall population exposures by encouraging disposal methods that would enhance dilution of any radionuclides released[5].

The extent to which the Yucca Mountain site eventually can be shown to be in compliance with a regulation that is evolving amid questions about the very nature of the regulatory structure that has been the basis for selection of the site, and assessment of its performance for nearly twenty years, is a significant concern. This reassessment is occurring even as the geologic and engineering disciplines are beginning to evolve data sets that are unprecedented in depth, breadth and specificity for evaluating the Yucca Mountain site for its waste isolation potential.

REFERENCES

1 United States Congress. Nuclear Waste Policy Amendments Act of 1987. Public Law 100-203, 100th Congress, December 22, 1987.

2. Unites States Department of Energy. Site Characterization Plan. Yucca Mountain Site, Nevada Research and Development Area, Nevada. Office of Civilian Radioactive Waste Management, Report DOE/RW-0199, December 1988.

3. United States Congress. Waste Isolation Pilot Plant Land Withdrawal Act. Public Law 102-579, 102d Congress, October 30, 1992.

4. United States Congress. Nuclear Waste Policy Act of

1982. Public Law 97-425, 97th Congress, January 7, 1983.

5. United States Environmental Protection Agency. Environmental Standards for the Management and Disposal of Spent Nuclear Fuel, High-level and Transuranic Radioactive Wastes. (40 CFR Part 191). Federal Register, vol. 50, no. 182, Sept. 19, 1985.

6. United States Nuclear Regulatory Commission. Disposal of High-level Radioactive Wastes in Geologic Repositories. Code of Federal Regulations, Title 10, Part 60.

7. United States Congress. Energy Policy Act of 1992. Public Law 102-486, October 24, 1992.

8. National Research Council, Board on Radioactive Waste Management, Waste Isolation Systems Panel, United States National Academy of Sciences. A Study of the Isolation System for Geologic Disposal of Radioactive Wastes. National Academy Press, Washington, D.C. 1983.

9. National Research Council. Technical Bases for Yucca Mountain Standards. Committee on Technical Bases for Yucca Mountain Standards, Board on Radioactive Waste Management, United States National Academy of Sciences. National Academy Press, Washington, D.C., 1995.

10. Civilian Radioactive Waste Management System Management and Operating Contractor. Total System Performance Assessment - 1995: An Evaluation of the Potential Yucca Mountain Repository. B00000000-01717-2200-00136, Rev. 01, November 1995.

11. Civilian Radioactive Waste Management System Management and Operating Contractor. Mined Geologic Disposal System Advanced Conceptual Design Report. B00000000-0171705705-0002,. Rev. 00, March 1996.

12. McKenzie, Daniel.G., III, Kalyan K Bhattacharyya, and Alden Segrest. Conceptual Subsurface Facility Design for a High Level Nuclear Waste Repository at Yucca Mountain. 37th Annual Meeting of the Institute of Nuclear Materials Management. July 1996.

13. United States Nuclear Regulatory Commission. NRC Staff Site Characterization Analysis of the Department of Energy's Site Characterization Plan, Yucca Mountain Site, Nevada. NUREG-1347. August 1989.

14. United States Nuclear Waste Technical Review Board. First Report to the United States. Congress and the United States Secretary of Energy, March 1990.

15. Dennis, A. W. (editor). Exploratory Studies Facility Alternatives Study: Final Report. Sandia National Laboratories, SAND91-0025. September 1991.

16. Geomatrix Consultants, Probabilistic Volcanic Hazard Analysis for Yucca Mountain, Nevada, Draft Report, Volumes I, II, March 1996.

17. Wittwer, C., Chen, G., Bodvarsson, G.S., Chornack, M., Flint A., Flint L., Kwickless, E., and Spengler, R.. Preliminary Development of the LBL/USGS Three Dimensional Site Scale Model of Yucca Mountain, Nevada. Lawrence Berkeley National Laboratory Report. LBL-37356/UC-814. June 1995.

18. United States Department of Energy. Civilian Radioactive Waste Management Program Plan. DOE/RW-0458. December 1994.

19. United States Department of Energy. Civilian Radioactive Waste Management Program Plan. Draft Revision 1. DOE/RW-0458. May 1996.