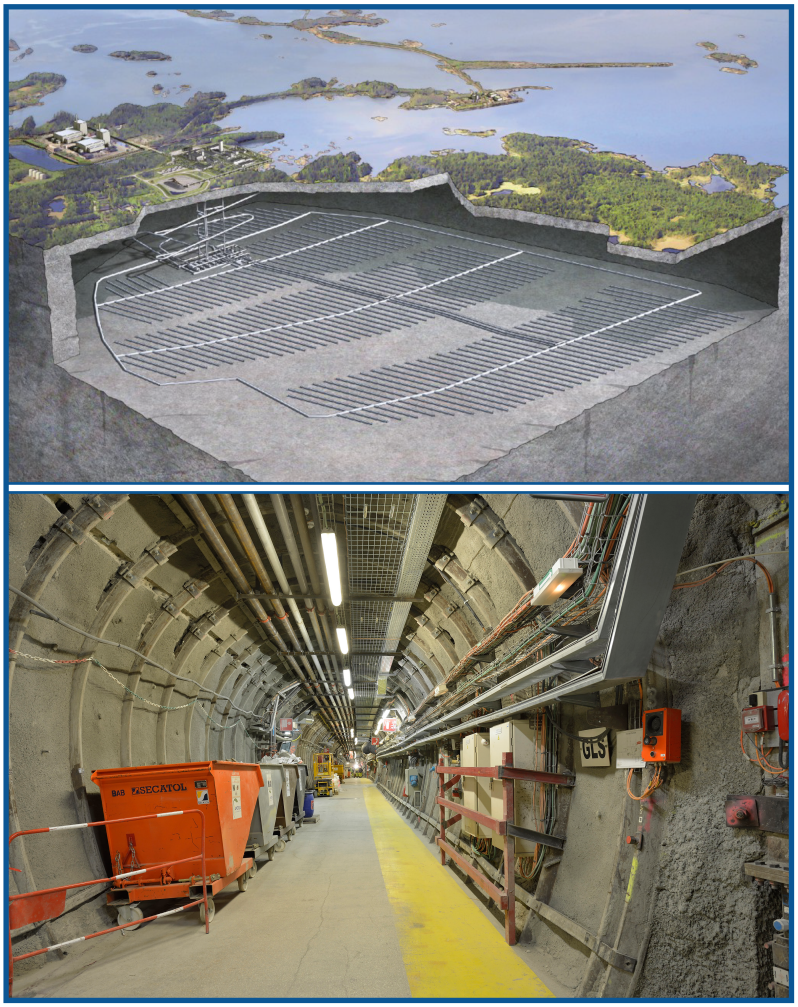
International Approaches for Deep Geological Disposal of Nuclear Waste: Geological Challenges in Radioactive Waste Isolation

Fifth Worldwide Review



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Prepared for: **U.S. Department of Energy**

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Chapter 1

Introduction

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1.1 Introduction and Motivation

An important issue for present and future generations is the final disposal of spent nuclear fuel (SNF) and high-level radioactive waste (HLW). Over the past forty years, the development of technologies to isolate both SNF and HLW as well as low- and intermediate-level nuclear waste (LILW) in underground rock and sediments has been found to be a challenging problem. In addition to waste from nuclear power generation and fuel processing, significant quantities of LILW are generated from various other sources, and while not as long-lived as spent nuclear fuel in terms of radioactivity, they also require management and ultimate disposal.

The first investigations on isolating radioactive waste underground were started in the United States in the 1960s, in a salt mine near Lyons, Kansas (Witherspoon and deMarsily, 1991), and in West Germany in 1965 using an underground research laboratory (URL) in the Asse salt mine (Langer et al., 1991). The first effort to study the problem of isolating HLW in granitic rock was initiated in 1977 at Stripa, Sweden, in an abandoned iron ore mine (Witherspoon and Degerman, 1978). In 1980, the Belgians started their HADES project at Mol with the construction of a vertical shaft, ~220 m deep, from which they developed a URL in the Boom clay (Neerdael, 1996). In 1983, Nagra in Switzerland started the Grimsel rock laboratory in crystalline rock some 450 m below the summit of the Juchlistock mountain in the Bernese Alps (McCombie and Thury, 1991). Since then, many URLs have been put into operation or are being planned, including international projects. For example, the Swiss waste program has developed a wide international scope of operations in their joint projects at Grimsel (in crystalline rock) and at Mt. Terri (in Opalinus clay).

The results from a review of a wide variety of investigations on the development of technology for geologic radioactive waste isolation from 19 countries were published in the *First Worldwide Review* in 1991 (Witherspoon, 1991). Subsequently, the results of investigations from 26 countries were published in the *Second Worldwide Review* in 1996 (Witherspoon, 1996). The results of investigations from 32 countries were summarized in the *Third Worldwide Review* in 2001 (Witherspoon and Bodvarsson, 2001). Finally, the latest results from 24 countries were compiled in the *Fourth Worldwide Review* in 2006 (Witherspoon and Bodvarsson, 2006).

Since publication of the last review report in 2006, radioactive waste disposal approaches have continued to evolve, and there have been major developments in a number of national geological disposal programs. Significant experience has been obtained both in preparing and reviewing cases for the operational and long-term safety of proposed and operating repositories. It has become apparent that disposal of radioactive waste is a complex issue, not only because of the nature of the waste, but also because of the detailed regulatory structure for dealing with radioactive waste. There are a variety of stakeholders affected, and there are a number of regulatory entities involved. In the U.S.A., federal government agencies involved in radioactive waste management in the United States include: the Environmental Protection Agency (EPA), the Nuclear Regulatory Commission (NRC), the Department of Energy (DOE), and the Department of Transportation. In addition, the states and affected Indian Tribes play roles as stakeholders in siting waste disposal facilities. In Europe, Finland has authorized construction of a final repository for SNF, and an application for a repository is under review in Sweden. France is planning to apply to build a final repository for high-level waste in the next few years. Although these three countries are frontrunners in the area of nuclear waste disposal, many more countries are in the process of addressing this issue.

Preparation and publication of this *Fifth Worldwide Review* on nuclear waste disposal is timely and important for assessing the lessons learned and developing future cooperation among countries. The Review provides an opportunity to share practical experiences on preparing for, developing, and documenting projects on nuclear waste disposal in the subsurface in various geologic environments in terms of requirements, expectations, and experience gained in judging the adequacy of cases for long-term repository safety. The Review helps identify potential issues that may arise as a repository program matures, and to understand the importance of safety cases in promoting and gaining societal and political confidence.

1.2 Objectives and Scope of the *Fifth Worldwide Review*

The overall objective of the *Fifth Worldwide Review* is to document the current state-of-the-art and major developments in a number of nations throughout the World pursuing geological disposal programs, and to summarize challenging issues and valuable experience that have been obtained in siting, preparing and reviewing safety cases for the operational and long-term safety of proposed and operating nuclear waste repositories.

The countries that are approaching implementation of geological disposal are increasingly focusing on the feasibility of safely constructing and operating their repositories in the short- and long term on the basis of existing regulations. Therefore, the scope of the Review covers current technical issues and challenges in safety case development, along with the interplay of technical feasibility, siting, engineering design decisions, and operational and post-closure safety.

1.3 Summary of the Current Status of the International Deep Geological Repository Programs

Brazil. The National Nuclear Energy Commission (CNEN) of Brazil is planning to give high priority to the construction of a final disposal facility for low- and intermediate level waste initially stored at nuclear power plants and fuel cycle facilities, as well as in intermediate of its Institutes and Centers. CNEN is going to begin developing details of the High Level Waste site selection regulation, as well as the establishing of an internal group of experts within the waste Management Coordination (COREJ). The COREJ group would be responsible for the complete licensing process of the disposal site. Additionally CNEN will establish another expert group in one of their research institutes with responsibility for the site selection, construction and operation of the future geological repository. It was concluded that the site selection process for radioactive waste repositories should avoid fractured zones, primarily because they may provide preferential pathways for migration of radionuclides to the environment, as well as adding to the complexity of models in the safety analysis.

Furthermore, there are minimum safety requirements established that must be met, such as: (i) the site must be characterized and modeled mathematically, (ii) the site cannot be located in an area where tectonic processes occur that could cause disruptions to the ground, (iii) the site must be far from facilities or activities (e.g., nuclear plants) which may likely adversely affect, or interfere with significantly, the environmental monitoring program; and (iv) the geologic terrain, saturated or not, should be able to prevent or retard the migration of radionuclides from the repository to groundwater. Such water should not flow readily to waterways, or aquifers potentially usable by the public, or into areas of highly permeable fractured rocks. Special attention should be given to the establishment of safety requirements for initial storage of radioactive waste from mining and processing of conventional ores containing radionuclides from the uranium and thorium series, since these storage facilities may be converted into final storage. The idea of constructing a final disposal facility in the Angra dos Reis region for the low level waste should be carefully analyzed.

Bulgaria. A comprehensive preliminary analysis has been conducted to elucidate the possibility of selecting a potential site suitable for the construction of a deep geologic HLW repository. The step-by-step analysis of Bulgaria's territory has been performed according to the methodological framework specifically developed in Bulgaria for selecting potential host rocks. Zoning of the Bulgarian territory has been carried out on the basis of exclusion criteria, corresponding to the specific geological and hydrogeological conditions of the country. An aggregated map was constructed at a scale 1:500,000, within which three regions of interest have been identified. Five prospective areas (each between 100 and 500 km^2) have been selected in these regions, which meet to the highest possible extent the basic preferable conditions, characteristics, and requirements for the construction of the geologic repository. Based on the set of 27 selection criteria, more than 30 suitable sites (each approximately 4-6 km^2) have been identified within the five prospective areas. At this stage of the preliminary site selection process, the data for the selection criteria have been collected and analyzed based on the information from prior investigations, mainly from oil and gas explorations. On the basis of multi-criteria comparative analysis five potential sites have been selected for further consideration: Varbitza, Kozloduy, Dekov, Komarevo and Zlatar sites, where the potential host rock is the Lower Cretaceous clayey marl; and Kozloduy site, where the potential host rock is Neogene (Miocene) clay. The main conclusion from the preliminary site selection analysis is that within Bulgaria, the possibility exists for the development of a deep geological HLW repository.

Canada. During the past decade, significant advances in the Canadian program for the long-term management of used nuclear fuel include dialogue and engagement with Canadians that has led the Nuclear Waste Management Organization (NWMO) to develop and recommend implementation of Adaptive Phased Management (APM). APM envisions that the nuclear used fuel would ultimately be placed within a multi-barrier deep geological repository excavated within a suitable sedimentary or crystalline setting with an informed and willing host community, and that the facility is safe such that it meets or exceeds regulatory requirements. The APM siting program was initiated in 2010 following two years of engagement with Canadians. By the end of 2012, a total of 22 communities (6 with potential sites in sedimentary rock and 16 in crystalline rock) had expressed interest in learning more about APM. Through a stepwise process, initial dialogue, engagement, and technical screening of these communities was conducted to ensure safety and assessment of well-being potential. As studies progressed, NWMO's engagement activities broadened into the surrounding region to include First Nation and Métis communities. As of 2016, nine communities remained in the APM siting process. NWMO's work program includes further learning in these communities and surrounding regions, further assessment of community well-being and partnership potential for the project, and more detailed site-specific studies and investigations of safety. Ultimately, one preferred site will be selected for the APM facility and will be subject to the necessary environmental assessment and regulatory approvals process to attain a site preparation and construction license. Confidence in the long-term safety of the multi-barrier repository design has been continually tested and refined by the NWMO through the APM technical program, which has three functional areas: Geosciences, Repository Safety, and Repository Engineering. Publically available safety case studies in crystalline and sedimentary rock settings have focused on illustrating long-term safety and the robustness of the multi-barrier repository concepts.

China. A three-phase long-term plan has been developed to guide R&D for geological disposal of high-level radioactive waste, with major milestones to build an underground research laboratory (URL) by 2020 and a national geological repository by 2050. The site for the URL will be selected from the potential sub-areas in Beishan, Xinjiang, and Inner Mongolia. Beishan, located in Gansu Province in Northwestern China, has been selected as the first priority site for China's HLW repository. Altogether 23 boreholes have been drilled in the Jiujing, Jijicao, Xinchang, Yemaquan, Shazaoyuan, and Suanjingzi sub-areas in Beishan in the period of 2000–2012. The results show that the rock mass is of high integrity, very low fracture density, very low hydraulic conductivity, and moderate in-situ stresses, indicating that the Gobi desert Beishan site has great potential for the construction of future geological repositories and a URL.

A multi-barrier concept has been proposed for the preliminary design of a geological repository in granite. The Gaomiaozi (GMZ) bentonite in Inner Mongolia has been selected as the buffer and backfill material. An experiment in a mock-up facility, to study GMZ bentonite properties under thermal-hydro-mechanical-chemical coupled conditions, is in operation. Comprehensive experimental studies on the mechanical characteristics of Beishan granite have been conducted as well. A series of theoretical models have been established to represent the behavior of Beishan granite under different stress and coupled conditions. A preliminary 3D discrete fracture network (DFN) modeling code with optimized algorithms has been developed to provide DFN models for mechanical stability evaluation and nuclide pathway analysis. Although progress has been made in many sectors, the Chinese program is still facing social, economic, scientific, technical, and engineering challenges. Continuous efforts will be concentrated on the Beishan site and its comparison with other sites, concept design of repository, design and construction of an underground research laboratory, and safety assessment, while other associated laboratory studies will also be conducted in the coming years.

Czech Republic. Deep geological repository (DGR) development in the Czech Republic is governed by a relatively tight schedule—2020 for the selection of two candidate sites, and 2025 for the selection of the final site—and represents, in terms of its multidisciplinary nature and the amount of funding, a unique and significant research challenge for the Czech Republic. The screening of geological conditions in the Czech Republic revealed that granitoids of the Bohemian Massif would provide the most suitable host rock mass for a deep repository. Extensive research commenced concerning the materials for the construction of the waste disposal containers, the various adsorbing and sealing materials, and other components to construct a future deep repository. The updated State Energy Concept envisages building new nuclear sources in the Czech Republic, which necessitated the enhancement of the capacity of a future repository and the re-assessment of all candidate sites in terms of space requirements. Extensive research of all aspects relating to the long-term safety of the entire waste disposal system is currently considered top priority. All the work involved in the project is aimed at meeting the set date of 2065 for the commencement of deep repository operation.

Finland. Since 2004, Posiva Oy, the company responsible for the final disposal of spent nuclear fuel of the Olkiluoto and Loviisa nuclear power plants in Finland, has constructed an underground rock characterization facility on the planned repository site in Olkiluoto, western Finland. This facility, called ONKALO, has provided an opportunity to carry out further site investigations, develop construction techniques, and test and demonstrate the engineered barrier system in the actual repository environment. As a result of these investigations and developments, application for a license to construct the encapsulation plant and geological repository was submitted in 2012. The Radiation and Nuclear Safety Authority in Finland (STUK) gave a positive review on the safety of the construction of the disposal facility in early 2015, and Posiva received the construction license in November 2015. An important aspect of the project is identification and assessment of the uncertainties related to the readiness for starting the implementation phase related to construction of the deposition tunnels and deposition holes according to the demanding requirements, installation of the buffer and backfill, demonstration of the mechanical strength of the disposal canister, and the thermal conductivity of the buffer. The next major milestone will be submission of the operation license application, scheduled for 2020. Posiva has already launched a project for compiling a successful application.

France. Research on solutions for disposal of high-level and intermediate-level long-lived radioactive waste has been ongoing since 1991. According to the Parliament's Planning Act of 2006, commissioning of the geological disposal of radioactive waste in the repository is planned by 2025. In 1998 the Meuse/Haute-Marne site was selected for licensing. The repositories will be located 500 m underground, in the Callovo-Oxfordian Argillite layer. The geological disposal project entered the industrial development phase in 2011. The site is expected to operate for about 120 years. French law requires that disposal be reversible, and therefore the waste packages must be physically retrievable. French law requires consultation via public debate to ensure that major projects effectively meet requirements. Andra, the French National Radioactive Waste Management Agency, aims to raise collective awareness of the existence of radioactive waste, and to provide a forum for rational, rather than polemical, discussion.

Germany. Taking into account that the German government decided to phase out nuclear energy for the industrial generation of electricity, Germany has been implementing a new approach for a disposal facility for heat-generating radioactive waste. (The categorization of the radioactive waste in Germany differs from standard international practice by dividing it into heat-generating waste and waste with negligible heat generation.) A location for the nuclear waste disposal site is to be sought which guarantees optimal safety for a period of one million years. After completion of underground exploration and comparison of sites, the selection of a site is expected in 2031. Commissioning is not expected before 2050 at the earliest. Germany currently is implementing three projects for storage and disposal of negligible heat-generating radioactive waste. The Asse nuclear facility, an abandoned salt mine, which was used as a disposal facility from 1971, is now decommissioned pursuant to the German Atomic Energy Act. The Konrad disposal facility, a decommissioned iron ore mine, is currently in the construction phase as a repository for negligible heat-generating waste. The Morsleben disposal facility (ERAM), a salt mine, which was used as a repository for radioactive waste from nuclear power plants in the former German Democratic Republic and the Federal Republic of Germany, is in process to be decommissioned. The Gorleben salt dome has been investigated since 1979 to assess its suitability as a disposal facility for radioactive waste. As part of a political decision, work at the Gorleben site has been reduced to mere maintenance since 2013 to keep essential underground workings open. Due to changes to legal frameworks, more research will be required in Germany, as well as a new search for a disposal facility. International collaboration is also indispensable for disposal facility research. It is also critical to address socio-technical issues, so that the process is transparent to interested and critical members of the general public and all stakeholders, and current scientific understanding of technical and social issues are communicated clearly to all.

Hungary. A surface-based geological research program in Hungary aiming at a HLW repository started more than two decades ago, and has led to identification of a potentially suitable geological formation at the southwest foot of the Mecsek Hills: the Boda Claystone Formation (BCF). The conceptual design of the Deep Geological Repository (DGR) was developed after the preliminary safety assessment, taking into account the inventory, heat load, packaging information, criticality, and radiology information. The current disposal concept considers a copper overpack on the canisters. The layout of the planned repository is based on disposal of canisters in vertical disposal holes drilled from disposal tunnels excavated within the BCF. The design of the encapsulation plant is based on the Swedish concept. The disposal tunnel system is planned for 500–800 m below ground surface. In the design, underground construction activities will be undertaken via vertical shafts. The disposal shafts will be connected to one another and to the service area by ventilation ducts and utility piping. The underground space will be constructed by conventional drill and blast methods. It is assumed that the large-section underground drifts will require rock bolts and a sprayed fibrous concrete lining with an average thickness of 10 cm, while for the small-section drifts (including the disposal drifts), rock bolts and a 5 cm thick sprayed concrete lining will suffice. However, due to numerous uncertainties, such as the lack of a defined back-end strategy for the management of spent fuel the disposal concept may be reassessed. In the borehole complex, geophysical, geotechnical and hydraulic measurements are performed. The screening of the space between the wells is carried out by surface-based geophysical methods, providing seismic reflection profiles.

The purpose of the surface-based exploration phase I is the selection of the target site and its general characterization. At the end of the current phase, the task is to designate within the 37 km^2 potential repository area the most appropriate 10–12 km^2 site. The research will continue on this site with the surface-based investigation phase II, at the end of which the 1–2 km^2 territory for the underground site and its associated surface facility area could be selected. Characterization will also take place in this phase. The surface-based investigation phase III aims at the preparation of the underground research laboratory. The Public Limited Company for Radioactive Waste Management (PURAM) intends to finish the site selection process by 2030, the planned end date of surface-based investigation. This site should include the potential location of surface and subsurface facilities and the URL as well.

India. The program of the development of a Deep Geological Repository (DGR) for high-level radioactive waste has been in progress in India for the last three decades. The program is mainly based on field and laboratory studies, which focus on the development of comprehensive databases of suitable types of host rocks available in India. The rock mass parameters coupled with waste characteristics provide inputs for the design of DGR and the analysis using commercially and specifically developed computer codes. A dedicated program is aimed at setting up an underground research facility for the development of methodology and technology related to site characterization, construction of disposal tunnels and pits, sealing and grouting of groundwater conducting zones, as well as the design of an engineered barrier system, waste transfer and emplacement technology, etc. Nationwide studies on granites spanning over 100,000 square kilometers have been undertaken. Such granitic bodies have been tested using state of the art monitoring technologies involving geographic information systems (GIS) and satellite based studies, geophysical investigations, and deep borehole drilling. A large amount of information on geological, structural, hydrogeological and geochemical parameters has been generated on samples retrieved from as deep as 600 m. A number of clay-sand admixtures are currently being evaluated for establishing the rock barrier function under the influence of the temperature field generated around the disposed waste overpacks. Granitic rock is under consideration for hosting a geologic repository 500 to 600 m deep, with an initial capacity of 10,000 overpacks and with provisions for further expansion. The Indian reference design is a 2-m-long overpack emplaced in a 5-m-deep pit, 0.85 m in diameter, in disposal tunnels of the repository with 50-cm-thick layers of highly compacted clay between the overpack and the rock, and the top of the disposal pit covered with a concrete plug. A field scale *in-situ* heater test, of six years duration, has been conducted in a gold mine in South India, to observe the evolution of the thermal field around disposed waste and changes in the strength and permeability of the rock mass, and to validate modeling results.

Japan. Since the 2011 Great East Japan Earthquake, tsunami, and reactor meltdowns, there has been increased public concern regarding major natural disasters causing accidents at nuclear facilities. To help rebuild public confidence required to site a nuclear waste repository, the Government of Japan has re-evaluated the technical feasibility of the geological disposal program based on state-of-the-art geosciences and recommended further promotion of the geological disposal program. The Nuclear Waste Management Organization (NUMO) has started to develop a comprehensive safety case that builds on both worldwide technical progress and understanding of relevant geological conditions in Japan. In cooperation with relevant organizations, NUMO has been developing key technologies required for the safe implementation of the geological disposal project since its establishment, which are being evaluated to provide the new safety case. This safety case will demonstrate flexibility to respond to a siting process that is still focused on volunteering and thus reflects the need to be able to compare alternative designs or siting options, which requires a performance assessment approach based on realism rather than conservatism. In particular, the Radioactive Waste Management Funding and Research Center (RWMC) has developed key technologies for remote operation of welding and inspection of the overpack, and transportation and emplacement of buffer materials in both vertical and horizontal concepts, which has been conducted at Horonobe underground research laboratory (URL). In addition, in order to demonstrate retrievability of waste packages, full-scale buffer material removal technology has been developed.

Research and development (R&D) on transuranic (TRU) waste disposal has been conducted by participating in the joint research framework of the European Union such as the Implementing Geological Disposal of Radioactive Waste Technology Platform (IGD-TP) and sharing results of international R&D. RWMC has been conducting studies on natural analogues for cement and bentonite to demonstrate long-term stability of multiple barriers for TRU waste disposal systems. RWMC has been conducting R&D on subsurface disposal for low-level waste with relatively higher activity, focusing on demonstration of construction technology of a cavern disposal facility located 50-100 m below the surface. An important aspect of work in Japan is providing information to the public on the current situation of disposal projects in other countries through the website of RWMC (http//www2.rwmc.or.jp).

Latvia. Latvia is considering an interdisciplinary conceptual approach to the development of multinational repositories, including: (a) building of an international stakeholder consensus that will be promoted by activating the international multilateral interaction between intra- and international stakeholders, (b) a gradual progress in reaching an intergovernmental consensus and multilateral agreements via observing various interests and resolving emerged controversies, and (c) knowledge and mental flexibility, which are basic prerogatives for elevating the level of mutual understanding and consensus. The following basic tasks are emphasized in Latvia: identification, prioritization, and motivation of stakeholders, establishing benchmarks for stakeholder engagement activities, and development and implementation of the stakeholder engagement strategy. One of the issues in Latvia is that the national (or a high-level) stakeholder is being represented by the national government, facing the task of seeking simultaneously an upward (or international) consensus via interacting with international stakeholders and partnering with national stakeholders, as well a downward (intra-national) consensus via interacting with multi–level intra-national stakeholders.

Lithuania. The Lithuanian spent nuclear fuel disposal program is currently aimed at the development of the initial site investigation program and a preliminary disposal facility design. Several geological formations were identified as potentially suitable for a deep repository for SNF: crystalline basement rock, Lower Cambrian clay, Permian sulfate deposits (anhydrite), Permian rock-salt, and Lower Triassic clay. A screening of the territory of Lithuania was performed based on the most important geological and hydrogeological requirements, such as simple tectonic structure and absence of aquifers, low neotectonic and seismic activity, good isolation, and favorable mechanical properties of the rock. The crystalline basement rocks were considered as one of the best candidates for a geological repository, and rock salt could not be regarded as having a high potential for disposal of SNF. The best prospects for the crystalline basement appeared to be located in the southeastern part of Lithuania, where these rocks are overlain by only 200–300 m of sedimentary cover. The prospects for clayey formations were limited to only the Lower Cambrian Baltija Formation and the Lower Triassic, since these best fulfilled the requirements for depth, thickness, lithological composition, and homogeneity of sediments. It was determined that a 100 km^2 area of crystalline rock is located between major fracture zones at an acceptable depth, which meets the requirements both with respect to tightness and stability. The Lithuanian scientists, with support of Swedish experts, are working on development of the repository concept and the generic safety assessment of a repository in crystalline rock.

A detailed repository design is highly specific to waste type and geological environment. Regardless of waste type, construction of the access and emplacement shafts and tunnels will involve the excavation of a substantial underground facility involving the removal of several hundreds of thousands of cubic meters of rock, and as much as millions of cubic meters for larger waste disposal programs. The Lithuanian repository concept is based on the KBS-3 concept developed by the Swedish Nuclear Fuel and Waste Management Company (SKB) for disposal of SNF in Sweden. The KBS-3H design, with horizontal canister emplacement, is proposed as the reference design for Lithuania.

Mexico. A temporary storage facility is currently located in Santa Maria Maquixco, State of Mexico, and is operated by ININ. It stores low and intermediate level radioactive wastes in solid form generated in non-energy activities in Mexico in the medical, industrial and research sectors, as well as spent radioactive sources. Mexico is proposing to dispose of high-level and long-lived wastes in a deep underground stable geological formation. A deep geological repository could be emplaced within the country's territory, or alternatively, Mexico could dispose of its wastes in an international co-owned, multi-shared facility. The current proposal in Mexico is for both HLW and LL-LILW to be disposed in a deep geological disposal repository. In the first stages of site selection there will be possibilities for investigations of different rock types—granite, salt, clay and tuff in Mexico. Another planned approach is the evaluation of nuclear waste disposal in existing mines. During the characterization and site confirmation stages, most of the work will be focused on the development of a license application. This usually takes from 10 to 20 years and involves moving from a general level of knowledge, obtained in the previous stage, to a site-specific level of knowledge and detailed safety assessments, incorporating engineering designs and the infrastructure required for submittal of the license application, construction, and operation. The major strategic objective for the future Waste Management Organization in Mexico should be the development of a deep geological facility within a reasonable timescale of 50 years, by the year 2065. By that time, the current Laguna Verde nuclear power plant Laguna Verde NPP (LVNPP) will have been shut down for 10 years (considering a total life span of 60 years). Also by that time, there will be spent fuel from the LVNPP and from other newer plants in temporary dry storage with a wide range of cooling times of between 10 and 75 years.

Slovak Republic. Two options are considered for the spent fuel and radioactive waste not disposable in the existing near-surface repository: (a) development of the Slovak deep geological repository, and (b) participation in activities potentially leading to implementation of a shared, international repository. The research activities mainly focused on the evaluation of geological conditions in two prospective localities and types of the formations: crystalline rock (the Tribec Mountains) and sedimentary rocks (Cerová vrchovina uplands). Field research of crystalline rock (granitoids) was conducted in the Central Tribec Mountains (Tatricum geological unit). Both tectonic and hydrogeological conditions for a repository seem to be favorable. Rock-quality designation (RQD) at depths of 150–250 m is about 90–95%. No indications of ore, mineral concentrations, or geothermal potential have been discovered in this region. Geological research activities in the sedimentary formations of the Cerová vrchovina Upland and the Rimavska kotlina Basin have focused on the evaluation of a mixture of siltstones and claystones. The maximum thickness of the Ciz Formation in the territory of Slovakia is 400-500 m, while the maximum thickness of the Lucenec Formation in Cerova vrchovina Upland is 1300 m. The thickness of both formations increases from the northern margin toward the south. The cumulative thickness of both formations varies between 1400 and 1600 m. In order to implement geological disposal in the Slovak Republic, strategic documents have delineated the following milestones:

- Elaboration of the research and development framework program in area of deep geological disposal and creation of internal conditions for its implementation – 2018,
- Creation and implementation of a system of economic incentives for municipalities affected by the development and, later, by operation of the repository – 2018,
- Adoption of the resolution on continuation or cessation of the double-track approach to geological disposal development, i.e. complex evaluation of the shared, international repository idea – 2020, In case of cessation of the double-track approach: decision on the deep geological repository site – 2030, and commissioning of the repository – 2065.

A detailed plan for the subsequent stage of the repository development (2017-2023) is currently under preparation.

Slovenia. Slovenia is a full member state of the European Union (EU) and is required to implement its national radioactive waste management program according to the EU waste management directive (EURATOM, 2011). One of the significant challenges before Slovenia is the shared responsibility with Croatia for the spent fuel and waste from shared Krško nuclear power plant. Both countries have to respect the obligations of the Bilateral Agreement to share the responsibility for waste management. No site investigations for a deep geological repository have been carried out in Slovenia, and no specific data for geological disposal are available. The reference scenario is made for a generic location in hard rock media. While hard rock is foreseen as the basic geological environment for the reference repository, it is not widespread in Slovenia. The hypothetical repository location is set at the depth of 500 m beneath the surface. *Hydrogeological properties* of Slovenian igneous and metamorphic rocks are quite comparable with hydrogeological conditions in Sweden. Fractured and weathered rock zones should be avoided. *Thermal properties* of the rock show that the temperatures at depth in areas with igneous and metamorphic rocks can be as much as 10 degrees higher compared to what is observed at Sweden sites (11°C to 14°C). This should be considered for canister and tunnel spacing calculations. The thermal conductivity of magmatic and metamorphic rocks in Slovenia varies between 2.5 and 3.4 W/m-K, which is not essentially different from the Swedish case. *Geotechnical rock quality* is expected to allow for classification of the largest part of repository to be situated within a "good" to a "very good" rock class according to the Bieniawsky Geomechanics classification system (RMR>70); and from "very good to "exceptionally good" rock class according to the Q-system (Q>20), where generally no support is required for a span of 10 m of the underground structure. The conceptual disposal concept for national repository follows the SKB KBS-3V model of disposal. Where necessary, some modifications have been introduced, either because of changes in the original Swedish model, or to adjust technical solutions to Slovenian conditions. The dual-track approach in Slovenian strategy includes both first, the option of multinational disposal is kept open, and second, the basic reference conceptual scenario for national geological disposal is included. In the revised national program the beginning of national disposal of spent fuel is assumed in or after 2065. In parallel to the national disposal program, the multinational disposal option is possible. Both options go in parallel until the beginning of construction of national repository between 2055 and 2065.

South Africa. Significant effort has been invested in planning to dispose disused (spent) and sealed radioactive sources in boreholes at the Vaalputs Radioactive Waste Disposal Facility, which is operated by Necsa, and is the only South African Low-Level radioactive waste-disposal facility. The facility is located about 100 km southeast of Springbok, in the Northern Province. It covers approximately 10,000 hectares, measuring 16.5 km from east to west, and 6.5 km from north to south at its narrowest point (see https://en.wikipedia.org/wiki/Vaalputs). Borehole disposal has been identified as the optimal disposal solution for the country's nuclear waste inventory.

Spain. Spain has the necessary infrastructure in place to manage spent fuel and radioactive waste, from institutional, administrative, technical, and economic and financial standpoints. It has also established appropriate measures to ensure the public's rights of access to information and participation. The Sixth General Radioactive Waste Plan (GRWP) approved by the Government in June 2006 planned a substantial modification to the management strategy to develop a Centralized Temporary Storage (CTS) facility as its top and most urgent priority. The permanent geological disposal of spent nuclear fuel (SNF) occupies since then a secondary place. The new plan has implied the deceleration of all activities relating to deep geological disposal. ENRESA (Empresa Nacional de Residuos Radiactivos, S.A), the public company that manages all radioactive waste produced in Spain, has essentially concentrated efforts in R&D activities, adapted to the current Spanish SNF/HLW management strategy. These activities have allowed technical knowledge to be updated and national working teams to be trained in the development of a permanent disposal option, participating in international research projects and in demonstration projects in overseas underground laboratories.

The Sixth General Radioactive Waste Plan, in effect since 2006, identifies the development of a centralized temporary storage facility as top priority, with eventual deep geological disposal to begin about 2050. Activities relating to deep geological disposal in Spain have therefore been aimed at developing the technologies required for its implementation, improving key areas of knowledge required for safety assessment, and integrating this knowledge within basic reference documents supporting future decision making. Activities related to the geological barrier have focused on progressing in process understanding and with techniques for characterizing the performance of compacted clay media and, to a lesser extent, of granitic media. All the activities have been performed within a framework of close and efficient international collaboration. The has resulted in the availability of equipment and capacities that will allow for implementation of solutions when they are considered appropriate. The transition from generic activities to specific activities is an issue that has yet to be decided and is not a short-term priority in the new ENRESA strategy.

Sweden. The Swedish program for nuclear waste is well underway. Applications for a final repository for spent nuclear fuel and an extension of the existing repository for low-level waste were submitted to the authorities in 2011 and 2014 respectively. Both repositories will be situated in Forsmark, about 150 kilometers north of Stockholm. The planning for a third repository for long-lived waste has also started, but many important milestones still remain. Long-term safety has been analyzed for both the Forsmark repositories. In the repository, the waste will be emplaced in a central structure and surrounded by one or more engineered barriers contributing to safety. The research program will therefore be concerned with the long-term performance of the waste package and the engineered barriers (concrete, bentonite, and possibly crushed rock). In addition, research may be required to improve the waste form by conditioning. The research program aimed at studying the long-term performance is a vital part of the planned extension of the Repository for Short-lived Waste (SFR). The conclusion in the safety analysis for the Repository for Spent Fuel was that a KBS-3 repository that fulfils the requirements could be built at Forsmark site, since the favourable properties of this site ensure the required long-term durability of the barriers. The conclusion is the same for the SFR repository, where the estimated risk is below the risk criterion of 10^{-6} during the assessment period.

Switzerland. The new Swiss Nuclear Energy Law and the corresponding Ordinance, both in force since February 1, 2005, define deep geological disposal as the way forward for the long-term management for all types of radioactive waste and spent fuel declared as waste. The concept of "monitored, long-term geological disposal" is based on combining passive safety during a monitoring period and retrievability "without undue effort" before repository closure. A new licensing process for site selection was approved and implemented by the federal government in 2008. This so-called "Sectoral Plan" process is driven by the long-term safety and feasibility of the geological repositories, and is based on a three-stage, stepwise, decision-making approach, with a strong participatory component from the affected communities and regions. Two separate geological repositories are being planned: one for L/ILW, and the other for spent fuel (SF), vitrified high-level waste (HLW), and long-lived intermediate-level waste (ILW). (Note: for the SF/HLW/ILW repository, the term "high-level waste repository" (HLW) is generally used.) The option of co-locating these facilities at a single site, however, is being kept open.

In Stage 1 of the Sectoral Plan process, The National Cooperative for the Disposal of Radioactive Waste (Nagra), on behalf of the waste producers, proposed six potential sites with favorable properties for construction, operation, closure and long-term safety of a deep geological repository. The identification of suitable geological siting regions was conducted in five steps. After extensive review by the authorities, followed by a broad public consultation, Nagra's siting proposals were approved by the Federal Government in November 2011. In Stage 2, the process of site selection continued with the goal of selecting at least two potential sites for each type of repository within the siting regions identified in Stage 1. In addition to the emplacement of the underground facilities, areas for surface facilities were identified as a result of an intense interaction with regional interest groups. In January 2015, following a safety-based comparison of the potential sites, Nagra submitted a proposal for two sites for Stage 3, where more detailed investigations will be performed, and the proposal for one site for each type of repository (or a combined repository) should be made. The host rock proposed for both repositories is the Opalinus Clay. Nagra's proposals are being reviewed by the authorities.

The various elements of a multi-barrier safety system can be summarized as follows:

- Waste matrix: Glass for high-level waste, UO_2 and MOX pellets within their cladding for spent fuel, and immobilization of the waste along with various low- and intermediate-level wastes.
- Disposal canisters/containers: Corrosion-resistant canisters with a lifetime of at least 1000 years for the vitrified HLW and SF; concrete containers for L/ILW and ILW.
- Backfilling of the disposal tunnels: Using pre-compacted granulated bentonite for the HLW and SF, and cement-based mortars for L/ILW and ILW.
- Geological barrier: Host rock and adjacent low-permeability confining units.
- Geosphere and geological situation at large.

In the case of the HLW repository, the underground facility will be constructed in Opalinus Clay at depths of 400–900 m below the ground surface. The thickness of the Opalinus Clay layer varies between 100 m and 130 m. The host rock has a very low hydraulic conductivity ($\leq 10^{-13}$ m/s), and its dominating solute transport mechanism is diffusion. Two operating underground research laboratories (URLs)—the Grimsel Test Site (GTS) and the Mont Terri Project—are located in two different host rocks. GTS, owned and operated by Nagra, is situated in crystalline rocks in the Swiss Alps and has been in operation since 1984 (www.grimsel.com). Mont Terri, which is owned by the Republic and Canton of Jura and operated by the Swiss Geological Survey of the Swiss Federal Office of Topography (swisstopo), is situated in Opalinus Clay in the Jura Fold and Thrust Belt and has been in operation since 1996 (www.mont-terri.ch).

United Kingdom. The UK Government adopted a policy of geological disposal, coupled with safe and secure interim storage, as the best available approach for the long term management of the UK's legacy of higher activity wastes. To identify potentially suitable sites for a geological disposal facility (GDF), the Government has developed an approach based on working with interested communities that are willing to participate in the siting process. An initial program of preparatory work is to be completed before formal discussions with communities can begin. A high-level program showing the stages of implementation, the activities at each stage and indicative timescales has been developed. The screening exercise will present an overview of relevant existing information on geology to a depth of about 1,000 meters beneath England, Wales and Northern Ireland. The overview will focus on aspects of the geological environment that are relevant to the long-term safety of the GDF. In particular, it will set out the available information on the distribution of suitable rocks with low groundwater flow in which a GDF could potentially be built. It will identify geological features, which may influence the movement of groundwater from GDF depths to the surface environment. Information will be included on the likely impact of future geological changes, such as sea level rise and future ice ages, and the distribution of minerals, hydrocarbons and other resources which may affect the likelihood of future civilizations inadvertently drilling into or mining the waste.

The principal aim of the Disposability Assessment process is to minimize the risk that the conditioning and packaging of radioactive wastes results in packages incompatible with geological disposal and the associated transport system, as far as possible in advance of the availability of waste acceptance criteria for a GDF. As such, it is an enabler for early hazard reduction on UK nuclear sites. A specification for a disposal system is derived from external considerations such as regulatory and stakeholder requirements as well as the nature, characteristics and quantities of the inventory for disposal. The concept of disposal in high strength rock is based on the idea that low heat generating waste packages are stacked in vaults that are backfilled on closure. For high heat generating waste, disposal units are lowered into deposition holes and surrounded by bentonite, and backfilled following deposition. The site-specific safety cases will be developed as a separate and parallel work stream to the generic safety cases. This strategy ensures that Radioactive Waste Management Limited has accepted, benchmark safety cases in place, whilst developing the site-specific ones. As potential sites are identified, site-specific safety cases will be developed to inform site assessment, optimization studies and regulatory submissions. The site-specific safety cases will be developed as a separate and parallel work stream to the generic safety cases rather than evolve the generic safety cases into site-specific ones. Eventually site-specific development and the safety cases will reach a sufficiently advanced state and be wide enough in scope for the generic safety cases to be no longer necessary, and all ongoing safety work will be site-specific.

Ukraine. Since the late 1990s, the study of potential geologic disposal sites has focused on the crystalline formations of the Chernobyl Exclusion Zone (ChEZ). In 1997–2000, the regional study of granitoid formations in the borders of the Korosten pluton and ChEZ was performed to assess the suitability of this territory for radioactive waste (RAW) disposal in a mined geological repository. The research was based on the results of geological, geophysical, and hydrogeological surveys at scale of 1:200,000. Two sites were selected within the ChEZ, Veresnia and Tolsty Les, as the most promising for the further study. At the same time, the feasibility of disposal of long-lived and high level RAW in Ukrainian mines was also studied. Research was conducted to assess the possibility of creating a borehole-type geological repository in this region. This work led to the preliminary conclusion that Archaean and Proterozoic crystalline rocks of the ChEZ and its vicinity are suitable for both types of geological repositories—mined and deep boreholes. Since the end of 2000s geophysical magnetic and gravity surveys, satellite imaging, and hydrogeological modeling of the ChEZ were performed for the refinement of new regulatory requirements. The result was the identification of a new area in the ChEZ, Zhovtneva, and replacement of the vast Tolsty Les area by its southwestern part with the new name, Novosilky area. Preliminary safety assessments have demonstrated the suitability of crystalline rocks of the ChEZ for geological disposal of RAW, as well as the necessity for detailed field investigations, including drilling, within promising areas.

USA. The principal focus of U.S. DOE's R&D activities is to develop a suite of options that will enable future decision makers to make informed choices about how best to manage the SNF from reactors and HLW. An additional objective is the demonstration of technologies necessary to allow deployment of solutions for the sustainable management of spent fuel that is safe, economic, and secure. The stated mission of the Used Fuel Disposition Campaign (UFDC) is to identify alternatives and conduct scientific research and technology development to enable storage, transportation and disposal of used nuclear fuel and wastes generated by existing and future nuclear fuel cycles. A comprehensive R&D program investigating a variety of geologic media is currently being executed in the UFDC for the disposal of UNF and HLW. The UFDC R&D program is designed to ensure that the technical information needed to implement new national policy for managing the back end of the nuclear fuel cycle is available for decision-making. Initially, UFDC focuses on generic research and development work undertaken today that will support future site-specific work. The research and development is focused on finding solutions and building confidence, and reducing uncertainties in the technical basis information related to challenges in nuclear waste repository siting. The UFDC conducts its R&D in collaboration with national laboratories, university, industrial, and international collaborators. The UFDC activities performed for the evaluation of deep geologic repositories include work performed under the following R&D topics:

- Crystalline Disposal R&D
- Argillite Disposal R&D
- Salt Disposal R&D
- Deep Borehole R&D
- Generic Disposal System Analysis
- International R&D
- Dual-Purpose Canisters

The design of a generic disposal system includes the evaluation of common features, events and processes (FEPs) that provide the bases for many of the fields of study. In addition, the participation in international underground research laboratory (URL) programs enhances the overall research value for the UFDC. For Deep Borehole R&D for example, the key future milestones include; design and fabrication of borehole canister, borehole construction, conduct canister emplacement test, and carry out science and engineering demonstrations. A key element in developing a safety case for any deep geologic repository is to quantitatively evaluate the response of various engineered and natural barriers limiting radionuclide transport to an accessible environment. This evaluation is provided by performance assessment methodologies such as Total System Performance Assessment (TSPA). In the U.S., the approach taken has been to construct a logical network for incorporating FEPs into quantitative methodologies that include appropriate representation of models and associated uncertainties based on simulations of thermo-, hydraulic, mechanical and chemical coupling processes in various types of geologic media for generic mined repository concepts and for a generic deep borehole disposal concept. Active collaboration with international programs, initiatives, or projects is considered very beneficial to the disposal research program, providing access to the decades of experience that some international programs have gained in various disposal options and geologic environments.

1.4 Development of Multinational Repository Concepts (Based on the Work by Arius Corporation)

The concept of developing a multinational geological disposal facility (MGDF) has been discussed for more than 40 years, and has generally been closely linked with wider consideration of multinational fuel cycle activities, often via initiatives taken by the International Atomic Energy Agency (IAEA). Despite the advantages of an MGDF, the cross-national political challenges clearly exist, and the understanding of the advantages and challenges has little changed with time. Over the last decade, Europe has taken the lead on MGDF concept development. The most comprehensive project to assess technical, economic, legal, security, political and societal aspects of developing an MGDF took place through the European Commission project called SAPIERR (Support Action: Pilot Initiative on European Regional Repositories), which ran from 2005–2009. The work was financed by the European Commission and carried out by a consortium of specialists from fourteen European Union countries.

The advantages that could be provided by an MGDF are as follows: Collecting and disposing in a central safe and secure location nuclear materials from widespread radioactive waste management (RWM) programs with different paces of development. An MGDF would, in principle, allow disposal of wastes at lower cost than in any single national GDF, which is particularly attractive for countries working together toward a shared, regional solution "at cost." A commercially-based MGDF operating within a strong, national political and regulatory framework is likely to be driven more efficiently than any national program and achieve more timely and cost-effective results for its users. Many countries with no nuclear power program, which have radioactive wastes, require geological disposal, so access to an MGDF would provide an ideal solution. The ability for any nuclear power nation to show that it has a closure solution for its RWM is key and underpins all arguments for the continuation or growth of nuclear power.

Challenges that have been experienced to date include: Inability to counter perceived public concerns about possible "dumping of other nations' nuclear waste in our country" (several national programs have presented active opposition to multinational initiatives, particularly in Europe). Unwillingness to face the political challenge, because some national politicians do not want to address arguments from the electorate that an MGDF 'might be here.' Opponents of developing an MGDF have emphasized that it is improper for a national RWM program to cite an MGDF as its eventual solution if no active project exists, as this is simply putting off finding a credible solution. Although a number of possible projects have been advanced worldwide over the last decades, it is essential that any country offering possible long-term storage or MDGF services has the strongest non-proliferation, nuclear security and environmental protection credentials. Few nations would be universally accepted in this respect. There is limited funding that has been provided by the European Commission, some US Foundations and the small nations involved with the European Repository Development Organization (ERDO) working group.

The 'dual track' concept that emerged from European projects during the last few years implies that any nation pursuing a multinational solution must also have an operational GDF program that could lead to a national solution. Two drivers were influential in establishing this approach: the first was to counter any accusation of 'wait and see,' because it is not credible for a national RWM program to rely entirely on a solution that is beyond its control and is not assured. The second driver was the requirement to satisfy international legal obligations. The IAEA Joint Convention requires regular and formal reporting, open to international scrutiny, which must present a credible national strategy and timeline for RWM. For the European Union countries, the need to have a clear and credible strategy was formalized in European Council Directive 2011/70/EURATOM, which sets out a legal timetable for EU Member States to establish a RWM program. This Directive very explicitly acknowledges the legality of transferring radioactive wastes between countries for the purposes of disposal. For transfers outside the EU, the host country would have to have a deep geological repository in operation, but this will not be the case for a long time. Discussions on multinational solutions have focused on countries with growing stocks of spent fuel to manage, as these present the greatest security issues and would involve the largest and most costly GDF development projects. However, relatively little attention in multinational discussions was given to non-nuclear power countries. Most of the advanced industrial countries have long-lived or high activity wastes in storage from medical, research or industrial applications. Some countries possess materials that they currently have no final solution for, other than permanent storage, for example Denmark, Ireland and Norway. An MGDF would provide the most obvious solution to these problems.

The Government of South Australia has recently initiated the consideration of the storage and the MGDF project. The Arius Association, with the support of the Sloan and Hewlett Foundations, has examined how the European ERDO model might be extended to countries in the Middle East and North Africa or in Asia. According to Arius, the initial predominant resistance, even hostility, to the MGDF concept, has turned into widespread acknowledgement of its advantages and realization that its development will benefit many industrialized countries. Multinational RWM solutions are now firmly embedded within the international nuclear power development landscape, and, with the considerable security and economic advantages that they offer, they will be seen as the norm for many countries in the future.

1.5 Summary of the WWR-5 Workshop and Discussion Sessions

1.5.1 Workshop Goal

The primary goal of the Fifth Worldwide Review (WWR-5) Workshop on International Approaches for Deep Geologic Disposal of Nuclear Wastes was to re-establish a series of worldwide reviews on this topic, and to provide a forum for learning from international, regional and national assessments of perspectives of nuclear waste disposal in geological formations. The Workshop was held at Lawrence Berkeley National Laboratory in Berkeley, California, on May 25-26, 2016. The presenters both in the conference room and through videoconferencing did an outstanding job of sharing their expertise, and covered many important topics during the workshop. The overall opinion of the participants was that the workshop was informative and worthwhile.

There were 45 participants in total representing 21 countries. 22 participants were present at LBNL, including 17 presenters from 12 countries, and there were 23 virtual (through videoconferencing) participants, with 10 presenters from 8 countries. The Workshop Agenda is given in Attachment 1 to this Report, and a list of the names, countries, and affiliations of participants is given in Appendix 2.

The Workshop included 4 discussion sessions, and topics of these discussion sessions are summarized below.

1.5.2 Special Discussion Sessions

Special Discussion Session #1 "*Summary of various disposal geological media and relevant technical issues*" Moderator: Jens Birkholzer (LBNL)

There are multiple ways to achieve safe and effective long-term isolation, and many concepts can work well. Different geological media have been proposed and explored for deep geologic disposal of radioactive wastes. In many countries and over a long time, considerable efforts have gone into understanding various potential disposal media. For those countries whose development program for deep disposal has not yet begun in a serious way, a careful discussion of characterization of different geological settings can be of major benefit. Those countries along with countries already involved in the programs of nuclear waste disposal can use the information to concentrate efforts accordingly, including various attributes, most importantly technical feasibility attributes, social attributes, and economical attributes related to the overall cost and benefits of a disposal program.

The discussion during this workshop session centered on the following question: What is known about various attributes of the common geologic settings being explored for deep geologic mined-repository disposal of radioactive wastes, and the linkage between geological media and relevant technologies? The session also focused on the integration of engineered barriers into the various geologic systems to produce an overall safe repository concept that isolates wastes. Comparing and contrasting these aspects would be useful for younger programs and provide a discussion forum on efficient approaches to utilizing the various geologic systems for safe waste isolation.

Special Discussion Session #2 "*How long will the dry storage systems last and what are implications for disposal programs?*" Moderator: Thomas Isaacs (CISAC, LLNL)

During the 2000s, dry storage systems were used in the United States, Canada, Germany, Switzerland, Spain, Belgium, Sweden, Russia, the United Kingdom, Japan, Armenia, Argentina, Bulgaria, Czech Republic, Hungary, South Korea, Romania, Slovakia, Ukraine and Lithuania. Spent nuclear-reactor fuel is being moved from the spent-fuel pools where it must spend its first few years to on-site dry-cask storage canisters. This technology is now mature, with several vendors and many years of experience in manufacturing, installation, and monitoring. The dry storage system "design life" is variously reported as being a few decades in duration, and sometimes the regulatory license for such casks is short – in the US NRC has in the past granted these licenses for 20 years; current requirements in 10 CFR 72.42 allow for 40 year licenses with a possible extension for an additional 40 years. But the design or regulatory "life" may be different from the actual duration before these casks will start to lose their integrity. The US NRC has a written position on this, documented in the 2014 "Continued Storage Rule" and supporting EIS, which is that "spent fuel canisters and casks would be replaced approximately once every 100 years." The comprehensive evaluation of technical concerns associated with extended dry storage is a difficult exercise.

From a standpoint of final disposition of these wastes, geologic disposal is needed regardless. However, if the dry storage systems can be relied on for X years, the question is: for how long can the need for a deep geologic repository for disposal of the spent fuel be delayed? In different countries, there are different drivers on the timing of accomplishing disposal, the reinvigoration of the nuclear industry being one primary one. The discussion topic was: What is known, technically, about how long the dry storage systems will actually last before their contents need to be removed, or the systems need to be refurbished or replaced? What are implications for disposal programs?

Special Discussion Session #3 "*Deep Boreholes*" Moderator: David Sassani (SNL)

For many years, the idea of using very deep boreholes to dispose of radioactive waste has been discussed, and several specific design ideas have been developed on paper in a number of countries including the United States, Sweden, the United Kingdom, and Russia (during the previous Soviet Union).

Recently, the idea has received increased attention, and a few new research projects have been undertaken. In the U.S.A., a new DOE initiative has begun to identify a site and implement a Deep Borehole Field Test (involving no nuclear waste) to evaluate the feasibility of the method for disposal purposes.

The topic has received attention recently because borehole-disposal proponents often claim that this method of disposal may offer unique flexibility for disposing of small waste forms and may be more rapidly deployed than would be feasible for a deep mined repository. Those who have less enthusiasm commonly state that deep-borehole disposal appears less feasible for some important radioactive waste streams, such as direct disposal of spent reactor fuel. Key points for the discussion included:

- What is known, technically, about the feasibility of deep borehole technology?
- For which types of radioactive wastes is it most suitable, and for which other types might it be generally unsuitable?
- What is known about the potential cost?

Special Discussion Session #4 "*Duration of the period of regulatory concern – technical issues and regulatory policy*." Moderators: Peter Swift (SNL), Mark Jensen (NWMO)

For a mined repository for deep geologic disposal of radioactive wastes, various countries have adopted (or proposed) a variety of different durations for the period of regulatory concern. This is defined as the period over which the repository's performance must be demonstrated in order to obtain a regulatory approval to proceed with the repository. As an example, in the U.S. the duration of regulatory concern for the Yucca Mountain repository has two parts, one part corresponding to evaluations out to 10,000 years, with additional consideration using the same modeling approaches out to 1,000,000 years.

The topic for discussion was not "*What should the period of regulatory concern be*?" That is an inherently political decision that is beyond the scope of discussion at this Workshop. The discussion topic here was: How difficult is it to demonstrate regulatory compliance for various periods of regulatory concern, as a function of the disposal scheme – meaning not only the geologic setting but also the engineering technology used for the repository? What is known technically about the issues, and why? What research is appropriate to help provide greater insight into this set of questions?

1.6 Acknowledgements

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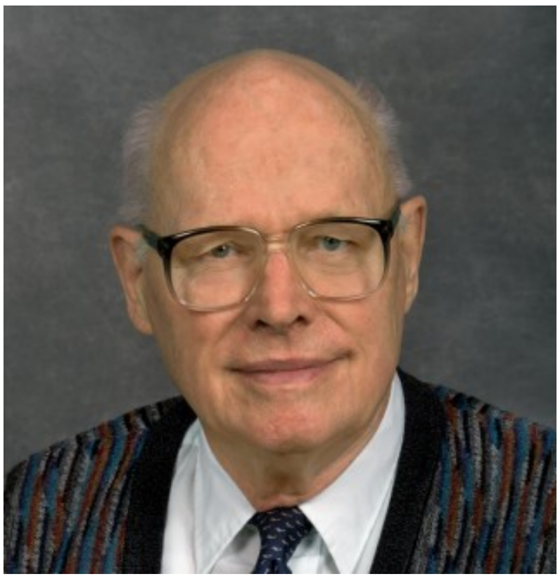
1.7 Dedication of the WWR-5

We dedicate this WWR-5 report to the memory of late Prof. Paul A. Witherspoon (1919-2012) -- http://eesa.lbl.gov/in-memoriam-paul-witherspoon/, who initiated the program of the review of national programs of nuclear waste disposal in geological formations in 1989, and published the previous four reviews (Witherspoon, 1991; 1996; Witherspoon and Bodvarsson, 2001, 2006). We also dedicate this report to the memory of Dr. Gudmundur "Bo" Bodvarsson (1952-2006), the former director of the Earth Sciences Division of Lawrence Berkeley National Laboratory, and Leader of the LBNL's Yucca Mountain project on potential nuclear waste disposal (http://eesa.lbl.gov/profiles/gudmundur-bo-s-bodvarsson/).



Prof. Paul A. Witherspoon

Dr. Gudmundur "Bo" Bodvarsson

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1.9 Acronyms

APM—Adaptive Phased Management

BCF—Boda Claystone Formation

ChEZ—Chernobyl Execlusion Zone

CISAC—Center for International Security and Cooperation (Stanford)

CNEN—National Nuclear Energy Commission of Brazil

COREJ—Internal group of experts within the waste Management Coordination in Brazil

CTS—Centralized Temporary Storage

DGR—Deep Geological Repository

DOE—Department of Energy

EPA—Environmental Protection Agency

ERDO—European Repository Development Organization

EU—European Union

FEP—Features, Events, Processes

GDF—Geological Disposal Facility

GIS—Geographic Information Systems

GMZ—Gaomiaozi

GRWP—General Radioactive Waste Plan

GTS—Grimsel Test Site

HLW—High-Level Radioactive Waste

IAEA—International Atomic Energy Agency

IGD-TP—Implementing Geological Disposal of Radioactive Waste Technology Platform

LBNL—Lawrence Berkeley National Laboratory

LILW—Low –and Intermediate-Level Nuclear Waste

LLNL—Lawrence Livermore National Laboratories

LVNPP—Laguna Verde Nuclear Power Plant

MGDF—Multinational Geological Disposal Facility

Nagra – National Cooperative for the Disposal for Radioactive Waste – Switzerland

NRC—Nuclear Regulatory Commission

NWMO—Nuclear Waste Management Organization

PURAM—Public Limited Company for Radioactive Waste Management

R&D—Research and Development

RQD—Rock-Quality Designation

RWMC—Radioactive Waste Management Funding and Research Center

RWM—Radioactive Waste Management

SAPIERR—Support Action: Pilot Initiative on European Regional Repositories

SFR—Short-Lived Waste

SKB—Swedish Nuclear Fuel and Waste Management Company

SNF—Spent Nuclear Fuel

SNL—Sandia National Laboratories

STUK—Radiation and Nuclear Safety Authority in Finland

swisstopo—Swiss Geological Survey of the Swiss Federal Office of Topography

TRU—Transuranic

TSPA—Total System Performance Assessment

UFDC—Used Fuel Disposition Campaign

UNF —Used Nuclear Fuel Disposition Research and Development

URL—Underground Research Laboratory

WW5—World Wide Review

Chapter 2

Radioactive Waste Management In Brazil Including Spent Fuel

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ABSTRACT. The authors review the main issues related to the management of radioactive waste in Brazil, including the management of spent fuel, and provide a comprehensive picture of the regulatory waste management status in the country. The article includes information about high-level waste storage and the current situation involving studies for selection of a site for a geological repository for the disposal of Brazilian high-level waste. The authors also address the technical aspects that need to be considered in Brazil to ensure construction of safe disposal facilities for radioactive waste, including description of different types of waste storage facilities existing in the country. Some aspects are highlighted: regulatory issues involving licensing, the importance of participation of the scientific community and society in the process, guidance for the application of basic requirements of safety and radiation protection, general safety aspects involved, and the current actions for the disposal of radioactive waste in Brazil.

2.1 Introduction

Articles 21 and 177 of the Brazilian Constitution state that the Federal Government has the exclusive competence for managing and handling all nuclear energy activities in the country, including the operation of all kinds of nuclear power plants. The nuclear activities shall be solely carried out for peaceful uses and under the approval of the National Congress. The Federal Government holds the monopoly for surveying, mining, milling, exploitation, and exploration of nuclear minerals, as well as the activities related to industrialization and commerce of nuclear minerals and strategic materials for the nuclear industry. The Brazilian Nuclear Energy Commission (CNEN) is responsible for receiving and disposing of all radioactive waste existing in the country.

CNEN (Comissão Nacional de Energia Nuclear—http://www.cnen.gov.br), the Brazilian Nuclear Energy Commission, was created in 1956, and currently has two technical directorates. Figure 2–1 shows the structure of CNEN.

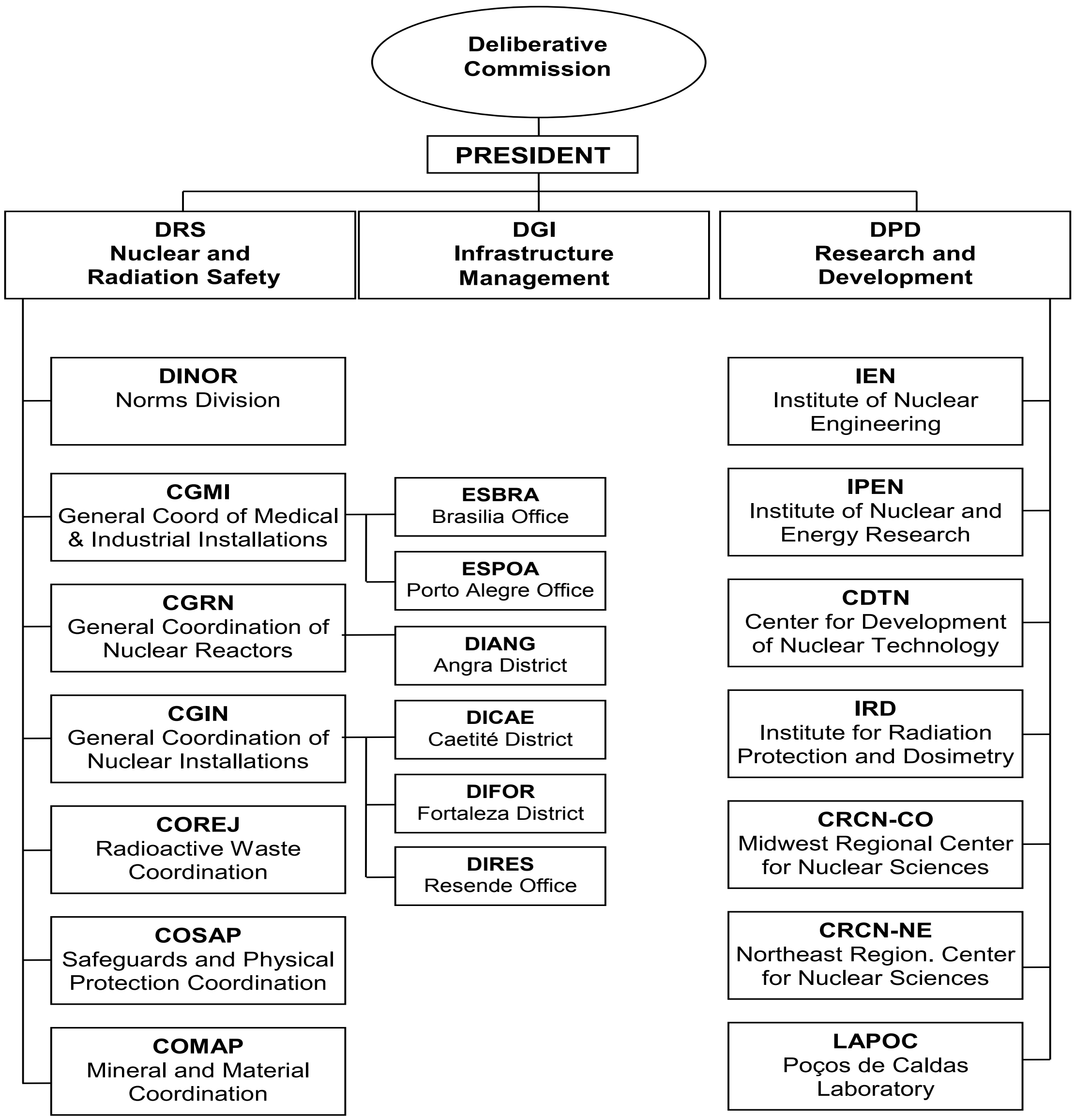

Figure 2–1. CNEN structure. Industrial, medical, and research installations, as well as Brazilian nuclear power plants in operation for the last 50 years, generate wastes that must be properly managed. To better oversee the safe construction, operation, and decommissioning of radioactive waste storage and disposal facilities, and to control the transport of radioactive materials, the radioactive waste management sector of the Brazilian Nuclear Energy Commission, CNEN, was transformed to Radioactive Waste Management Coordination in 1995 (COREJ).

DRS (Diretoria de Radioproteção e Segurança Nuclear) is responsible for radiation protection and nuclear safety. The DRS directorate is responsible for radiation safety issues in the country, including regulating, licensing, and controlling nuclear energy utilization, coordinating various activities, such as nuclear waste management (COREJ—Coordenação de Rejeitos—Waste Management Department). DRS duties also include the so-called "regulatory research," conducted for regulatory purposes.

DPD (Diretoria de Pesquisa e Desenvolvimento) is responsible for research and development activities in the country, and its directorate controls the three main Brazilian Research Institutes: IEN (Instituto de Engenharia Nuclear), located in Rio de Janeiro state; CDTN (Centro de Desenvolvimento da Tecnologia Nuclear), located in Minas Gerais state; and IPEN (Institudo de Pesquisas Energeticas e Nucleares), located in São Paulo state. Each has a nuclear research reactor in operation. DPD is also responsible for the production of radioisotopes for medical, industrial and research uses in the country.

2.2 The Brazilian Nuclear Program

The Brazilian Nuclear Program oversees the operation of several nuclear installations and thousands of installations that use radioactive material for medical, industrial and research purposes, including the following:

- Two nuclear power plants in operation (Angra 1 and 2), and one under construction (Angra 3). Angra 1 is a 657 MWe gross/626 MW net, 2-loop pressurized water reactor (PWR). Angra 2 is a 1345 MWe gross/1275 MWe net, 4-loop PWR. Angra 3, under construction, will be a 1312 MWe gross/1229 MW net, 4-loop PWR. All reactor plants are located at Angra dos Reis City, Rio de Janeiro state.
- Two uranium mines and mills, operated by Nuclear Brazilian Industry (INB). One is currently closed and the other is still in operation. The first uranium mine, located in Poços de Caldas city, Minas Gerais state, operated from 1982 until 1991. All the economically recoverable uranium was extracted, and currently no mining activity is under way. The second mine and milling complex is located in Caetité, Bahia state, with a capacity of 100 t/year of U_3O_8 and the production of lanthanum chloride and cerium hydroxide. This mine has been operational since year 2000, with reserves of 100,000 t of U_3O_8, and a capacity of 400 t/year of yellow cake (U_3O_8), which can be expanded to 800 t/year.
- One fuel element complex named FEC, at Rio de Janeiro state, also operated by INB. The factory includes a reconversion and a fuel fabrication plant. The enrichment plant is expected to be operational in 2016.
- Four research reactors: the first is IEA-R1 (1956), located at the Nuclear and Energetic Research Institute (IPEN), on São Paulo University campus, in São Paulo city, with 5 MW power. The second, named IPR-R1, is a 100 kW Triga reactor, operating since 1960 at CDTN, located on the campus of Federal University of Minas Gerais in Belo Horizonte. The third research reactor is named ARGONAUTA (1965) and is located at the Institute of Nuclear Engineering-IEN, on the campus of the Federal University of Rio de Janeiro, in Rio de Janeiro city. The last one, the IPEN MB-01, located at IPEN, is a water tank type critical facility rated 100 W and is the outcome of a national joint program, developed by CNEN and the Navy.
- Pilot scale fuel cycle facilities, including a plant for the conversion of uranium to UF_6, and another for uranium enrichment, both belonging to the Navy.
- Approximately 4,000 medical, industrial, and research facilities that use radioactive sources and equipment (not including *x*-rays).
- One industrial facility for processing monazite sands.
- Many installations with NORM materials (phosphates, zirconite, petroleum, tantalite, columbite, etc).

The state company responsible for the fuel cycle in Brazil is "Nuclear Industries from Brazil" (INB: Indústrias Nucleares do Brasil—http://inb.gov.br). To ensure the fuel cycle facilities are in accordance with the safety principles required by national and international authorities, the nuclear installations are subject to a set of requirements established in a series of CNEN nuclear regulations, based on the

International Atomic Energy Agency (IAEA) standards. These regulations include the elaboration and approval of a "Site Location Report," a "Preliminary Safety Analysis Report," and a "Final Safety Analysis Report." The operator prepares and submits the reports in accordance with CNEN regulatory guides and standards in order to obtain the license from the regulatory body.

Reports must be prepared and submitted in accordance with regulatory guide CNEN NE 1.04 "Licensing of Nuclear Installations" (CNEN NE 1.04, 2002), which, in the case of uranium ore mining and milling operations, is further supplemented by regulatory guide CNEN 1.13—"Licensing of Uranium and/or Thorium Mining and Milling Facilities" (CNEN NE-1.13, 1989). Additionally, all nuclear facilities go through an environmental licensing process, including an Environmental Impact Study, which describes the safety conditions relevant to the environment and surrounding population. IBAMA (Instituto Brasileiro do Meio Ambiente e dos Recursos Naturais Renováveis, or Brazilian Institute of Environment and Renewable Natural Resources) conducts this process, together with CNEN. In some cases, the state environmental protection agencies can participate, too.

2.3 Classification of Radioactive Waste in Brazil

One of the common definitions of radioactive waste is "Material, whatever its physical form, remaining from practices or interventions, containing radionuclides in quantities exceeding clearance levels (specified in the Standards) and for which no further use is foreseen." According to the International Atomic Energy Agency (IAEA), radioactive waste was initially grouped into five categories, as shown in Table 2–1. Subsequently, the IAEA (1994) re-classified the radioactive wastes into three categories, taking into account the disposal options, as shown in Table 2–2, where short-lived applies to radionuclides with half-lives up to approximately 30 years (such as ^{137}Cs, ^{90}Sr, ^{60}Co, ^{85}Kr, ^{3}H, etc).

In 2009, the IAEA published a third classification for radioactive waste (IAEA, 2009), as shown in Table 2–3. This new classification has taken into account some missing points identified at IAEA technical meetings, and included all types of radioactive waste, a direct link with disposal options, as well as reflected experience gained in developing, operating and assessing the safety of disposal facilities. Brazil adopted the third classification that is shown in Table 2–3.

Table 2–1. Initial IAEA Classification of Radioactive Waste.

Category	Description
I High Level Waste: Long-Lived	- High βγ Energy Emitters - High Level of α Emitters - High Level of Radiotoxicity - High Heat Generation
II Intermediate Level Waste: Long-Lived	Intermediate βγ Energy Emitters High level of α Emitters Intermediate Level of Radiotoxicity Low Heat Generation
III Low Level Waste: Long-Lived	Low βγ energy Emitters High level of α Emitters Low/intermediate Levels of Radiotoxicity Negligible Heat Generation
IV Low Level Waste: Short-Lived	Low βγ Energy Emitters Negligible Level of α Emitters Low Level of Radiotoxicity Negligible Heat Generation

Table 2–2. IAEA (1994) Classification Of Radioactive Waste, Taking Into Account The Disposal Options.

Category	Characteristics	Disposal
1. Exempt Waste (EW)	The levels of radioactivity in the waste are below the clearance values, based on a trivial annual radiation dose for the public of no more than 0.01 mSv.	No Radiological Restriction
2. Low and Intermediate Level Waste (LILW)	Radioactive waste activity levels above the clearance levels for (EW) and heat generation ≤ 2 kW/m^3.	
2.1. Short-Lived Waste (LILW-SL)	Long lived α emitters concentration ≤ 4000 Bq/g and the average of all radionuclides concentration in the package < 400 Bq/g.	Near Surface or Geological
2.2. Long-Lived Waste (LILW-LL)	Long lived α emitters with concentrations above the previous values.	Geological Only
3. High Level Waste (HLW)	Heat generation > 2kW/m^3 and α emitters concentrations above the levels established for Category 2.1	Geological Only

2.4 Radioactive Waste Management Regulations, Laws and Conventions in Brazil

On September 29, 1997, the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management was open for signature at the Headquarter of the International Atomic

Energy Agency in Vienna. Brazil signed the Convention on October 11, 1997, and ratified it by the Legislative Decree No. 1019 on November 14, 2005. Brazil deposited the instrument of ratification with the Depositary on February 17, 2006, and the promulgation of the Convention was done in the Decree No. 5935 on October 19, 2006 (DECREE Nº 5.935, 2006).

One of the obligations of the Parties to the Convention is the preparation of periodic national reports, describing the measures taken to implement each obligation of the Convention, including a description of the policies and practices related to spent fuel and radioactive waste management, as well as an inventory of related material and facilities.

The main Brazilian regulations for radioactive waste management in force are CNEN-NN-8.01 (2014) and CNEN-NN-8.02 (2014), which use the classifications given in Table 2–3, except for high-level waste. The first of these establishes general criteria and basic requirements for safety and radiation protection related to the management of radioactive waste. The waste covered by the regulation is that with low and medium level radiation and radioactive waste with very short half-life. The second regulation establishes the general criteria and basic requirements of safety and radiation protection related to the licensing of initial, intermediate and final deposits of radioactive waste of low and medium levels of radiation, in compliance with the Brazilian Law 10 308/2001 (2001). This law established the main regulatory criteria in the country for site selection, construction, licensing, operation, supervision, costs, damages, liability, and warranties related to deposits of radioactive waste, among other measures.

Two more important regulations for low-level radioactive waste are in force in Brazil: CNEN-NE-6.06 (1989) for site selection for waste disposal, and CNEN-NE-6.09 (2002) for the acceptance criteria for waste disposal. As of today, no specific safety criteria have been established in Brazil for site selection and disposal of the high-level waste generated in the country, although there are safety criteria regulations for the storage of spent fuel.

Spent fuel is not currently considered high-level waste, because there is no final government decision about reprocessing or not of the stored spent fuel. At present, spent fuel is stored in pools inside the reactor plant. The general safety requirements for the management of spent fuel are contained in the safety requirement for siting, design and operation of the nuclear reactors; see Regulation CNEN-NE-1.04 (1984). Additional requirements are established in Regulation CNEN-NE-1.26 (1997) for the operational phase, and Regulation CNEN-NE-1.14 (2002) for the necessary reporting requirements.

Table 2–3. IAEA (2009) Classification of Radioactive Waste

Category	Characteristics
Exempt Waste (EW)	Waste that meets criteria for clearance, exemption or exclusion from regulatory control for radiation protection purposes
Very Short Lived-Waste (VSLW)	Waste that can be stored for decay over a limited period of up to a few years and subsequently cleared from regulatory control. This class includes waste containing primarily radionuclides with very short half-lives often used for research and medical purposes.
Very low level waste (VLLW)	Waste that does not necessarily meet the criteria of EW, with no need for a high level of containment and isolation, and, therefore, is suitable for disposal in landfill type facilities with limited regulatory control. Such landfill type facilities may also contain other hazardous waste. Typical waste in this class includes soil and rubble with low levels of activity concentration. Concentrations of longer-lived radionuclides in VLLW are generally very limited.

Category	Characteristics
Low Level Waste (LLW)	Waste that is above clearance levels, but with limited amount of long-lived radionuclides. Such waste requires robust isolation and containment for periods of up to a few hundred years and is suitable for disposal in engineered near-surface facilities. This class covers a very broad range of waste. LLW may include short-lived radionuclides at higher levels of activity concentration, and long-lived radionuclides, but only at relatively low levels of activity concentration.
Intermediate Level Waste (ILW)	Waste with long-lived radionuclides, which requires a greater degree of containment and isolation than that provided by near-surface disposal. However, ILW needs no or limited provision for heat dissipation during the storage and disposal. ILW may contain long-lived radionuclides, in particular, alpha emitting radionuclides that will not decay to a level of activity concentration acceptable for near-surface disposal during the time of the institutional control. This type of waste requires disposal at greater depths of the order from tens of meters to a few hundred meters.
High Level Waste (HLW)	Waste with a high level of activity concentration, which generates significant heat due to the radioactive decay process or waste with large amount of long-lived radionuclides. Disposal in deep, stable geological formations usually several hundred meters or more below the surface is a generally recognized option for disposal of HLW.

2.5 Main Aspects of the Brazilian Waste Disposal Law 10 308/2001

According to the Brazilian Federal Waste Law (2001), radioactive waste deposits can be of four types, as follows:

1. *Initial deposits*: Initial deposits are those constructed by the operator for the safe storage of the radioactive waste directly on site.
2. *Intermediate deposits*: Intermediate deposits are those constructed with the purpose of waste storage prior to disposal, waste treatment and conditioning. The intermediate deposits in Brazil were built in the Research Institutes of CNEN, and receive waste generated from industrial, medical, and research facilities.
3. *Final deposits*, also known as repositories: Final deposits are those constructed for the final disposal of radioactive waste with no intention of retrieval.
4. *Temporary deposits*: Temporary deposits can only be constructed for the interim storage of radioactive wastes generated in the case of nuclear or radiological accidents.

The above-mentioned law also establishes the following provisions:

- Initial, intermediate, and final deposits shall be constructed, licensed, operated, and administered in accordance with criteria, procedures and standards established by CNEN.
- Disposal of radioactive waste in liquid or gaseous form is prohibited in final repositories.
- The design, construction, and operation of initial deposits of radioactive waste, as well as related costs, are the responsibility of the operator of the facility that generates the waste.
- The design, construction, and operation of intermediate and final deposits of radioactive waste are of CNEN's responsibility. Services can be entrusted to third parties, however the full responsibility remains with CNEN.
- The surveillance of initial, intermediate, and final deposits will be performed by CNEN, in its field of legal competence, without prejudice to the exercise by other government organizations of their legal competence.

- Initial deposits used for waste storage in mining and milling facilities can be converted into final deposits, with the express permission of CNEN. The holder of the authorization for the operation of these facilities will fully bear the costs related to site selection, design, construction, installation, licensing, administration, operation, and security of these deposits.
- Site selection for initial, intermediate and final deposits must comply with criteria, procedures and standards established by CNEN.
- Sites selected for final disposal facilities that do not belong to the government will be declared of public utility and dispossessed.
- Construction of radioactive waste deposits of any nature on oceanic islands, on the continental shelf and in Brazilian waters is prohibited.
- CNEN will bear the costs related to site selection, design, construction, installation, licensing, administration, operation, and physical security of intermediate and final deposits.
- Treatment and disposal of radioactive waste must have their costs reimbursed to CNEN by the waste generator. The costs are calculated taking into account the following items:
- Volume of waste to be placed in the deposit.
- Activity concentrations of isotopes present in the waste.
- Expenses related to the licensing, construction, operation, maintenance and security of the deposit.
- Projects related to national defense are exempt from payment of costs associated with radioactive waste treatment and disposal.
- The holder of the authorization to operate a deposit has financial obligations related to personal radiation damage, property and environmental damage caused by the stored radioactive waste.
- The holder of the authorization for operation of an initial deposit has financial obligations related to personal radiological damage, property and environmental damage during the transport of waste to intermediate or to final deposits.
- CNEN has financial obligations related to personal radiological damage, property and environmental damage during the transport of waste from intermediate to final deposits.
- The Municipalities that have initial, intermediate, or final deposits shall receive monthly compensation. This compensation may not be less than 10% of the costs paid to CNEN for storage or disposal of radioactive waste.
- It is an obligation of CNEN to receive and transfer monthly to the Municipalities the compensation amount paid by the operators of the waste generating facilities.

2.6 Financial Compensation Model for Radioactive Disposal Sites

In 2003, CNEN approved a Technical Note developed by Heilbron (2003), which established methodology for financial compensation for Brazilian Municipalities where deposits are placed, based on the provisions of the aforementioned radioactive waste law. This model was revised in 2009, and takes into account:

- The volume of waste and whether it has been treated or not
- If the waste comes from mining activities
- The time required for storage and type of deposit (initial, intermediate, final, or temporary)
- The concentration of radioactive materials in the waste
- The properties of the radionuclides, such as type of emission, half-life, etc.
- The demography and population density near the site of the deposit
- A maximum period of institutional control of 300 years, as recommended worldwide

According to the Brazilian Law on Fees (BRAZILIAN FEDERAL LAW No 9.765/1998, 1998), the final disposal cost per waste volume for low-level waste (cubic meter) was established as R$ 10,000.00 (≈3,000 $US). For the calculation of the financial compensation to be paid monthly to the Municipality, the following methodology was developed.

The monthly amount to be paid to the City Council (R$/month), V_m, is determined from:

$$V_m = F_m \times V_{RC}/T \qquad (2\text{-}1)$$

where F_M is the percentage of the amount received by CNEN from the operator (V_{RC}), and to be paid to the City Council; V_{RC} is the cost of CNEN for waste disposal; T is the maximum period of the institutional control established for near-surface repositories (T = 3,600 months). $\mathbf{F_m}$ is determined from:

$$F_m = F_b + F_{dd} \qquad (2\text{-}2)$$

where F_b is the minimum percentage established by the Law (F_b = 10%); F_{dd} is a factor that allows for an increase in the payment value according to the demographic density (Table 2–4). Thus, the values to be paid to the Municipality are in the range from 10% to 15% of the total waste disposal expenses borne by CNEN. The values of F_{dd} are given in Table 2–4. The costs to be paid by the operator, V_{RC}, is given by:

$$V_{rc} = V_r \times (C_r x k_2) x\ k_1 \qquad (2\text{-}3)$$

and the parameters taken into account are explained in Table 2–5.

Table 2–4. F_{dd} Factor as a Function of Demographic Density

F_{dd}	Demographic Density Of The County (inhabitants/km^2)
1.0	< 500
1.1	≥ 500 and < 1000
1.2	≥ 1000 and < 2000
1.3	≥ 2000 and < 3500
1.4	≥ 3500 and < 5500
1.5	≥ 5500

Table 2–5. Parameters considered in Equation 2–3

V_r	Volume of waste to be disposed—m^3
C_r	Waste reference cost per cubic meter—R$/ m^3
k_1	Cost correction factor due to the type of waste, radionuclide concentration, type of deposit, half-life of the nuclide, etc. that affects the cost of disposal
k_2	Factor to be applied when the costs associated with storage are basically borne by the operator (e.g., mining and milling waste)

The k_1 factor can be estimated as follows:

$$k_1 = F_d + F_r + F_{mc} \tag{2-4}$$

where F_d, is the percentage factor to be applied to the cost to take into account the type of disposal facility: F_d = 5%, Waste in Initial Deposit; F_d = 10%, Waste in Intermediate Deposit; and F_d = 20%, Waste in Final Deposit. F_r, is the percentage factor to be applied to the cost to take into consideration if the waste has or has not undergone treatment: F_r = 5%, For treated wastes; F_r = 10%, For semi-treated wastes; and F_r = 20%, For non-treated wastes;

F_{mc}, Percentage factor to be applied to the cost to take into account radionuclides concentration and half-lives which are associated to the risk of disposal: F_m, Is associated with the half-life of the radionuclide in the waste; and F_c, Is associated with the concentration activity of the waste

The factor F_{mc} accounts for half-life and concentration, and is given by

$$F_{mc} = F_m + F_c \tag{2-5}$$

where F_m = 40%, waste containing beta and gamma emitters with half-lives less than one year and with concentration of alpha emitters less than 3,700 Bq/g; F_m = 50%, waste containing beta and gamma emitters with half-lives between one year and approximately 30 years and with alpha emitters concentration equal to or greater than 3700 Bq/g; and F_m = 60%, waste with mainly alpha emitters with half-lives greater than 30 years. F_c values are different for waste containing naturally occurring radioactive materials (NORM), and other types of waste: for NORM Waste - F_c = 25% for radioactivity from 100 Bq/g to 1,000 Bq/g, F_c = 60% for radioactivity from 1,000 Bq/g to 10,000 Bq/g, and F_c = 100% for radioactivity more than 10,000 Bq/g.

The range of variation of k_1 is from 56% to 90%. The k_2 values are applied to the reference cost, C_r, and are determined as following: k_2 = 1/1,000, for NORM waste (large volumes, but low activity), and k_2 = 1, for all other wastes. In the case of waste in a final disposal, the same methodology is used, but the value of T varies with time, and Equation (2-1) is given by

$$V_m = F_M \times Vrc \frac{V_{RC}}{360} \times f_{d,n} \tag{2-6}$$

where $f_{d,n}$ is the 30-year time constant factor month payment (for months of 1-30, 31-60, 61-90, 91-120, 121-150, and so on); $f_{d,n} = 2^{-n}$ (n = 1 to 9) and $f_{d,10} = f_{d,9}$. Note that $f_{d,n}$ = 1/2, 1/4, 1/8, etc., and

$$\sum_1^{10} f_{d,n} = 1 \tag{2-7}$$

In the case of high-level waste, it is recommended that CNEN will carry out a more detailed study of some parameters in order to take into consideration the different costs of disposal and the necessity of the institutional period, as well as the high cost of the necessary research including transport costs associated to the spent fuel.

2.7 Final Waste Disposal Facilities for Environmental Protection

According to the IAEA (1994), the waste disposal facilities can be classified into two categories: *Near-surface Repository*, and *Geologic Repository*. A near-surface repository is constructed at depths of a few meters from the surface (not greater than 30 m) and can be either a simple trench or a more sophisticated construction with engineered barriers, usually made in concrete. Many countries have used this type of disposal site for the final, safe disposal of their low-level waste (LLW), including Brazil (Goiânia), England (Drigg), France (La Manche and Centre de L'Aube), Canada (Irus), Japan (Rokkasho), Spain (El Cabril), Sweden (Oskarshamn), and USA (Barnwell).

The establishment of an institutional period of control for near-surface repositories is very important to diminish the long-term risk of intrusion that could lead to higher doses for the intruder. The institutional period can be passive (without the presence of personnel on the site) or active (with personnel on the site, for monitoring and physical protection, among other activities).

Contrary to the near-surface repository, the geologic repository is located at great depth to eliminate the risk of human intrusion or animal contact with the waste. This kind of disposal facility is recommended for high- and intermediate-level wastes, and can belong to one of the three categories:

- Cavities, specially man constructed
- Old abandoned salt mines
- Natural cavities.

The only Brazilian Waste Disposal Facility is a near-surface type, and was built following the Goiânia accident involving a teletherapy source of approximately 1375 Ci of ^{137}Cs. The decontamination work performed in the contaminated area generated approximately 3,500 cubic meters of very low and low level waste. Figures 2–2 and 2–3 show the location and provisional storage of Goiânia waste. Two near-surface repositories with engineered barriers were constructed about 30 km from Goiânia city. One repository, named CGP, received the very low level waste (approximately 40% of the total volume), and the second one was filled with more robust engineered barriers for placement of the remaining 60% of low level waste generated (Figure 2–4).

In the specific case of the very low level waste, the environmental impact report was not required by the Brazilian environmental agency, since the safety assessment report prepared by CNEN showed that the risks associate with the disposal of that waste would be negligible within a few years (Heilbron et al., 1994; 1996).

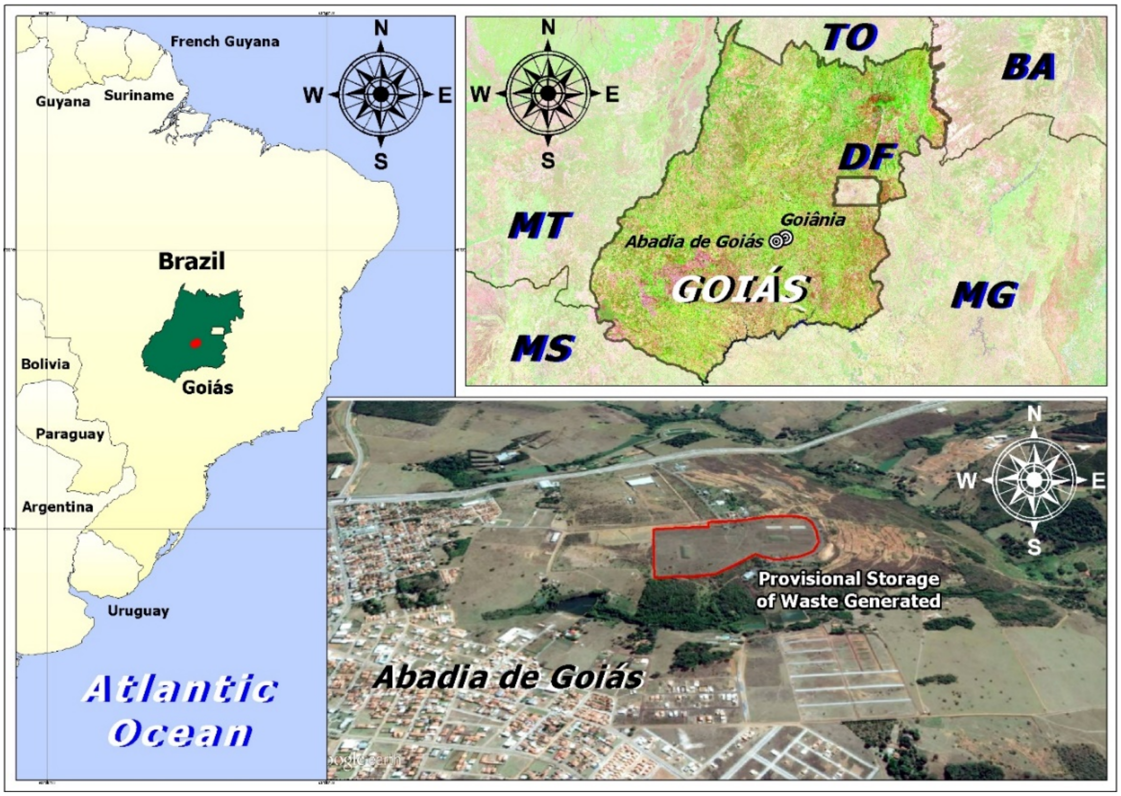

Figure 2–2. Provisional Storage of Waste Generated in Goiânia

Figure 2–3. Provisional Storage of Waste Generated in Goiânia

Figure 2–4. Final disposal sites in Goiânia, showing the two near-surface disposal units (vaults), and the necessary waste treatment before final disposal of packages.

In 2014, CNEN jointly with the Geology Department of the State University of Rio de Janeiro, and with the support of FAPERJ, started a project aimed at the development of a GIS system, including the main technical information, for the site selection of a geological disposal in the state of Rio de Janeiro for the HLW existing in Brazil.

2.8 Management of Spent Fuel

For Angra reactors the spent fuel is stored in swimming pools; an additional wet storage facility is planned, complementing the current on-site storage capacity. ELECTRONUCLEAR Company will accept responsibility for operation of the initial waste storage facility. For Angra 1, a Westinghouse project, the new fuel dry storage room and the spent fuel pool are located in the Fuel Handling Building, having connections with the reactor via the fuel transfer system and the refuelling machine. The path of the nuclear fuel inside the plant up to the reactor is: the entrance gate, the cask opening area inside the fuel building, the new fuel storage area, the transfer canal (or temporarily in the spent fuel pool), the fuel transfer system, the refuelling machine, and the reactor core.

For Angra 2 and 3, a Siemens project, the dry new fuel storage room and the spent fuel pool are also located inside the Reactor Building. The path of the nuclear fuel inside the plant up to the reactor is: the entrance gate, the auxiliary portico, the equipment lock, the cask opening area, the new fuel storage area, the refuelling machine, the spent fuel pool, and the reactor core.

Both units are equipped with fuel storage racks of two different designs. The first group, named Region 1, or compact racks, was designed to receive fresh and irradiated fuel assemblies at maximum reactivity for the specified core design, without taking credit for burnup. The second group, named Region 2 or super compact racks, was designed to receive fuel assemblies that have reached a certain minimum burnup.

In Angra 1, the compact and super compact racks, made of stainless steel, have boron between the storage cells. In Angra 2, the compact and super compact racks use borated steel plates as the construction material of the cells, and were supported by criticality safety studies, appropriate periodic inspection, and testing in all storage places. Criticality in the new fuel and in the spent fuel storage areas is prevented both by physical separation of fuel assemblies, by boron shields, and by borated water as appropriate.

The storage capacity and occupation of the two Brazilian nuclear power plants storage systems under operation are shown in Table 2–6.

Table 2–6. Spent fuel storage capacity (i.e., number of fuel assemblies) at Angra.

Storage Local	Angra 1		Angra 2	
	Capacity	Occupation	Capacity	Occupation
New Fuel Storage Room	45	0	75	0
Region 1 Spent Fuel Pool	252	142	264	39
Region 2 Spent Fuel Pool	1000	508	820	233
Reactor Core	121	121	193	193

Assuming a regular lifetime of 32 operating cycles for each unit, and that in each cycle a third of the core is replaced, then Angra 1 has enough storage capacity for its entire lifetime, and Angra 2 has storage capacity for about 14 cycles. Both units have redundant residual heat removal systems fed by redundant electrical safety buses, with provisions from the plant house load supply, redundant external electrical supplies and redundant Diesel generators. The sources of cooling water are closed circuits, cooled by seawater open circuits. Instrumentation in the cooling and purification systems of the fuel pools detects radiation and high or low temperatures, emitting alarms to prompt the operator for actions.

Each unit is designed for a regular lifetime of 32 operating cycles. According to the national electric power demand, the refuelling policy is to operate with 11 equivalent full power monthly cycles, with a one-month refuelling outage. Studies are being carried out to gradually increase the cycles to 18 months, since longer cycles reduce waste generation and doses during refuelling outages. Shutdowns, refuelling, and startups of the plants are conducted in such a way to reduce the amount of radioactive waste.

Table 2–7 lists the main characteristics of the fuel elements used in the Brazilian research reactors. Table 2–8 shows the inventory.

Table 2–7. Fuel Element Characteristics.

Facility	Fuel Type	Fuel Material	Enrichment	Cladding Material
IEA-R1	MTR	U_3O_8-Al U_3Si_2-Al	LEU 19.9%	Aluminum
IPR-R1	TRIGA	U-Zr-H	LEU 20%	Aluminum/SS
ARGONAUTA	MTR	U_3O_8-Al	LEU-19.0-19.9%	Aluminum
IPEN-MB-01	Pin PWR	UO_2 Pellets	LEU 4.35 %	SS

Table 2–8. SF Inventory at Brazilians Research Reactors.

Facility	# of FA in Present Core	Average # used per year	SFA Storage		SFA % Average Burnup
			At RR	Outside RR	
IEA-R1	24 LEU, Silicide-9; Oxide-15	~18, expected for 120 h/week, 5 MW	16 wet	0	~30
IPEN-MB-01	680 pins	NA	0	0	NA
IPR-R1	59 rods (LEU)	NA	0	0	~4
IEN-R1	8 LEU	NA	0	0	NA

NA = not applicable

2.9 Waste Management Considerations in Brazil

The low- and intermediate level radioactive waste in Brazil come from the facilities of the nuclear fuel cycle, including the two nuclear power reactors, the processing of monazite sands, the mining and milling facilities of conventional ores that are associated with uranium and thorium, the use of radioisotopes in medicine, industry, research, and from the decontamination work performed in Goiânia following the radiological accident that occurred in 1987.

The radioactive waste and nuclear materials generated over the last 40 years in Brazil are currently stored in various nuclear and radioactive facilities located in four states of the Federation, owned or supervised by CNEN.

The waste generated by the Uranium Mine and Milling Facilities, although significant in volume, is kept at the respective sites in dams specially built for this purpose. There are, presently, about 600 metric ton of "mesothorium" with an estimated ^{228}Ra activity of 1.85 TBq (50 Ci) stored by INB in a trench and 0.2 TBq (6 Ci) stored in the shed (78 m^3).

Since the Brazilian Regulatory Body was suspended in 1989, the authorization was given to several manufacturers to use radioactive sources in lightning conductors. An estimated 75,000 lightning rods with an overall activity on the order of 3.7 TBq (100 Ci) of ^{241}Am were installed all over the country, and part of them were already moved to CNEN's Institutes for dismantling.

After a certain decay period, most of the soluble radioactive wastes generated from using short-lived radioisotopes in medical institutions and research laboratories can be discharged into sanitary sewerage systems, if concentrations and total activities are not exceeding the limits specified by the Brazilian specific regulations. The low level radioactive wastes generated by the nuclear reactors (shown in Figures 2–5 and 2–6) are stored at the sites in sheds (see Figure 2–7).

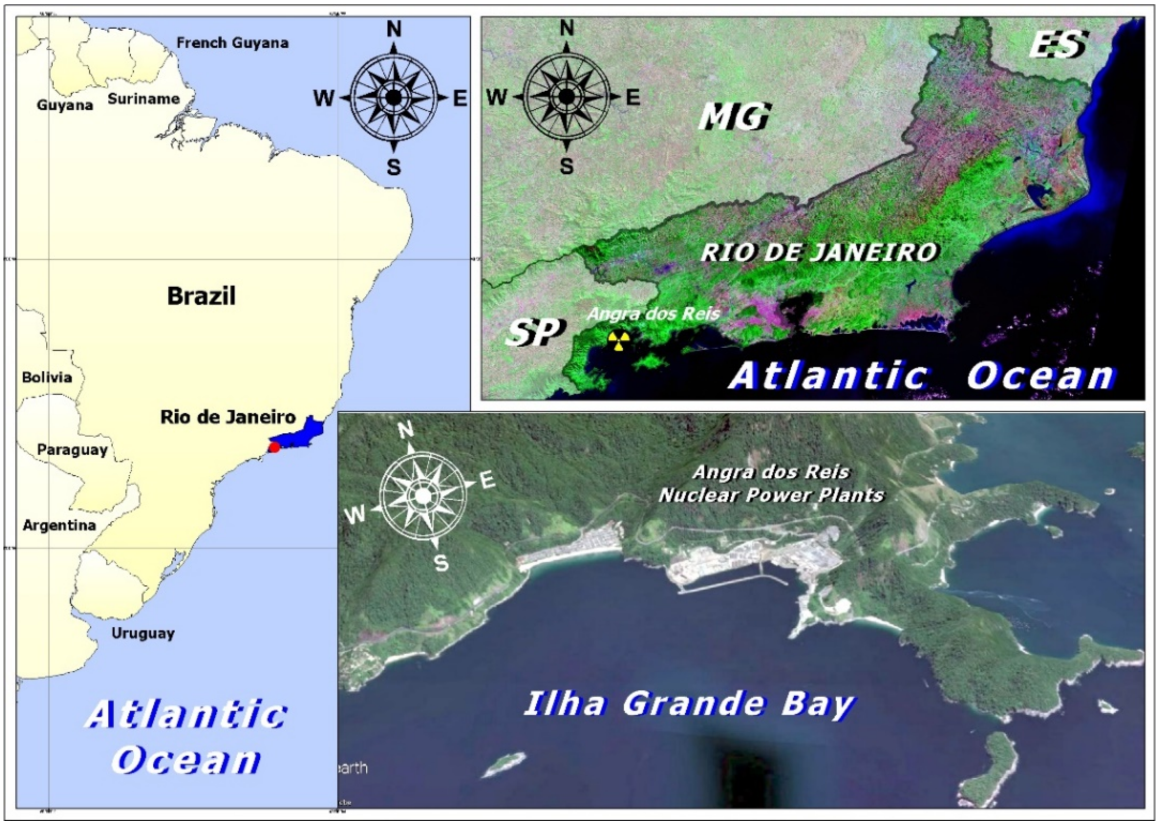

Figure 2–5. Location of Brazilian Nuclear Power Plants.

Figure 2–6. A view of Brazilian NPP Site.

Figure 2–7. Left: Waste Storage Shed in the NPP. Right: Waste Storage Shed in the Nuclear Reactor Site.

Spent fuel elements removed from the power reactors (Angra 1 and Angra 2) are stored in pools located in respective reactor building, awaiting the decision of reprocessing or final disposal.

Radioactive waste from medical, industrial, and research facilities that cannot reach clearance levels are mainly stored in CNEN Research Institutes. Figure 2–8 illustrates some of CNEN's storage facilities and treatment systems.

Figure 2–8. Waste Storage Facilities in Brazil and treatment systems of CNEN research institutes.

As established in the Federal Laws No. 4118 (Law 4118/62 of 1962.07.27, 1962), No. 6189 (Law 6189/74 of 1974.12.16, 1974), and No. 7781 (Law 7781/89 of 1989.06.27, 1989), CNEN is responsible, among other duties, for receiving and disposing of radioactive waste in the country, as well as for issuing regulations and safety standards related to treatment and disposal of radioactive waste.

The waste policy adopted in Brazil is similar to that adopted by several countries around the World and is consistent with the IAEA recommendations. The final disposal of radioactive waste of low and very low levels of radiation is performed in near-surface repositories, with the necessary engineered barriers. In the case of high-level wastes, a geological repository must be used for the safe disposal.

Because of the complexity and costs involved in site selection, construction and safety assessment of a final disposal facility, low-level radioactive waste can be stored in intermediate deposits, while high level waste must be stored in pools. It is worth mentioning that political and psychosocial aspects related to the subject of radioactive waste disposal (“not in my backyard” syndrome) contribute enormously to the difficulties faced by the Brazilian Government in the accomplishment of the national waste disposal policy. Because no technical solution regarding reprocessing or disposal of spent fuel has been developed in Brazil, it may require some time to make a decision, probably until international consensus is achieved.

2.10 Site Selection Regulation for Near-Surface Repositories for LLW

As previously mentioned, in 1989, CNEN published the regulation CNEN-NE-6.06 that deals only with criteria’s for site selection for near-surface deposits of low level radioactive waste. Although, no activities for the site selection for geological repositories of the high-level waste have been undertaken so far, specific criteria established by the CNEN-NE-6.06 regulation may also apply for the selection of geological repositories. These criteria are as follows: region of interest, preliminary areas, potential areas, and candidate sites. The site selection must comply with the technical analysis, selectively and sequentially, comprising various levels of detail data and information. In particular:

- The site should be located preferably on public lands
- The region of interest and preliminary areas must be identified on a regional scale
- Potential areas should be described on a less detailing meso-scale
- Local candidates must be described in detail on a local site scale

A series of technical requirements was established in the CNEN-NE-6.06 standard for site selection, including the need for ecological, socio-economic, geological, and physiographic studies. According to the Brazilian regulation, these studies should be performed by technical experts—geologists, hydrologists, physicists, geographers, engineers, ecologists, biologists, radiation protection, meteorologists, chemists, lawyers, etc. Ecological studies must include predictions of the impact of the repository operation on physical, chemical, and biological components of the environment, locally and in the near field, as well as identification of the species and ecosystem’s changes. The social-economic studies must include evaluation of demographics, jurisdiction of land and water, agricultural activities, industrial activities, transportation, information and assessment of direct and indirect benefits to the local and surrounding population.

Geological studies must include characterization of mineralogical and chemical properties of sediments, as well as tectonic and other processes that could cause the destruction of the geological repository. Physiographic studies should address collection of the information related to hydrographic, meteorological, and climatological features of the region.

2.11 Participation of the Scientific Community

Aiming to perform the safety assessment of repositories for low- and intermediate level of radioactive waste, CNEN, jointly with the Federal University of Rio de Janeiro (UFRJ-COPPE), developed a code for safety assessment of repositories. The code was developed in consonance with the IAEA standards to provide a fully transparent process of work with the scientific community. Example of a graphical user interface of the developed code MIGRAD is shown in Figure 2–9.

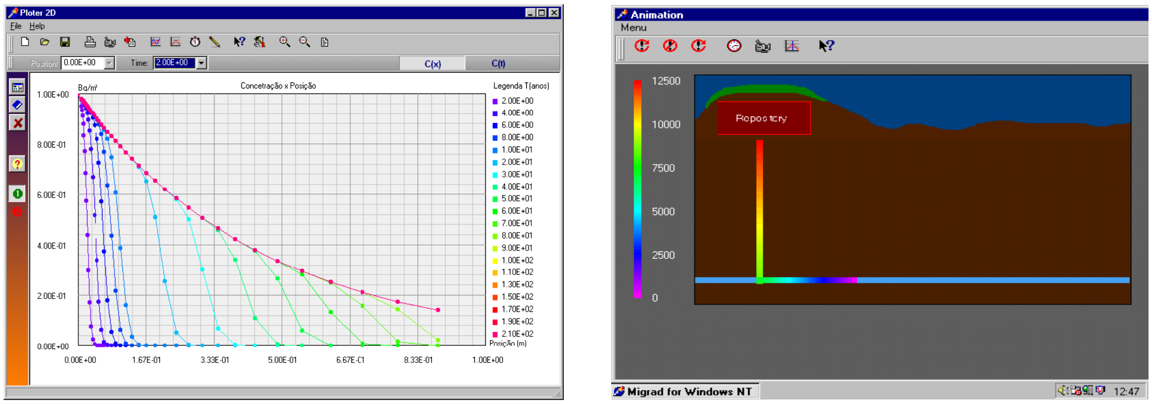

Figure 2–9. Example of the user interface of the MIGRAD code developed by CNEN-COPPE.

CNEN also participated in a project conducted by IAEA on the practical application of safety assessment methodologies named ISAM (IAEA, 2004). Overall, about 200 experts from 40 countries were involved in this project. One of the main tasks of this project was the development of the "safety cases" scenarios for three types of near-surface repositories. Accordingly, three working groups were created in the ISAM project:

1. The working group on the development of a database of various scenarios for the evaluation of different features, events, and processes (FEP).

2. The working group on modeling with the focus on (i) safety assessment model compilation, including a list of models to be used as a basis for the development of safety assessment codes and software), (ii) Parameter database, including the ranges of near-surface safety assessment parameters, (iii) Definition of modeling and data terms, and (iv) a list of codes available for safety assessment, including the analysis of their limitations.

3. The working group on confidence building with the focus on the compilation of worldwide safety indicators, regulations and safety assessment documents, gathering information about communications, materials that have been used with different audiences, and quality assurance and uncertainty.

A technical scientific working group was created in 2015, with experts from CNEN and the State University of Rio de Janeiro, in order to establish criteria for selection of a deep geological disposal site using the GIS system. The study is expected to focus initially on the state of Rio de Janeiro, where the NPP and the U enrichment plants are located.

The project, which started in April 2015 and will last for five years, is intended to study internationally recommended criteria, especially those recommended by the IAEA for the site selection for the construction of geological repositories for the safe disposal of high-level nuclear waste. The studies will be based initially on the analysis of the existing geological conditions in the state of Rio de Janeiro, using the current geological map of the state of Rio de Janeiro, scale 1: 400,000, prepared by the Geology Department of the State University of Rio de Janeiro (UERJ) (Silva and Heilbron, 2015). The site selection rating for choosing such areas will combine a set of geological, biological, physiographic, socio-economic and environmental data, which will first be integrated and analyzed by means of Geographic Information System (GIS) in order to select geological candidate areas in the state of Rio de Janeiro.

2.12 Communication and Involvement of Society

Communication is part of a process known as "Confidence Building," which includes all activities that are carried out as part of safety evaluation and communication of safety analysis results for a given facility, following a systematic safety assessment process. Confidence building focuses on three key questions:

- How to gain a level of confidence in the assessment of technical results?
- How to provide the technical results to convince the regulators to make decisions regarding a disposal facility?
- How to convince the public and other stakeholders that the impacts from a disposal facility are within acceptable limits?

In 1997, the IAEA launched a three-year coordinated research project on the Improvement of Safety Assessment Methodologies for Near Surface Disposal Facilities, which attracted an interest of about 700 experts in 72 IAEA Member States. In 2004 project results were published in a set of documents (IAEA, 2004). The results of a questionnaire survey are summarized in Tables 2–9 and 2–10. In particular, Table 2–9 summarizes the main types of audiences, percentage of respondents communicating with audiences, and the perceived importance of each audience from the point of view of the respondents.

A detailed description of methods of communicating with different audiences is given in Heilbron (2005). The survey revealed several frequently asked questions common to all audiences:

- How hazardous is radioactive waste?
- What sites have been studied?
- What is the future of the repository?
- Is the repository safe?
- What are the plans for future management of radioactive waste in the country?
- Where does the waste come from?
- What will happen in the longer term?
- Can the public put in jeopardy due to transporting the waste on public roads or through communities and towns?
- What else are you hiding from us?

Examples of topics raised by stakeholders are:

- The strategy and status of radioactive waste management activities,
- The environmental impact of radioactive waste (along with impacts on vegetation and animals)
- The public protection, licensing conditions, and plans for deep geological disposal of radioactive waste.

Table 2–9. Safety Assessments and Perceived Importance Based on the Surveys

Audiences	Respondents (%)	Perceived Importance
Regulatory Bodies	88	High
Academic and Scientific Organizations	100	Medium
The Public	94	Medium
The Media	94	Medium
Government Bodies	100	Low
Non-Governmental Organizations	88	Low

Table 2–10. Communication Methods and Tools and their Relative Efficiency (ranked on a scale from 1 to 5, with the most effective being 5)

Methods and Tools	Regulator	Academic and Scientific Organization	Public	Media	GO*	NGO**
Pamphlet, Brochure, Leaflet	2	2	4	3	2	3
Video Tapes	2	3	3	3	2	3
Visitor Centre, Facility tour,	2	3	4	4	3	4
Presentation at School	1	3	3	1	1	1
CD	3	3	3	2	2	2
Web Pages	2	2	3	3	2	2
Technical Paper	4	4	1	2	3	3
Progress Report for Government	4	3	1	3	4	2
Paid Advertising	2	2	3	3	1	3
Press Conferences	2	2	3	4	2	2

**GO-Governmental Organizations; **NGO-Non Governmental Organizations*

It was found that most organizations expend little efforts to determine the most effective ways to provide information and gather feedback from various audiences. Examples of other issues that were identified are:

- Safety cases appear to be the main method of communicating results to the regulatory authorities. This raises the question of whether the safety cases should be prepared with only the regulator in mind.
- The experts are quite good in providing technical information, but not so good at making it easy to understand. Science journalists play an important role in the communication with the public.
- In the past, people had more faith in science and technology. We must now learn to live with controversy, which means that we must be prepared to engage in a genuine dialogue.
- People living near the repository usually ask simple questions, and they know that there is no facility free of risk. Can communications play an important role showing them that the risk may be very low compared with the societal benefits?

- In some countries, operators should develop two reports, a very technical one for the competent authorities, and the other one in an accessible language for external audience's presentations/debates.

Contrary to the scientific method of the risk evaluation as the product of the probabilities of the event occurrence and the event consequences, people perceive the risk as a probability of the consequence only. The above questions can be condensed into four main groups:

1. What is "Safety" from the public point of view?
2. How can the Safety (or Safety Case) be communicated using plain language and what are supporting tools for communication?
3. What should be done to enhance communication with the public and stakeholders?
4. How should we become more trustworthy organizations?

Thus, societal participation is critical for the success of the Brazilian nuclear program.

2.12.1 On-Site Safety Assessment System

CNEN must specify an individual target risk acceptable for the final disposal of waste. This risk should have a value of less than 5×10^{-5} per year (which is equivalent to an annual dose of 1 mSv, according to ICRP-60 (1991)), and must take into consideration the following factors:

- The annual radiological risk from natural radiation;
- Possible exposures arising from future nuclear activities in the vicinity of the geological disposal site;
- Uncertainties in the estimation of future risk;
- The institutional control measures; and
- Chemical toxicity associated with the waste disposed.

The risk objective should not be less than "de minimis," i.e., a risk associated with such a small dose of radiation that it is not of concern. The values 50 μSv/a and 1 person-Sv/a have been proposed internationally for "the minimis doses" (IAEA, 1996).

2.12.2 Surveillance Program and Additional requirements for Safety Assessment

The surveillance program should complement the analysis and evaluation of safety throughout the licensing process. Using inspections and audits, this program verifies compliance for the activities related to site selection, design, construction, operation and closure requirements established by CNEN or specified in licenses, permits or standards. The program also includes the development of additional safety and security requirements, and corrective actions and sanctions. An adequate safety assessment must take into consideration the following aspects:

- Time of predictions—most of the experts are in agreement that the maximum time of predictions should be 10,000 years, which corresponds to the beginning of the next expected glacial area. These predictions will demonstrate compliance with the requirements of acceptable individual risk. The individual dose calculations should be performed using the risk (i.e., the likelihood of serious and fatal cancers) factor of ICRP 60 (0.05/Sv) (ICRP 60, 1991), instead of the early established earlier of 0.02 Sv.
- Balance between the technical and economical aspects—in selecting the optimal option for the final disposal of radioactive waste, CNEN shall use the best judgment to balance between the

need and the cost to establish safety requirements and the acceptance of a certain degree of institutional control.

Among 31 countries with nuclear reactors, only five are using bitumen for waste immobilization: Japan—Tsuruga and Mihama, Sweden—Barsebäck, Forsmak-1, 2 and 3, Spain, Switzerland—Gösgen, and Brazil—Angra-II. The waste immobilization system planned at Angra II Nuclear Power Plant uses bitumen. From 433 reactors in operation worldwide, only 17 (4%) use bitumen for the immobilization of radioactive waste of low and medium levels of radiation. Special attention must be given by CNEN to this system because immobilization with bitumen requires rigorous chemical control in view of chemical reactions that present risk of fire and explosion.

The waste immobilization system at the Angra I Nuclear Power Plant must be evaluated by CNEN to verify if the criteria for waste acceptance in final disposal facilities have been met. The first safety barrier existing in near-surface deposits must have high mechanical strength, low rate of leaching of radionuclides, good self-absorption range, etc.

From the transport regulation point of view, the drums containing immobilized waste (either concrete or bitumen) must pass the tests for Type A packages provided in Standard CNEN-NE-5:01 (transport of radioactive material). The feasibility of employing external concrete packaging (VBAs) to involve drums containing immobilized waste generated in Angra I during transport need to be evaluated, since the quality assurance program for the manufacturing of these VBAs must be approved by CNEN.

2.13 Conclusions

The CNEN has the legal authority to receive all radioactive waste generated in the country, to build and operate intermediate and final disposal facilities, and to evaluate the safety and conduct inspections in these facilities. In Brazil, the Ministry of Environment is responsible for the environmental licensing of radioactive waste repositories, and requires that the operator of a final disposal facility develop an Environmental Impact Report (RIMA) to be discussed in public audiences with CNEN as well as all stakeholders involved.

Since 2001, CNEN has the legal attribution of licensing radioactive waste deposits. CNEN must give high priority to the construction of a final disposal facility for low and intermediate level waste stored in initial deposits of nuclear power plants and fuel cycle facilities, as well as in intermediate deposits of its Institutes and Centers. The idea of constructing two repositories, one for the low-level waste generated by the nuclear power plants, and another for all other waste under storage would result in a significant increase in cost, in view of the need for selection and characterization of two sites, operation of two deposits, licensing of two deposits, etc. Furthermore, the costs associated with the acceptance of the deposit by the community would also be doubled.

CNEN should begin immediately the elaboration of the High Level Waste site selection regulation as well as the establishment of an internal group of experts within the waste Management Coordination (COREJ) responsible for the complete licensing process of the disposal site and another expert group in one of our research institutes (responsible for the selection, construction and operation of the future geological disposal site).

The site selection process for radioactive waste repositories must avoid fractured zones, due to the complexity of modeling for safety analysis, primarily because of preferential pathways available to migration of radionuclides to the environment. Furthermore, there are minimum safety requirements established internationally that must be met, such as: (i) the site must be characterized and modeled mathematically, (ii) the site cannot be located in an area where tectonic processes occur that could cause disruptions on the ground, (iii) the site must be far from facilities or activities (e.g., nuclear plants) likely to adversely affect or distort significantly the environmental monitoring program; (iv) the geological terrain, saturated or not, should be able to prevent or retard the migration of radionuclides from the repository to groundwater. Such water should not flow readily to waterways, aquifers potentially usable by the public or areas of highly permeable fractured rocks.

Special attention should be given to the establishment of safety requirements for initial storage of radioactive waste from mining and processing of conventional ores containing radionuclides from the uranium and thorium series, since these storage facilities may be converted into final deposits.

The Agenda 21 Program, that resulted from the well known Earth Summit (Rio-92), which brought together 102 heads of states, established in its chapter 22 that "Governments should refrain from promoting or allow the storage or disposing of radioactive wastes near the marine environment, regardless of their level of radioactivity." It should also be emphasized that the Brazilian president signed, on 26 January 1997, a decree creating the Agenda 21 Sustainable Development Commission. Therefore, the idea of constructing a final disposal facility in the Angra dos Reis region for the low level waste should be carefully analyzed.

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2.15 Acronyms

CDTN—Centro de Desenvolvimento da Tecnologia Nuclear

CNEN—Brazilian Nuclear Energy Commission

COREJ—Coordenação de Rejeitos—Waste Management Department

DPD—Diretoria de Pesquisa e Desenvolvimento

DRS—Diretoria de Radioproteção e Segurança Nuclear

EW—Exempt Waste

FAPERJ—Fundação de Amparo à Pesquisa do Estado do Rio de Janeiro (http://www.faperj.br/)

GIS—Geographic Information System

HLW—High Level Waste

IAEA—International Atomic Energy Agency

IBAMA—Instituto Brasileiro do Meio Ambiente e dos Recursos Naturais Renováveis

IEN—Instituto de Engenharia Nuclear

ILW—Intermediate Level Waste

INB—Indústrias Nucleares do Brasil, Nuclear Brazilian Industry

IPEN—Institudo de Pesquisas Energeticas e Nucleares, or Brazilian Institute of Environment and Renewable Natural Resources

LILW—Low an Intermediate Level Waste

LILWA-LL—Long-Lived Waste

LILW-SL—Short-Lived Waste

LLW—Low Level Waste

NORM—Naturally occurring radioactive materials

PWR—Pressurized Water Reactor

RIMA—Environmental Impact Report

UERJ—State University of Rio de Janeiro

VBA—External concrete packaging

VLLW—Very Low Level Waste

VSLW—Very Short Lived Waste

Chapter 3

Site Selection Approach to Geological Disposal of High-Level Waste in Bulgaria

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ABSTRACT: Since the Fourth Worldwide Review in 2006, the State Enterprise Radioactive Waste (SE RAW) has carried out a comprehensive preliminary analysis of geological conditions in Bulgaria with the purpose of selecting a potential site suitable for the construction of a deep geological disposal facility for high level nuclear waste (HLW). The developed methodological framework included a step-by-step analysis and evaluation of the Bulgarian territory for selecting potential host rocks and a further detailed survey to study options to house HLW disposal in the deep geological environment. On the basis of multi-criteria comparative analysis, five potential sites with respective host rocks have been selected for further consideration. In four of these sites the potential host rock is Lower Cretaceous clayey marl, in one—Miocene clay. The present report summarizes the results of the preliminary site selection. First, the methodology applied for the selection of potential sites and their respective host rocks is described. Then, a short overview of the five candidate sites is given. We conclude that according to the preliminary analysis, there is potential for development of a deep geological HLW repository in Bulgaria.

3.1 Introduction

In Bulgaria, under the present strategy for management of high-level waste (HLW) and spent nuclear fuel (SF), geological disposal is considered to be a suitable option to permanently guarantee safe isolation of HLW and intermediate-level long-lived radioactive wastes (ILW). Disposal at depths of several hundred meters is expected to be the most ethical, sustainable, and safe approach to the management of these types of waste. Construction of a long-term interim surface repository for HLW and ILW with a period of administrative control of at least 100 years is considered optimal. It is believed that during this period of controlled storage new data and technical solutions will be obtained, which may significantly change the methods for HLW management. At the same time this will make it possible to avoid serious errors in the final disposal in a geological formation (Strategy 2011).

Although it might be desirable to extend the duration of interim storage of SF and HLW for even longer, the principle of equity between generations requires that the present generation take responsibility for the wastes generated by activities that they have benefited from. It is therefore necessary to continue the efforts for construction of a deep geological repository.

Since the Fourth Worldwide Review in 2006, the State Enterprise Radioactive Waste (SE RAW) has completed a comprehensive preliminary analysis of the geological conditions in Bulgaria to identify potentially suitable sites for construction of a deep geological disposal facility. This analysis includes the zonation of the Bulgarian territory and identification of several candidate sites for further detailed survey and a study of the host geological environment for deep HLW disposal. The analysis was carried out by a multi-disciplinary team of experts from several Institutes of the Bulgarian Academy of Sciences under the guidance of The Geological Institute (Karastanev et al. 2011). This analysis and the assessment of options for construction of a deep geological repository is based on a critical review and compilation of the results of previous surveys and studies of the natural (primarily geological) conditions of the territory of the country, and on the analysis of relevant socio-economic factors.

The present report summarizes the most important results of the preliminary site selection analysis (Karastanev et al. 2011). First we describe the methodology for selecting potential sites and respective host rocks. Then we give a short overview of the five preferred potential sites.

3.2 Methodology of Preliminary Site Selection Analysis

3.2.1 Basic principle of the main methodological approach

The basic principle for selecting a suitable site is that the host rock should possess, to the highest possible extent, favorable qualities of a natural barrier against radionuclide migration. The natural conditions of the host geological formation, together with the engineered structures, should ensure reliable isolation of radionuclides over a time period long enough for their decay to safe levels. The geological setting of nuclear disposal sites is considered to be the major natural barrier against radionuclide migration. The repository structures, and the amount and a method of packing of wastes (so-called engineered barrier) must be compatible with the characteristics of the host formation and the surrounding geoenvironment. The overall system for geological disposal should provide adequate safety and guarantee isolation of radionuclides from the biosphere for at least 100,000 years.

To comply with this basic principle, the host rock must meet to the maximum extent the following main *preferable conditions, characteristics and requirements* with respect to:

i. Geological-tectonic structure:

   a. clear and indisputable geological-tectonic structure; spatially homogeneous lithology; clear paths of possible radionuclide migration to the biosphere;

   b. host rock, situated within the range of one tectonic unit, relatively stable throughout the longest possible time period (at least with attenuated tectonic processes during the Late Neogene and the Quaternary); formation not disturbed by active faults and geodynamic zones; low enough potential for adverse neotectonic events to ensure the stability of the geological environment around the repository.

ii. Hydrogeological conditions:

   a. host rock, composed of non-fractured and water impermeable rocks, practical absence of groundwater flow, mass transfer that is realized only or primarily by diffusion processes at molecular level;

   b. simple hydrogeological conditions susceptible to reliable modeling and prediction of the paths of possible radionuclide migration.

iii. Geochemical and physical characteristics – the geochemical and mineral composition of the host rock, as well as the thermochemical and thermophysical properties of the rock should guarantee:

   a. long-term stability of the geological and engineering barriers;

   b. effective and long-term functioning of the entire system for disposal and isolation of radionuclides from the biosphere;

iv. Engineering geological and mining geotechnical conditions:

   a. allowing the construction of the repository using technologies that are established in practice and cost-effective;

   b. ensuring the technological possibility of retrievability of the disposed RAW;

v. Sustainability of the natural environment – long-term sustainability of the natural environment, which guarantees preservation of the integrity of the potential host formation for 100,000 years and more; in this context preference is given to formations, located in:

   a. zones with absence of sharp gradients and anomalies of the gravitational, magnetic and geothermal fields, as well as anomalies in the regional stress field;

   b. terrains with stable geomorphological conditions that have not been subjected to dramatic changes during the Quaternary;

   c. terrains with low seismicity (seismic acceleration $a \leq 0.15$g);

   d. terrains with absence of hazardous geological processes as karst, volcanism and diapirism, landslides, rock-falls, intensive weathering, erosion, etc.;

   e. terrains with absence of floods and other adverse hydrological phenomena.

vi. Potential anthropogenic impacts:

   a. the location of the host rock should prelude disturbance of the natural and engineered barriers by current or future human activities;

vii. Socio-economic conditions:

   a. the public acceptability of the selected site together with the entire repository system is a crucial primary condition for implementing HLW and SF disposal;

   b. preference is given to areas near nuclear facilities, where local citizens are used to the presence of nuclear facilities, distant from state borders, and with lower economic importance (including water, mineral and other resources), as well as lower density of population.

The Bulgarian territory has been analyzed and evaluated based on a step-by-step approach recommended by the IAEA (2006) and OECD-NEA (2004), which is depicted in Figure 3–1.

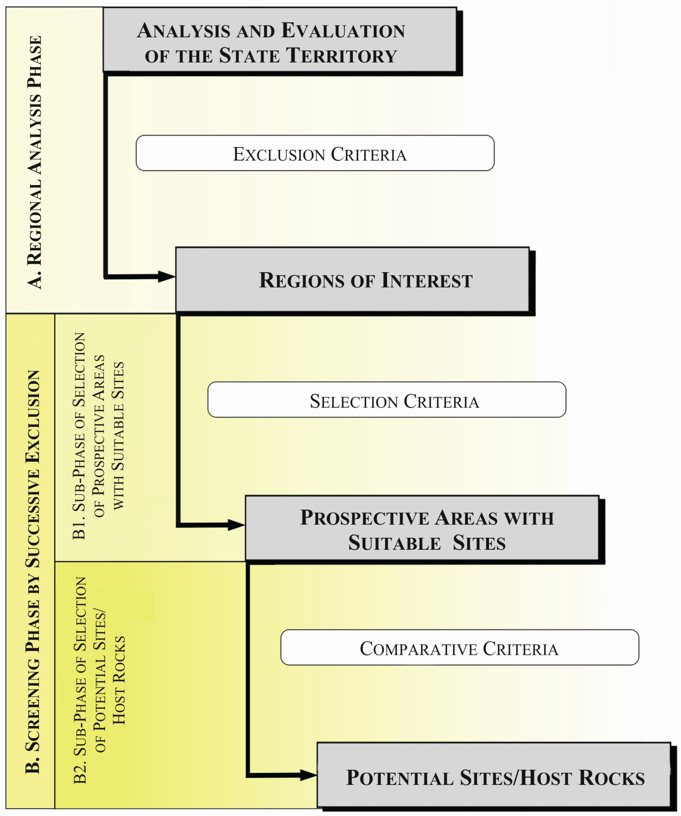

Figure 3–1. Methodological flow chart of selection of potential host rocks

3.2.2 Site Selection Analysis Phases

The preliminary site selection analysis was implemented in two phases: *A. Regional Analysis Phase* and *B. Screening Phase by Successive Exclusion* as suggested in the safety guide of IAEA (1994).

3.2.2.1 Regional Analysis Phase A

Phase A includes analyses and evaluations of the whole state territory in scale 1: 500,000. Based on a set of exclusion criteria, large areas with unfavorable conditions for selecting potential geological blocks for HLW disposal are eliminated. The remaining regions of interest (with an area exceeding 500 km^2) represent territories with more favorable geological-tectonic, seismic, hydrogeological, engineering geological, geomorphological (topographic), socio-economic and other characteristics.

Table 3–1. Site Selection, Regional Analysis Phase A: Exclusion Criteria

Category	Criterion for Exclusion	Reason
Geological-Tectonic	Changeable and inhomogeneous lithology	Greater uncertainty in simulations of RN transport for safety assessment
	Fold and thrust	
	Active tectonism	
	Significant past tectonism	
Seismic	Predicted MSK degree IX during 10,000 years	Seismicity could disrupt natural barrier
	Seismic acceleration $\geq$ 0.4 g during 10,000 years	
Hydrogeological	Host rock in contact with aquifer thickness > 200 m and aquifer depth < 1 km	Protect groundwater quality
Engineering Geological	Presence of karst or karst-suffusion process	Could disrupt engineered or natural barrier
	Volcanism and diapirism	
	Floods or other hazards	
Geomorphological (Topographic)	Ground elevation > 600 m	Protect watersheds, rapid erosion, weathering, uncertain hydrogeochemical modeling of RN transport
Socio-Economic	Population density > 80/km^2; resort or tourist area, national park;	Legislative protection

The following exclusion criteria, considered equally important, have been applied (also summarized in Table 3–1):

GEOLOGICAL-TECTONIC – regions with complex geological-tectonic structure due to significant spatial lithological heterogeneity, with fold and thrust structures, and with active tectonic activity and significant tectonic disturbances are excluded.

<u>Reason</u>: the complex geological-tectonic structure increases the uncertainties in safety assessment and does not allow for reliable prediction and modeling of radionuclide migration.

SEISMIC – territories that have predicted seismic activity of degree IX or higher on the MSK scale (The Medvedev–Sponheuer–Karnik scale, also known as the MSK or MSK-64, is a macroseismic intensity scale used to evaluate the severity of ground shaking on the basis of observed effects in an area of the earthquake occurrence (Sponheuer and Karnik 1964) for a period of 10,000 years and seismic acceleration a ≥ 0.4g are excluded.

Reason: the seismogenic impacts and their consequences may drastically alter and geological and engineering barriers of the repository.

HYDROGEOLOGICAL – regions where the potential host rock will be in hydraulic connection with the aquifer having strategically important reserves of fresh groundwater are excluded. Also, areas where the aquifer thickness exceeds 200 m, down to 1000 m depths are excluded.

Reason: the location and structure of the geological repository should prevent the influx of inadmissible radionuclide concentrations into groundwater throughout the life cycle of the repository facility.

ENGINEERING GEOLOGICAL – regions prone to karst and karst-suffusion processes, volcanism and diapirism (after the Late Neogene), and other hazardous geological and climatic processes (including floods), which cannot be overcome by reasonable and cost-effective engineering solutions are excluded.

Reason: the post Late Neogene terrains with manifestation of volcanism and diapirism may cause the repository instability; the presence of karst causes the hazard of rapid radionuclide migration through groundwater to a great depth and over large area; the manifestations of the processes of the geological hazard (including floods) may threaten the engineering stability of the repository or at least provoke cessation of the normal functioning of the repository system.

GEOMORPHOLOGICAL (TOPOGRAPHIC) – territories more than 600 m above sea level are excluded.

Reason: the most important watersheds in Bulgaria are found in the mountain regions; these areas have highly variable relief, a high degree of development of exogeodynamic processes (erosion, weathering, etc.), large hydraulic flow gradients within aquifers; if located at a high altitude, the repository may increase the risk of more distant downward radionuclide migration.

SOCIO-ECONOMIC – the excluded territories are municipalities with density of population exceeding 80 people/km^2, resort and tourist complexes of national importance, national parks, in accordance with the legislation for protection of environment, reserves, protected areas, natural phenomena and landmarks, mine workings, and terrains near fire- and explosion-hazardous sites.

Reason: the deployment of the repository in such areas would be socially and economically unacceptable.

Based on the exclusion criteria listed in Table 3–1, large areas of the territory of the country have been rejected as unfavorable (non-prospective) and regions of interest have been distinguished, which are shown on the 1:500,000 map (Figure 3–2).

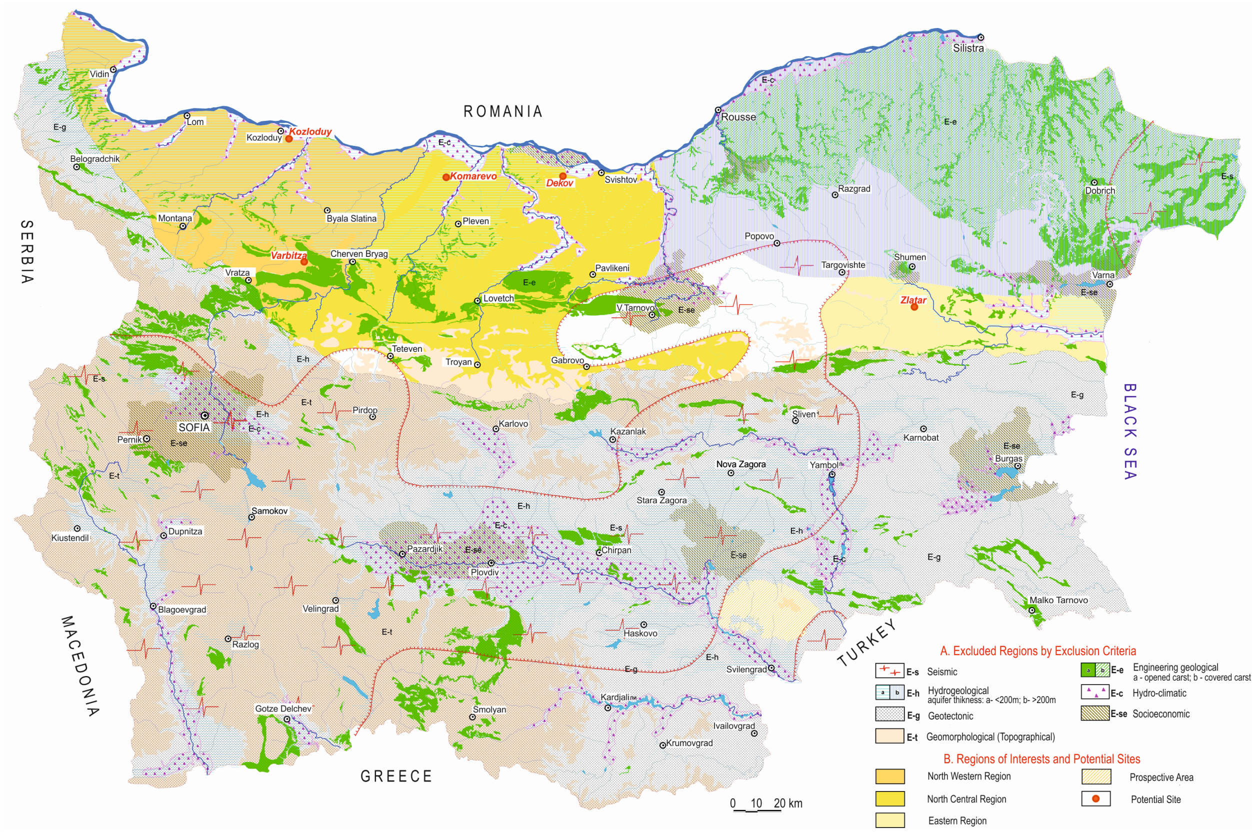

Figure 3–2. Map showing territory of Bulgaria excluded in Phase A, and location of potential sites (after Karastanev et al. 2011)

3.2.2.2 Screening by Successive Exclusion—Phase B

Phase B consists of two sub-phases (Figure 3–1): B1 - Sub-Phase of Selection of Prospective Areas with Suitable Sites, and B2 - Sub-Phase of Selection of Potential Sites/Host Rocks.

3.2.2.2.1 Selection of Prospective Areas with Suitable Sites Sub—Phase B1

Five prospective areas have been identified (in scale 1:100,000) within the regions of interest, determined in Phase A, each between 100 to 500 km². These prospective areas meet to the highest extent the basic preferable conditions, characteristics and requirements (see Section 3.1 above). More than 30 suitable sites (each approximately 4-6 km²) have been identified within these prospective areas on the basis of the following selection criteria:

i. The geological structure should provide maximum isolation of the wastes and restrict radionuclide migration to the biosphere; in particular this applies to the following:

   a. preference is given to formations with simple geological-tectonic structure, characterized by uniformity and homogeneity of the lithological varieties in vertical and horizontal direction, and with clear paths of radionuclide migration to the biosphere to enable reliable modeling of the mass transfer processes;

b. the geochemical characteristics and mineral composition of the host rock should favor the retention of radionuclides; in this context evaporites (rock salt, anhydrite), argillaceous rocks (clay, clayey marl, schist), hard non-fractured rocks (granite, basalt, gneiss, tuff, gabbro) are considered suitable;

c. the thermochemical and thermophysical parameters of the host rock should ensure the long-term stability of the geological and engineered barriers;

d. geodynamic processes affecting the host formation should have attenuated in the Late Neogene and the Quaternary; the formation should not be disturbed by active faults and geodynamic zones; the potential for adverse neotectonic events should be low enough to ensure invariability of the geological environment around the repository. Therefore preference is given to areas with stable neotectonic circumstances, and without seismic activity, active faults, or sharp gradients or local anomalies of the gravitational, magnetic and geothermal fields, or regional stress field.

ii. Preferred sites are located within the area with the degree of VIII or lower seismic intensity according to the MSK scale for a 10,000-year period and seismic acceleration a ≤0.15g; zones considered as unfavorable possess IX or higher degree of predicted seismic intensity according to the MSK scale, according to the shakeability map for a period of 10,000 years (Boncev et al. 1982) and $a \geq 0.27$g;

iii. Preference is given to formations built on non-fractured and water impermeable rocks practically absent of groundwater; with simple hydrogeological conditions, susceptible to certain modeling and prediction of the migration paths;

iv. The mining and geotechnical conditions should ensure the construction of the repository using accessible and proven technologies that will guarantee safe operation of the system; homogenous host rock is preferable, built of non-fractured rock or semi-rock varieties, or consolidated dense clays with good sorption parameters;

v. The preferable areas are with stable geomorphological conditions, which have not undergone dramatic changes during the Quaternary Period;

vi. The sites are situated in such a manner that there is a very low probability of:

a. manifestation of hazardous geological processes, as well as other unfavorable natural phenomena, and

b. disturbance of the geological and engineering barriers as a result of current or future human activities should be negligibly low;

vii. The information about the meteorology and hydrology in the region is sufficient to adequately predict the influence of the changes in climatic on hydrological conditions;

viii. Preference is given to sites near present and future nuclear facilities, which will facilitate the socio-economic acceptability; in addition this minimizes RAW transportation risk for the population and environment;

ix. Priority is given to well-studied areas, requiring a minimum of additional geological and other explorations and analyses.

3.2.2.2.2 Selection of Potential Sites with Respective Host Rocks Sub-Phase B2

During this sub-phase the 30 sites screened in by sub-phase B1 were further analyzed against 27 criteria (Figure 3–3). At this stage of preliminary site selection, the data relevant to each criterion were collected and analyzed. Data from previous investigations conducted on various occasions, but mainly for oil and gas explorations, presented in numerous reports and scientific publications, thematic maps (geologic, tectonic, seismic, hydrogeologic, engineering geologic, hydrological, climatic) in different scale, statistical reference books, and others were obtained. (Note: The list of references in Karastanev, ed., et. al., 2011 includes citations from 211 published documents -- monographs, papers, maps, etc., and 98 unpublished reports of geological explorations or other materials.) On the basis of this multi-criteria comparative analysis, five potential sites with respective host rocks have been selected (Karastanev et al. 2011) for further consideration – Varbitza, Kozloduy, Dekov, Komarevo and Zlatar sites.

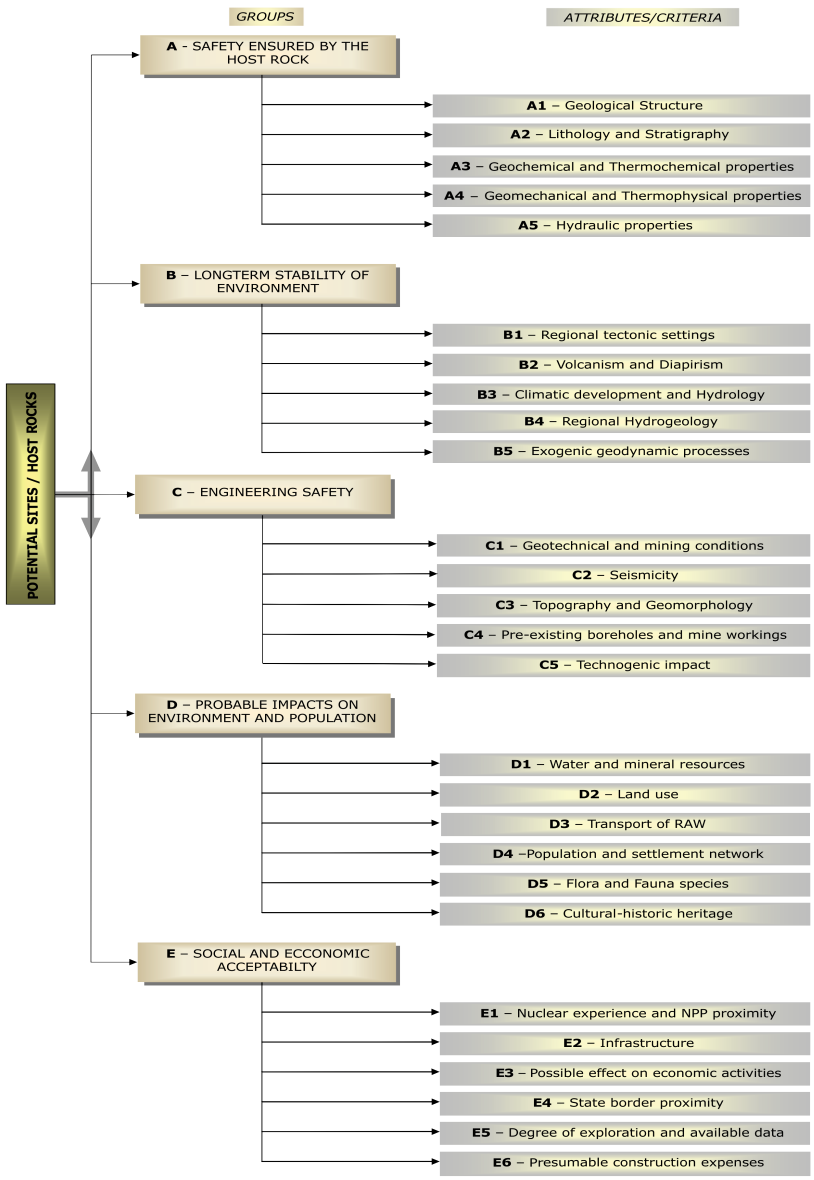

Figure 3–3. Groups and attributes of multi-criteria comparative analysis for selection of potential sites and host rocks

3.2.2.3 *Overview of the Potential Sites and Respective Host Rocks*

The location of the potential sites is shown in Figure 3–2. All five potential sites are situated in Northern Bulgaria, which is part of the so-called Moezian Platform. This is a stable tectonic structure enclosed between the Carpathians and the Balkanides, where thick terrigenic-carbonate sediments were deposited. The basin has changed with time and transformed from an internal marine basin into an internal continental one. Major tectonic activities in this structure ceased during the Jurassic, i.e. 150 million years ago. Since then, the Moezian Platform has been subjected mainly to fluctuating movements. The rest of the country falls within the Alpine-Himalayas orogeny where the contemporary seismotectonic processes are generally much greater.

Four sites – Kozloduy, Komarevo, Dekov and Varbitza, are located in Western North and Central North Bulgaria, seismically the most calmest territory of the country, without contemporary and historically documented earthquakes with M≥4.0 (Figure 3–4). The Zlatar site is located in the southern periphery of the North Bulgarian Arch, where higher seismic activity is observed. The active faults are established within the range of the Alpine orogenic belt, situated around the Moezian platform.

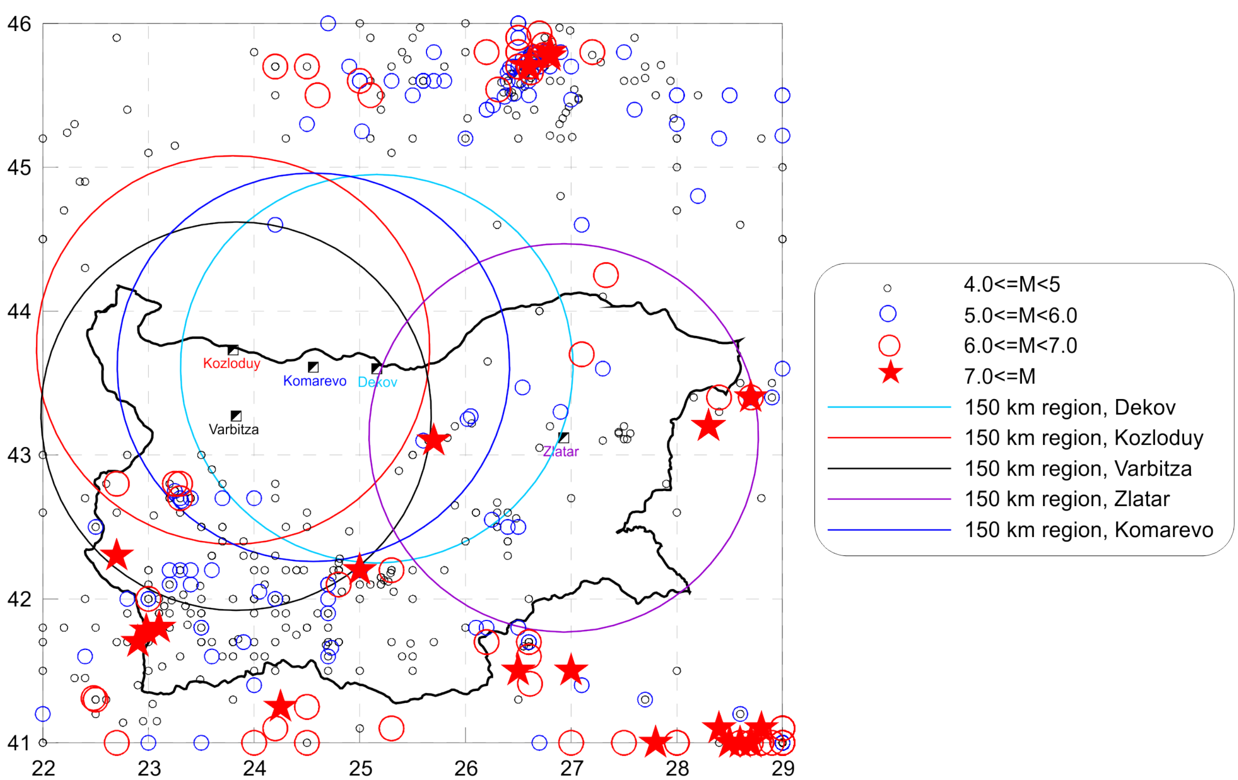

Figure 3–4. Regional seismicity around the potential sites – earthquakes of magnitudes M≥4.0 (after Karastanev, et al. 2011)

3.2.2.4 *Varbitza site*

The host rock is Lower Cretaceous clayey marl (Sumer Formation – *smK₁ap*) with a thickness at the site of about 1000 m starting from the surface (Figure 3–5). The most appropriate depths for a geological disposal of HLW is between 350 m and 500 m. The site is situated on a Pliocene denudation surface at the altitude of 310-330 m and sloped 3-4º to the north-northeast. The distance to the Kozloduy Nuclear Power Plant (NPP) along the existing road network is 90 km.

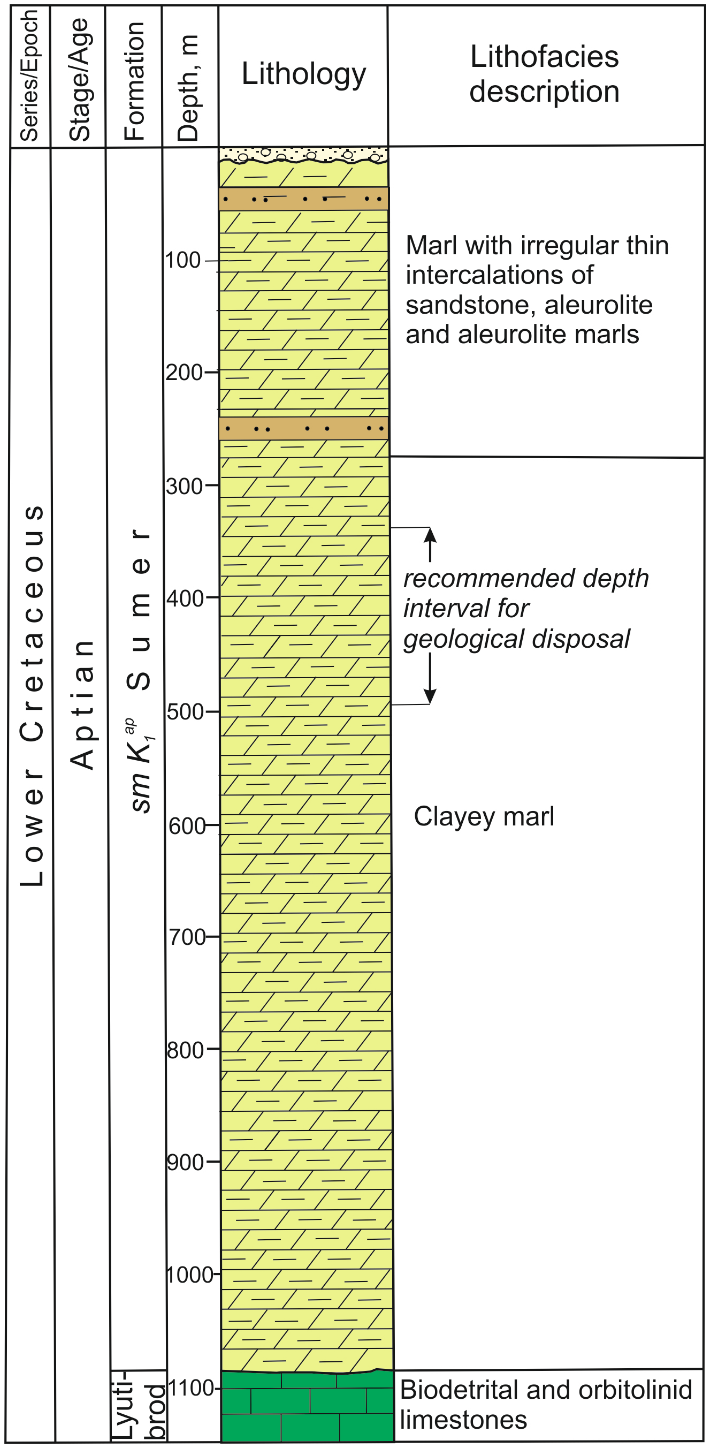

Figure 3–5. Lithostratigraphic column in the area of Varditza site (after Karastanev et al., 2011)

The Sumer Formation is largely homogeneous geologic media. It is composed of hard clayey marls with rare intercalations of sandstones, encountered mainly in the upper part of the profile. At depth they are rarer, thinner, and spatially irregular. The marls have a very dense structure, built of silt particles with clayey-carbonate cementing. Their carbonate content is about 20% and the clayey content is about 30%. The unconfined compressive strength is higher than 10 MPa (Kamenov and Iliev, 1963).

This site is suitable from a hydrogeological point of view, with practically no water-bearing formations in the Sumer Formation and the presence of non-fractured water-impermeable sediments. The Sumer Formation is covered by an impermanent formation within the uppermost weathered part of the marls. An aquifer is present below the depth of 1000-1100 m in the Lower Cretaceous limestones, which has so far no practical importance.

In general, the construction of the repository chambers at depths of several hundred meters will likely not be difficult from geotechnical and mining-technologic viewpoints.

3.2.2.5 *Kozloduy Site*

The host rock is the Miocene clay of the Smirnenski (*smN₁*) and Krivodol (*krN₁*) Formations with a thickness of about 600 m starting from the depth of 130 m (Figure 3–6). The most appropriate depths for geological disposal of HLW are between 300 m and 500 m. The site is situated about 2 km to the southeast of the Kozloduy NPP on the non-flooded three-loess river terrace. The land surface has an altitude of 45-60 m and is slightly sloped (2-3˚) to the north-northeast.

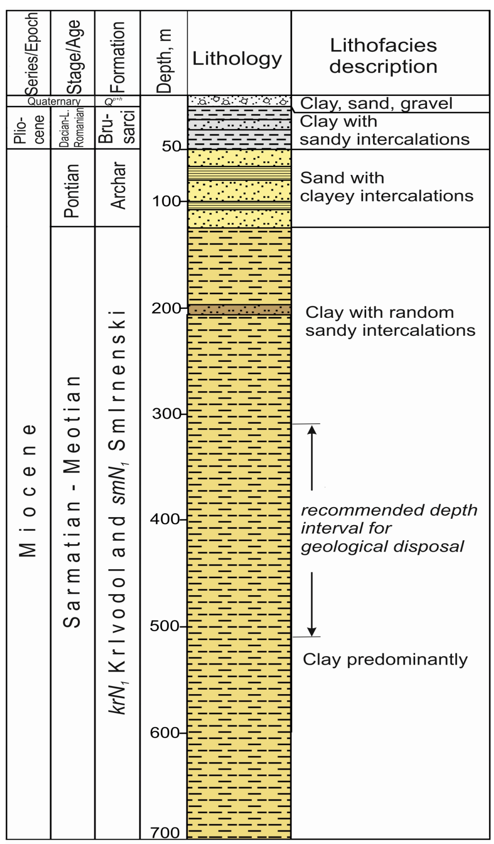

Figure 3–6. Lithotratigraphic column in the area of Kozloduy site (after Karastanevet al., 2011)

The host rock has heterogeneous structures on a macro level in vertical and horizontal direction. Sand layers unevenly intercalate thick packs of clay with a thickness of up to several meters (Figure 3–6). The Smirnenski Formation and the upper part of the Krivodol Formation are accepted as the most suitable interval for the construction of a geological repository. The clays are dense, with predominant clayey fraction content compared to the silty and sandy fractions.

The hydrogeological conditions are relatively complex. Aquifers are found above the Smirnenski Formation in the alluvial deposits and especially in the sands of the Archar Formation, which are about 80 m thick in the area of the Kozloduy site. Based on the results of prior investigations, an aquifer is located at depths of about 1000 m and deeper in the limestone and sand layers of the Middle Eocene.

Compared to the other host rocks, building a geological repository in the Miocene clay of the Kozloduy site at depths exceeding 300-400 m will probably require an application of special and expensive mining-construction technologies. The main advantage of this site is its immediate proximity to the Kozloduy NPP.

3.2.2.6 Dekov site

The host rock is Lower Cretaceous marl (Trumbesh Formation – tK_1^{ap}) with a thickness of about 500 m (Figure 3–7) starting from the surface. The most appropriate depths for geological disposal of HLW are between 300-500 m. The site is located on the second non-flooded terrace of the Danube River with an altitude of 36-40 m. The distance to the Kozloduy NPP along the existing road network is 160 km. In the event of construction of the Belene NPP, the advantage of this site is its proximity to the new plant (about 10 km by road).

The marls of the Trumbesh Formation are dense, with low porosity and permeability, and with massive structure. The marl layers are embedded subhorizontally. They have irregular intervals of thin sandstone and limestone intercalations, mainly in the upper levels of the host geological block. The marls are covered by relatively thin Quaternary alluvial and Aeolian deposits, containing the aquifers in alluvial deposits of the second non-flooded terrace of the Danube River.

The Razgrad Formation is embedded in depth under the Trumbesh Formation. Its upper part is composed of clayey marls (Figure 3–7). Lower Cretaceous limestones are found under the sediments of this formation, which are part of the highly productive Upper Jurassic-Lower Cretaceous aquifer, developed at depths between 800 m and 1600 m. This aquifer is of high practical importance on a regional scale. The Hauterivian-Barremian and a great part of the Aptian deposits (Razgrad and Trumbesh Formations) are considered to be the upper aquitard for the Upper Jurassic-Lower Cretaceous aquifer.

The unconfined compressive strength is evaluated to be higher than 10 MPa (Techn. design, 1974). The geotechnical and mining-technological conditions will not hamper the construction of the repository chambers to a depth of several hundred meters.

3.2.2.7 Komarevo Site

The host rock at this site is also Lower Cretaceous marl of Trumbesh Formation – tK_1^{ap}, which is situated at depths between 300 m and 1000 m (Figure 3–8). The site is situated in the generally flat terrain between the Iskar and Vit Rivers at an altitude of 120-130 m. This site has the least population density nearby. The distance to the Kozloduy NPP along the existing road network is 90 km.

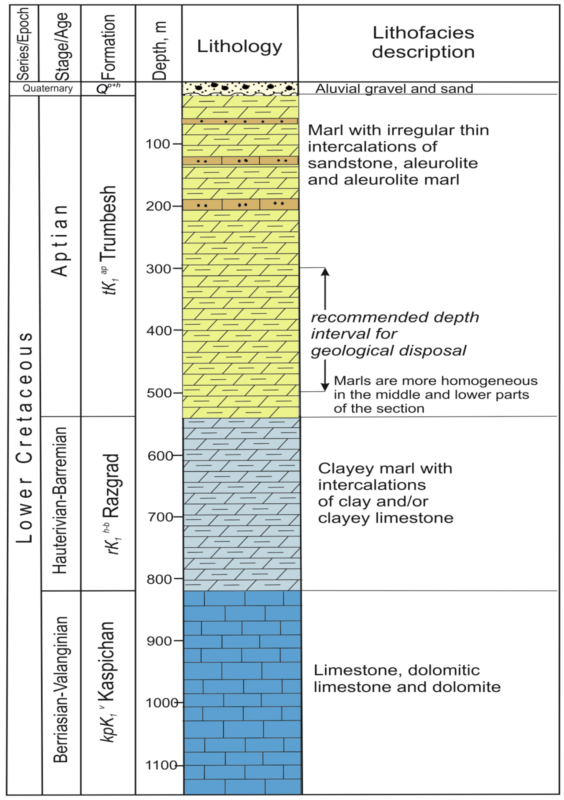

Figure 3–7. Lithostratigraphic column in the area of Dekov site (after Karastanevet al. 2011)

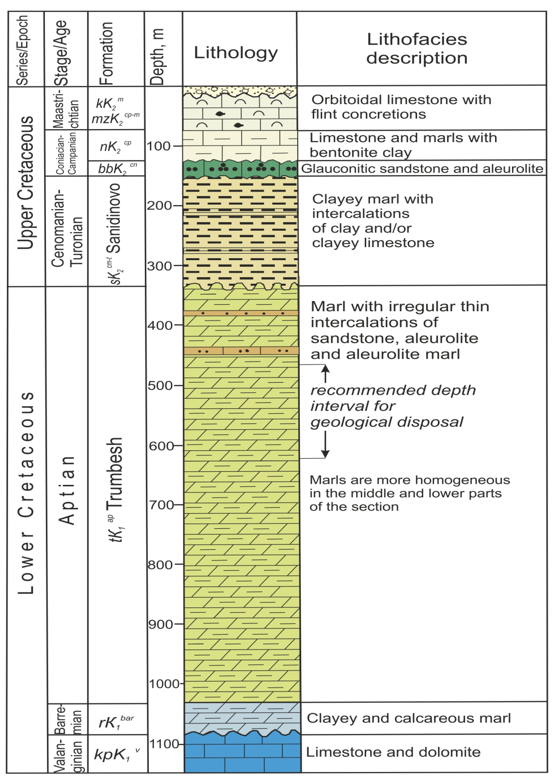

Figure 3–8. Lithostratigraphic column in the area of Komarevo site (after Karastanev et al. 2011)

The host rock—the marls of the Trumbesh Formation, are relative homogeneous, especially in the lower and middle parts of the section below a 450 m depth (Figure 3–8). Their mineral composition consists mainly of clayey minerals, quartz, and calcite in the form of carbonate cementing. Thin layers of loose sandstone and aleurolite intercalate them irregularly. The sandstone intervals are probably not interconnected and are isolated by dense marl packs. The marls have low porosity and permeability, and possess good isolating properties. The layers are bedded horizontally and sub horizontally, so that vertical migration of fluids is expected to be insignificant. The marls of the Trumbesh Formation are more than 700 m thick. Taking into account the thickness of the underlying marls of the Razgrad Formation and the overlying marls of the Sanidinovo Formation, the total thickness of the potentially hosting formation will be about 1000 m (Figure 3–8). Lower Cretaceous limestones are embedded at depths below 1100 m. They are a part of the highly water abundant Upper Jurassic-Lower Cretaceous aquifer. In the upper parts of the geological section calcareous sediments are situated, which are fractured, karstified and with relatively complex hydrogeological conditions. In general, the mining-technological conditions of this site are worse than those of the "Dekov" site, because the suitable depth interval for geological disposal is deeper (below 450 m).

3.2.2.8 Zlatar Site

The host rock is Lower Cretaceous marl of the Gorna Oryahovitsa Formation ($gK_1^{h\text{-}ap}$), which is outcropped in a vast area of about 200 km² and has a thickness of more than 600 m (Figure 3–9). The average altitude of the site is about 290 m. The relief is typical for marl terrains – soft, with rounded low hills and relatively wide and shallow valleys between them. This is the most remote site from the Kozloduy NPP – the distance along the existing road network exceeds 400 km.

The marls of the Gorna Oryahovitsa Formation are characterized by lithological homogeneity in vertical and horizontal directions. Inhomogeneous thin layers (up to several centimeters) of sandstone or aleurolite are encountered at different depths. The marls have dense structure, and composed of silt particles with carbonate cementing, clayey minerals, calcite, and fine terrigenic admixtures, including quartz, muscovite, and biotite. They are more clayey in the upper part of the formation, slightly aleurolitic, but more calcareous with depth. Illite is predominant in the clayey component.

The physico-mechanical properties of marls depend on the ratio between the clay and carbonate content. The structure becomes more massive and the strength increases with increasing carbonate content. However, the increasing clayey content enhances the sorption and retention capacity of the medium.

According to data from oil and gas explorations, the marls represent in fact a natural barrier with excellent isolation properties. They are considered as the upper aquitard of the highly water abundant Upper Jurassic-Lower Cretaceous aquifer. The hydraulic conductivity of marls is estimated to be less than 10^{-11} m/s, with no flow until the so-called threshold pressure-head gradient, which in this case ranges from 20 to 40, is exceeded (Mollov 1993).

The unconfined compressive strength of marls is estimated to be 17-18 MPa (Kosev 1978). In general it can be expected that the construction of the repository chambers at depths of several hundred meters will not encounter difficulties from geotechnical and mining-technological viewpoints.

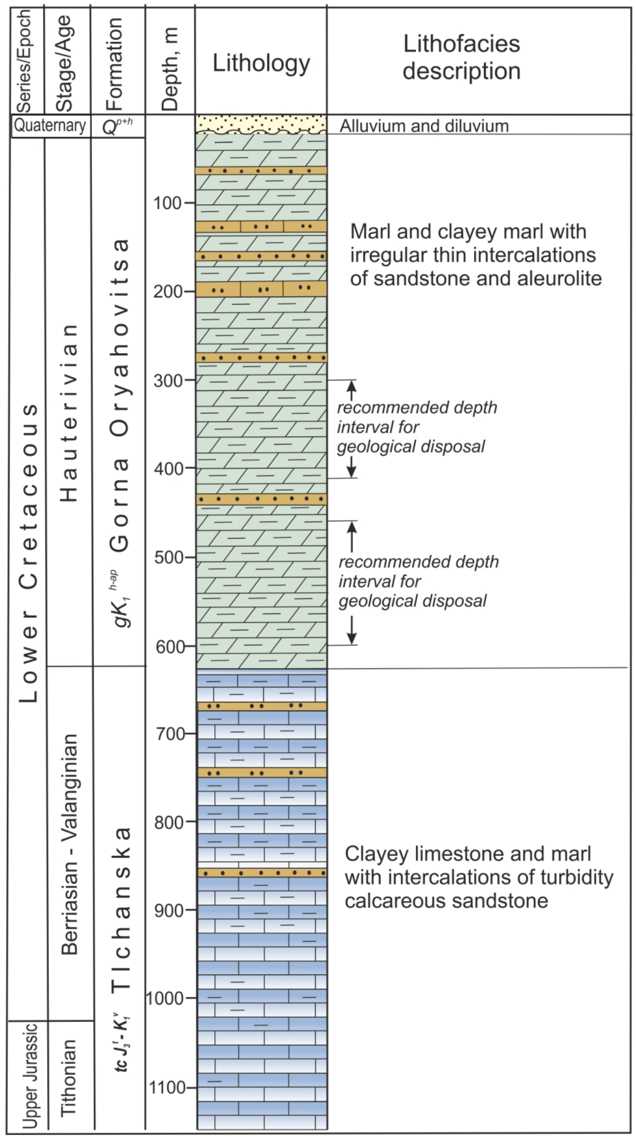

Figure 3–9. Lithotratigraphic column in the area of Zlatar site (after Karastanev et al. 2011)

3.3 Conclusions

Since the Fourth Worldwide Review in 2006, a comprehensive preliminary analysis has been conducted of the conditions within Bulgaria in order to elucidate the possibility of selecting a potential site suitable for the construction of a deep geological HLW repository. The step-by-step analysis and evaluation of Bulgaria's territory has been performed according to the methodological framework specifically developed in Bulgaria for selecting potential host rocks.

Zoning of the Bulgarian territory has been carried out on the base of exclusion criteria, corresponding to the specific geological and hydrogeological conditions of the country. An aggregated map is composed in scale 1:500,000, within which three regions of interest have been identified. Five prospective areas (each between 100 to 500 km²) have been selected in these regions, which meet to the highest possible extent the basic preferable conditions, characteristics and requirements for the construction of the geological repository.

Based on the set of 27 selection criteria, more than 30 suitable sites (each approximately 4-6 km^2) have been identified within the prospective areas. At this stage of the preliminary site selection process, the data for the selection criteria have been collected and analyzed based on the information from prior investigations, mainly from oil and gas explorations. On the basis of multi-criteria comparative analysis five potential sites with respective host rocks have been selected for further consideration:

- Varbitza, Dekov, Komarevo and Zlatar sites, where the potential host rock is the Lower Cretaceous clayey marl; and
- Kozloduy site, where the potential host rock is Neogene (Miocene) clay.

The main conclusion from the performed preliminary site selection analysis is that within Bulgaria, the possibility exists for the development of a deep geological HLW repository.

3.4 References

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3.5 Acronyms

HLW—High Level Nuclear Waste

ILW—Long-Lived Radioactive Wastes

SERAW—State Enterprise Radioactive Waste

SF—Spent Nuclear Fuel

Chapter 4

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

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ABSTRACT: Following the Government of Canada decision issued in June 2007, the Nuclear Waste Management Organization (NWMO) has been implementing Adaptive Phased Management, Canada's plan for the long-term care of its used nuclear fuel. Adaptive Phased Management is both a technical method and a management system. The approach has, as its end point, the placement of used nuclear fuel in a multi-barrier deep geological repository excavated within a suitable crystalline or sedimentary geologic setting to safely contain and isolate the waste. Since 2010, the NWMO has been leading the process to select a site for a deep geological repository, based on the identification of a willing host and development of a strong repository safety case. The NWMO has also maintained a technical program to advance methods as applied in geoscience, safety assessment and engineering design that test and enhance confidence in understanding repository safety. This chapter outlines progress toward long-term nuclear used fuel management in Canada.

4.1 Introduction

For just over half a century, Canadians have been using electricity generated by CANDU (abbreviation from "Canada Deuterium-Uranium reactor") nuclear power reactors situated in Ontario, Quebec and New Brunswick. In that time, Canada's commercial nuclear reactors have produced about 3,100 terawatt hours of electricity and just over 2.5 million used CANDU fuel bundles, which are made from naturally occurring uranium ore. A view of a typical CANDU fuel bundle is shown in Figure 4–1. Each fuel bundle is about the size and shape of a fireplace log with a total mass of approximately 24 kilograms. If used nuclear fuel bundles could be stacked like cordwood, all of Canada's used nuclear fuel could fit into about seven hockey rinks reaching from the ice surface to the top of the boards. Currently, all of Canada's used nuclear fuel is safely managed in surface facilities licensed for interim (temporary) storage at the nuclear reactor sites in Ontario, Quebec and New Brunswick, and at Canadian Nuclear Laboratories (formerly Atomic Energy of Canada Limited) facilities in Whiteshell, Manitoba, and Chalk River, Ontario.

Figure 4–1. A general view of the CANDU fuel bundle. Each fuel bundle is about the size and shape of a fireplace log with a total mass of approximately 24 kilograms.

Under the Nuclear Fuel Waste Act (2002), the Nuclear Waste Management Organization (NWMO) has a legal obligation to provide long-term management of all of Canada's used nuclear fuel that exists now or which will be produced in the future. The NWMO's mandate is to develop and implement, collaboratively with Canadians, a management approach for the long-term care of Canada's used nuclear fuel that is socially acceptable, technically sound, environmentally responsible and economically feasible.

4.2 Adaptive Phased Management

Since June 2007, the NWMO has been implementing Adaptive Phased Management (APM), the approach selected by the Government of Canada for long-term management of Canada's used nuclear fuel waste. The APM approach emerged from a three-year (2003 to 2005) dialogue with Canadians and best met the objectives that Canadians said were important – safety, security, protection of the environment, community well-being, fairness and economic viability (NWMO 2005). APM is both a technical method and a management system. The key attributes of APM, which are outlined in the final study report to Government and described on the NWMO website at www.nwmo.ca, include:

- Centralized containment and isolation of used nuclear fuel in a deep geological repository in a suitable crystalline or sedimentary rock formation.
- A series of steps and clear decision points that can be adapted over time.
- An open, inclusive, and fair siting process to identify an informed and willing host community.
- Opportunities for people and communities to be involved throughout the implementation process.
- Provision for optional temporary shallow storage at the central site, if needed.
- Long-term stewardship through the continuous monitoring of used fuel.
- Ability to retrieve the used fuel over an extended period should there be a need to access the waste or take advantage of new technologies.
- Financial surety and long-term program funding to ensure the necessary money will be available for the long-term care of used nuclear fuel.

APM will ensure that Canada's used nuclear fuel will be safely and securely contained and isolated from people and the environment in a deep geological repository using a multi-barrier system (Figure 4–2).

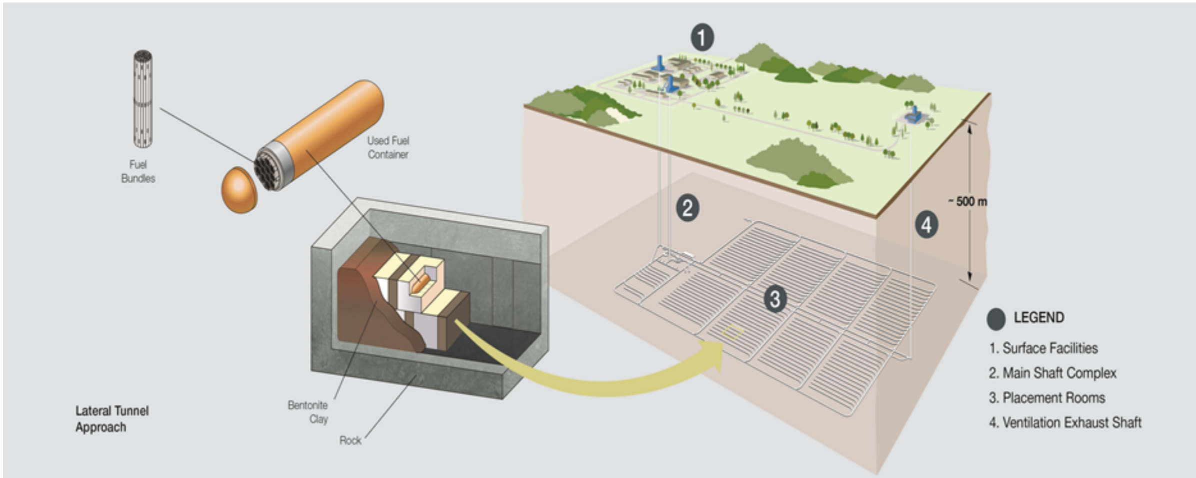

Figure 4–2. Schematic diagram of a deep geological repository.

4.3 Siting Process

The process for selecting a site for long-term management of Canada's used nuclear fuel grew out of a two-year dialogue with interested Canadians and Aboriginal peoples (2008 to 2009). Above and beyond ensuring the safety of people and the environment, those who participated in this dialogue told the NWMO that it should be a community-driven process and that the APM project should be implemented in a way that fosters the well-being of the community and area in the long term.

Canada's site selection process relies on collaboration for its implementation. The initiative to enter the process and proceed through each step must come from communities interested in learning about the APM project and the siting process. The process is designed to give communities the time and information they need to make a decision that is right for them. Throughout the site selection process, the NWMO provides resources for communities to support their involvement in the process.

There are nine steps in the APM site selection process, which are summarized in Table 4–1. Step 1 is a broadly based program to inform Canadians about APM and the process itself, which began in May 2010 (NWMO 2016) will continue over the lifetime of the project. This step is followed by increasingly detailed studies: initial screenings (Step 2), preliminary assessments (Step 3), and detailed site evaluations (Step 4).

Between 2011 and 2013, 22 interested communities completed their initial screenings (Step 2) (Figure 4–3). By the end of 2015, the first of two increasingly detailed phases of preliminary assessment in Step 3 were completed for all interested communities.

Technical studies during the initial phase of Step 3 (Phase 1) provided a preliminary examination of the potential suitability of the local geology to safely contain and isolate used nuclear fuel. These studies, which were primarily desktop based, were purposefully focused on repository safety for people and the environment as the overriding goal, and with this in mind, the questions they explore included:

- Are the characteristics of the rock at the site appropriate to ensuring the long-term containment and isolation of used nuclear fuel from people, the environment and surface disturbances caused by human activities and natural events?

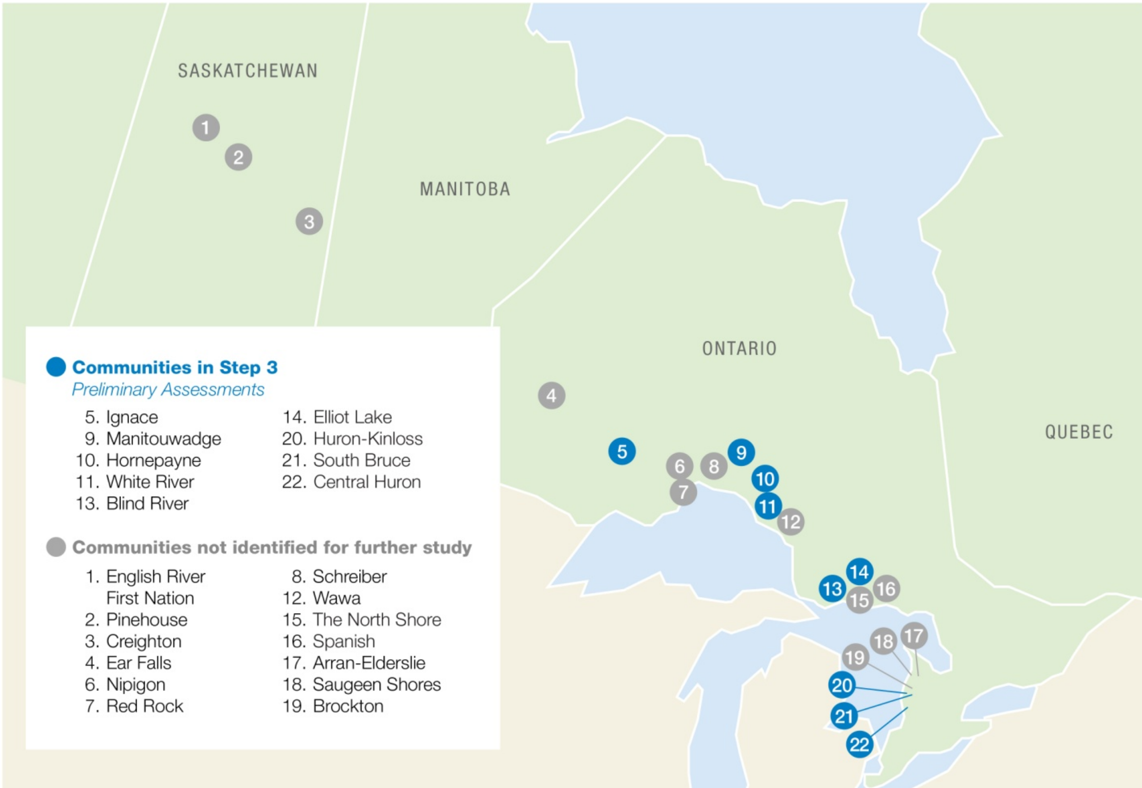

Figure 4–3. Twenty-two communities in the APM siting process, of which nine are continuing to undergo Step 3 preliminary assessments in 2016

Table 4–1. APM site selection process (NWMO, 2016)

Step 1	The NWMO initiates the siting process with a broad program to provide information, answer questions and build awareness among Canadians about the project and siting process.
Step 2	Communities identify their interest in learning more, and the NWMO provides detailed briefing and conducts an initial screening.
Step 3	For interested communities that successfully complete an initial screening, a preliminary assessment of potential suitability is conducted in two phases.
Step 4	Detailed site evaluations are completed in one site identified as having strong potential to meet project requirements in Step 3 preliminary assessments.
Step 5	Acceptance to host the repository is confirmed.
Step 6	Formal agreement to host the repository is ratified, subject to all regulatory requirements being met and regulatory approval received.
Step 7	An independent, formal, and public process is conducted under the Canadian Nuclear Safety Commission's regulatory framework to ensure that all requirements are met.
Step 8	Construction and operation of an underground demonstration facility proceeds.
Step 9	Construction and operation of the facility proceeds.

- Is the rock formation stable and is it likely to remain so over the very long term in a manner that will ensure the repository will not be substantially affected by geological and climate change processes, such as earthquakes and glacial cycles?
- Are conditions suitable for the safe construction, operation and closure of the repository?
- Is future human intrusion unlikely (e.g., exploration; natural resource development)?
- Can the geologic conditions be practically studied and described at a scale that supports demonstration of long-term safety?
- Can a transportation route be identified or developed for the safe and secure transportation of used nuclear fuel from the locations where it is currently stored?

Phase 1 of Step 3 assessment activities have also included community well-being studies that are designed to develop a better understanding of the community and how its well-being (social, cultural and economic) might be affected by the APM project. Conducted in collaboration with the community, they explore the potential for the APM project to align with values and aspirations of the community over the long term and contribute to the well-being of the community and surrounding area. Key activities in this early phase include the preparation of a community profile, discussions with the community about vision, priorities, and objectives, and initial assessment of potential to foster well-being, as defined by the community, through the implementation of the project in the area. Factors considered in assessing the potential to foster well-being, in addition to those identified by Aboriginal Traditional Knowledge, include:

- Potential social, economic, and cultural effects during the implementation phase of the project;
- Potential for enhancement of the community's and the region's long-term sustainability;through implementation of this project;
- Potential to avoid ecologically sensitive areas and locally significant features;
- Potential for physical and social infrastructure to adapt to changes resulting from the project; and
- Potential to avoid or minimize effects of the transportation of used nuclear fuel.

Phase 1 of Step 3 provided an opportunity to foster community relationship building, including Aboriginal and other communities in the siting areas. These partnerships play a crucial role in the implementation of APM.

Since 2014, the NWMO has been focusing the 2nd Phase of Step 3 on preliminary assessment activities of a smaller number of study areas. These Phase 2 preliminary assessments provide more detailed technical and safety assessments, as well as more intensive assessments of community well-being, interest and willingness for the APM project. Together, these assessments are designed to provide the NWMO and communities with an expanded understanding of potential suitability for a specific siting area to host the project, the interest in the project, and the ability for the project to be implemented in a way that fosters the area's well-being. At the same time, NWMO engagement with First Nation and Métis communities in the area, and surrounding municipalities broadens. The project will only be implemented with the involvement of the interested community, the First Nations and Métis communities in the area, and surrounding communities working in partnership.

Phase 2 of Step 3 includes technical and safety assessments focusing on geoscientific suitability, engineering, transportation, environment and safety on progressively smaller study areas. Phase 2 geoscience work involves a series of preliminary field investigations including:

- High-resolution geophysical surveys;
- Initial geological mapping (i.e., observing general geological features in the study area);

- Detailed geological mapping;
- Seismic reflection surveys; and
- Drilling and testing of deep boreholes

These initial field studies are being planned and implemented in collaboration with the interested communities, and First Nation and Métis communities. To date local knowledge holders have greatly enhanced the NWMO's planning and participated in the field mapping activities in the Phase 2 study areas. The data collected improves our understanding of the geology of the study areas. Communities undergoing Step 3 preliminary assessments in 2015 are shown in Figure 4–3.

Several more years of study will be required before a specific site is identified for consideration and detailed characterization is initiated in Step 4. The preferred site must meet robust technical requirements focused on safety. The preferred site must also be appropriate considering the social, economic, cultural and spiritual practices and preferences of those in the area.

4.4 Optimizing Repository Designs

In recent years, the NWMO has examined several different container and placement designs to represent the possible configurations of a deep geological repository. These designs and illustrations were based on international experience primarily for light water reactor used fuel.

In 2014, the NWMO completed the design of an engineered barrier system specifically created for used CANDU fuel and initiated a plan to proof-test the system's safety performance. This optimized design uses modern design concepts and tools and manufacturing methods (Hatton, 2015). The resulting engineered barrier system is robust and simple, and enhances safety to protect the environment and people over the long time periods required in a deep geological repository (see Figure 4–2).

The Canadian engineered barrier system consists of:

- A used fuel container;
- A buffer box; and
- A placement system

The Canadian used fuel container is specifically designed for used CANDU fuel and thus is smaller and lighter than the container designs of other national radioactive waste management organizations. The design was selected following the assessment of different alternatives (size / geometry / capacity), which considered engineering design, vessel manufacturability and automation, encapsulation plant operations and placement considerations in a repository.

The Canadian used fuel container has a steel shell to provide the strength needed to withstand repository loads, including glaciation. Copper is coated onto the outside of the steel bulk container to provide corrosion protection. Copper has been selected for its excellent corrosion performance in a repository environment, particularly as it is not subject to pitting in anaerobic conditions.

While the NWMO does not expect the copper to undergo any significant corrosion after placement in the repository, the coating thickness was selected after considering all possible corrosion mechanisms. Based on this analysis, the worst case scenario would involve about 1.27 millimeters of corrosion over one million years (Scully and Edwards 2013). To allow a safety factor, the Canadian used fuel container will have a coating of 3 millimeters of copper bonded directly to the steel core.

The copper-coated used fuel container is able to avoid problems associated with potential creep failure. The bulk steel container is copper coated by electro-deposition. On final assembly in a hot cell, the steel exposed weld zone is copper coated using cold spray technology and machined smooth.

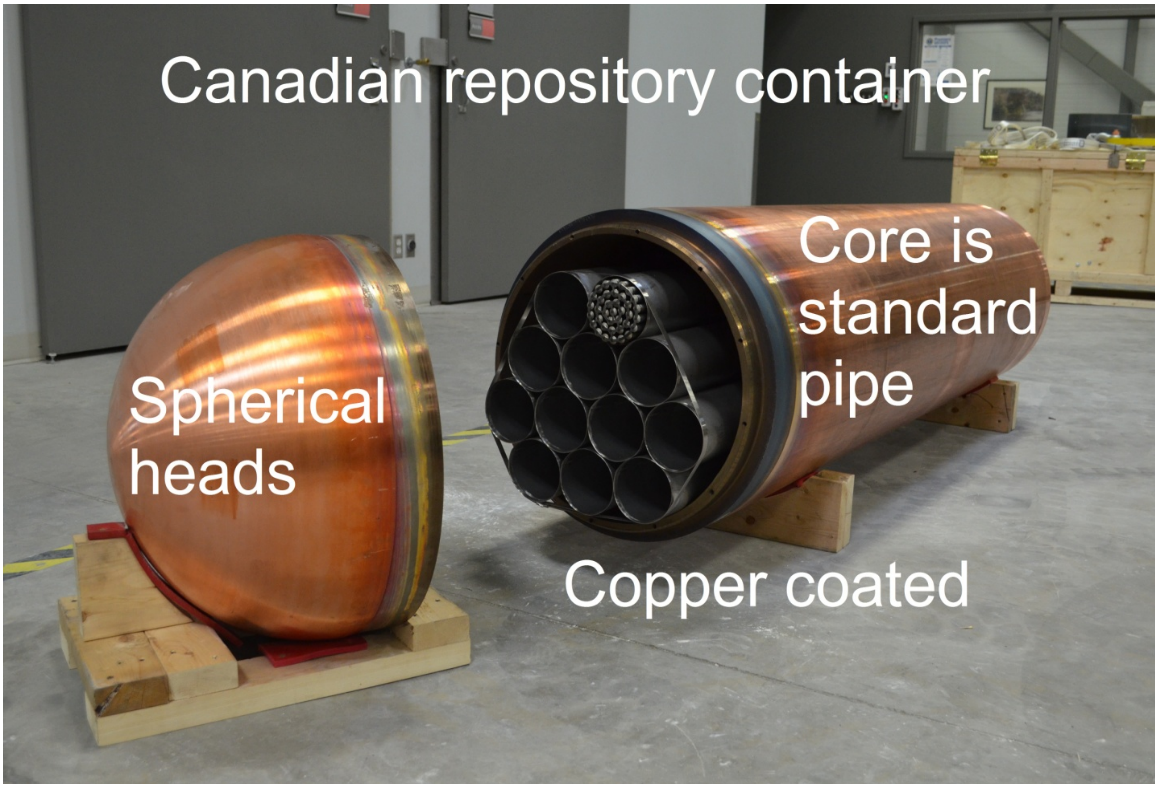

Figure 4–4. Canadian used fuel container

The Canadian used fuel container has a mass of 2.7 tons when filled with 48 used CANDU fuel bundles and is 2.5 meters in length. A full scale prototype used fuel container is shown in Figure 4–4.

Experimental studies of buffer sealing materials have shown that the use of highly compacted bentonite clay with a dry density greater than 1.6 g/cm^3 will inhibit microbial activity (Stroes-Gascoyne et al. 2007). To achieve this density, the used fuel container will be placed inside a highly compacted bentonite buffer box and enclosed in a sheet metal shell at the surface facility to prevent damage during transfer to, and placement within the repository (see Figure 4–5).

Highly compacted bentonite backfill and spacer blocks fill the void space surrounding the buffer box to effectively seal the used fuel containers within the placement room (see Figure 4–6). The dimensions of the spacer blocks can be adjusted depending on the thermal conductivity of the rock surrounding the placement room in order to limit the maximum temperature on the surface of the container to 100 degrees Celsius.

The Canadian used fuel container and buffer box assembly will be placed in the deep geological repository using simple standard equipment. The placement will be undertaken with a remotely operated wheeled vehicle, as illustrated in Figure 4–6. The buffer boxes will be stacked using slip skid technology to avoid the need to leave any placement skids inside the room (Hatton, 2015). Buffer boxes will be placed two high in a room excavated approximately 3 meters wide and 2 meters high using standard drill and blast excavation methods.

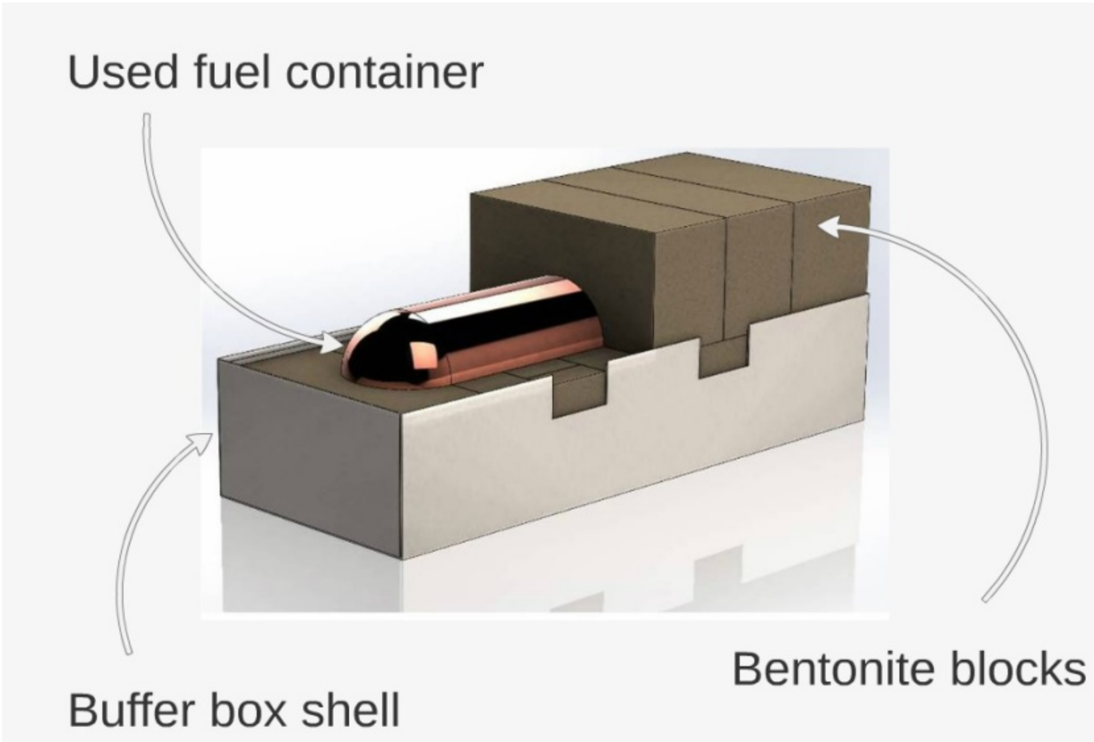

Figure 4–5. Canadian buffer box

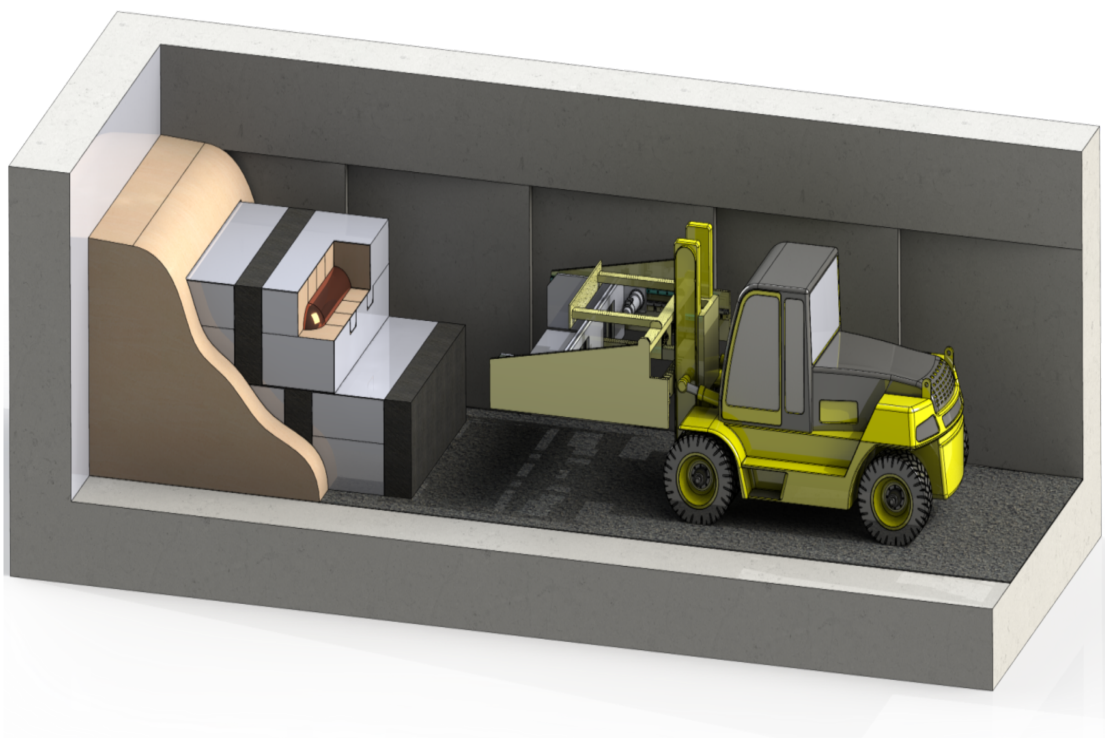

Figure 4–6. Canadian slip skid placement system

4.5 Further Increasing Confidence in Safety

The repository safety program is evaluating the long-term safety of potential candidate sites and repository designs in order to assess and improve the safety of the proposed facility. In the near term, before any candidate site has been proposed, the safety objective is addressed through the performance of post closure safety assessment case studies and continuous improvement of knowledge related to features and processes most influencing long-term repository performance.

Case studies provide feedback to the NWMO on the safety-relevant aspects of a deep geological repository for used nuclear fuel. Through identifying the factors that are important for safety, they provide, in part, direction for further work to improve repository design and safety assessment.

In 2011, the NWMO undertook an illustrative post closure safety assessment for a hypothetical repository site in crystalline rock. In 2012, this post closure safety assessment was submitted to the Canadian Nuclear Safety Commission (CNSC) for pre-project review (NWMO 2012). In 2013, the NWMO completed an illustrative post closure safety assessment for a hypothetical repository site in sedimentary rock, which was also submitted to the CNSC for pre-project review (NWMO 2013).

Post closure safety assessments are meant to show, with an appropriate degree of confidence, that the repository will remain safe over a period of one million years or more. This is approximately the time period for the radioactivity in used nuclear fuel to approach that of an equivalent amount of natural uranium. Because of the long periods involved, the NWMO's safety assessments evaluate whether the multiple engineered and natural barriers can safely contain and isolate the used nuclear fuel without human intervention.

These case studies considered varied geologic conditions that may be encountered at depth at potential repository sites, and calculated radiological and non-radiological impacts to humans and non-human biota for multiple future scenarios. These scenarios included the expected normal evolution of the repository, variance of key repository features, events and processes, and a number of disruptive events or "what-if" scenarios. Their main purpose was to show how the post closure safety assessment approach is consistent with the CNSC Guide G-320, Assessing the Long Term Safety of Radioactive Waste Management (CNSC 2006).

The NWMO's safety assessment program includes the extension of work from these two case studies and includes a preliminary assessment of preclosure or operational safety of an APM facility. These studies will provide further information on how the repository design can be improved.

4.6 International Collaboration

An important aspect of the NWMO's technical program is collaboration and interaction with national radioactive waste management organizations in other countries. The NWMO has formal agreements with SKB (Sweden), POSIVA (Finland), NAGRA (Switzerland) and ANDRA (France) to exchange information arising from their respective programs on nuclear waste management. These countries have advanced programs and are developing High Level Waste (HLW) or used nuclear fuel deep geological repository concepts in either sedimentary or crystalline rock environs. These national programs are, in particular, adept with respect to repository siting, application of site characterization methodologies, repository engineering design, in-situ repository prototype demonstrations and regulatory approvals.

A key component of the APM technical program has been international participation in applied research and demonstration projects in underground research laboratories (URLs). In this respect, the NWMO has actively participated in activities at the Äspö Hard Rock Laboratory (crystalline) in Sweden, the ONKALO Hard Rock Laboratory (crystalline) in Finland, and both the Mont Terri URL (sedimentary) and Grimsel URL (crystalline) in Switzerland.

A key motivation for involvement is to advance confidence and scientific consensus on the understanding of phenomena, both at the repository scale and the far-field scale, that most influence repository performance and implementation. The URL experience also provides a strong basis for international exchange and comparison from which shared lessons learnt aid in the refinement of reliable site characterization methodologies, the confirmation of repository technology development strategies, and the scientific basis that underpins a repository safety case.

Since 2005, the NWMO has been involved in more than 15 URL experimental or demonstration projects including:

- Gas Permeable Seal Test (Grimsel);
- Disturbances, Diffusion, Perturbation and Retention Experiment (Mont Terri);
- Deep Borehole/Porewater Characterization Experiment (Mont Terri);
- Iron Corrosion in Opalinus Clay (Mont Terri);
- Full Scale Emplacement Experiment (Mont Terri);
- Long-term Diffusion – Opalinus Clay (Mont Terri);
- Fracture Parameterization for Repository Design and Post closure Analyses (ONKALO/Äspö HRL);
- Engineered Barrier Task Force (Äspö HRL); and
- LASGIT (Äspö HRL)

For example, the Mont Terri URL located in St-Ursanne, Switzerland has been excavated into the Mesozoic age Opalinus clay formation and has been host to more than 47 projects that have benefitted from collaboration by 15 international organizations (see Figure 4–7).

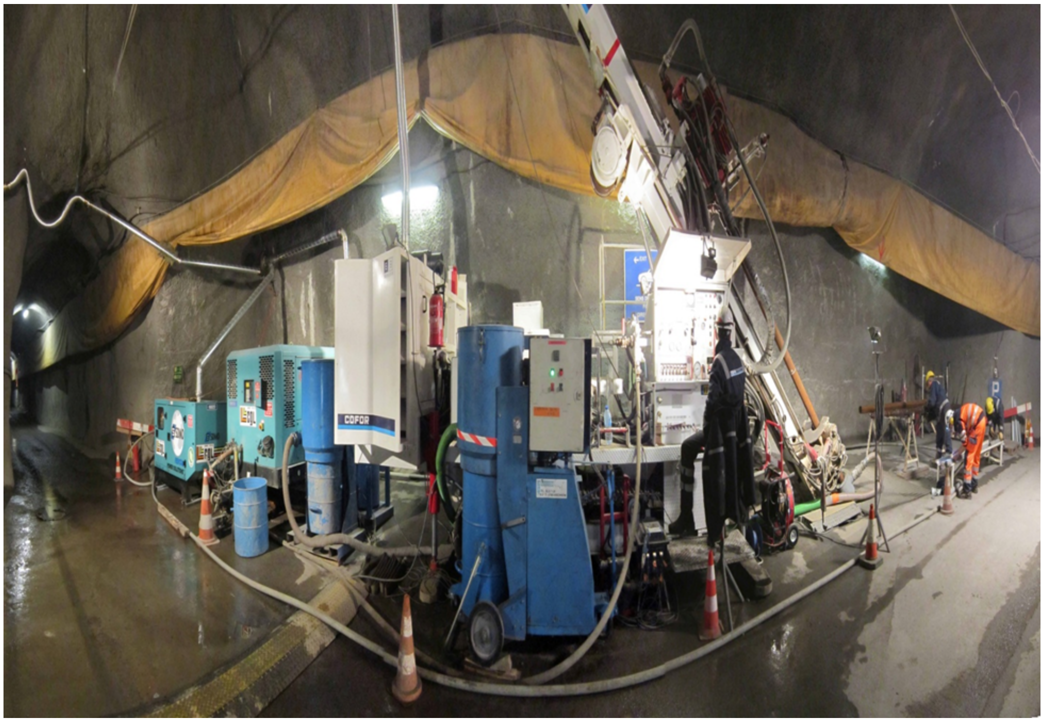

Figure 4–7. International collaboration at the Mont Terri Underground Laboratory in Switzerland (Photo Swisstopo)

Figure 4–8. Greenland Analogue Project investigating effects of glaciation and permafrost.

Another example of joint international studies is the Greenland Analogue Project initiated in 2009. The project, undertaken by SKB, Posiva, and the NWMO, was devised to study, through field investigation and numerical modelling, the influence of glacial and permafrost conditions on a deep-seated crystalline groundwater system. The field site was located at the land-terminating portion of the Greenland ice sheet (Russell Glacier), near Kangerlussuaq on the southwest coast. The Greenland Analogue Project was divided into three integrated project areas: i) surface based ice-sheet studies; ii) ice-sheet bed studies; and iii) geosphere investigations, the field work for which was conducted between 2009 and 2012 (see Figure 4–8). A comprehensive data report and final report, detailing the results of the four-year field campaign and the synthesis of a conceptual model for the ice-sheet perturbed groundwater system will be published in 2016.

Finally, since 2005 the NWMO has continued to participate in the international radioactive waste management program of the Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA). Members of this group include all the major nuclear energy countries, including waste owners and regulators. The NWMO has had continued membership on several NEA Committees and working groups, including: i) the Radioactive Waste Management Committee (RWMC); ii) the Integration Group for the Safety Case (IGSC); iii) the Forum on Stakeholder Confidence; and iv) the Working Group on the Characterization, the Understanding, and the Performance of Argillaceous Rocks as Repository Host Formations (i.e., the Clay Club). Active participation by the NWMO in initiatives launched by these international groups, such as the Safety Case Symposium (2013); the IGSC Feature, Events and Processes Database Project; the Reversibility and Retrievability Project; the Preservation of Records, Knowledge and Memory Project; and FEPCAT project for argillaceous sediments have been very beneficial to the repository program.

4.7 Conclusions

During the past decade, significant advances have occurred in the Canadian program for the long-term management of used nuclear fuel. Key amongst these advances has been the dialogue and engagement with Canadians that led to the development and recommendation by the Nuclear Waste Management Organization (NWMO) of Adaptive Phased Management (APM). APM is Canada's plan for a long-term nuclear waste management strategy for its used nuclear fuel. The APM approach, selected by the Federal Government of Canada in 2007, is being implemented by the NWMO. APM envisions that the nuclear used fuel would ultimately be placed within a multi-barrier deep geological repository excavated within a suitable sedimentary or crystalline setting with an informed and willing host, and that the facility is safe such that it meets or exceeds regulatory requirements.

The APM siting program was initiated in 2010 following two years of engagement with Canadians. By the end of 2012, a total of 22 communities (6 in sedimentary rock and 16 in crystalline rock) had expressed interest in learning more about APM. Through a stepwise process, initial dialogue, engagement, and technical screening of these communities was conducted to ensure safety and assessment of well-being potential. As studies progressed, NWMO's engagement activities broadened into the surrounding region to include First Nation and Métis communities in the area and surrounding communities.

In 2016, nine communities remain in the APM siting process. NWMO's work program includes further learning in these communities, including First Nation and Métis communities and those in the surrounding area, further assessment of community well-being and partnership potential for the project, and more detailed site-specific studies and investigations of safety. Ultimately, one preferred site will be selected for the APM facility and will be subject to the necessary environmental assessment and regulatory approvals process to attain a site preparation and construction license.

Confidence in the long-term safety of the multi-barrier repository design has been continually tested and refined by the NWMO through the APM technical program. This program has three functional areas: i) Geosciences; ii) Repository Safety; and iii) Repository Engineering. While advances have been made in each of these areas, recent design optimization and on-going proof-testing of a Canadian copper coated used fuel container is providing a means to further establish a basis for understanding enhanced repository safety. Publically available safety case studies in crystalline and sedimentary rock settings have focused on illustrating long-term safety and the robustness of the multi-barrier repository concepts. Technical collaboration with international radioactive waste management organizations and active involvement in the NEA OECD throughout the past decade have created a stronger basis with which to communicate and support the safety case for Canada's deep geological repository for used nuclear fuel.

4.8 References

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4.9 Acronyms

APM—Adaptive Phased Management

CANDU—Canada Deuterium-Uranium reactor

CNSC—Canadian Nuclear Safety Commission

HLW—High Level Waste

IGSC—Integration Group for the Safety Case

NEA—Nuclear Energy Agency

NWMO—Nuclear Waste Management Organization

OECD—Organization for Economic Co-operation and Development

RWMC—Radioactive Waste Management Committee

URL—Underground Research Laboratory

Chapter 5

Geological Disposal Program for High Level Radioactive Waste and the Plan for the Underground Research Laboratory in China

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ABSTRACT: This chapter introduces the latest progress in China's geological disposal program for high level radioactive waste (HLW) and the plan for an underground research laboratory. The Chinese government has decided that the installed capacity of nuclear power plants (NPPs) will reach 58 GW by 2020, and an additional 30 GW under construction. The spent fuel generated from those NPPs will reach 83,000 tons by 2050. The Chinese policy is that the spent fuel from light water reactors should be reprocessed first, followed by vitrification, and then final centralized geological disposal. The preliminary repository concept is a shaft-tunnel model, located in saturated zones in granite, while the final form for disposal is vitrified HLW. The geological disposal program is divided into three stages: (1) Laboratory Studies and Site Selection for HLW Repository (2006–2020); (2) Underground in-situ tests (2021–2040); (3) Repository construction (2041–2050). One of the major milestones is to complete the construction of an underground research laboratory (URL) by 2020. In the program, the five major R&D areas include: studies on strategies, regulations and standards; site selection and site characterization; engineering design for underground research laboratory and repository; safety assessment study; and radionuclide migration study. The strategy for the construction of the URL has also been proposed. With support of the China Atomic Energy Authority, comprehensive studies are underway, and achievement has been made in site selection, URL planning, engineered barrier system and other areas. The Beishan site is considered the first priority site for China's HLW repository.

5.1 Introduction

To safely dispose of the high level radioactive waste (HLW) generated from nuclear power plants (NPPs) and other nuclear facilities, the former Ministry of Nuclear Industry proposed a preliminary long-term program and has been conducting R&D for the final disposal of high level radioactive waste (HLW) since 1985 (Yang 1992). In 2006, the first government document for HLW disposal, "R&D Guidelines for Geological Disposal of High Level Radioactive Waste," was jointly published by the China Atomic Energy Authority (CAEA), the Ministry of Environment Protection (MoEP), and the Ministry of Science and Technology (MST), with the objective to build China's national high level radioactive waste repository in 2050 (CAEA 2006).

In 2007, the top Chinese government organization, the State Council, approved the "Medium- to Long Term Plan for the Development of Nuclear Power Plants in China (2006-2020)" (SCC 2007), indicating that the installed capacity of nuclear power plants (NPP) should reach 40 GW by 2020, with another 18 GW NPPs under construction. According to the plan, the NPPs will supply 4% of China's electric power by 2020. This will require construction of about 30 more nuclear reactors (1,000 MW-grade). As a consequence, the total spent fuel generated from those NPPs during their lifetime will reach about 83,000 tHM. This spent fuel must be stored, reprocessed and disposed of in safe manner.

On March 11, 2011, a severe nuclear accident occurred in Japan's Fukushima Dai-ichi nuclear power plant following a big earthquake and devastating tsunami. That accident had a significant impact on the worldwide nuclear industry, and many lessons can be learned from it. The Chinese government therefore conducted a series of safety checks and reviews, and suspended approval of new NPP applications. After a year and a half review and intensive discussion, in October 2012 the State Council approved an updated "Medium-to-Long Term Plan for the Development of Nuclear Power Plants in China (2011–2020)," which reaffirmed its support for nuclear power development in China. The updated plan is that the installed capacity of NPP should reach 58 GW by the year 2020, while other 30 GW under construction. The electricity produced by NPPs will account for 4% of the China's electric power in total.

In China, the CAEA is the government organization in charge of development plans and projects for HLW disposal, while the MoEP and its affiliated institute, the National Nuclear Safety Administration (NNSA), are the regulatory bodies. Implementation activities for radioactive waste disposal are currently managed by China National Nuclear Corporation (CNNC), while Beijing Research Institute of Uranium Geology (BRIUG) is the current lead institute.

Estimates from the Chinese nuclear power development plan are that the spent fuel accumulated from light water reactors will be 2,000 t by 2015. After 2020, about 1,000 t of spent fuel will accumulate each year. There is also another type of spent fuel—the one from CANDU reactors, which are in operation in a Qinshan-III nuclear power plant. The annual amount of spent fuel generated from the Qinshan-III nuclear power plant is about 200 MTU (Metric Tons of Uranium).

The Chinese policy for prioritizing high-level radioactive waste disposal is that the spent fuel from light water reactors should be reprocessed followed by vitrification, and then final geological disposal. The CANDU spent fuel will not be reprocessed. The preliminary concept for China's repository will be a shaft-tunnel model, located in saturated zones in granite. China has built a pilot reprocessing plant, which has begun operations since 2015. The siting study for a commercial reprocessing plant was started in 2007; construction should start about 2020 (CAEA 2006).

5.2 The Long-Term Plan for High Level Radioactive Waste Disposal

In 1985, the former Ministry of Nuclear Industry of China proposed a research and development (R&D) program for the deep geological disposal (DGD) of HLW (Yang 1992). The program included four phases: (1) technical preparation phase; (2) geological study phase; (3) in-situ test phase; and (4) repository construction phase. The objective of the program was to build a national geological repository in granite by 2040 for disposal of vitrified waste, transuranic waste, and HLW from decommissioning. Discussion on the long-term strategies was published in 2004 (Wang 2004a; Zhang 2004; Wang 2004b).

In 2006, a three-phase long-term plan for the geological disposal of HLW was set in the "R&D Guidelines for Geological Disposal of High Level Radioactive Waste" (CAEA 2006). Table 5–1 shows the three major phases. One major milestone is to select a site for the underground research laboratory and complete the construction of the laboratory before 2020. The final goal of the program is to build China's national repository for HLW around 2050.

Table 5–1. The 3-phase long-term plan for geological disposal of HLW in China

Phase	Period	Milestones
Phase 1: Laboratory Studies and Site Selection for HLW Repository	2006–2020	Repository sites should be preliminary selected, site characterization preliminary completed, a site for an underground research laboratory (URL) confirmed, the construction of an URL completed, technical capabilities in major areas preliminary established through laboratory studies.
Phase 2: Underground In-situ Tests	2021–2040	Site characterization should be completed, the repository site confirmed, most in-situ tests in the URL completed, technical capability for construction of repository established, detailed design for repository completed.
Phase 3: Repository Construction	2041—the middle of 21st Century	The repository construction is completed around 2050, the demonstration for HLW disposal is conducted, while vitrified HLW is accepted and disposed in the repository.

In 2007, the State Council approved the "Medium- to Long Term Plan for the Development of Nuclear Power Plants in China (2006-2020)" (SCC 2007), which says "the construction of an Underground Research Laboratory for Geological Disposal of High Level Radioactive Waste should be completed before 2020." Again it sets a very clear target for China's HLW disposal program in the near future.

In October 2012, the goal to complete the construction of a URL for China's geological disposal by 2020 was reconfirmed in China's Plan for Nuclear Safety and Prevention of Radioactive Pollution for the 12th Five-Year Plan and the Period between 2016 and 2020.

In the period of the 12th Five Year Plan (2011-2015), the program focuses on the following areas:

1. Site selection and site characterization for HLW repository: Drilling in the Beishan area in Gansu Province; Aqishan, Tianhu and Yamansu sites in the Xinjiang Autonomous Region; as well as the Alashan area in Inner Mongolia Autonomous Region, etc.
2. Site selection and preliminary feasibility study for an underground research laboratory in Beishan area.
3. Large-scale test on Chinese Gaomiaozi (GMZ) bentonite.
4. Conceptual design for underground research laboratory and repository.
5. Study on migration behavior of key radionuclides.
6. Methodological studies on safety assessment of disposal system.

5.3 Plan for China's Underground Research Laboratory

URLs are underground facilities in which characterization, testing, technology development, and/or demonstration activities are carried out in support of the development of geological repositories for high level radioactive waste disposal. Generally, URLs include "generic URLs" and "site-specific URLs." Facilities developed for research and testing purposes at a site that will not be used for waste disposal are called "generic URLs," while those that are a potential site for waste disposal and a precursor to the development of a repository at the site are called "site-specific URLs."

In accordance with China's long-term plan for geological disposal (CAEA 2006), study on the strategy and plan for China's URLs has been carried out since 2006, with construction expected to be completed by 2020.

In 2005, Ju Wang initially proposed the concept of "area-specific URL." In 2014, the concept of "area-specific URL" was formally proposed (Wang 2014). An area-specific URL refers to as a facility at a site within an area under consideration for an HLW repository, or located near a future repository site, and may be a precursor to the development of a repository at the site. It acts as both "generic URL" and "site-specific URL" to some extent. It can be aimed at confirming host rock suitability, conducting general research and development, guiding the layout of disposal tunnels and design of the repository, and demonstrating the technological operations.

Where a general area for a repository has been identified, but a specific site has not been selected, an area-specific URL can be built, as long as the site has similar geological, hydrogeological, engineering geological conditions and environments as a future in-depth repository site.

The area-specific URL has a potentially important role, i.e., if the site characterization and experiments conducted in the URL prove that the site is suitable for a repository, the process to select and confirm a site will accelerate accordingly.

In 2014, the CNNC proposed the strategy and plan for China's URL. The major considerations of the strategy are:

- Build an "area-specific URL" in a representative rock formation within the area that has been identified as the most suitable area for a geological repository
- The URL will be large-scale with full functions
- The URL will be about 500 m deep
- The URL should be expandable
- The URL will support technology development and demonstration, site characterization, public acceptance etc.
- Open for domestic and international cooperation
- Build a URL in granite first, and then a URL in a clay formation

The plan for the construction of China's URLs in the coming seven years has four stages:

1. Site selection and characterization for the URL.
2. Feasibility study for the URL.
3. Preliminary design and detailed design for the URL.
4. Construction of the URL.

Based on nationwide comparison and intensive consultation, in July 2011 the CAEA and the MoEP decided that the Beishan site in Gansu Province can be regarded as the first priority site for China's HLW repository. This important decision has provided a sound basis for URL site selection. Therefore, during stage 1, the site selection will be conducted in the Beishan area in 2016-2017. One preferred site and one backup site in granite will be selected among the five sub-areas in Beishan area: Jiujing, Xinchang, Yemaquan, Shazaoyuan and Suanjingzi.

During the feasibility study stage, the site will be confirmed, preliminary approval from the Gansu Province and the MoEP should be obtained, the data needed for design should be acquired, and other specific reports, such as an environment impact assessment report and a safety assessment report, should be prepared. Also, the design criteria for the URL should be determined.

Construction of the URL is estimated to take about four years. In order to reach the target date of 2020, the CAEA has approved two important projects. The first project is "Studies on Construction Technologies and Safety Technologies for URL." The second one is "Site Selection and Preliminary Design for URL." Currently, BRIUG is leading these two projects, and the "Beishan Exploration Tunnel," which is a 50-meter deep underground facility for construction technology development, has been under construction since June 26, 2015.

5.4 Progress in Site Selection and Site Characterization

5.4.1 Site Selection Process

Site selection for China's HLW repository started in 1985. The whole siting process was divided into four stages (Wang 2001; Wang 2004b; Wang 2006; Wang 2010): nationwide screening, regional screening, area screening, and site confirmation. During the siting process, the following factors were considered: socioeconomic factors; natural factors, including geological conditions, future natural changes, hydrogeological conditions, and geochemical conditions; human activities; construction and engineering conditions; waste transportation; environment protection; land use; social impact; and public acceptance. Since 1986, the following activities have been conducted for site selection:

1. Nationwide screening (1985–1986): six regions were selected as the potential regions: Southwestern China Region, Eastern China Region, Inner Mongolia Region, Southern China Region and Northwestern China Region, and Xinjiang.
2. Regional screening (1986–1989): Based on the results from the previous stage, further investigation was conducted, resulting in selection of 21 candidate areas. In the Northwestern China Region, the Beishan area in Gansu Province is considered as the most important area.
3. Area Screening (1990–present): Since 1990, major efforts have been concentrated on Beishan area, Gansu Province NW China. However, since 2011, drilling in granite intrusions in Xinjiang and Inner Mongolia was also conducted in order to find suitable sites for the purpose of comparison with the Beishan site. Investigation of clay formations also started in 2014.

5.4.2 Site Characterization

Since 1990, efforts have been concentrated on the Beishan area. Activities include regional geological setting, crust stability, geological characteristics, hydrogeology, and methodological studies for site characterization.

Within the Beishan area, eight granite intrusions have been chosen as candidate sub-areas for an HLW repository. Among them, three sub-areas (Jiujing, Xinchang-Xiangyangshan, Yemaquan) have been chosen as the sub-areas with the most potential. In 2012, two new sub-areas, Shazaoyuan and Suanjingzi, were proposed for further investigation.

During 1999–2015, site characterization was performed at the five sub-areas. Surface geological, hydrogeological, and geophysical investigations were conducted. Twenty-three boreholes (BS01—BS23, fifteen deep boreholes and eight shallow boreholes) have been drilled and used for pumping tests, injection tests, borehole televiewer, and borehole radar survey, sample-taking, and in-situ stress measurement. Through the 23 boreholes in the five sub-areas, rock and groundwater samples have been collected, and deep geological environment parameters have been measured. The suitability of the region has been further confirmed through a series of site characterization results.

5.4.3 Geology of Beishan Area

The Beishan area, located in northwestern China's Gansu Province (Figure 5–1), is the first priority area for China's HLW repository. The topography of the area is characterized by flatter Gobi and small hills with elevations ranging between 1,600 and 2,000 m above sea level. The height variation is usually several tens of meters. The seismic intensity of the area is below Grade VI, and no earthquakes with Ms > 4.75 have taken place based on historical records. Since the Tertiary period it has been a slowly uplifting area without obvious differential movement. Surface geological mapping, geophysical survey, and borehole investigation have been conducted in the Jiujing, Xinchang-Xiangyangshan, Yemaquan, Shazhaoyuan and Suanjingzi sub-areas. The results have shown that the rock mass is of high integrity (Figure 5–2), low fracture density, very low hydraulic conductivity, and moderate in-situ stresses, indicating that the Gobi desert Beishan site has great potential for the construction of future geological repositories and an URL (Xu et al. 1996).

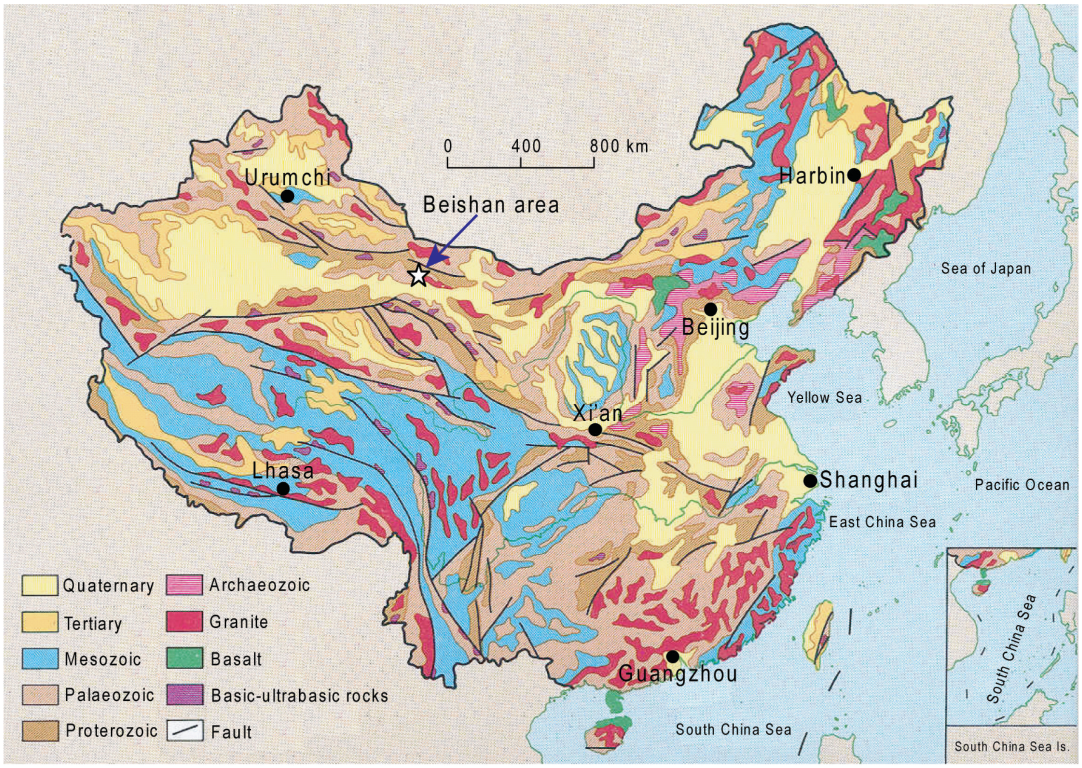

Figure 5–1. The location of the Beishan area, Gansu Province, northwestern China.

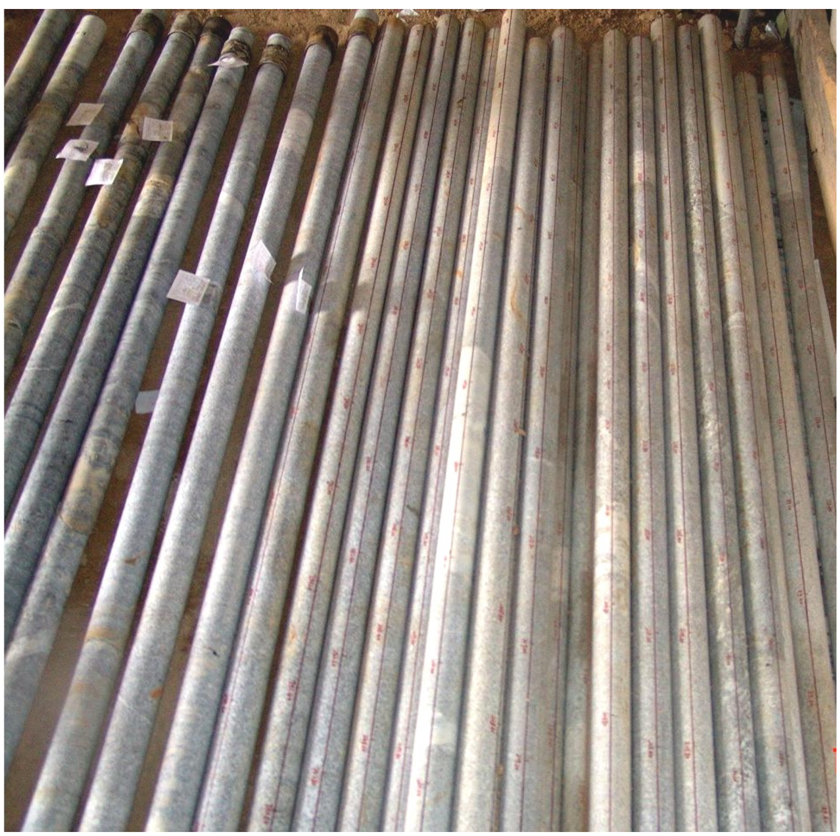

Figure 5–2. Intact 63 mm diameter drill cores extracted from borehole BS16.

5.4.4 In-Situ Stress in the Beishan Area

Knowledge of the in-situ stress field in the rock mass at depth is important to provide stress boundary conditions for underground excavation design and stability evaluation. During the last 15 years, in-situ stress was measured in nine deep boreholes in the Jiujing, Jijicao, and Xinchang sub-areas in the Beishan area, using a hydraulic fracturing method. Information on the stress magnitudes and orientations in these three sub-areas was obtained from 136 measurement points at depths ranging from 30 to 700 m. These stress data provide a comprehensive view of the state and characteristics of regional in-situ stresses in the Beishan area (Wang et al. 2006; Zhao et al. 2013b).

As shown in Figure 5–3, the horizontal stresses gradually increase with depth in each sub-area, although there is scatter in the measurement results. In all test locations, 81% of the maximum horizontal stress (σ_H) values are larger than the vertical stress (σ_v) values, indicating that the in-situ stress field is dominated by tectonic horizontal stress field rather than by the overburden load. Within the measurement depth range, the magnitudes of σ_H are all less than 25 MPa. The linear regression analysis further shows the variations of the principal stresses with depth. It is seen that a piecewise distribution of the principal stresses reveals approximately three stress domains with increasing depth. At shallow and medium depths, the stress fields are characterized by $\sigma_H > \sigma_h > \sigma_v$ and $\sigma_H > \sigma_v > \sigma_h$, respectively, where σ_h is the minimum horizontal stress. At greater depths (below 800 m), the maximum principal stress changes from horizontal to vertical (i.e. $\sigma_v > \sigma_H > \sigma_h$). In addition, the magnitude of σ_H at the depth of 450 m is about 15 MPa, which is approximately 50% and 25% of the maximum horizontal stresses, respectively, measured from the crystalline rock-based Äspö Hard Rock Laboratory (Andersson and Martin 2009) in Sweden and the URL in Canada (Martin 1997) at the same depth. The stress state in the Beishan area is therefore very favorable for the stability of underground excavations due to the relatively low stress magnitudes, compared with the strength of crystalline rocks and small difference between the maximum and minimum principal stresses.

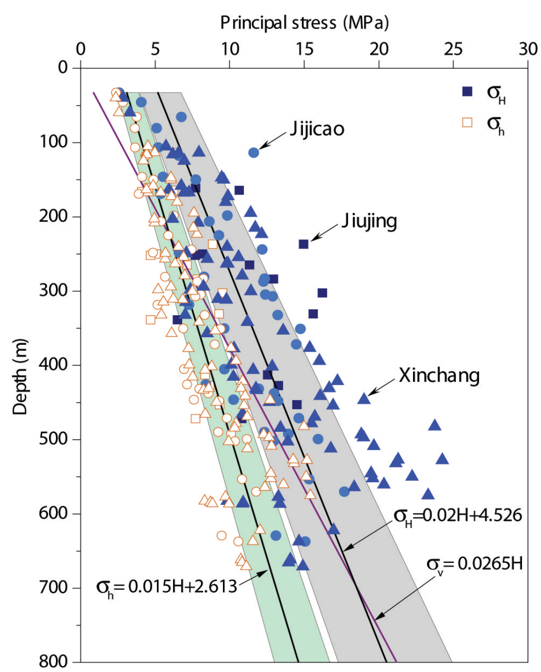

Figure 5–3. Variation of the principal stresses with depth in the Beishan area.

The orientations of the maximum horizontal stresses at 39 measurement test intervals were identified from the borehole impression measurement, and the results are presented in Figure 5–4. The fracture impression results show that in the three sub-areas, σ_H is oriented mainly in the northeastern direction, except for two test intervals in borehole BS05, where it is oriented in the northwestern direction. Hence, it can be concluded that the regional in-situ stress field in the Beishan area is influenced by the northeastern trending tectonic stress.

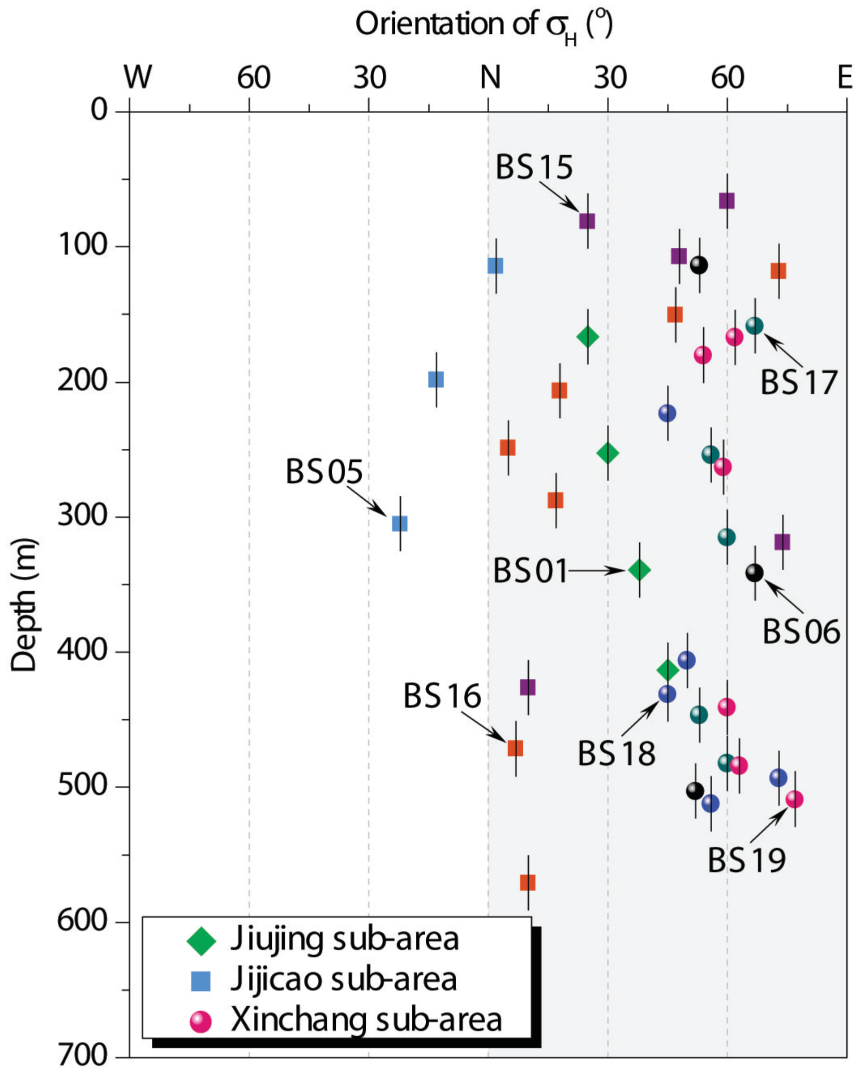

Figure 5–4. Variations of orientations of the maximum horizontal stresses with depth in the Jiujing, Jijicao, and Xinchang sub-areas.

5.4.5 Hydrogeology of the Beishan Area

The Beishan area is an arid Gobi desert area, with an average annual precipitation of 70 mm, while annual evaporation is 3000 mm. More important, there is no yearlong stream or other surface water body in the area. Therefore, the Beishan area is also poor in groundwater resources. Pumping tests carried out by local geological teams in the 1980s have shown that most of the wells produce less than 50 m^3/d. The groundwater in the Beishan area can be divided into three categories: (1) an upland rocky fissured unit; (2) a valley and depression pore-fissure unit; and (3) a basin pore-fissure unit. The upland rocky fissured unit is the most prevalent one in this area. This type of groundwater occurs in weathered and structural fractures, its recharge is primarily from precipitation infiltration, with discharge mostly through evaporation and lateral outflows into the fracture water-bearing zones, intermountain areas, and valley depressions.

The current water table at the potential site area is 28–46 m below the surface. Chemical analysis shows that the groundwater in the Beishan area is of Cl-SO_4-Na type, with pH ranging between 7–9, and the total dissolved solid (TDS) greater than 2 g/L. The *in-situ* water sample from depth of 440 m in borehole BS03 shows the TDS value of 4.15 mg/L, pH of 7.58, while the dominant ions are Na^+, Cl^- and SO_4^{2-}.

5.4.6 Permeability of Rock Mass

The permeability study of rock mass was conducted by using a "Double Packer Hydraulic Test System," produced by Golder Associates GmbH. The test system is characterized by reliable packer isolation, pressure monitoring above, below, and between packers, and reliable pumps. The system is computer controlled and can be used to conduct hydraulic test for rock mass with hydraulic conductivity less than 10^{-15} m/s. That system was used in boreholes BS05, BS06, BS15, BS16, BS17, BS18 and BS19 in the Jijicao and Xinchang sub-areas. Figure 5–5 presents the hydraulic conductivity distributions along these boreholes at depths between 400 and 600 m. The data show that for most of the test intervals, the hydraulic conductivity values are very small, mainly falling between 10^{-9} and 10^{-13} m/s. In a few test intervals hydraulic conductivity is larger than 10^{-8} m/s, due to the influence of local fracture zones. The low permeability of the rock mass around the boreholes not only reflects the good rock quality in the two sub-areas but also provides a very favorable condition for the construction of HLW disposal facilities and for the permanent isolation of radionuclides.

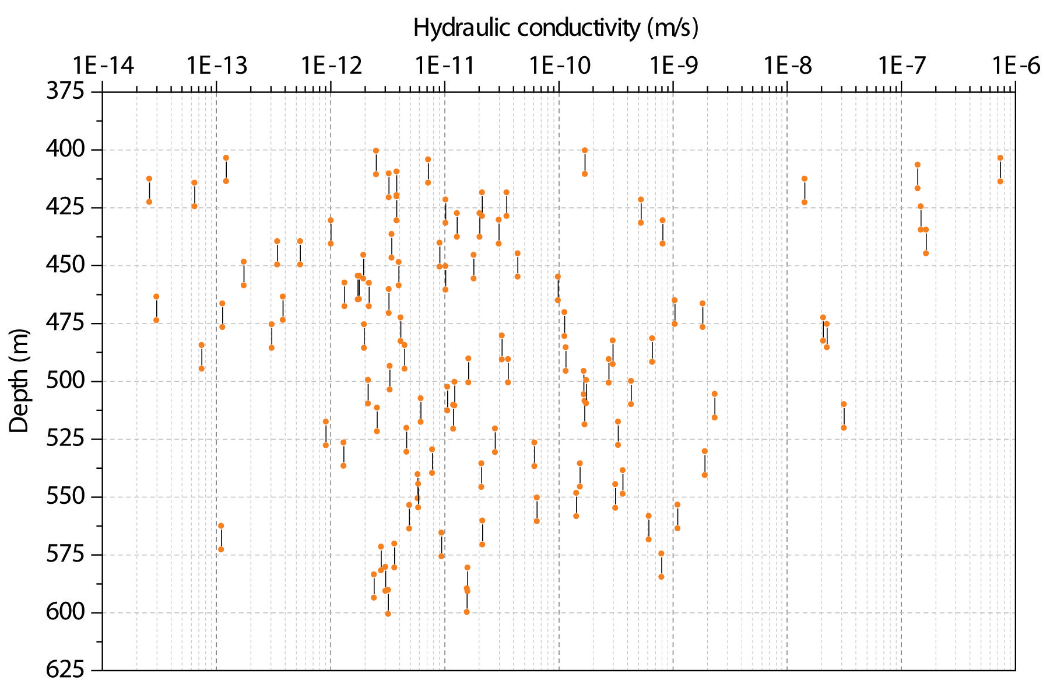

Figure 5–5. Hydraulic conductivity distributions along the boreholes at depths between 400 and 600 m in the Jijicao and Xinchang sub-areas.

5.4.7 Performance Assessment of the Beishan Area

The obtained data show that the Beishan area has positive socio-natural conditions for the construction of an HLW repository. The first performance assessment study of the total disposal system was conducted by Chen (2009) using GoldSim software. Further performance assessment study has confirmed the suitability of the area (Zhou et al. 2013a). The assessment model was developed based on the FEP (features-event-process) analysis and scenario development for the "normal scenario" (Zhou et al. 2013b), i.e., the expected evolution of such a repository. The repository concept has a shaft-tunnel access, with vitrified HLW glass encapsulated in a carbon-steel overpack surrounded by a bentonite buffer located in saturated zones within granite. The natural barrier consists of nearly intact near-field granite within fractured far-field granite. The biosphere is a desert environment with no agriculture and limited grazing activities due to poor water resources. The assessment period for radionuclide releases extends from the end of a 1000-year containment period to up to 10^6 years. Three variants for the "normal scenario" are also assessed, including no sorption by corrosion products, a new highly transmissive fractured zone forming within the intact repository host rocks, and impacts arising from heterogeneous flow paths in the host rocks. The results show that, for the reference case (with nearly intact host rock), no release of radionuclides to the biosphere is observed. For the extremely conservative variant scenarios, calculations indicate limited radionuclides reaching the biosphere over a million year period with insignificant annual dose, and the maximum annual dose from a single waste package is only 1.45×10^{-4} μSv per year (Zhou et al. 2013a).

5.5 Progress in Engineered Barrier Studies

5.5.1 Conceptual Design of Geological Repository

The preliminary concept for China's geological repository in granite is shown schematically in Figure 5–6. The engineered barrier system (EBS) includes the vitrified waste, waste canisters, buffer materials, backfill, and seals. Currently, the Gaomiaozi (GMZ) bentonite is considered as the candidate buffer and backfill material for China's HLW repository. The thermal analysis of disposal hole and the disposal area has been studied by Zhao (2009).

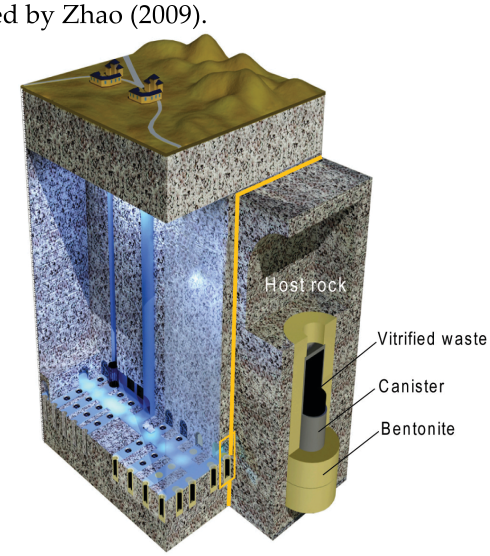

Figure 5–6. The conceptual model for China's HLW repository in granite.

5.5.2 Buffer/Backfill Material Study

The GMZ bentonite deposit, which occupies a mining area about 72 km^2, is located in northern China's Inner Mongolia, 300 km northwest of Beijing. The deposit, with bedded ores, was formed in the late Jurassic period. Ore minerals include montmorillonite, quartz, feldspar and cristobalite, etc. The reserve is about 160 million tons with 120 million tons of Na-bentonite. The major ore body of the deposit extends about 8,150 m with a thickness range from 8.78 to 20.47 m.

The GMZ bentonite is characterized by a high content of montmorillonite (>70%), which gives it a high Cation Exchange Capacity (CEC = 77.30 meq/100 g), a large plasticity index (I_P = 275), and a large specific surface area (S = 570 m^2/g). Preliminary researches have been conducted on the swelling, mechanical, hydraulic, and thermal properties of the GMZ bentonite. Results have shown that GMZ bentonite is a good buffer/backfill material (at a dry density of 1.6 Mg/m^3) for its relatively high thermal conductivity (1.51 W/m at a water content of 26.7%) (Ye et al. 2010), quite low saturated hydraulic conductivity (1.94×10^{-13} m/s at temperature of 25°C) (Wen 2006), a relatively high unconfined compression strength (1.74 MPa and a water content of 23.6%), and quite a high swelling pressure (3.17 MPa) (Liu et al. 2001). Chen et al. (2006) completed the experimental investigation by determining the water retention curves of the GMZ bentonite and showed its high retention capacity which is necessary for ensuring the containment function of the engineered barrier systems.

5.5.3 China-Mock-Up Facility for Bentonite Study

To study the behavior of the GMZ bentonite under coupled thermal-hydraulic-mechanical conditions, a mock-up facility, named China-Mock-up (see Figure 5–7), has been designed and installed in the Beijing Research Institute of Uranium Geology (Liu et al. 2013; Liu et al. 2014) The China-Mock-up is mainly made up of eight components, namely compacted bentonite blocks, steel tank, heater and corresponding temperature control system, hydration system, sensors, gas measurement and collection system, real-time data acquisition and monitoring system, as shown in Figure 5–8. When the experiment is completed, this facility will be dismantled, and its fillings will be studied in detail. Until now, the Mock-up has been working for about four years, and many important data have been obtained (Chen et al. 2014b).

The temperature evolution in buffer sections II, III, and VII is given in Figure 5–9 to Figure 5–11. Note that the temperature has continuously increased with time, especially in the inner rings. Moreover, the distribution of temperature is non-uniform vertically, and much higher in the central part. In addition, some fluctuations of temperature are recorded because of season-related room temperature changes. Meanwhile, the temperature distribution is also influenced by a complex coupling mechanism. Because saturation affects thermal conductivity, the temperature distribution is also dependent on the saturation process in the compacted bentonite.

Figure 5–7. The China-Mock-up facility.

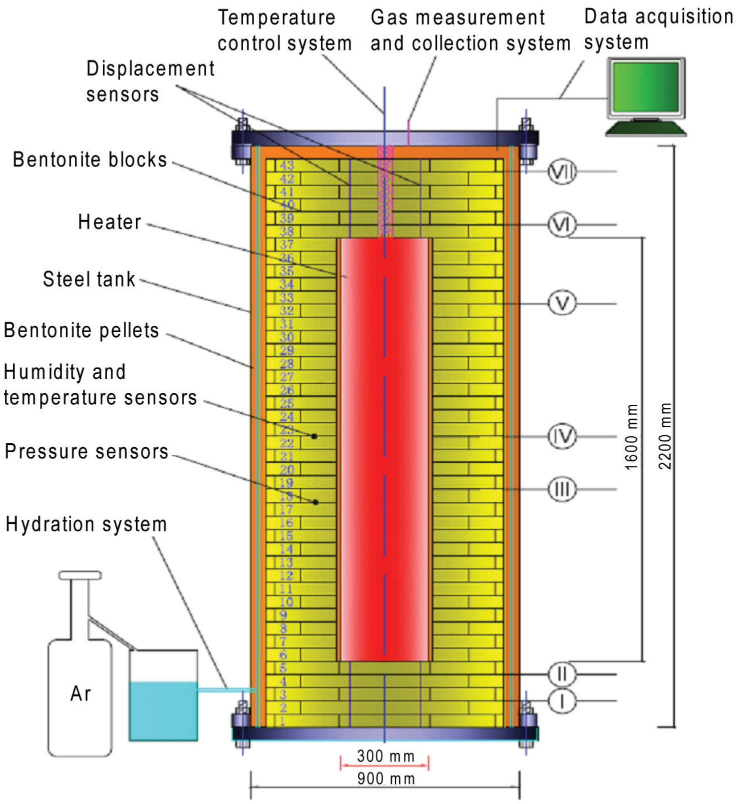

Figure 5–8. Main components of the China-Mock-up.

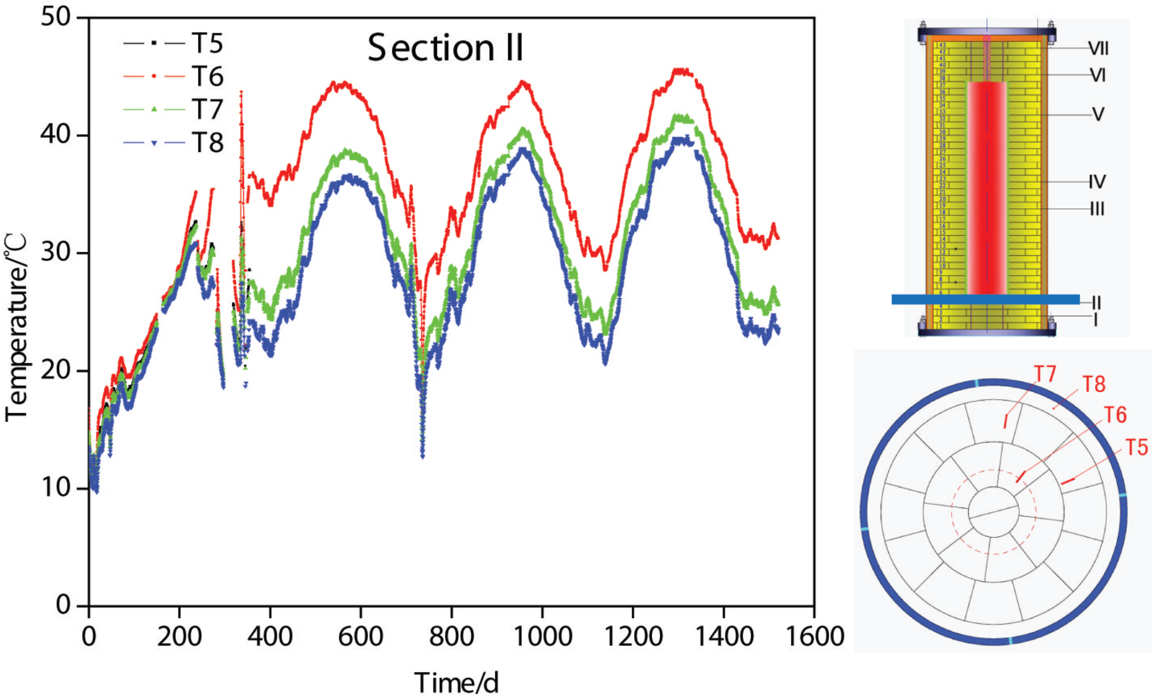

Figure 5–9. Temperature evolution in section II.

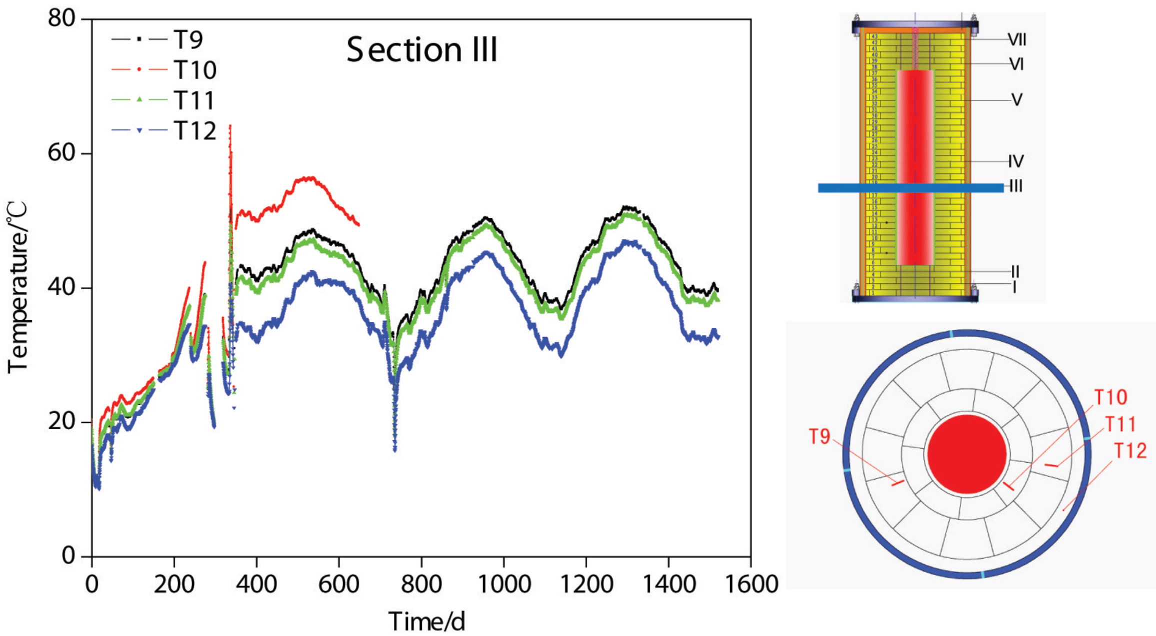

Figure 5–10. Temperature evolution in section III.

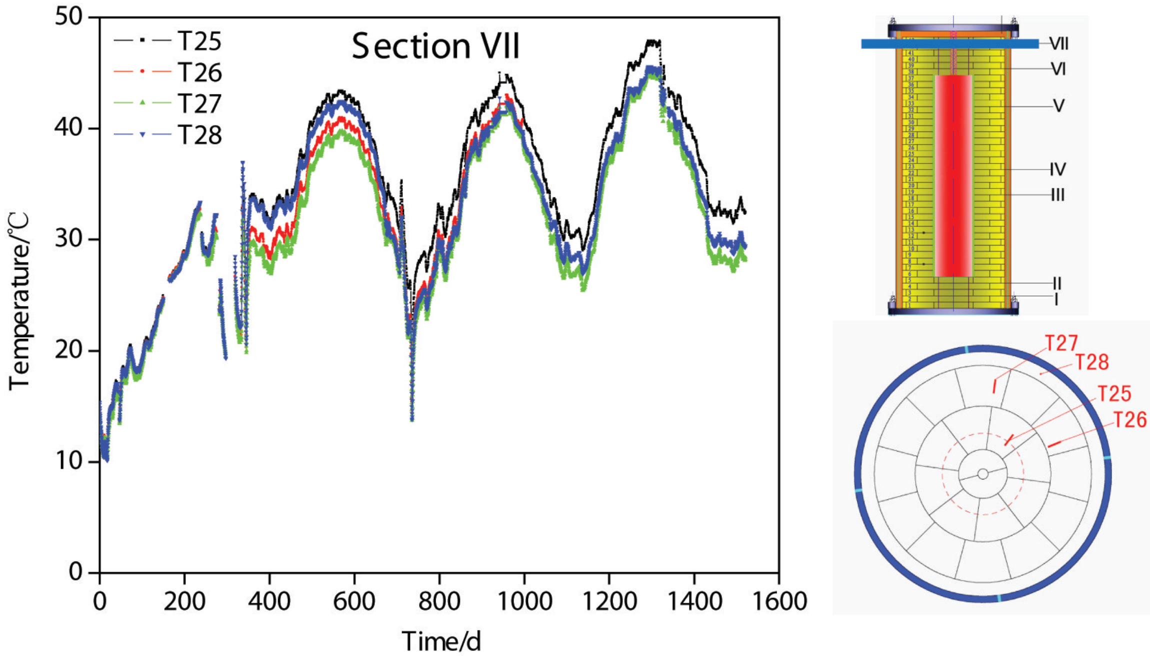

Figure 5–11. Temperature evolution in section VII.

The relative humidity (RH) distribution of China-Mock-up is illustrated in Figures 5–12 to 5–14. Note that the variation of relative humidity in these areas is much more complex. The compacted bentonite is progressively saturated with time in section II, and the distance to the outer boundary has a significant influence on the saturation process. In the zone at section III close to the heater, the decrease of relative humidity can be observed. This phenomenon can be attributed to the competitive mechanism between the saturation process induced by the water penetration, and the drying effect by the high temperature of the electrical heater. The desaturation phenomenon indicates that, due to the low permeability of the compacted bentonite, the drying effect is dominant at the beginning in the zone close to the heater. Then, with the increase of the injection water, the saturation process is dominant after 500 days, and the humidity increases gradually. The variation of relative humidity with time at the top of the China-Mock-up facility is illustrated in Figure 5–14. Thanks to the longer distance to the heater, note that the desiccation induced by the high temperature is less evident in this area. However, with the increase of temperature in in the central part of the section, the desaturation phenomenon is still noticeable. In addition, the relative humidity is sensitive to temperature fluctuation.

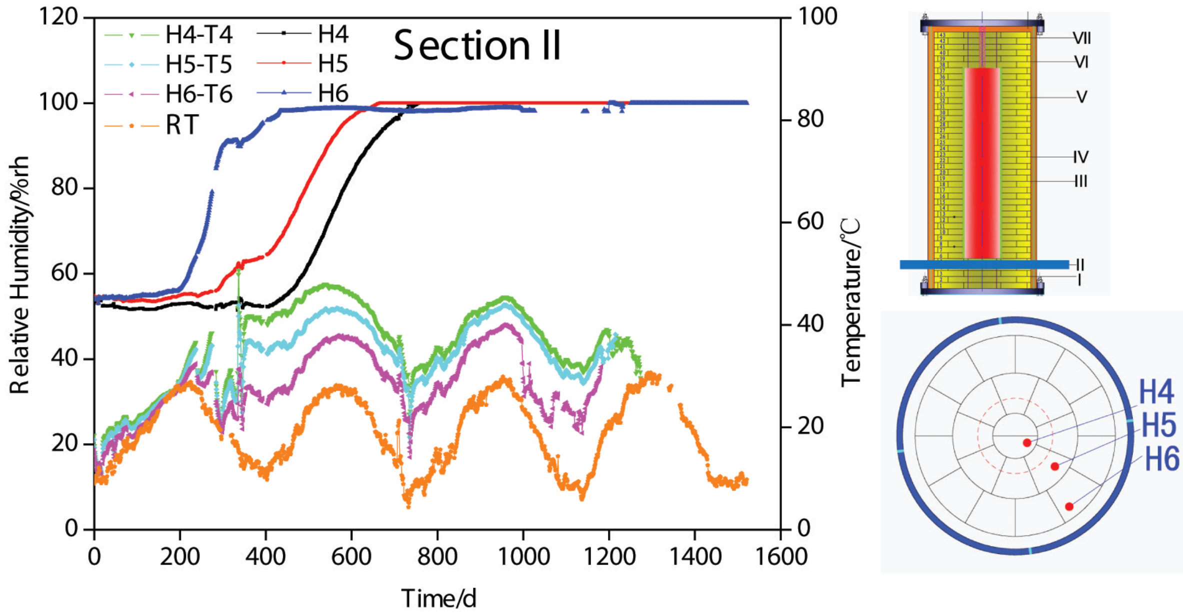

Figure 5-12. Relative humidity evolution in section II.

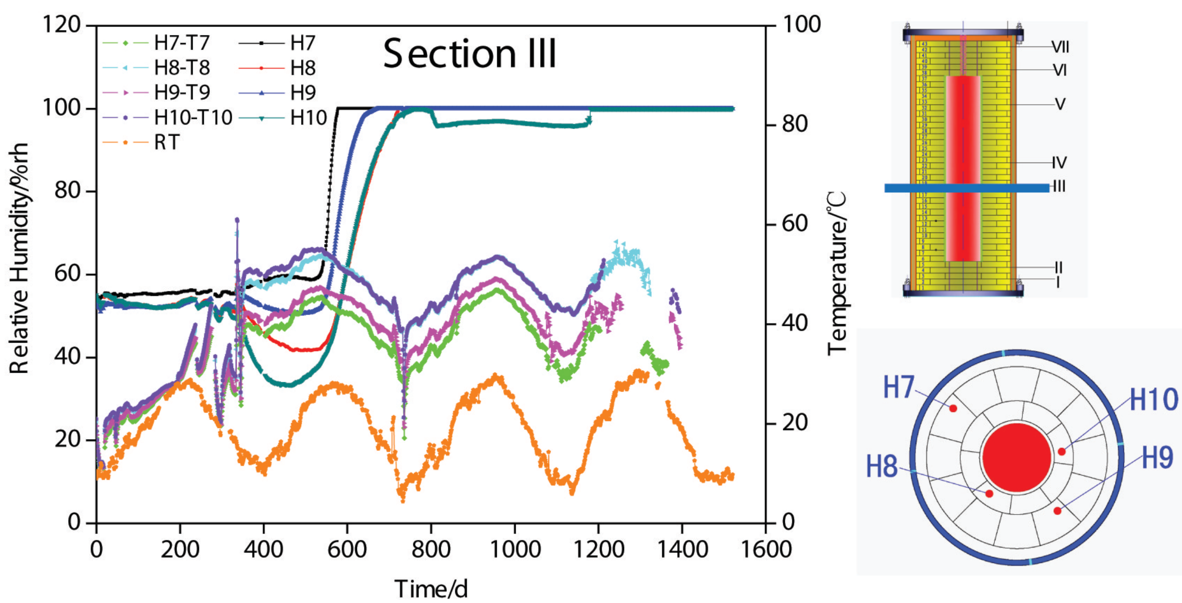

Figure 5-13. Relative humidity evolution in section III.

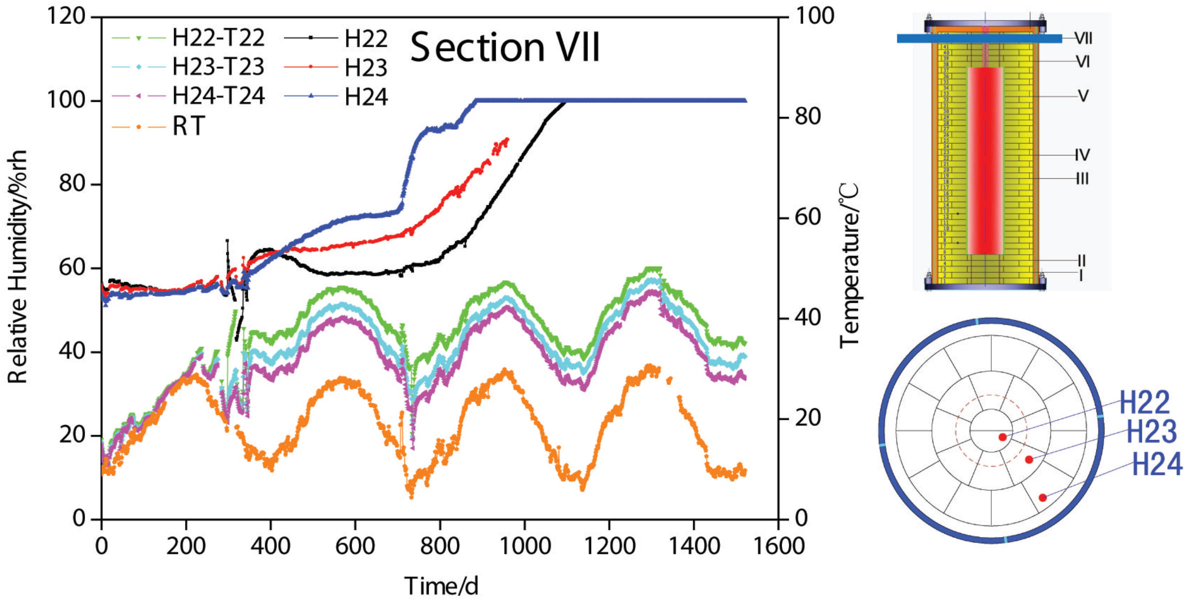

Figure 5–14. Relative humidity evolution in section VII.

5.6 Progress in Rock Mechanics Studies

5.6.1 A New Rock Classification System for HLW Disposal

Rock mass classification is widely considered as a practical and useful method to evaluate the "quality" of rock mass in underground engineering. However, the conventional methods are mainly concentrated on the properties of rock masses relevant to constructability. In light of the avoidance strategy in geological disposal (i.e., avoidance of adverse rock conditions or geological structures), the main objective of rock mass classification is to select a suitable rock volume for HLW disposal.

With the purpose of developing a comprehensive and practical method to evaluate the suitability of the host rock for HLW disposal, a new rock mass classification system named Q_{HLW} system was proposed (Chen et al. 2015b). In this system, the rock suitability evaluation of two different scales is considered, namely the repository and tunnel scales. The system is developed on the basis of the Q-system (Barton et al. 1974), and considers both the long-term safety and constructability requirement of the host rock for disposal. Thus, some additional parameters, including the fracture zone, groundwater chemistry, and thermal effect, are also taken into account in light of their significant influence on the long-term safety of HLW disposal. The avoidance strategy is highlighted in the proposed system by excluding the rock volume with unfavorable conditions, particularly adjacent to the large-scale fracture zones. As a preliminary validation, the proposed system is applied to the Beishan area. Using the classification method at the repository scale, the most suitable disposal site in Xinchang sub-area of the Beishan area is identified. The rock classification at the tunnel scale was also carried out along two deep boreholes (i.e., boreholes BS16 and BS18). The proposed system was found to be useful for identifying suitable rock volume of different scales for HLW disposal.

5.6.2 Basic Mechanical Properties of the Beishan Granite

In the Beishan area, the dominant rock types include monzogranite, granodiorite, and tonalite. The rocks at depth are relatively isotropic in texture and composition, with low porosity. The test results indicate that most uniaxial compressive strength (UCS) values of the core samples collected from different boreholes are between 100 and 240 MPa and are not sensitive to depth, as presented in Figure 5–15. The cracking processes of the Beishan granite subjected to compression and tensile conditions, and their influences on the macroscopic mechanical behaviors were comprehensively studied (Zhao et al. 2013a; Liu et al. 2014). Based on experimental investigations, a coupled elastoplastic damage model was developed (Chen et al. 2015b). The overall macroscopic mechanical behavior of the Beishan granite under different confining pressures is well reproduced by the proposed model. The established model provides an effective means to predict the mechanical response of rocks around the openings in the Beishan area.

By analyzing acoustic emission (AE) responses of Beishan granitic rocks in uniaxial compression, a new method (called the cumulative AE hit method) was proposed for determining the crack initiation stress (σ_{ci}) (Zhao et al. 2015), which is considered by some to be an important parameter for estimating field spalling strength of brittle rocks (Martin 1997; Diederichs 2007; Andersson et al. 2009). Based on the in-situ stress measurement results and established σ_{ci} of rocks, it is predicted that rock spalling may not occur around excavations in the Beishan area. Furthermore, rock samples were tested under true-triaxial stress condition to investigate the rock characteristics of the Beishan granite (Zhao et al. 2014). Experimental results revealed that under normal ground condition, rocks in the Beishan area may not experience rock-mechanical problems when the tunnels are located at medium depth (< 1000 m).

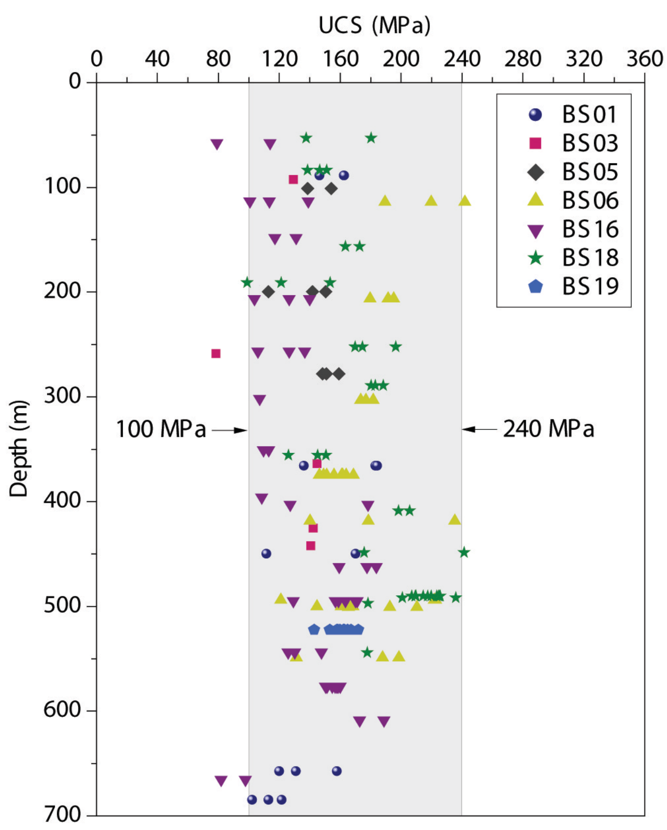

Figure 5–15. Variation of UCS with depth for different boreholes in the Beishan area.

5.6.3 Thermal-Hydro-Mechanical Coupling Behavior of the Beishan Granite

As the last barrier to the biosphere, the host rock is located in a thermal-hydro-mechanical coupled environment. To design and construct repositories successfully, it is essential to understand the thermal-hydro-mechanical coupling behaviors of the host rock. The hydro-mechanical coupling behavior of the Beishan granite in compression, particularly the damage evolution and its effect on the permeability of damaged rock, was systematically studied using the MTS 815 rock mechanics test system (Chen et al. 2014a), as shown in Figure 5–16. According to the analysis of experimental data and recorded AE events, it was confirmed that the permeability properties of damaged rock are strongly influenced by the initiation and growth of microcracks. An empirical model was proposed to represent the influence of damage evolution on axial permeability variation under different stress conditions.

A series of constant loading tests under different temperatures and confining stresses has been performed to study the time-dependent deformation of the Beishan granite (Liu et al. 2007; Lin et al. 2009). The results showed that when the temperature was increased to 90°C, the long-term strength of the Beishan granite decreased about 10%. Under the same confining pressure, the steady state volumetric strain rate increased when the stress ratio increased. For a given stress ratio, the strain rate increased significantly with decreasing confining pressure. On this basis, by incorporating the damage evolution process, a damage-mechanism-based creep model has been proposed to describe the full creep stages of the Beishan granite (Chen et al. 2014c).

Figure 5–16. The MTS 815 rock mechanics test system.

5.6.4 Three-Dimensional Discrete Fracture Network Modeling

Granitic rocks inevitably contain faults, fractures, and joints at different scales. A reliable 3D discrete fracture network (DFN) model of surrounding rock is one of the key models for the safety assessment of the HLW disposal. A modified max-min distance method of selecting initial cluster centers was first proposed by adding a factor of density into the distance calculation. This modification made the centers closer to the actual dominant occurrences. To evaluate the performance of different validity indexes, a stochastic approximation method was realized under different distance measures, and the optimal combination of validity index and distance measure was then proposed. The field orientation data of Beishan deep rock mass were successfully clustered by using the above achievements (Figure 5–17). A 3D DFN modeling code was developed, and automatic density analysis of fracture traces on disposal holes was realized (Figure 5–18).

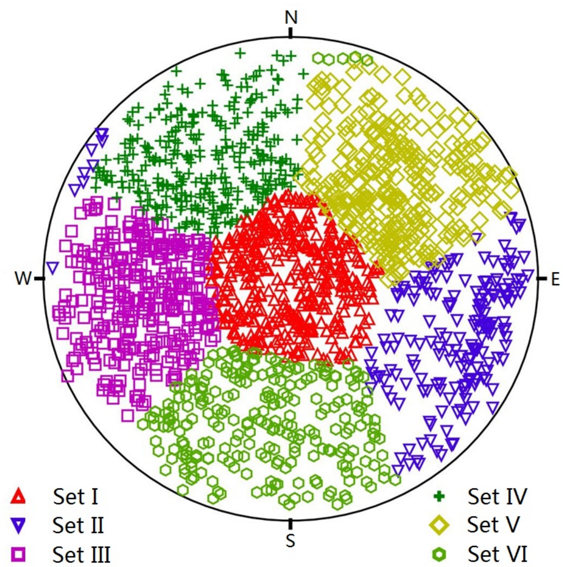

Figure 5–17. Clustering results of orientation data of BS17 borehole in Beishan area.

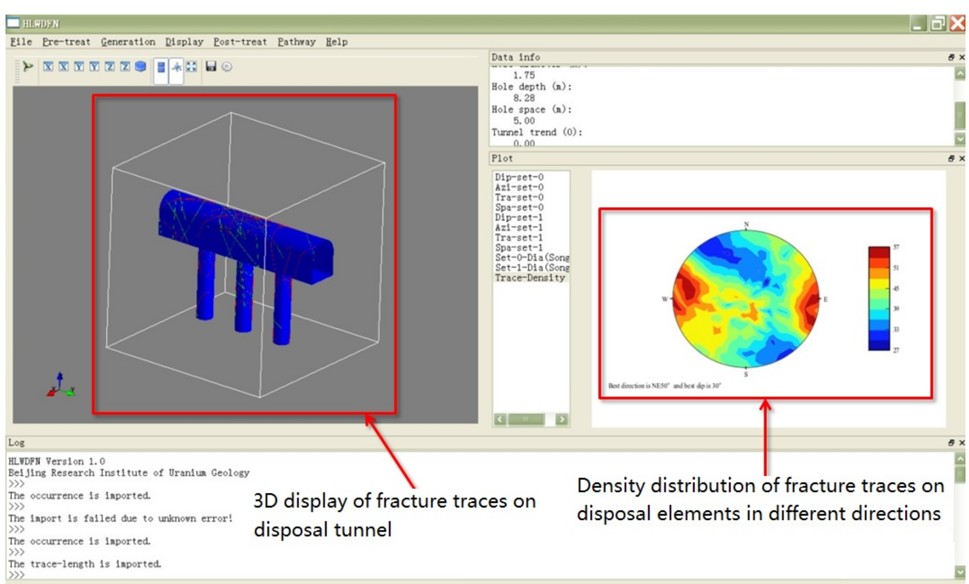

Figure 5–18. 3D DFN modeling code and trace density analysis

5.7 Summary of Key Scientific Challenges

The safe disposal of HLW/SNF (spent nuclear fuel) is a challenging task in scientific and technological areas, because the HLW/SNF has to be fully, reliably isolated for hundreds of thousands years, even millions of years. The radionuclides, such as Np, Pu, Am, Tc, etc., are highly radioactive, toxic, and with long half-life. If these radionuclides escape from the repository, they can cause tremendous harm to the biosphere. Therefore, deep geological repositories are constructed to isolate the HLW/SNF. However, the construction of such repositories faces several key issues, including:

- How to select a suitable site, and how to evaluate its suitability
- How to select engineered barrier materials to effectively isolate HLW/SNF
- How to design and construct a deep repository
- How to assess the long-term safety of the disposal system

To solve these challenging issues, many large-scale R&D projects have been carried out in the world (Wang 2007; Kickmaier 1997), including:

- Development and testing of excavation techniques, e.g., the projects in the Belgian URF, and the Finnish Olkiluoto site;
- Studies of Excavation Damaged Zone: e.g., the ZEDEX experiment at Äspö, the EDZ experiments at Grimsel Test Site and the Mt. Terri Tunnel in Switzerland;
- Site characterization studies: e.g., full-scale deposition holes in a research tunnel at Olkiluoto, and development of geophysical methods at Grimsel and Stripa underground laboratories;
- Hydrogeological tests: conducted in most of URLs;
- In-situ radionuclides migration tests: conducted in most of the URLs;
- Simulation of effects caused by emplacement of radioactive waste: e.g., the TSS project at Asse, FEBEX project at Grimsel, Heater test at Stripa, T-H-M test in Whiteshell, Drift-scale Heater Test in ESF (Exploratory Studies Facility), the international DECOVALEX project;
- Demonstration of engineered barriers system: e.g., RESEAL project in the Belgian URF, FEBEX project at Grimsel, buffer and container testing at Whiteshell, borehole sealing test at Stripa;
- Prototype repository study: e.g., the prototype repository test at Äspö, the EU's ESDRED project;
- Natural analogue and anthropogenic analogue studies: e.g., the Oklo project, the Brazilian Poços de Caldas project, the Cigar Lake project, the Australian ARAP project, the Lianshanguan uranium deposit study in China.

Most of the above R&D projects are carried out to address the following key scientific challenges:

- Prediction of the evolution of a repository site
    - Because HLW/SNF contains many radionuclides with long half-lives, the wastes must be isolated for a very long time, as long as $(1\sim10)\times10^5$ years. Therefore, the accurate prediction of a repository site should be carried out, including the prediction of the

geologic stability, regional geologic conditions, regional and local groundwater flow, climate changes, landform, and geologic hazards (volcanism, earthquakes, faulting etc.).

- Characterization of deep geological environments
    - The geologic repositories are usually located at depth between 300–1000 m, where the environment is characterized by high temperature, high in-situ stress, reducing conditions, groundwater flow, and radiation caused by waste. The understanding of the deep geological environment is key to safety of a disposal system.
- Behavior of deep rock mass, groundwater, and engineering material under coupled conditions (intermediate to high temperature, in-situ stress, hydraulic, chemical, biological and radiation processes, etc.)
    - Compared with the shallow rock mass, the deep rock mass is characterized by heterogeneity and discontinuities, while the deformation is also discontinuous. Due to the excavation of repositories and to radiation, the environment and behavior of deep rock mass will experience great changes. At present, the behavior of deep rock mass and geomechanical evolution of the disposal structures under coupled conditions (intermediate to high temperature, in-situ stress, hydraulic, chemical, biological and radiation process) is a scientific frontier. Many international projects are created to conduct further studies in this area (for example, Birkholzer, 2015).

- Behavior of Engineering Material Under Coupled Conditions
    - The engineering materials for repositories include waste forms (such as waste glass), canister (carbon steel, copper etc.), buffer and backfill materials. These materials play an important role in preventing the intrusion of water and the migration of radionuclides. The behavior of such materials under coupled conditions (intermediate to high temperature, in-situ stress, hydraulic, chemical, biological, and radiation process) is much different from behavior under usual environmental conditions.
- Geochemical Behavior of Transuranic Radionuclides with Low Concentration and Its Migration
    - Radionuclide released from the repositories will be transported with groundwater flowing through fractures, and diffuse into the rock matrix. The migration behaviors of radionuclide depend on the groundwater flow and complicated geochemical processes. At present, we have little knowledge on the geochemical process of radionuclides such as Np, Pu, Am, Tc. The speciation, complexation, colloids, biological effect of those radionuclides under realistic repository conditions are challenging topics. Some of the radionuclides, such as ^{99}Tc, ^{129}I and ^{3}H, are difficult to retard, so selecting suitable materials to retard them is also challenging.
- Safety Assessment of Disposal System
    - The geological disposal system is complex, composed of many subsystems (waste forms, canister, buffer material, near-field, far-field, biosphere, groundwater, etc.), which will experience complicated and long-term coupled processes for 10,000 years. The detailed safety assessment of the system is a most difficult challenge to current computational and technical capabilities.

All of the above key scientific challenges are cross-cutting scientific challenges, and are related to many research disciplines such as geology, hydrogeology, radiochemistry, rock mechanics, engineering science, material science, mineralogy, thermodynamics, nuclear physics, radiation protection, and computer science. Only comprehensive and integrated R&D will help the final success of disposal of HLW.

5.8 Conclusions

The Chinese government has paid great attention to the issue of geological disposal of HLW. A three-phase long-term plan has been published to guide R&D for geological disposal of high level radioactive waste, with major milestones to build an underground research laboratory by 2020 and a national geological repository by 2050. Necessary resources have been arranged for the geological disposal program.

A strategy to build an "area-specific underground research laboratory" has been proposed. The site for the URL will be selected from the potential sub-areas in Beishan, Xinjiang, and Inner Mongolia. The CAEA has approved two important projects: "Studies on Construction Technologies and Safety Technologies for URL" and "Site Selection and Preliminary Design for URL." Construction of China's first URL is planned to be completed by 2020.

Beishan, located in Northwestern China's Gansu Province, has been selected as the first priority site for China's HLW repository. Altogether 23 boreholes have been drilled in the Jiujing, Jijicao, Xinchang, Yemaquan, Shazaoyuan and Suanjingzi sub-areas in Beishan in the period of 2000–2012. The results have shown that the rock mass is of high integrity, very low fracture density, very low hydraulic conductivity, and moderate in-situ stresses, indicating that the Gobi desert Beishan site has great potential for the construction of future geological repositories and a URL.

A multi-barrier concept has been proposed for the preliminary design of a geological repository in granite. The GMZ bentonite in Inner Mongolia has been selected as the buffer and backfill material. An experiment in a mock-up facility, to study its properties under thermal-hydro-mechanical-chemical coupled conditions is in operation.

A new rock mass classification system has been developed to evaluate the suitability of the host rock for HLW disposal. Comprehensive experimental studies on the mechanical characteristics of Beishan granite have been conducted. A series of theoretical models have been established to represent the behaviors of Beishan granite under different stress and coupled conditions. A preliminary 3D DFN modeling code with optimized algorithms has been developed to provide DFN models for stability evaluation and nuclide pathway analysis.

Although progress has been made in many sectors, the Chinese program is still facing social, economic, scientific, technical, and engineering challenges. Continuous efforts will be concentrated on the Beishan site and its comparison with other sites, concept design of repository, design and construction of underground research laboratory, safety assessment, while other associated laboratory studies will also be conducted in the coming years.

One of the key tasks for China's geological disposal program is to complete the building of a URL before 2020. Resources have been allocated to the URL project, and it is the most challenging work facing the Chinese program in the coming years approaching 2020.

5.9 Acknowledgments

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5.11 Acronyms

AE—Acoustic Emission

BRIUG—Beijing Research Institution or Uranium Geology

CAEA—China Atomic Energy Authority

CNNC—China National Nuclear Corporation

DFN—Discrete Fracture Network

DGD—Deep Geological Disposal

EBS—Engineered Barrier System

EDZ—Excavation Disturbed Zone Experiment

ESF—Exploratory Studies Facility

FEP—Features-Event-Process

GMZ—Chinese Gaomiaozi

HLW—High Level Radioactive Waste

MoEP—Ministry of Environment Protection

MST—Ministry of Science and Technology

NNSA—National Nuclear Safety Administration

NPP—Nuclear Power Plant

R&D—Research and Development

RH—Relative Humidity

SNF—Spent Nuclear Fuel

TDS—Total Dissolved Solid

UCS—Uniaxial Compressive Strength

URL—Underground Research Facility

URL—Underground Research Laboratory

Chapter 6

Progress of the Czech Deep Geological Repository Program

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ABSTRACT: The program for the development of a deep geological repository for the disposal of spent nuclear fuel and radioactive waste in the Czech Republic was launched in the late 1980s. The program received a significant boost in 1997 following the passing of the Atomic Act, which established the Radioactive Waste Repository Authority (SÚRAO) as a state organization responsible for deep geological repository development. The initial stages of repository development involved the screening of geological conditions throughout the Czech Republic, which revealed that, as a result of the prevailing geological composition of the Bohemian Massif, granitoids would provide the most suitable rock mass environment for deep repository siting purposes. At the same time extensive research commenced concerning the materials to be used in the construction of the waste disposal containers, the various absorbing and sealing materials and other components which would make up the future deep repository. The first Reference Project concerning the repository's above-ground and underground sections was developed at the end of the 1990s. Subsequently, the updated State Energy Concept, which envisages the building of new nuclear sources in the Czech Republic, necessitated the enhancement of the capacity of the future repository and the re-assessment of all the candidate sites in terms of space requirements. Work on repository siting and site characterization as well as research work and the optimization of the design of the repository are being conducted with increasing intensity. Indeed, the extensive research of all aspects relating to the long-term safety of the entire waste disposal system is currently considered top priority. All the work involved in the project is aimed at meeting the set date of 2065 for the commencement of deep repository operation.

6.1 Introduction

On November 8, 2012 the Czech Government officially issued a State Energy Policy and approved the main elements of the energy strategy (Ministry of industry and trade 2014a) (http://www.mpo.cz/zprava161030.html). According to the Policy, major priorities consist of a balanced mix of generating sources, the efficient use of the full range of domestic sources available, and the maintenance of a surplus balance within the power system with sufficient reserves available at all times. The Policy assumes a slight decline in gross electricity generation from 93,443 GWh in 2015 to 88,541GWh in 2040, approximately 5 %. The Concept also assumes the strengthening of the role of the nuclear energy sector via the construction of two new generating units at the Temelín nuclear power plant (NPP) by 2025 and the extension of the lifetime of the existing four units at the Dukovany NPP, together with the construction of a fifth unit at this site by 2040. As a result, the nuclear sector share of electricity generation will increase to approx. 50% by 2040.

At present in the Czech Republic, almost 50 % of electricity is currently produced at coal-fired power plants, with slightly more than 33% produced at nuclear power stations. The assumed strengthening of the role of the nuclear energy sector in the Czech Republic will naturally be reflected in the volume of radioactive waste produced, primarily spent nuclear fuel (SNF), which will eventually have to be disposed of. This change is reflected in the Updated Concept of Radioactive Waste and Spent Nuclear Fuel Management official notice of which was taken by the Government on December 15, 2014 (Ministry of Industry and Trade 2014b).

The Concept states that the nuclear power station operator should focus on the direct disposal of SNF in a deep repository (DGR); however, it does not exclude the option of fuel reprocessing, particularly in view of the advanced fuel cycles of fourth generation reactors, the adoption of which would reduce both the potential hazard level and volume of the waste disposed of. The Concept also mentions that even though the variant of disposing of waste in an international regional repository currently seems unrealistic, developments in this respect should be monitored.

Concerning the amount of SNF to be disposed of, the Updated Concept of Radioactive Waste and Spent Nuclear Fuel Management bases estimates on data contained in the so-called Update of the Reference Design (hereinafter the Updated Reference Project) (Pospíšková et al. 2012). Currently it is envisaged that 4,420 tonnes of waste not acceptable in near-surface repositories and 9,910 tonnes (of heavy metal (HM)) of spent nuclear fuel from currently operational and new nuclear sources with an expected lifetime of 60 years will be disposed of in the DGR.

The Concept sets out the following milestones concerning DGR development and construction:

Year	Milestone
2020	Selection of two potential sites
2025	Selection of the final site
2050	Commencement of DGR construction
2065	Commencement of DGR operation

The above data form an outline framework for current activities relating to the research and development of the Czech deep geological repository.

6.2 Design

Research relating to DGR design has intensified during recent years. Work on the optimization of the Reference Project commenced (Holub et al. 2003) almost immediately following its completion (Holub 1999). For the purposes of optimization the DGR was divided into individual modules consisting of structures of the same or similar importance and related in terms of technological, transport or other considerations (e.g., the transport service module, the workshops and warehouses module, the SNF disposal module etc.). Moreover, a total of five designs relating to the above-ground and underground layouts of the repository complex were defined and compared for optimization purposes (and two optimal designs are presented in Table 6–1), concerning which the DGR layout included in the Reference Project of 1999 formed the basic version. Further variants consisted of the preparation of spent nuclear fuel (SNF) and radioactive waste (RAW) for disposal outside the deep repository complex and the consideration of the disposal of waste generated from potential SNF reprocessing. Individual modules were optimized in relation to the definition of individual variants. The assessment of individual variants took into consideration economic, safety and socio-political aspects as well as overall feasibility. The variant relating to the preparation of waste for disposal outside the repository complex was evaluated most highly primarily because of the consequent considerable reduction in the required surface area of the repository as well as favorable assessments in terms of safety and a number of socio-political aspects. The initial variant as set out in the Reference Project of 1999 was rated in second position.

A conceptual study prepared in 2004 compared the various advantages and disadvantages of the horizontal and vertical disposal concepts. The study also considered a potential two-level disposal variant (i.e., one level constructed below the other). The following detailed comparison of the horizontal and vertical disposal concepts (Březina et al. 2004; 2005) concluded that:

- Horizontal SNF disposal is realistic and advantageous in terms of the required excavation work, i.e., the volume of excavated material is reduced
- Horizontal SNF disposal will lead to a reduction in the surface area of the SNF disposal horizon
- The disadvantage of horizontal disposal lies in the relatively demanding nature of waste handling associated with this concept and the underground transportation of super containers

It was recommended that research into underground transportation and handling techniques should continue, and that verified calculations should be performed for horizontal disposal configuration in terms of residual heat build-up. Calculations concerning vertical SNF disposal were performed as part of the Reference Project of 1999 (Holub. 1999).

Both expert estimates and data from foreign literature were employed in the compilation of the report. Based on the results of the above studies, SÚRAO subsequently commissioned a project titled "Update of the Reference Design of a Deep Geological Repository in a Hypothetical Locality", the final report on which was completed in 2011 (Pospíšková et al. 2012). This reference design project was conceived as a set of partial model documentation concerning the permitting process for DGR siting. In technical terms it was designed to be a model construction project, the parameters of which corresponded to the current level of knowledge. The final site has not yet been selected, therefore a hypothetical site was chosen with properties representing the site to be selected in the future.

The design, evaluation, and selection of the optimal variants of the technical design of the DGR formed one of the first topics addressed by the reference design project. A multi-criteria evaluation was performed for individual modules considered significant in terms of the conceptual design such as the SNF and RAW disposal containers (the materials to be used and the structural design) and the layout of the repository's above-ground and underground sections. Table 6–1 shows the differences between the variants suggested in the 1999 Reference Design and the 2011 Reference Design.

The base assumptions and the concept of the variant proposed in the 2011 Reference Design (Pospíšková et al. 2012) are summarized below:

- The design provides the potential for simultaneous construction and operation of the repository
- The major part of the installations and equipment which have a substantial impact on operational safety and radiation protection such as the reloading node, hot cell, and related equipment is situated in underground caverns at the 0.0 meter horizon. This design option will also result in a considerable reduction in the repository's surface area provided that the morphology of the candidate location is suitable
- The disposal horizon will be situated at a depth of approximately 500 meters below the surface

- A vertical shaft will be employed for the removal of excavated material and the transportation of personnel and materials
- An inclined tunnel will be constructed for the transportation of waste and machinery
- Transportation in both the inclined tunnel and the disposal horizon will be rail-less

Table 6–1. Main differences between variants of 1999 Reference Design and 2011 Reference Design

	1999 Reference Design	2011 Reference Design
Disposal inventory	Direct disposal of SNF Disposal of waste not acceptable in near-surface repositories	Direct disposal of SNF Disposal of waste not acceptable in near-surface repositories (Alternative scenario: disposal of waste generated from potential SNF reprocessing)
Transport	Road	Railway
Reloading node	Surface area	Underground area
Underground access	Personnel: vertical shaft SNF and RAW: vertical shaft	Personnel and material: vertical shaft SNF and RAW: incline drift
Disposal borehole layout	Vertical	Horizontal

The basic function of the DGR above ground section throughout the various operational stages of the DGR is to provide or ensure the provision of those services required for the construction of the DGR's above-ground facilities, the excavation of the underground section, including in the stage which involves the simultaneous disposal of waste, services required for the handling of excavated material, activities involving safe disposal, those operational services necessary for the safe and smooth operation of both the DGR's above-ground and underground facilities, compliance with the requirements of supervisory and state administration authorities and legislation, activities relating to environmental protection, the protection of the DGR complex and the surrounding environment, and the protection of DGR personnel against potentials risks, however small, related to DGR operation.

The proposed concept of the underground section and the design thereof are based primarily on three main issues: meeting design requirements and relevant legislation and ensuring both the functionality of the facility and a high level of operational security.

The extent of the underground facility, the dimensions of the various installations, mine working profiles, and the total area required will be determined by the total volume of SNF and RAW to be disposed of. It is planned that SNF and other RAW will be disposed of at a disposal horizon of -500 meters. SNF packed into supercontainers will be emplaced in large-diameter horizontal disposal boreholes while other RAW packed into concrete containers will be emplaced in disposal chambers.

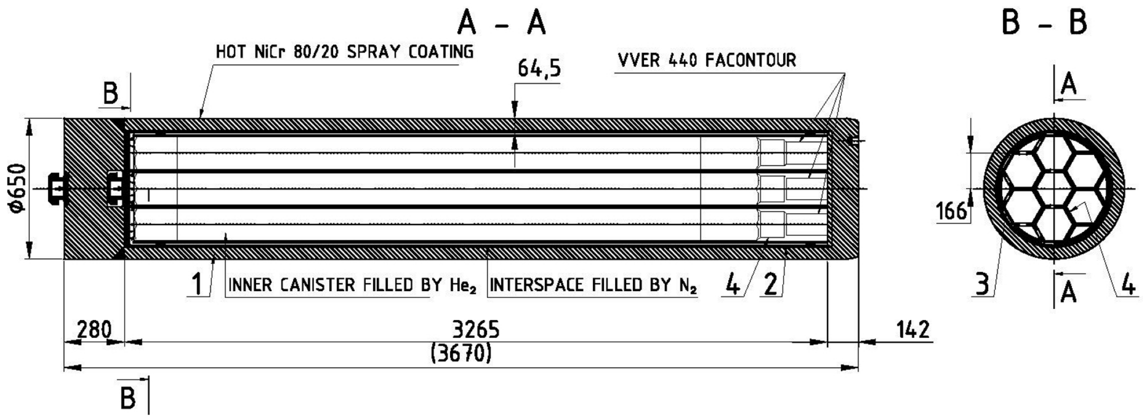

Figure 6–1. Design of disposal container for SNF from VVER 440 reactors: 1–outer case, 2–inner case, 3–fixing of fuel assembly, 4–fuel assembly. After Pospíšková et al. 2012

The super container will consist of a disposal canister which, in the case of VVER440 reactor SNF, will be 3,670 mm long and will have a diameter of 650 mm; it will contain 7 fuel assemblies, will be double-walled and will be coated with an anti-corrosive layer (Figure 6–1). The inner case will be made of 5 mm-thick stainless steel with a flat bottom welded onto the case. An aluminum alloy insert will facilitate loading and the fixing of the fuel assemblies in the case as well as enhance its heat transfer characteristics. The outer case will be made from a low alloyed material with a wall thickness of 45 mm, a flat bottom and a hermetically welded lid. A protective anti-corrosion coat of paint will be applied to the surface of the outer case to protect it from the corrosion effects of the surrounding environment. It is proposed that the protective layer will consist of a 0.5 mm-thick sprayed coat of NiCr 80/20.

The waste disposal canister for SNF from VVER1000 reactors and from new nuclear sources will be similar to that described above; it will differ only in terms of size (a length of 5,050 mm and diameter of 701 mm) and in the number of fuel assemblies (three instead of seven).

The bentonite rings in which the disposal canister will be placed will be 700 mm thick for disposal canisters containing VVER400 fuel assemblies and 674 mm thick for those from VVER1000 reactors and new nuclear sources. The bentonite bottom and lid will both be 700 mm thick. The basket of the supercontainer will be made from perforated steel sheeting.

The Updated Reference Project (Pospíšková et al. 2012) envisages the disposal of 2,050 supercontainers loaded with VVER440 fuel assemblies, 1,130 supercontainers with VVER1000 fuel assemblies, 2,700 supercontainers with fuel assemblies from new nuclear sources and 5 supercontainers with fuel from research reactors.

The concrete disposal container for other radioactive waste with outer dimensions of 1.7 x 1.7 x 1.5 meters will consist of an outer and an inner casing made from 10 mm-thick steel sheeting with welded inner and outer 15 mm-thick bottoms. The upper parts of these casings will be welded to the upper flange ring containing threaded holes. The space between the casings will be filled with concrete. The surface of the disposal container will be covered with a ZnAl protective sprayed coating. Each concrete container will contain four waste casks. The Updated Reference Project envisages the placement of a total of 2,990 concrete containers.

Four sections are allocated for SNF at the -500 meter disposal horizon (Figure 6–2). Assuming a 10 % reserve for unfavorable geological conditions with regard to the drilling of boreholes, a total of 251 large-diameter boreholes with a diameter of 2,166 mm and length 300 m will be necessary for the disposal of the predicted amount of SNF. The bentonite spacing blocks between the supercontainers will have a thickness of 500 mm.

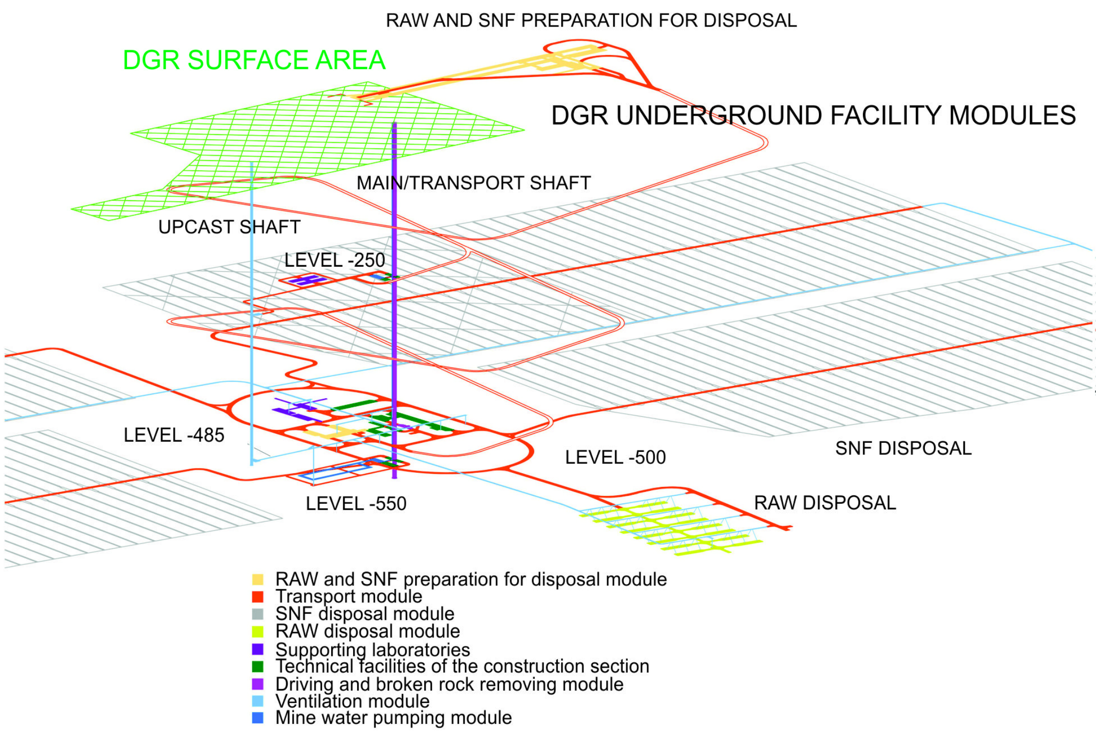

Figure 6–2. Modules of deep geological repository underground facility. After Pospíšková et al. 2012

Other RAW will be emplaced in a separate section at the -500 meter horizon, which will have space for 2,990 concrete containers. 16 disposal chambers (15 plus 1 reserve), 55 meters long, 10.50 meters wide and maximum 4.80 meters high, will be excavated. Each chamber will accommodate 204 concrete containers. A further variant under consideration is that of the construction of disposal chambers for other RAW at the -250 meter horizon. This variant will be realized provided that safety analysis documentation proves the acceptability of such a solution.

The DGR underground section has been designed as a multilevel mine working featuring four basic horizons (Figure 6–2):

1. The 0.0 meter horizon is primarily intended for the preparation of RAW and SNF for disposal; it will include the reloading node, the hot cell and related installations and will be accessible from ground level via a horizontal tunnel (a double- or single-track railway tunnel).
2. The -250 meter horizon will be used for the pumping of mine water to the surface. This horizon will also accommodate an underground research laboratory.
3. The -500 meter horizon will comprise sections for SNF disposal, chambers for the disposal of other RAW, the supercontainer preparation center, the confirmation laboratory and technological support facilities for both the DGR construction and waste disposal sections. The -485 meter horizon, which will form a subsidiary horizon to the -500 meter horizon, will comprise ventilation shafts, a ventilation station, outlets for ventilation stacks from installations placed at the -500 meter horizon and ventilation shafts serving the SNF and RAW disposal sections.
4. The -550 meter technological horizon is intended for the collection and pumping of mine water.

It is planned that the underground repository will accommodate SNF from currently operational NPPs, i.e., the 4 Dukovany NPP generating units and the 2 Temelín NPP units, as well as from planned new nuclear sources (2 units at the Temelín NPP and 1 unit at the Dukovany NPP). Further, it is intended that RAW originating from the decommissioning of existing NPPs and planned new sources which cannot be stored in near-surface repositories will also be disposed of in the deep repository.

It is assumed that the total area of the above-ground section will cover around 234 thousand m^2 and will be fenced. The area intended for the handling of radioactive materials (the control zone) and related facilities will cover approximately 21 thousand m^2 and security will be ensured via the appropriate physical protection system.

The underground area required will clearly depend on the amount of SNF and RAW as well as the disposal system finally selected. Based on available data and the expected volume of the disposal inventory assuming the adoption of horizontal disposal technology, the area required for SNF disposal has been estimated at approximately 2.72 million m^2 plus a 370 thousand m^2 reserve. An area of 40 thousand m^2 will be required for the disposal of other RAW. The DGR underground section will cover a total area of approximately 4.4 km^2.

It is assumed that the total volume of the underground section will be approximately 1.8 million m^3, of which the capacity of large-dimension horizontal boreholes around 547.5 thousand m^3 and the capacity of the chambers intended for the disposal of other RAW around 63 thousand m^3.

6.3 Siting

In compliance with a Government Decision of 2004, geological work at the selected candidate sites was suspended until 2009. In the following year a team of specialists led by Czech Geological Survey specialists drew up a plan of activities with concern to a hypothetical site, including financial considerations and an estimate of the required time period. This document subsequently formed the basis for the planning of further work at individual candidate sites.

A pre-feasibility study was then prepared the aim of which was to prove that both the above-ground and underground sections of the deep repository could be constructed at the sites according to the design set out in the Updated Reference Project 2012; the study subsequently proved the potential for the construction of both sections of the DGR at all the candidate sites (Holub et al. 2012).

The deep repository development program further assumes that geological investigation work will commence at all six sites initially selected for site characterization (Woller 2006) and at one newly-selected site. Sites 1-6 are situated in granitoid rock massifs. The newly-included Kraví hora site, located in the east of the Czech Republic on a regional rock massif known as the Moldanubicum (Moldanubian zone) is composed of granulite, migmatized gneisses to migmatite. The site is situated near the Rožná mine, the only currently operational uranium mine in Europe, and in the immediate vicinity of the Skalka central storage facility for SNF (currently under construction).

The areas of a number of sites to be subjected to geological investigation have, for a number of reasons, been slightly modified, and applications for the establishment of investigation areas have been drafted in compliance with the Geological Act. For application purposes the names of individual sites were changed, since the sites were originally named after selected communities located within the sites, a number of which, following the modification of the extent of exploration areas, were no longer included within site boundaries. The defined investigation areas comprise several communities (a minimum of three and in one case nine). Consequently, it was decided that names referring to one community would be replaced by a more general geographical name. At the end of April 2015 SÚRAO obtained a decision from the Minister of the Environment approving investigation areas at the Horka, Kraví hora, Magdaléna, Čertovka and Čihadlo sites. A decision from the Minister of the Environment concerning the remaining two sites is expected during 2015.

The new site names and the investigation areas within the sites are provided in Table 6–2 and the geographical position of the sites is shown in Figure 6–3 which also presents a simplified geological sketch map of the Bohemian Massif, the geological unit which forms the highly predominant part of the Czech Republic and the individual zones within it. Bohemian Massif presents a large denudation window with extent of some 100,000 km^2. It is one of the number massifs (Iberian, Armorican (AM), Central (CM) and others) that form the Variscan continental Europe.

The first investigation stage, which will involve all the sites within which an investigation area will be established in 2015, will consist of detailed remote sensing using state-of-the-art techniques, particularly the collection of new generation high-resolution radar data (e.g., Alos-Palsar L-band) and detailed geological, structural and hydrogeological mapping. Rock samples will be collected during the mapping stage for use in petrographic and microstructural studies, chemical analysis, tests to determine physical and mechanical properties etc. In addition, water samples will be taken from watercourses, wells and existing deep boreholes for hydrogeological purposes.

Geophysical measurements along 200m-distant profiles will be taken covering the whole surface areas of the candidate sites. Measurement will include the use of magnetometry surveys, geoelectric methods, resistivity measurements, VLF (very low frequency) measurements and radiometry. In addition, gravity and reflex seismic measurements will be taken to the extent necessary for the creation of 3D model of site.

Geochemical samples will be gathered along the same profiles at which geophysical measurements will be collected. Surface geochemistry techniques will be used for the identification of inhomogeneities by means of an element migration in the zone of hypergenesis model according to J. K. Burkov and D. V. Rundquist (Ahrens 1979). The samples will be subjected to whole rock analysis in order to determine a wide range of minority (Au, Ag, As, Ba, Be, Bi, Cd, Co, Cs, Cu, Ga, Hf, Hg, Mo, Nb, Ni, Pb, Rb, Sb, Sc, Se, Sn, Sr, Ta, Th, Tl, U, V, W, Y, Zn, Zr, REE) and majority (SiO_2, Al_2O_3, Fe_2O_3, CaO, MgO, Na_2O, K_2O, MnO, TiO_2, P_2O_5, Cr_2O_3) elements including the total amount of sulfur and carbon and losses on ignition. Two methods will be used for the evaluation of the results of the chemical analysis of the samples the first of which will consist of the identification of mineralized zones, rock fractures, fault zones and mineralized objects and the second the application of the Burkov-Rundquist model of the migration of elements in endo- and hypergenous environments.

It is expected that the results obtained, i.e., the evaluation of remote sensing, geological and structural maps, and geophysical and geochemical anomaly maps will allow the preparation of surface models of the sites and the formulation of assumptions concerning at-depth development based on which it will be possible to optimize the positioning of various types of boreholes including those to a depth of 1,000 meters. The drilling of boreholes to different depths will form the initial step in the next stage of geological investigation. It is possible that the spatial area of some of the sites investigated will subsequently be modified and, in extreme cases, certain sites will be rejected.

6.4 Underground Research Facilities

The research required for the DGR development program is currently being conducted at three underground facilities the geographical position of which is shown in Figure 6–3.

Work has been underway since 2003 in the Bedřichov water supply tunnel in northern Bohemia which was excavated in Variscan granitoids of the Krkonoše-Jizera Composite Massif. Interestingly, one part of the tunnel was excavated using tunnel boring machine (TMB) technology while drill and blast technology was used for the other. Following the first stage of research which focused on the compilation of geological documentation (Woller, 2006) the second stage focused on the area in which the excavation technology was changed. Geophysical measurements (resistivimetry and seismic) have been taken repeatedly with the aim of identifying both differences in rock deformation caused by excavation and variations in the measured values over time. In addition, temperatures at various distances from the tunnel wall have been measured and changes in temperature due to the temperature of water flowing through the tunnel monitored as were the amount and chemical composition of the groundwater which leaks into the tunnel along individual fractures, seasonal variations and dependence on climatic conditions at the surface. Movement along Sudeten trend (NW-SE) and the Krušné hory trend (NE-SW) fractures has also been monitored over the long term employing high-accuracy 3D dilatometers; movement of a few tenths of millimeters was subsequently detected (Klomínský, Woller, 2010). All the collected data is regularly processed in the form of models. Significant attention has recently been devoted to the optimization of data collection methods as well as methods to be employed for the automated trouble-free transmission of data from the underground complex to the processing center. The data gathered has been utilized in the Czech contribution to the international DECOVALEX project. Most of the work performed in the tunnel can be characterized as generic research.

The Josef Underground Research Center is situated 50 km south of Prague. The original mine complex, which is made up of 7,854 meters of galleries, was excavated between 1981 and 1991 with a view to the mining of the substantial gold deposits in the area. In 2007 the Faculty of Civil Engineering of the Czech Technical University (CTU) in Prague opened an underground laboratory at the site, named the Josef URC (underground research centre). The rock environment consists of weakly metamorphosed volcanic and volcanic-sedimentary rock (basalts, andesites, rhyolites, tuffs and tuffites) permeated with younger intrusive rock (granite and albite granite). The main gallery with a cross-section of 9 m² and a length of 1,853 meters proceeds in a NNW–SSE direction; several mine workings situated between 90 and 150 meters below ground level are connected to this gallery. A number of experiments and educational programs are currently underway at the Josef URC, some of which include SÚRAO´s involvement.

The third facility, the Bukov Underground Research Laboratory (URL), the construction of which is currently nearing completion, is situated in eastern Bohemia near the Kraví hora candidate site and the

Rožná uranium mine (Figure 6–2). The research facility consists of a 70 meter-long corridor and two chambers, is located at a depth of 600 meters, and is connected by a 320 meter-long gallery with the uranium mine. This facility will play an important role in the overall DGR research and development plan, is situated within the control zone of a fully-operational uranium mine. This facility will allow for conducting experiments involving radionuclides, and will provide the opportunity which is rare not only in the Czech Republic, but in other countries.

Table 6–2. New names of sites and investigation areas (numbers correspond to the locations in Figure 6–3)

No.	Original name (Woller 2006)	New name	Investigation area (April 2015)
1.	Lubenec-Blatno	Čertovka	established
2.	Pačejov Nádraži	Březový potok	
3.	Božejovice-Vlkšice	Magdaléna	established
4.	Lodhéřov	Čihadlo	established
5.	Rohozná	Hrádek	
6.	Budišov	Horka	established
7.	(Site did not exist in 2006)	Kraví hora	established

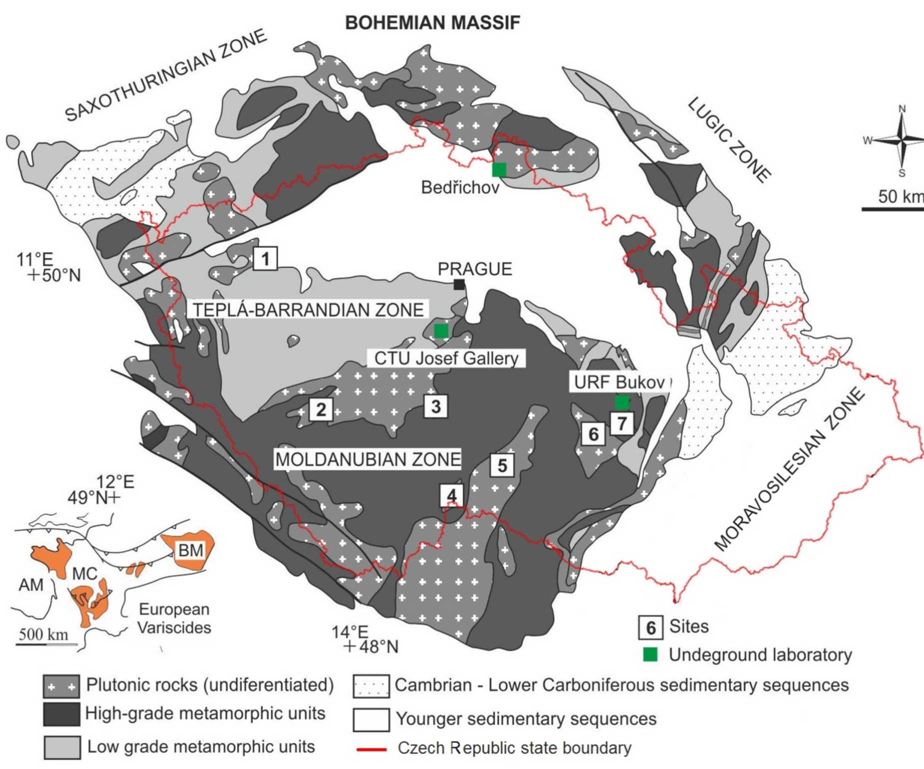

Figure 6–3. Schematic geologic map of the Bohemian Massif, depicting the state border of the Czech Republic, locations of sites for characterization and underground research facilities. Site numbers correspond to the numbers in Table 6–2.

The rock environment is made up of gneiss and migmatite. Two horizontal boreholes, 150 meters and 100 meters long have been drilled which will allow for the characterization of the geological environment of the area surrounding the Bukov URL as well as for the testing of various geophysical methods, the conducting of hydraulic tests and general monitoring. It is planned that a further borehole will be drilled from the surface to the URL, which will, among other things, provide information on the development of the geological, hydrogeological and geotechnical parameters of the rock massif with depth.

Excavation work and the compilation of detailed geological, structural and geotechnical documentation are currently nearing completion. The facility will be used principally for the in-situ long-term testing of materials to be used in the construction of waste disposal containers, the testing of new methods concerning rock environment characterization, etc.

6.5 Research

Extensive research activities involving the whole range of issues concerning DGR development have formed an integral part of the Program for the Development of the Deep Geological Repository in the Czech Republic from the very outset. Research activities can be divided into four stages, the first of which relates to 1994 to 1997. This stage, prior to the establishment of the Radioactive Waste Repository Authority, was characterized by a lack of relevant legislation. A number of research reports relating to individual parts of the disposal system were compiled during this period; however, the vast majority is currently considered to be of purely historical significance.

The second stage concerns the period 1998 to 2006 at the beginning of which the Atomic Act and related regulations came into force and the Radioactive Waste Repository Authority (SÚRAO) was established as a state organization responsible, among other things, for the development of a deep repository including related research. The research conducted during that period was of a continuous nature and included the development of the first long-term projects (e.g., the Ruprechtov natural analogue, the study of glass dyed using uranium-bearing pigments, the Mock-Up CZ project, pyrochemical fuel reprocessing, the research of temperature-impacted bentonites etc.), initial calculations of the source term and the drafting of reports concerning Czech participation in international projects which were drafted for the first time.

In 2006 SÚRAO commissioned a research project entitled “The Research of Near-Field Processes in a Deep Geological Repository for Spent Nuclear Fuel and High-level Radioactive Waste” which involved the participation of the Nuclear Research Institute Řež and three universities. The aim of the project was to identify the scientific and technical bases for the assessment of the safety functions of “the capture and minimization of the leakage” qualities of near-field processes in deep repositories for high-level radioactive waste (HLW) and spent nuclear fuel in the disposal system concept set out in the DGR reference project. One extensive research report (Vokál ed., 2008) and a number of partial reports addressed practically all aspects of near-field processes and put forward recommendations for future research.

In 2007 SÚRAO commissioned a further research project entitled “Research into Processes of the Far Field Interaction of a Deep Repository for Spent Nuclear Fuel and High-level Radioactive Waste” which was conducted by the Czech Geological Survey and the Technical University in Liberec. The basic goal of the research was to define a scientific and technical framework for the evaluation of the main function of the far field in the natural barrier. The research dealt with those physical, chemical and geological processes in the geosphere that might retard the transport of radionuclides. (Pačes and Mikšová eds. 2013).

The latest stage of research commenced in 2015 with the approval of the "Medium-term Research and Development Plan for Deep Repository Siting in the Czech Republic for 2015–2025". This document is based on the Updated Concept of Radioactive Waste and Spent Nuclear Fuel Management, official notice of which was taken by the Government in 2014. On the basis of a detailed analysis of work conducted in the Czech Republic and other countries over the previous 20 years or so, research and development targets were identified relating to the selection of two DGR candidate sites in 2020 and the selection of the final site in 2025. The identification of the research to be performed during these periods is founded both on the need to characterize candidate sites to a level sufficient for the comparison of the sites in terms of feasibility, safety and repository impact on the environment and the need to prepare methodologies, tools and procedures for repository safety and feasibility assessment in such a way that the results of research activities at individual sites can be assessed. It is intended that the document will be updated on a regular basis (approximately every five years) in compliance with knowledge and experience gained through research and development work and its utilization to date in the site selection process. The opinions of all the stakeholders concerned, namely the State Office for Nuclear Safety and other state institutions (e.g., the Ministries of Industry and Trade and the Environment and the Czech Mining Authority), as well as of representatives of the various communities concerned and the local public will be included in this document.

The following range of research issues will be considered in the document:

- Requirements, suitability indicators and criteria for DGR siting
- Source term
- Geological characterization of the rock environment, its development and stability;
- Research and development work required for safety assessment purposes (operational and long-term safety)
- Research and development work required for the assessment of the technical feasibility of the DGR (technical design, waste disposal containers, DGR thermal dimensioning, handling and transport, absorbing materials, sealing plugs, sealing system and underground construction work)
- Research and development required for the preparation of the EIA study
- Research and development for the preparation of the monitoring program
- Research and development concerning a study of the impact of the DGR on the various social and economic aspects involved.

The document also identifies the research and development work that should be conducted for the selection of two DGR candidate sites in 2020 and the selection of the final site in 2025. Certain activities, particularly those related to long-term safety assessment and design, can begin prior to the commencement of exploration work at the sites and, indeed, certain activities must begin prior to the commencement of exploration work in order that the results can be adequately assessed.

A number of respected Czech universities and academic institutions, namely the Czech Technical University in Prague, the Masaryk University in Brno, the Technical University in Liberec, the Institute of Geonics of the Czech Academy of Sciences (see http://www.ugn.cas.cz), the Institute of Rock Structure and Mechanics of the Czech Academy of Sciences and other research institutions which are deeply involved in international research projects, primarily the ÚJV Řež, are currently participating in the research program.

6.6 International Cooperation

The research and development component of the DGR project significantly exceeds normal requirements governing the construction of other nuclear facilities due to the necessity to prove long-term safety in a time horizon of thousands and even hundreds of thousands of years following closure. Moreover, the period of time over which DGR development and operation will extend, up to and including closure, will be significantly longer than that for other nuclear facilities. These aspects clearly place extremely high demands on research and development relating to the construction of the deep repository.

SÚRAO, when preparing the research and development plan, carefully considered a recommendation put forward by the NAPRO (National Programs) working group established by the European Nuclear Energy Forum (ENEF) for the preparation of National Programs which specifies the following three approaches to successfully accomplishing the set research and development targets:

- SÚRAO´s own research at the national level which is required for the implementation of the DGR project in the Czech Republic;
- joint research activities at the bilateral and international levels utilizing common sources and knowledge, particularly as part of EU-initiated Framework Programs for research and technical development (current program: Horizon 2020);
- on the basis of bilateral research agreements with organizations which already have an advanced DGR development program.

All these approaches currently are and will continue to be employed in order to achieve the targets set out in the Czech DGR development program.

The establishment of the IGD-TP (Implementing Geological Disposal Technology Platform) represented a significant milestone in terms of international cooperation, and the opportunity it presents to participate in joint projects with more advanced countries in terms of DGR development is seen as being of exceptional importance. Currently, Czech organizations are involved in five out of seven defined priority areas (Table 6–3).

Table 6–3. Overview of IGD–TP projects with the involvement of Czech organizations

Area	Project title	Note
Waste forms and their behavior	CAST	In progress
Demonstration of DGR sealing plugs under operational conditions	DOPAS	In progress
Increasing trust in computer codes used in safety analysis	PEBS	A new CEBAMA project under preparation
Monitoring program	MoDeRn 2	Continuation of modern 1 under consideration
Long-term stability of bentonite in crystalline rock	BELBaR	

The participation of Czech specialists in projects organized by the IAEA and NEA-OECD is seen as being of particular benefit. The results of the LTD (Long-Term Diffusion) experiment underway at the Grimsel underground laboratory in Switzerland (GTS) in cooperation with Nagra and a number of other participating organizations are of major importance for the Czech national program. This long-term experiment is concerned with slowing down the transport of radionuclides in terms of their diffusion from fractures into crystalline matrices; the experiment involves the study of radionuclides in a natural environment. Two further projects including Czech participation are underway at the GTS: the LASMO (Large Scale Monitoring) experiment which is concerned with the long-term monitoring of structural-tectonic changes in a crystalline rock massif and the potential impact on the long-term stability of the massif, and the MaCoTe (Material Corrosion Test) experiment which involves the long-term assessment of the velocity and mechanism of the corrosion processes affecting waste container materials.

The participation of Czech specialists in the DECOVALEX (Development of Coupled Models and their Validation against Experiments) project is also of major significance—the sixth phase of the project is currently underway. The Czech team is concerned with the study of water flow, tracer transport and reactive transport in a granite massif based primarily on tunnel inflow (flow rate and quality) measured in the Bedřichov tunnel. The main issue concerns the inhomogeneity of water flow, i.e., the heterogeneous distribution of water as a result of conduits of different size (faults, fractures) and the relationship between water quantity and flow velocity.

The Ruprechtov natural analogue experiment conducted in north-western Bohemia (Woller, 2006) was completed in 2014 with the compilation of an extensive experience report. During the course of the project relevant documentation in the form of publications, conference proceedings, reports (including internal company reports) and diploma and Ph.D. theses was produced and entered into the database (Noseck and Havlová eds. 2014).

6.7. Role of the Public in the Decision-Making Process Regarding DGR Siting

The deep repository and its siting in the Czech Republic, as in other countries, is both an important and, sometimes, controversial and sensitive issue in terms of acceptance primarily by the public concerned.

The Dialogue on the Deep Repository working group, an advisory body for the Minister of Industry and Trade, was established in 2010 with the remit of improving the level of involvement of the local public in the siting process and promoting transparent dialogue between all the parties concerned with DGR development.

DGR projects for HLW and SNF disposal are of a multidisciplinary nature and have been underway in many countries worldwide for a number of decades. As far as the Czech project is concerned, statutory representatives of the local communities of the sites concerned and non-profit organizations have made it clear that it is necessary to seek a new approach to realizing this important national project which is more in compliance with local interests and views.

The status of the Dialogue working group was upgraded at the end of 2014 to that of advisory body to the Government Council for Raw Materials and Energy Strategy, i.e., it now falls under the umbrella of the national Government. Consequently, the conclusions and recommendations of the working group can now be presented directly to the Government for discussion.

The main objective of the Dialogue working group is to improve the transparency of the DGR siting process and strengthen the role of the communities concerned, the public concerned and the micro-regions in which the candidate sites are located. The Dialogue working group is searching for a way in which to modify relevant legislation and strengthen the power of the communities concerned in the DGR siting process. The working group is currently working on preparations for the introduction of special draft legislation according to which the decision on the selection of two candidate sites and the final site in 2020 and 2025 respectively as specified in the Concept will be approved by the Government with the full consideration of the views of the communities concerned. Should the decision not contain the views of the communities concerned or should their opinions be unfavorable, it is proposed that the Government will decide only following a public hearing and subsequent discussion in the Senate and, finally, on the basis of the position taken by the Senate. It is generally considered that such a stipulation will guarantee that the voice of the communities concerned and the local public is heard and will be fully respected in the final decision. The structure of the Dialogue working group is shown in Table 6–4.

Table 6–4. Structure of the Dialogue working group (that was established in 2010 and reorganized in 2015

Organization/Institution	**Number of representatives**
Representatives of local municipalities (2 for each site)	14
State administration authorities (MPO, MŽP, SÚRAO, SÚJB*)	4
Local non-profit organizations (1 for each site)	7
National non-profit organizations	2
The Parliament (Chamber of Deputies and the Senate)	2
Expert public (a legal expert, technical expert and sociologist)	4

*MPO—Ministry of Industry and Trade, MŽP—Ministry of the Environment, SÚRAO—Radioactive Waste Repository Authority, SÚJB—State Office for Nuclear Safety

At present, prior to the commencement of the first stage of geological investigation work, the DGR project involves seven sites covering a total of 40 municipalities and more than 18,000 inhabitants. The representatives of the individual sites in the Dialogue working group thus represent a relatively large group of the population with a wide range of opinions on the issues involved. This provides a unique opportunity for finding a well-balanced and widely acceptable solution.

The draft legislation will not only aim to contribute to the strengthening of the role of the communities concerned and the local public in the DGR site selection process but it also suggests a more balanced method with respect to compensation both for the communities concerned and the relevant micro-regions for which future DGR construction may well provide an important economic development boost. The specific amount of compensatory contributions and support will be subjected to further discussion as will detailed stipulations in the legislation which specify the procedure concerning how individual communities will agree on their final position. In short, it is important that an acceptable approach be identified in terms of the legislative background to the DGR siting decision-making process in the Czech Republic which fully complies with the decision-making principles applied in other European countries and worldwide in this respect.

6.8 Conclusions

DGR development in the Czech Republic is governed by a relatively tight schedule—2020 for the selection of two candidate sites, and 2025 for the selection of the final site—and represents, in terms of its multidisciplinary nature and the amount of funding, a unique and significant research challenge for the Czech Republic. Without the involvement and concentration of the research capacities available in the process and the support of the whole of Czech society it will simply not be possible to fulfill this task. SÚRAO and the senior state authorities involved are well aware of this fact and will make every effort to achieve the various objectives in an atmosphere of mutual cooperation.

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6.10 Acronyms

CTU—Czech Technical University

DGR—Deep Repository

ENEF—European Nuclear Energy Forum

GTS—Grimsel Underground Laboratory in Switzerland

HLW—High-Level Radioactive Waste

IGD-TP—Implementing Geological Disposal Technology Platform

LASMO—Large Scale Monitoring

LTD—Long-Term Diffusion

NAPRO—National Programs

NPP—Nuclear Power Plant

RAW—Radioactive Waste

SNF—Spent Nuclear Fuel

SURAO—Radioactive Waste Repository Authority

TMB—Tunnel Boring Machine

URL—Underground Research Laboratory

VLF—Very Low Frequency

Chapter 7

Towards Implementation of the Spent Nuclear Fuel Repository in Finland

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ABSTRACT: Since 2004, Posiva Oy, the company responsible for the final disposal of spent nuclear fuel of the Olkiluoto and Loviisa nuclear power plants in Finland, has constructed an underground rock characterization facility on the planned repository site in Olkiluoto, western Finland. This facility, called ONKALO, has provided an opportunity to carry out further site investigations, develop construction techniques, and test and demonstrate the engineered barrier system in the actual repository environment. As a result of these investigations and developments, application for a license to construct the encapsulation plant and geological repository was submitted in 2012. The Radiation and Nuclear Safety Authority in Finland (STUK) gave a positive review on the safety of the construction of the disposal facility in early 2015, and Posiva received the construction license in November 2015. Recently Posiva has identified and assessed the uncertainties related to the readiness for starting the construction phase, and the start of the construction has been scheduled for 2016. The next major milestone will be submission of the operation license application, scheduled for 2020. Here, we summarize the progress made in Finland's nuclear waste disposal program since the Fourth Worldwide Review in 2006, and outline the program for the near future.

7.1 Introduction

In 2001, the Finnish Parliament ratified the decision-in-principle related to the disposal of spent fuel from Finnish nuclear power reactors. According to the decision, the repository would be located at Olkiluoto, in the municipality of Eurajoki, on the western coast of Finland, and the disposal would be based on the KBS-3 concept. KBS-3 is an abbreviation of kärnbränslesäkerhet, nuclear fuel safety, which is a technology for disposal of high-level radioactive waste developed in Sweden at SKB—see https://en.wikipedia.org/wiki/KBS-3. The next step of the program was the construction of an underground rock characterization facility on the planned repository site. The facility is intended for final assessment of the previous conclusions on the suitability of the Olkiluoto site for a safe geological repository. The construction of the underground facility (ONKALO) started in 2004. Altogether about 5 km of tunnels have now been excavated, and the underground investigation program has been active for about 10 years. Positive outcome of the assessment based on ONKALO experience made it possible to proceed to submission of the application for the construction license for the disposal facility in 2012. The radiation and Nuclear Safety Authority in Finland (STUK) gave a positive review on the safety of the project in early 2015, and Posiva received the construction license in November 2015.

In the period that has passed since the Fourth Worldwide Review in 2006 (Witherspoon and Bodvarsson 2006), the focus of the Posiva program has been on preparation and submission of the construction license application for the encapsulation plant and the geological repository. The comprehensive application consists of the Preliminary Safety Analysis Report (PSAR), the Safety Case portfolio, and a large number of technical documents. In parallel, progress has been made in construction, investigation, and development of the technology needed for encapsulation and disposal of the spent nuclear fuel.

Here, we summarize the progress made in Finland's nuclear waste disposal program since the Fourth Worldwide Review, and outline the program for the near future. A recent account of progress made since 2006 can also be found in the reports for the three-year research and technological development programs (Posiva 2010, Posiva 2013a, Posiva 2015).

7.2 Construction of ONKALO

7.2.1 Site Description

Since the ONKALO is intended to become a part of the final repository, it has been designed and constructed according to the requirements for nuclear facilities (Figure 7–1). Specifically, this means that the construction must comply with the quality assurance criteria posed by STUK, the Radiation and Nuclear Safety Authority in Finland. A specific graded quality assurance (QA) program for ONKALO, based on regulatory guidelines and IAEA safety guides, has been launched to complement the ISO 9001-based QA applied by Posiva for its normal RTD (Research, Technical Development and Design) activities.

The excavation works for ONKALO started in September 2004. The tunneling work was carried out using drill and blast technique. The vertical shafts were excavated by raise boring. The main characterization level at the depth of 420 m was reached in 2010. Another major step in the construction was reached in 2014 when the two vertical shafts for ventilation down to 455 m level were finalized. Preparations for the raise boring of the shafts were particularly challenging, because the predicted amount of water leakage was too high, and needed to be reduced to the required level by pre-grouting.

Figure 7–1. Posiva's ONKALO site in Olkiluoto in summer 2014. The ramp and entrance to the ONKALO access tunnel are visible in the center of the image. Posiva has constructed a project office and various technical buildings at the site. The infrastructure is designed so that it can be utilized for the operation of the future disposal facility.

The final design of the ONKALO layout has been modified during progress. One major change was that the previous plans for constructing an additional characterization level at 520 m depth were considered unnecessary. The deepest parts of ONKALO now extend to 455 m depth. The inclination of the tunnel is 1:10, and the length of the access tunnel is about 5 km.

The ONKALO premises in 2015 consist of four main sections: the inclined access tunnel, the vertical shafts, the demonstration area at 420 m level, and the technical facilities at 437 and 455 m levels (Figure 7–2). Also, five dedicated investigation tunnels were constructed at several depth levels to study properties and processes of the bedrock. Personnel access to ONKALO is, so far, arranged by vehicles driving in the access tunnel. The personnel elevator will be installed in a later phase.

Posiva has gained remarkable experience and collected significant amount of information during the construction of ONKALO. Especially the demonstration area at the 420 m level has provided an opportunity to learn how to construct deposition tunnels and deposition holes according to a wide range of challenging requirements. We excavated four demonstrations tunnels that are dimensioned and implemented as if they were actual deposition tunnels. We also constructed 10 test deposition holes into the demonstration tunnels to learn how to implement exact deposition holes, and to use them for testing the installation of the engineered barrier system.

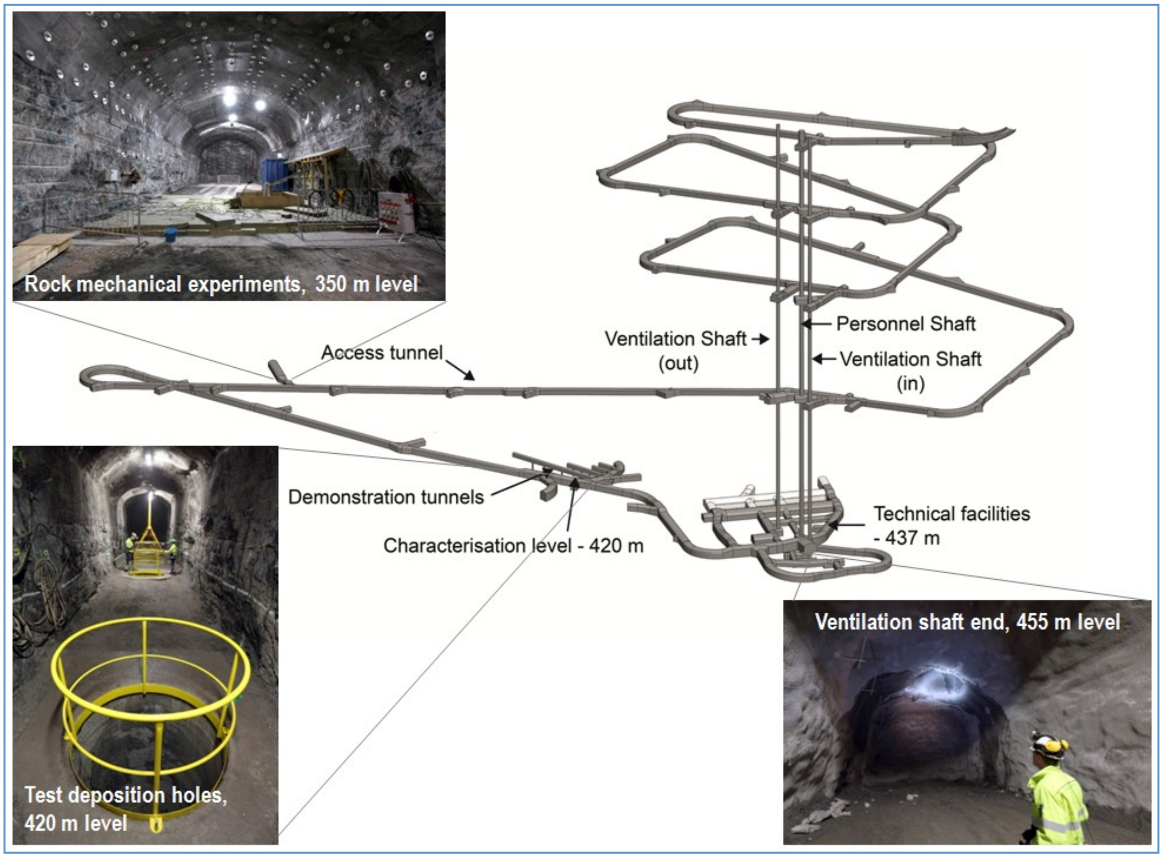

Figure 7–2. Layout for the ONKALO premises in 2015. ONKALO consists of the access tunnel (5 km), vertical shafts for ventilation and future personnel access, demonstration area at the characterization level 420 m and technical facilities. ONKALO will serve as the access route to the future repository.

Currently the tunnel excavations are minor. We are focusing on the demonstrations of the final disposal concept at the 420 m level. Recently we carried out a testing campaign for the bentonite buffer and disposal canister installations. At the moment, a test plug for the disposal tunnel end is constructed in the demonstration area. Various long-term investigation programs, such as the rock spalling experiment, the sulphate reduction experiment and the rock matrix diffusion experiment, continue in the dedicated investigation areas.

7.1.2 Site Investigations and Monitoring

Since the Fourth Worldwide Review, the site investigations both above ground and underground have been active. With the construction of ONKALO, the focus of the site investigations at Olkiluoto moved underground, to the conditions near the repository volume. The Program for Repository Host Rock Characterization in ONKALO (ReRoC, Aalto et al. 2009) provides an overview of the plans, which are designed to obtain the relevant site-specific knowledge. It includes geological mapping in the tunnel and in the shafts, as well as investigations in pilot holes and characterization drillholes. The experimental studies carried out in different locations in the tunnel aim to obtain information on rock properties that are comparable to the rock at the disposal depth.

Around 30 pilot holes and characterization holes (typical length from 100 to 200 m) were core drilled in ONKALO. Also about 400 technical core drilled holes, lengths from 5 m to 150 m, were drilled, mainly for grouting purposes but used also for investigations. Additionally about 200 short holes, typically only a few meters long, were core-drilled for various experiments. The entire facility is being equipped for gathering data on host rock hydrogeological, hydrogeochemical, rock mechanical and geophysical conditions. An important part of the investigations program has been the prediction-outcome process, in which models were used to predict the conditions ahead. These predictions assisted in the further design of the facility, and form an important part of the learning process.

Field work at the surface has continued as well, consisting of deep drillings, groundwater sampling, geophysical and geohydraulic measurements, geological mapping, and various monitoring networks. The number of deep-cored boreholes at Olkiluoto is now 58. The majority of the deep boreholes are targeted to characterize the bedrock at the repository depth between 400 and 450 m. In recent years the above-ground site investigations have focused on finalizing and continuing the various borehole investigations and sampling programs. To achieve the safety of work, multi-disciplinary characterization of the Olkiluoto bedrock has been extended. The work is particularly focused on specific topics that require further clarification, such as the rock mechanical properties, the deformation zones, and the groundwater characteristics. Results from this work were published in the Olkiluoto Site Description reports (Posiva 2009; Posiva 2011).

In recent years the focus on the site investigations has shifted to the effort for a more detailed understanding of functional processes in the bedrock. A detailed site model of the rock volume near the demonstration area in ONKALO at the planned repository level has been developed (Figure 7–3). This model is targeted especially for the Rock Suitability Classification (RSC) needs. Posiva's rock suitability classification system, developed for locating suitable rock volumes for the repository design and construction, is presented by McEwen et al. (2012).

The Olkiluoto Monitoring Program has continued since 2004, when construction of ONKALO started. It comprises rock mechanics, hydrological, hydrogeochemical, and surface environment studies to monitor natural changes within the geosphere and biosphere of Olkiluoto, as well as the effects of ONKALO construction and other human activities. In addition, controlling the use of foreign materials in the construction of ONKALO has been a part of the program. The aim of the monitoring has been to observe changes in the host rock and surface environment that may affect the long-term safety of final disposal of spent nuclear fuel. Additional aims are to obtain data on the properties of the site and the environmental impact of the project, and to obtain information on the response of the host rock to the excavation for the benefit of further planning of construction, operation, and eventual closure of the disposal facility. The updated monitoring program (Posiva 2012a) was implemented 2012, and will continue until the projected beginning of the operational period of the repository. The update also presents generic plans for monitoring during operation. The new objectives to monitor the performance of the engineered barriers and radioactive releases into the environment are included in the update.

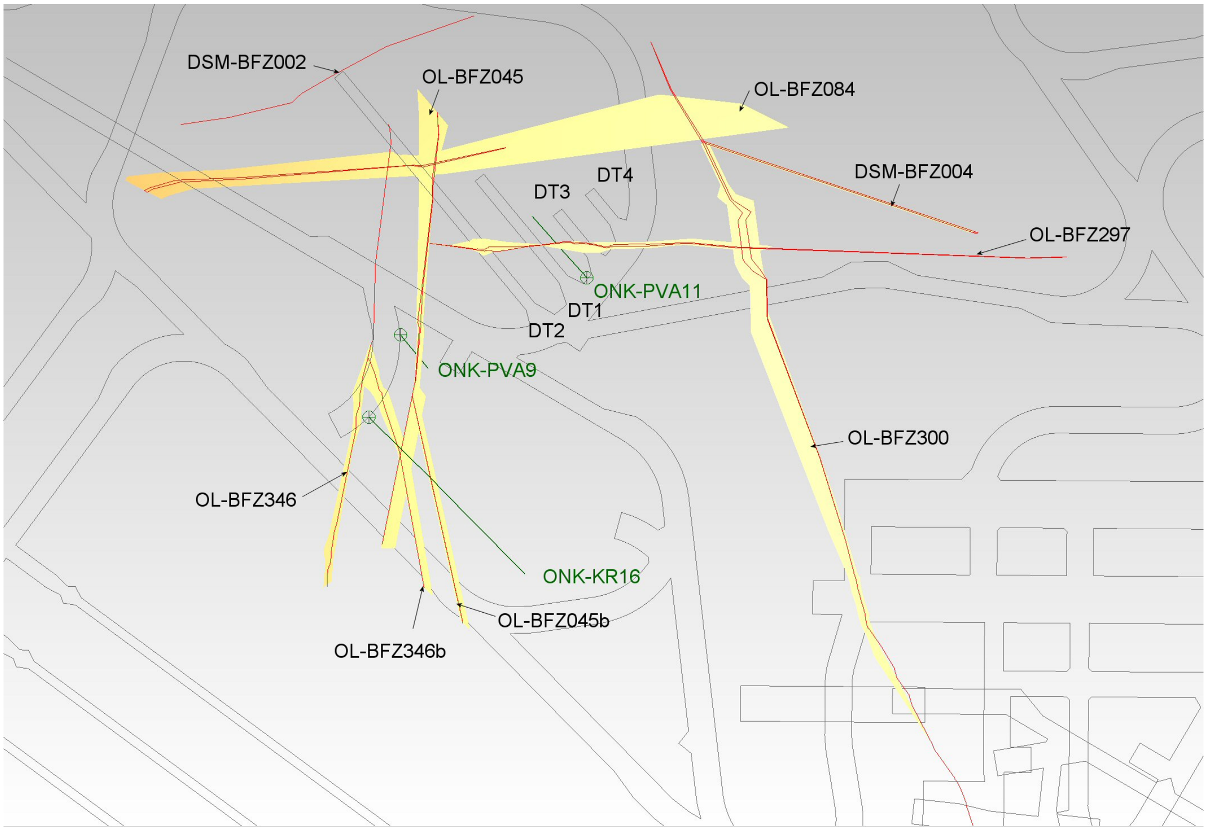

Figure 7–3. Detailed scale geological model for the demonstration area in ONKALO (420 m level). The modeled brittle deformation zones (BFZ) are shown by red (core of the zone) and yellow (influence zone). The modeling is based on observations from the tunnels and from the adjacent boreholes. The core drilled boreholes are marked green. DT1-DT4 refers to demonstration tunnels 1–4.

7.3 Development of Technology and Disposal Concept

The advancement in the facility design for the encapsulation plant and geological repository (also called disposal facility) was published in the reports, Facility Description 2006, 2009 and 2012 (Tanskanen 2007, Tanskanen 2009, Palomäki and Ristimäki 2013). The planned facility complex will consist of the above–ground encapsulation plant and auxiliary buildings, and the underground access tunnel, shafts, technical facilities, and repository (Figure 7–4). The encapsulation plant and the repository will be directly connected by a shaft, in which a lift will transfer disposal canisters. The facility planning, especially for the encapsulation plant, has reached the detailed system design phase.

Posiva has compiled Production Line Reports for each engineered barrier (Canister, Buffer, Backfill, Closure, Underground Openings). These reports describe the design, production and initial state of the particular barrier (Raiko et al. 2012, Juvankoski et al. 2012, Keto et al. 2013, Sievänen et al. 2012, Posiva 2013b). Based on the feedback received for the construction license material, in 2014 we compiled a development program to summarize the remaining research, development and testing work needed for demonstrating the functioning of the final disposal concept. This work consists of studies, modeling and a variety of tests to demonstrate the feasibility and safety of the disposal concept for the safety case of the operation license application.

A major achievement in the development has been the design and production of the machinery prototypes for engineered barrier installation. Posiva has produced prototypes for transferring bentonite blocks, installing bentonite buffer into the deposition holes, transferring and installing disposal canisters, and installing backfilling blocks into the deposition tunnels (Figure 7–5). These prototypes were successfully tested at the site in the demonstration tunnels in ONKALO. Further development and testing are ongoing. Also, significant work has been done for analysis of the purchase and potential suppliers of buffer and backfill materials and disposal canister manufacturing. The work involves development of quality control procedures.

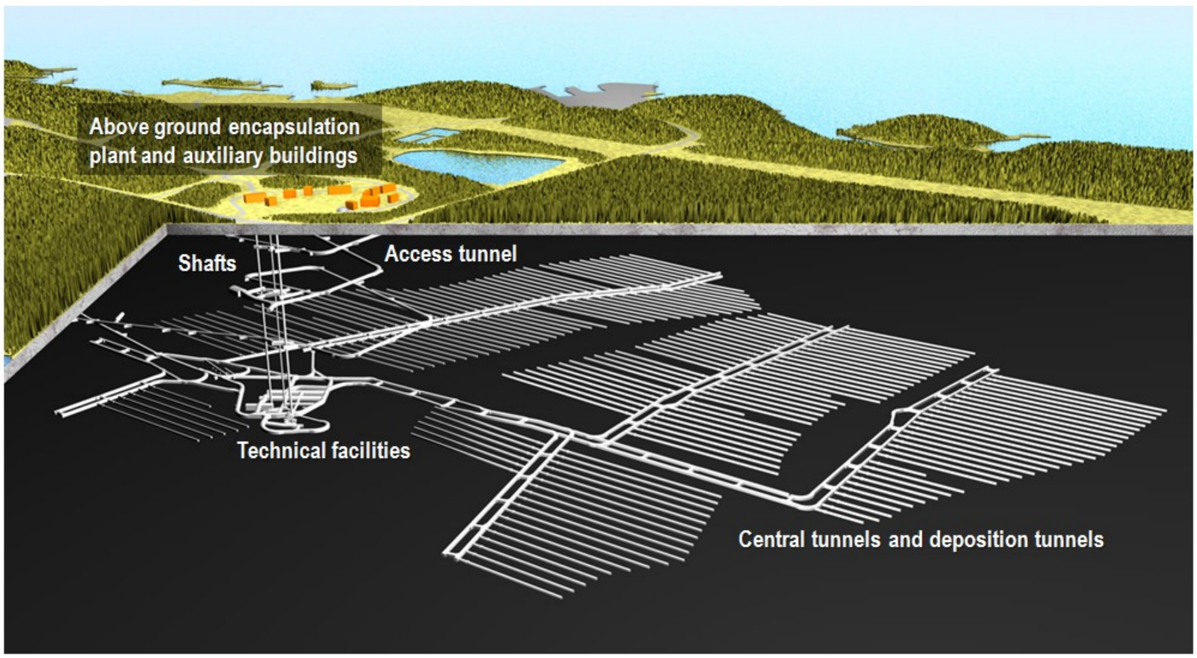

Figure 7–4. Posiva's planned facility complex that consists of the above-ground buildings, connections to the repository level, and the repository itself, including central tunnels, deposition tunnels and deposition holes.

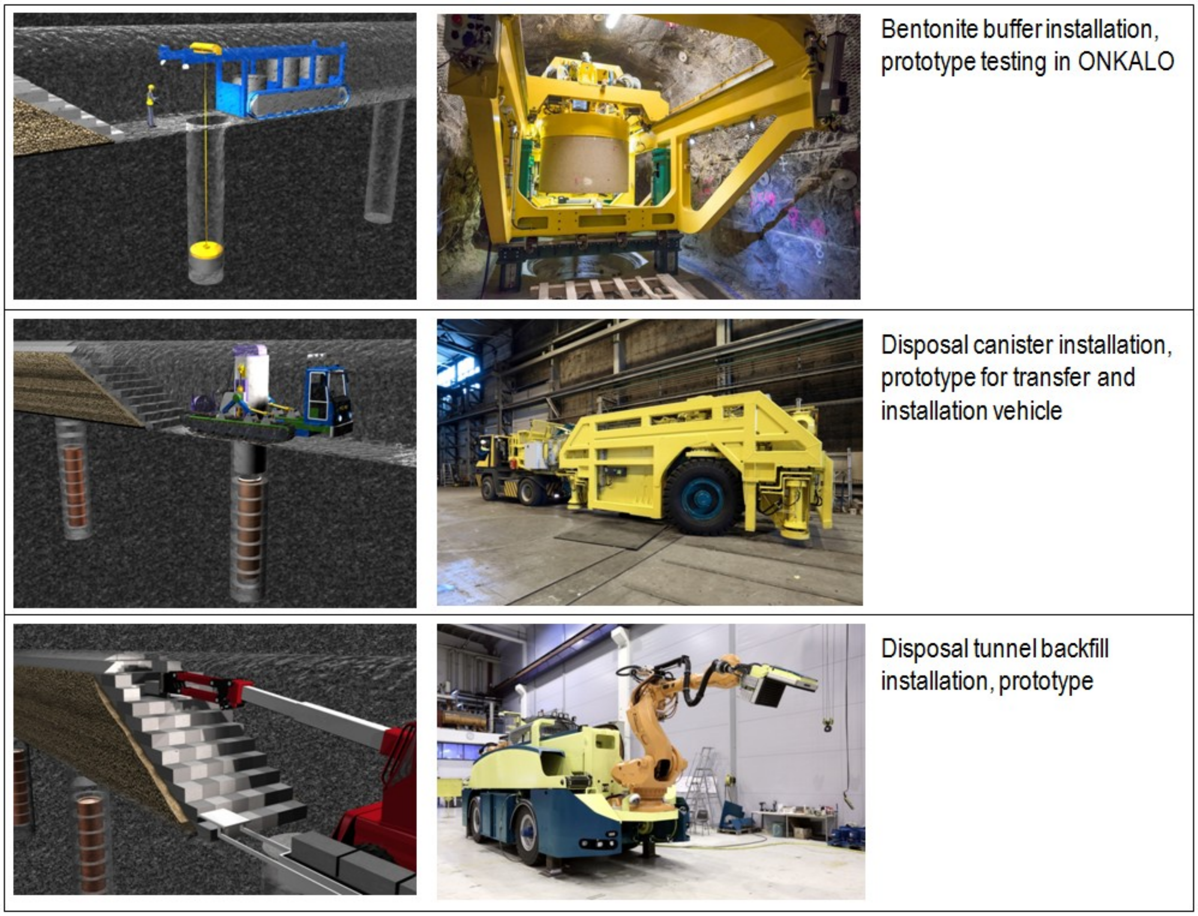

Figure 7–5. Posiva's prototype machinery for the installation of engineered barrier system. The prototype tests for bentonite buffer, disposal canister and tunnel backfill installation continue in the demonstration tunnels in ONKALO.

The sealing of the disposal canister has been one of the main technical challenges in the KBS-3 concept. Posiva's own testing has focused on the high-vacuum electron-beam welding technology, but we have also participated in the work carried out by SKB in Sweden on the friction-stir welding technology. Based on the experience by SKB concerning the success of this method, and after a careful feasibility study between the two sealing techniques, Posiva decided to focus on the friction-stir welding in 2014.

An important project is being carried out in cooperation with SKB to assess the feasibility of a horizontal version of the KBS-3 concept. In this concept, the disposal canisters are emplaced directly in the tunnels, but in a horizontal position. The basic components and dimensions remain the same as in the reference vertical design. We plan to decide between the vertical and horizontal disposal within a few years.

7.4 Construction License Application

The basis for Posiva's construction license application material is the Preliminary Safety Analysis Report (PSAR), which consists of a general part with 11 documents, 28 thematic reports, and a set of about 180 system descriptions. A significant part of the application is the long-term safety assessment, the Safety Case (TURVA-2012), which consisted of 11 main reports (Posiva 2012b, Posiva 2012c, Posiva 2012d, Posiva 2012e, Posiva 2012f, Posiva 2012g, Posiva 2013c, Posiva 2013d, Posiva 2013e, Posiva 2013f, Posiva

2014), and tens of background reports. In addition, Posiva submitted separate reports on security, emergency preparedness and safeguards arrangements, a proposal for a classification document, project management documents, surveys of safety culture and quality management during construction, a design probability-based risk analysis (PRA) and fulfillment reviews on STUK's YVL Guides. During the processing of the construction license application Posiva has also provided over 100 additional documents and reports to STUK for review, fulfilling requests and Posiva's own development needs.

Posiva's application was subjected to a public hearing by the Ministry of Employment and Economy in 2013. The Ministry also asked for opinions from the designated sources, such as STUK, Ministry of the Interior, and the Ministry of the Environment. The Ministry received a total of 32 opinions. The statements comment on a wide range of long-term and operating safety issues related to the nuclear waste disposal facility, and in general comment on the construction license application processing.

Connected to the handling of the application, Posiva communicated the information related to the nuclear investment project to the European Commission, according to the Article 41 of the Euratom Treaty. On the basis of the information submitted, the Commission considers that the investment project fulfills the objectives of the Euratom Treaty and the Council Directive for the management of spent nuclear fuel and radioactive waste, and that the project promotes responsible management of spent nuclear fuel in Finland and the European Union. The Commission recognizes the complexities and long time–horizons associated with such large-scale projects. The project's long-term implementation requires establishment of data storage system that is capable of operating throughout the repository lifetime, including the post-closure phase. Well-defined mechanisms are required for the implementation of safeguards, and for transferring the data collected during the facility’s operation to the state.

STUK's comprehensive safety assessment was completed in early 2015. According to the assessment, the spent nuclear fuel encapsulation plant and the final disposal facility (repository) can be built safely. STUK's assessment considered the safety of the lifetime of the project, starting from the construction and commissioning, to covering operation, decommissioning and closure, as well as the long-term safety of the repository. Posiva's management system and safety culture were evaluated, in addition to the plans on the management of facility aging, retrievability of the disposed fuel, and spent fuel transports. In addition, STUK carried out inspections in accordance with the periodic inspection program, which had already started with the construction of ONKALO. Inspections covered Posiva organization's readiness for the construction of nuclear waste facility. According to STUK, the plans Posiva has presented are adequate and appropriate for the construction phase. However, STUK made a total of seven remarks and exclusions in the safety assessment. Some of these requirements must be addressed before construction can start, and the remaining either during the construction phase, or, at latest, by submission of the operation license application. STUK has also revised its observations in the decisions submitted directly to Posiva.

7.5 Next Steps

Posiva has now reached an important phase in the final disposal project. The positive outcome of the safety review for the construction license application and the granting of the license in 2015 have advanced the project substantially.

Recently Posiva has identified and assessed the uncertainties related to the readiness for starting the implementation phase. The main uncertainties relate to construction of the deposition tunnels and deposition holes according to the demanding requirements, installation of the buffer and backfill, demonstration of the mechanical strength of the disposal canister, and the thermal conductivity of the buffer. Posiva has scheduled the decision on initiating the implementation for 2016. The construction of the encapsulation plant and further underground premises is expected to start in around 2016–2018. The next major milestone will be submission of the operation license application, scheduled for 2020 (Figure 7–6). Posiva has already launched a project for compiling a successful application.

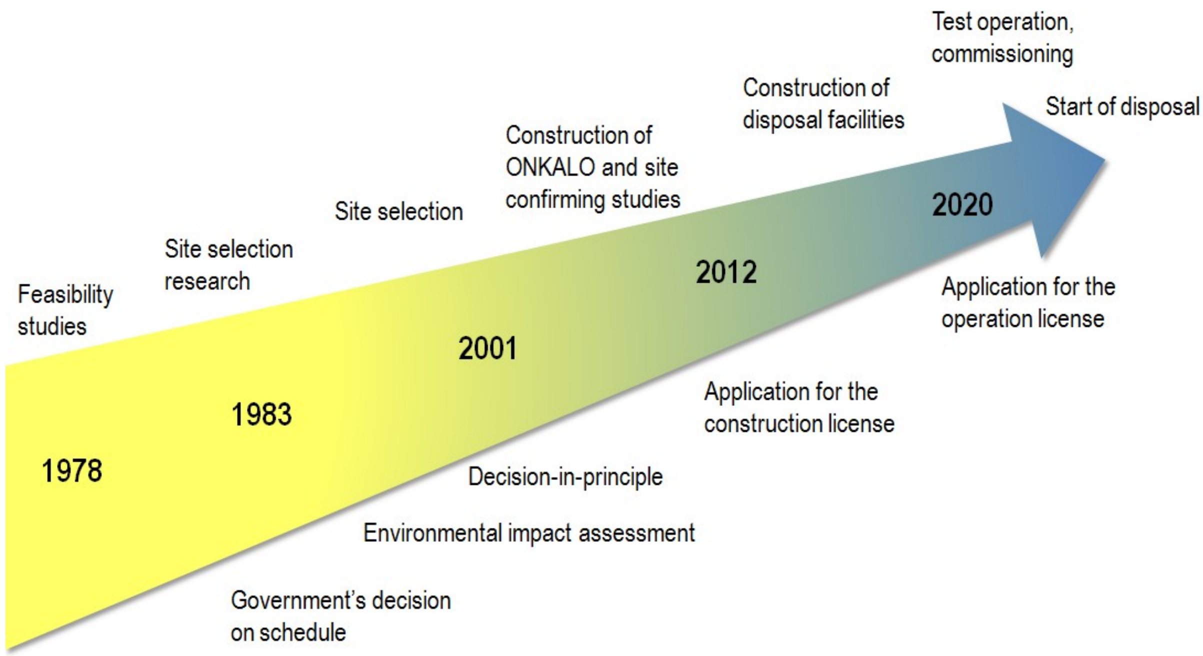

Figure 7–6. Past and future steps of the final disposal program in Finland.

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7.7 Acronyms

BFZ—Brittle deformation zones

ISO—International Organization for Standardization

KBS-3—An abbreviation of kärnbränslesäkerhet, nuclear fuel safety, which is a technology for disposal of high-level radioactive waste developed in Sweden at SKB—see https://en.wikipedia.org/wiki/KBS-3)

PRA—Probability-based risk analysis

PSAR—Preliminary Safety Analysis Report

QA—Quality Assurance

ReRoC—Program for Repository Host Rock Characterization (see Aalto et al. 2009)

RSC—Rock Suitability Classification

RTD—Research and Technical Development

STUK—The Radiation and Nuclear Safety Authority in Finland

YVL—Regulatory Guides on nuclear safety. Group A: Safety management of a nuclear facility, Group B: Plant and system design, Group C: Radiation safety of a nuclear facility and environment, Group D: Nuclear materials and waste, Group E: Structures and equipment of a nuclear facility (see http://www.stuk.fi/web/en/regulations/stuk-s-regulatory-guides/regulatory-guides-on-nuclear-safety-yvl-)

Chapter 8

Progress Towards Geological Disposal of High-Level and Intermediate-Level Long-Lived Radioactive Waste at an Industrial Scale: The Cigéo Project in France

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ABSTRACT[1]: Research on solutions for disposal of high-level and intermediate-level long-lived radioactive waste started in France in 1991. The Parliament passed a Planning Act in 2006, proposing geological disposal of radioactive waste, with commissioning of the repository by 2025. In 1998 the Meuse/Haute-Marne site was selected for licensing. The repositories will be located 500 m underground, in the Callovo-Oxfordian Argillite layer. The geological disposal project entered the industrial development phase in 2011. The underground facilities are designed to receive vitrified waste resulting from the reprocessing of spent nuclear fuel produced by all French reactors during their service lives. They will also receive intermediate-level and long-lived waste. Surface facilities will receive, check and prepare waste packages before disposal. Waste disposal will be gradual. The site is expected to operate for about 120 years. French law requires that disposal be reversible, and therefore the waste packages must be physically retrievable. French law requires consultation via public debate to ensure that major projects effectively meets requirements. Andra, the French National Radioactive Waste Management Agency, aims to raise collective awareness of the existence of radioactive waste, and to provide a forum for rational, rather than polemical, discussion. The ultimate purpose is to get society to think about and take responsibility for the issue.

8.1 Legal Process and Developments of the Cigéo Project

8.1.1 Historical background and the French Research Law of 1991

The geological disposal program for high-level and intermediate-level long-lived radioactive waste in France has been in progress for 25 years now. Following the failure of efforts to find study sites, a

[1] In the framework of the license application for Cigéo, in the early 2016, Andra submitted a first set of documents to the French Nuclear Safety Authority (NSA). The NSA requested to conduct an international review, and the review conclusions were published at https://www.asn.fr/Informer/Actualites/CIGEO-revue-internationale-du-dossier-d-options-de-surete. The set of documents includes the safety options for the operations and for the long-term, a first draft of the master plan of operations, and the approach to reversibility, including technical options for retrievability.

moratorium was declared by the Prime Minister in 1990. The period that followed was used to analyze and understand the reasons for this failure and to put forward a step-wise process to make further progress in finding solutions for radioactive waste disposal. The outcome of this effort was the Act of 30 December 1991 (French Law 91-1381), a research law providing for a 15-year research program, which would study alternative solutions. Within this context, detailed research work began on partitioning (i.e., reprocessing to separate minor actinides), and transmutation (i.e., process to modify nuclides in either less radioactive or short-lived radionuclides), and long-term storage, alongside research on geological disposal. From more than 30 candidate sites for geological disposal, three were selected by the Government in 1993 for further investigations carried out from the surface. Preliminary studies gave rise to a first set of performance and safety assessments to support the license application to build and operate underground laboratories. The three selected sites:

- The first is in granite, in the Vienne "department," 500 km southwest of Paris. Granite was covered by an 80-m thick sedimentary formation
- The second is in an indurated clay formation, in the Paris Basin, 240 km east from Paris, at the border between two "departments," the Meuse and the Haute-Marne, later giving rise to the Meuse-Haute-Marne site
- The last is in a clay formation, in the Gard "department," 720 km south from Paris. Field investigations revealed a 400-m thick, high quality indurated clay formation.

Of the three applications, only one license to build and operate an underground research laboratory was approved in 1998, which was for the Callovo-Oxfordian argillaceous formation of the Meuse/Haute-Marne site, near the village of Bure.

The reasons for rejecting the site in granite was that the massif is too small to accommodate the amount of radioactive waste to be disposed, and thus constructing an underground laboratory would not be relevant. Moreover, the National Assessment Board, launched by the same 1991 law, would have preferred an outcropping granite, above which there would not be any water resource. Lastly, the candidate site in the Gard District (Southern France) posed a scientific challenge related to its long-term geodynamic evolution. The project was shelved in the face of strong local opposition.

Along with the political decision to build an underground laboratory in Meuse/Haute-Marne, provided a license was granted, the Government launched a new program for finding another granitic site. Due to an unfavorable national context and to local opposition, the mission entrusted to three high-level civil servants never succeeded.

8.1.2 Construction of the Underground Laboratory and Launch of the Experimental Program

By the end of 1999, enough progress had been made to start detailed investigations and first experiments. The underground laboratory was constructed, with two vertical shafts down to –500 m, and a first set of galleries. The general architecture of the underground laboratory in 2005 is shown in Figure 8–1. As a major rock mechanics experiment, digging of the main shaft was stopped at –445 m, and a small preliminary experimental gallery was constructed. The first migration experiments were undertaken to get first performance results in the clay formation as early as possible. From this same gallery, a set of boreholes was drilled in the direction of the shaft to be dug. These boreholes were used to monitor digging of the shaft from –445 to –490 m depth, and provided important information for further rock mechanics works and modeling.

Innumerable results from investigations and experiments were collected, analyzed and described in the Dossier 2005. That document concluded that a geological repository within the Callovo-Oxfordian argillite layer is feasible, that the repository will be safe in the short and long term, and that disposed waste will be retrievable, which is a requirement introduced by the 1991 Law and confirmed by the government in 1998.

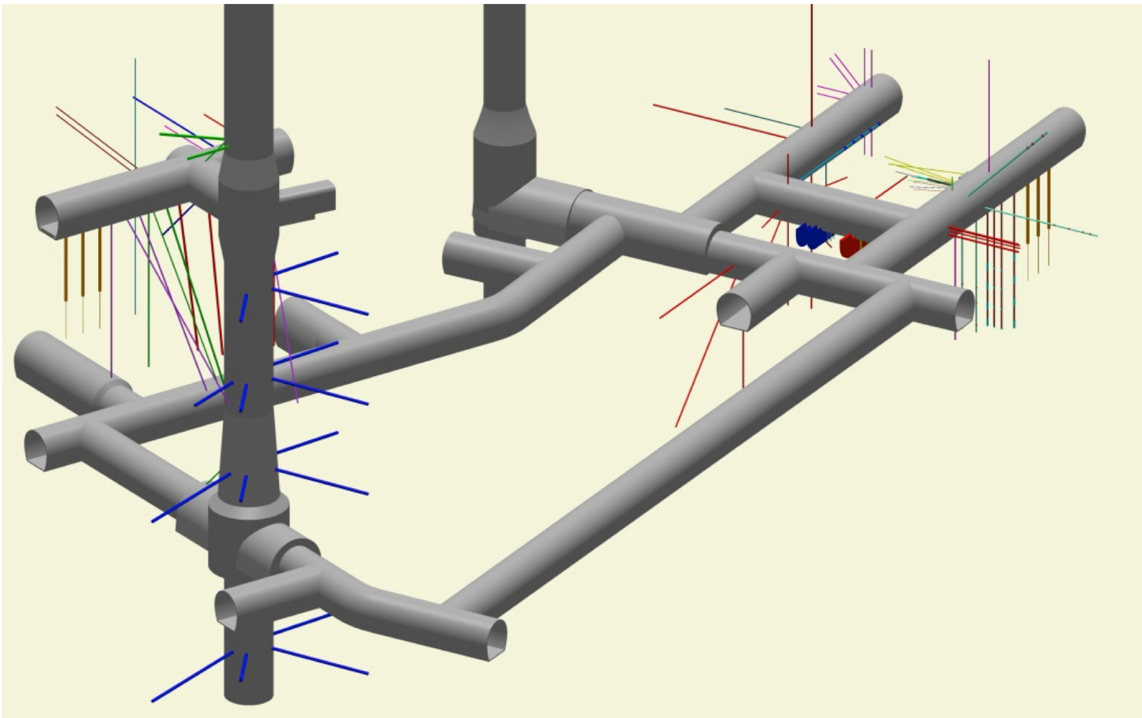

Figure 8–1. Architecture of the Bure underground laboratory in 2006

The main characteristics of the formation include:

- A thickness of approximately 130 m
- A lateral continuity over several tens of kilometers, without any major fracture
- A mineralogy on the order of 50% clay minerals, with the remainder being composed of carbonates and silicates, which ensure the mechanical strength of the formation
- A porosity showing the occurrence of small size pores, and low water content equal to less than 15% saturation
- The absence of water movement, with hydraulic conductivity measured at different scales and locations being less than 10^{-12} m/s
- A high radionuclide retention capability

Those characteristics have also helped delineate a 250 km^2 zone (Transposition Zone), marked with a blue line in Figure 8–2, which is limited by natural boundaries within which the properties of the Callovo-Oxfordian argillite were considered as identical to those observed or measured from the underground laboratory.

8.1.3 Taking stock of the 15 year research results

The publication of the Dossier 2005 marked the conclusion of the 15 years of research prescribed by the 1991 Law. The document compiled the full set of technical and scientific information needed for a sound decision. In accordance with existing rules, the Dossier 2005 underwent several assessments, including an international assessment led by the OECD/NEA (2006); cf. *Safety of Geological Disposal of High-level and Long-lived Radioactive Waste in France: An International Peer Review of the "Dossier 2005 Argile" Concerning Disposal in the Callovo-Oxfordian Formation*. The Peer Review approved the quality of the submitted results, and formulated complementary needs which justified the continuation of research programs.

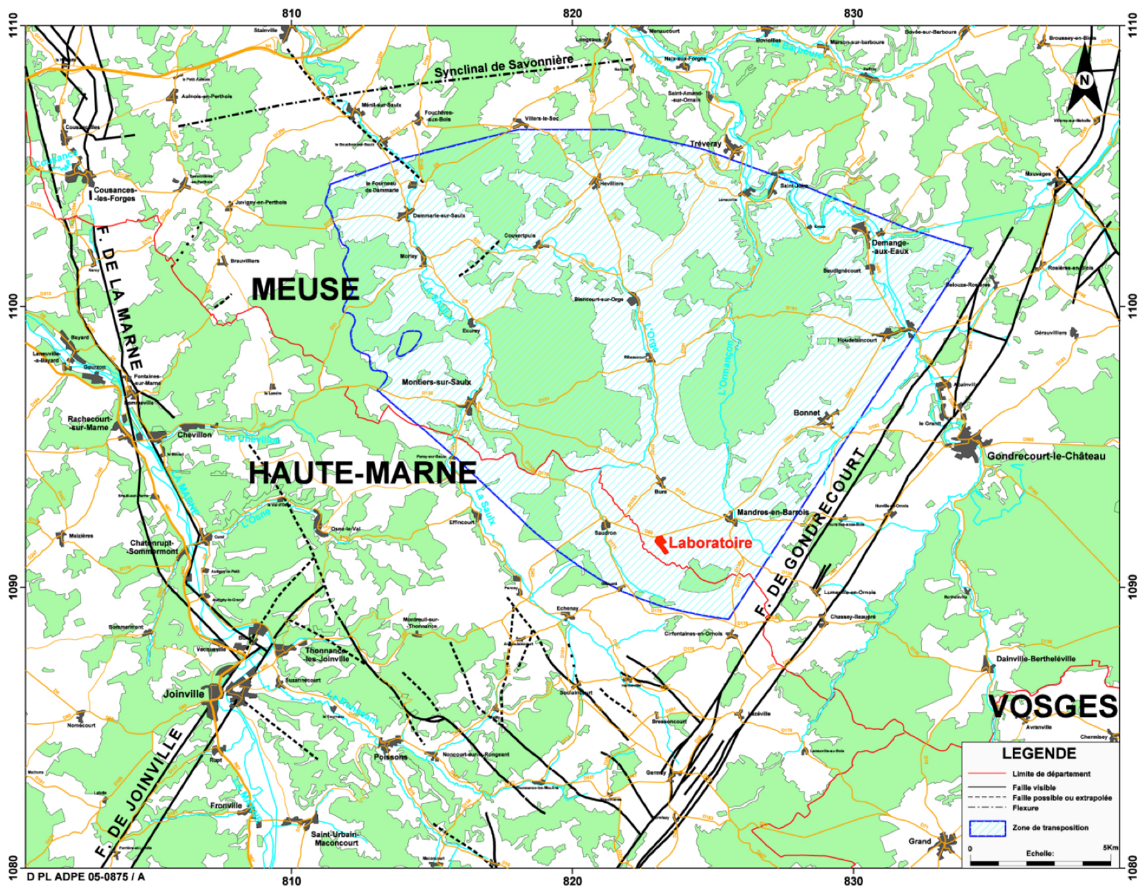

Figure 8–2. Definition of the 250 km² transposition zone (delineated in blue)

In parallel with all the assessments, the government initiated a national public debate to ascertain the opinions of the various stakeholders. The debate provided an opportunity to respond, notably to the concerns of nearby residents about the potential future impacts of a geological repository.

All the results acquired during this first phase of experiments, including experiments at the targeted depth of 490 m, were also incorporated into the Dossier 2005, and submitted to the French Government and Parliament, at the same time as the partitioning, transmutation and long-term storage reports.

The same year, the Government organized a public debate on the issue of radioactive waste management. Based on all gathered information, and after consulting the stakeholders, a new law was debated and passed by Parliament. This was the Planning Act of 28 June 2006, which sets out the main spent fuel management options, together with a roadmap for the disposal of high-level and intermediate-level long-lived waste. According to the Planning Act, a license application for the geological repository was to have been submitted to the Nuclear Safety Authority by 2015, and commissioning of the facility must take place in 2025. As for any great project to be launched in France, a Public Debate must be organized before the license application.

Field investigations were thus stepped up within the transposition zone, especially with additional boreholes, in order to get the same level of information as at the underground laboratory. Two-dimensional seismic surveys also led to better imaging of the underground structures, mostly focused on the argillite Callovo-Oxfordian formation. From these additional investigations, a restricted 30 km² zone named ZIRA (for Zone of Interest for Detailed Survey), a restricted investigation zone, as shown in Figure 8–3 was defined.

In 2009, Andra, the French National Radioactive Waste Management Agency, proposed to build the underground disposal facilities some 5 km north of the Bure underground research laboratory, in the ZIRA. After consulting the stakeholders at the local and national levels, the site of the underground facilities was chosen by the Government in 2010. A thorough three-dimensional seismic survey was then undertaken to develop a more detailed image of the underground in the ZIRA, while additional investigations and the experimental work inside the underground laboratory are still ongoing. With the geological disposal project moving toward an industrial phase, a dedicated name was given, Cigéo, for Centre Industriel Géologique (or Industrial Geological Center). The effort in research and development (R&D) was thus moving from a science-dominated approach, intended to generate knowledge, to a more technological vision. Nevertheless, it was also understood that, given the importance and the lifetime of the future repository, strong R&D support would be required.

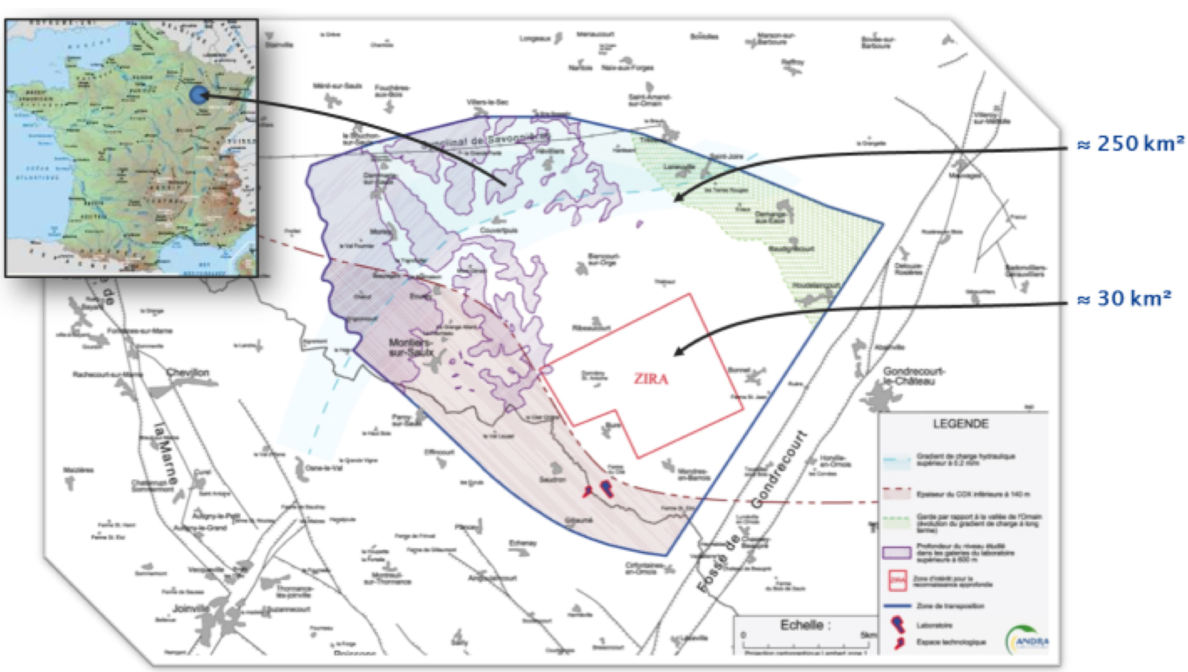

Figure 8–3. Map of the 30 km² zone of interest for the underground disposal

8.1.4 The 2013 Public Debate

Before applying for the license, the project was to be submitted for a National Public Debate. This is a mandatory procedure for any great project in France. A detailed document presenting and describing the Cigéo project was thus prepared and published (The Cigéo Project 2013), so that all stakeholders could express their opinions and suggestions.

In October 2012, toward the end of the industrial design phase of the project, Andra, the Waste Management Organization and project owner of the Cigéo project, referred the matter to France's National Public Debate Commission (CNDP), the independent administrative authority responsible for organizing public debates. By submitting the project as an outline, Andra was able to present the project in accordance with the Aarhus Convention at a sufficiently advanced stage to include its fundamental characteristics, but early enough in the design process to take account of the outcome of the public debate when initial development was to begin in 2014.

The preparation of the debate, its first phase, began in November 2012 with the CNDP's decision to hand the organization of the debate to a special public debate commission (CPDP). Andra was responsible for producing project presentation documentation for the public debate, under the CPDP's guidance, and for setting dates and terms for the public meetings, which would take place during the second phase of the debate.

Once the supporting documentation for the public debate had been approved by the CNDP, and dates and terms of the public meetings phase were set, the second phase of the public debate was scheduled to run from mid-May to mid-October 2013, with a break in August, the traditional summer holiday period in France. It would consist of the distribution of the supporting documentation to a wide audience, particularly via a dedicated website (The Cigéo Public Debate Website—http://www.cigeo.com/en/the-public-debate), 15 public meetings, and relations with the press to relay all the information. Members of the public would also have the opportunity to ask questions or air their opinions via the public debate website.

The main non-governmental organizations opposed to the project quickly expressed their unhappiness with the fact that the debate had been organized in parallel with the national energy transition debate, and were particularly concerned about the usefulness of the debate in the light of a previous experience in 2005/2006, which took place in parallel to the International Review of Dossier 2005. The first two public meetings were prevented from taking place, prompting the CNDP in July 2013 to review the timetable and the terms of the debate. The meetings phase was extended until mid-December, and it was decided that local “mini-meetings” and around 10 open debates between experts live on the Internet would be organized to replace the public meetings. In addition, it was agreed that a citizens’ conference would also be organized.

By the end of the meetings phase in mid-December 2013, public participation in the debate had been highly satisfactory in terms of both quantity (over 1500 questions asked, nearly 500 opinions expressed, 150 “cahiers d'acteur,” and several dozen written contributions submitted) and quality. Although it had not been possible to hold any of the public meetings or mini-meetings, the open debates had allowed some well-argued discussions to take place between experts with often very conflicting opinions, demonstrating that calm debate on the subject was possible.

The citizens’ conference run by the CNDP from December 2013 to February 2014 was the final component of the public debate. Now held up as an example by both the CNDP and specialists, the conference was designed to gather the views of members of the public not involved in the project, including some who live in the two departments affected by the Cigéo facility. The unanimous opinion reached by the panel demonstrated pragmatism and common sense, and showed that discussion of the subject of radioactive waste management need not be restricted to "experts" alone.

In early May 2014, three months after the publication of the report and the conclusions of the debate by the CNDP, Andra presented the changes it would make to the project in response to the debate. These were published in a resolution passed unanimously by Andra's Governing Board (see Follow-Up by Andra of the Cigéo Public Debate 2014), and are based on the conclusions of the debate but also on the recommendations of Andra's various assessment bodies, which examined the project proposal in 2013. Andra made four changes to the project, explained its proposals concerning reversibility, and made several commitments for the future.

8.1.5 Follow-up of the Cigéo Public Debate

To decide on what action to take on the Cigéo project and to present the government with proposals, Andra took the results of the public debate and the opinions expressed during the citizens' conference into account, as well as the various opinions and recommendations it received over the course of 2013 from its assessors (ASN—Nuclear Safety Authority, CNE—National Assessment Board), the French Environmental Authority, and the French High Committee for Transparency and Information on Nuclear Safety. Four new actions were proposed:

1. The integration of a pilot industrial phase when the facility starts up

This pilot industrial phase will make it possible to test, under real conditions, all of the disposal functions: the technical measures taken to control operating risks, the capacity to retrieve waste packages being disposed of, the disposal monitoring sensors, the techniques for sealing cavities and galleries, etc.

Cigéo will move into normal operation after Andra has reviewed the pilot industrial phase.

2. The establishment of a regularly revised master plan for disposal operations

Andra proposes to develop a master plan for the operation of disposal, drawn up in consultation with stakeholders. State-approved and revised regularly, this plan will be a genuine disposal management tool for the entire operational period.

3. Changes to the calendar

Andra has decided to prepare the license application (a Demand Creation Authorization, or DAC) to create Cigéo in two stages:

- In 2015, Andra submitted the master plan for disposal operations to the government, and a set of safety options and technical retrievability options to the Nuclear Safety Authority
- In 2018, Andra will complete the DAC on the basis of this information and final pre-project studies, to obtain the license decree for construction of Cigéo in 2021

Subject to approvals, the construction of the disposal facility could begin in 2020, and the commissioning, beginning with a pilot industrial phase, could take place in 2025.

4. The involvement of civil society in the project

To allow for greater involvement of civil society in decision-making regarding the Cigéo project, Andra has decided to conduct a consultation process to draw up and revise the master plan for operating Cigéo, to contribute to the development of pluralistic expertise on the management of radioactive waste, to explore ways of opening up the Permanent Observatory of the Environment, and to set up a pluralistic committee to provide guidance on the consideration of societal issues in its activities. The Permanent Observatory was built to establish the reference environmental state at the Bure site, and also to keep samples for future investigations, even after several decades, in case of need. The Permanent Observatory will also monitor the environment during the operations of the Cigéo project.

A proposal regarding reversibility was also made. To meet the requirements for reversibility of disposal, the conditions for which are to be laid down by the French Parliament, Andra will use a phased approach that will allow next generations to choose how to pursue disposal operations, including the ability to retrieve waste packages during the expected hundred years of operation, if they so wish.

Finally, Andra is making three commitments for the project going forward: (1) to ensure safe disposal, which remains the absolute top priority, (2) to preserve and develop the local area in close collaboration with local stakeholders, and (3) to control disposal costs, without reducing safety and security.

8.1.6 The Research Program at the Meuse/Haute-Marne Underground Research Laboratory

Since 1999, Andra has been investigating and characterizing the Callovo-Oxfordian clay rocks (the "argillites"), which could host a reversible deep geological disposal of High-Level and Intermediate-Level Long-Lived radioactive Waste (HLW and IL-LLW), now called the Cigéo project. As shown in Figure 8–3, the site is located on the eastern boundary of the Paris Basin, at the border between the Meuse and Haute Marne departments. The site's cross-section (Figure 8–4) shows that the Callovo-Oxfordian clay layer lies at a depth between 420 m and 550 m, and its thickness varies between 130 m and 160 m. Characterization studies are carried out thoroughly in the Meuse/Haute-Marne Underground Research Laboratory (URL), of which an aerial view is displayed in Figure 8–5 and its initial general architecture (2006) in Figure 8–1, following a step-wise process according to the objectives and the Cigéo project's main milestones.

More than 15 years of Research and Development work has been implemented in the URL to support the project development and the reports of the Cigéo project during this period. The URL will continue to play a key role for the next steps of the Cigéo project: the licensing application in 2017, the first industrial pilot phase from 2025, and then the industrial phases according to an operation master plan over about 120 years to achieve the emplacement of the radioactive waste inventory in the Cigéo underground disposal facility.

In the framework of the 1991 Research law, the underground research laboratory construction started in 2000 with shaft-sinking operations (Figure 8–6). Knowledge gained about the overlying layers crossed, especially their hydrogeological characteristics, was essential for designing the shafts and access ramps to the future Cigéo underground facilities.

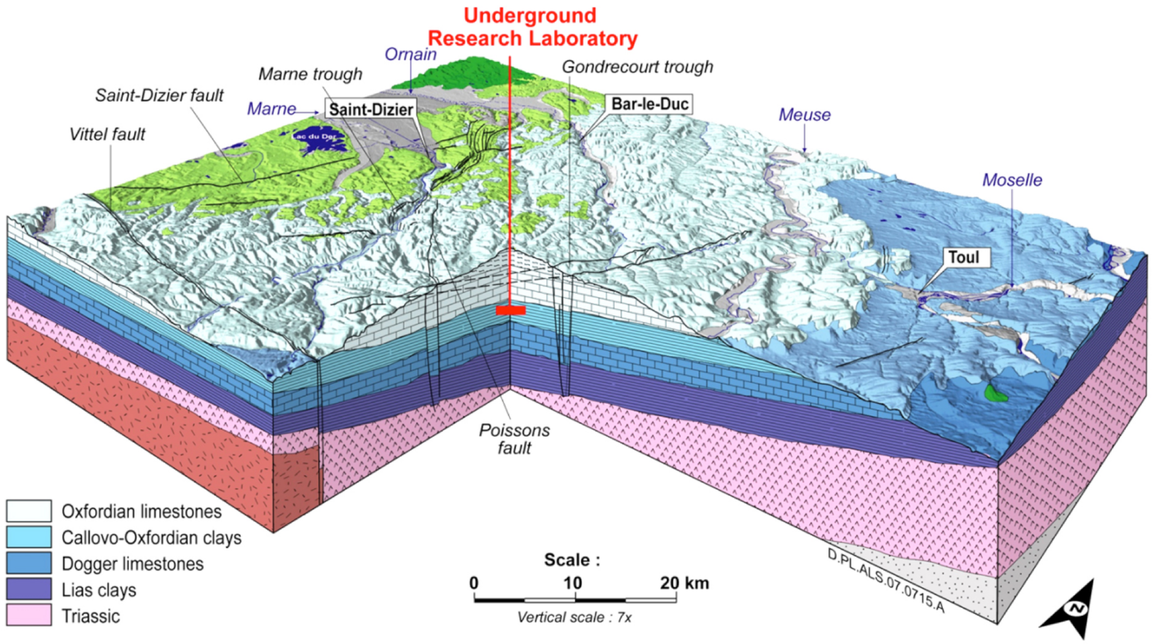

Figure 8–4. Geological structure of the Paris sedimentary basin at the Bure site

Figure 8–5. Aerial view of the Bure underground laboratory

Figure 8–6. Underground laboratory shaft sinking (2000)

When the argillites layer was reached, experiments were first conducted to achieve the main objectives required for the assessment of disposal post-closure safety and construction, in the context of the Dossier 2005, the report on feasibility as requested by the 1991 Research Law. Objectives required for the assessment of disposal post-closure safety and construction are the following:

1. To characterize the confining properties of argillites (water permeability, diffusion coefficients, adsorptive capacity, porosities, pore water chemistry)

2. To demonstrate that the construction and operation of a deep geological repository would not introduce preferential pathways for radionuclide migration (characterization of damaged zone as illustrated in Figure 8–7) around underground structures, evaluating effect of structure dimensions, method of drilling and direction vs. minor and major in situ mechanical stresses (shape, extension, profile of water permeability from surface)

3. To demonstrate the capability to build underground structures (method and stress-strain behavior of liner)

Based on the results presented in Dossier 2005 and the related audits and reviews, the Parliament adopted the 2006 Planning Act, which states that the deep geologic disposal was the reference solution for managing high-level and intermediate-level long-lived radioactive waste, and that it could be built in a formation already investigated by an underground research laboratory, i.e., the Callovo-Oxfordian argillite.

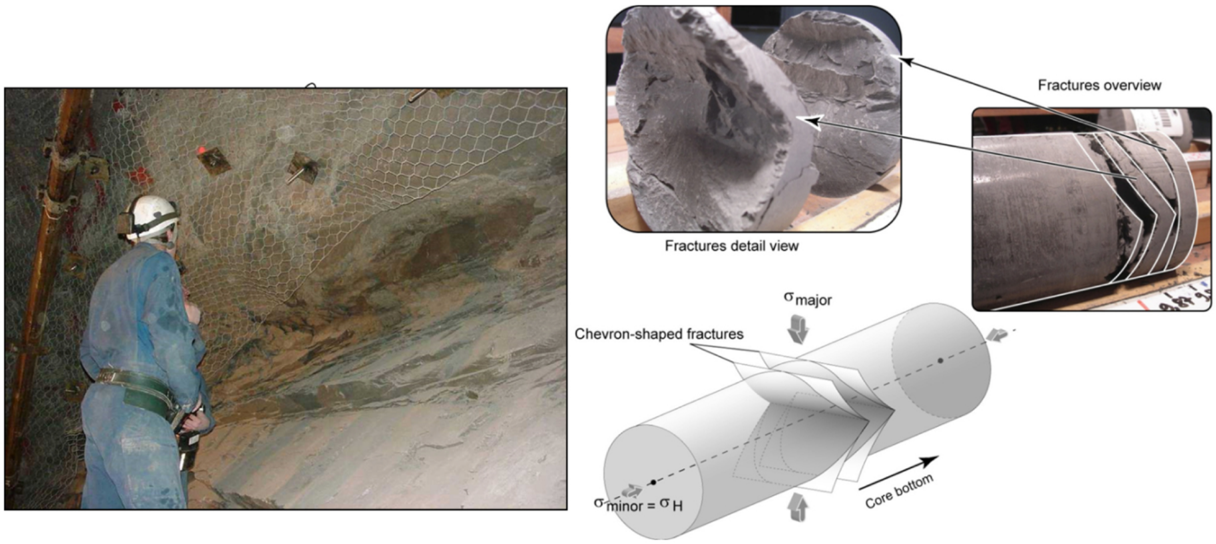

Figure 8–7. Rock mass behavior during and after excavation of the argillite formation at the Bure underground laboratory

The next major step for the Cigéo project was the license application in 2015.[2] Research and development at the URL then focused on three main directions:

1. To consolidate the characterization of the argillites properties, in particular through long-term experiments.
2. To specify and optimize technologies to be used for constructing the underground facilities (drifts, disposal cells, seals). An example of tunneling and lining of galleries is shown in Figure 8–8.
3. To take stock of the THMCR (Thermo-Hydraulic-Mechanical-Chemical-Radiological) behavior of the engineered components and of the Callovo-Oxfordian near field to design the Cigéo components, making sure to fulfill requirements, especially those in connection with reversibility and retrievability of waste packages, and with operational and post-closure safety objectives.

Along with the scientific experiments, a full set of technological ones was developed for the underground research laboratory to answer these three objectives. For example, several excavation technologies (hydraulic hammer, road-header, tunnel boring machine) and various stiff and flexible structural supports (bolts, sliding arches, shotcrete, concrete wedges, steel casing) were tested for the construction of access galleries and specific disposal cells for HLW and IL-LLW disposal packages. All experimental galleries and cells were instrumented to measure the short-term and long-term hydromechanical behavior. These various configurations give insight into the impact of construction methods on the Excavation Damaged Zone extent and its hydromechanical evolution with time (e.g., self-sealing…), and on the progressive loading of the supports, which are key issues for designing the intermediate-level long-lived waste Cigéo disposal drifts.

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[2] In the 2006 Planning Act, the license application for Cigéo was requested by 2015. In France new regulations were published defining new conditions for new construction, especially construction requiring license application files based on detailed design, instead of preliminary design as assumed in 2006. Later Public Debate and discussions with regulators and the Nuclear Safety Authority led to an agreement which allows the license application file to be submitted progressively from 2015 to 2017 to accommodate this change. At the same time, the Nuclear Safety Authority will begin its review as early as possible to attain the Cigéo commissioning target date in 2025.

For characterization of phenomenological processes, main focuses were:

- Strain-stress processes in Excavated Damaged Zone, in relationship to fractures (creeping, swelling…)
- Gas transfer in Argillites (intact and damaged) and at interfaces between Argillites and engineered components
- Thermo-hydromechanical response of Argillites under thermal loading
- Degradation processes of engineered materials (concrete, metal, bentonite, glass) and effect on Argillites
- Transfer of radionuclides in Argillites under disposal situations (salts, thermal loading, organic components)

Particular effort was given to seals, to assure meeting expectations for the licensing of Cigéo. Design of the in situ Seal&Plug experiment is illustrated in Figure 8–9. Construction and hydraulic and hydromechanical performances of the drift seal are tested and studied; two concepts of seals are considered: the first with a hydraulic key within the Excavated Damaged Zone and second without. These two concepts are based on a bentonite clay core between two low pH cementitious walls. A mock-up to demonstrate the construction of a seal without key at large scale was built in surface facility and several experiments on key were conducted in the URL (construction, emplacement of swelling clay bricks and hydromechanical-gas behavior of keys). Finally the Seal&Plug experiment on the hydromechanical performance of a seal was implemented. The experiment is still in progress in the URL.

Figure 8–8. Illustration of a gallery dug with a tunneling machine and lined.

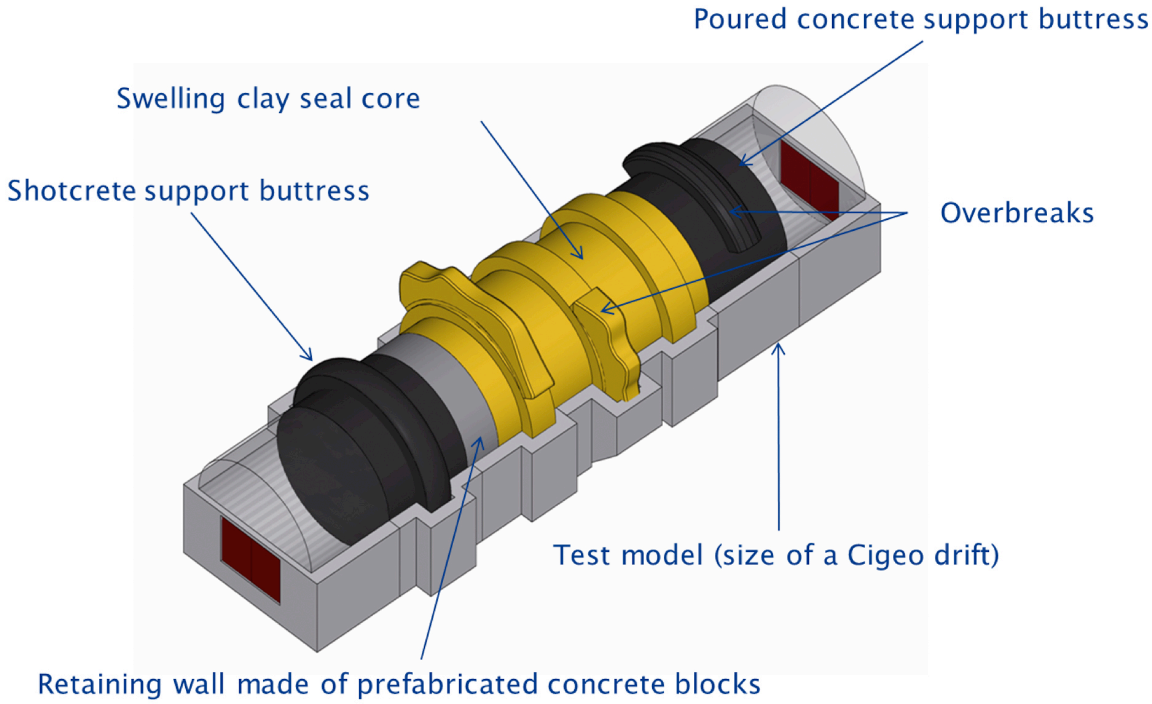

Figure 8–9. In-situ sealing test design at the Bure underground research laboratory.

Following the Public Debate in 2013-2014, an industrial pilot phase was introduced as the first step of the industrial development of Cigéo. The URL program is consequently focused on technological improvements and demonstration of the different disposal systems for the first industrial pilot phase of Cigéo, according to the "Technology Readiness Level" Scale. In the meantime, THMCR characterization studies are still ongoing, in order to improve our understanding of the THMCR processes and the behavior of components.

Currently, the Meuse/Haute-Marne URL has about 1.5 km of drifts, 650 boreholes, and 10,000 sensors. According to the next steps of the Cigéo project, the Meuse/Haute-Marne URL is granted to operate until 2030.

8.2 The Cigéo Project

8.2.1 Waste inventory

In France, the Cigéo project was established to design a deep geological disposal repository for long-lived ILW and HLW generated during the lifetime of existing nuclear facilities, either under operation or licensed. By law, only end waste that cannot be disposed of in a near-surface repository for safety reasons will be accommodated in a deep geological repository. Current French nuclear policy considers complete reprocessing of all spent fuel.

Table 8–1. The French inventory for Cigéo (2014)

Waste type	Total conditioned volume	Conditioned volume produced at the end of 2010	
	[m³]	[m³]	%
ILW	73,600	42,900	27
HLW	10,100	2,700	58
Total	83,700	45,600	54

8.2.2 High-level waste (HLW)

HLW is generated in France as a by-product of reprocessing used nuclear fuel. During recycling, used fuel is dissolved so that the uranium and plutonium are separated from fission products and minor actinides that are non-reusable residues. These highly radioactive residues are incorporated into a molten glass. The mixture is then poured into a stainless steel container. This container and its content constitute the HLW primary packages. They are stored at the two main reprocessing sites, La Hague (operated by Areva) and Marcoule (CEA). Prior to disposal, each HLW primary package is due to be overpacked in a steel disposal container.

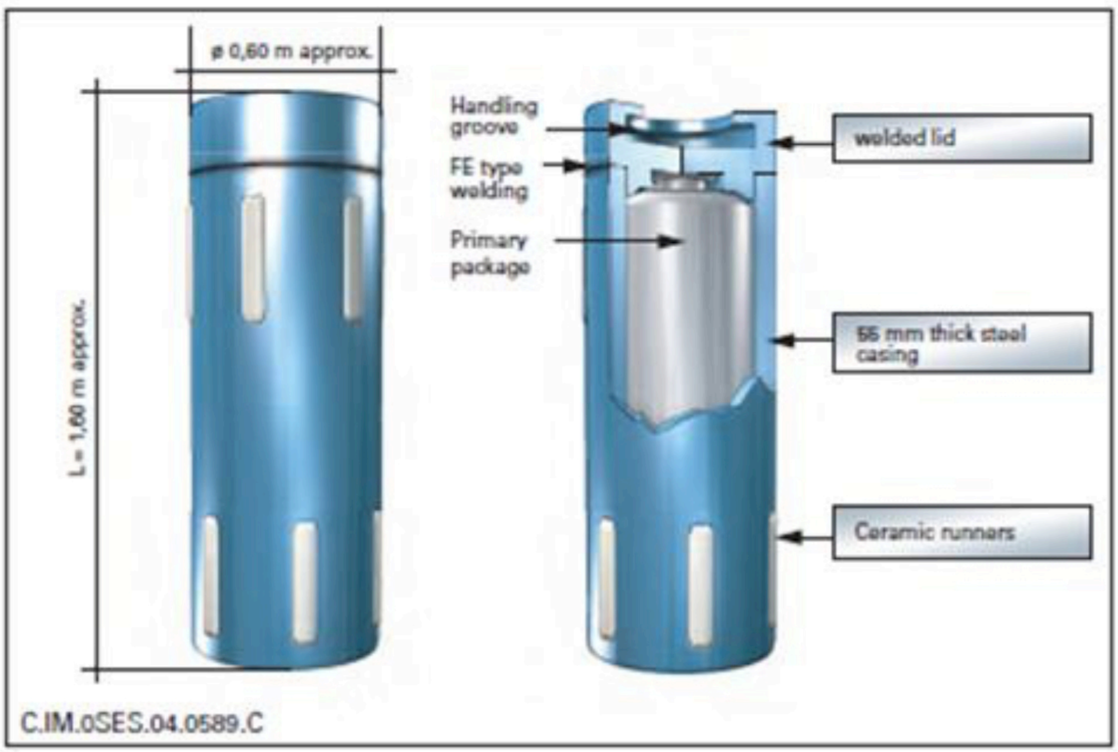

Figure 8–10. HLW waste over-pack principle

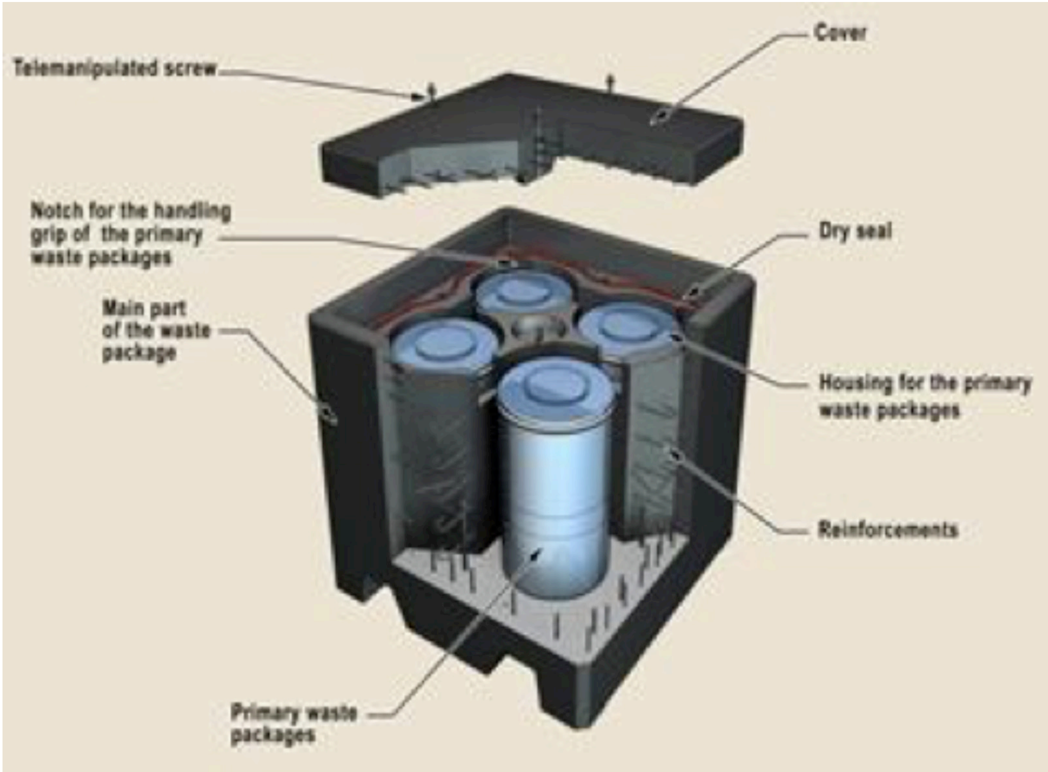

Figure 8–11. Illustration of ILW waste disposal containers.

8.2.3 Intermediate-level waste (ILW)

Long-lived ILW includes the metal structures (cladding, hulls and end caps) resulting from the used fuel reprocessing. This waste also comes from residues originating from the operation, maintenance and dismantling of nuclear facilities (maintenance and dismantling waste, solidified waste issued from effluent treatment, etc.). To reduce its volume, a significant fraction of solid long-lived ILW is compacted to form pucks, which are then packed into concrete or metal containers.

Pending the commissioning of the deep geological repository, long-lived ILW primary waste packages are stored at production or conditioning sites, mainly La Hague (Areva), Marcoule (CEA) and Cadarache (CEA).

To facilitate disposal operations, before emplacement, IL-LLW might be placed into precast concrete robust containers (Figure 8–11), providing complementary protection during the transfer underground.

8.2.4 Compatibility of Cigéo with potential direct disposal of spent fuel

Current French policy is complete reprocessing of spent nuclear fuel. However, the French National Management Plan for Radioactive Materials and Waste stipulates that Andra has to carry out studies on direct geological disposal of spent nuclear fuel, and to verify, as a precautionary measure, that Cigéo remains compatible with the hypothesis of direct disposal of spent nuclear fuel, in case these elements should be considered in the future as waste. Waste packages and disposal concepts were defined, and technological tests were implemented. In 2005, Andra concluded that direct geological disposal of spent nuclear fuel was feasible.

Cigéo is designed to be flexible enough to adapt to potential changes in France's energy policy and the consequences of such changes on the nature and volumes of the subsequent waste generated. Given the volume of existing waste to be disposed of, the impact of a change in France's energy policy would not affect Cigéo's operation until sometime around 2070.

8.2.5 Main Features of the Meuse/Haute-Marne site

The site lies in the eastern section of the Paris Basin. It is a geologically simple domain with a succession of layers of limestone, marl and argillaceous rock deposited in ancient oceans. The argillaceous layer studied by Andra in Meuse/Haute-Marne was deposited around 160 million years ago. The geometry of the geological layers is simple and even. It is homogeneous across a wide surface area. The thickness of the argillaceous layer is more than 130 meters.

This argillaceous rock has properties that enable radionuclides contained in waste to be confined over very long periods:

- Its very low permeability limits the flow of water through the layer, and prevents radionuclides from being carried away by convection. Migration of soluble chemical elements occurs very slowly by means of diffusion.
- The chemical composition of the rock, and of the water contained in it, limits the dissolution of many radionuclides, such as those of the uranium series (actinides), and thus prevents the radionuclides from migrating through the rock.
- In addition to its very low permeability, its confinement properties stem from the argillaceous nature of the rock, which is made up of stacked sheets between which radionuclides can be trapped.

8.2.6 Concept design

The geological repository project includes the underground installation for radioactive waste disposal and various surface supporting facilities. In particular, the surface facilities host the services for receiving, controlling and preparing waste packages prior to disposal emplacement.

As a result of the excellent retention properties of the Callovo-Oxfordian argillite, the disposal concept for HLW is simple. Robust conventional materials such as carbon steel may be used; there is no need for a complex engineered barrier system in addition to the glass matrix, since waste will be efficiently contained first by the over-pack and vitrified matrix, and then by the clay that represents a natural barrier.

The disposal concept for HLW consists of placing waste packages within horizontal cavities measuring 70 cm in diameter. These cavities are lined to ensure the mechanical stability of the structures throughout their operating phase and to ensure retrievability.

The disposal design for long-lived ILW is more compact since they are less exothermic. The disposal design consists of stacking the waste packages in horizontal vaults.

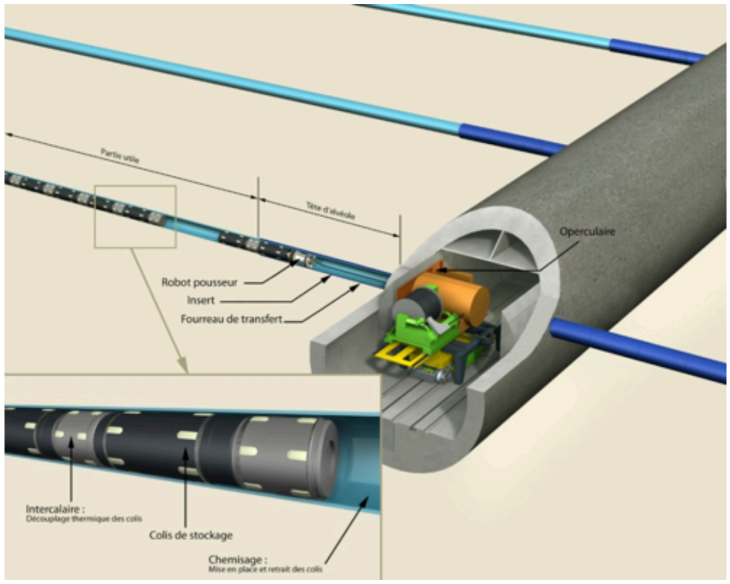

Figure 8–12. Illustration of repository cells design for HLW packages

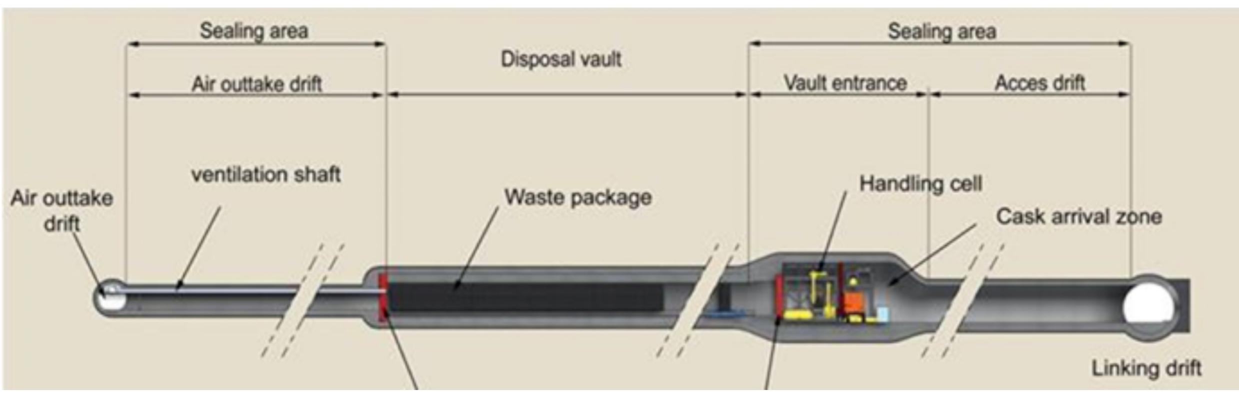

Figure 8–13. Illustration of repository design for ILW packages

8.2.6.1 Performance assessment

Operational safety assessment

Compared to existing nuclear facilities, a geological repository is a specific case due to the existence of underground facilities with, for example:

- Radioprotection, ventilation and containment of radioactivity during transfer of the waste package from the surface to the underground disposal cell, and during its emplacement in the disposal cell
- Operation conditions in case of fire
- Underground co-activity of nuclear operation and construction works

As an input to the industrial design of Cigéo for operation, Andra has defined reference safety guidelines, based on the French regulatory texts for nuclear facilities, basic safety rules generally applied to nuclear surface facilities, and in addition, where relevant, regulatory texts relevant to mines and tunnels. Andra's own experience gained through the previous iterations and corresponding inquiries and recommendations of the Nuclear Safety Authority was also used as input. These guidelines were set up with contributions from experienced nuclear facility operators, experts in industrial risks, and fire safety specialists in tunnel studies. They deal with nuclear safety functions (e.g., containment of radioactive materials, radioprotection, criticality risk control), internal hazards (co-activity, chemical and toxic risks, fall or collision, fire) and external hazards (earthquake, plane fall, flood).

According to the common "defense in depth" principle for nuclear facilities, these guidelines for the industrial design of Cigéo operations consider the successive levels of defense of:

- Preventing the risk and its spreading by design and construction
- Detecting incidents and early management by design and operational methods (detection and control systems, survey procedure)
- Controlling incidents and limiting their consequences by safeguarding systems, specific procedures

Post closure Safety Assessment

The post closure Safety Assessment is based on the body of scientific knowledge acquired. It includes quantitative evaluation of the overall performance in the form of indicators that characterize the possible impact of the repository on man and the environment. It appraises the validity of the safety functions initially established for the repository, and is a test of the whole system's robustness. The safety assessments have shown that the repository's very long-term impact will be orders of magnitude below that of naturally occurring radioactivity.

8.3 Communications and Integration to the Territory

Disposal facilities for radioactive waste are projects of national interest. However, they have to be launched locally, and thus are also, even especially, projects which have to become integrated into a host territory and its social, political and natural environment. With two facilities commissioned these last 20 years and two projects under study, France has wide experience setting up such industrial bodies. Understanding the local socioeconomic and industrial environment is the first essential step, which will then lead to opportunities for effective information dissemination and dialogue.

8.3.1 Information and dialogue

Aware of the need for more information and debate about waste, Andra has launched a process of information and consultation with stakeholders and citizens. It aims both to raise collective awareness of the existence of radioactive waste that requires safe disposal, and to introduce some rationality to the subject so that it would no longer be dealt with in a purely polemical way. Ultimately the purpose was to get society to think about and take responsibility for the issue, at both national and local levels.

Andra's communication, in its widest sense, is an attempt to build greater links between society and the issue. To do this, it uses five pillars: surprise, inform, explain, be visible, dialogue and debate.

Figure 8–14. Examples of advertisements.

8.3.2 Surprise

The purpose of surprising the public is to generate curiosity about the subject. First of all this means accepting the existence of radioactive waste and making this known. Several actions were carried out with this aim, e.g. the design of a slightly provocative advertisement placing a drum of radioactive waste next to a household rubbish bin, with the question: "On average, each person in France generates 2 kg of radioactive waste each year. What should be done with it?" Surprise was also generated with advertisements inviting the public to visit its sites: "Radioactive waste doesn't vanish as if by magic" and "Radioactive waste: challenge received wisdom" using graphics borrowed from the nuclear protest camp. A few examples are displayed in Figure 8–14.

8.3.3 Inform

Informing means communicating in an appropriate way with all sectors of the public, frequently, regularly, transparently, and with no taboos. All types of communication media are used: educational and corporate brochures, periodical booklets, videos, websites, etc.

Figure 8–15. The four editions of Andra's newspaper

To create a greater sense of familiarity, Andra issues a monthly newsletter to all stakeholders in the vicinity of its sites to keep them well informed about progress with its activities. Also, each Andra facility has its own website (www.andra.fr/andra-aube, www.andra.fr/andra-manche, www.andra.fr/andra-meusehautemarne). Finally, a quarterly newspaper is produced, with more than 250,000 copies disseminated to all sectors of the population (local communities, local representatives, political decision-makers, economic players, etc.). As shown in Figure 8–15, four different editions, one for each site (Manche edition, Aube, edition, Meuse/Haute-Marne edition) plus a national one, are issued to explain what is happening at each of the sites, but also to show everything that Andra is doing and allow everyone to have a say, even those who do not share Andra's views.

8.3.4 Explain

To help the stakeholders understand the subject of radioactive waste disposal requires explaining through demonstrations and education outreach. Andra welcomes the public to tour its repository sites, and industrial and research facilities. With around 15,000 visitors per year, including nearly 3,000 to the underground research laboratory at Andra's Meuse/Haute-Marne Centre near Bure, Andra's centers are its primary showcases for what it is accomplishing. There's no better proof than showing someone what you're doing! Explaining also means using educational media for all sectors of the public, from children to experts: website www.dechetsradioactifs.com; an online tour of the underground laboratory at www.andra.fr/visite_virtuelle_laboratoire; events and educational videos on a Dailymotion channel: www.dailymotion.com/andra, e.g., www.dailymotion.com/video/xpuesk_dechets-radioactifs-evitez-les-idees-recues_tech. Finally, explaining means communicating directly with those who will pick up the baton from the current adult generation, by organizing lectures for students and educational workshops for younger children, in liaison with education managers from the French Education Ministry.

8.3.5 Be visible

The aim of being visible is to demystify and "normalize" the subject, whether through exhibitions on different topics organized close to Andra sites, in Paris or anywhere else in France (www.andra.fr/laradioactivite, www.argiles-expo.org) or through Andra's participation in events (annual science fairs, trade fairs) held in the vicinity of its facilities. Around the Meuse/Haute-Marne Center, Andra even deployed an "infobus," designed to bring information straight to the heart of neighboring towns and village, so that local communities did not have to travel to obtain information.

Figure 8–16. Two examples of themed exhibitions: left, "Radioactivity from Homer to Oppenheimer", created and produced by Andra; right, "Clay, a history of the future", produced with Andra's support

These days, being visible also means having a presence on social networks—Facebook, Twitter, YouTube—which have experienced a phenomenal rise in the last few years. These are all places where radioactive waste is discussed, and where Andra should have a presence to answer internet users' questions, and prevent untruths about radioactive waste or its activity from taking root.

Finally, being visible also means taking part in the local life around its sites (talks, sponsorships, development of a local sponsorship policy, participation in local activities such as hunting, etc.), to make Andra a committed, recognized player in the local communities' cultural, economic, educational and social scenes.

8.3.6 Dialogue and debate

Dialogue and debate means creating opportunities for conversations with and involvement of local communities, the general public and experts, and the Local Information and Oversight Committee (CLIS) for the underground research laboratory; to present and weigh up our arguments, including with people who do not share our point of view. With this in mind, Andra offers stakeholders numerous opportunities to meet, responds to requests to send Andra representatives to debates and round table discussions, etc., and participates online in forums, chats, etc. Because the press is the leading liaison for information, Andra developed extensive relations with journalists to ensure they could knowledgeably cover all aspects of the subject of radioactive waste, and especially that they would not give voice only to polemical debate.

A concrete example of stakeholder involvement is the study of the siting of the underground facility of Cigéo. In 2009 Andra proposed to the French Government a 30-km² underground zone (ZIRA, or zone of interest for detailed survey) located in the heart of a 250-km² zone defined in 2005. Technical criteria related to safety and geology (thickness of the layer, depth, etc.) were taken into account, as were criteria related to planning and local integration of the project, such as compatibility with siting of the ramp in the bordering Meuse/Haute-Marne area, the potential siting of access shafts in a wooded area, and avoidance of siting the facility under inhabited areas.

8.3.6.1 Socioeconomic and industrial insertion

Andra in Meuse/Haute-Marne today

Andra's Meuse/Haute-Marne site generates more than 300 direct jobs (Andra employees and service providers) in connection with its various facilities: the Underground Research Laboratory, the Technological Exhibition Facility, the Core Sample Library, the monitoring stations of the Perennial Observatory of the Environment, and the Ecothèque.

Andra also has a strong policy to develop and support relations with local economic stakeholders, and indirect jobs in the region. Since 2010, Andra has supported an annual event called, "Become an Andra subcontractor," which targets local small and medium size enterprises. It encourages businesses to learn about Andra's requirements and procedures, and to prepare for future markets. Énergic ST 52/55, an association that brings together businesses in the energy and public works sectors, is also involved in developing skills in partnership with Andra.

This joint policy is paying off. In 2011, the two regions accounted for 10% of the total amount of invoices concerning the Cigéo project. This amount represents work with more than 250 local public and private businesses. Of these businesses, 30% are located in the Meuse department and 30% in Haute-Marne. These direct and indirect jobs in the region also boost other jobs as a result of spending by employees in the area. According to a survey in 2010, 52% of Andra employees lived less than 20 km from the site, and 97% less than 45 km. In addition, 77% lived locally on a permanent basis, 51% were homeowners and 57% had children in primary school.

Current economic support for the project

Two public interest groups have been launched in the Meuse and Haute-Marne departments to manage facilities and amenities for easing the construction and operation of the underground laboratory and Cigéo, and to carry out planning and economic development initiatives at the department level. They must also support initiatives relating to training and to developing, promoting and disseminating scientific and technological knowledge. Each public interest group received a €30 million grant in 2012.

Waste producers EDF, CEA, and Areva also have policies aimed at promoting local economic development. Examples include building new facilities (EDF nuclear spare parts workshop in Velaines, EDF and Areva archive buildings in Bure and Houdelaincourt, respectively, and the CEA Syndièse Biofuels project in Haute-Marne), helping local firms to develop specialist expertise and boost their business with nuclear operators, and working on initiatives to control energy demand more effectively and reduce CO_2 emissions from buildings.

Cigéo requirements and amenities: a development plan for the local territories

The construction of Cigéo requires preparing its host environment, which is actually a rural area. Not only must the necessary infrastructure be built (means of transport, water and electricity supplies, digital networks, etc.), but a strategic framework must be implemented for employment, economic development, and drawing new residents to the area.

The French Government requested that a development plan for both the Meuse and Haute-Marne departments be prepared. This plan is being prepared under the aegis of a State representative, the Prefect of the Meuse department, acting as coordinating prefect in coordination with local bodies (local authorities, consular chambers, etc.). Andra and nuclear energy companies are also participating in plan preparation.

The draft scheme has already identified the issues in each field, as well as scenarios. It will be rounded out with operations to be carried out and their timetables, both to be established in consultation with regional development bodies. As such, it will boost the economic impact of the construction of Cigéo and its benefits for the area, as well as coordinate the actions of the various parties involved.

Material and infrastructure requirements

Infrastructure must be developed to supply Cigéo with water and electricity. Roads will have to be built for access to the site.

Once Cigéo is commissioned in 2025, water requirements are estimated at around 100 m^3 per day, the average consumption of a population of 700. During the initial Cigéo construction phase (2020–2025), water requirements will rise to around 500 m^3 per day, which is the average consumption of a 3,500-inhabitant city. This amount covers workers' requirements (catering, sanitation facilities, etc.) and those relating to the operation of Cigéo (water for making concrete, reserves for fighting fires, etc.). All site water will be recovered, inspected and dealt with as appropriate. One design objective is to limit liquid discharges. That is why Andra plans to reuse wastewater from the site as much as possible, if necessary after treatment.

Cigéo's electricity requirements are estimated at around 90 MW. The power supply network company is working on connecting Cigéo to a power supply based on Andra's requirements. It will also be the project owner for the electrical substation connecting up to the 400 kV grid.

Building materials (aggregates, cement, etc.) for repository structures will need to be transported to the site. Studies are under way to see whether some of these materials can be transported by rail or water. Possible routes for oversized loads have been studied as part of the interdepartmental development plan. Andra's staff transport plan will focus on promoting carpooling and collective transport. Special arrangements will also need to be made for access to Cigéo's two surface sites (access to the shaft zone, diversion of the secondary road near the ramp zone, etc.).

Jobs

Cigéo is an industrial development project for the community. Construction and operation will span more than 100 years. According to Andra's estimates, between 1,300 and 2,300 people will be involved in building the first Cigéo installations over the period 2019–2025, in addition to the 335 currently involved in activities relating to the underground laboratory. During the operating phase, between 600 and 1,000 people will be employed on the site permanently to operate Cigéo while construction work continues. These preliminary construction and operating estimates will be consolidated in later studies.

In addition to these direct jobs on the site, Cigéo will generate indirect jobs, particularly with suppliers and service providers in the region, as well as jobs produced by spending of Cigéo employees in the area (purchases, housing investment, etc.). Cigéo will boost the development of local businesses. In addition, as Cigéo is guaranteed to remain in operation for over a century, some companies will most likely decide to set up locally, thus generating new business in the area.

8.3.7 Evolution of the Landscape over the Next 10 Years

In conclusion, we can point out that the Cigéo project is progressing according to the request from the political representation, through the 2006 Planning Act. A rural community in eastern France is preparing to host a major industrial project, Cigéo, which will bring to the region not only a high scientific and technological added value, but also significant opportunities for development of the territory. It is the legitimate price that the nation owes to local people for their solidarity, welcoming the radioactive waste produced by nuclear power for the benefit of all the French.

Figure 8–17. Artist conception of the nuclearized surface facility of Cigéo

Figure 8–18. Artist conception of the non-nuclearized surface facility

Following the study and regulatory files instruction phases, the Cigéo project should now take another dimension. The objective remains that of an industrial commission, with an initial pilot phase starting in 2025. The rural environment will by then evolve to integrate the industrial project and the new population that will invest locally. Artists' views of the new sites, the first on a current agricultural sector, the second on a forest area, are presented in Figures 8–17 and 8–18.

The nuclearized facility is designed to receive waste packages by rail, and to control and prepare them for disposal, especially with their disposal overpack. Transfer to the underground disposal zone will be done with an about 5km-long funicular operated through a ramp.

The second surface facility is designed to transfer workers to and from underground, as well as to carry the equipment needed for underground construction downward and the excavated earth upward.

8.4 Acknowledgments

The Cigéo project is the work of all engineers in Andra. The authors would like to thank all of them for their continuous efforts to protect our environment and that of the future generations.

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8.6 Acronyms

Andra—French National Radioactive Waste Management Agency, or Agence nationale pour la gestion des déchets radioactivs

ASN—Nuclear Safety Authority

CEA—Commissariat à l'Énergie Atomique (atomic energy organization)

Cigéo—Centre Industriel Géologique (Industrial Geological Center)

CLIS—Local Information and Oversight Committee

CNDP—France's National Public Debate Commission

CPDP—Special Public Debate Commission, or Commission Particulière du Débat Public

CNE—National Assessment Board

DAC—Demand Creation Authorization

HLW—High-level Waste

IL-LLW—Intermediate-Level Long-Lived Radioactive Waste

OECD/NEA—Organisation for Economic Co-operation and Development / Nuclear Energy Agency

THMCR—Thermo-Hydraulic-Mechanical-Chemical-Radiological

ZIRA—Zone of interest for detailed survey

Chapter 9

Current Status Of Nuclear Waste Disposal In Germany

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ABSTRACT: In 2011 the German government decided to phase out nuclear energy for the industrial generation of electricity. Against this background, Germany resolved to take a new approach for a disposal facility for heat-generating radioactive waste. On the basis of a transparent and scientific process, a location is to be sought which guarantees optimal safety for a period of one million years. After completion of underground exploration and comparison of sites, the selection of a site is expected in 2031. Commissioning is not expected before 2050 at the earliest. The categorization of the radioactive waste in Germany differs from standard international practice by making a subdivision into heat-generating waste and waste with negligible heat generation. Germany currently is implementing three projects for storage and disposal of negligible heat-generating radioactive waste. The Asse nuclear facility, an abandoned salt mine, which was used as a disposal facility from 1971, is now decommissioned pursuant to the German Atomic Energy Act. The Konrad disposal facility, a decommissioned iron ore mine, is currently in the construction phase as a repository for negligible heat-generating waste. The Morsleben disposal facility (ERAM), a salt mine, which was used as a repository for radioactive waste from nuclear power plants in the former German Democratic Republic and the Federal Republic of Germany, is in process to be decommissioned. The Gorleben salt dome has been investigated since 1979 to assess its suitability as a disposal facility for radioactive waste. As part of a political decision, work at the Gorleben site has been reduced to mere maintenance since 2013 to keep essential underground workings open. Due to changes to legal frameworks, more research will be required in Germany, as well as a new search for a disposal facility. International collaboration is also indispensable for disposal facility research. It is also critical to address socio-technical issues, so that the process is transparent to interested and critical members of the general public and all stakeholders, and current scientific understanding of technical and social issues are communicated clearly to all.

9.1 Introduction

After the decision to withdraw from nuclear energy use for electricity generation, Germany resolved to take a new approach to selecting a disposal facility for heat-generating radioactive waste. On the basis of a transparent and scientific process, a location is to be sought which guarantees optimal levels of safety for a period of one million years. The previously investigated site in the Gorleben salt dome will be included in the site selection process in just the same way as every other location under consideration.

The legislation stipulated that a "Commission for the Storage of High-level Radioactive Waste" with a pluralistic membership was to define the basic requirements before the implementation of the actual site selection process. The members of the commission include government representatives from federal and state levels, as well as scientists and stakeholder representatives, who work together on the fundamental decision-making aspects for the subsequent site selection procedure. The recommendations have been presented to the German Bundestag (national parliament) in July2016, and are to be enacted by parliament. The site selection process, which will then commence, is scheduled to be complete by 2031. The commission's work is largely based on the 2002 recommendations of the "Committee on a Site Selection Procedure for Repository Sites" (AkEnd 2002), including a recommendation for a phased procedure based on concrete geoscientific and socio-scientific criteria. The recommendations attracted a great deal of attention among the general public and scientific circles, but were never followed up by any implementation.

In addition to a future disposal facility for heat-generating waste, the Konrad Disposal facility (see Section 9.2) is also being developed for waste with a negligible amount of heat generation. Its commissioning is not expected before 2022.

9.2 Radioactive Waste in Germany

Spent fuel and the radioactive waste from reprocessing in other European countries, as well as all other types of radioactive waste, are scheduled to be disposed of in a long-term (i.e., permanent) disposal facility in Germany. Waste of this kind is generated in Germany (BMUB 2015) from:

- Operations of commercial, experimental, demonstration, and research reactors
- Decommissioning of commercial, experimental, and demonstration reactors, as well as research and teaching reactors, and other nuclear facilities
- Enrichment of uranium and the production of fuel elements from nuclear industry
- Basic research and applied research
- Radioisotope applications in other research institutes, universities, commercial and industrial operations, hospitals, and doctors' surgeries
- The military sector
- In the future, conditioning of spent fuel intended for direct disposal in a disposal facility

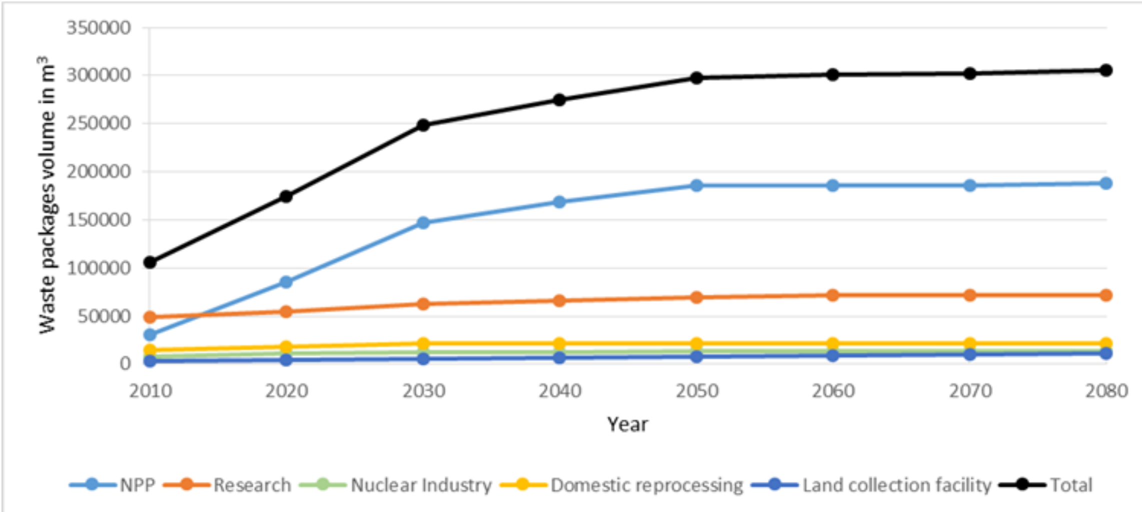

Figure 9–1. Expected volume of accumulated radioactive waste with negligible heat generation, from various waste originators, to be emplaced in the Konrad disposal facility, according to the valid license (plan approval) through 2080 (BMUB 2015).

Categorization in Germany differs from standard international practice by dividing waste into heat-generating waste and waste with negligible heat generation. Heat-generating waste includes spent fuel and radioactive waste from fuel element reprocessing. According to the IAEA classification, this waste can be largely classified as high-level radioactive waste. Apart from a few exceptions, the remaining radioactive waste is waste with negligible heat-generation, and is classed pursuant to the IAEA classification as low-level and medium-level radioactive waste (Figure 9–1).

9.2.1 Heat Generating Radioactive Waste

It is assumed that reactors in nuclear power plants will produce around 10,500 Mg heavy metal (HM) of waste which must be disposed of in a facility in the Federal Republic of Germany. This amount will be temporarily stored in around 1,100 storage casks. Table 9–1 shows the amount and type of waste expected from reprocessing which will have to be dispatched to a disposal facility in the Federal Republic of Germany.

An amount in the 10 to 12 Mg HM range is expected from the experimental, demonstration, and research reactors. In the case of BER-II in Berlin, contracts exist governing the return shipment of spent fuel to the country of origin.

Table 9–1. Predicted amounts (including current inventories) of radioactive waste from reprocessing that will have to be disposed of in a facility in the Federal Republic of Germany (as of December 31, 2014) (BMUB 2015).

Types of nuclear wastes	Ingot molds (*Kokillen*)	Containers
Vitrified high-level radioactive waste from France (CSD-V)	3,024	108
Vitrified intermediate-level radioactive waste from France (CSD-B)	140	5
Intermediate-level radioactive waste from France compacted under high pressure (CSD-C)	4,104	152
Vitrified high-level radioactive waste from the United Kingdom (UK-HAW)	571	21
Vitrified high-level radioactive waste from reprocessing at Karlsruhe (HAW-WAK)	140	5
Total	**7,979**	**291**

9.2.2 Radioactive Waste with Negligible Heat Generation

Around 47,000 m³ of low-level and medium-level radioactive waste was stored in the Asse II mine. Recovery of this waste is expected to generate at least 90,000 Mg of unconditioned waste. After conditioning, this will generate around 175,000 to 220,000 m³ of waste for subsequent disposal in a disposal facility. Assuming that no further utilization takes place, uranium enrichment is estimated to produce up to 100,000 m³ of containerized waste filled with depleted uranium.

9.3 Political Background and Institutional Framework

9.3.1 Political Background

The use of nuclear fission for the commercial generation of electricity in the Federal Republic of Germany will end in 2022 at the latest. The transfer of spent fuel from nuclear power plants to the processing plants has been illegal since 1 July 2005. According to the Atomic Energy Act (AtG), the federal government is obliged to establish facilities for the secure storage and final disposal of radioactive waste in a disposal facility. The German government plans to emplace all types of radioactive waste in deep underground geological formations. In line with this strategy, the Konrad Mine in Salzgitter is currently being converted into a disposal facility for radioactive waste with negligible heat generation. A site for a disposal facility for heat-generating radioactive waste will be defined by a selection procedure.

In 2015 the German Ministry of the Environment, Nature Conservation, Building and Nuclear Safety (BMUB) presented, for the first time, a disposal program encompassing all types of waste (Program for the responsible and safe management of spent fuel and radioactive waste—National program). The National Disposal Program's fundamental aspects are subject to the provision of possible changes resulting from recommendations by the Commission for the Storage of High-level Radioactive Waste discussed in Section 1 (BMUB 2015), and are characterized by the following key elements:

- Radioactive waste is explicitly disposed of as an act of national responsibility. Disposal in a disposal facility is to take place within the country. Irradiated fuel elements from facilities used for the fission of nuclear fuels, but not for the commercial generation of electricity, can be transported in accordance with the statutory regulations to the country that provided or produced the fuel elements for research reactors.

- Repositories are to be established at two locations: the Konrad Disposal facility for radioactive waste with negligible heat generation, and a disposal facility chosen in accordance with the Site Selection Act specifically for heat-generating radioactive waste.
- The radioactive waste in the Asse II Mine is to be recovered and to be taken into consideration in the search for a disposal facility site in accordance with the Site Selection Act. Depleted uranium from uranium enrichment already generated or to be generated, should be taken into consideration on a precautionary basis when searching for a site for a disposal facility pursuant to the Site Selection Act in the event that such waste is not used for other purposes.
- A final decision on the disposal facility site for all of this waste, taking into consideration all of the technical, economic and political aspects, can only be made when the criteria for the storage of waste in a disposal facility have been defined pursuant to the Site Selection Act, and when sufficient information is available on the volume and properties of the radioactive waste to be recovered from the Asse II Mine, and the time when this waste becomes available for disposal.
- The timing for the dismantling of all reactors in commercial nuclear power plants, as well as other nuclear facilities and installations which are to be decommissioned during the period under consideration, should be scheduled to coincide with the availability of a disposal facility so that the radioactive waste with negligible heat development produced during the dismantling process can be disposed of in the Konrad Disposal facility.
- The Konrad Disposal facility is scheduled to be commissioned in 2022. The emplacement operations for the 303,000 m^3 of waste defined in the planning process should not exceed 40 years.
- The site for a disposal facility for heat-generating waste in particular should be defined by 2031 at the latest pursuant to the Site Selection Act. The disposal facility is to be commissioned around 2050.
- The first partial authorization for the disposal facility for heat-generating waste in particular, is then to be followed by the authorization of an incoming storage at the site for all irradiated fuel elements and reprocessing waste, with the aim of thus establishing the prerequisite conditions that allow work to start on emptying the current interim storages.
- The irradiated fuel elements and the reprocessing waste are intended to be stored at the existing interim storage sites until this can take place.
- The emplacement of low-level and medium-level radioactive waste in the Morsleben Disposal facility for radioactive waste has finished. The disposal facility is to be decommissioned and safely sealed for the long term. The polluter-pays principle, in the sense of the obligation to act, applies to the disposal of radioactive waste until it is handed over to a disposal facility or a state collection center. This means that the parties handling radioactive materials bear the responsibility for ensuring that any radioactive residues that are generated, or any dismantled or disassembled radioactive components from the facilities, are recovered safely or disposed of as radioactive waste in accordance with the regulations. Radioactive waste from industry, medicine, and research must first be delivered to a collection center run by the state (state collection centers), where it is to be temporarily stored. The state collection centers hand over the radioactive waste temporarily stored at the centers to a disposal facility.

9.3.2 Institutional Framework

The adoption of the Site Selection Act created the legal framework for setting up a new regulatory authority. The Federal Office for the Regulation of Nuclear Waste Management (BfE) opened for business on 1 September 2014. It regulates the site selection procedure and supports BMUB on aspects involving the disposal of radioactive waste in a disposal facility. The BfE will not take up its regulatory tasks until the Commission has finalized its work and the decision-making criteria have been laid down in law.

BfE is currently responsible for ensuring that the site selection procedure is refinanced. This involves BfE implementing an apportionment procedure to issue bills of costs and advance payment notices to the parties responsible for generating the waste. BfE, as a regulatory authority, will be responsible in the future for these statutorily planned duties as part of the site selection process:

- Exercising state supervisory responsibilities during the implementation of the site selection procedure.
- Examining the proposals of the project executors involving surface exploration.
- Defining site-specific surface and underground geological exploration programmes and testing criteria.
- Examining the site selection of the project executor for the underground exploration.
- Issuing an official notice on whether the site selection procedure was undertaken pursuant to the specifications and criteria of the Site Selection Act, and that the selection proposal of the project executor for the underground exploration satisfies these specifications and criteria.
- Implementing strategic environmental audits.
- Final recommendation for a site.
- Informing the general public as part of the site selection procedure.

BfE undertakes the following after the commissioning of the Konrad Disposal facility, after the decommissioning planning permission decision for the Morsleben Disposal facility, and after the statutory definition of the disposal facility site pursuant to the Site Selection Act:

- Planning permission and authorization pursuant to the Atomic Energy Act (AtG) and their rescission (i.e., planning permission and authorization).
- Issuing authorisations in accordance with mining law, and other permits and approvals required by mining law involving authorisation procedures pursuant to the AtG for the construction, operation, and decommissioning of federal German facilities for safeguarding and final disposal in a disposal facility pursuant to the AtG, and in consultation with the responsible mining authority.
- Mining supervision pursuant to the Federal Mining Act involving federal German facilities for safeguarding and final disposal in a disposal facility pursuant to the AtG.
- Issuing permits or licences in accordance with water law in consultation with the responsible water authority.

The Federal Office for Radiation Protection (BfS) is currently responsible as the operator of the Asse Mine (Section 12.4.1), the Konrad Mine project (Section 12.4.2), the Morsleben Disposal facility for radioactive waste (Section 12.4.3), and is the project executor pursuant to the Site Selection Act. With the resolution adopted on March 2nd, 2015, the Commission for the Storage of High-level Radioactive Waste proposes the establishment of a new operative organization. The BMUB is currently engaged with its implementation.

9.4 National Disposal Facility Projects

9.4.1 The Asse Mine

Potash salt and rock salt were mined from 1909 to 1925, and 1916 to 1964, respectively, in the Asse II Mine located 10 km southeast of the city of Wolfenbüttel. The production of salt ended March 1964. The decommissioned mine was purchased by the federal government in 1965, and, after an initial test phase, used for storage of radioactive waste. Starting in 1971, Asse II was used as a disposal facility to store a large proportion of low-level and medium-level radioactive waste from the Federal Republic of Germany. A total of 125,787 barrels and containers with low-level and medium-level radioactive waste were emplaced in the mine up until 1978 (BfS 2015).

The mine lies in the Asse ridge. The rock salt and potash salt layers of the Zechstein Formation had been tectonically stressed after they were deposited. This gave rise to a complicated internal structure, which also significantly influenced the geometry of the underground workings. Numerous extraction chambers were created in the Asse II Mine during the salt mining phase. These underground workings are tightly stacked, one above the other, in the southwest flank of the mine (Figure 9–2). In some parts, the salt was mined directly up to the surrounding rock. In some places, this means that the distances between the chambers are no more than a few meters. Because of the deconsolidation of the rock salt and the surrounding rock, water saturated with salt is currently flowing into the mine in the upper part of the southern flank at a depth of around 500 to 575 m (BfS 2015).

In addition to the mining regulations, the operations for the ongoing open-mine maintenance and the decommissioning of the Asse II Mine are also subject to the Atomic Energy Act stipulations applying to federal German facilities for the safeguarding and disposal of radioactive waste in a disposal facility. The facility is to be decommissioned as quickly as possible, pursuant to the German Atomic Energy Act. The responsible operator is the Federal Office for Radiation Protection (BfS) (NMU 2015).

The decommissioning options include flooding the mine with magnesium chloride solution, backfilling with solid material (ballast, Sorel cement), relocating parts of the radioactive waste within the mine, or recovering all of the waste drums. According to the latest analysis, recovering the waste from the Asse II Mine is the best option for handling the radioactive waste stored there. During a test phase for data collection, boreholes were initially drilled into two emplacement chambers, followed by the opening of the chambers and the test recovery of some of the waste stored in these chambers (BfS 2015).

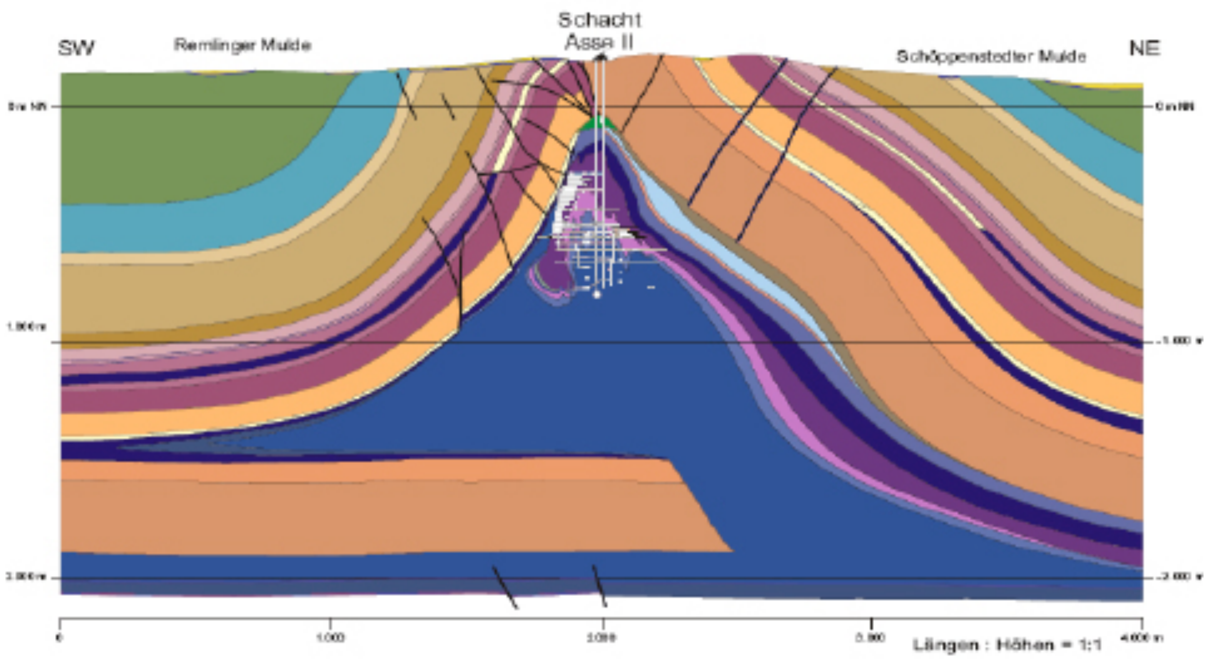

Figure 9–2. Asse II Mine underground workings in the Asse salt structure (blue: Hauptsalz of the Staßfurt Series).

Between 1993 and the end of 2013, 735 million Euros were spent on Asse II. The total cost for decommissioning the Asse II Mine cannot be estimated on the basis of the information available to date. The beginning of retrieval is estimated for 2033 (BMUB 2015).

9.4.2 The Konrad Disposal Facility

The Konrad Mine planned for a disposal facility is a decommissioned iron ore mine located close to the city of Braunschweig in the state of Niedersachsen. Iron ore has been mined in this region since 1867. Production at the Konrad Mine was terminated in 1976 because it was no longer profitable.

Because of its favourable geological conditions, the Konrad mine is unusually dry compared with other metalliferous mines. This is why the iron ore formation lying at depths between 800 m and 1300 m was chosen for the disposal of radioactive waste (Figure 9–3).

In 1982, an application for initiation of a planning permission procedure for the Konrad Disposal facility Project was submitted by PTB (Federal Institute for Physics and Metrology), the legal predecessor of BfS. The licensing documents, the so-called Konrad Plan, were submitted in 1990. According to the "agreement" of 11 June 2001 between the Federal Government and the power utilities, the planning permission procedure for the Konrad Disposal facility was completed, and a resolution was adopted on 5 June 2002. The planning permission decision for the Konrad Disposal facility permits the disposal of up to 303,000 m^3 containerized radioactive waste with negligible heat generation in the Konrad Mine (Wallner et al. 2005). The decision to construct and use the Konrad disposal facility was finally confirmed by the uppermost court in 2007, after considering all legal objections to the planning permission decision. According to the progress made so far, completion of the conversion work is projected for 2022 (BfS 2015).

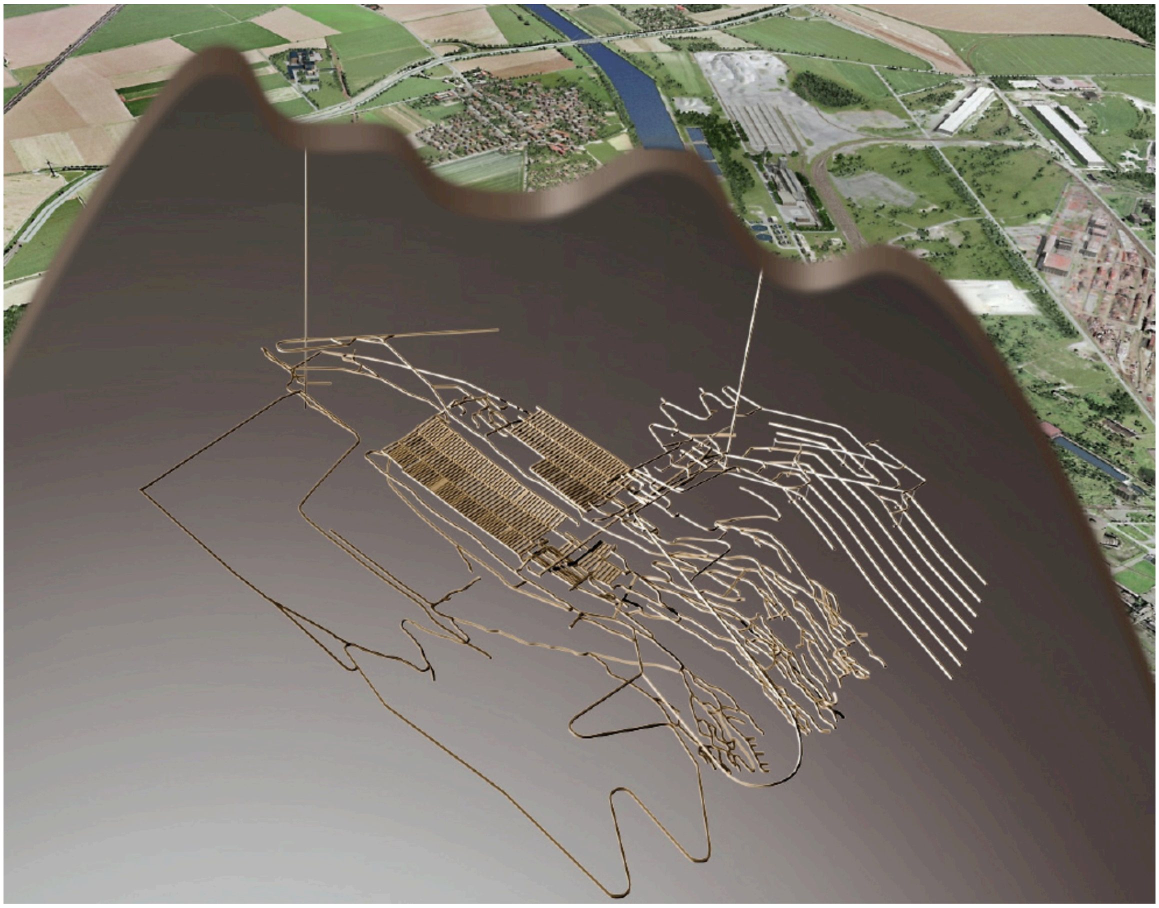

Figure 9–3. Konrad Mine underground workings (BfS 2015).

The total costs calculated for the construction of the Konrad Disposal facility are currently considered to be approximately 2.9 billion Euros. All in all, total cost is calculated at 7.5 billion Euros for all phases until decommissioning of the mine. Approximately one-third is to be financed by the public purse, and approximately two-thirds by the private generators of the waste, in proportion to the volumes of waste produced by each generator (BMUB 2015).

9.4.3 The Morsleben Disposal facility (ERAM)

The mine near Morsleben was used for the extraction of potash salt and rock salt before it was converted to the Morsleben Disposal facility for radioactive waste in 1971. Initially, operating waste from the nuclear power plants in the German Democratic Republic (GDR) were stored here until 1998, and subsequently also operating waste from nuclear power plants in the Federal Republic of Germany. The total volume of low-level and medium-level radioactive waste is 36,754 m^3 (BfS 2014).

The Morsleben Disposal facility lies in a salt structure formed by the Zechstein formation. Unlike salt diapirs with simple structures, the Morsleben structure is significantly modified by tectonic processes. The intensely folded rock salt and potassium salt layers are studded with anhydrite blocks. The thickness of the formation varies between 350 m and 550 m (Figure 9–4).

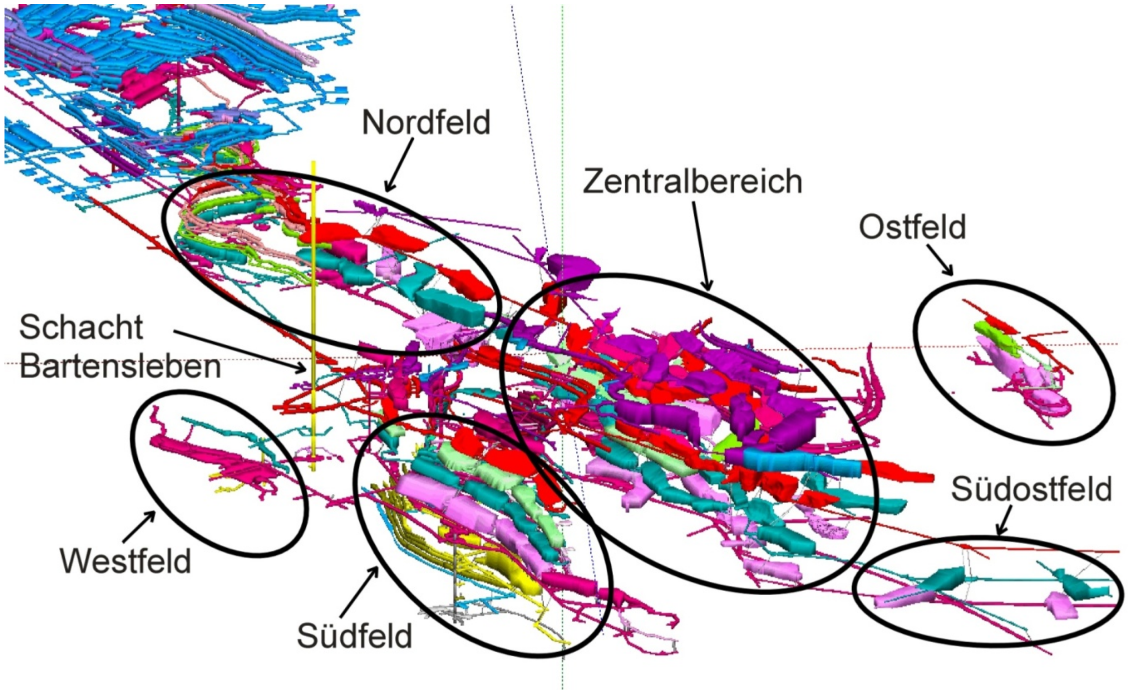

Figure 9–4. Morsleben Disposal facility underground workings BfS (2015).

After the authorization for storing radioactive waste ran out in 2002, following an amendment to the German Atomic Energy Act, a planning permission process was initiated to decommission the disposal facility. The resulting planning permission decision is currently being implemented.

The costs for the Morsleben Disposal facility since German reunification in 1990 and the end of 2013 were approximately 1.1 billion Euros. The total project costs are estimated at around 2.4 billion Euros (BMUB 2015).

9.4.4 The Former Gorleben Disposal Facility Project

The Gorleben salt dome in the western part of the state of Niedersachsen has been investigated to assess its suitability as a disposal facility for radioactive waste since 1979. The investigation program comprised the surface and underground geological, geotechnical, and mining exploration work, as well as the processing and assessment of all questions required to evaluate the suitability and long-term safety of the site. For three decades comprehensive investigations to study the internal structure of the salt dome, as well as the cover rock and surrounding rock, were carried out at the Gorleben site. Interim findings were presented in comprehensive reports in 1983, 1990 and 1995. An overall description of all exploration findings was presented in a four-volume site description report (Klinge et al. 2007, Köthe et al. 2007, Bornemann et al. 2008, Bräuer et al. 2012) (Figure 9–5).

Work on the Gorleben site was suspended for the first time as a result of an agreement reached between the German government and the power utilities on 14 June 2000. This moratorium was valid for a period of at least three years and a maximum of ten years. After restarting work for a short time in 2010, exploration was terminated again in 2012 as a result of political negotiations on a new search for a disposal facility site in Germany. The exploration findings were assessed as part of a "Preliminary safety analysis Gorleben (VSG)" (GRS 2013). As part of a political decision, work at the Gorleben site has been reduced to mere maintenance since 2013 to keep the essential underground workings open.

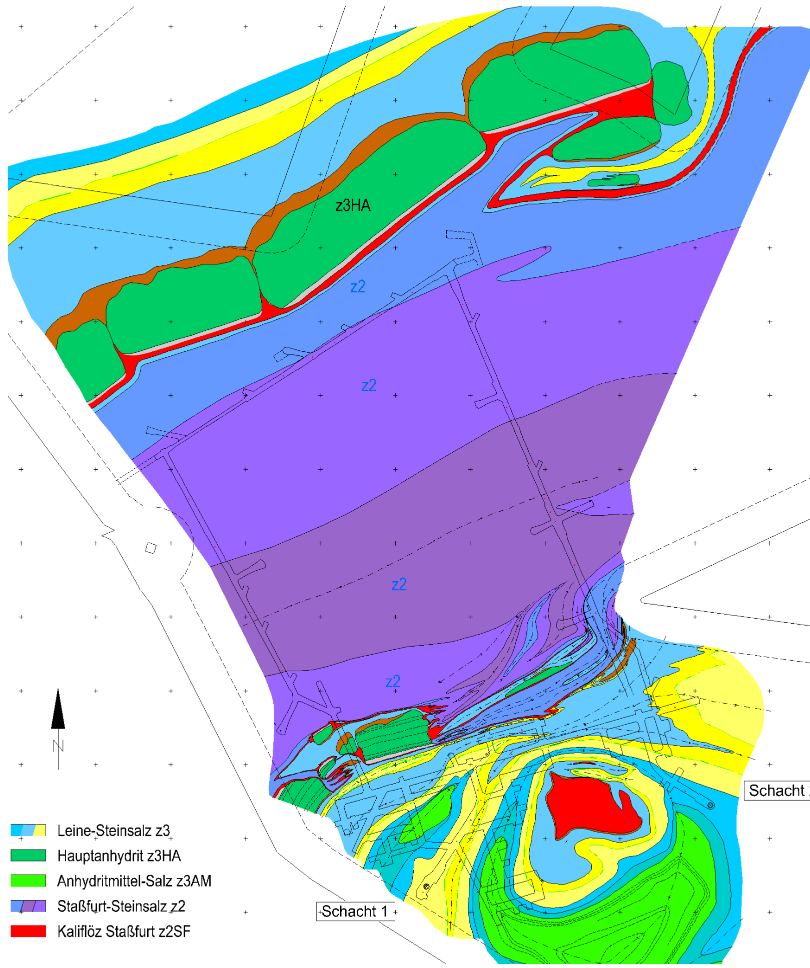

Figure 9–5. Horizontal cross section at the 840 m-level. Exploration area 1 of the Gorleben Salt dome.

Costs totalling around 1.74 billion Euros have been expended on the Gorleben project, from 1977 to the end of 2013. Further costs for Gorleben depend on a number of factors, including how long the essential underground workings are to be kept open. Therefore they cannot be calculated precisely at this time (BMUB 2015).

9.5 German R&D Activities

According to the Atomic Energy Act, the German government is responsible for the provision of nuclear repositories for radioactive waste in Germany (Section 12–3). Specific research programs designed for this purpose are therefore necessary to take into consideration and incorporate the required state-of-the-art science and technology. The programmatic basis for this is the 6th Energy Research Programme of the federal government, which had been established under the management of the former Federal Ministry for Economic Affairs and Technology (BMWi 2011).

The research funding for the departments is provided in Germany by three ministries: Federal Ministry for the Environment, Nature Conservation, Building and Nuclear Safety (BMUB), the Federal Ministry for

Economic Affairs and Energy (BMWi), and the Federal Ministry for Education and Research (BMBF). This involves BMUB controlling the facility and site-related research activities; BMWi co-ordinating the application-oriented and site-independent basic research activities; and BMBF funding the basic research.

Due to changes to legal frameworks, more research will be required in Germany, as well as a new search for a disposal facility. According to its funding concept, "Research for the disposal of radioactive waste (2015–2018)," published by BMWi in 2015, this ministry is responsible for this revised approach. Associated research activities are focused on the following main objectives (PTKA, 2015):

- Creating the scientific and technical basis for the construction of a disposal facility for heat-generating waste.
- Developing the methods and technologies required for specific measures to prepare the disposal facility, as well as for the conception, construction, operation, and commissioning of a disposal facility, associated with the continuous further development of the state-of-the-art of science and technology.
- Making a substantial contribution to the development, further advancement, and maintenance of scientific-technical competence, and promoting the influx of young staff, for the purposes of nuclear disposal in Germany.

The following thematic changes have arisen compared to the previous funding concept (PTKA 2015):

- More intense research activity, covering a range of potential host rocks, in particular the analysis of disposal facility systems in all relevant host rocks (rock salt, claystone, crystalline rocks).
- The analysis of longer interim storage periods, in particular with respect to the safety of waste and containers.
- Scientific investigations on alternative disposal methods instead of direct disposal in a disposal facility in a mine.
- More intense incorporation of socio-technical issues.

There is global consensus in the scientific-technical community that heat-generating high-level radioactive waste should be disposed of in nuclear repositories in deep underground geological formations. In all of the concepts, the isolation of the radioactive waste is to be achieved by suitable barriers (natural and engineered). The natural geological barrier is given a particularly high priority in the concept pursued in Germany (AkEnd 2002). Potential host rocks, consisting of rock salt formations, claystones, and crystalline rocks, are present in adequate quantities in Germany.

Because of the previous focus on investigating rock salt formations, there are differences in the degree of understanding of the potential host rocks. The realigned research on nuclear repositories in Germany therefore pursues an approach to reduce these deficits. According to PTKA (2015), this means:

- In the case of rock salt formations: working on the identified R&D needs for a disposal facility in salt domes, and clarifying conceptual questions on the location of a disposal facility in horizontally bedded salt formations.
- In the case of claystones: expanding the R&D projects, and completing the instruments available for system analysis.
- In the case of crystalline rocks: clarifying basic scientific issues.

In addition to research at a national level, international cooperation activities are also indispensable for disposal facility research. They not only help expand the state-of-the-art of science and technology, but also play an important part in improving acceptance by the general public thanks to the broad international validation of the research findings. Cooperation between German research organizations and international partners is based on the following components:

- Treaties with state institutions,
- Bilateral treaties with disposal facility organizations,
- Involvement in international committees, and
- Involvement in international rock laboratories

The most important component at the scientific-technical level is the collaboration in international rock laboratories, in which Germany participates, in particular because of a shortage of in-situ investigation possibilities within its own borders. German research organizations have participated in international rock laboratories for many years (Table 9–2).

Table 9–2. Rock laboratories in which German institutes are involved together with international partners (after PTKA)

Rock laboratories	Geological formations	Countries	German participation since	German institutions*	International partners*
Grimsel	Granite	Switzerland	1983	BGR, FZK, GRS	CH, E, JP, USA
Aspö	Granite	Sweden	1995	BGR, FZK, FZR, GRS, TU-C	S, E, F, FIN, JP, USA
Mt. Terri	Claystone	Switzerland	1996	BGR, GRS	CH, B, E, F
Mol	Clay	Belgium	1997	BGR, GRS	B, F
Tournemire	Clay	France	1997	BGR, GRS, Uni-HD	F
Bure	Claystone	France	2000	BGR, GRS	F

* BGR (Federal Institute for Geosciences and Natural Resources); FZK (Research Center Karlsruhe); FZR (Research Center Rossendorf); GRS (Gesellschaft für Anlagen- und Reaktorsicherheit); TU-C (Technical University Clausthal); Uni-HD (University of Heidelberg); CH (Switzerland); E (Spain); JP (Japan); S (Sweden); F (Finland); B (Belgium); F (France)

In addition to geoscientific and technical research activities, it is also critical to address socio-technical issues, so that the process is transparent to interested and critical members of the general public and all stakeholders, and current scientific understanding of technical and social issues are communicated clearly to all. Research activities are undertaken in Germany with the aim of creating a comprehensive social dialogue, and encouraging participation of the general public in the search for a disposal facility site. The ENTRIA research platform established in 2013 addresses this aspect, in particular, and bundles research projects that unite technical and socio-scientific approaches (ENTRIA 2014).

The nuclear safety and disposal facility research projects funded as part of the 6th Energy Research Programme cost 77 million Euros in 2014. A growing proportion is expected to involve disposal facility research as a result of the research required during the implementation of the Site Selection Act.

9.6 Conclusions and Outlook

The Site Selection Act legislating the search and selection of a site for a disposal facility for heat-generating radioactive waste came into force in Germany in July 2013. This act defines new regulations for a selection procedure for a site for a disposal facility. As part of this procedure, all of the potentially suitable host rock formations (rock salt, claystone, and crystalline rock) for a disposal facility for high-level radioactive waste in Germany are to be considered.

In the run-up to the implementation of the Site Selection Act, the "Storage of High-level Radioactive Waste" Commission began its work in May 2014. The main focus of its work was developing recommendations for action on other disposal options, general safety specifications, exclusion and evaluation criteria, and criteria for correcting potential errors during the ongoing process, as well as recommendations concerning organization and public relations work. After the commission completed its work in July 2016, the federal government's time schedule stipulates that after enacting the decision-making principles into federal law, site selection should begin. Selecting sites for underground exploration is to follow the end of surface exploration in 2023. The decision on a site is expected in 2031, after completion of the underground exploration and a comparison of the sites. The subsequent approval process is expected to take several more years. Commissioning is not expected before 2050 at the earliest.

According to the coalition agreement reached between the governing parties, disposal facility research in Germany is to be intensified in parallel with the implementation of the Site Selection Act. The investigations involved must take into consideration all of the feasible host rocks in Germany, and all of the different disposal facility concepts. International cooperation in rock laboratories, in particular, is indispensable as part of this strategy. The geoscientific and technical issues involved in disposal facility research in Germany will be investigated mainly in international rock laboratories.

In addition, an equal amount of effort is to be put into research projects looking at issues concerning the participation of the general public and the transparency of the site selection procedure. This will also incorporate experience gained from site selection procedures in other countries. The ambitious goal of selecting a site for a disposal facility in Germany by 2031 is to be achieved by integrating the findings from the site investigation work and the results of research activities.

9.7 Acknowledgements

Sincere thanks are given to Dr. Ingo Böttcher from the German Ministry of the Environment, Nature Conservation, Building and Nuclear Safety (BMUB) who kindly provided the current information about the new German National Programme for the responsible and safe management of spent fuel and radioactive waste.

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9.9 Acronyms

AkEnd—Committee on a Site Selection Procedure for Repository Sites

AtG—Atomic Energy Act

BfE—Federal Office for the Regulation of Nuclear Waste Management

BfS—Federal Office for Radiation Protection

BMBF—Federal Ministry for Education and Research

BMUB—Federal Ministry for the Environment, Nature Conservation, Building and Nuclear Safety

BMWi—Federal Ministry for Economic Affairs and Energy

GDR—German Democratic Republic

HM—heavy metal

PTB—Federal Institute for Physics and Metrology

PTKA—Project Management Agency Karlsruhe

VSG—Preliminary safety analysis Gorleben

Chapter 10

Current Status of Geological Disposal Projects in Hungary

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ABSTRACT: Lessons learned and efforts made in geological disposal of radioactive waste in Hungary are summarized. After a brief introduction of the national context, we present our experience in siting and construction of the country's first geological low and intermediate level waste repository, the National Radioactive Waste Repository in Bátaapáti, which has been operating since 2012. We also discuss the more challenging geological research program for the siting of a future deep geological repository for high level waste.

10.1 Introduction

Generation of radioactive waste in Hungary started simultaneously with the introduction of the use of isotope technology in the early 1960s. Later, the commissioning of the four VVER-440 units of Paks Nuclear Power Plant (NPP) (1982–1987) increased the generation of radioactive waste significantly. Nowadays, the operation of Paks NPP generates every year approximately 330 spent fuel assemblies (120 kgHM/assembly), 5 m^3 of high level waste (HLW), and about 450 m3 of low and intermediate level waste (LILW). Small scale waste generators other than the NPP (e.g., research institutes, hospitals) add only a few cubic meters (5-15 m^3 of LILW) each year. A more comprehensive and detailed inventory of radioactive waste and spent fuel in Hungary can be found in the 14th Medium and Long Term Plan of the Public Limited Company for Radioactive Waste Management (KHTT 2015).

Being a European Union member state, Hungary is subject to the measures of the Council Directive 2011/70/EURATOM, which establishes a community framework for the responsible and safe management of spent fuel and radioactive waste (hereinafter: Directive). This was transposed into the national legislation by the last major amendment of the fundamental law in the field, the Act CXVI of 1996 on atomic energy (hereinafter: the Act), and its relevant executive orders in 2013.

In line with the Directive, the Act declares that Hungary is ultimately responsible for the management of the spent fuel and radioactive waste generated in the country. It also requires that a national policy and a national program for radioactive waste and spent fuel management be prepared and revised as necessary.

The Hungarian national program for radioactive waste and spent fuel management was completed and approved by the Government on 24 August 2016 after the approval of the national policy by the

Parliament in April 2015. The national policy sets the basic principles of radioactive waste and spent fuel management, and the boundary conditions for the national program. The national policy also addresses the necessary high-level policy issues (including the back-end of the nuclear fuel cycle).

By distributing the tasks and responsibilities among the key players (government, authorities, waste management organization, etc.), the Act establishes the corresponding institutional framework. In this institutional framework, the Hungarian Atomic Energy Authority (HAEA) is the competent and independent authority of nuclear facilities (e.g., NPPs, research and training rectors, and interim spent fuel storage facilities) and radioactive waste disposal facilities. On the other hand, the national radioactive waste management organization assigned by the Government is the Public Limited Company for Radioactive Waste Management (PURAM).

10.2 National Policy and National Program

The national policy reaffirms that the Hungarian state takes the ultimate responsibility for the safe management of any radioactive waste and spent fuel (or HLW from its reprocessing) generated in the country. Consequently, the final residuals have to be disposed of within Hungarian borders, unless an international agreement between Hungary and another country enters into force in the future, where the appropriate disposal is achievable in an already operating repository.

The radioactive waste flowchart of the Hungarian national program is shown in Figure 10–1. Quantitatively the most significant waste stream is the LILW short-lived stream. Due to the existence of two operating LILW repositories in Püspökszilágy and Bátaapáti, the final disposal of this waste category is solved in Hungary.

Hungary's oldest operating repository, the Radioactive Waste Treatment Disposal Facility (RWTDF) is a near-surface LILW repository established in 1976 in Püspökszilágy, which receives waste only from non-NPP producers. Here a safety enhancement and security upgrade program is ongoing. It is expected that disposal activities can be maintained in the facility until the end of the time frame (2084) set out in the national program.

From the point of view of geological disposal, the National Radioactive Waste Repository (NRWR) in Bátaapáti is of interest, because its granite host rock (200–250 m beneath the surface) serves as a real safety barrier. This is discussed in detail in Section 2.

At present, there is no deep geological repository suitable for disposing of HLW or spent fuel in Hungary. The spent fuel assemblies of the Paks NPP are stored in the Interim Spent Fuel Storage Facility (ISFSF) in the neighborhood of the NPP.

An important feature of the Hungarian national program is that the decision maker has not yet found it necessary to make a final decision on the back-end of the fuel cycle. Although reprocessing of spent fuel has never been done in Hungary in the past, the option of future reprocessing is worth preserving.

Nevertheless, it is clear that a domestic deep geological repository is necessary for Hungary, regardless of any decision on the back-end of the nuclear fuel cycle. Not only reprocessing of spent fuel, but also operation and decommissioning activities, will inevitably lead to some amount of HLW.

Being aware of this, the national policy requires a flexible (reversible) yet active approach that we call "DO and SEE" policy. It means that instead of allowing delay in real actions (for instance, until the final political decision on the back-end is known), a real and ongoing research program for a deep geological repository is required.

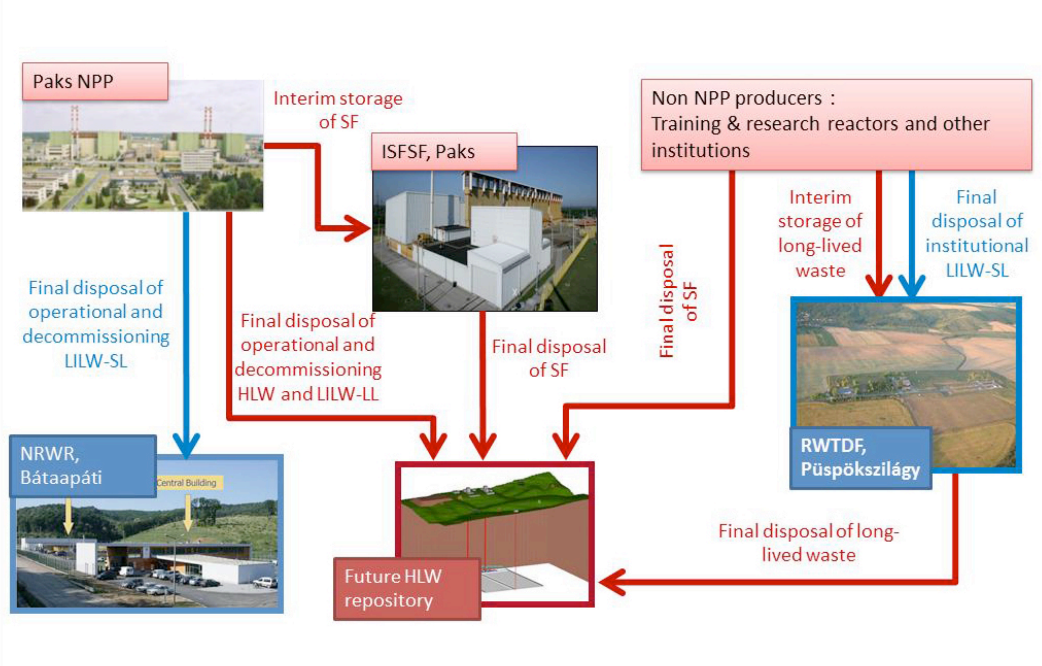

Figure 10–1. The logical scheme of the Hungarian national program. Blue line stands for LILW-SL, red line is for HLW and LILW-LL streams both requiring deep geological disposal.

While the final decision on reprocessing is not available, the research program and other planning activities in the implementation of the national program should be based on a reference scenario, which is currently the direct disposal of spent fuel in a domestic deep geological repository, together with other HLW arising from operation and decommissioning.

Obviously the flexible nature of the "DO and SEE" policy cannot be sustained forever. Eventually, when research and repository development activities require that the waste packages be characterized, it will become necessary to make a clear and final decision on the back-end of the nuclear fuel cycle.

A surface-based geological research program in Hungary aiming at a future HLW repository started more than two decades ago, and has already led to identification of a potentially suitable geological formation at the southwest foot of the Mecsek Hills: the Boda Claystone Formation (BCF). Further details on achievements and future challenges of this program are discussed in Section 10.3.

10.3 Public Limited Company for Radioactive Waste Management (PURAM)

The executor of the projects in Bátaapáti and in the southwest Mecsek Hills, both relevant from the point of view of geological disposal, is the national radioactive waste management organization, the Public Limited Company for Radioactive Waste Management (PURAM).

PURAM is a 100% state-owned enterprise, established by the Government in 1998, as required by the Act on atomic energy. The financial source of PURAM's activities is the Central Nuclear Financial Fund, a segregated state fund within the national budget, which is exclusively earmarked for radioactive waste and spent fuel management purposes that are defined in law.

Regarding PURAM's activities, other than the technical implementation of the national program, the importance of confidence-building efforts towards the local and wider public cannot be overemphasized. Since the beginning of both projects above, it has been clear that without firm public (especially local) acceptance, even technically well-founded disposal projects are destined to fail. To achieve its goals, PURAM needs to build a trusting and transparent relationship with the local communities (municipalities) around its facilities (i.e., the ISFSF in Paks, the NRWR in Bátaapáti, the RWTDF in Püspökszilágy), and the research area in the southwest Mecsek Hills. An important element of the trust-building practices in Hungary is that municipalities near a facility are supported in establishing so-called social control and information associations. These associations are allowed to carry out public control on PURAM's operation at the repositories, and serve as an excellent way to educate locals about radioactive waste management in general. According to the Act on atomic energy, the social control and information associations can obtain financial assistance from the Central Nuclear Financial Fund to finance the operating costs of the associations and some municipality development investments in the member municipalities.

10.4 National Radioactive Waste Repository

10.4.1 History

Since the extension of the RWTDF facility in Püspökszilágy to provide additional capacity to meet the waste disposal needs of the nuclear power plant was not feasible, a national project was launched to resolve the problems of disposing LILW of nuclear power plant origin. The preparation for the siting work was commenced in the framework of this project.

The entire area of the country was surveyed, based on data from the literature. In promising regions, where the local municipalities also accepted a potential repository site, preliminary investigation work was carried out to identify geological structures that could potentially be acceptable for the construction of a surface or subsurface disposal facility. In 1996, recommendations were formulated in the final geological, technical, safety and economical assessment reports for additional investigation work in the region of Bátaapáti to check the acceptability of the subsurface granite formation for the construction of an underground repository (Figure 10–2).

The investigation process was carried out in several stages (Figure 10–3), with the goal of proving the suitability of the target geological formation (granite) to host a low and intermediate level radioactive and short-lived waste repository. A wide range of surface and borehole-based geological and geophysical tools was used. Results and interpretations of these investigation programs led to the conclusion in 2003 that the Bátaapáti site met all the requirements and was suitable for the disposal of LILW (Balla et al. 2003). Between 2004 and 2007, additional underground investigations were carried out along two access tunnels, while a local referendum in 2005 resulted in high local public support to construct a LILW repository in Bátaapáti. In the meantime, the Hungarian Parliament gave the decision in principle to start the preparations to construct the repository in 2005. The surface facilities were completed in 2008, and since then have been in normal operation. So far, more than 6000 drums filled with LILW have been transported to the site from the Paks NPP, and prepared for final disposal. In 2012 the underground section of the repository system was opened, and the facility started to accept wastes in the first chamber. Lately there have been two main parallel operations at the site: (1) construction of additional chambers for future wastes, and (2) disposal of wastes. The ongoing activities at the site and in the repository system follow the principle of "design as you go," meaning that many aspects of the project (waste forms, chamber geometry, orientation, etc.) may change in the future to maximize the safety and efficiency of the entire facility.

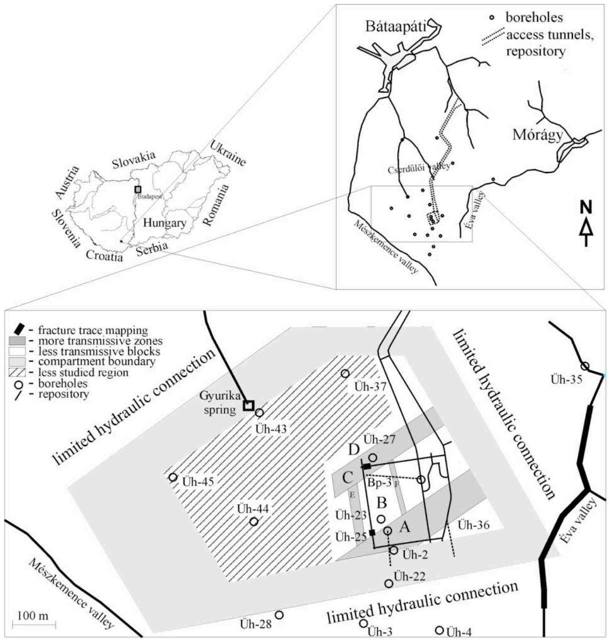

Figure 10–2. Generalized map of the study area (Benedek and Molnár 2009). A, C, E, and F indicate the more transmissive zones, and B and D are the less transmissive blocks. Note that in this figure only zones and blocks determined during field activities are displayed. In the western and northern reaches of the site, the location of more transmissive zones is uncertain due to a lack of data.

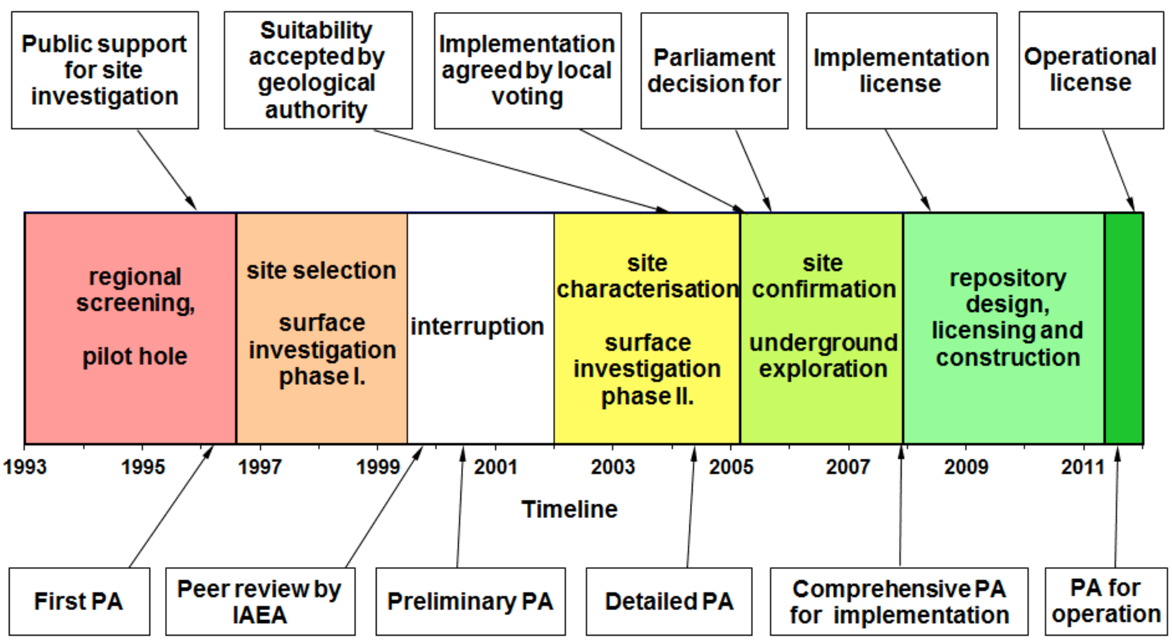

Figure 10–3. Main steps in the time-line of the Bátaapáti project.

10.4.2 Geological and Hydrogeological Setting

The Bátaapáti site is located in the southern part of Hungary (Figure 10–2) in the Mórágy Block, which is composed of intrusive igneous sequences of the Paleozoic Mórágy granite formation (Balla et al. 2003). The general strike of the intrusive body is NE–SW, and several rock types are present, such as porphyritic monzogranite, monzonite, and aplitic rocks (Király and Koroknai 2004). The contact between the different rock types is not always clear. Monzonite enclaves of different size (from small 1–2 cm to 1–2 km bodies) are usually enclosed within monzogranite. The magmatic body is frequently cut by Cretaceous sub-volcanic dykes. The upper section of the entire granitic complex is strongly altered and weathered. The Paleozoic rocks are covered by Quaternary, Miocene and Pliocene sediments, with a thickness of about 50 m on the hilltops, and thinning towards the valleys.

The hydraulics of the granitic rocks is primarily controlled by the fracture systems formed through several tectonic events (Benedek and Molnár 2013). The hydraulic interference tests carried out in several boreholes and the results of the hydrogeological monitoring system operating at the site indicate that the rock domain can be subdivided into several hydraulic compartments with very limited hydraulic communication (Benedek et al. 2009, Figure 10–4).

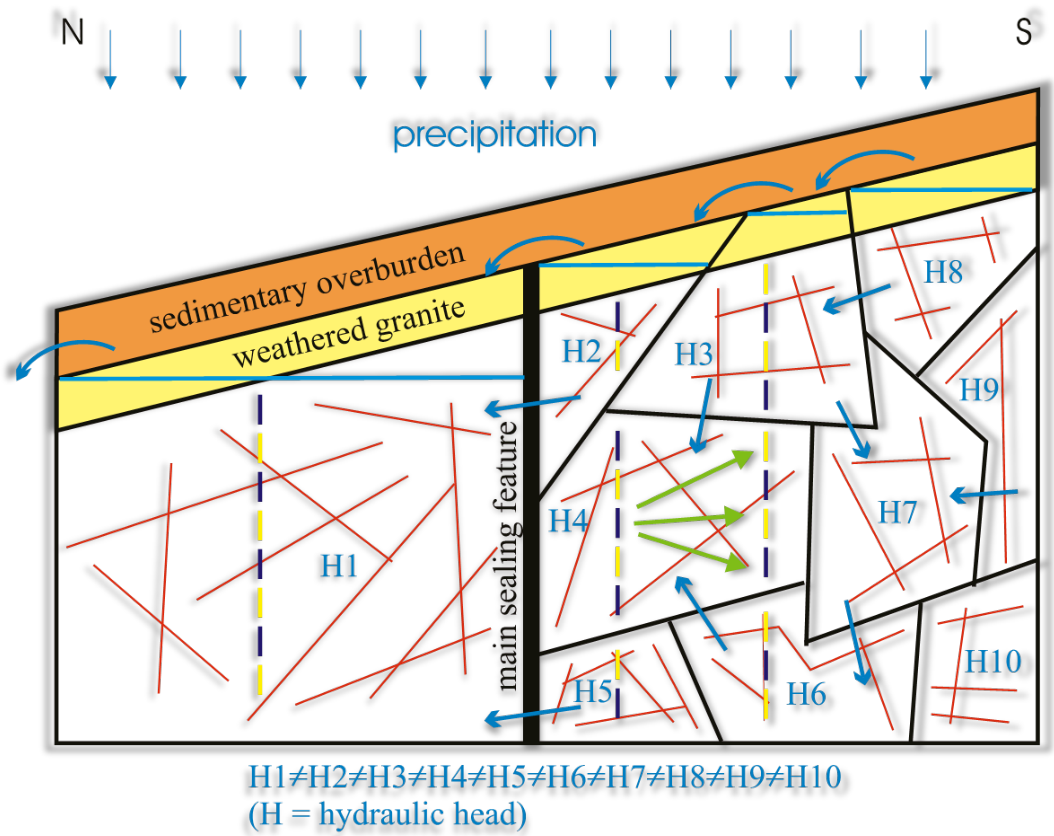

Figure 10–4. Schematic cross-section of the studied area (Benedek et al. 2009). The frequency of faults (black) is higher in the south (S), than in the north (N). The boundary between the northern and southern parts is represented by the main sealing feature. Yellow and blue violet sections of boreholes represent different packer intervals. Only packer intervals located in the same compartment are hydraulically connected (green arrows). Blue arrows indicate very limited flow through boundaries and overflow.

At the boundary of different compartments, boreholes penetrated zones affected by intense tectonics, such as sheared fault zones with intense mineral alteration, and formation of clay minerals. Another observation was that within individual compartments, the hydraulic heads are almost uniform, with only very small spatial variations. The relatively uniform pressures are attributed to the presence of large,well-connected, highly transmissive features providing for hydraulic connectivity within each compartment. As a consequence of the small hydraulic gradients within compartments, it is very difficult to determine flow directions. The practical implications of the site's hydrogeological conceptual model and the conceptualization of the fracture system are listed in Benedek et al. (2009) and Benedek and Molnár (2013).

10.4.3 Description of the Facility

The site of the National Radioactive Waste Repository (NRWR) consists of two parts: the surface and the underground facilities. Both parts of the site are divided into two segments: the radiation protection controlled zone, and the supervised zone.

Since the construction is taking place in parallel with the radioactive disposal, both surface and underground areas have two parts. One is where radioactive waste management operations take place; the other part of the area serves the needs of construction and the enlarging of the underground facility. Two access tunnels connect the surface and the underground parts of the facility. One tunnel serves as access to the disposal chambers that is part of the radiation protection controlled zone, and the other serves as access to the construction area, situated in the supervised zone (Figure 10–5).

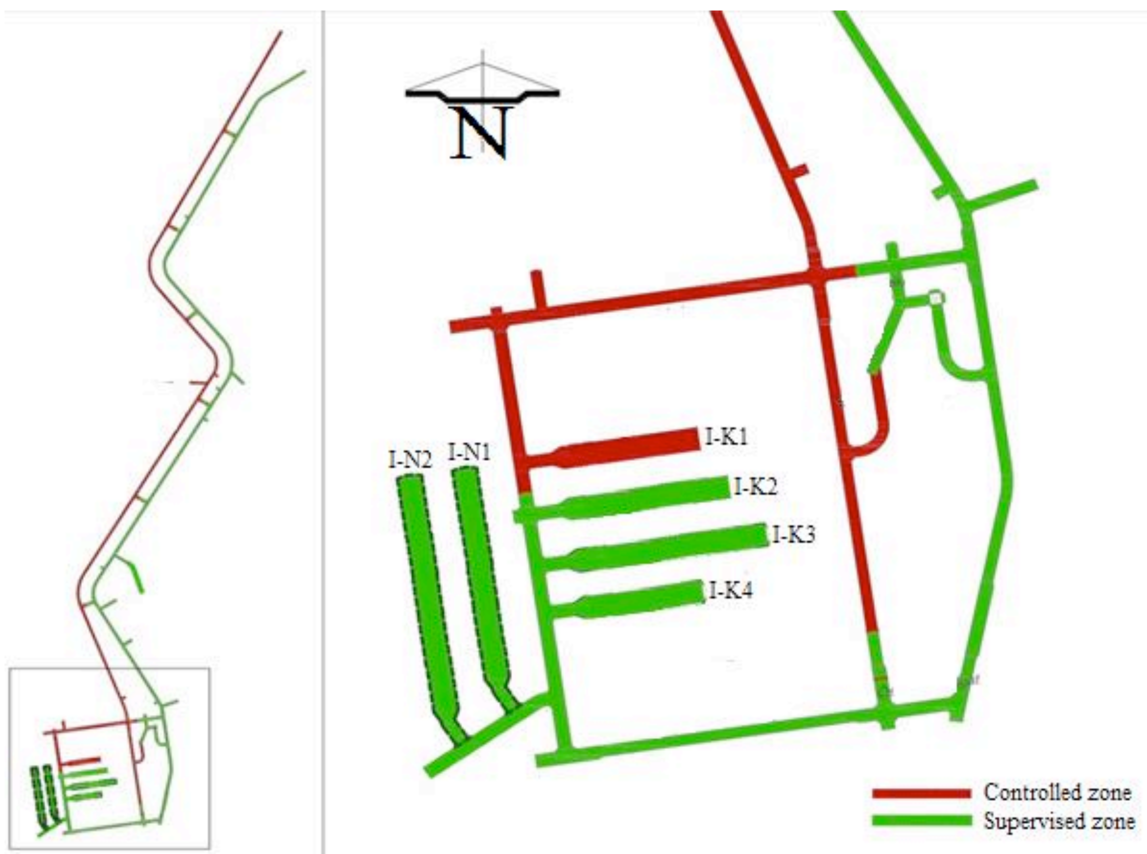

Figure 10–5. Underground facilities with the currently operating chamber (I-K1), the ones under construction (I-K2, I-K3 and I-K4), and with planned chambers (I-N1, I-N2).

Conditioning drums in ferro-concrete containers

Transporting ferro-concrete containers to the underground area

Final disposal in chamber I-K1

Figure 10–6. Steps towards the final disposal of the drummed waste in the NRWR: conditioning ferro-concrete enclosures, transportation, and final disposal.

On the surface, the administrative building, security center, and construction yard can be found in the supervised zone. The technological storage building, where the waste shipments are received from the Paks NPP, is situated in the controlled zone. Currently compacted solid waste is received in the form of 200 L drums. The surface construction area serves as a depot for construction material and the machinery yard, as well as the center of construction operations.

Historic waste in 200 L drums is stored in the technological storage building until the drums are cemented into reinforced concrete disposal containers, nine drums in each container. The drums are fixed with inactive mortar, and, after seven days of hardening, containers are transported into the underground disposal chamber I-K1 (Figure 10–6).

10.4.4 Need for Optimization and Possibilities

Optimization of the disposal system in Bátaapáti NRWR was needed because of several changes in the waste production and the potential layout of the underground repository. One of the driving forces was that the waste producer, Paks NPP, has managed to improve the liquid radioactive waste treatment system so that it results in less volume of the cemented liquid waste. In addition, new waste types (filters from the treatment system) are to be introduced. On the other hand, PURAM started to optimize the disposal system of the NRWR, taking into account the new waste and package types. The main aim of this optimization project was to use the available disposal rooms more effectively while maintaining the required level of safety.

To deal with the new situation, PURAM, Paks NPP, and several subcontractors coordinated to start a design project. The result was a new disposal design based on a new waste package with a thin-walled steel container that can be used to accommodate four drums of solid waste (compacted, non-compacted, filters, etc.). Liquid waste is incorporated into the grout used to fill the space between the drums and walls of the container (Bérci et al. 2010). This new type of waste package is called compact-waste package (CWP) (Figure 10–7) (Gyöngyösi et al. 2013).

The low permeability concrete container serves as a chemical and hydraulic barrier in the original disposal concept of the NRWR (chamber I-K1). The safety of the reinforced concrete container is not neglected. Instead, a reinforced concrete vault is erected and serves as a barrier in the disposal chamber. This vault will host the steel containers. Sections will be sealed by cement backfill after receiving about 400 containers, which is the planned annual shipment rate of the CWPs from the NPP.

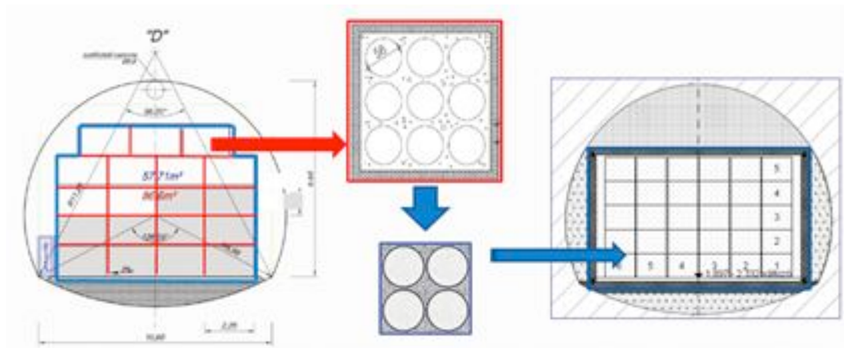

Figure 10–7. Change of the disposal concept of the NRWR: Placing several containers into a reinforced concrete vault, rather than using reinforced concrete for each container, maximizes efficiency of disposal.

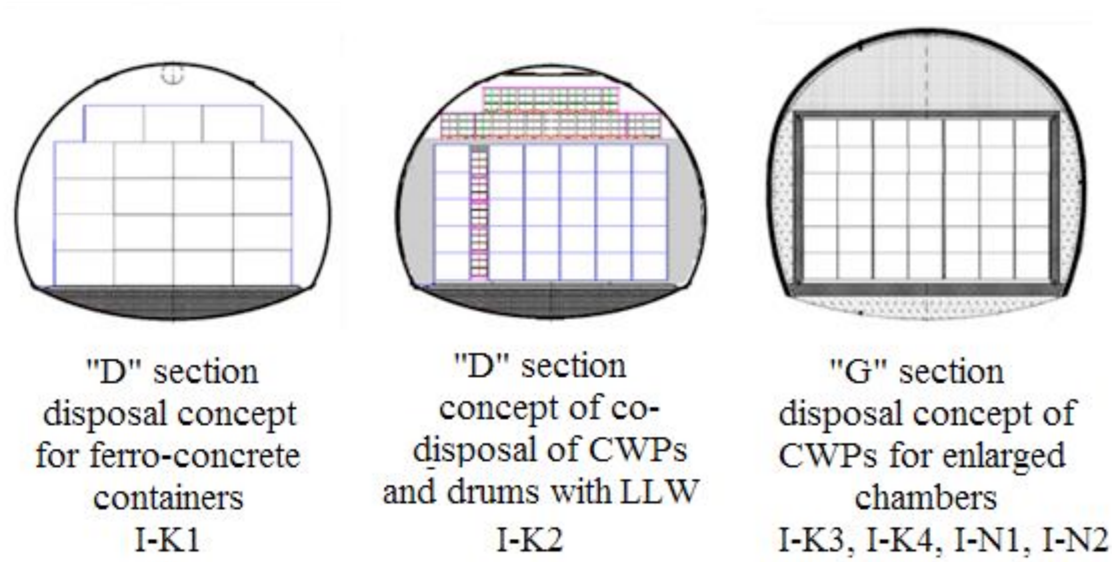

Figure 10–8. Comparison of disposal concepts, from left to right: currently used disposal concept (I-K1); planned concept for I-K2, with co-disposing CWPs and 200 l drums containing compacted LLW; disposal CWPs in the redesigned chambers (valid for the chambers being under construction or being planned).

Further optimization can be attained either by combining the two types of disposable waste packages, or by redesigning disposal chambers to achieve a higher active-inactive ratio in the chamber volume. Disposal chamber I-K2 was constructed with same cross-section “D” as chamber I-K1 (see Figure 10–8). It is feasible to dispose of drums containing LLW beside the CWPs in the vaults. Disposing drummed LLW on the top of the vault seems technically viable as well. A new design of the chambers (section “G”, Figure 10–8) allows PURAM to dispose CWPs with greater efficiency.

10.4.5 Results of the Safety Case Supporting Construction License Modification

To acquire authorization for the new disposal concept, PURAM issued a request to the licensing body to change the construction license of NRWR. One document supporting the license request was the safety case, summarizing all available knowledge about the new concept. Operational safety and post-closure safety were assessed. The results were used to derive preliminary waste acceptance criteria for the CWPs. Operational safety was assessed from both non-radiological and radiological points of view (Baksay et al.

2014). Non-radiological hazards were assessed for the waste management operations and the interaction between operations and construction. All systems were assessed using HAZOP methodology. The conclusion was that no "unacceptable level risk" could be identified for the new disposal concept, as described in Czakó et al. (2014a) and Czakó et al. (2014b).

The assessment showed that the optimized disposal concept results in higher dose rates around the CWPs than for the original design of reinforced concrete containers. Thus careful radiological planning of the operations will be necessary, with special attention paid to the backfilling of the disposal vaults.

Probabilistic analysis of the post-closure radiological impacts was performed. The results showed that the highest possible annual effective dose is expected to be well under the dose constraint (Baksay et al. 2014; Bőthi et al. 2014) defined for the NRWR site (Figure 10–9).

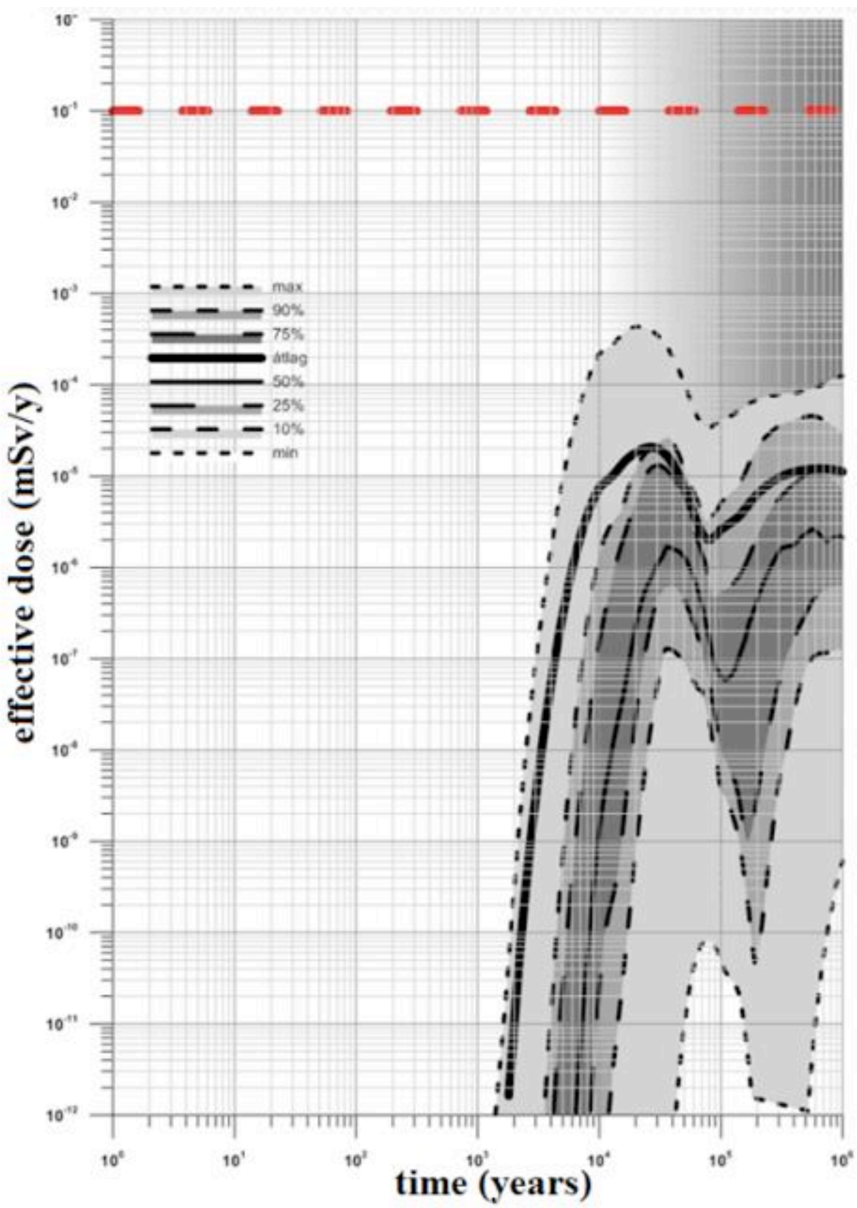

Figure 10–9. Results of the stochastic analysis of the post-closure radiological impact.

10.4.6 Future Plans

Hydrogeological interpretations and safety assessment calculations have indicated that the low conductivity clayey core of the tectonic faults at compartment boundaries plays an important role in the safety case (Benedek et al. 2009; Bőthi et al. 2010). These features minimize the hydraulic gradient within hydraulic compartments and the hydraulic communication between compartments, and favor diffusive transport processes for contaminant migration. However, the access tunnels penetrated these naturally sealing barriers during construction of the underground facilities (Figure 10–2). Therefore there is a need to demonstrate technologies to plug the access tunnels. To do that, a large-scale demonstration experiment is planned (Megyeri et al. 2014). The experiment's main goal is to demonstrate the ability to rebuild the long-term isolation of the hydraulic compartments. A schematic of the experiment is shown in Figure 10–10. The key component of the design is the low permeability bentonite brick section, which should provide the necessary hydraulic sealing of the access tunnels. The experiment is under construction now, and will be implemented in 2017.

In addition to experiments to ensure long-term safety of the site, the construction of additional chambers is considered. Currently only one chamber is in operation. This chamber receives concrete containers containing steel drums of radioactive wastes (Figure 10–6). Another already-excavated chamber is intended to fill up with wastes in 2017. The waste form is going to be different from that of the first chamber to maximize space availability for waste disposal. In this particular chamber, waste drums are placed in steel containers filled up with active concrete (Figure 10–7). The steel containers will be placed within a concrete vault. Two additional chambers are under excavation, and two more are in the planning stage.

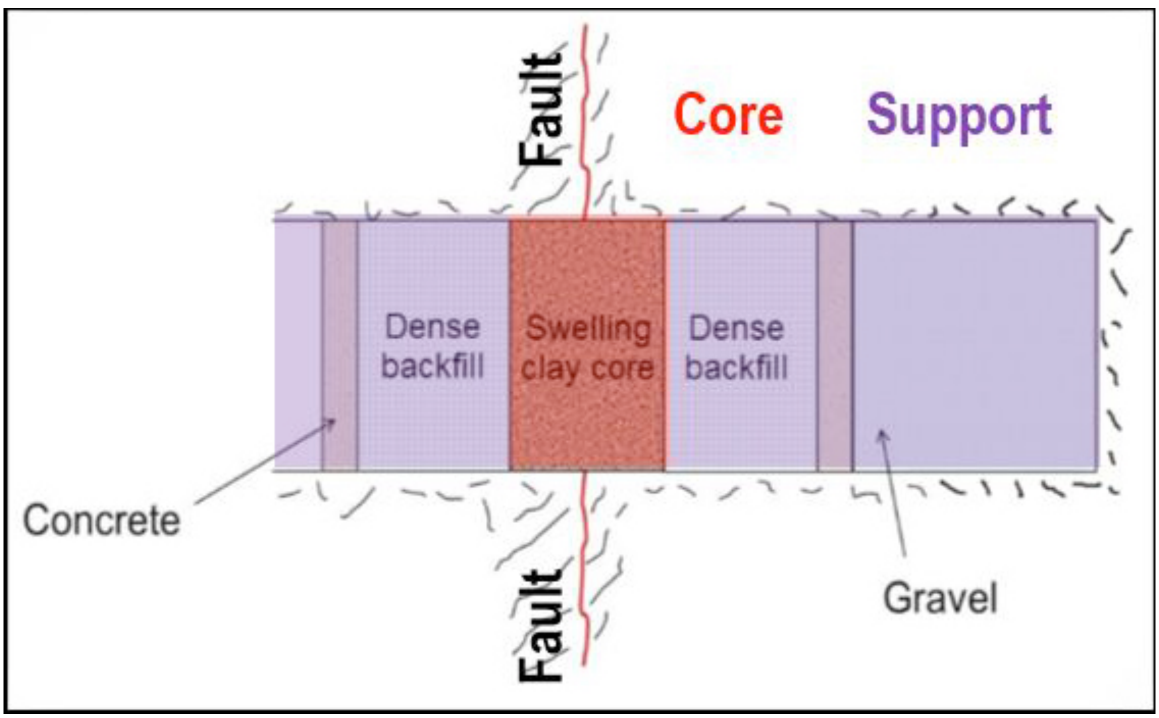

Figure 10–10. Generalized layout of the tunnel sealing experiment.

10.5 Siting Program for a Deep Geological Repository

10.5.1 History

The idea that the Boda Claystone formation (BCF) in the West Mecsek Hills might be suitable to host a deep geological disposal facility dates to 1983. This is the bedrock (layer) of the uranium-rich sandstone formation that has been mined in the past. To determine the likely location of the repository and to characterize the host formation, several projects were initiated during the following two decades. These early projects did not follow the principle of gradual understanding, but were adapted to the circumstances and opportunities of those times, such as the existing uranium mine openings.

The preparations for disposal of HLW and long-lived LILW started in 1993. In 1994, an exploration tunnel was excavated in the Mecsek Uranium Mine, reaching the BCF, and on-site underground data acquisition began at a depth of ~1100 m. The formation itself was explored underground by a tunnel extending 500 m into the claystone. The tunnel was used as an Underground Research Laboratory (URL), providing a large amount of on-site geological data. In 1998 the mining activities ceased and the mine was flooded, so the opportunity to perform underground investigations ended. The results of the investigation programs confirmed the potential suitability of the BCF, considering its extent, isolation, and geotechnical capabilities. An additional program was proposed to investigate characteristics of the formation in detail.

However, when PURAM was established in 1998, it systematically re-evaluated its strategy and disposal concept. In 2000 it initiated a country-wide screening process to identify potential host rocks. This country-wide screening was carried out by evaluating the potential host rock formations in detail, based on information that was available in literature, including the previous investigations in the URL. Thirty-two lithological formations were identified that may be potentially suitable for a deep geological repository within the territory of Hungary. This comprehensive investigation confirmed that the BCF has the highest potential among the suitable host rocks for a HLW repository.

Two short surface-based investigation programs have taken place since 2000, the first in 2005 and the second in 2014. Both used a wide range of geoscientific tools, such as boreholes, seismic sections, and other geological-geophysical techniques. However, financial obstacles emerged during these programs: the first project was interrupted, while the second was suspended.

The goals of geological investigations were to characterize the BCF, and to reduce the investigation area to 10 to 15 km^2. The current geological investigation program covers surface activities, including geological and geomorphological mapping, hydrogeological reambulation, trenching, and the drilling of six deep boreholes accompanied by seismic profiling.

10.5.2 Geological and Hydrogeological Setting

The ~265 million year old (Middle Permian-Guadalupian age) BCF is explored on the surface and underground in southwest Hungary, in the southwest Mecsek Hills, near the city of Pécs. The known areal extent of the formation is about 150 km^2. The BCF is part of the Permian-Triassic sedimentary sequence that makes up the Western Mecsek Anticline.

The lithofacies represent facies varieties of an intermittent saline playa lake in a desert-to-semidesert environment (Konrád et al. 2010). The properties of the BCF have been determined in part by the extreme climate, inflow, and geochemical conditions of the sedimentation, and in part by its diagenesis. These conditions led to the formation of an extremely high proportion of sedimentary albite in the rock, which

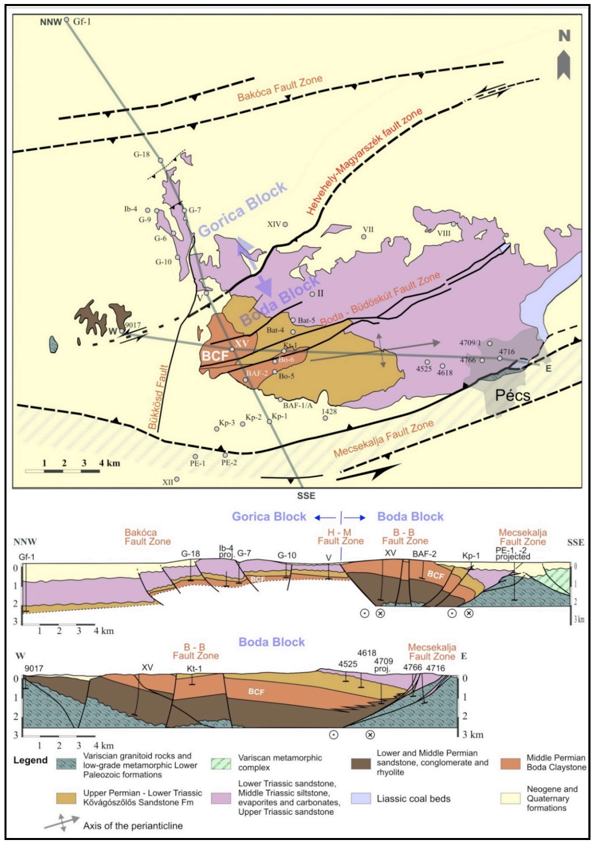

Figure 10–11. Geological map and cross sections of the Western Mecsek Anticline, including the surface outcrop of the BCF (compiled by Konrád 2014).

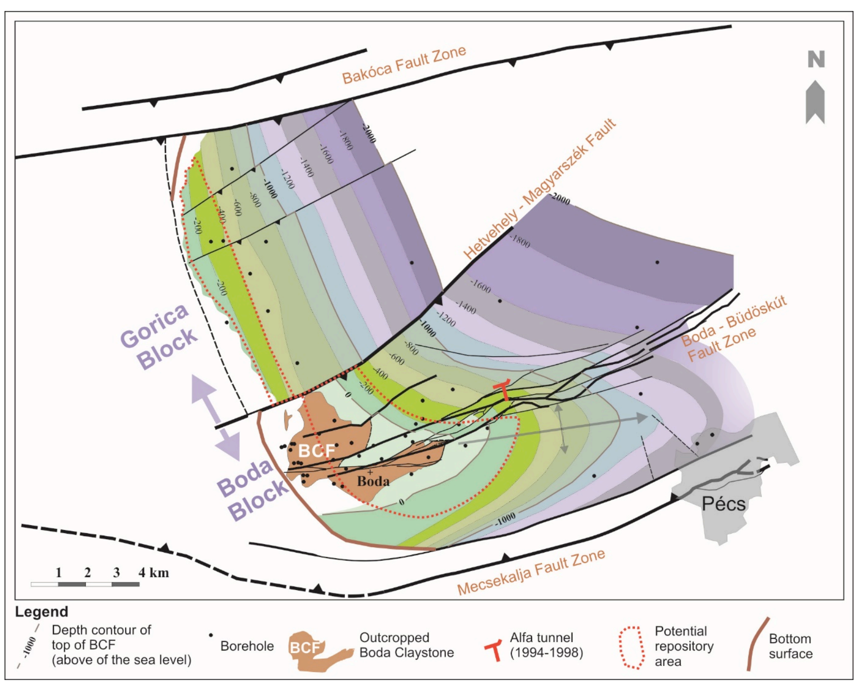

Figure 10–12. Depth contour of the top of the BCF, including the surface outcrop of the BCF (compiled by Konrád and Hámos 2014).

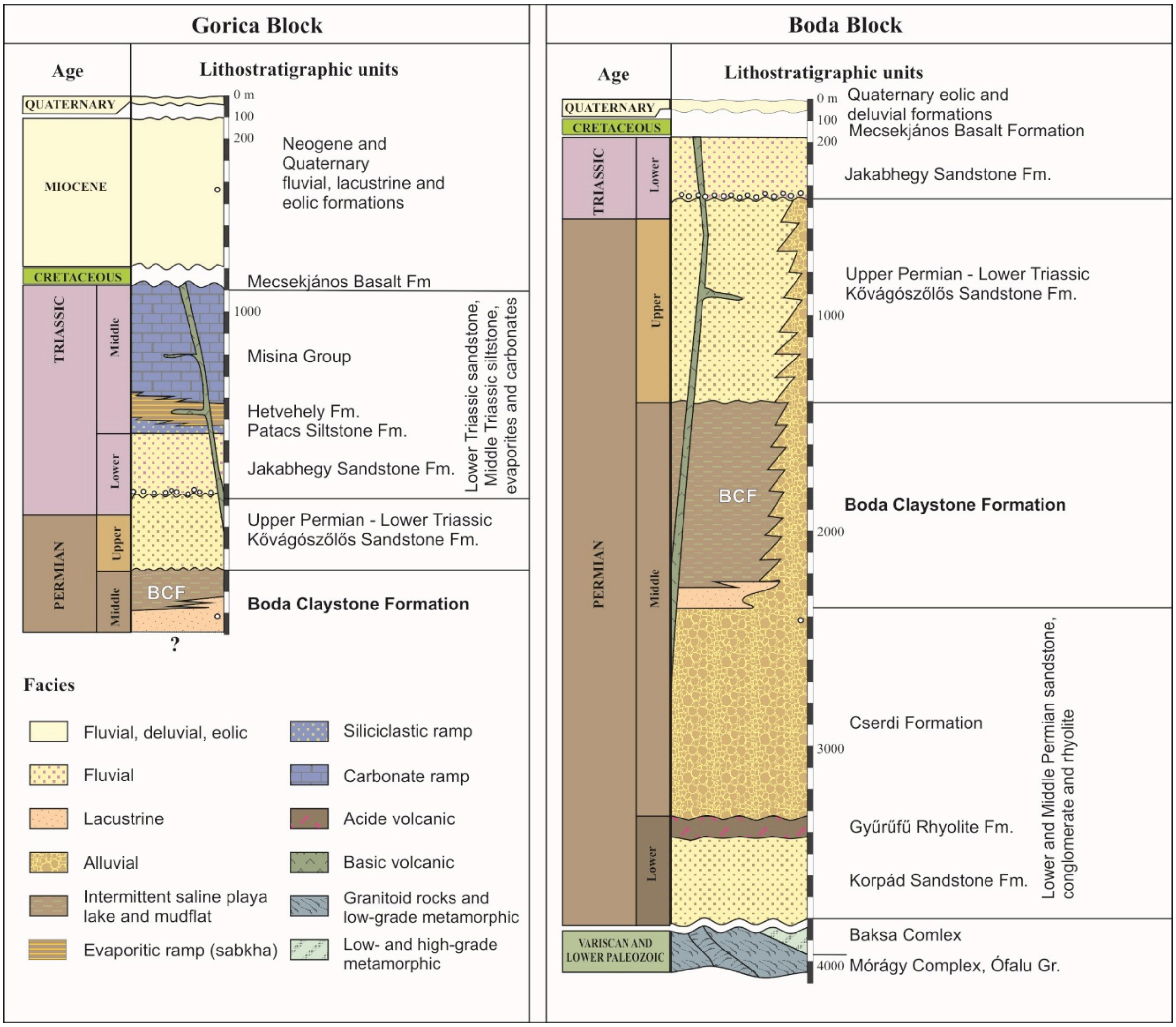

Figure 10–13. General geological/lithological columns for the comparison of lithostratigraphical units in the Boda and in the Gorica Blocks . Compiled by Konrád (2014).

is the most characteristic mineral of the BCF. The lack of organic material is also remarkable. Intensive oxidation processes influencing the rock's mechanical properties are not expected to occur within the lifetime of the repository.

Since sedimentation conditions were nearly constant for millions of years, an extremely thick quasi-homogeneous sequence was formed. Due to minor, cyclic changes in the depositional environment, the formation can be divided into six characteristic rock types, but the main body of the formation (approximately 90%) consists of one rock type, the albitic claystone. Within this host rock there is a unit extending up to 37 km^2, which has been identified as a potential disposal area at depths between 500 and 900 m below the surface. Based on the geometry (thickness) and level of consolidation, two different areas, separated by a fault zone, can be distinguished: one in the formation's central and southern part, which is called the Boda Block, and another in the northern part, called the Gorica Block (Figures 9–11 and 9–12). In the two blocks the thickness of the BCF, the mineralogical composition, and the degree of diagenesis are different (Figure 10–13).

The main rock-forming minerals of the BCF are: clay minerals (dominant are illite-muscovite and chlorite), authigenic albite, quartz, carbonate minerals (calcite and dolomite), and hematite (Varga et al. 2005; Varga et al. 2006). In addition, barite, anhydrite, authigenic K-feldspar, and detrital constituents were identified in trace amounts (Figure 10–14).

Due to its long tectonic history, the BCF has a discontinuous character. Detailed surface and underground geological mapping reveal traces of four different tectonic periods. Joints and fractures are completely filled by various clayey, carbonatic or sulfatic materials. Depending on the density, type, and orientation of discontinuities, the measurable hydraulic conductivity of the actual rock zone can vary within a relatively broad interval (typically 10^{-9} to 10^{-13} m/s, with values of up to 10^{-8} m/s). In the southwest Mecsek Hills region, the geological formations are usually hydrogeological units as well. But in some cases a single hydrogeological unit is built up from several geological formations. The BCF is an aquitard from hydrogeological point of view, and its hydrologic storage capacity is elevated only close to the surface.

At greater depths the larger tectonic zones may have significant hydrologic storage and transmissivity. The hydraulic conductivity K of the weathered zone and the tectonic zones is in the range of $K = 10^{-9}$ to 10^{-7} m/s, but the sound rock has $K = 10^{-13}$ to 10^{-10} m/s.

Bulk porosity and hydraulic conductivity of the intact rock matrix is very low (0.6–1.4%; 10^{-15} m/s). The average unconfined strength exceeds 100 MPa. Due to the burial (thermal) history of BCF, the possible impact of heat production of HLW (e.g., alteration of clay minerals, thermal softening) is assumed to be limited compared to younger argillaceous formations.

Figure 10–14. Tectonic joints on the wall of the URL exploration tunnel filled with calcite veins.

The aspect of geotechnical stability does not restrict the construction of a repository at depth. Because of its geochemical and geotechnical character, the BCF does not show complete self-healing behavior, even at the maximum depth of final disposal (1000 m). However, some partial self-healing effects have been recognized.

Porosity, determined using the Hg-porosity with pressure up to 2000 bar), and gas permeability measurements indicated the microporous nature of the rock and rock microfracturing. The secondary porosity (aperture and frequency of fractures) determines the water storage capacity of the rock, as well as the nature of water flow. Based on the results of *in situ* hydrodynamic tests, permeability of the claystone was estimated to be between 10^{-19} and 10^{-15} m^2. Based on the chemical analytical results, the salinity of the BCF pore water was estimated to be between 1.5 and 7 g/l, and water was of Na^+-SO_4^{2-} HCO_3^- and Na^+ HCO_3^- SO_4^{2-} types.

10.5.3 Disposal Concept and Preliminary Safety Assessment

Based on the reference scenario of direct disposal for spent nuclear fuel, the first, and so far last, conceptual design for the long-term management of spent fuel was developed in 2005 and then revised in 2008. Initial design of the disposal facility was based on the Swedish concept, despite differences in lithology.

In 2005, a preliminary safety assessment (Dankó et al. 2005) was carried out to assess the suitability of the host formation. This assessment was based on data gathered during exploration of the area before the year 2000. This assessment confirmed that the BCF is a suitable potential host rock due to its low porosity, low permeability, and high isotope retardation capability. However, it must be stated that in 2005 the available information on the BCF was not comprehensive, while some concepts included in the models were hypothetical.

The conceptual design of the Deep Geological Repository (DGR) was developed after the preliminary safety assessment. Assessments of inventory, heat load, packaging information, criticality and radiology were carried out to help design the DGR in the conceptual phase.

The current disposal concept considered a copper overpack on the canisters. The layout of the repository was based on the assumption that disposal of canisters will take place in vertical disposal holes drilled from disposal tunnels excavated within the BCF. The design of the encapsulation plant was based on the Swedish concept as well. The disposal tunnel system was planned for 500–800 m below ground surface, along with associated surface facilities. In the design, underground construction activities will be undertaken via vertical shafts. The disposal shafts will be connected to one another and to the service area by ventilation ducts and utility piping. The underground space will be constructed by conventional drill and blast methods. It is assumed that the large-section underground drifts will require rock bolts and a sprayed fibrous concrete lining with an average thickness of 10 cm, while for the small-section drifts (including the disposal drifts), rock bolts and a 5 cm thick sprayed concrete lining will suffice.

Due to numerous uncertainties, such as the lack of a defined back-end strategy for the management of spent fuel (and thus the expected inventory of the highly active residue of nuclear energy production), or of an appropriate repository area, as well as unknowns regarding the main geometry and geological characteristics of the selected site, the disposal concept will be reassessed before each decision point (e.g., each decision about narrowing of the research area, location of the repository site, building of the URL and/or deep geological repository, decisions about the disposal facility operations). Such revisions may result in changes to the original design.

10.5.4 Future Plans

Figure 10–15 shows the BAF long-term research program schedule, divided into phases of the geological investigation and development of repository program. PURAM has developed an investigation plan for the second stage of surface Phase I, which relaunched the investigation program. This is a continuation and completion of the aborted first phase of the 2005 program. The purpose of this research is to classify the BCF general site, to acquire geological data and information for the safety assessment, and to reduce uncertainties. The ongoing research project in southwest Mecsek Hills covers 87 km^2.

The green line in Figure 10–15 is the boundary of the investigation area, while the territory inside the brown line (37 km^2) shows the potential disposal area. This is a surface-based projection in which BCF is located in a favorable depth range (500 to 900 m). Research will include assessing geomorphological risks, creating the geological space model, describing the hydro-geological conditions, evaluating the geodynamic processes, and characterizing the host rock. The drilling of 800–1500 m deep boreholes plays a key role in the investigation of the BCF formation and its immediate environment. Core samples from the boreholes are documented, sampled, and inspected (Figure 10–17).

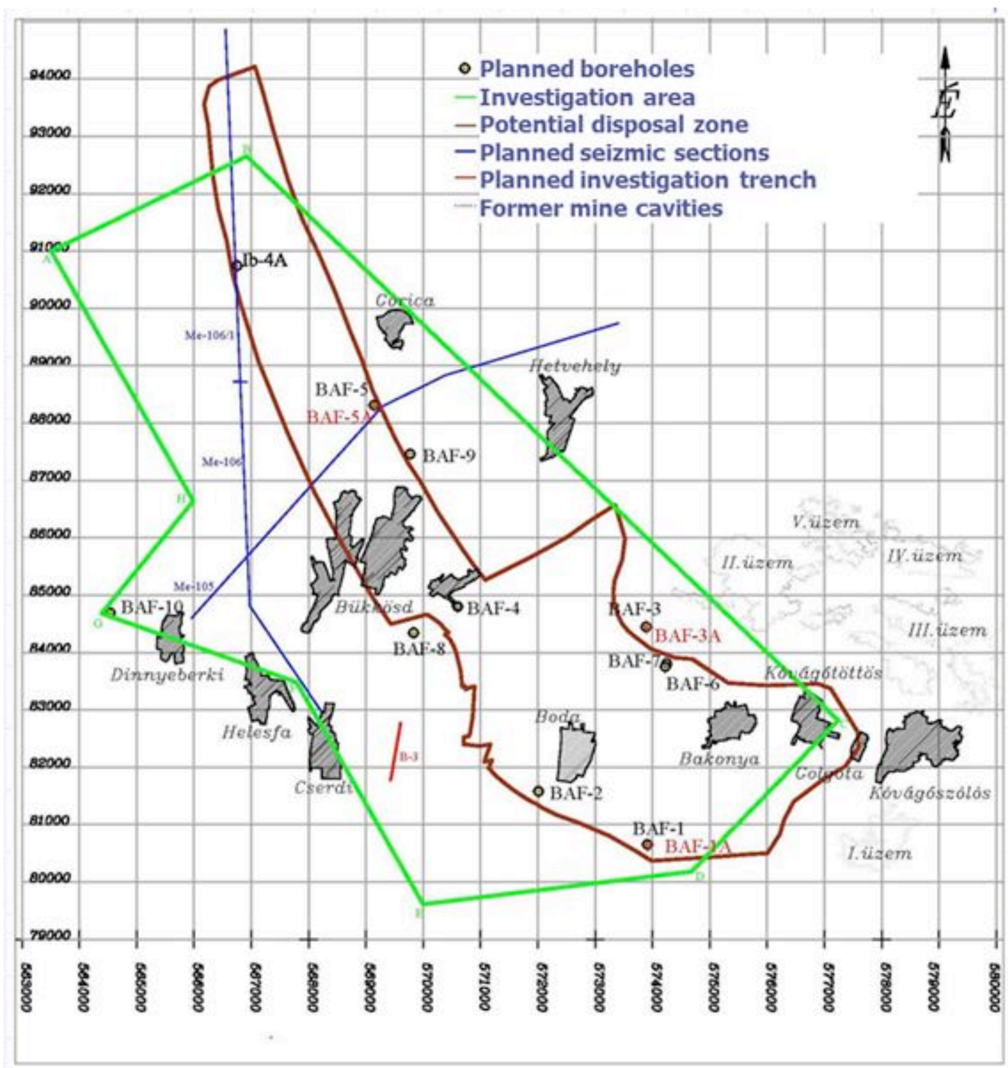

Figure 10–15. Map of the investigation area in 2014.

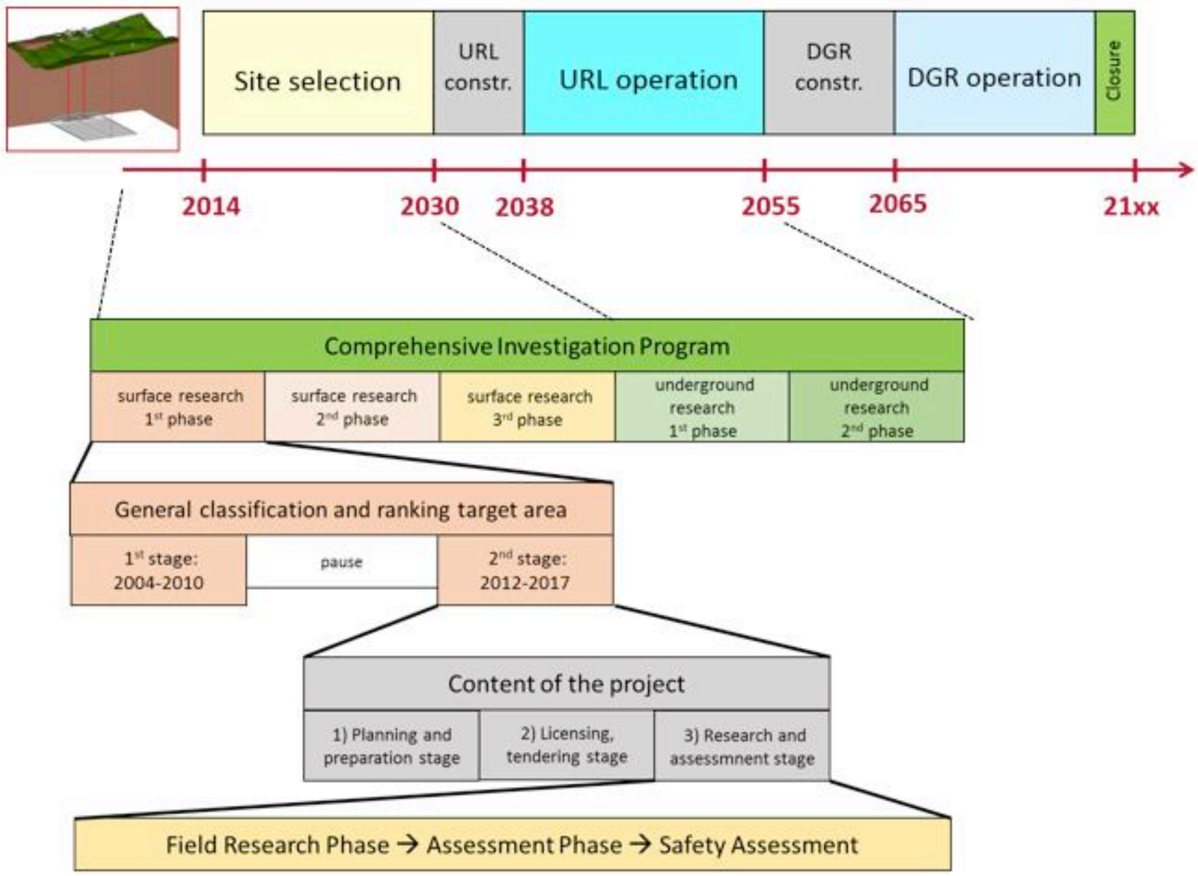

Figure 10–16. Schedule of investigation.

Figure 10–17. BCF cores from the BAF-2 borehole (539–544 m), Boda Block.

In the borehole complex, geophysical, geotechnical and hydraulic measurements are performed. The screening of the space between the wells is carried out by surface-based geophysical methods, providing seismic reflection profiles.

The purpose of the surface-based exploration phase I is the selection of the target site and its general characterization. At the end of the current phase, the task is to designate within the 37 km^2 potential repository area the most appropriate 10–12 km^2 site. The research will continue on this site with the surface-based investigation phase II, at the end of which the 1–2 km^2 territory for the underground site and its associated surface facility area could be selected. Characterization will also take place in this phase. The surface-based investigation phase III aims at the preparation of the underground research laboratory. PURAM intends to finish the site selection process by 2030, the planned end date of surface-based investigation. This site should include the potential location of surface and subsurface facilities and the URL as well.

As part of the on-going conceptualization, PURAM has initiated work on a Project Development Plan, which has the goal of determining the decision points and their contexts for future research and development of the deep geological repository. PURAM is committed to the "DO and SEE" policy based on a phased approach.

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10.7 Acronyms

BCF—Boda Claystone Formation

CWP—Compact-waste package

DGR—Deep Geological Repository

HAEA—Hungarian Atomic Energy Authority

HLW—High-Level Waste

ISFSF—Interim Spent Fuel Storage Facility

LILW—Low and Intermediate Level Waste

NPP—Nuclear Power Plant

NRWR—National Radioactive Waste Repository

PURAM—Public Limited Company for Radioactive Waste Management

RWTDF—Radioactive Waste Treatment Disposal Facility

URL—Underground Research Laboratory

Chapter 11

Host Rock Characterization, In-Situ Experiments, Numerical TMH Simulations and Natural Analogue Studies in the Indian Deep Geological Disposal Program

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ABSTRACT: Granitic rock is under consideration for hosting a geologic repository 500 to 600 m deep, with an initial capacity of 10,000 overpacks and with provisions for further expansion. The Indian reference design is a 2-m-long overpack emplaced in a 5 m deep pit, 0.85 m in diameter, in disposal tunnels of the repository with 50 cm thick layers of highly compacted clay between the overpack and the rock, and the top of the disposal pit covered with a concrete plug. The design has been analyzed for its spatial and temporal stability under the combined impact of thermal and mechanical stresses with the help of computer codes FLAC 3D, UDEC 3D, etc.

A field scale *in-situ* heater test, of six years duration, has been conducted in a gold mine in South India, to observe the evolution of the thermal field around disposed waste and changes in the strength and permeability of the rock mass, and to validate modeling results. A more realistic simulation is planned to be conducted in the Underground Research Laboratory at 200-300 m depth in granite rock.

Nationwide studies on granites spanning over 100,000 square kilometers have been undertaken. Such granitic bodies have been tested using state of the art monitoring technologies involving GIS and satellite based studies, geophysical investigations, and deep borehole drilling. Large amount of information on geological, structural, hydrogeological and geochemical parameters has been generated on samples retrieved from as deep as 600 m. A number of clay-sand admixtures are currently being evaluated for establishing the rock barrier function under the influence of the temperature field, which is generated around the disposed waste overpacks.

11.1 Introduction

High-level radioactive wastes (HLW) contain heat-emitting fission products such as isotopes of Cs and Sr, as well as long-lived actinide isotopes. The permanent disposal of such wastes therefore necessitates their isolation from the accessible biosphere over extended periods of time. The duration of such time frames are set by national regulatory authorities and depends mostly on the half lives of the involved radionuclides. The period of concern for long-term safety may run into few tens of thousands of years for spent fuel disposal, but only a few thousand years for waste arising from reprocessing.

Specifically designed and constructed underground Deep Geological Repositories (DGR) in suitable geological formations and at appropriate depth are under consideration worldwide for ultimate deposal of HLW comprising either vitrified waste arising from reprocessing of spent fuel or the spent fuel itself. The proven occurrences of millions of years old rocks at depths with almost negligible alteration can be considered as one of the most convincing indications supporting a possibility of the nuclear waste disposal within these rock types. While the feasibility of this mode of disposal has been demonstrated to a large extent by extensive experimentation in the Underground Research Laboratories (URL) worldwide in last four decades, the construction and operation of the DGR for actual disposal of such waste has yet to happen. The major impediment to realization the DGR has not been the lack of technology or methodology, but rather the socio-political aspects.

The reference geological disposal system considered in India involves disposal of Vitrified Waste Product (VWP) in a vertical mode within a pit of suitable dimension excavated in galleries of the underground DGR about 400-600 m depths in granitic rocks (see Section 10.2). The canister filled with VWP and stored in metallic overpacks thus constitutes the basic disposal unit. Protective layers of clay backfill and buffer as components of an engineered barrier system are proposed to be inserted between the rocks and the overpacks to retard the transport of radionuclides that may eventually be released from the VWP. These buffer layers are also designed to provide a cushion to protect the overpack against possible rock movements and retard the ingress of groundwater into the disposal pit. In most cases the long-term safety of geological disposal beyond the thermal phase (~300 years) induced by heat-emitting fission products relies upon the natural barrier functions of the rocks around the disposed waste. Therefore, selection of a suitable host rock and a suitable site constitutes the most critical stage in the development of such repositories (Bajpai 2004, 2006; Mathur et al. 2001).

A number of geological domains and host rocks are currently under study worldwide with a view to understanding their suitability in providing isolation and confinement to such wastes. Among these, granites (Canada, Sweden, Korea, Finland, etc.), and argillaceous rocks (Belgium, France) emerge as important rock types as the natural barrier for a DGR (Bajpai 2004, 2006).

The responsibility of ensuring the long-term safety of the public and the environment greatly relies on the efficacy of protective properties of host rocks, serving as natural barriers, which require an application of systematic scientific studies for the selection of such rocks. In most cases, initial suitability criteria within larger regions are methodically narrowed down through stage-wise investigations with introduction of finer and micro-level criteria toward later stages of investigations. The important desirable characteristics of a suitable host rock include high strength, thermal conductivity, and sorptive capacity for radionuclides of concern. Low thermal expansion and hydraulic conductivity, together with homogeneous chemical and mineralogical compositions capable of limiting unfavorable chemical reactions between the rock, fluids, and waste package are other important properties. The geological domains suitable for a DGR are also expected to lie in a low seismicity area, with minimum forest cover, agriculture, groundwater, surface water, and mineral abundance, as well as absence of structural pathways like faults, shear zones, dykes, etc.

In India, the need for a DGR will arise only by 2060; nevertheless, research and development work has been in progress for the last three decades in the fields of *in situ* experiments, natural barrier characterization, numerical modeling, conceptual design, and natural analogues of waste forms and repository processes. Keeping in line with international developments, the initial focus of work in 1980s mainly centered on setting up a generic URL in some of the underground metallic mines in India. These efforts finally culminated in the development of an underground chamber in a gold mine located in South India. This URL functioned for about ten years in 1980s and was used to conduct a two-component version of rock-canister interaction experiment. In the early 1990s until the turn of the millennium, India undertook extensive field geological and structural screening of a few vast regions occupied by granites with a purpose to develop a comprehensive database of host rock characteristics and assess their suitability as the host rock of a DGR. The current efforts within the Indian geological repository program are directed toward setting up a purpose-built URL in suitable host granitic rocks. The geological repository program has been amply supported by integrated research and development activities on buffer and backfills, natural analogues, numerical modeling, *in situ* testing of rock mass properties, etc. (Bajpai 2008a, 2008b; Narayan et al. 2007).

11.2 Conceptual Design

A conceptual design and layout of a DGR, with a capacity of tens of thousands of overpacks has been developed based on the analysis using suitable computer codes of site-specific data on host rock properties, geological conditions, overburden stresses, depth of underground excavations, and radiological and thermal characteristics of VWP's (Goel et al., 2003). The facility is planned at 400-600 m depths and occupies an area of 2×2 km^2. The conceptual design of the DGR includes one main shaft (6 m) for accessibility and another ventilation shaft (4 m). The facility comprises two orthogonal transportation tunnels, each 800 m long. A total of 63 disposal tunnels (each 110 m long), each holding about 40 waste overpacks, aligned at right angles to transportation tunnels have been included in the design (Figure 11–1). The disposal pit depth for hosting a 2-m-long overpack has been set at 5 m with a diameter of 85 cm. A layer of compacted smectite clay bricks with a maximum thickness of 50 cm is proposed to be inserted between the overpack and the rock mass (Figure 11–2).

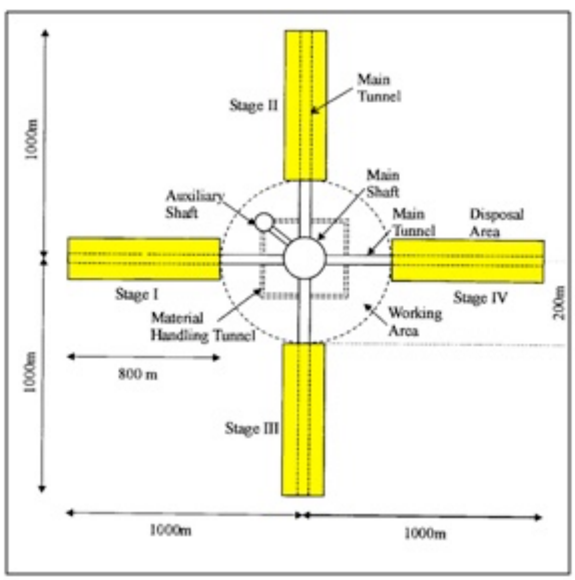

Figure 11–1. Conceptual design layout of DGR. Sixty-three disposal tunnels (not shown) are perpendicular to the main transportation tunnels in the yellow disposal areas. About 2500 disposal pits would be located in each of the four disposal areas, with one overpack vertically emplaced in each pit.

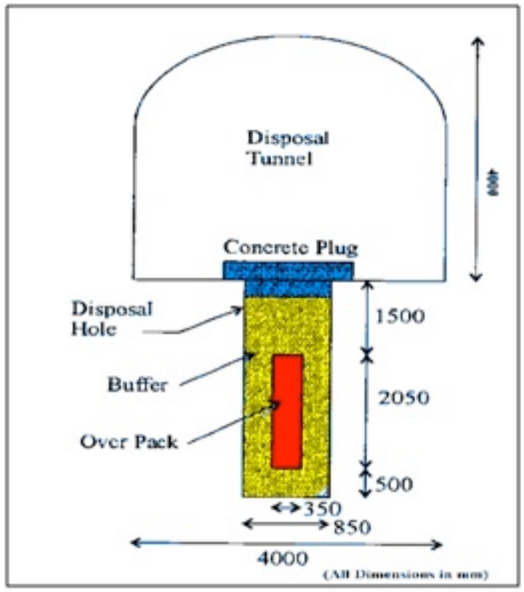

Figure 11–2. Cross-section of disposal tunnel, showing overpack containing waste emplaced in disposal pit.

Due to the prevalence of closely spaced vertical fractures in granites, vertical pits are positioned to avoid intersecting fracture joints. Similarly, to achieve the tunnel stability, the disposal tunnels are oriented parallel to the direction of the maximum principal stress and are planned to be excavated by state-of-the-art excavation methods. For mitigating any adverse effect due to the thermal load, use of admixtures of clay and sand or clay and crushed host rocks with good thermal conductivity is desirable. The DGR design is developed incrementally to accommodate uncertainties that remain unresolved during site characterization. The design undergoes continual refinement as more site data are acquired during the site evaluation program (Yadav et al., 2007; Varma et al., 2005; Bajpai et al., 2006b).

11.2.1 Conceptual design using numerical modeling

Components of the conceptual design, viz. single disposal pit, multiple disposal pits, disposal tunnels with multiple disposal pits, the complete repository, etc., have been analyzed with the application of commercial computer codes such as UDEC and FLAC^{3D}. These codes are used for prediction of temperature and stresses as functions of time and distance from the emplaced waste. The analysis of the advance of heat flux across single overpack with thermal flux of 500 W/m^2 in a disposal pit backfilled with bentonite clay, and also the analysis of thermal history of the entire DGR with tens of thousands of disposed overpacks reveal that granite host rock and clay-based buffers facilitate smooth dissipation of heat and maintain the temperature below 100°C in all parts of the DGR, thus meeting the design requirement to maintain temperature well below 100°C throughout the DGR evolution. The 1000-year predictions of the temperature field at various distances from the disposal tunnels showed that the temperature will be well below 100˚C, so that vaporization of groundwater may not be a significant factor adding to vapor-pressure-induced groundwater flow and associated enhancement of radionuclide transport. The thermal history of the complete DGR loaded with 10,000 overpacks indicates that the structure is expected to remain quite stable after 500 years without any significant impact from the combined thermal and mechanical stresses. Maximum instability has been observed after 50 years following the emplacement, where minor spalling of rocks is induced by thermal and mechanical stresses. Combined thermal and mechanical stresses resulting from excavation of disposal pits and emplacement of waste have been estimated well below 32 MPa, and hence disposal pits in granites with unconfined compressive strength of 102-150 MPa can be considered stable without requiring any support (Table 11–1). Another 3D analysis of displacement around the excavated face of disposal tunnels (300 m deep) for a DGR has shown better stability of elliptical and horseshoe-shaped tunnel as compared to other shapes. In these cases, the maximum principal stress, 20.23 MPa, occurs at the side walls of the elliptical tunnel and then decreases along the side walls. The maximum vertical displacement, 1.70 mm, occurs at the crown of the tunnel, which is 0.02 mm more than for a circular shape tunnel at the same depth. Therefore carefully selected granites with good mechanical strength and thermal conductivity can provide a stable DGR (Bajpai 2009a; Singh et al., 2009).

Table 11–1. Typical stress concentrations induced by excavation and thermal loading around D shaped disposal tunnel as modeled with UDEC

Location	σ_1 (MPa) (Excavation)	σ_1 (MPa) (Excavation + thermal)	Stress-Strength Ratio (σ_1/ σ_c)	
			Excavation	Thermal
Crown	18.1	18.46	0.15	0.15
Floor Centre	7.41	9.97	0.06	0.08
Face-Arch Intersection	17.36	20.1	0.15	0.17
Floor Corner	25.92	31.95	0.22	0.27

11.2.2 Field scale URL experiments

In the pursuit of development of technology and methodology for emplacement of overpacks with vitrified high level wastes and evaluation of the response of the host rock to the thermal field generated by emplaced waste, URL based experimentation was conducted in early 1980s in an underground gold mine located in South India (Narayan and Bajpai, 2007). This generic URL was operated for about 10 years. During this period, the experiment to simulate waste overpack-rock interaction was hosted at a depth of one kilometer. As the mine did not expose granites in any abandoned section, the experiment was placed in amphibolites rock. These massive amphibolites (~2700 Ma old) form part of a Greenstone belt with occurrence of diapiric granites. The experiment chamber was characterized for its geological and hydrogeological characteristics prior to installation of heaters replicating VWP. The amphibolites host rock at the experimental site is characterized by lower silica (48%) and higher concentration of Al_2O_3 (17%) than those in granite. It has a density of 3,000 kg/m^3, average compressive strength of 266 MPa, specific heat 920.7 J/kg-K, linear thermal expansion coefficient of 1.13×11^{-5}/K and thermal conductivity 2.261 W/m-K. These parameters indicate very good rock mass properties, which at times better than average parameters for granites.

11.3 Experimental Set Up

A main heater with a total of 6 heating elements each with 1.5 kW, in carbon steel casing of 2 m length with a diameter of 35.5cm was used to simulate Indian overpack. Besides, another eight auxiliary heaters with 3 m length, 101.6 mm diameter and 2.0 kW heat output were emplaced around the main heater. The main heater was placed in 368 mm diameter borehole drilled to a depth of 4.5 m, whereas auxiliary heaters were placed in 140-mm boreholes with depth of 5 m. These boreholes were drilled at a radial distance of 1 m from the main heater. Four extensometers were placed in 76-mm-diameter boreholes with depth of 6 m. Another 40 boreholes of 76 mm diameter and 6 m depth were drilled for installation of thermocouples. A total of five stress meters were also installed in boreholes with 38 mm diameter and 6m depth (Figures 11–3, 11–4, and 11–5). After switching on the heaters, temperature was initially recorded hourly, and subsequently daily.

The experiment was continued for 1,280 days. The evolution of thermal field was predicted using finite element modeling using indigenously developed computer codes. Figure 11–5 shows the layout of the experiment, with the main heater, auxiliary heaters, and boreholes where temperature, rock stress, and rock strain were measured. The predicted and measured temperatures were compared; the match was better in some boreholes than in others. Poor matches were attributed to instrument malfunction in some cases, and to inhomogeneity of the rock in others (Mathur el al 1998).

An example of a borehole with a good temperature match is in Figure 11–6. No significant degradation in confining rock mass has been recorded, and the pattern of movement of the heat front has been found in conformity with that predicted through numerical simulations.

As the experiment did not include clay based buffer layer, a new experiment is planned to test the complete engineered barrier system. As preparatory work, granite block scale experiment with central heater and clay layers is in progress to generate various design parameters for a field scale test.

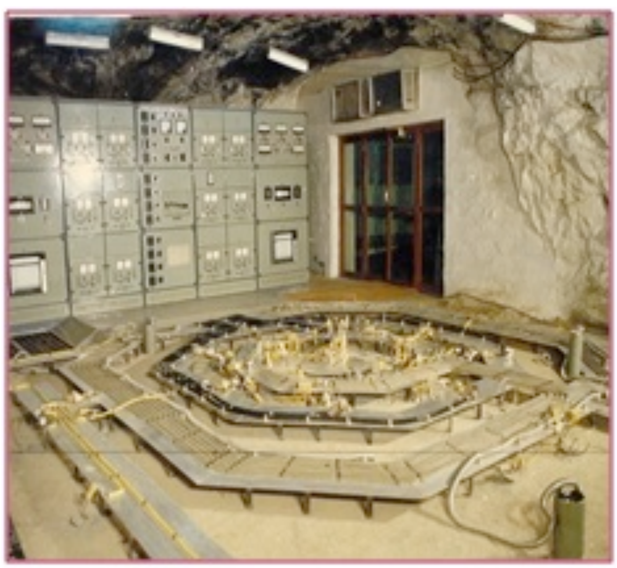

Figure 11–3. View of URL based Heater Test at 1 km depth

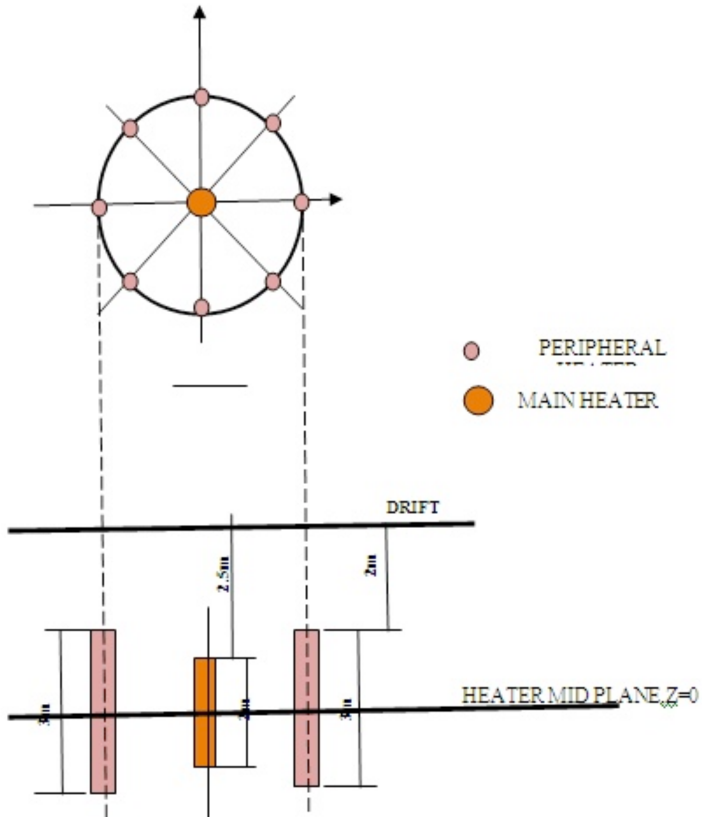

Figure 11–4. Plan and cross section of the experimental set up

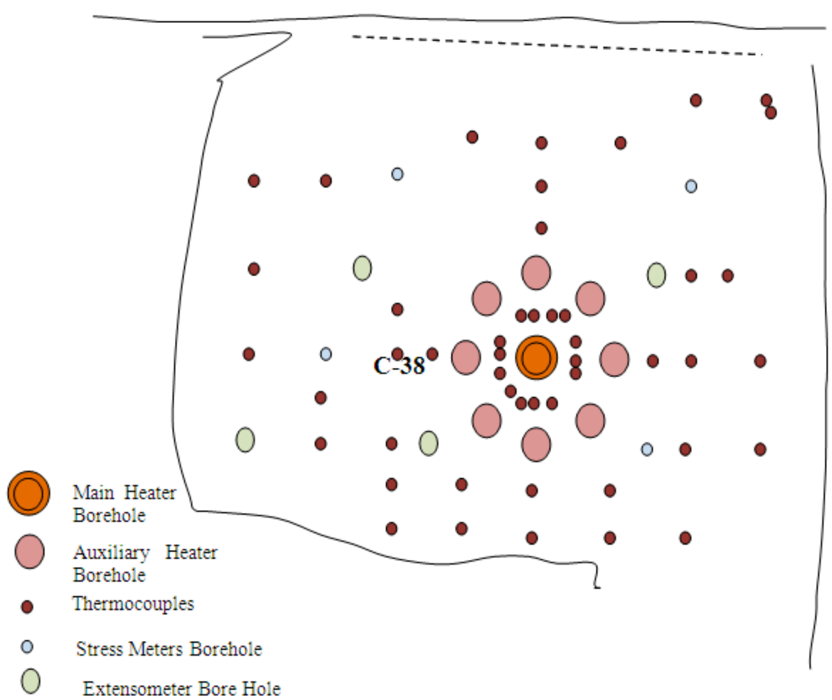

Figure 11–5. Plan view of the layout of various sensors, showing the location of C-38 borehole

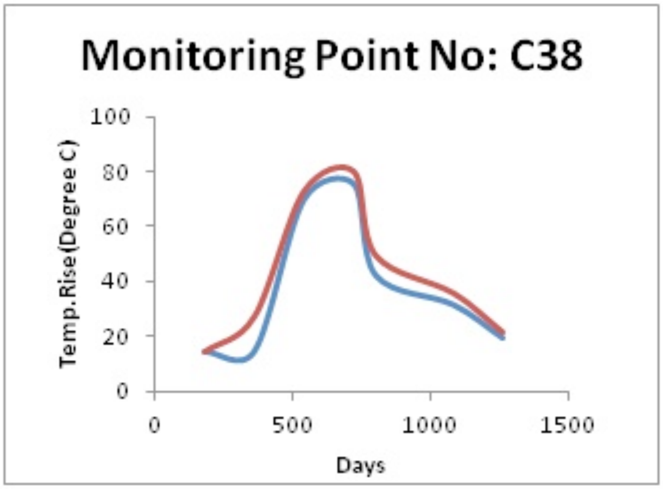

Figure 11–6. Measured (red) and predicted (blue) temperature profiles in borehole #C 38

11.4 Some Important Indian Host Rocks

11.4.1 Geological Conditions and Host Rock Evaluation Methodology

Peninsular India with an area about 3 million km^2 shows a wide range of rock types ranging in age from Achaean to recent in varying geographic, climatic, and seismic domains. The major geological units are represented by granite and associated crystalline, Deccan basalts, Proterozoic meta-sedimentary basins and Indo-Gangetic alluvium plains. Among these, granites and associated crystalline and basaltic rocks have offer good potential as a natural barrier capable of providing isolation and confinement of radioactive waste over stipulated time periods. Granites and basalts constitute about 20% (0.60 million km^2) and 16% (0.52 million km^2) of the total area of the country respectively.

The Himalayas, extending over more than 2000 km across the country on its northern periphery covering over 0.34 million km^{2}and constituting almost 14% of the total geographic area of the country, have been excluded from site selection because they are located in an active seismic zone with highly complex subduction-induced thrust-dominated structure and geology. Besides, occurrence of very rich faunal and floral resources and the presence of tributary streams of main Indian rivers further limit their suitability for hosting such repositories. Similarly, recent alluviums, popularly known as Indo-Gangetic alluvium, covering about 13% of the area constitutes some of the most fertile land with huge groundwater resources and dense population, and, therefore, excluded from consideration as host rock for deep geological repository in the Indian context. Proterozoic meta-sedimentary basins, occupying almost 7% of the country's geographic area, are characterized by occurrence of a number of mineral deposits and are also involved in complex basement-cover metasediments tectonics. Such basins, mostly confined in North West, Central India, South India and North East India, have therefore been kept at lower priority.

Basaltic rock, popularly known as Deccan Basalts in central and western India, represents one of the most spectacular volcanic provinces in the world. The fissure eruption related to passing of the Indian plate over plume produced a large number of volcanic flows spread over 0.52 million km^2 mainly in the states of Madhya Pradesh and Maharashtra, with a maximum thickness of about 2 km in Western Ghat along the west coast of India. They comprise number of subhorizontal lava flows. The maximum thickness of individual flows has been estimated on the order of 70-80 m. These flows are analogous to Hanford basalt flows in the USA, which was at one time considered as a host rock for geological repository (Saling and Fentiman, 2002). The occurrence of thicker flows in thickly populated or dense forest area, together with their preponderance in high rainfall area of Western Ghat limit their suitability as host rock for a DGR in India. Besides, their limited thickness, generally less than few meters, occurrence of columnar joints, higher hydraulic conductivity, especially along flow contacts, and presence of sedimentary interbeds, are other constraints. In terms of rock mechanical parameters, these basalts have sufficient strength (150-200 MPa), if unaltered, but the strength is anisotropic due to the presence of direction preferred anisotropy imparted by mineralogical laminations along flow contact. In view of these, the basalt has been assigned lower priority as suitable host rock in India and only laboratory based characterization to produce data on their mechanical, thermal, hydraulic and radiological characteristics is in underway. There is a plan to investigate these rocks at few locations in greater details in future to offer more flexibility in siting process.

The clay host rock of younger age, rich in highly sorptive smectite and of extremely low hydraulic conductivity, like those under consideration in Belgium, France and Switzerland, do not occur with sufficient spatial and depth persistence in India. Nevertheless, other argillaceous rocks like shales/argillites with significant thickness are known to occur in some of the Proterozoic basin viz. Vindhyan System of Central India and Cuddapah System of Andhra Pradesh. These are thinly laminated shales with late stage intrusions coupled with their brittle nature resulting in multiple sets of fractures. These hardened clays are often devoid of any swelling potential but at times contain occurrences of economic minerals. Among these shales, Shirbu shales of Vindhyan System in Central India and Tadpatri Shales of Cuddapah System in Andhra Pradesh show some degree of potential as host rocks. Laboratory based characterization of argillites from these two stratigraphic systems is in progress.

Granites especially those associated with relatively younger magmatism (500-700 Ma) like Malani Igneous Suite of northwestern India and few older granites (~2500 Ma) occurring in Central and Eastern India also constitute promising regions from the point of view of their suitability as host rock for deep geological repository. Among these Bundelkhand granites and Dongergarh granites are noteworthy. In India, granitic terrains spreading over 0.1 million km^2 meeting initial site requirements have been evaluated so far with a view to generate comprehensive data bases on host rock characteristics and socio-economic aspects (Mathur et al., 2001; Bajpai et al. 2006a, 2008b). A systematic host rock evaluation methodology has been developed and applied in stages on the larger regions occupied by granites. The main objective of these efforts has been the development of a database on host rock characteristics. The evaluation process is classified into three stages (Narayan and Bajpai, 2007, Bajpai, 2009a).

Stage I: In the initial stage, most of the information pertaining to geology, hydrogeology, structure, and aspects related to socio-political and economic factors is derived from secondary datasets, mainly involving reports published by national agencies such as the Geological Survey of India, Survey of India, Indian Metrology Department, National Land Use and Soil Survey Department, Groundwater Survey Departments, and National Geophysical Research Institute. This information is integrated in a GIS environment, preferably on 1:250,000 scale. The common GIS used during this stage are ILWIS and Arc View. During this integration ample use of satellite imagery such as LISS III and IV obtained from the Indian Remote Sensing Satellite (IRS) series is made to generate information on gap areas. This approach has yielded valuable information on distribution of seismic events, lineaments, major hydro-geological zones, and geological and structural details. Close evaluation of a series of such large scale maps helped in carving out three major regions occupied by granites in Northwest, Central, and Eastern India with areas of 15,000, 60,000, and 15,000 km^2 respectively. During this exercise, certain specific criteria were laid down to undertake relative assessments of these regions. These assessments, with the help of secondary datasets, identified a few regions of 100 to 150 km^2 area for further investigation.

Among criteria related to tectonics and instability, the location of the area within seismic zones in the Indian shield, along with the occurrences of major structural discontinuity like faults, shears, etc., have been assigned the greatest importance. Consequently only regions falling in seismic zones I and II, with maximum ground acceleration of 0.1 and 0.2 g, respectively, were considered. Among geological requirements, homogeneity of the rock mass with sufficient depth persistence (~1 km) and area extent (min. 4 km^2) have been considered as essential requirements. Besides, absence of intrusive, chemical durability, good mechanical and thermal strength and absence of mineral deposits have also been assigned important consideration. The criteria considered under hydrogeology mainly included absence of surface water bodies and high rain fall, lower recharge, and relief. Presence of sparse population, distance from industrial and commercial areas, better accessibility and favorable political climate constituted socio-economic criteria during this stage of investigations. Based on these criteria, 20 attributes were defined. The information thus generated, through secondary as well as primary data sets, involving application of satellite data and selected field checks, was subsequently transformed to numerical scores by assigning maximum score points and suitable weighting factor to individual attributes. Summation of total score points for all attributes for all the zones reveals their relative suitability.

Stage II: The second stage investigation mainly focused on the zones screened in Stage I investigations of large regions, and essentially involves data generation on 1:50,000 and 1:25,000 scales. These zones 100 to 150 km^2 in area were divided into a 5×5 km^2 grid and systematically evaluated by means of studies on geomorphology, soil thickness, rock types, weathering pattern, jointing, land use, etc. The attributes were again transformed to numerical scores to rank these zones. Sensitivity analysis was also performed to gauge the relative importance and impact of individual attributes. One of such zones was taken up for third stage investigations involving geological and structural mapping with the help of plane table and theodolite, pitting and trenching on 1:5000 scale. The focus of these mapping was on outcrop mapping, correlation studies, demarcation of heterogeneity like dikes, shear zones, and detail fracture mapping and short borehole drilling. These studies helped in demarcating an area of about few square kilometers wherein geological and topographic features are in conformity with the requirements. This zone has been explored with the help of ground geophysics as well as borehole geophysics. The core samples retrieved from array based boreholes has been subjected to intensive studies on fracture characteristics and other rock mass parameters like core loss, Rock Quality Designation, Rock Mass Rating, Q-system Rock Mass Classification, etc.

Stage III: The third stage investigations included detailed geological and structural surveys on 1:1,000 scale, and geophysical surveys using resistivity, gravity, magnetic etc., on the 50×50 m grid. The data obtained through such surveys were analyzed using Magmod, and three dimensional site models to a depth of one kilometer were produced. These models were validated based on the results of deep drilling to the 600 m depth. The representative cross section of the zones delineated based on the results of these investigations are depicted in Figures 11–7 and 11–8.

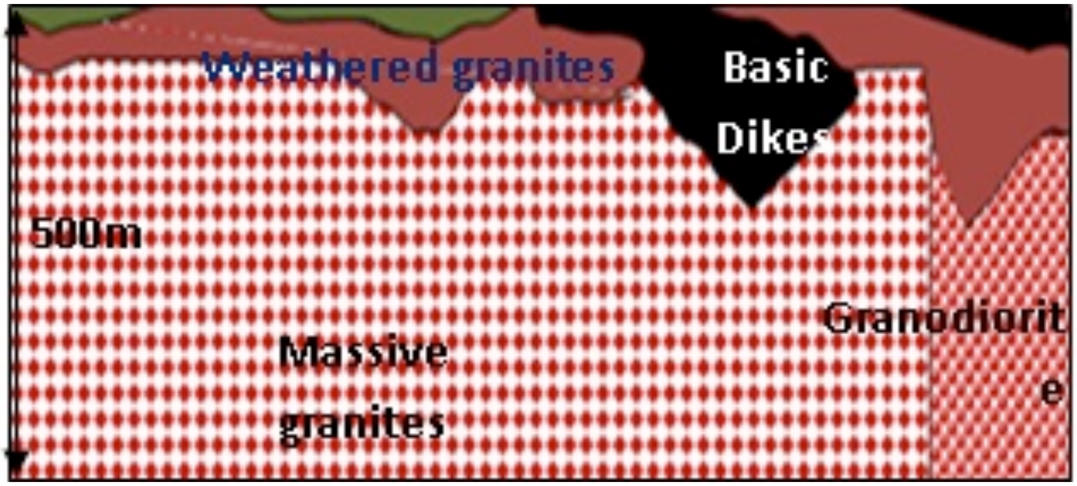

Figure 11–7. Geological cross section of a promising candidate site up to a depth of 500m constructed using geological-drilling-geophysical datasets

Figure 11–8. High rock mass quality granite (RQD 100%) from repository depth zone

Such systematic studies have yielded comprehensive parametric values on Indian granites for use in research and development activities involving URL based experiments. The exercise has also been useful in assessing the relative importance and sensitivity of various rock mass parameters and thus would be used in future site selection activities.

11.4.2 Laboratory Based Studies

Granites explored during the field investigations are mostly leucocratic, hard, compact, and massive, with specific gravity ranging from 2.44 to 2.7 g/cm^3. Generally these have well developed equigranular hypidiomorphic textures, often perthitic, with occasional porphyritic texture (Bajpai et al., 2006c). They show wide variation in their quartz content, ranging from 21 to 52%. Similarly, alkali feldspar ranges from 33 to 71%. Plagioclase is generally less, and the rock is of adamellite variety. Hornblende has been observed in some granite in the range of 0.81% to 3.69%. Biotite and chlorites occur as accessory minerals in these granites. The presence of interlocking perthitic texture in host rock gives them greater strength (150-200 MPa), suitable for underground excavation without any support, and the higher biotite content, coupled with secondary sericite and other clay minerals, accounts for higher sorption capacity of these granites for radionuclides in high level waste. Among granites, perthitic intergrowth-rich granites are also known for their very low hydraulic conductivities.

Variation in strength and hydraulic conductivities of these rocks has also been evaluated as function of temperature. The rocks show minor reduction in their strength up to a temperature of 65°C, (~5%), however strength improves by >10% due to sealing of cracks at 160°C.

Studies have been conducted on granite samples to evaluate the process of micro-crack formations in granites under the combined loading of stresses and heat. Maximum cracking is observed in samples heated at 100°C and 400°C (Figures 11–9 and 11–10). At 400°C, almost all grains cracked. Compressive strength of these rocks increases up to 100° C and then decrease slightly at 200° and again increases at 400° C

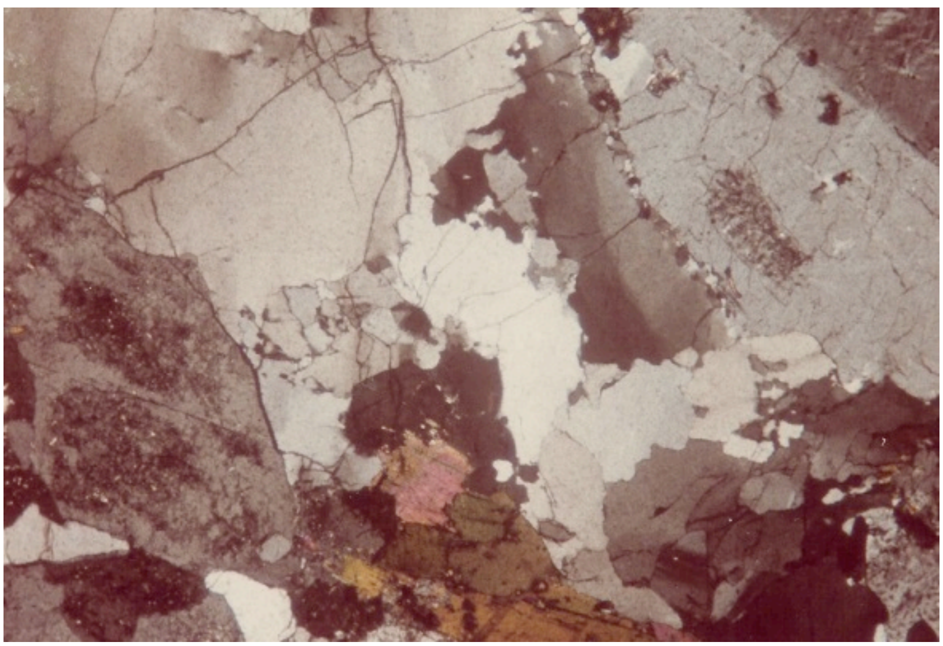

Figure 11–9. Photomicrograph showing microcracking in granites at 100^{0}C

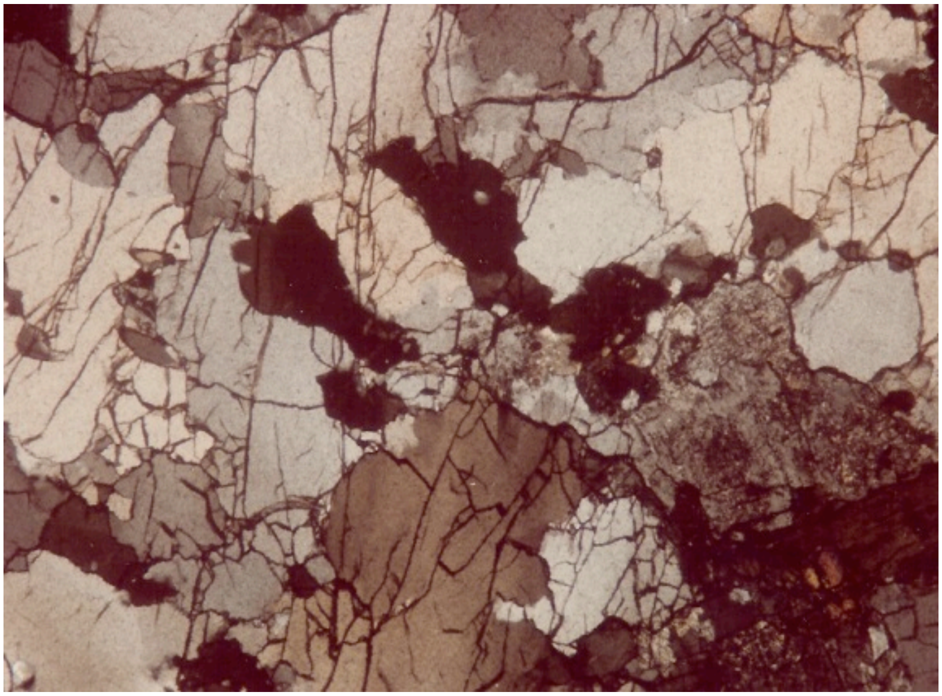

Figure 11–10. Photomicrograph showing intense microcracking in granites at 400^{0}C

The causes of micro-cracking include transformation of quartz to other polymorphs with resultant residual stresses. At 200°C crack density is higher and induced compressive stresses on neighboring cracks make crack propagation difficult. At 400°C, coalescence of micro cracks is the dominant phenomenon. Similarly, the permeability to N_2 gas (a surrogate for hydraulic conductivity), shows an increase up to 35% at 65°C and 315% at 160°C (Dwivedi et al., 2008). No significant acceleration in micro-crack formation has been observed up to 100°C (Goel et al., 2003, Singh et al., 2009). Host rocks have also been evaluated by acoustic measurements for assessment of damage they undergo when subjected to a combined load of stress as well as heating. The damage has been expressed as the ratio of volume of the intact to fractured portions in sample expressed as %. The stress loading conditions have been expressed as stages for a particular temperature. The studies reveal increase in damage with increasing stress loads with almost 57% damage under the stresses of 173 MPa even in the absence of any heating. However in a DGR stresses of these magnitudes are not expected. With increasing heating, the damage does not show much difference as compared to one observed in unheated samples (Figures 11–11 and 11–12). Rather, at times there is reduction in damage percentage at high temperature. This is due to the compact interlocking texture of host rock and sealing of micro-fractures due to differential expansions experienced by minerals. Table 11–2 shows the magnitude of damage in rocks under these experimental conditions.

Table 11–2. Damage in rocks due to stress and temperature

Deformation Stage	Stress (MPa)	Damage (%)	Temperature (°C)
Stage 0	0	14	Unheated
Stage 3	136	27	
Stage 4	154	43	
Stage 5	173	57	
Stage 0	0	21	200°C
Stage 3	175	32	
Stage 4	188	43	
Stage 5	197	42	
Stage 0	0	35	400°C
Stage 3	140	50	
Stage 4	159	50	
Stage 5	165.	50	

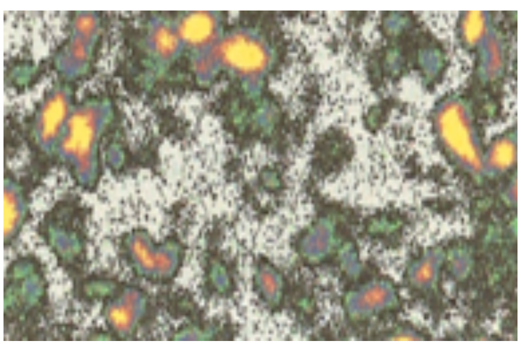

Figure 11–11. Micro-cracking (yellow) under room temperature and at 173MPa

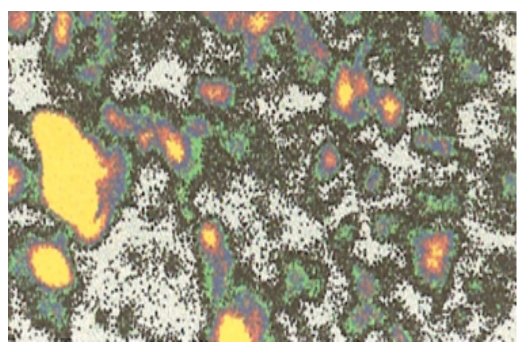

Figure 11–12. Micro-cracking (yellow) under room temperature and at 173MPa/200^{0}C

11.5 Development of Engineered Barrier System

Clay-based backfills and buffers are very important components of engineered barrier systems of a DGR. The conceptual design of a DGR would require about 0.3 to 0.5 million tonnes of good quality clays for use as backfill and buffers (Bajpai et al., 2007, 2008c). Therefore dedicated research and development activities are in progress to identify and characterize suitable clay deposits. India has good reserves of swelling clays located in various parts of the country, amounting to almost 50 million tonnes. Many deposits have been located by Geological Survey of India and State Department of Geology and Mining. Among these, deposits of Rajasthan, Gujarat, Jammu and Kashmir, and Uttar Pradesh are important in terms of tonnage and quality. Indian clays differ from most clays under evaluation worldwide on account of their very high iron content (~5-10%).

Currently, Na-smectite-rich clays of Barmer, Rajasthan, are being characterized in detail. Rich deposits of about 20 million tonnes have been identified in this area. These clays contain mainly Na-montmorillonite (50-60%) and quartz along with small amount of kaolinite, calcite, pyrite and organic matter (Figures 11–13 and 11–14). Clays from these deposits contain more than 84 % clay-size particles. Free swelling volume ranges from 20 to 30 mL/g and is directly proportional to montmorillonite (smectite) content of these clays (50-65 %). The temperature field around disposed waste overpack in DGR for periods of 1, 2, 5, 15, 20, 50, 100, and 500 years is not expected to exceed 100°C. Temperature in buffers will also not exceed 100°C. The free swelling volume and other key geotechnical and sorption parameters of these clays remain unaffected up to a temperature of 150°C, hence indicating good suitability of these clays as buffers. This in turn also rules out temperature-induced alteration of montmorillonite to non-swelling clay. Based on the mineralogical composition, due to the presence of calcite and gypsum, these clays are immune to the adverse impact of pyrite. The presence of biotite and iron oxides add to their enhanced sorption of radionuclides of concern. Thus, these clays possess very suitable mineralogy to serve as backfill and buffers in a DGR (Table 11–3).

Figure 11–13. Bentonite clay deposits in NW India overlying ferruginous layers

Table 11–3. Major oxide compositions of some important clays

Oxides	Barmer Clay (%)	Kutch Clay (%)	Mineral
SiO_2	57.2	56.7	Quartz
Al_2O_3	16.4	18.6	Feldspar, smectite
Fe_2O_3	11.3	12.2	Pyrite, magnetite
CaO	2.1	1.6	Calcite, gypsum
MgO	3.4	1.2	Biotite
Na_2O	0.9	0.9	Na smectite, halite
K_2O	1.4	0.5	Orthoclase

Figure 11–14. Na-smectite clay from Barmer, Rajasthan

The buffers proposed for a DGR are generally in the form of highly compacted clay bricks. The hydraulic conductivity of these bricks is thus another important factor. As buffers are planned to be used as admixture of sands and clay, studies have also been carried out on such admixtures. The compaction and other geotechnical properties of the admixture of 50% bentonite clays and 50% sand have been evaluated. This admixture is characterized by distinctly high Maximum Dry Density (MDD) of more than 1.67 g/cm^3 and low Optimum Moisture Content (OMC) of less than 21%. The high MDD and low OMC are expected to impart a greater swell potential to this admixture. The large swelling potential of bentonite clays and suitable compaction characteristics apparently make it promising to use clay materials as buffer in geological repository (Table 11–4). The focus area of research in near future would be centered on the evaluation of impact of the mineralogical composition on evolution of near field geochemistry, propogation of the thermal front across the disposed waste canister, buildup of pore pressures, and thermal degradation of barrier materials.

Table 11–4. Geotechnical properties of some clay-sand admixtures

Sample	Maximum Dry Density (Mg/m^3)	Optimum Moisture Content (%)	Swelling Potential (%)	Swelling Pressure (MPa)	Saturated Hydraulic Conductivity (cm/s)
50% Barmer Bentonite + 50% sand	1.85	14.7	108	0.84	5.42E-11
50% Barmer Bentonite + 50% sand	1.94	11.5	79	0.78	6.65E-11
50% Wandh Bentonite + 50% sand	1.70	15.0	87	0.86	9.05E-11
50% Hamla Bentonite + 50% sand	1.67	15.2	63	0.71	1.20E-10
50% Pundi Bentonite + 50% sand	1.74	14.7	55	0.93	6.24E-11

11.6 Natural Analogues

Natural analogues generally refer to naturally occurring material or processes that replicate in total or in part a material used in DGR or a process that may be operational in DGR subsequent to emplacement of radioactive waste. The studies have gained lot of attention in recent past as a strong tool to demonstrate the performance of various components of a DGR in the distant future. Studies on natural analogues of waste glasses, backfill and buffer clays, concretes, metal and DGR-related near field geochemical processes thus constitute an important area within Indian Geological Repository Program. A large number of such analogues have been discovered in India. Some of natural analogues identified are discussed in brief below (Bajpai, 2009b).

Shock Basaltic Glasses: Basaltic glasses produced as a result of meteorite impact about 50,000 years ago, in the district of Maharashtra have been studied to demonstrate their analogy with Na borosilicate glasses being used in India for vitrification of radioactive waste. These glasses show alteration phases such as analcime ($NaAlSi_2O_6.H_2O$), saponite, Mg-rich clay, gyrolite (Ca_2SiO_7 $(OH)_2H_2O$), tobermorite (Ca_5 $(OH)_2Si_6O_{16}H_2O$) and zeolite. Thus they show remarkable similarity in their alteration mechanism, product and alteration layer thickness with those observed with Na Borosilicate glasses (Bajpai et al., 2005a; Bajpai, 2009b; Shrivastava et al., 2008). Figures 11–15, 11–16 and 11–17 show some of the important features of these glasses under experimental conditions.

Obsidian Glasses: Obsidian glasses (65 Ma) form Osham hills in Haryana have been experimentally altered to study the sequence of secondary mineral formation in a Parr Reactor. The studies reveal the formation of quartz, smectite, analcime and other clay minerals similar to those formed in experimentally altered Na borosilicate glasses (Nishi Rani et al., 2010a, 2010b).

High Cs thermal springs: Thermal springs associated with the Puga geothermal field in Ladakh district of Jammu and Kashmir has a surface temperature of 56°C and a discharge rate of 15 L/s, a very high reservoir temperature (180-260°C), and flows along fault planes in granites. Thermal water contains 30 mg/L of Cs and the topsoil cover around these springs has been estimated to contain about 400 tonnes of Cs. It is also established that about 10 tonnes of Cs is being discharged into the environment per year by these springs. The presence of hot groundwater, Cs source at depth, and flowing water along faults all together mimic geochemical processes of repository DGR (Bajpai et al., 2006d).

U-bearing thermal springs in granites: A group of hot springs with high uranium content (~90 ppb) draining uraninite minerals located at about 250 m depth in Garo Hills district of Meghalaya have been studied to understand processes responsible for migration of uranium and its precipitation along fracture-filling minerals. Studies here revealed that despite the presence of fast-flowing hot water and a highly permeable path in the form of fault zone, uranium has not migrated more than 85 m in the past few million years (Bajpai et al., 2005b).

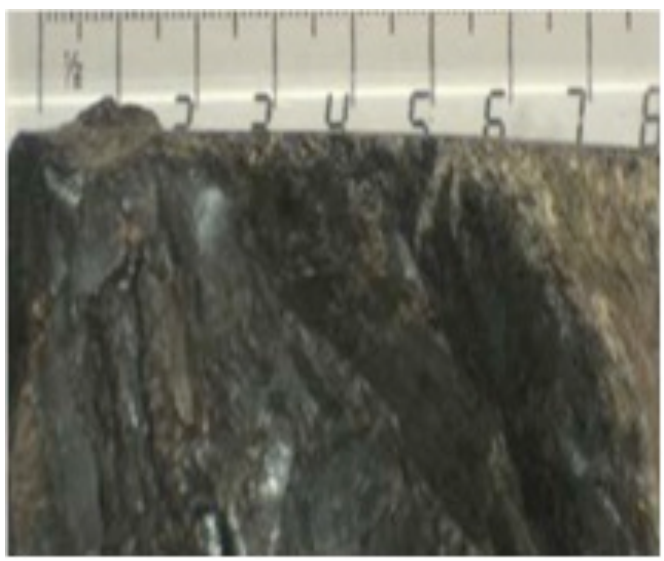

Figure 11–15. Natural basaltic glasses exposed in Osham Hills NW India

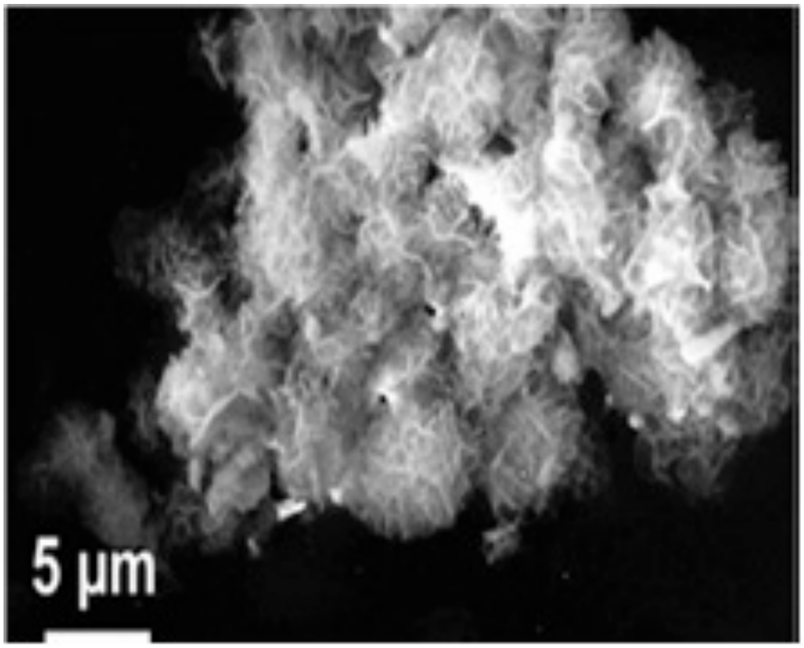

Figure 11–16. SEM image of the experimentally altered basaltic glasses

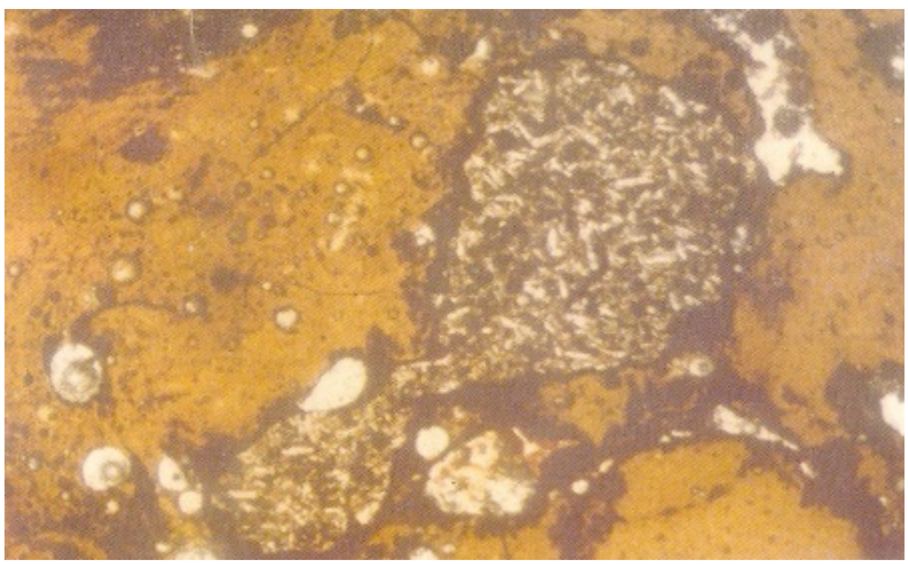

Figure 11–17. Palagonite formation as alteration product of basaltic glasses

To demonstrate capability of such host rocks in providing very long term protection against radiation hazards, natural analogues studies have also been carried out, which demonstrated good potential of basalts and granites in arresting movement of isotopes of uranium and cesium.

11.7 Conclusions

A well defined program of the development of a Deep Geological Repository for high level radioactive waste has been in progress in India for the last three decades. The program is mainly based on field and laboratory studies, which focus on the development of comprehensive databases of suitable types of host rocks available in India. The rock mass parameters coupled with waste characteristics provide inputs for the design of DGR and the analysis using commercially and specifically developed computer codes. A dedicated programme is in progress toward setting up an underground research facility for the development of methodology and technology related to site characterization, construction of disposal tunnels and pits, sealing and grouting of groundwater conducting zones, as well as the design of an engineered barrier system, waste transfer and emplacement technology, etc.

11.8 Acknowledgments

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Chapter 12

Progress in the Geological Disposal Program in Japan

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ABSTRACT: The Second Progress Report on Research and Development for the Geological Disposal of High-level Radioactive Waste in Japan (H12 Report) demonstrated the feasibility of safe and technically reliable geological disposal. On the basis of this report, the Act on Final Disposal of Specified Radioactive Waste was enacted and Nuclear Waste Management Organization of Japan (NUMO) was established in 2000 to carry out the geological disposal.

Since the Great East Japan Earthquake and the subsequent tsunami and reactor meltdowns occurred in 2011, there has been increased public concern regarding the potential for major natural hazards to cause accidents at nuclear facilities. The Government re-evaluated the technical feasibility of the geological disposal program based on state-of-the-art geosciences and implementation processes. Following the re-evaluation, the Government concluded to further promote the geological disposal program and revise the Basic Policy for Final Disposal as a Cabinet decision in May 2015 to include "Scientifically Preferable Areas" as a new criteria to the siting process in addition to the previous approach solely based on open solicitation.

NUMO and relevant R&D organizations such as JAEA and RWMC have been carrying out R&D activities to increase technical reliability for geological disposal. NUMO has started to develop a generic safety case that builds on both worldwide technical progress and also increased understanding of relevant geological conditions in Japan. The safety case will incorporate recommendations in key technical areas from the H12 Report.

12.1 Introduction

The safe management of radioactive waste, particularly vitrified high-level waste (HLW) from fuel reprocessing, has been a major concern in Japan. In 1999, the Second Progress Report on Research and Development for the Geological Disposal of High-level Radioactive Waste (HLW) in Japan (hereinafter referred to as "H12 Report") (JNC, 2000) was published by the Japan Nuclear Cycle Development

Institute (now the Japan Atomic Energy Agency – JAEA), integrating the results from 25 years of R&D activities (Masuda and Kawata, 2001). This demonstrated the feasibility of safe and technically reliable geological disposal based on a generic study. On the basis of the H12 Report, the Act on Final Disposal of Specified Radioactive Waste ("the Final Disposal Act" or "the Act", in short) came into force and Nuclear Waste Management Organization of Japan (NUMO) was established in 2000. The Final Disposal Act specifies a step-wise siting process consisting of the nomination of preliminary investigation areas (PIAs) based on area-specific literature survey, a selection of detailed investigation areas (DIA's) by surface-based investigation, and then selection of the repository construction area by detailed investigation that includes characterization with the underground test facility. Low-level waste generated from reprocessing of spent nuclear fuel and mixed-oxide fuel fabrication (termed "TRU waste" in Japan) was also included in the inventory to be considered through an amendment of the Act in 2007 based on the generic feasibility study of TRU waste disposal conducted by JNC and the Federation of Electric Power Companies (FEPC) published in 2005 (JNC and FEPC 2005).

NUMO initiated the siting process by an open solicitation of volunteer municipalities in 2002 (NUMO 2002), with development of siting factors for selecting PIAs (NUMO 2004a) and a Repository Concepts Catalogue (NUMO 2004b). More details on the evolution of the Japanese HLW program in its early stage of implementation of geological disposal has been described in the *Fourth Worldwide Review* (Kitayama, et al., 2006).

There was only one case where a mayor officially applied for the literature survey, but the application was withdrawn soon thereafter due to objections from local residents. After this case, in late 2007, an additional process where the Government can propose potential candidate sites was introduced, running in parallel to volunteering in order to advance the site selection. So far, however, no volunteer municipality has appeared and no candidate host rock type has been specified.

Since the 2011 Great Tohoku earthquake and resulting tsunami and reactor meltdowns, there has been increased public concern regarding nuclear issues and the potential of major natural hazards to cause accidents at nuclear facilities. In response to this, and to help rebuild the public confidence required to site a repository, the Government re-evaluated the geological disposal program in terms of technical feasibility based on state-of-the-art geosciences and implementation process.

In parallel with the siting process, NUMO has been developing key technologies required for the safe implementation of the geological disposal project in close collaboration with relevant R&D organizations, such as JAEA, Radioactive Waste Management Funding and Research Center (RWMC), etc.

This paper summarizes the progress and evolution of the geological disposal program in Japan since the *Forth Worldwide Review* (Kitayama et al. 2006).

12.2 National Nuclear Policy and Legal Framework

12.2.1 Development of new policy

The Great East Japan Earthquake and the Fukushima-Daiichi nuclear power plant accident (1F accident) in 2011 increased nationwide concerns about the safety and reliability of nuclear facilities, including the feasibility of safe geological disposal in Japan. The new administration established in December 2012 has been formulating a robust and sustainable energy policy from a range of perspectives, including assurance of stable energy supply and reduction of energy cost. In April 2014, it adopted the new Strategic Energy Plan that outlines Japan's energy policy for achieving stable, economic and environmentally friendly energy supplies. The Plan defines nuclear power as an important base-load power source, and maintains the nuclear fuel cycle program as same as before.

As part of the activities for formulating the Plan, multidisciplinary Working Groups were established in 2013 by the Government (Ministry of Economy, Trade and Industry, "METI") to re-evaluate both the technical feasibility of geological disposal and the implementation process. The Working Groups concluded that:

- Potentially favorable geological environments for geological disposal exist and selecting them is feasible, even considering the latest geoscientific knowledge.
- Periodic re-evaluation of safety based on the latest knowledge and communicating this with the general public are inevitable.

Based on the Working Groups' guidance, the Strategic Energy Plan concluded that geological disposal of HLW is the issue that should be addressed by the current generation who has benefitted from nuclear power generation. Actions towards the extended storage of HLW ensuring reversibility and retrievability so that future generations could select a better solution should be immediately implemented.

The Plan also concluded that the Government will take the initiative to find proper solutions toward the final disposal of HLW. It requires the Government to promote site selection process with nationwide understanding by identifying "Scientifically Preferable Areas". As a result of this Government commitment, the siting process deadlock is expected to be broken.

The essential outcomes of the Working Groups were finally reflected in the amendment of the "Basic Policy for Final Disposal" ("Basic Policy" in short) in 2015 as the following new policies on geological disposal program:

- Geological disposal is clearly identified as the final solution for HLW. For development of a geological repository, reversibility and retrievability should be included to allow future generations to select alternative options. In parallel, the feasibility and safety of the geological disposal will be iteratively reevaluated, based on the latest scientific knowledge, continuing relevant R&D activities.
- Based on the latest scientific knowledge, the Government will identify areas in the country preferable for promoting the site selection process, i.e. "Scientifically Preferable Areas", and will promote activities toward better understanding of the geological disposal in identified areas and encourage the acceptance of the first stage of the siting process.
- The Government will establish a scheme for consensus building of interested local municipalities using the same local/regional involvement approaches in some foreign programs.

The Basic Policy also requires the Japan Atomic Energy Commission (JAEC) to review the implementation activities by the Government and NUMO, and for the Nuclear Regulatory Authority (NRA, see next section) to be actively involved in the siting process.

The identification of Scientifically Preferable Areas by the Government will be carried out in advance of the three-step siting process defined in the Act as illustrated in Figure 12–1. The Working Groups are discussing the requirements and relevant criteria to be applied for identification of such areas from both geoscientific (igneous activity, fault activity, etc.) and social scientific (environmental preservation, land use, etc.) viewpoints. Although still under discussion, coastal areas including those underwater and islands have been suggested by the Working Groups as a Scientifically Preferable Area after previously excluding those areas whose long term stability of the geological environment could be significantly affected by the natural processes and events such as volcanoes and active faults.

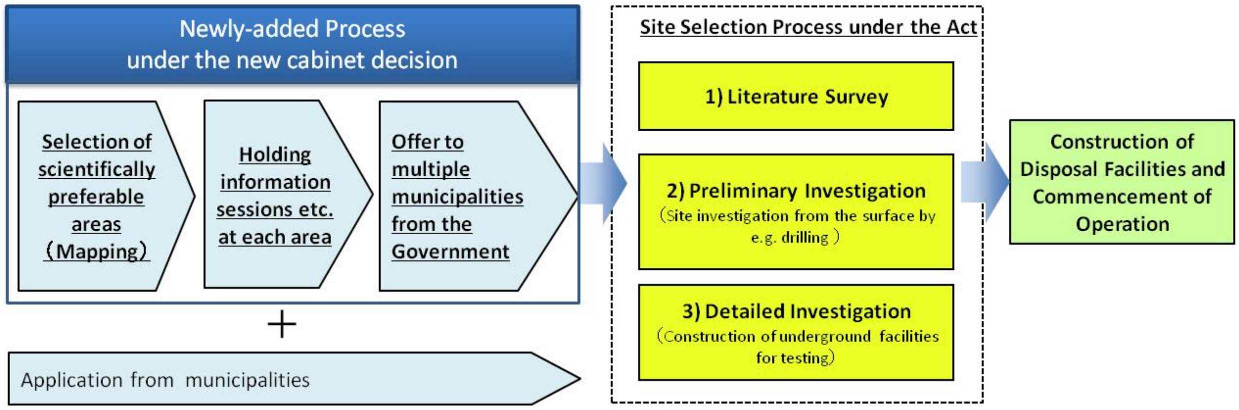

Figure 12–1. Process with identification of "Scientifically Preferable Areas"

12.2.2 Safety regulation

After the 1F accident, the Nuclear Regulatory Authority (NRA) was established in 2012 as a new regulatory organization merging the Nuclear Safety Commission and Nuclear and Industrial Safety Agency of METI. On December 18, 2013 new regulatory standards were enforced regarding radioactive waste disposal facilities for near surface pit disposal and trench disposal. Regulation for intermediate depth disposal was discussed by the 2015 NRA sub-committee which took into consideration radioactive waste generated from decommissioning of NPP. The discussion was focused on classification of radioactive waste to be disposed for an intermediate depth repository, requirements for repository design and institutional control, assessment scenarios, and dose criteria for safety assessment. Although related, the discussion on safety regulation for HLW geological disposal has not yet started. In addition to conventional radioactive waste, the 1F accident-derived radioactive waste included nuclear fuel debris; its future management is now one of the highest priority issues from a regulatory standpoint.

12.3 Siting Activities and Public Relations

Since its establishment, NUMO has made efforts to continue technical R&D in order to increase reliability and practicality for safe geological repository (see next section) and to increase public relations (PR) and involvement activities to promote awareness of the disposal project's importance and safety to date. For its immediate goal of launching initial literature survey to explore suitable sites for constructing a repository, NUMO held an open solicitation to call volunteer municipalities and has been actively promoting public dialogues in collaboration with the Government, electric power companies, and other related organizations (Kitayama et al. 2006).

Initially, NUMO activities focused on increasing public awareness of geological disposal projects through organization of open fora and roundtable meetings in parallel with nationwide advertising on television and newspapers. In January 2007, Toyo town in Kochi prefecture officially indicated its interest in NUMO's repository site selection program and applied for the literature survey, but withdrew it in April of the same year due to strong opposition. The lessons learned from this experience were that a majority of the public felt insecure in the safe geological disposal of high-level radioactive waste and that more efforts were needed to obtain better understanding of the importance and safety of the project. As a result NUMO has focused on more face-to-face dialogues in grass-root level discussion meetings and workshops.

After the Great East Japan Earthquake and associated 1F accident in 2011, the nuclear industry lost public trust, which significantly affected the ability for NUMO to continue its PR activities. Following the recommendation by METI that NUMO's PR activities be thoroughly reevaluated, NUMO organized an advisory committee that consists of outside experts. Based on the committee's guidance, in 2013 NUMO formulated and implemented its new activity plan which includes:

- Face-to-face dialogues with the public through open symposia in nationwide major cities (Figure 12–2) and by visiting local communities with its communication vehicle "Geo Mirai" (Figure 12–3), in which mockups of engineered barriers and a 3D-animation mini-theater are installed inside to explain the evolution of the geological disposal project and repository safety functions, with 'fun' hands-on bentonite clay experiment;
- Support for the development of geological disposal educational programs and materials for school teachers;
- Assistance for college debate classes on the geological disposal of high-level radioactive waste (Figure 12–4).

Figure 12–2. Open symposia in major cities

Figure 12–3. Communication vehicle “Geo Mirai"


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Figure 12–4. Assistance for college debate classes

Since 2013, NUMO has been carrying out nationwide consensus-building activities on the importance and safety of geological disposal project, applying a two-track approach.

Symposia are being held in major cities, each consisting of 3 components: 1) providing information from NUMO; 2) panel discussion among local residents and college students, a journalist, a coordinator, and NUMO staff to explain technical and project-related issues; 3) Q&A session for NUMO staff to answer all questions from the audience. Local panelists are mainly women and young university students, who are important targets to improve understanding.

During the symposia held at 29 cities in 2014, large numbers of questions were asked by the audience, which were considered to represent a representative sample of the Japanese public, and classified as shown in Figure 12–5.

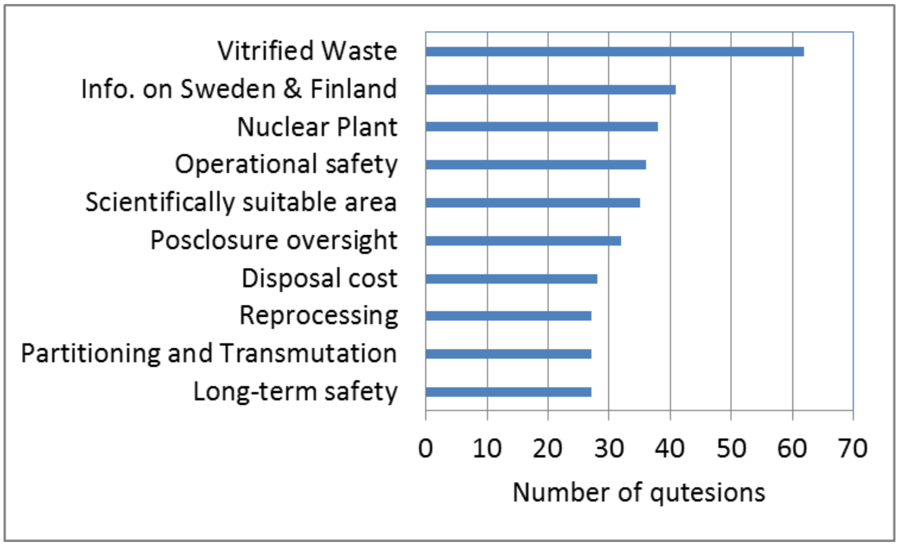

Figure 12–5. Classified questions from the symposia at 29 cities in Japan (Kaku, et al., 2015b).

The most frequently asked questions were on the vitrified waste, followed by those on information about more advanced Finnish and Swedish radioactive waste disposal programs, Japanese nuclear energy policy, operational safety, Scientifically Preferable Areas, post-closure oversight, disposal cost and post-closure safety. An analysis of the questions indicates that i) the public feels more comfortable when they understand the geological disposal issue within a wider framework of nuclear program, ii) the public are very interested in the international advancement of disposal programs when compared with Japan, and iii) the importance of operational safety is clearly highlighted as high as post-closure long-term safety. These observations have been reflected not only further to communicates with general public, but technical R&D program to increase reliability of disposal technology (see next section).

Along with the progress in new policy discussions mentioned in the previous section, a series of joint Agency of Natural Resource and Energy (ANRE) of METI and NUMO nationwide symposia have been held at major cities in Japan since 2015 to promote both public dialogue on new Basic Policy and technical discussion from the Working Groups on the selection of Scientifically Preferable Areas (see ANRE/METI website (http://www.chisou-sympo.jp/) and NUMO website (http://www.numo.or.jp/chisou-sympo/)). Open workshops are also being held by ANRE/METI for in-depth discussion with the general public. Those activities are aimed at increased understanding of the need for geological disposal and to provide a social environment to support and pay respect to municipalities that are interested in hosting the repository.

NUMO has been conducting nationwide annual public opinion surveys since 2003 to analyse the effects of its PR activities . It is useful data for observing trends in public awareness of the geological disposal project. Each year, NUMO has conducted an internet survey of 2,000 adults living in the country, consisting of about 200 adults of different age and gender from 10 different regions.

Figure 12–6 presents the results of the survey on awareness of geological disposal project from 2003 to 2014. About 50% of people answered “Don’t know” of the geological disposal project up until 2005 but, in 2006, it decreased to 16.5% which continued at similar level to 2014. It could be the effect of the public communication efforts, which expanded beginning in spring 2005.

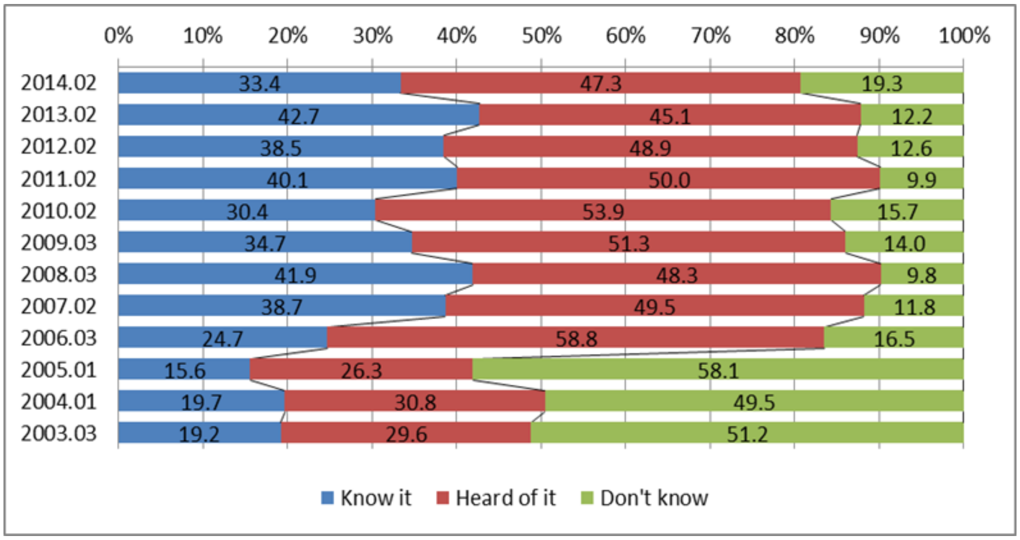

Figure 12–6. Survey results of the awareness of the geological disposal project (March 2003 - February 2014) (Kaku et al. 2015b).

NUMO also carried out the survey on the awareness of HLW disposal using television, newspaper, internet, magazine and a PR brochure. According to the survey, between 2002 and 2006, television and newspaper channels showed higher awareness by 60-80%, compared with other channels. These activities are summarized in detail by Kaku et al. (2015a; 2015b).

12.4 Technical R&D Programs

In cooperation with NUMO, relevant R&D organizations such as JAEA, RWMC (Radioactive Waste Management Funding and Research Center), CRIEPI (Central Research Institute of Electric Power Industry), AIST (National Institute of Advanced Industrial Science and Technology) and NIRS (National Institute of Radiological Sciences) are promoting R&D activities to support government policy-making and formulation of regulations and implementation of the geological disposal program under a "Coordination Executive" which is managed by ANRE/METI. These R&D activities are conducted in accordance with the comprehensive R&D program which is provided and revised by the Coordination Executive every 5 years through the peer review of outcome from the previous phase by independent experts. As a specific feature of the current R&D program (fiscal year 2013-2017), it has been organized to conduct studies not only of key issues at present for HLW and TRU waste disposal, but for the development of advanced technology to provide more reliable options, e.g. new waste form, overpack material, etc., and preliminary feasibility assessment of spent nuclear fuel direct disposal.

In December 2012, the JAEC published "Research and Development on Nuclear Power in the Future Should Be (Statement)" and pointed out a need of R&D on both disposal of vitrified waste and direct disposal of spent fuel in Japan. The new Strategic Energy Plan has also mentioned a need of R&D on direct disposal of spent fuel as an alternative option to maintain a wide range of options for disposal. The options have led to new R&D activities on direct disposal of spent fuel in Japan at the initiative of the "Coordination Executive" mentioned above. JAEA summarized and released its first progress report

December 2015. The R&D on technical issues stemming from the first progress report are being carried out.

In this section the R&D activities by NUMO and relevant organizations (focusing on JAEA and RWMC) since last *Worldwide Review* are reviewed.

12.4.1 NUMO Activities

Following public concerns in the wake of the 1F accident and a move by the Government to more strongly support moving forward with siting a geological repository, NUMO has started to develop a comprehensive safety case that builds on both worldwide technical progress and also increased understanding of relevant geological conditions in Japan (Fujihara et al. 2015; Fujiyama et al. 2015). The following is an overview of the NUMO safety case under development.

The NUMO Safety Case has been extended in key areas from the H12 Report, which could be recognized as the first comprehensive safety case in Japan. It includes the assessment of extreme geological events during long-term repository evolution, developing design options for increased flexibility, widening discussion of both operational and post-closure safety, and scenario development based on risk-informed approaches. In cooperation with relevant organizations NUMO has been developing key technologies required for the safe implementation of the geological disposal project since its establishment and published a comprehensive technical review in 2011 (NUMO 2011), which is being revised to include the new safety case. This safety case will demonstrate flexibility to respond to a siting process that is still focused on volunteering and thus reflects the need to be able to compare alternative designs or siting options, which requires a performance assessment approach based on realism rather than conservatism.

Safety strategy as an implementation policy is set up in the NUMO Safety Case, considering the current constraints that include legal requirements and the current state of the Japanese disposal program, i.e. candidate sites have not been specified yet, safety regulations are still under development, as well as the basic policies and the premises mentioned above. Management strategy, siting and design strategy, and safety assessment strategy are set up as components of the safety strategy.

When neither sites nor any host rock types are yet specified, geological and hydrogeological models of the types of potential host rock environments are developed by interpreting and synthesizing the latest geoscientific information. Repository design and safety assessment are thus performed for those geological models, which provide underpinning evidence to demonstrate the technical feasibility and the safety for the various types of Japanese geological environments.

Taking into account international norms and the "Final Disposal Plan" of the Act published in 2008, the key points of the disposal concept are as follows:

- A multi-barrier system consisting of engineered barriers and the geological barrier;
- Mined geological repository;
- Required capacity of the facility is more than 40,000 canisters of vitrified waste for HLW and more than 19,000 m^3 for TRU waste.

A suitable repository site will be identified in three stages as specified in the Act: literature survey, preliminary investigation and detailed investigation. A logical, comprehensive, and progressive basis for the siting process has been developed, which involves explicit exclusion siting factors that are set on the basis of geological attributes associated with the dynamic tectonic setting in Japan. NUMO needs to prepare and synthesize the reliable investigation and evaluation methodologies to select suitable repository sites where the key safety functions (isolation and containment) of the host geological environment are secured for a long period of time. In the NUMO Safety Case, feasibility of selecting a suitable repository site in Japan is assessed. Fundamental information and basic concepts for staged repository site selection will be presented, including safety functions and factors affecting safety. Advanced methodologies for precluding the potential impacts of natural disruptive events and processes have been provided. Key concepts, technical knowledge bases, and basic methodology for geosynthesis of relevant information into representative spatial and temporal models of site evolution are also being documented. The illustrative site descriptive models (SDMs) have been developed for subsequent repository design and safety assessment following the process outlined in Figure 12–7 (Ota et al. 2015).

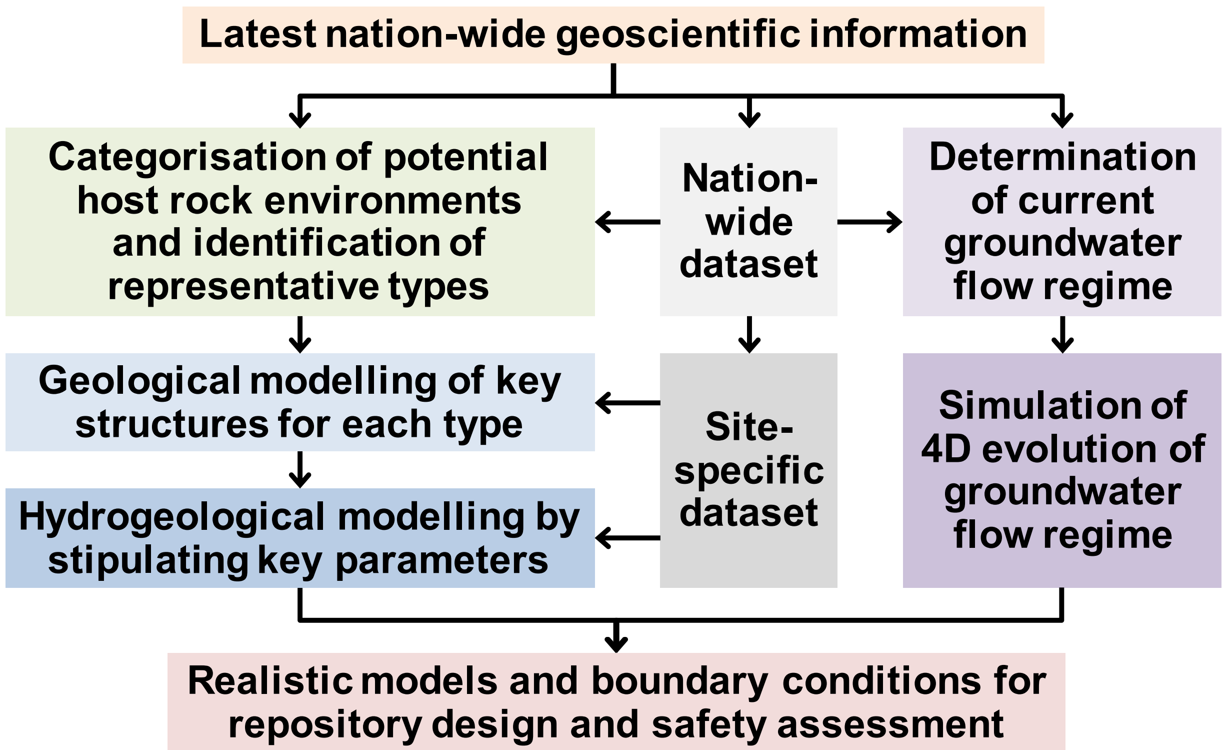

Figure 12–7. Flowchart of the process for developing potential host rock environment models (Ota et al. 2015)

Facilities that fulfill the required safety functions on the basis of SDMs will be designed, taking into account expected future regulations and constraints set by environmental impacts on the site. In the later siting stages, NUMO will:

- develop designs in a stepwise manner based on the SDM developed at each stage, which will be used to judge feasibility of implementation;
- develop technologies of construction, operation and closure on the basis of specific repository designs and SDMs, which become more detailed as the project progresses;
- verify and validate such technologies through demonstration tests to assure that they are mature prior to initiating construction and operation.

Design methodologies should be developed to maintain flexibility for a range of potential geological conditions encountered in Japan. In the NUMO Safety Case, alternative repository concepts are presented, which are applicable for a wide range of potential geological conditions. The repositories should be technically feasible to construct and fulfill the safety functions required to isolate and contain radioactive nuclides in any selected geological formation. The methodology is then demonstrated by carrying out a full repository design study, tailored to an illustrative SDM in this safety case. The engineering feasibility of construction, operation, and closure of the repository is also evaluated based on techniques demonstrated in domestic and international underground laboratories and related R&D facilities. The design system conducting these procedures is shown in Figure 12–8.

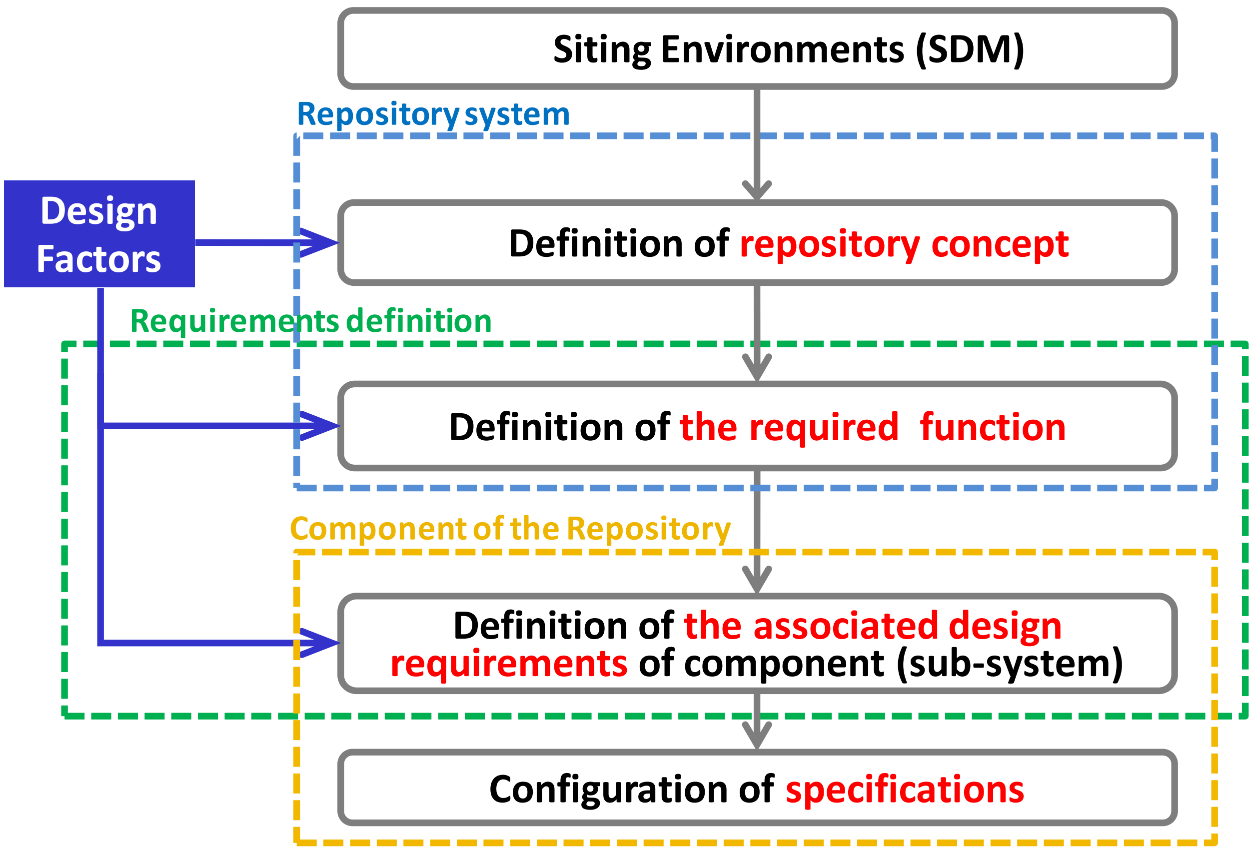

Figure 12–8. The overview of repository design system (Fujiyama et al. 2015)

The outline of repository design tailored to the geological models is shown by Kubota et al. (2015).

Both pre- and post- closure safety of the designed repository are assessed in the Safety Case within the context set by a specific SDM. In the siting stages, this will proceed in an iterative manner and the resulting output will support decisions made at the end of each stage from a safety perspective. The technology required to assess scenario development, modeling, database development, interpretation, etc. will be maintained at the current state-of-the-art.

Prior to closure, radiological protection for local residents and workers under repository operation needs to be ensured. As a result of the 1F accident, Japanese regulatory guidelines for nuclear facilities are under revision, and those for geological disposal have not yet been discussed in detail. In developing a methodology for assessing operational safety of geological disposal, relevant guidelines for other nuclear facilities are considered in the NUMO Safety Case. These include vitrified waste storage, which functions similarly in HLW repository facilities at the surface, including waste acceptance, waste inspection and encapsulation in a metal overpack, etc. In addition to nuclear safety guidelines, the Mining Safety Act and Industrial Safety and Health Act for civil engineering are also used as a reference when planning underground repository operations. As a first stage of methodology development, radiological safety has been highlighted, focusing on activities relevant to HLW handling and transport based on repository design and defined procedures of construction, operation and closure. The outline of assuring operational safety is presented in Suzuki et al. (2015).

For post-closure long-term safety, there is a need to develop an assessment approach and methodologies which can be applied to specific sites and the repository tailored for each site. The safety for defined criteria and other relevant requirements also need to be developed. In the NUMO Safety Case, procedures and methodologies to assess long-term safety are demonstrated for illustrative geological disposal systems and 'realistic' site geological conditions as defined by the illustrative SDMs. A risk-informed approach has been incorporated into the safety assessment, based on both international guidelines and recent national discussions on safety regulations. Each scenario is developed and classified with consideration of its probability of occurrence during the assessment period and defined target doses (Inagaki et al. 2015; Kurosawa et al. 2015). Figure 12–9 shows a preliminary assessment result. The outlines of advanced methodologies focusing on the scenario development (Kurosawa et al. 2015) and radionuclide transport models to assess the post-closure safety (Ishida et al. 2015) have been presented.

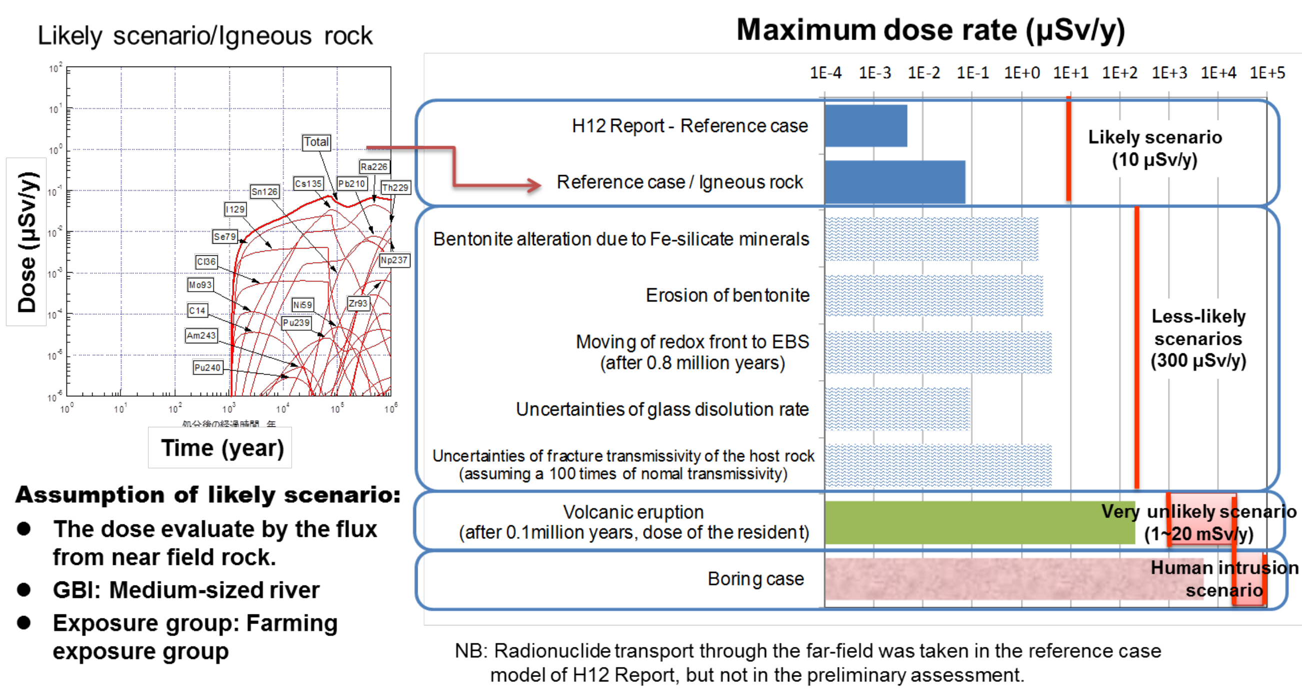

Figure 12–9. Examples of preliminary assessment results for post-closure safety (Fujiyama et al. 2015)

12.4.2 Promoting R&D collaboration

NUMO is actively participating in international projects, in particular, projects at URFs (Underground Research Facilities) operated by foreign implementers in order to tackle key R&D issues that establish technical capabilities with necessary human resources to gain technical experience and knowledge with the aim of establishing organizational trust, as NUMO has no R&D infrastructure such as URFs, lab test facilities, etc. at present.

NUMO is also promoting collaborative activities with domestic R&D organizations. It has been conducting a collaborative project with CRIEPI at its Yokosuka site, on borehole investigation techniques, methodology to develop site descriptive models, and other issues for the site investigation, a new phase of the project being prepared. Collaboration with JAEA is being expanded for key R&D issues such as mechanistic model development for glass matrix corrosion, alteration of bentonite and radionuclide sorption.

12.4.3 JAEA Activity

In 2005, JAEA was established as the only comprehensive research institute on atomic energy in Japan merging JNC and Japan Atomic Energy Research Institute (JAERI). R&D on HLW disposal is stipulated as a mission of JAEA in the Act on JAEA, which contributes to both implementation and safety regulation.

JAEA has three research centers dedicated to R&D on geological disposal technologies (Figure 12–10), including two purpose-built generic URLs (Underground Research Laboratories) (e.g. Koide et al., 2015), one at Mizunami in crystalline rock (http://www.jaea.go.jp/04/tono/miu_e/index.html), and the other at Horonobe in sedimentary rock (http://www.jaea.go.jp/english/04/horonobe/index.html).

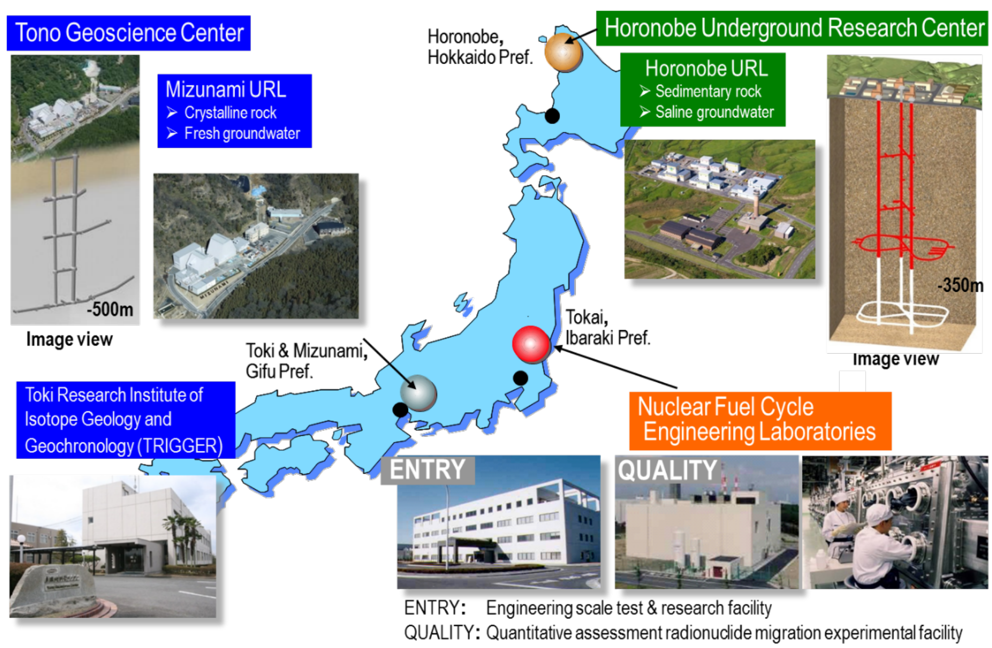

Figure 12–10. JAEA's R&D Facilities for HLW Geological Disposal

Both URL projects proceed in three overlapping phases, "Phase I: Surface-based investigation", "Phase II: Investigation during construction" and "Phase III: Investigation in drift". During phase I, techniques have been developed for characterizing the deep geological environment based on surface-based investigations. This takes into account data requirements for the design of underground facilities and infrastructure, along with associated safety assessment.

The results of Phase I in both URL projects were summarized separately as an URL Phase I Report (Saegusa and Matsuoka 2010; Ota et al. 2010). In Phase II, data obtained served to verify the results from the surface-based investigations and, additionally, characterize the perturbations caused by the excavation process. Such perturbations (e.g. changes in groundwater flow and rock mechanical properties) are monitored and compared to prior model predictions (e.g. Tsuruta et al. 2009; Daimaru et al. 2010; Sato et al. 2014). As research galleries have been constructed at the depth of 500 m for Mizunami, and 350 m for Horonobe by 2014, R&D programs using excavated galleries (Phase III) are actively being conducted.

As a part of the Phase III R&D program, at the Mizunami URL, groundwater recovery experiment is on-going to study recovery processes of the geological environment around the drift after backfilling and to develop long-term monitoring technology. At the Horonobe URL, a full-scale engineered barrier system (EBS) test for H12 vertical emplacement concept was set up and groundwater injection has been initiated in 2015.

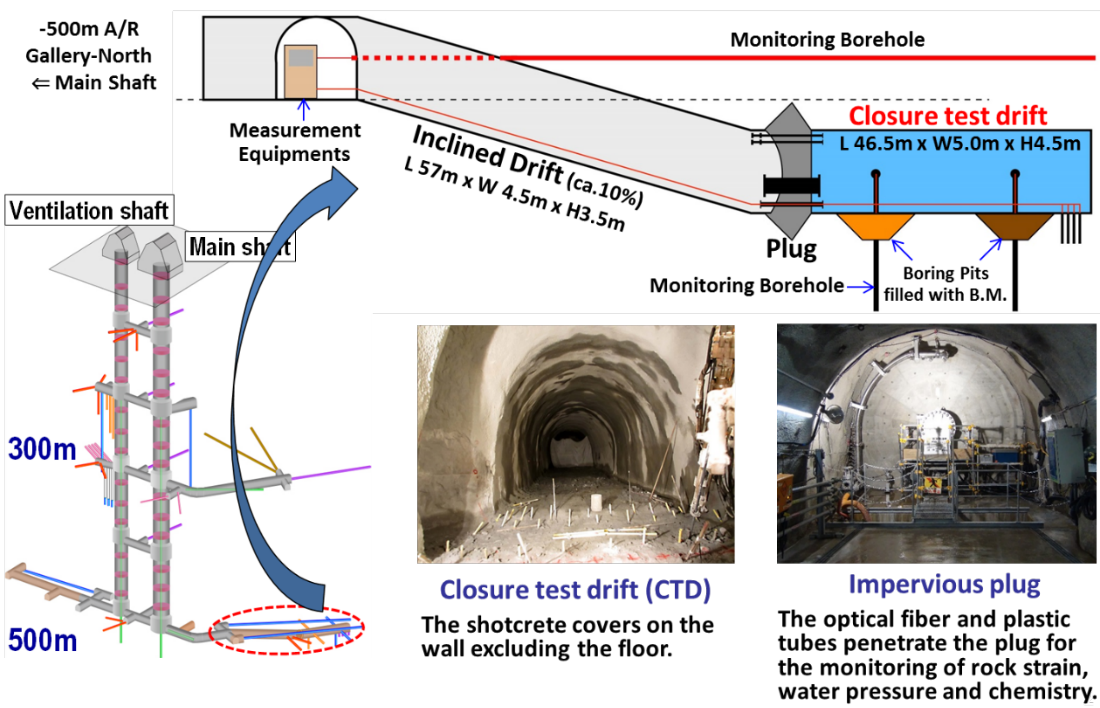

Figure 12–11. Groundwater recovery test at Mizunami URL

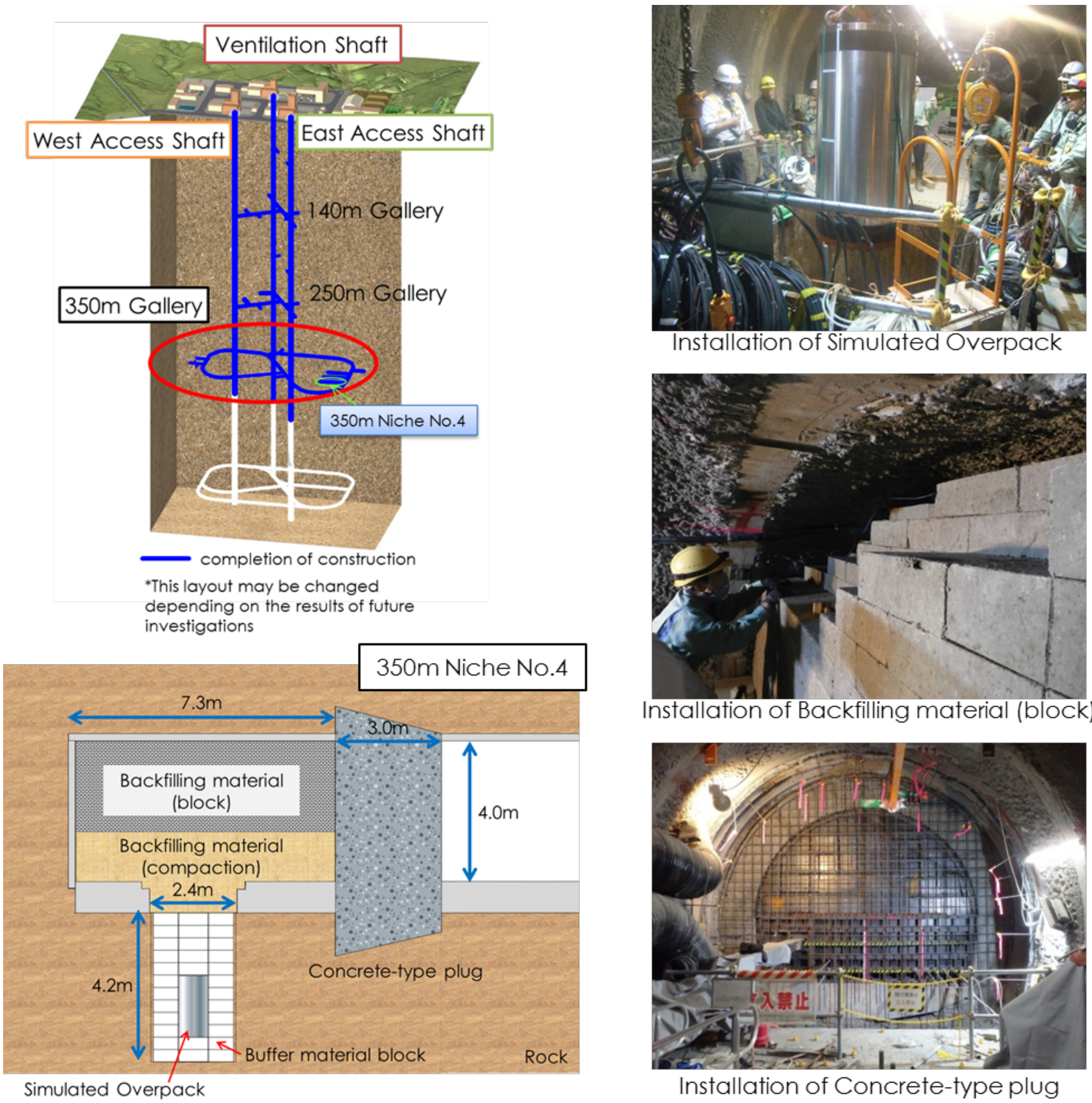

Figure 12–12. Full-scale EBS test at Horonobe URL

Various R&Ds at the URLs have been carried out under collaboration with relevant domestic research institutes (e.g. RWMC, CRIEPI, and AIST) and universities, and also foreign R&D organizations, such as Nagra, Andra and KAERI.

In parallel with URL activities, a study of the long-term stability of the geological environment has been conducted at the Tono Geoscience Center aiming at establishing techniques for estimating long-term evolutions of geological environment by characterizing Neotectonic events and processes such as crustal movements and volcanic activities etc. on nationwide scale.

JAEA has also been developing system evolutions models for near-field taking account of coupled phenomena and non-linear processes based on state-of-the-art knowledge. Radionuclide migration studies, such as glass dissolution, sorption and diffusion in the buffer and host rock, are also on-going to enhance mechanistic understanding and to support performance assessment models and related parameter settings (e.g. Tachi et al. 2011; Tachi et al. 2014). The relevant fundamental data are compiled as databases at the JAEA website. "Buffer material Database (https://bufferdb.jaea.go.jp/bmdb/index_e.jsp)" and "Thermodynamic, Sorption & Diffusion Database (https://migrationdb.jaea.go.jp/)" are available. Advanced scenario development methodology has also been provided under collaboration with NUMO, which has been applied to the NUMO Safety Case.

In order to manage expanding knowledge on geological disposal including the R&D outcomes described above, JAEA has been developing a Knowledge Management System (KMS) (e.g. Makino et al. 2012). As a part of the KMS, a supporting system to conduct performance assessment (PA) calculation and reporting on a web browser known as the Electronic Performance Assessment Report (e-PAR) was established. By using this system, PA calculation could be performed in effective and quality-assured manner. The e-PAR provides user-friendly interface and functions for the users who do not have detailed know-how to use PA calculation code.

JAEA's R&D activities in the second mid-term research program (fiscal years 2010–2014) were synthesized and published at the end of September 2014. The key outcomes were integrated and published in the web-based report, "CoolRepH26", which is an advanced reporting system built on the website (http://kms1.jaea.go.jp/CoolRep/; now only available in Japanese). R&D subjects for the next research phase were also identified through review by internal and external experts and published at the same time.

JAEA has been working in close cooperation with international and foreign R&D organizations. Currently it has cooperation agreements with 12 organizations from 8 countries in addition to participation in multilateral collaborations.

12.4.4 RWMC Activity

RWMC is the only Japanese organization specific to R&D on radioactive waste management, and its research areas cover all types of radioactive wastes, from low-level to high-level waste. It has been contributing towards support for development of national policies and safety regulations, promoting public understanding of geological disposal, and expanding technological options for waste disposal from power utilities. Development and demonstration of repository engineering technology is one of the specific features of R&D activities at RWMC, which is carried out at laboratories and URFs in collaboration with Japan Nuclear Fuel Ltd. (JNFL), JAEA, and foreign partner organizations. RWMC has been conducting its R&D with close cooperation with international colleagues. Currently it has cooperation agreements with 14 organizations from 12 countries. Its international partners include implementation bodies such as Andra in France, SKB in Sweden, Posiva in Finland, Nagra in Switzerland and KORAD in Korea, etc.

RWMC's R&D activities in the field of geological disposal are being conducted in two divisions (which are called "projects" in RWMC), such as Repository Engineering and EBS (Engineered Barrier System) Technology Research Project, and EBS Material Research and Assessment Project (Figure 12–13), under the initiative of the "Coordination Executive".

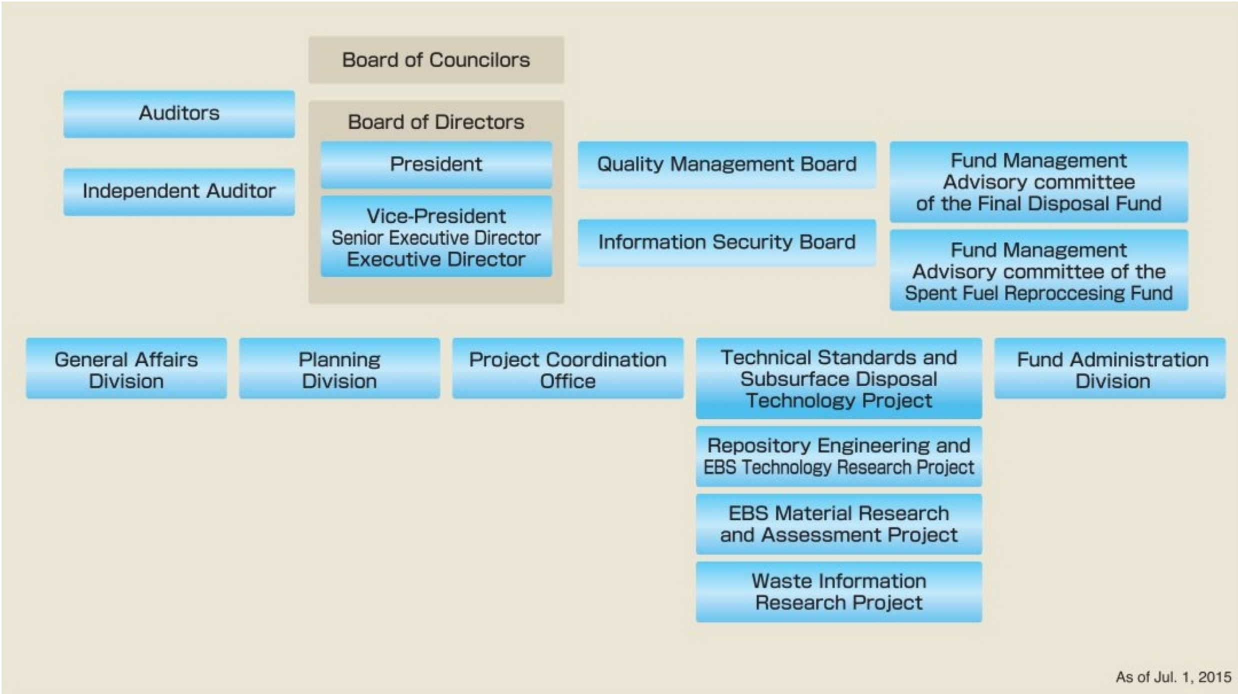

Figure 12–13. Projects for R&D in RWMC's organizational structure

The followings are a brief summary of R&D activities in geological disposal being conducted at each division of RWMC.

12.4.4.1 Repository Engineering and EBS Technology Research

RWMC has developed key technologies for remote operation of welding and inspection of overpack, and transportation and emplacement of buffer materials in both vertical and horizontal concepts which has been conducted at Horonobe URL (e.g. Asano and Aritomi 2010; Kawakami et al. 2010; Nakashima et al. 2010; Takao et al. 2010; Asano et al. 2011). In addition, in order to demonstrate retrievability of waste packages, full-scale buffer materials removal technology has been developed.

Other areas for technology development being conducted at this division are (1) quality assurance and long-term integrity of EBS and (2) monitoring technology of repository.

12.4.4.2 EBS Material Research and Assessment

In the EBS material research and assessment division, research on safety assessment for geological disposal of TRU wastes generated at spent fuel reprocessing and MOX fuel fabrication plants, is being conducted.

R&D on TRU waste disposal has been conducted by participating in the joint research framework of European Union such as IGD-TP and sharing results of international R&D. From the view point of safety assessment for TRU waste disposal in Japan, C-14 is a key nuclide and a corrosion model is under development for zircaloy used for fuel cladding and stainless steel used for hardware under the repository conditions, because C-14 will be released through corrosion of radio-activated metals (e.g. Sakuragi et al., 2014). Furthermore, RWMC has been conducting studies on natural analogue for cement and bentonite system to demonstrate long-term stability of multiple barriers for TRU waste disposal system (e.g. Fujii et al. 2013).

12.4.4.3 Other Studies in RWMC

Other than R&Ds in geological disposal, RWMC has been conducting R&Ds on intermediate depth disposal for low-level waste with relatively higher activity. In this field, its research focuses on demonstration of construction technology of cavern disposal facility located at 50-100 m below the surface (Akiyama et al. 2011).

Another important activity being conducted by RWMC is collection and provision of up-to-date information regarding disposal projects in foreign countries, which covers site selection, relevant legislation, safety regulations and so on, for the purpose of supporting national policy makers. It has also been providing information to the public on current situation of disposal projects in other countries through the website of RWMC. (http//www2.rwmc.or.jp)

12.5 Concluding Remarks

The H12 Report demonstrated the feasibility of safe and technically reliable geological disposal based on a generic study. On the basis of the Report, the Final Disposal Act was enacted and NUMO was established in 2000 to carry out the geological disposal.

Since the 2011 Great East Japan Earthquake, tsunami and reactor meltdowns, there has been increased public concern regarding major natural disasters to cause accidents at nuclear facilities. In response to this and to help rebuild public confidence required to site a repository, the Government re-evaluated the technical feasibility of the geological disposal program based on state-of-the-art geosciences and implementation process.

Following the re-evaluation, the Government concluded to further promote geological disposal program. The new Strategic Energy Plan adopted by the Government in April 2014 states that the geological disposal of HLW should be addressed, and action to implement the disposal should not be postponed. In accordance with this plan, the “Basic Policy for Final Disposal” was revised as a Cabinet decision in May 2015 to include a newly added siting process with identification of “Scientifically Preferable Areas”.

NUMO has started to develop a comprehensive safety case that builds on both worldwide technical progress and also increased understanding of relevant geological conditions in Japan, incorporating extensions in key technical areas from the H12 Report. In cooperation with relevant organizations, NUMO has been developing key technologies required for the safe implementation of the geological disposal project since its establishment, which are being reflected to provide the new safety case. This safety case will demonstrate flexibility to respond to a siting process that is still focused on volunteering and thus reflects the need to be able to compare alternative designs or siting options, which requires a performance assessment approach based on realism rather than conservatism.

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12.7 Acronyms

AIST—National Institute of Advanced Industrial Science and Technology

ANRE—Agency of Natural Resource and Energy

CRIEPI—Central Research Institute of Electric Power Industry

EBS—Engineering Barrier System

FEPC—Federation of Electric Power Companies

HLW—High Level Waste

JAEA—Japan Atomic Energy Agency

JAEC—Japan Atomic Energy Commission

JNC—Japan Nuclear Cycle Development Institute

JNFL—Japan Nuclear Fuel Ltd.

METI—Ministry of Economy, Trade and Industry

NIRS—National Institute of Radiological Sciences

NRA—Nuclear Regulatory Authority

NUMO—Nuclear Waste Management Organization of Japan

PA—Performance Assessment

PIA—Preliminary Investigation Areas

PR—Public Relations

R&D—Research and Development

RWMC—Radioactive Waste Management Funding and Research Center

URF—Underground Research Facility

URL—Underground Research Laboratory

Chapter 13

On Interdisciplinary International Approach for Geological Disposal in Latvia

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ABSTRACT: An interdisciplinary conceptual approach to the development of multinational repositories is considered in Latvia. This approach includes: a) building of an international stakeholder consensus that will be promoted by activating the international multilateral interaction between intra- and international stakeholders, b) a gradual progress in reaching an intergovernmental consensus and multilateral agreements via observing various interests and resolving emerged controversies, and c) developing the knowledge and mental flexibility, which are basic prerogatives for elevating the level of mutual understanding and consensus.

13.1 Introduction

The continuing growing public concerns about possible nuclear risks and decision making policy in nuclear areas, including safe disposal of the generated radioactive waste (RW) as inevitable principal back-end of any nuclear activity, may endanger the development of novel advanced projects of efficient use of nuclear energy. These risks and related problems require innovative approaches to find secure, reliable, and confident solutions of forthcoming use of nuclear energy, which would provide the basic conditions for safe and sustainable development of global society in the safety of nuclear energy.

According to Article 11 of the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management, "Each Contracting Party shall take the appropriate steps to ensure that at all stages of radioactive waste management individuals, society and the environment are adequately protected against radiological and other hazards." Latvia is aware of necessity to provide safe disposal for all types of radioactyive waste. Latvian situation displays a typical small-country case without nuclear activities and nuclear waste, because in 2005 in the frame of USA-Russia-Latvia Inter-Governmental agreement, in co-operation with IAEA, the fresh fuel of the shutdown Salaspils Research Reactor (SRR) has been moved out of Latvia, and in turn, in 2008 in the frame of the same agreement the spent fuel of the shutdown Salaspils Research Reactor has been moved out of Latvia.

The strategy and policy of radiation and nuclear safety in Latvia, as a small country, does not foresee to build a NPP nor a geological repository, taking into account relatively small volumes of total RW (100-200 cbm, mainly disused sealed sources—DSS). Latvia is following the recommendations of the Convention (Article 1) to achieve and maintain a high level of safety worldwide in spent fuel and radioactive waste management, *through the enhancement of national measures and international co-operation, including where appropriate, safety-related technical co-operation*. For Latvia, the best option is participation in the design of a regional or international repository mainly due to economic and geological reasons. A possible way to increase safety and security is to develop shared multinational nuclear projects, namely: (a) construction of shared multinational repositories for safe deep disposal of spent fuel and high-level RW (McCombie et al, 2001, McCombie and Chapman 2006, Štefula 2004), and multinational arrangement of advanced nuclear power plants (NPPs) and research facilities.

Latvia has already developed appropriate legislation for international cooperation in RW management by issuing specific Cabinet Regulations on *Generic Principles for transboundary Exchange* of RW as well as implementing the Euratom Directives on the RW shipment inside and through EU territory (2006) and the one establishing a Community framework for the responsible and safe management of spent fuel and radioactive waste (2011). In addition, Latvia has submitted to the management of the amended European Commission (EC) research pilot project SAPIERR (Support Action: Pilot Initiative for European Regional Repositories; Chapman et al., 2005) the data on RW inventory, specifying the inventory to be disposed as a long-lived RW as well as existing legislation basis concerning RW management and issues of regional/international RW disposal options.

Bearing in mind that the primary problem in realization of such regional disposal is construction of geological repository in one of the countries, thus, to solve the disputable issue: which will be the host country for the repository, and reach an agreement from the all interested parties. This would require consensus among all involved parties for successful realization of a shared nuclear project.

Such consensus must be reached among all potentially affected stakeholders not only within each country but also among the several participating countries. So, according to the pilot project SAPIERR, as well as the European Repository Development Organisation (ERDO) for implementation the shared repository concept (Euratom 2011), an actual task emerges – to find out efficient solutions of such an interdisciplinary complex set of technical, societal, political, economic and psychological issues.

13.2 The Most Significant Aims of Multilevel Stakeholder Involvement

In the framework of the problem of stakeholder involvement and consensus seeking on an international scale, besides the existence of multiple structural levels of involved stakeholders, one can also foresee a multitude of necessary consensus reaching stages in long term-scale, taking into account that development of a RW repository is a time-consuming process, which usually prolongs several decades. In such a time-period, interests of involved countries and relations between countries likely might change (IAEA, 2004). In particular, one would be necessary to develop mutually concerted solutions for management of possible economic risks for all involved parties by taking also into account existing differences in the economic development and national financial and economic policies.

Recognizing that for reaching acceptance of multinational repositories (in comparison with national ones) there is necessary to put forward all our best efforts, one should emphasize a real need to develop and implement interdisciplinary research studies, as well as develop practical advanced methods of stakeholder involvement aimed at reaching consensus at all levels and finally – acceptance to arrange the repository.

In particularly, according to the framework of the EC pilot project SAPIERR, namely, “multiple siting options should be maintained over a long time and the ultimate selection of preferred sites should be an open process in which all technical, societal, economic and political issues are tabled simultaneously” (Chapman et al. 2005). The guidelines for siting regional repositories should be based on the consensus reached by international participants, taking into account potential benefits of a shared repository, and should include the structure and processes required to assure the agreed level of stakeholder involvement.

13.3 An Extended Concept of the Stakeholder Framework

Taking into account that in our current task there appear qualitatively novel stakeholder classes, for the purposes of stakeholder involvement and consensus reaching problems pertaining to multinational repositories one should specify the basic structure and functions of main stakeholder categories. For considering this issue let us refer to recent approach (Weightman et al., 2006), by emphasizing the following basic tasks:

- Identification and prioritization of stakeholders
- Identification of motivations of each stakeholder
- Establishing benchmarks for stakeholder engagement activities
- Development and implementation of the stake engagement strategy

Particularly, the extended stakeholder community shall recognize all relevant international administrative, professional, business as well public bodies and their recommendations, for example:

- European Union Council and EC (for European countries)
- International business consortiums and projects, including those managed by EC
- International associations of national and top-level intra-national stakeholders dealing with waste management agencies and national regulatory authorities, such as Arius (Association for Regional and International Underground Storage), and the recently established ERDO (in 2009)

A novel problem that has recently arisen is that the national (or a high-level) stakeholder is being represented by the national government, who is facing the task of seeking simultaneously an upward (or international) consensus via interacting with international stakeholders and partnering with national stakeholders, as well a downward (intra-national) consensus via interacting with multi–level intra-national stakeholders.

Although a serious legal framework has been established, including the Joint Convention on the Safety of Spent Fuel Management and on the Safety of RW Management, EC and International Atomic Energy Agency (IAEA) (Chapman et al. 2005; Euratom 2011; IAEA 2004; Boutelier 2006) the national stakeholders are interacting in a psychologically stressed environment, which is characterized by often diverse political, legal, economic and social conditions and interests.

Taking into account that multinational RW repositories are foreseen mainly for disposal of spent nuclear fuel and nuclear material in the form of high-activity RW and long-lived disused sealed sources, we can also refer to the recent international instrument, namely, the "Strengthening the Global Nuclear Safety Regime" (IAEA 2006). This document addresses mainly nuclear installations and contains a list of relevant stakeholders and their groups.

Withal, a detail specification of the structure of such novel stakeholder framework should be worked out during the process of development of a multinational facility concept and appropriate international programme, in particular, the development of the institutional framework, which would result in identification of possible optimal structures for efficient programme management, as well as of legal forms for cooperation and "a study of the sharing of benefits and liabilities among multinational partners" (Chapman et al. 2005).

13.4 A Possible Approach to Stakeholder Interaction on International Scale

Recently, efforts have been made to investigate technical, economical, societal and environmental problems of RW disposal safety. These have resulted in a series of IAEA, OECD Nuclear Energy Agency (NEA) and other publications (OECD NEA 2004; IAEA 2006)] where possible ways of stakeholder involvement and stressed the actual necessity to understand their concerns are analyzed in parallel with significant advances in general approaches to consensus building issues (Butler and Rothstein 2006).

Possible routes for stakeholder involvement and communication on the international scale can be based on the synergetic approach to solve stakeholder interaction issues (Dreimanis and Salmins 2005), aimed at self-organization of various stakeholder categories and their development - in particular, via activating and diversifying interaction among stakeholders - into a harmonized stakeholder community having common strategic goals.

As a key mode of this interaction has been indicated, above all, mutual learning of various stakeholder groups, aimed to elevate their knowledge level, as well as to enhance mutual understanding, the whole process of mutual learning and educating of stakeholders will be capable to emerge in a *knowledge creating stakeholder community* being capable to use novel communication and knowledge management forms at all stages and levels of decision-making.

Such kind of web-based communication of all level stakeholders levels on the international scale is expected to be especially important pertaining to the self-organization goals of stakeholders already on a multinational level as well as for promotion of activities aimed at reaching their mutual understanding and consensus building. In particular, just for the case of geological repository arrangement such web-based approach—being a prospective way for all stakeholders to access permanently updated information - has already been markedly developed as an instrument to share information sharing among stakeholders, with the final aim to ensure socially informed and grounded decision-making (Yoshimura et al. 2006).

Such global-scale, web-based communication possibilities will be especially useful instruments for the development of international cooperation, particularly—in the framework of Global Nuclear Safety Regime (IAEA 2006)—between national and intra-national stakeholders (i.e., governmental authorities in nuclear safety, operators, etc.), thereby developing relevant communication of international stakeholders, being represented (beside the intergovernmental organizations IAEA, OECD/NEA) by such entities as:

- Multinational networks among regulatory authorities:
    - International Nuclear Regulators Association (INRA)
    - Network of Regulators of Countries with Small Nuclear Programmes (NERS) being especially important for cooperation of small countries having a necessity and interest in shared disposal solutions
- International networks among operators, such as World Association of Nuclear Operators
- (Stakeholders in international nuclear industry:
    - The World Nuclear Association (WNA)
    - Suppliers of services and equipment
    - Non-governmental organizations, mass media

Such international entities and organizations are called to form a proper structural framework for development of such highly actual for the current period of nuclear energy management initiatives as Multilateral Nuclear Approaches (MNA). In this frame, in addition, there is foreseen to establish multinational, and regional MNAs for new facilities which are based on joint ownership (IAEA 2006).

The development of partnerships between international and local organizations (Zadeh 1973) would serve as one of the key elements of democratic dialogue—via observing the whole set of various interests—and, correspondingly, can be regarded as a prerequisite for reaching shared understanding of a disputable problem, and finally, as a factor promoting multi-stakeholder consensus building, taking into account a whole set of possible real serious local distinctions and circumstances (Boutelier et al. 2006), namely:

- Different national legislation and time schedules
- Transportation policy and negative public reaction being a potential source of probable disputes and controversies among partnering countries
- The cost allocation

13.5 Multilevel Confidence Building of Stakeholders in Nuclear Activities

13.5.1 Confidence Building at a Global Scale

One of the basic elements in systemic societal optimization of nuclear energy management is building stakeholder confidence (including public) in official management policy and corresponding institutions and authorities. The world-wide scale of nuclear activities emphasizes the importance of multi-level confidence building, first of all, at a global scale, via the United Nations activities aimed at reaching political settlement of controversies and discrepancies related to the use and proliferation of weapons of mass destruction, with a possible use of a novel approach—based on social self-organization—of minimization of controversies.

In the area of nuclear energy management, just the IAEA activities are aimed at reaching safety and security of the use of nuclear energy and materials, at assurance of international community of peaceful use of nuclear materials as well as at efficient interaction with society. Activities of such bodies (established by IAEA) as the International Nuclear Safety Group (INSAG) and the International Nuclear Information System (INIS) highly promote (a) multinational cooperation, aimed to harmonize the global safety approaches the nuclear legislation transparency, (b) to maintain public confidence in all activities and to provide stability in decision-making, extend wide participation in the decision-making to stakeholders, and (c) the increase of the nuclear knowledge and "awareness" levels about safe use of nuclear energy.

13.5.2 European Union (EU) Actions and Initiatives

On the regional level, first of all the EU bodies and their activities succeed safe management of nuclear power plant running, decommissioning and RW disposal, in particularly, via (a) establishing a system for development of safety and standards for nuclear installations in the EU, (b) carrying out regular nuclear material inventory and safety inspections, and (c) analyzing of environmental, economic and social issues of nuclear energy activities. Significant contribution to social optimization has been given by the governance projects TRUSTNET, COWAM, RISCOM, RISKGOV, CETRAD, CARL and OBRA within the EC Framework programs FP6 and FP7—with the main goal to provide mechanisms for all stakeholders to have access to the knowledge.

13.6 Conclusions

Guided by recent international trends in the development of multinational RW repositories we have proposed an interdisciplinary approach towards stakeholder communication and building their consensus on the international scale. In line with the emphasized significance to increase internal variety of stakeholders via their social learning, tolerant communication and creative flexibility in the decision-making, the whole set of stakeholders community in their consensus building efforts might be called to develop "the same creative multilateral engagement and active international cooperation" (Petts, 1998).

The proposed approach could be extended also to solving similar societal-technical problems for arrangement of other multinational nuclear facilities (NPPs, research units) as well as national facilities supposedly having trans-boundary context. On the basis our analysis one can recommend to develop, in the frame of international cooperation projects, further systemic interdisciplinary studies preferably having goal-oriented status.

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Chapter 14

Approach to Disposal of Radioactive Waste and Spent Nuclear Fuel in Lithuania

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ABSTRACT: The Lithuanian spent nuclear fuel disposal program is currently aimed at the development of the initial site investigation program and a preliminary disposal facility design. The near-future activities will include planning of a geological repository design, investigation and characterization of prospective regions, development of a repository concept, and a general safety assessment, as well as long-term international cooperation.

14.1 Introduction

The Ignalina nuclear power plant (NPP), the only NPP in Lithuania, was a vital component in Lithuania's energy balance because it produced more than 70% of the total electricity in Lithuania until it was decommissioned in 2009. The main reason for this high percentage was that nuclear power had a significantly lower production cost than other forms of power under the economic and technical circumstances in Lithuania. The Ignalina NPP had the only two RBMK-1500 reactor units constructed; this was the last and most advanced version of the RBMK-type reactor design series. The Ignalina NPP reactors were commissioned in December 1983 and August 1987. The original design lifetime for these reactors was projected to be 2010–2015. After the nuclear accident in Chernobyl, safety systems at the Ignalina NPP were re-evaluated, and it was decided to decrease the maximum thermal power of the units from 4,800 to 4,200 MW. This limited the maximum electric power to about 1,250 MW per unit. On October 5, 1999, the Seimas (the Parliament of Lithuania) approved the National Energy Strategy, in which it was stated that the first unit of the Ignalina NPP would be shut down before the year 2005, taking into consideration substantial long-term financial assistance from the European Union (EU), G7, and other states, as well as international institutions. On October 10, 2002, the Seimas approved an updated National Energy Strategy in which it was stated that the first unit would be shut down before the year 2005 and second unit by 2009 if funding for decommissioning were available from the EU and other donors. Following this decision, Unit 1 and Unit 2 were finally shut down on December 31, 2004, and December 31, 2009, respectively.

The RBMK-1500 reactors were designed for nuclear fuel with 2% enrichment of ^{235}U. Use of the new uranium-erbium fuel at the Ignalina NPP started in 1995, and at the end of 2001, the first batch of nuclear fuel with 2.6% enrichment was loaded into the reactor. Later on fuel with 2.8% enrichment was also used. During operation of the Ignalina NPP, about 21,571 spent nuclear fuel (SNF) assemblies, containing about 2,400 tons of uranium (110–112 kg per assembly), had been generated by 2010.

14.2 Low and Intermediate Level Waste

14.2.1 Current Situation and Plans for Treatment, Conditioning, and Interim Storage

Currently the complex for raw solid radioactive waste storage at the Ignalina NPP consists of four buildings (155, 155/1, 157, and 157/1), with auxiliary systems and equipment for operations in each building. Buildings 155 and 155/1 are intended for storage of short-lived low/intermediate level waste (LILW-SL), and they are assembled monolithic buildings. Buildings 157 and 157/1 are intended for storage of LILW-SL and also for long-lived low/intermediate level waste (LILW-LL). Buildings 157 and 157/1 are separated into sections by solid partitions, with each section storing a specified group of wastes. At this time only compaction (baling) of the solid, very low-level combustible waste with a low-pressure compactor is performed. These bales are also stored in these buildings. Wastes from research, medicine, and industry also are transported to the NPP and stored in these buildings. A Safety Analysis Report (SAR) on these facilities was prepared during 1999–2000, and the regulatory authorities issued a license for its operation. A project for modernizing the entire management system for solid LILW-SL and LILW-LL is being implemented by German company NUKEM. The Lithuanian Energy Institute is the local subcontractor for preparation of the Environmental Impact Analysis (EIA) Program/Report, the Safety Analysis Report, and licensing support. The purpose of this project is to retrieve the radwaste from the existing storage facilities (Buildings 155, 155/1, 157, 157/1), and segregate this waste into two streams: one suitable for landfill disposal (very low level waste), and the other one for processing in the new treatment and conditioning plant for disposal in a Near-Surface Repository (NSR). Construction of the new storage facility for temporary storage of the conditioned LILW-SL before disposal in the NSR, and an interim storage facility for the LILW-LL and spent sealed sources, is also included in this project (Poskas et al. 2012).

Liquid waste is collected in large concrete tanks, for temporary storage, before being evaporated. Evaporator bottoms are bituminized and transferred to the waste storage facility (Building 158). This is an aboveground, two-story, assembled concrete monolith building consisting of 12 steel-lined vaults for loading the bitumen compound (bitumen and evaporator bottoms mixture). A preliminary long-term safety assessment of this facility was also performed by SKB (Sweden) with the participation of the Lithuanian Energy Institute (Poskas et al. 2000). This assessment led to the conclusion that this radioactive waste facility could be converted into a disposal facility if a multilayer earth cover was used, but a more detailed analysis is necessary. At present, this facility is licensed as a storage facility, based on the Safety Analysis Report prepared in 1999–2000.

Ion-exchange resins and perlite mixtures are also stored in the tanks before solidification in the cementation facility. Framatome ANP GmbH entered into the agreement with the Ignalina NPP at the beginning of 2002 for the erection of a cementation facility and an interim storage facility for cemented waste. The Radioactive Waste Solidification System processes the ion-exchange bead resins plus filter aid (Perlite) and solid particle sediment from evaporator concentrates plus filter aid generated and already collected in the controlled area during plant operation, and converts them into a suitable condition for storage. This is achieved by solidifying the liquid waste in the binding agent cement, or, as necessary, in a mixture of cement and additive. The Lithuanian Energy Institute was acting as the local subcontractor to Framatome ANP GmbH for support of the preparation of the EIA Program/ Report and SAR, and licensing support of the project. The implementation of the cementation facility and the interim storage building was successfully finalized in March 2006.

An old "Radon" type facility for institutional waste (from research, medicine, and industry) is located at Maisiagala (40 km from the capital Vilnius). This facility was designed for disposal of institutional waste, and is typical of "Radon" facilities constructed in the early 1960s (1964 in Lithuania) throughout the former Soviet Union. It was closed in 1989. Waste at this site was disposed of in a reinforced concrete vault with internal dimensions 14.75 × 4.75 × 3 m (200 m^3 in volume). The vault was only partially filled with waste (about 60%) during its operation. When loaded, the waste was interlayered with concrete. In addition, sealed sources were placed in two stainless steel containers, each with a volume of 10 L. Medical sources were placed within biological shielding. At the end of the operation, the remaining volume was filled with concrete and sand, and sealed. A preliminary safety assessment of this facility was performed by SKB (Sweden), with participation of the Lithuanian Energy Institute (Poskas et al. 2000). After that the Radwaste Management Agency (RATA) applied for a project (2004–2006), which was aimed at safety assessment and upgrading of the Maisiagala facility. A consortium led by French company Thales, and including the Lithuanian Energy Institute and some other organisations, developed a database for the Maisiagala repository inventory in 2004. Data from Data Record Books were recorded into the database, and analysis of the radionuclide inventory was performed. In parallel, additional site data were collected. Based on this updated information, a SAR on the existing disposal facility was prepared, and proposals for safety improvements were suggested. After review of the preliminary SAR by Regulatory Authorities, it was decided to upgrade and license the Maisiagala facility as a radwaste temporary storage facility because safety requirements for a disposal facility cannot be met. The State nuclear power safety inspectorate (VATESI) license was issued in 2006. The current plan is to retrieve the waste from the Maisiagala temporary storage facility and dispose of it in an NSR to be constructed at Sabatiske, near the Ignalina NPP. Spent sealed sources will be transferred to the Ignalina NPP site for storage. The preparation of the Final Decommissioning Plan is ongoing.

14.2.2 New Near-Surface Disposal Facilities

Lithuania legislation allows very low-level, short-lived waste to be disposed of in a simple near-surface repository of landfill type. Here treated and untreated radwaste that meets acceptance criteria, defined during safety assessment, could be placed. Activities related to the implementation of a landfill repository for very low-level waste at the Ignalina NPP site were started in 2003. A reference design, recommendations on site selection, and preliminary waste acceptance criteria were prepared in 2003. The Lithuanian Energy Institute performed a site comparison, and proposed preliminary waste acceptance criteria for the selected site during 2005–2006. A contract for design of the landfill repository and buffer storage facility, and construction of this buffer storage facility, was signed at the end of 2007. Implementation of the project is going on. The total volume of the landfill repository will be 60,000 m^3. The disposal of waste will be organized in campaigns (4,000 m^3 of VLLW each). Between campaigns, waste will be stored in the buffer storage. Operation of the landfill repository is planned for 2018–2038 period. After closure, active control (with monitoring of the site) and passive control will be performed until 2068 and 2138, respectively.

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

Siting of a near-surface repository in Lithuania was started in 2003. The Lithuanian Energy Institute prepared the siting criteria that were used by the Lithuanian Geological Survey and the Institute of Geology and Geography in selecting two sites for the repository close to the Ignalina NPP. In 2004, the Environmental Impact Assessment Program was prepared and approved by the Lithuanian government. An Environmental Impact Assessment Report was also prepared by the consortium led by the Lithuanian Energy Institute. The Galilauke site was found to be preferable. Relevant Lithuanian Authorities approved the report, but neighboring country (Belarus) objected because the site was close to the border. It was then decided to investigate a new site, at Sabatiske, which is very close to the Ignalina NPP. Peer review of the siting process was performed by IAEA experts in December 2005. In 2006 the EIA Report was updated, including both Galilauke and Sabatiske sites, and in 2007 it was approved by the Ministry of Environment. Finally, the Sabatiske site for the NSR was approved by the Lithuanian Government on November 11, 2007. At the moment the design project is under development by the consortium: AREVA (France), ANDRA (France), the Lithuanian Energy Institute (Lithuania), Specialus Montazas-NTP (Lithuania), Pramprojektas (Lithuania). The total volume of the near-surface repository will be 100,000 m^3. Operation of the near-surface repository is planned for 2021–2038 period. After closure, the active control and the passive control will be performed until 2138 and 2338, respectively.

14.3 Spent Nuclear Fuel and Long-Lived Waste

14.3.1 Interim Storage of SNF and Long-Lived Waste

According to Lithuania's legislation, it is forbidden to reprocess spent nuclear fuel (SNF) in Lithuania's territory, but it is not forbidden to reprocess SNF in other countries and return secondary waste to Lithuania. However, SNF reprocessing is not anticipated under current conditions, because the use of reprocessed SNF is limited and secondary waste must be treated similarly to SNF and disposed of in a deep geological repository. In 1992, the decision was made to build an interim dry SNF storage facility with a lifetime of about 50 years at the Ignalina NPP site. After the tendering process, GNB (Gesellschaft für Nuclear Behalter) casks have been chosen. Twenty of these casks are ductile cast iron CASTOR RBMK-1500 casks; the remaining ones are metal-concrete CONSTOR RBMK-1500 casks. Both types of casks are designed to load 102 half-assemblies, which are arranged in the basket of a special configuration. In March 2000, the Ignalina NPP received its operational license, and regular loading of the CASTOR RBMK-1500 casks was started. The last CASTOR cask was loaded and transported to the storage site on September 20, 2000. At the same time, the licensing procedure for CONSTOR RBMK-1500 casks was under way, and regular loading of these CONSTOR casks started at the end of June 2001. The capacity of the existing SNF dry-storage facility at the Ignalina NPP is for 118 casks (20 CASTOR and 98 CONSTOR casks). The facility design operation time is until 2050.

As indicated above, during operation of the Ignalina NPP, 21,571 SNF assemblies were accumulated at the Ignalina NPP up to 2010. Of these 6,018 were accommodated within the existing SNF dry-storage facility at the Ignalina NPP. It was decided to build a new interim storage facility accommodating the remaining 15,553 SNF assemblies that are stored now in the water pools. This SNF will be loaded in the higher-capacity (182 fuel half-assemblies) CONSTOR RBMK-1500 casks. After the tendering process in January 2005, Consortia GNS (Gesellschaft für Nuklear-Service mbH**)**, RWE NUKEM GmbH, Germany, was awarded the contract. The Lithuanian Energy Institute, as the local subcontractor, is supporting the

Consortia in Environmental Impact Assessment Program and Report, and also Preliminary and Final Safety Analysis Reports preparation and licensing.

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

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

14.3.2.1 Strategy for disposal

The first Strategy on Management of Radioactive Waste in Lithuania, approved by the Lithuanian government in 2002, defines a number of activities related to spent-nuclear-fuel disposal:

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

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

In 2008, the Strategy for Radioactive Waste Management was revised, and subsequently the National research program for 2008–2012 was developed and approved by the Lithuanian Government (Government of Lithuania 2008).

With regard to EC Directive 2011/70/EURATOM, a new National program on the management of spent nuclear fuel and radioactive waste (with content prescribed in the Directive) was approved by the Government on December 23, 2015.

The strategic purpose of the program is as follows: through proper management of all existing and future radioactive waste (RW) and SNF in Lithuania, protect people and the environment against the harmful effects of ionising radiation and leave no undue burden for future generations. Implementation of safety principles requires that RW and SNF should be put in long-term isolation from people and the living environment, and that safety should be ensured by passive means. This could be achieved by disposing of SNF and RW in repositories. Storage of SNF and RW is a short-term solution that does not ensure safety, and it cannot be an alternative for waste disposal in a repository. The Program sets four goals:

1. Reduction of RW amount

2. Achievement of a high level nuclear safety, radiation protection and environmental protection while managing SNF and RW
3. Assurance of long-term safety of SNF and long-term RW
4. Striving to ensure transparency of SNF and RW management and provide objective information on waste management to the Lithuanian society

The 2015 National Radioactive Waste Management Development program foresees only one alternative for SNF and long-lived RW disposal, which is the implementation of a geological repository in a suitable geological formation in Lithuania. The construction and commissioning of the repository is planned for completion in 2066.

While developing the implementation plan of a geological repository in Lithuania, the analysis of the requirements defined in IAEA documents, related to site characterization, step-by-step development of the repository design, preparation and update of safety case and safety assessment at each step of the development of the disposal facility, and quality management, was carried out. The recommendations to support these requirements have been assessed. The process of the geological repository implementation in Lithuania could be divided into the following steps:

Step 0: Research studies of possibilities to implement a geological repository in Lithuania (2000-2015)

Step 1: Site selection for a representative investigation borehole (2016-2021)

Step 2: Site selection for additional boreholes

Step 3: Site selection for an underground research laboratory (repository)

Step 4: Repository site confirmation

Step 5: Repository design development

Step 6: Repository construction and preparation for operation

Step 7: Repository operation and closure

Step8: Post-closure period

An R&D program for period 2016–2021 will take into account the IAEA comments and recommendations (Chaplow et al. 2011). The following types of studies will be carried out: select the site for a representative borehole, develop the site descriptive model, develop or update a series of repository concepts, and develop safety case studies. The program for the representative borehole drilling and investigations will be updated, and the program will be demonstrated based on the preliminary safety case for a repository. Prior to the beginning of drilling of the borehole, this program will be reviewed by foreign experts.

14.3.2.2 Geological and hydrogeological investigations

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

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

A screening of the territory of Lithuania was performed based on the most important geological parameters describing the suitability of formations, such as simple tectonic structure, absence of intraformation aquifers, low neotectonic and seismic activity, good isolation, and favorable mechanical properties of the rock. Then, the areal distribution of these favorable formations was defined (Figure 14–1). The screening was also based on an evaluation of the general geological parameters—such as lithological homogeneity of the prospective layer, thickness, depth and lateral extent, and tectonic structure—of the candidate formation. It was concluded after these investigations that rock salt could not be regarded as having a high potential for disposal of SNF. The crystalline basement rocks were considered as one of the best candidates for a geological repository. The best prospects for the crystalline basement appeared to be located in the southeastern part of Lithuania (Figure 14–1), where these rocks are overlain by only 200–300 m of sedimentary cover. The prospects for clayey formations were reduced to only the Lower Cambrian Baltija Formation and the Lower Triassic, since these best fulfilled the requirements for depth, thickness, lithological composition, and homogeneity of sediments. Thus, the extended studies on media selection in 2001–2002 led to the conclusion that crystalline rock and argillaceous rocks are the primary candidates for a geological repository in Lithuania.

In 2003, activities were focused on more detailed studies of these geological media, because desktop studies were not sufficient for making a prioritization of these media. The direct observation and characterization of cores from reference wells were carried out for detailed lithofacies evaluation of candidate formations, and analysis of similarities and differences between the cores of the different lithofacies zones at different depths. At the same time, a sampling for a preliminary analysis of mechanical properties was carried out. After such investigations, among clayey formations, the Lower Triassic formation was selected as the first priority, and the Cambrian Baltija Formation as the second priority. With regard to crystalline rock, it was determined that an area of 100 km^2 could be found between major fracture zones at an acceptable depth and with a normal fracture content that fulfills the desired requirements both with respect to tightness and stability. Furthermore, there is broad experience and competence in the field of investigations concerning SNF disposal in crystalline rocks in Sweden. Thus, crystalline rocks were selected for further characterization in 2004.

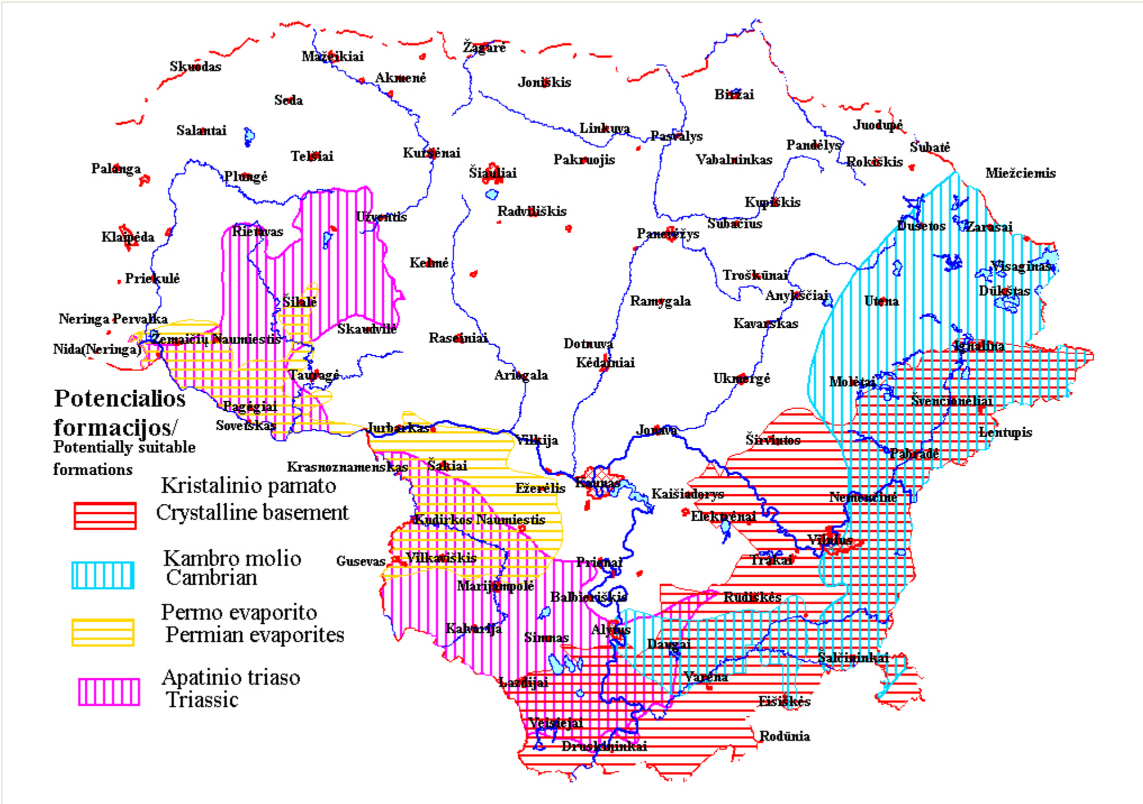

Figure 14–1. Geological formations in Lithuania that are potentially suitable for disposal of spent nuclear fuel (Kanopienė et al. 2005).

Laboratory investigations were performed on crystalline rocks to gather general information on the strength parameters of these rocks as a function of different petrological composition. Rock samples were taken from some old cores collected in boreholes drilled in southern Lithuania during deep geological mapping at a scale 1:200,000. Depths of the samples varied from 427 to 606 m and were accepted as representative of the candidate formation. The main characteristics of these crystalline rocks, such as density, porosity, unconfined compression strength, and rock temperature, were collected from different literature sources. All the data were filed systematically, and a simple statistical analysis was performed. Tectonics and hydrology of the crystalline basement were analyzed using existing archival information.

Based on a four-year period investigation of crystalline rocks and clay formations, the following is a summary of the conclusions:

- After compiling available geological data and inspecting selected drill cores, only the Lower Cambrian Baltija Group and the Lower Triassic clayey formations emerged as potential alternatives to crystalline rocks. They fulfill the requirements with respect to the depth, thickness, distribution, and lithological composition. The Lower Triassic has a higher priority than the Lower Cambrian.
- The crystalline basement rock can be considered as one of the best prospects for a geological repository. The best crystalline-basement prospects appear to be located in southeastern Lithuania where these rocks are overlain by only 200–300 m of a sedimentary cover. Extensive geological information is available for southeastern Lithuania, and this makes it very attractive for future studies.

- A dense network of faults is located in southern Lithuania, yet large-enough blocks, suitable for a geological repository, are present. Nontectonic blocks of order 10 × 10 km can be found. The best rock-type prospects are represented by cratonic (anorogenic) granitoid intrusions that, in some places, have formed large massifs. These rocks are the least damaged by tectonic activity. Other rock types (gneisses, mafic intrusions, and migmatites) have (in various places) formed only a weakly fractured block, which may also be a prospective repository site. Because of the very low seismic activity in this part of Lithuania, the tectonic stress is very low and should not affect tunnel construction. However, no instrumental observations of the stress field are available.
- Hydrogeological well tests indicate that the tectonic zones are water-saturated, whereas the homogeneous blocks are of very low permeability. The salinity of the formation water does not exceed 30 mg/L (except in some rare anomalies), which is favorable for engineered barriers. The water flow field of the basement is not well understood as yet.

In 2008–2012, several alternatives for the final disposal of SNF and long-lived RW in Lithuania were being investigated:

- A geological repository for disposal of SNF and other long-lived RW (at a depth of approx. 500 m)
- A repository at intermediate depths for long-lived low and intermediate level RW
- BOSS (Borehole disposal of sealed sources) boreholes for disposal of small amounts of spent sealed sources (depth of approx. 100 m)
- Deep boreholes for disposal of SNF in crystalline rocks (depth of 3–5 km)

Also analyzed were the advantages and disadvantages of two repositories (for SNF and long-lived RW separately) in different areas, two repositories (for SNF and long-lived RW separately) at different periods of time, and one repository (for SNF and long-lived RW).

The investigations were summarised in the report (GROTA 2010). The report shows that a feasibility study of the BOSS (BOrehole disposal of Sealed radioactive Sources) method for disposal in medium depth boreholes was carried out in order to evaluate the alternatives for final disposal of spent sealed sources (SSS) and long-lived radioactive waste (not disposing it of in a geological repository) in Lithuania. The study included an analysis of the suitability of hydrogeological, hydrogeochemical and geological conditions of the Ignalina NPP region (the Stabatiškė site and its surroundings) for drilling BOSS system boreholes. However, it was concluded that the BOSS method is possible only for disposal of fairly small amount of waste, hence other alternatives for long-lived RW disposal were analyzed simultaneously.

The study also contained an evaluation of possibilities to dispose of the irradiated graphite and other long-lived radioactive waste and SSS of the Ignalina NPP that cannot be disposed of in a near-surface repository at a moderate depth and in a geological repository. Based on the data, the analysis was focused at first on repository construction possibilities at the Stabatiškė site close to the Ignalina NPP and later on other alternative locations in Lithuania.

The Stabatiškė site for the drilling of the deep investigations borehole was proposed in order to concentrate the radioactive waste in the area of the Ignalina NPP. To this end, all actual data on the conditions of Stabatiškė site and geological-hydrogeological conditions of its surroundings was collected and structured. Stabatiškė site and its surroundings already have a number of boreholes of various natures (except from those for the approval of suitability of a deep geological repository); the deepest are of 728 m and 620 m depth. This allowed preliminary distinguishing of suitable and unsuitable intervals for disposal of the RW in the vertical geological cross-section.

The aforementioned report together with some additional documentation was submitted for review to IAEA. In 2011 the IAEA Expert Team analyzed the submitted information and summarized their comments in the mission report (Chaplow et al. 2011). Comments of the Expert Team touch upon analysis of geological investigations, selection of potential locations, safety evaluation, as well as work with the public. Several comments of greater importance to highlight:

- A preliminary safety case based on the adopted waste inventory and a defined disposal concept should be included in the draft report and used to inform the site selection process and the definition of requirements for further investigations. It is important that all parts of the geological sequence that might contribute to the safety performance of the repository are investigated, rather than focusing the investigations only on the potential host formation.
- The proposals for drilling a deep investigation borehole at the Stabatiškė site should be revised. The proposals for further investigations should be demonstrably based on the preliminary safety case for a repository and should include considerations of:
    - What information is needed from the investigations;
    - How this information will be obtained; and
    - How the information will be used.
- The Expert Team has concluded that the proposals to drill a new deep borehole are premature and are not currently justified in relation to the current status of the Program. The Expert Team recommends that the proposals should be reconsidered and revised.

14.3.2.3 Development of the Disposal Concept and Safety Assessment Studies

During 2001–2004, the Lithuanian Energy Institute, with support of Swedish experts, was working on development of the repository concept (Poškas et al. 2005a) and the generic safety assessment (Poškas et al. 2005b) of a repository in crystalline rock in Lithuania.

A detailed repository design is highly specific to waste type and its geological environment. Regardless of waste type, construction of the access and emplacement shafts and tunnels will involve the excavation of a substantial underground facility involving the removal of several hundreds of thousands of cubic meters of rock, and as much as millions of cubic meters for larger waste disposal programs. Geological repositories now being considered have underground dimensions varying from a few square kilometers to about twenty square kilometers, depending on the inventory of waste, its thermal output, and the repository design.

The proposed repository concept for Lithuania is based on the KBS-3 concept developed by SKB for disposal of SNF in Sweden. The KBS-3H design, with horizontal canister emplacement, is proposed as the reference design for Lithuania. The scheme of the repository is shown in Figure 14–2.

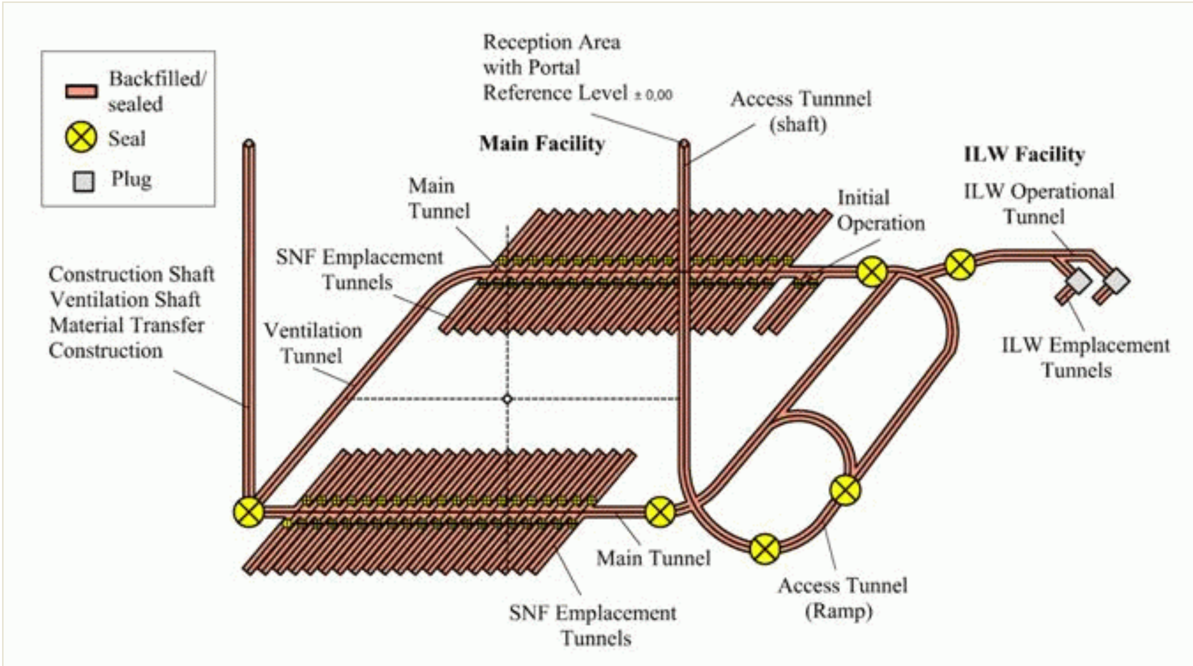

Figure 14–2. The scheme of the repository after final sealing and closure of the facility (Poškas et al. 2005a).

The generic repository concept for SNF disposal in crystalline rock in Lithuania envisions the waste disposal in horizontal emplacement tunnels with a diameter of 1.85 m and length of 250 m. The repository would be constructed in the crystalline basement at a depth of 300–500 m. Disposal canisters with RBMK-1500 SNF would be emplaced with a distance between canisters of 1.2 m. A copper canister is assumed for the disposal of SNF in the basement rock in Lithuania. The proposed canister for SNF is designed with two components: an outer corrosion protection of copper, and a cast iron insert with channels for the RBMK-1500 SNF half-assemblies. This is to improve the mechanical strength of the canister, as is being done in Sweden and Finland. The copper canister has a wall thickness of 50 mm and should be made of oxygen-free copper with low phosphorus content. The canister insert will be made of cast iron with a minimum wall thickness of 50 mm. Preliminary data for the RBMK-1500 SNF reference canister suggest a 1,050 mm diameter and a 4,070 mm length. One disposal canister can hold 32 RBMK-1500 fuel half-assemblies, and for Lithuanian SNF disposal, about 1,400 canisters would be necessary. The required area for the repository construction would be about 0.4 km^2.

The repository concept is described at the level of detail needed to perform a generic safety assessment and cost analysis. Determination of thermal evolution, criticality, and other important disposal characteristics for RBMK-1500 spent nuclear fuel emplaced in a copper canister supports the development of this concept.

Cost estimates for the disposal of SNF and long-lived intermediate-level waste in the crystalline basement in Lithuania is presented in Poškas et al. (2005a). This preliminary assessment is based on the experience accumulated during development of the Swedish KBS-3 concept, as applied to the Lithuanian case. The method used in analysis is based on the application of a concept known as the "successive principle," which has been used especially as a tool for managing uncertainties that may develop due to unforeseen events in the future. The input data for the calculations are obtained from the "most likely" costs, or the so-called reference costs, by means of conventional (deterministic) calculations. These calculations are based on a functional description of each facility, resulting in layout drawings, equipment lists, personnel forecasts, etc., under established fixed conditions but without allowances for variations and uncertainties. To provide guarantees to cover losses resulting from future unforeseen events, reasonable additional costs (cost variations) are included in the calculations. The influence of a specific variation on the costs is evaluated, and the result provides a mean value of the future costs and the standard deviation of the cost for a desired degree of confidence.

A generic safety assessment of the Lithuanian repository is presented in Poškas et al. (2005b). Because of the assumed similarities in the repository environment and repository concept, the selection of scenarios is based on experience from the safety assessment performed in Sweden. For this stage of generic safety assessment, only two scenarios were chosen: base scenario and canister defect scenario. In performing this safety assessment, Lithuanian parameter values were used as much as possible.

To analyze system evolution under the above-mentioned scenarios, thermal evolution, criticality, and dose assessment of the copper canister loaded with RBMK-1500 SNF were performed. Analysis of thermal-evolution effects at the top of the crystalline rock has shown that, over a period of one million years after deposition of the spent fuel, the heat released from the canister with RMBK-1500 SNF will have only a marginal impact on thermal conditions at the top of crystalline rock.

An essential component of the safety assessment was to calculate radionuclide release and doses to critical group members. The computer codes COMPULINK7, CHAN3D, provided by SKB were used to calculate radionuclide release through the near-field and far-field regions from the canister with an initial defect (canister defect scenario). For dose assessment, the computer code AMBER 4.4 (UK) was also used. The results from calculations showed that most of the analyzed radionuclides identified as safety-relevant will be effectively retarded in the near-field (by the bentonite buffer). Release rates of ^{135}Cs, ^{129}I, ^{99}Tc, and ^{226}Ra are the most significant. The release rates from the near-field are dominated by ^{135}Cs and ^{129}I. At longer times, the release rate is dominated by ^{226}Ra, which is formed by an in-growth from the chain decay of ^{238}U. Modeling of the transport of radionuclides through the far-field was performed for the most dominating radionuclides in the near-field. The values of parameters that have influence on radionuclide transport were selected from the study area in Lithuania, and if they were not yet available, values from the Beberg site (Sweden) were used. Results of total dose behavior demonstrated that the dose constraint of 0.2 mSv/y will not be exceeded in a period of a million years, and in fact will be lower by two orders of magnitude than this constraint. The total dose rate for the initial period is dominated by non-sorbing radionuclide ^{129}I; at later times it is dominated by ^{226}Ra.

Therefore, it should be emphasized that the existing experience is mainly gained from the international cooperation. As it was indicated above, an effective start-up of competence development was a multiyear cooperation project with Swedish International Project for Nuclear Safety (SIP) (“Competence development in the area of SNF disposal”), 2000–2005. Later on, to ensure effective transfer of knowledge and practices existing in other countries, the Lithuanian Energy Institute participated in various research projects coordinated by the International Atomic Energy Agency (IAEA), e.g., Neerdael and Finsterle, 2010) and the IAEA’s Coordinated Research Project “Treatment of Irradiated Graphite to Meet Acceptance Criteria for Waste Disposal,” 2010–2013. The institute’s researchers also took advantage of IAEA fellowships, training courses and workshops. To become a part of the common research community on nuclear issues in Europe, the Lithuanian Energy Institute takes part in numerous EC funded projects. EC 7th FP research project on “Treatment and Disposal of Irradiated Graphite and other Carbonaceous Waste (CARBOWASTE)” was launched in 2008 for a five-year period. The Lithuanian Energy Institute was involved in a large 30-partner consortium addressing retrieval, characterization, treatment, reuse, and disposal of irradiated graphite and other carbonaceous waste. Together with partners of another research project, “Fate of Repository Gases, 2009–2013 (FORGE),” Lithuanian Energy

Institute experts focused on understanding gas generation, its migration through the engineered and natural barriers of a geological repository, and consideration of these processes in quantitative assessment of repository performance. Within the EC 7th FP project "CArbon-14 Source Term" (CAST, 2013–2018), Lithuanian Energy Institute researchers, together with the experts from 32 other organizations, seek to develop understanding of the potential release mechanisms of carbon-14 from radioactive waste materials under conditions relevant to waste packaging and disposal in underground geological disposal facilities. The project focuses on release of carbon-14 from irradiated metals (steels, Zircaloys), irradiated graphite, and ion-exchange materials as dissolved and gaseous species.

Besides the research projects, Lithuanian Energy Institute experts were involved in coordination and support action type projects, such as "New MS Linking for an Advanced Cohesion in Euratom Research (NEWLANCER)" (2011–2013), and "Sustainable network for independent technical expertise for radioactive waste disposal (SITEX)" (2012–2014). Currently Lithuanian Energy Institute experts are also involved in an on-going EC 7th FP project, "Building a platform for enhanced societal research related to nuclear energy in Central and Eastern Europe (PLATENSO)" (2013–2016), and EC HORIZON 2020 project SITEX 2 (2015–2017).

The overview of the main research papers related to the performance and safety assessment of the geological repository in Lithuania is presented in Poskas at al. (2015)

14.4 Conclusions

Starting in 2000, research and international cooperation activities undertaken in the field of SNF disposal in Lithuania have resulted in:

- Prioritization of potential geological formations for SNF disposal in Lithuania
- Development of a proposal for a preliminary repository concept (including designs of the repository and a canister)
- Preliminary calculations of temperature distribution, criticality, and dose rate from disposal canisters, modeling of radionuclide and gas migration from the repository
- Greater knowledge of the field of SNF disposal by Lithuanian researchers and practitioners

The Lithuanian SNF disposal program is at the stage of initial site investigations and a preliminary facility design. The main points in the near-future activities in the field of SNF disposal include planning the implementation of geological repository, investigating and characterizing prospective regions, development of repository concept and general safety assessment, and increase of competence through international cooperation, and maintaining ongoing competence for the long term.

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14.6 Acronyms

BOSS—Borehole disposal of Sealed radioactive Sources

CARBOWASTE—Carbonaceous Waste

EIA—Environmental Impact Analysis

EU—European Union

FORGE—Fate of Repository Gases

GNB—Gesellschaft für Nuclear Behalter

IAEA—International Atomic Energy Agency

LILW-LL—Long-Lived Low/Intermediate Level Waste

LILW-SL—Short-Lived Low/Intermediate Level Waste

NEWLANCER—New MS Linking for an Advanced Cohesion in Euratom Research

NPP—Nuclear Power Plant

NSR—Near-Surface Repository

PLATENSO—Building a platform for enhanced societal research related to nuclear energy in Central and Easter Europe

RATA—Radwaste Management Agency

RW—Radioactive Waste

SAR—Safety Analysis Report

SIP—Swedish International Project for Nuclear Safety

SITEX—Sustainable network for Independent Technical Expertise for radioactive waste disposal

SNF—Spent Nuclear Fuel

SSS—Spent Sealed Sources

VATESI—The State nuclear power safety inspectorate

Chapter 15

Mexico's Plans for the Disposal of Radioactive Waste in a Deep Geological Repository

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ABSTRACT: The start of operations of a deep geological repository for high-level wastes is still far in the future for Mexico. The proposed schedule of activities for the implementation of such a facility in the country show that siting and licensing should begin in 2045, construction in 2055 and operation in 2065. However, as has been the experience of many countries that are pursuing this kind of repository, the planning and preparations required for it are not trivial and take long periods of time to develop. To be able to achieve the intended goals on the dates proposed, the most important activities have to start in the short-term future. Mexico's intent is to establish a modern radioactive waste management system for all different types of radioactive waste, and central to that plan is the implementation of a deep geological repository for the disposal of high-level and long-lived wastes.

This chapter describes the plans and activities that Mexico is contemplating in order to have a deep geological repository operational by 2065. Although work has not surpassed the planning stages, this proposal is a summary of all tasks and activities that have to be planned and prepared in order to begin work on the subject and have the required personnel, expertise, resources and infrastructure in a timely

manner. The considerations enlisted here, although well-known to countries that have already begun work and are advanced in activities such as the site selection and characterization process, can be of great interest to other countries that find themselves in the early development stages of a national geological disposal program and are in need of guidance on the best way to proceed and the best strategies to pursue.

15.1 Background

Mexico is engaged in establishing a modern Radioactive Waste Management system. As part of that system, plans are being made to set up solutions for the long term disposal of the different types of radioactive wastes. The effort involves all major organizations in the Mexican Nuclear Industry: the Secretariat of Energy (SENER – Secretaría de Energía), the National Power Utility (CFE – Comisión Federal de Electricidad), owner of the Laguna Verde Nuclear Power Plant (LVNPP), the National Commission for Nuclear Safety and Safeguards (CNSNS – Comisión Nacional de Seguridad Nuclear y Salvaguardias), and the National Institute for Nuclear Research (ININ – Instituto Nacional de Investigaciones Nucleares).

The regulatory framework of the Mexican energy sector is going through important changes. With the "energy reform" of 2013 and 2014, the fundamental laws and other legislation were modified to allow for private participation in strategic areas that were until then reserved to the Mexican State. This new regulatory framework is intended to attract investments and modernize the energy sector in order to boost the economy by improving the industrial capacity and the national productivity, and by enhancing Mexico´s competitiveness with transparency and a strong regard for the environment.

Both the oil industry and the electricity sector underwent modifications, and in this last area it took shape in the establishment of a deregulated market and the opening up of possibilities for private generation. Nuclear power, however, remains a strategic area controlled by the Mexican State. This means that all the activities related to the use of nuclear energy to produce electricity can only be performed by the State or a state owned company, including waste management and confinement.

A reform of the nuclear sector could be the next to come. A new nuclear law would bring a much needed update to a 30-year-old regulation. It could incorporate new schemes for commercialization of radioactive minerals, the implementation of export controls, and the creation of a waste management organization, among other things, and would open new possibilities that could bring more dynamism to the industry.

15.2 Current Status of Radioactive Waste Activities In Mexico

15.2.1 Present Legal Framework

The Mexican Regulation regarding nuclear matters is comprised of the following:

- Article 27 of the Political Constitution of the Mexican United States. It establishes that all natural resources within the Nation's territory are owned by the State, and the exploitation of nuclear fuels is reserved for the State. It also states that nuclear energy can only be used for peaceful purposes.
- Regulatory Law of Constitutional Article 27 on Nuclear Matters, the "Nuclear Law." First issued in 1985, it regulates exploration, exploitation and benefit from radioactive minerals, including exploitation of nuclear fuel, use of nuclear energy, scientific research, nuclear technology, and radioactive waste and spent nuclear fuel management.
- International Treaties. Mexico is a part of, and has signed, a number of international treaties in nuclear matters. The list of the main international treaties signed by Mexico is shown in Table 15–1.

- General Code of Radiological Safety (RGSR – Reglamento General de Seguridad Radiológica). This code establishes criteria and basic requirements related to radiological safety matters, including: the Dose Restriction System, obligations of radiological safety personnel, radiological incidents and accidents, measures in case of imminent danger to population, authorizations, licenses or permissions, inspections, and sanctions.
- Mexican Official Standards (NOM – Normas Oficiales Mexicanas), Nuclear Series (NUCL). These Standards are of compulsory compliance in the country. Table 15–2 lists the Standards related to waste management currently in existence.

Table 15–1. International Treaties undersigned by Mexico in the nuclear field.

Title	Signature date	Effective date (Entry into force)
Treaty of Nuclear Weapon Proscription in Latin America	February 14th, 1967	September 20th, 1968
Treaty of Not Nuclear Weapon Proliferation	July 1st, 1968	January 21st, 1969
Agreement to safeguard implementation related to the Treaty of Nuclear Weapon Proscription in Latin America and Treaty of Not Nuclear Weapon Proliferation	September 27th, 1972	September 14th, 1973
Subsidiary Arrangements related to the Agreement to safeguard implementation related to the Treaty of Nuclear Weapon Proscription in Latin America and Treaty of Not Nuclear Weapon Proliferation	September 17th, 1972	September 14th, 1973
Convention about Early Notification of Nuclear Accidents	September 26th, 1986	June 10th, 1998
Convention about Mutual Assistance in case of Nuclear Accident or Radiological Emergency	September 26th, 1986	June 10th, 1998
Convention about Physical Protection of Nuclear Materials	April 4th, 1988	May 4th, 1988
Vienna Convention about Civil Liabilities for Nuclear Damages	April 25th, 1989	July 25th, 1989
Convention about Nuclear Safety	November 9th, 1994	October 24th, 1996
Additional Protocol for the Agreements between Mexico and IAEA to safeguard implementation	March 29th, 2004	March 4th, 2011
Convention Amendment about Physical Protection of Nuclear Materials	Not applicable	August 1st, 2012

Table 15–2. List of NOMs in radioactive waste management.

Number	Title
NOM-004-NUCL-2013	Radioactive Waste Classification
NOM-008-NUCL-2011	Radioactive Contamination Control
NOM-018-NUCL-1995	Methods to Determine Activity Concentration and Total Activity in Radioactive Waste Packages
NOM-019-NUCL-1995	Requirements for the Near Surface Disposal of Low-level Radioactive Waste Packages
NOM-020-NUCL-1995	Requirements for Radioactive Waste Incineration Facilities
NOM-021-NUCL-1996	Lixiviation Test for Solid Radioactive Wastes
NOM-022/1-NUCL-1996	Requirements for Low-level Radioactive Waste Disposal near surface. Part 1. Siting
NOM-022/2-NUCL-1996	Requirements for Low-level Radioactive Waste Disposal near surface. Part 1. Design
NOM-022/3-NUCL-1996	Requirements for Low-level Radioactive Waste Disposal near surface. Part 1. Construction, Operation, Closure, Post-Closure and Institutional Control.
NOM-036-NUCL-2001	Requirements for a Treatment and Conditioning Radioactive Waste Facilities
NOM-028-NUCL-2009	Radioactive Waste Management in Radioactive Facilities where Open Sources are used.
NOM-039-NUCL-2011	Specifications for Exemption of Ionizing Radioactive Source and the Practices which use them
NOM-035-NUCL-2013	Levels to Classify a Solid Waste as Radioactive Waste
NOM-041-NUCL-2013	Annual Incorporation Levels and Discharge Concentrations

15.2.2 Current Radioactive Waste and Spent Fuel Management Facilities in Mexico

As depicted in Figure 15–1, there are currently five radioactive waste and spent fuel management facilities in Mexico.

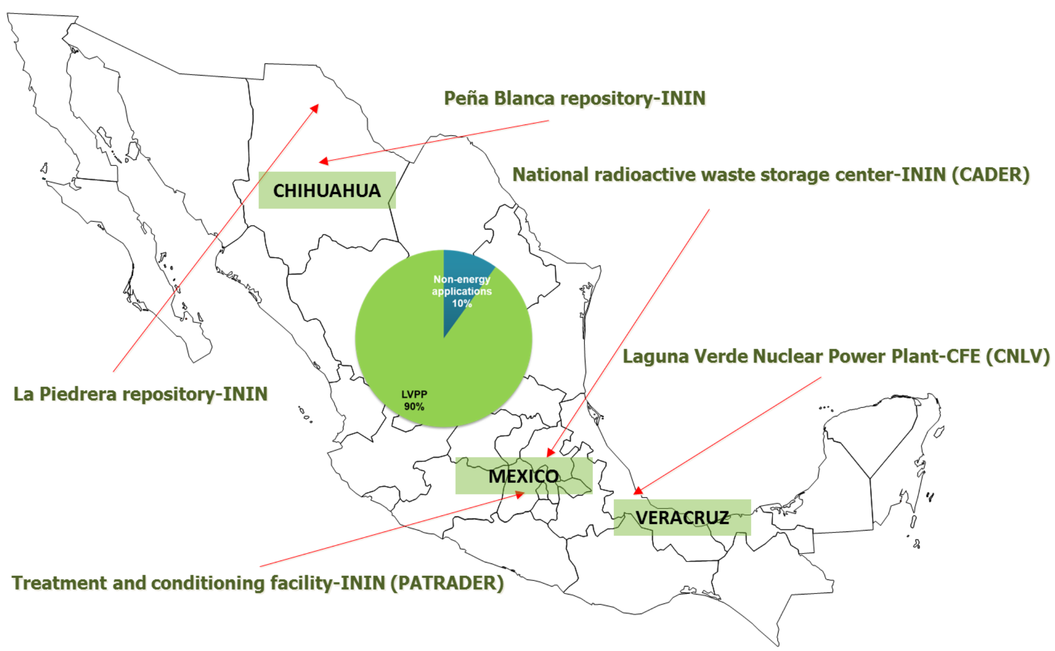

Figure 15–1. Spent nuclear fuel and radioactive waste management facilities in Mexico. The figure also depicts percentage of waste generated in energy and non-energy applications.

15.2.2.1 Radioactive Waste Treatment Plant (PATRADER – Planta de Tratamiento de Desechos Radiactivos)

Operated by ININ and located at its premises in the State of Mexico, this facility processes and conditions the radioactive wastes generated by the use of radioactive material in the medical, industrial and research fields.

15.2.2.2 Radioactive Waste Storage Facility (CADER – Centro de Almacenamiento de Desechos Radiactivos)

A temporary storage facility operated by ININ and located in Santa Maria Maquixco, State of Mexico. It stores low and intermediate level radioactive wastes in solid form generated in non-energy activities in Mexico in the medical, industrial and research sectors, as well as spent radioactive sources.

Figure 15–2. Entrance to the CADER

15.2.2.3 Laguna Verde Nuclear Power Plant

The Laguna Verde Nuclear Power Plant (LVNPP) is located in the Municipality of Alto Lucero, State of Veracruz. The Nuclear Power Plant is authorized to process, condition and store all radioactive wastes from its own operation. All storage facilities at the Laguna Verde NPP are temporary in nature (as is usual for NPPs) and for low- and intermediate-level waste include the Dry Solid Radioactive Waste Storage (DDRSS – Depósito de Desechos Radiactivos Sólidos Secos) for dry solid waste and the At Site Temporary Storage (ATS – Almacén Temporal en Sitio) for wet and solid wastes.

For storing spent nuclear fuel, there are spent fuel pools in each of the two reactor buildings. As of May 2015, 85% and 70% of the capacity of the two fuel pools was used up. Therefore, the plant initiated in the first quarter of 2015 the construction of an Independent Spent Fuel Storage Installation (ISFSI) with the capacity to store all fuel assemblies during its 30-year expected lifetime. The facility will be able to hold 130 double application dry casks made of inner steel cylinders and external concrete containers. Initial loading of 13 storage modules began in June 2016 and will be finished before the year's end. The expected lifetime of the fuel casks is 40 years under the climatic and environmental conditions of the Laguna Verde site.

15.2.2.4 La Piedrera

"La Piedrera" repository is located south of Ciudad Juárez, in the State of Chihuahua. The facility was built to dispose of the radioactive wastes generated as a result of the 1983 accident occurred in Ciudad Juárez where a teletherapy cobalt-60 source was melted with iron and used to fabricate rebar.

15.2.2.5 Peña Blanca

The "Peña Blanca" uranium mine is located north of Ciudad Aldama, in the State of Chihuahua, and presently stores uranium mining tails and residues from the operation and closure of the uranium and molybdenum extraction plant in Ciudad Aldama.

15.3 The Need for a National Policy for the Management of Radioactive Waste

Radioactive materials have been used in Mexico since the 1950s. The National Regulatory body, the CNSNS, supervised the safe handling and management of radioactive residues and established an efficient regulatory framework for that purpose. However, at present all regulations regarding radioactive waste management in the country are focused on temporary storage solutions.

The growth of nuclear applications in medicine, industry, research, and operations of two nuclear power reactors for the past 25 years have imposed the need for the country to establish a radioactive waste management plan for the long-term, a plan that will include not only temporary storage facilities, but also installations for the definitive disposal of different types of radioactive and nuclear wastes.

To accomplish this, SENER is coordinating the efforts to define and implement both the National Policy and the National Strategy for the Management of Radioactive Waste. At this stage of maturity of the Mexican nuclear industry, a national policy is needed to ensure the public that the State and its different organizations are currently taking care of the storage of radioactive wastes and will continue to do so in the long-term future. As well, a national policy is essential to assign responsibilities among the organizations involved in order to achieve the minimum safety criteria for protection to the public and to the environment in accordance with international standards. It is also necessary, through a national policy, to define and establish financing mechanisms for radioactive waste management activities now and in the future, minimizing or eliminating the burden of cost on future generations. And finally, but no less important, is the need to define standardized management criteria for all public and non-public entities involved, with the purpose of homogenizing methodologies and optimizing available resources.

One of the central concepts of the proposal for a national policy that is presently being analyzed and evaluated includes the establishment of a Waste Management Organization in the country. Following the experience of other countries worldwide with nuclear power applications programs, it has become clear that there is a need for a waste management organization that will develop and implement an efficient and unified strategy for the final disposal of significant quantities of radioactive wastes and spent fuel coming from different sources. Unlike countries where the management of radioactive wastes is entrusted to existing organizations such as research institutions, a dedicated waste management organization would be responsible for the management of these kinds of waste, and would have a budget committed for that purpose. SENER and other nuclear industry entities have discussed the need to establish a Waste Management Organization in Mexico.

SENER, the LVNPP, CNSNS and ININ are currently developing a proposal for a new policy and plans for the long-term management of radioactive wastes. .The expert opinion of international advisors has also been taken into consideration as a result of workshops and projects implemented in collaboration with the European Commission (EC) and the International Atomic Energy Agency (IAEA). For example, a collaborative project with the EC was conducted from 2012 to 2015, in which experts from a consortium of well-known enterprises, specializing in waste management activities (ENRESA, Empresarios Agrupados, IBERDROLA and Westinghouse from Spain; Belgoprocess from Belgium; and COVRA from the Netherlands), provided consulting services to Mexican specialists and produced a proposal for a national strategy. The plan for the disposal of radioactive wastes described below in this Chapter was developed by these organizations, and is part of the current proposal for a National Policy for the Management of Radioactive Waste in Mexico.

15.4 The Plan for the Disposal of High-Level and Long-Lived Radioactive Wastes

High-level and long-lived radioactive waste must be contained and isolated from humans and the environment for thousands of years. It must be kept away from the biosphere and it must be protected in such a way that makes intentional human intrusion difficult or close to impossible to achieve without special technical means. Precautions should be taken to ensure that any eventual release of radioisotopes from the repository does not pose an unacceptable risk to the population or the environment. A repository of the deep geological type does not require any active management or intervention to provide safety after its period of operation, closure and institutional surveillance. The principal idea of a design of a deep geological repository is the integrity and confinement capacity of the geosphere, as many mineral deposits and paleontological sites in different areas of the world have remained stable and unchanged for millions of years. In this sense, deep geological disposal is the only option that can ensure continued safety of high-level and long-lived waste for as long as it remains a hazard.

The repository takes advantage of several protective characteristics of the natural geological environment to achieve the long-term safety, which is enhanced by designing and placing additional engineering barriers and by conditioning the waste to be in a stable form.

The use of deep geological formations for the disposal of high-level and long-lived radioactive wastes was first proposed by the National Academy of Sciences of the United States in 1957 [11], and has been the basis for the disposal programs of most countries belonging to the Organization for Economic Co-operation and Development's Nuclear Energy Agency (OECD-NEA) ever since.

Mexico is proposing to dispose of high-level and long-lived wastes in a deep underground stable geological formation. A deep geological repository could be emplaced in a site within the country's territory, or alternatively, Mexico could dispose of its wastes in an international co-owned, multi-shared facility.

An international, multi-shared facility is a beneficial solution for countries that have small nuclear programs, especially if located in the same geographical area. This could be the case for Latin American countries pursuing nuclear power programs. Alternatively, Mexico is located next to the United States and Canada, both countries with large nuclear power programs, which brings the possibility of arranging a common North American solution. However, because such possibility is currently uncertain, it is necessary for the country to continue with its plans for the development and implementation of its own facility.

Although the realization of a safe long-term disposal system is possible, the construction of this system will require the supervision from a responsible organization and the continued supply of technical, human and financial resources. Operational costs and maintenance activities will be necessary for long periods of time. There is a factor of uncertainty in the future existence of a disposal system like this and the possibility of placing an unwarranted responsibility on future generations if the strategy for the establishment of a repository is not well planned in advance.

The responsibility for the site selection, design, construction and operation of such a facility in Mexico would fall on the national waste management agency described before. To finance these activities, a fund will need to be established that begins to accrue funds in the short term, given that the cost for such installations run into the billions of dollars. Not providing the sufficient funding for these activities would breach the ethical principle of not imposing undue burden on future administrations or generations.

The waste management organization will be responsible for establishing a program for the construction of an underground disposal facility that should include the creation jointly with the government of a dedicated financing fund and the procurement of the required human resources and technical capabilities.

Important elements that should be considered to structure a long-term program for geological disposal are:

- Planning a site selection process to find suitable candidate geological sites based on a well-established set of criteria.
- Development of a conceptual or basic design of the deep geological repository.
- Development of capabilities to carry out state-of-the-art safety assessments of the disposal system.
- Establishment of a comprehensive Research and Development (R&D) plan to fulfill all aspects of the design of the repository, including acquiring the required technologies and developing personnel training programs for the characterization of the potential designated sites and the building of the repository.
- Addressing societal issues and needs when carrying-out the site selection process and understanding the need of having a strong public support or acceptance for the resulting site.

Considering all the concepts explained previously, it is clear that a program for the implementation of a geological repository for high-level and long-lived wastes will require a long period to mature. Finland, for example, which became in the end of 2015 the first country in the world to obtain a license from its Government to begin construction of a final disposal facility for used nuclear fuel, has begun the extensive R&D work leading to the implementation of the repository at Olkiluoto more than 40 years ago. Construction work is expected to begin in late 2016. For Sweden and France to reach the current point of development required more than three decades of work: Sweden started in the early 1980s and France in the late 1970s. In Sweden, the Swedish Nuclear Fuel and Waste Management Company (SKB) submitted an application to the Radiation Safety Authority (SSM) to build the country's first repository for used nuclear fuel at Forsmark in March 2011, with a final decision expected by 2017. In France, an application to the regulators for the construction of the Industrial Center for Geologic Storage (CIGEO), an underground repository for high-level and long-lived wastes along the border between the Meuse and Haute-Marne departments, is expected to be submitted in 2017. Other countries like the US, Canada or Switzerland, who pursue repositories of the geological kind, also began their programs long time ago.

15.5 Types of Waste Considered for Disposal

The current proposal in Mexico is for both HLW and LL-LILW to be disposed in a deep geological disposal repository. Both HLW and LL-LILW require similar levels of isolation from the public and from the environment, and therefore it would be economically advantageous to dispose these two types of waste in the same facility. It is also advantageous in terms of reaching the public acceptance, which is the reason Germany, France Sweden, Switzerland and the Netherlands favored such solution.

It should be considered that there are also potential safety issues associated with the combined disposal of these wastes, notably issues related to heat and gas generation, to the chemistry of the waste and the interactions with the near field environment. These issues have to be researched and addressed as part of the design and development phase of the repository.

15.5.1 High-level waste

In Mexico HLW is defined as a category of waste with high activity, with presence of alpha-emitters and significant heat generation. It could include the long-lived wastes arising from the reprocessing of SNF or actual spent nuclear fuel bundles, if reprocessing is not chosen as an option for the Mexican nuclear power plant's fuel cycle. Waste products of the reprocessing of nuclear fuel would be subject to a vitrification process and would finally end up in the form of packaged vitrified HLW.

HLW would also include materials coming from the interior of the nuclear reactors (reactor internals) that are removed after decommissioning of the power plants. The nuclear power plants are the major source of HLW as a result of their operation (in the form of SNF), their decommissioning, or as a result of the reprocessing of SNF.

Mexico does not have yet an official policy on reprocessing. However, the disposal principles of reprocessing products and SNF are conceptually very similar. It is possible that at some point, Mexico decides to dispose of both reprocessing products and also SNF directly in a deep geological repository.

15.5.2 Long-lived low and intermediate level radioactive waste

According to the IAEA definition, the LL-LILW has the following characteristics:

- A significant concentration of beta and gamma emitter radionuclides with a half-life greater than 30 years
- Alfa-activity in the range between 400 Bq/g to 4,000 Bq/g
- Heat generation below 2 kW/m^3, meaning heat dissipation is not an issue during handling and transportation.

LL-LILW is generated in nuclear fuel cycle activities, such as fuel reprocessing, reactor operation and reactor decommissioning. It is also generated in medical and industrial applications and in nuclear research. LL-LILW from the nuclear fuel cycle can be materials contaminated by the primary reactor coolant or activated by the neutron flux, such as control rods, core grids or core shrouds. From the fuel cycle there is also fuel assembly reprocessing waste like fuel cladding hulls, spacers and liquid waste contaminated with activated particles generated due to the dissolution of spent fuel pellets. LL-LILW from research, medical, industrial and other uses include liquid and solid waste such as concentrates, precipitates, organic liquids, laboratory materials and equipment, disused glove boxes and filters. Other sources of LL-LILW are disused components of research reactors, instrumentation from irradiation experiments, samples from reactor material monitoring programs and decommissioning waste. In general, long-lived waste is contaminated by the whole spectrum of radionuclides generated during reactor operations and include actinides, fission and activation products.

LL-LILW requires stricter disposal solutions with more robust engineering barriers than those for short-lived LILW disposal. Whereas short-lived LILW can be disposed of in a near surface facility, according to the IAEA recommendations and the practical experience obtained in other countries, LL-LILW need to be disposed in a geological repository. A geological repository of only 100 meters deep could be sufficient to effectively isolate some categories of LL-LILW.

15.6 Waste Volume Estimations

The estimation of the volume of wastes to be disposed of in the deep geological repository is summarized in Table 15–3, including the HLW, SNF and LL-LILW that will be produced in the current and future nuclear power plants in Mexico. The wastes produced as a result of the decommissioning of all the nuclear power plants considered is also taken into account. The assumptions for the estimations are that the current power plant (with two reactors) will operate for a total extended period of 60 years, and that 3 new power plants with two reactors each will be constructed and operated in Mexico in the future. As a matter of course, there will remain some uncertainty regarding the real magnitude of the nuclear program in the long-term future and whether a policy for reprocessing is adopted or not. But it is considered that these factors would only impact the final capacity of the underground repository and could be dealt with adjustments to the storage volume either in the design phase or once the facility is already in operation. The bulk of the investments in such installations are concentrated in the excavation and construction of the access amenities; after those are taken care of, expanding the capacity of the underground chambers or boreholes by a fraction will not impact the overall budget in any significant way.

Table 15–3. Estimation of the volume of wastes that will require disposal.

SCENARIO	SNF (tU)	SNF (Fuel assemblies)	LL-LILW (m^3)
LVNPP + 8 ABWR reactors (Operating for 60 years)	16,130	11,879 BWR 77,736 ABWR 89,615 Total	1,139

15.7 Expected Characteristics of Geological Formations and Repository

The typical depth for geological repositories varies between 100 m and more than 1000 m, depending on the hosting rock characteristics and the type of waste to be confined. Some other desired characteristics of the host formation are:

- Sufficient depth for an effective isolation of the repository from the biosphere, so that the transport of contaminated groundwater and gases to the environment is reduced to a minimum.
- Sufficient depth to reduce the chances of potential human intrusion.
- Isolation from major disrupting natural phenomena in the long term, like the effects of erosion.
- Slow and predictable changes in the geological conditions of the host formation.

Studies of various types of geological formations have been made around the world to investigate their suitability to host a repository. Some of the favored formations are:

- Hard rock formations, such as granitic or other crystalline formations. These are being studied as possible repository sites in Canada, Finland, Hungary, Japan, Sweden, and were also studied in the past in Spain and Switzerland;

- Clay formations, ranging from over indurated clays to mudstones. Countries that are studying these formations are Belgium, France, Japan, the Netherlands and Switzerland.
- Salt formations. Salt layers in the form of salt domes are being assessed in Germany and the Netherlands. This type of formation was also studied in the past in Spain and the US.
- Volcanic formations. Volcanic formations under consideration are tuff and basalts. The former Yucca Mountain project was situated on a welded tuff formation.

15.8 Repository Design

Among the waste management organization's tasks, one of the most important will be the development of the conceptual design of a deep geological repository. The conceptual design will use the latest technologies available and will be based on geological data collected from the potential repository sites and the specific characteristics of the waste intended for the disposal.

There are two main configurations planned for underground repositories: galleries and boreholes. Galleries are relatively wide "rooms" built underground to place the waste. In the borehole type facility, the waste is placed inside small diameter holes bored either vertically or horizontally branching from underground corridors.

An important component of the repository concept is the engineered barriers to be designed to complement the host rock formation to safely and efficiently contain the wastes—in particular, to minimize heat generation, gas generation, criticality, radiation fields and leaching propensity. Typical engineered barriers considered for underground disposal are:

- Metal containers, also called canisters or over-packs. These are designed to maintain the wastes' integrity for the first 1,000 years of the repository.
- Buffer material. Used around the canisters to fill up voids and fractures in the rock formation and to prevent water from flowing around the containers.
- Backfilling material. Used to fill transport or access galleries.
- Seals. Sort of a "plug" to provide isolation of galleries or boreholes from transport and access tunnels.

Depending on the type of formation, additional barriers would have to be included; for instance, in the case of water bearing formations or unsaturated formations.

15.9 Site Selection

The development of a geological repository facility will require identifying potential repository sites. A site selection process will need to be structured to systematically evaluate the geological characteristics of different candidate sites. According to the IAEA [5], the siting selection process for a radioactive waste disposal facility includes the following stages:

- Conceptual and planning stage
- Area survey stage
- Site investigation stage of detailed site specific studies and characterization, and
- Site confirmation stage.

The site selection detailed geological investigations should also include preliminary safety evaluation, analysis of potential infrastructure resources, transportation issues, and stakeholder engagement. The site to be selected should have favorable geologic conditions to isolate the waste, taking into account that the disposal site should be constructed in the location to be approved by the stakeholders and the inhabitants of nearby communities.

In the first stages of site selection there will be different possibilities for investigations. As said before, there are different rock types that could host a deep repository, like granite, salt, clay and tuff. A country as large as Mexico may have several types of most of these rocks. A national screening process will lead to the preselection of a number of sites for further investigation. The potential for re-using existing mines is a possibility that should not be overlooked.

During the characterization and site confirmation stages, most of the work is focused on the development of a license application. It is a period that usually takes from 10 to 20 years and involves moving from a general level of knowledge obtained in the previous stage, to a site-specific level of knowledge and detailed safety assessments, incorporating engineering designs and the infrastructure required for submittal of the license application, construction and operation.

15.10 Research & Development Plan

R&D will be an important part of the waste management scheme in Mexico. It is needed to support all stages of waste management and to complement the technological needs that the industry is not able to provide. Not all activities in waste management can be considered technically mature at present, so a significant dose of in-house development will be needed to solve the technical problems that arise during the implementation of solutions for waste handling. Of all the phases in waste management, it is precisely in the areas of HLW and LL-LILW handling and long-term disposal, where the need for research and development are higher.

A comprehensive R&D plan will be formulated along with the conformation of different specialized expertise groups or teams to perform research in areas such as:

- Development of conceptual models for a deep geological repository
- Site characterization and site selection
- Long-term performance of repositories in different geological formations, and modeling and simulation
- Site confinement
- Design and testing of engineering barriers
- Waste and repository monitoring systems
- Waste characterization,
- Waste decontamination, waste minimization
- Behavior of waste under long-term storage conditions
- Container systems
- Transportation of radioactive waste
- Dismantling of facilities

R&D is an area that can benefit greatly from international collaboration, and for a country like Mexico, who is in the beginning phase of research in waste handling activities, it could be advantageous at a given point to participate in an international joint effort. International cooperation has been a beneficial mechanism for extending and maximizing common scientific knowledge and for reducing costs by means of collaboration, technology transfer and experience sharing.

International organizations such as the International Atomic Energy Agency (IAEA), the Organization for Economic Co-operation and Development's Nuclear Energy Agency (OECD-NEA), and the European

Atomic Energy Community (EURATOM), offer collaboration alternatives. One such possibility is that of the IAEA's Underground Research Facilities Network (URF Network) where Member States who own underground research facilities and related laboratories dedicated to the investigation of geological disposal of radioactive waste, offer their facilities for use by other Member States. The facilities are available for personnel training, demonstration of waste disposal technologies, scientific visits and collaboration projects for the joint solution of key technical issues.

15.11 Time Frame for the Design and Construction of a Deep Geological Repository

The major strategic objective for the future Waste Management Organization in Mexico should be the development of a deep geological facility within a reasonable timescale of 50 years, by the year 2065. By that time, the current power plant (LVNPP) will have been shut down for 10 years (considering a total life span of 60 years). Also by that time, there will be spent fuel from the LVNPP and from other newer plants in temporary dry storage with a wide range of cooling times of between 10 and 75 years. The timeline for different events related to the disposal of HLW and LL-LILW in a deep geological repository is shown in Table 15–4.

Table 15–4. Timeline of the planned disposal of HLW and LL-LILW in a deep geological repository.

EVENT	YEAR
Start of operations LVNPP temporary SNF storage	2017
Site selection and licensing	2045-2055
Construction of deep geological repository	2055-2065
Operation of deep geological repository	2065-2085
Dismantling of surface facilities, deep geological repository	2075-2080
Institutional control	2080-

15.12 Closing Remarks

Mexico, as an owner of a nuclear power plant and an active user of radiological materials for peaceful applications, has a responsibility to plan for the safekeeping of the resulting residues at present and in the long-term future. The first steps have been taken with the planning work leading to proposals for a national policy and for a national strategy for the management of spent nuclear fuel and radioactive waste. The approval of these documents would later pave the way for the implementation of a Waste Management Organization. These activities are key to continue the modernization process of the waste management system in Mexico. With these in place, it will be possible then, to set about the task of recruiting the required human resources, develop the necessary training, R&D programs, and begin work for site selection and for the design and construction of a deep geological repository.

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15.15 Acronyms

ATS—Almacén Temporal en Sitio - Site Temporary Storage for wet and solid wastes.

CFE—Comisión Federal de Electricidad - National power utility

CNSNS—Comisión Nacional de Seguridad Nuclear y Salvaguardias - National Commission for Nuclear Safety and Safeguards

DDRSS—Depósito de Desechos Radiactivos Sólidos Secos - Dry Solid Radioactive Waste Storage

HLW—High-Level Radioactive Waste

ININ—Instituto Nacional de Investigaciones Nucleares - National Institute for Nuclear Research

ISFSI—Independent Spent Fuel Storage Installation

LL-LILW—Long-Lived Low and Intermediate Level Radioactive Waste

LVNPP—Laguna Verde Nuclear Power Plant

NEA—Nuclear Energy Agency

NOM—Normas Oficiales Mexicanas - Mexican Official Standards

NUCL—Nuclear Series

R&D—Research and Development

RGSR—Reglamento General de Seguridad Radiológica - General Code of Radiological Safety Code

SENER—Secretaría de Energía -Mexican Nuclear Industry: the Secretariat of Energy

SNF—Spent Nuclear Fuel

Chapter 16

Development of the Slovak Deep Geological Disposal – Status In 2015

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ABSTRACT: The paper describes an infrastructure of nuclear energy utilization in the Slovak Republic and the geological disposal development within its frame. Three stages of the repository are identified and described: the first stage (1996-2001) of initial investigations, the second (2002-2012), when the geological disposal development was abandoned and limited mainly to a few geological investigation activities, and the current stage beginning in 2012. Establishing five localities in the both crystalline and sedimentary rock environments is a main output of the first two stages. In the near future, the development will continue in two main directions: site selection activities and communication with affected municipalities, including creation of the legislative framework for incentives for communities affected by the repository implementation. The milestones for implementation of geological disposal in Slovakia are: siting the repository – 2030, and commissioning – 2065.

16.1 Introduction

There are two nuclear localities in the Slovak Republic: Jaslovske Bohunice and Mochovce (marked as EBO and EMO respectively in Figure 16–1). In Jaslovske Bohunice, the first Czecho-Slovak nuclear power plant (NPP) A1 (HWGCR, 150 MW_e, commissioned in 1972, shut down in 1979) is currently in the second stage of decommissioning. Similarly, the NPP V1 (double-block PWR/WWER, 2 x 440 MW_e commissioned in 1979, 1980, and shut down after Governmental decision in 2006, 2008) is also in decommissioning at the present time. The NPP V2 (double-block PWR/WWER, 2 x 440 MW_e commissioned in 1984, 1985) continues to operate. Additionally, Jaslovske Bohunice also has the spent fuel storage facility (see below) and the facility comprising the waste management technologies on the site called "Technologies for treatment and conditioning of radioactive waste".

In Mochovce, there are two double-block NPPs, similar to the previously mentioned NPP V2: EMO 1,2 commissioned in 1998, 2000; and MO 3,4, which is under construction at the present time. The facility called "The final treatment of liquid radioactive waste" also operates on the Mochovce site. The National repository for low-level radioactive waste, in operation since 2001, is located 1.5 km away from the Mochovce NPP site. The NPPs under operation or construction are owned and operated by the Slovenske Elektrarne, a.s. (Slovak Electric, Inc.); the company ENEL, Inc. is its majority shareholder. Recently, the Slovak Government has declared an intention to construct a new NPP in Jaslovske Bohunice (single block/reactor PWR, generation III+), with maximal power output 1700 MW_e.. It is expected to be commissioned in 2029.

The back end of the fuel cycle in the former Czechoslovakia was based on close co-operation with the former Soviet Union. Under bilateral agreements, the Soviet partner delivered fresh fuel and accepted spent fuel for reprocessing without return of high-level radioactive waste. During the years 1983-1986, part of NPP-V1 spent fuel (697 spent fuel assemblies) was shipped, after appropriate on-site storage, to the Soviet Union. The rest of the spent fuel assemblies had been stored in at-reactor pools and later in the interim storage facility. The wet type spent fuel interim storage facility with capacity of 5040 spent fuel assemblies was commissioned in 1987, with its original intention being to store the spent fuel assemblies for 10 years before shipping to Soviet Union. After political changes in late 1980s and early 1990s, the contractual relations on the fuel cycle back-end between the Slovak Republic and Russian Federation were modified. The changes are reflected in the Slovak national spent fuel management strategy.

After the decision to store the spent fuel in Slovakia, Slovak Electric, Inc. decided to extend the capacity and lifetime of the spent fuel interim storage facility in Jaslovske Bohunice. After reconstruction, the facility has greater storage capacity (14,112 fuel assemblies), improved overall safety, particularly seismic resistance, and an extended operating lifetime of at least 50 years. Increased storage capacity was achieved by reracking the spent fuel assemblies into racks with higher capacity (originally 30, now 48 fuel assemblies). This new capacity was considered to be sufficient for expected lifetime of all reactors installed in Jaslovske Bohunice. Now, after pre-term shut down of NPP V1, there should be free space for about 1500 spent fuel assemblies.

Recently, it has been decided that the Slovak republic will have one centralized spent fuel storage facility, in Jaslovske Bohunice, for storage of spent fuel from both nuclear localities. Therefore, it is planned to extend the Bohunice interim spent fuel storage facility during the present decade. Currently, the extension is going through the Environmental Impact Assessment process. Construction of the modular type dry storage facility (storage containers [canisters] emplaced in the concrete storage modules) functionally connected with the existing wet storage has been determined to be the best option for the planned extension. The total capacity after construction of the first two extension stages will be 18,600 spent fuel assemblies. The modular approach makes it possible to accommodate additional extensions as the need arises.

According to the recent infrastructural development, the Nuclear and Decommissioning Company, Plc. (JAVYS, a.s., owned by the state and having been established as a consequence of privatization of Slovak Electric in 2006) is responsible for activities related to the back end of peaceful use of nuclear energy (i.e. NPP decommissioning; centralized treatment, conditioning, and storage of radioactive waste; centralized spent fuel storage; development and operation of repositories). Generally, the nuclear energy back-end activities are financed from the National Nuclear Fund (formerly the State Fund for Nuclear Facilities Decommissioning and Spent Fuel and Radioactive Waste Management). The Fund provides financial tools to cover decommissioning of nuclear facilities, including the management of decommissioning waste, centralized spent fuel storage, and development of a deep geological disposal facility. The Fund is mainly constituted by legislatively established contributions from owners and operators of NPPs, as well as payments by the operators of the electricity transmission and distribution systems.

The history of deep geological disposal development since dissolution of the former Czech-Slovak Federation (1993) can be divided into three stages:

- the first stage (1996-2001; see Section 2),
- the interim stage after the development of a repository was abandoned (2002-2012; see Section 3),
- the current stage after the repository development project was resumed in 2012 (see Section 4).

16.2 First Stage (1996–2001)

R&D for deep geological disposal of spent fuel (and radioactive waste not acceptable for disposal in the existing national near surface repository) in the Slovak Republic began in 1996, in close relation to previous federal activities. As the first step in development activities, the former federal document titled "Project of Deep Disposal Development" was revised to reflect the specific geological conditions of the Slovak Republic. The Directorate General of Slovak Electric (a state company at this time) was intended to act as the project implementer, and DECOM, Ltd. was contracted to act as the project coordinator.

The project formally consisted of activities, which were divided into the following key areas (the main contractors are given in parentheses):

- Design and implementation studies (EGP Invest, Ltd., Uhersky Brod, Czech Republic and Energoprojekty, Plc., Bratislava, Slovak Republic)
- Studies of the source term (Nuclear Research Institute, Plc., Prague, Czech Republic)
- Near field (Nuclear Research Institute, Plc., Prague, Czech Republic)
- Far field (State Geological Institute of Dionýz Štúr, Bratislava, Slovak Republic)
- Siting (State Geological Institute of Dionýz Štúr, Bratislava)
- Safety assessment (Nuclear Power Plants Research Institute, Plc., Trnava, Slovak Republic)
- Public involvement (AEA Technology, Harwell, U.K. and Decom Slovakia, Ltd., Trnava, Slovak Republic)
- Legislative framework (Decom Slovakia, Ltd., Trnava, Slovak Republic)
- Quality assurance of the project (Decom Slovakia, Ltd., Trnava, Slovak Republic), and
- The project coordination (Decom Slovakia, Ltd., Trnava, Slovak Republic).

The main results and the project status at the end of the first stage are described in following sections.

16.2.1 Design and Implementation

There were three principal results related to the design:

- Elaboration of preliminary reference design of the repository, i.e. the design for a hypothetical site and two alternative geological host environments (sedimentary and crystalline). Conceptual ideas for the deep-geological-repository operational phase were focused on various aspects of surface and underground activities presented from technological, economic, and feasibility perspectives. These activities described transportation and reception of spent fuel and radioactive waste packages, encapsulation, manipulation, access to underground shafts or tunnels, emplacement of containers (disposal borehole or tunnel), and auxiliary and control systems.
- Conditions for implementation of the underground laboratory (generic research or confirmation laboratory) considered for the disposal program in the Slovak Republic. This involved the consideration of possible alternatives, the proposal of technical research and economic issues, and the role of public involvement. A confirmation laboratory at a potential repository site was found to be the more appropriate solution at this time.

- Elaboration of the preliminary feasibility study based on knowledge and experience acquired during the given period, current status of mining technologies, and worldwide experience.

16.2.2 Source Term

Nuclear Research Institute, Plc. described the physical and chemical properties (inventory and species of radionuclides) of the WWER-440 spent-fuel assemblies after interim storage. Activity of selected isotopes, total spent-fuel assembly activity, fuel-assembly thermal power, and the contribution of selected isotopes were calculated for 5, 10, 50, 70, 100, 1,000, and 10,000 years after refueling. The effect of cladding and WWER-440 fuel-assembly construction parts on the total inventory of radionuclides and heat production were also calculated, considering the various enrichments: 1.6%, 2.4%, 3.6%, and a planned 3.82%, and various fuel burn-ups: from 12,000 MWd/t U up to 46,000 MWd/t U.

Possible mechanisms of radionuclide leaching from spent fuel and vitrified or cemented forms of radioactive waste were also reviewed. Data regarding radionuclide leaching were gathered from technical literature, with the source term estimated for selected radionuclides and a hypothetical repository. The study was focused on investigating the mechanisms of radionuclide release from spent-fuel cladding and waste glass and cement matrices in a repository environment, as well as identifying critical groups of radionuclides and their characteristics. The results describing the expected behavior of spent-fuel cladding in a deep geological repository environment were also summarized. It was found that zirconium-niobium cladding could be considered an effective barrier against radionuclide release. The resulting data could be usable as input for the future activities regarding the repository concept, disposal container design, and demonstration of the repository safety.

16.2.3 Near Field

The first step in studying the near field was a critical review of the current status of research, engineered-barrier modeling, available information on the materials suitable for engineered barriers, and the disposal container. Physical and chemical properties of sealing materials were also described. Attention was given to processes important for radionuclide retention in engineered barriers and factors influencing these processes, as well as to the transport characteristics of the critical group of radionuclides.

Within the given area, the first proposal for the disposal container design was also elaborated. The container could contain seven WWER-440 spent-fuel assemblies. Carbon-steel (80 mm) coated with a nickel layer (3 mm) was proposed for the outer wall; the inner wall would be made of stainless steel (5 mm), the inner cask of an aluminum alloy. This design facilitates the handling of containers with fuel assemblies and improves heat removal. Such a container would ensure subcriticality, effective heat removal, and pressure resistance up to 20 MPa. The total container weight with encapsulated spent fuel would be 7.7 metric tons. Container handling would require additional shielding to ensure a surface effective dose of 0.1 mSv/h.

16.2.4 Far Field

Worldwide experience in modeling geological barriers and groundwater flow (as dominant factors in radionuclide migration) was reviewed by the State Geological Institute of Dionýz Štúr, Bratislava. Special attention was paid to analysis of groundwater flow mechanisms in saturated and unsaturated environments and transport of dissolved substances, including procedures for verification and validation of the models. Basic information about coupled processes, and the advantages and disadvantages of deterministic and stochastic models were reviewed.

For sites identified as potentially suitable for a repository, three-dimensional numerical models of the geological barrier were prepared. These models reflected the current status of knowledge about geology, petrography, seismicity, neotectonics, hydrogeology, and geochemistry of potential disposal sites. Interactions between host environment (granites or clays) and engineered barriers, as well as the possible alteration of the host-rock environment and engineered-barrier materials (induced by expected hydrogeochemical processes), were analyzed.

16.2.5 Geology and Hydrology

Geological investigation activities were based mainly on the research of archive information on the Slovak territory geological environment.

Generally, the Slovak territory (49,016 km²) is located in the Western Carpathians mountain chain. The Carpathian arc is a tectonically complicated Alpine-type geological structure within the Alpine chain of Europe. From a geologic point of view, the Western Carpathians are structured into several tectonic units (Figure 16–1). Only a few of these units contain rock environments identified as potentially suitable for a deep geological repository site; the five localities marked in Figure 16–1 were recommended for more detailed investigation, including investigation in situ.

A characteristic feature of the Western Carpathians is the Neogene post-tectonic basins infilled with Miocene sediments (predominantly clays, claystones, sands, and sandstones). The overall thickness of Neogene sediments is several thousand meters. The Neogene pelitic sediments were also identified as potentially suitable for siting a repository in Slovakia.

Figure 16–1. Schematic geological map of Slovak part of Western Carpathian. Candidate sites for deep geological repository identified in 2004: 1 - Tribec Mts.; 2 - Veporske vrchy Mts.; 3 – Stolicke vrchy Mts.; 4 – Rimavska kotlina Basin; 5 – Cerova vrchovina Upland.

16.2.5.1 Tectonic and Seismic Environment

In the Western Carpathians, tectonic processes have resulted in a network of discontinuities (faults and fissures) affecting the bedrock properties. Down to depths of ~100 m, fissures provide good conduits for groundwater flow. At deeper levels, the fissures are usually gravitationally closed. The simplified tectonic cross section of the Western Carpathians is shown in Figure 16–2.

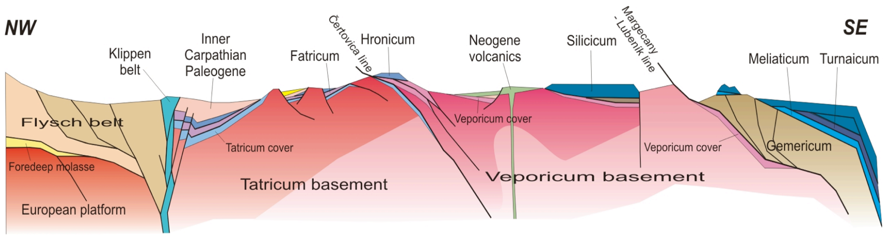

Figure 16–2. Simplified (schematic) tectonic cross–section of the Western Carpathians (not to scale)

Tendencies for vertical movement of the Earth's crust, along with denudation processes, decisively affect the depth of a repository. Long-term geodetic measurements indicated the block structure of the Western Carpathians; the average rate of sinking blocks is approximately 0.5–1.5 mm/year. To assess the possible effects of relief, the geomorphologic models were developed to predict the future relief for a period with potential negative effects on the repository (from 10,000 up to 100,000 years).

Prospective sites for the repository are located away from recorded earthquake epicenters (Figure 16–3).

16.2.5.2 Climatic and Hydrogeological Conditions

Situated in Central Europe, the Slovak Republic experiences temperate climatic conditions. Annual average precipitation varies between 400 and 1000 mm, reaching 2000 mm/year in high mountainous regions. The studies pointed out that hydrogeological properties of the Western Carpathian crystalline complexes are (at this point) not much known at greater depth. In terms of a deep geological repository, the main deficiency of these complexes is their relatively large petrophysical heterogeneity, resulting from tectonic effects. An unfavorable result of tectonic disturbances (besides manifestations of brittle tectonics associated with local increases of permeability) is the frequent occurrence of ductile zones, which may indicate (in some sections) an environment with increased permeability even at greater depths.

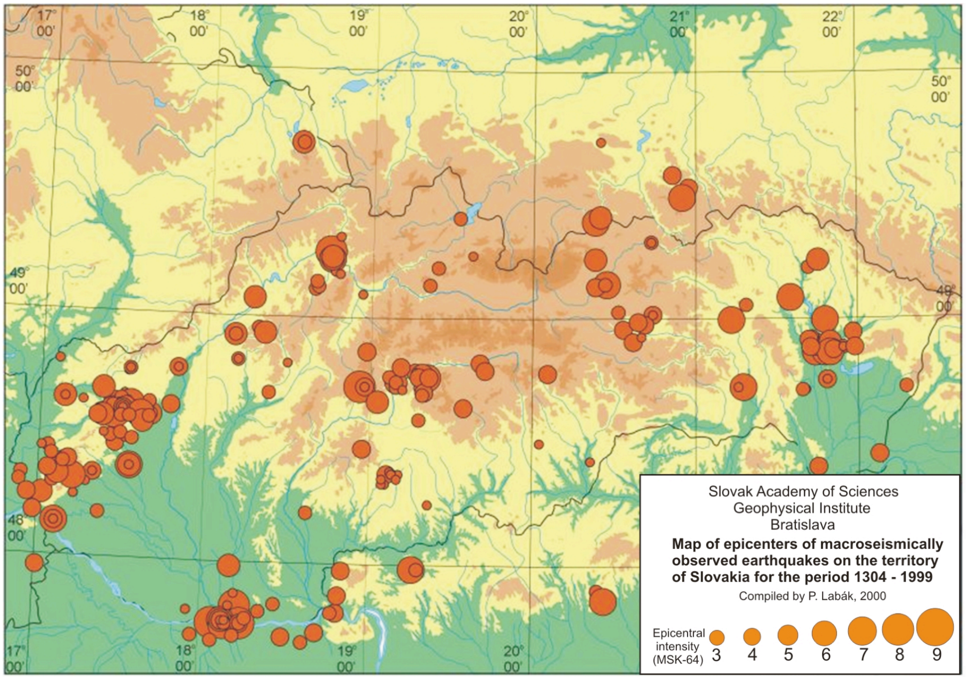

Figure 16–3. Map of epicenters of macroseismically observed earthquakes on the territory of Slovakia for the period 1304 – 1999

16.2.6 Identification of potential geological repository sites

Issuing from worldwide experience and site selection criteria and processes as recommended in the IAEA standard document (Safety Series No. 111-G-4.1 at the time), a preliminary set of site-selection criteria for a deep geological repository was proposed. Three groups of criteria were proposed for site selection:

1. Geologic and tectonic stability of prospective sites (seismic activity, faulting, folding, uplift of the territory, etc.)
2. Characteristics of host rock (lithologic homogeneity, hydrogeology, low hydraulic conductivity, absence of groundwater resources, favorable geotechnical conditions, rock stress, thermophysical and geological characteristics, absence of mineral resources)
3. Conflict of interests (natural resources, natural and cultural heritage, protected resources of drinking or thermal waters).

Owing to progress in program activities and a growing knowledge from exploration of different sites, the siting criteria were supplemented and adapted in 2001. A revised set of criteria was based on the first one, Scandinavian experience, and experience gained from the Slovak program. The evaluation process reflects progressive growth in the qualitative evaluation used in the siting process. Qualitative evaluation of the suitability of host rock includes specified rules and requirements in order of preferences – requirements – criteria:

- Preferences – world-wide accepted principles and conditions of host-rock or site suitability - advisable but not prescriptive,

- Requirements – specified principles and conditions of host-rock or site suitability – obligatory for host-rock or site selection,
- Criteria – defined qualitative and quantitative suitability measures to determine limiting values for host rock or site selection.

Evaluation of the geological environment should be focused on all its components: rock, water, morphology, and geodynamic phenomena, in prediction of its evolution for more than 100,000 years. Conditions to be evaluated can be divided into the following groups:

- Geological
- Hydrogeological and hydrogeochemical
- Engineering geology
- Geomorphologic (area surface stability).

The evaluation should consider also basic requirements for protection of the environment, and social and economic requirements, i.e.:

- Areas with a higher degree of legal protection, or mineral and underground water resources should be avoided,
- Areas with lower population density and more favorable demography should be preferred,
- Beneficial effects of the repository on the site should be enhanced, and negative should be minimized.

In 1996, the program started with a critical review of archived information relevant to site selection. This review included a survey of published and archival regional geology, hydrogeology, engineering, and geophysical data, and led to identification of 15 areas potentially suitable for a deep geological repository within Slovak territory. Seven areas featured granitoid rocks, four clayey formations, three metamorphosed rocks, and one flyschoid rocks. The accuracy of the assessment corresponded to a map scale of 1:200,000. The next four-year phase focused on screening the data for the 15 potentially suitable areas. Limited field verification and some technical measures (geophysical profiles, shallow boreholes) were performed during this stage. A series of maps on a scale of 1:50,000 were compiled for each area, including important geological and hydrogeological factors and data on mineral resources. With respect to preliminary criteria, each area was assessed for its geology, tectonics, vertical movements, seismicity, mineral resources, geothermal potential, geochemistry, and engineering geology. The result of this stage was the ranking of the selected areas into three groups:

- Areas recommended for further investigation - these areas are not expected to contain excluding factors.
- Abandoned but not excluded areas - these areas may be considered as backup if other sites prove unsuitable or for any reason unacceptable. At these sites, lithologic structure, tectonic classification, and presence of ore indicators probably rule out (though not conclusively) selection of a sufficiently large and homogeneous rock block.
- Excluded from further investigation—these areas contain utilizable geothermal energy below the prospective host rock; thin, hydraulically homogenous rock formations; and/or lithologic heterogeneities.

Finally, three distinct areas (five localities) were determined as prospective sites for construction of a deep repository (see Figure 16–1). Their total areal extent is 320 km^2. Three localities are situated in granitoid rocks (granites, granodiorites, tonalites) of Hercynian (Variscan) Early to Upper Carboniferous age and two are situated in argillaceous Neogene complexes. More detailed geologic characterization of sites, possibly reducing their extent and number were consequently performed here. New field investigations (geophysical measurements – electric, gravimetric, magnetic, seismic) and shallow drilling down to 250 m (including hydrogeological and geophysical logging) were performed. More detailed maps were compiled for each site (scale 1:25,000); it was expected that the number of prospective sites would be further reduced in the next stages of repository development.

16.2.7 Safety Assessment

Activities supporting the repository safety demonstration were generally oriented to the development of safety assessment methodology. They started by the critical review of safety concepts for a national deep geologic repository (DGR) and corresponding approaches to modeling. Methodologies were examined for various safety issues, with the aim of identifying and assessing their applicability for conditions in the Slovak Republic.

First, scenario development methodologies were studied. Emphasis was placed on points of international consensus and global issues related to deep disposal. It was agreed that an internationally accepted scenario development methodology, supporting tools, and an international database of features, events, and processes (FEPs) should be implemented. Then, appropriate conceptual and mathematical models for safety assessment of particular repository subsystems were selected, and the need for modifications and adaptations was assessed. Results from this work provided preliminary information about worldwide-accepted experience and knowledge in safety assessment, both in establishing conceptual models and methodology. Studies also were done to address the issue of natural analogs for deep geological disposal.

16.2.8 Public Involvement

It was already understood that the prospects for a geological repository should be enhanced by closer and earlier involvement of host communities. Key features of that involvement were identified:

- Better-focused and more open information,
- Closer contact between the public and the nuclear community,
- Need to establish real public influence and control.

Five key aspects of public involvement need to be addressed in any public awareness program: information, communication, participation, acceptance, and compensation. The future activities should focus on the following aspects of the public involvement:

- Informing the public on radioactive waste management (presentation of nuclear energy and radioactive waste management in media, changing negative attitudes and responding to arguments against nuclear energy, and identifying constructive ways to respond to opposition).
- Investigating the socio-economic effect of DGR development and its potential impact on the public (how to mitigate or eliminate any negative impact and maximize the benefit for the host community).
- Developing a program for public involvement in decision-making and information processes during repository development.
- Publishing a brochure explaining the alternatives related to managing the back end of the fuel cycle, possible solutions for the Slovak Republic, basic principles of deep geological disposal, and the national program for the general public.

16.2.9 Project Coordination Activities

At first, DECOM, Ltd. performed the standard project coordination activities, under contract with the project implementer, e.g. proposals and evaluation of plans, evaluation of the project outputs and organization of corresponding review meetings, establishing the archival system for storage and maintenance of the program documents and data, proposals for the annual progress reports, etc. Additionally, DECOM established the project quality management system and analyzed national and international legislation from the point of view of repository development and implementation needs. Particular attention was given to application of provisions of the Environmental Impact Assessment Act to the implementation of a geological repository. The existing national infrastructure of radioactive waste and spent fuel management was also analyzed and compared with infrastructures of other countries with similar levels of peaceful use of nuclear energy. Finally, the project coordinator represented the Slovak program in a major number of international activities at the first stage period (which also continued later, after the project was abandoned). In particular, it began with organization of annual seminars to exchange the experience learned from activities dealing with the repository development and operation. The seminars were originally co-organized with Czech radioactive waste management organization SURAO, and later also with the institutions involved in radioactive waste and spent fuel disposal from Austria, Hungary, and Slovenia.

16.3 Interim Stage (2002–2012)

Unfortunately, in 2001 the Slovak Electric top management decided to abandon the deep geological disposal development program. This decision was followed by 11 years during which some individual activities of corresponding geological research and investigation were continued by the State Geological Institute of Dionýz Štúr and financed from budget of the Ministry of Environment, Directorate for Geology and Natural Resources. The State Geological Institute and DECOM, Ltd. maintained the international relations during this period, mostly by participating in international R&D projects (DECOM, Ltd., for instance, played a significant role within the European projects dealt with the shared international repositories idea – see below).

The activities of the State Geological Institute of Dionýz Štúr were mainly focused on research of geological conditions in two of the selected prospective localities. The research projects were oriented mainly toward methodological aspects of geological studies of crystalline rock environment (The Tribec Mountains; site 1 in Figure 16–1) and sedimentary rocks (Cerová vrchovina uplands; site 5). As mentioned above, the scope of work was narrower due to budgetary limitations, as funding was provided only from the state budget by the Slovak Ministry of Environment.

16.3.1 Crystalline rock environment

The project "Tribec - determination of geological, geophysical and environmental factors for sitting of high radioactive waste deep geological repository" (Marsina et al., 2002) followed the research concerning the crystalline rock environment performed in the first stage. The main objective was comprehensive site research, particularly focusing on development of methodological aspects of crystalline rock environment and geological research in the context of deep geological disposal. Field research was focused on the prospective area Central Tribec Mountains (Tatricum geological unit).

This prospective site is an area of granitoid rocks in the southern Tribec-Zobor block in the Tribec Mountains. The Zobor Massif, one of the largest in the Western Carpathians, is formed of granitoids characterized by a uniform and almost invariable composition, from leucocratic granites, fine- to medium-grained granites – granodiorites, to massive medium-grained granodiorites – tonalites (Figure 16–4). Tectonic deterioration of the site is low in general (Figure 16–5), and thus hydrogeological conditions for a repository seemed to be favorable. At present, there are not known limiting structural-tectonic phenomena in the relief of the plutonic body. The drilling survey (Kováčik et al., 2001) had revealed an increase in rock quality with depth (homogeneity, low tectonic deterioration, etc.). Rock-quality designation (RQD) at depths of 150–250 m is about 90–95%. No indications of ore mineral concentrations or geothermal potential have been discovered in this region. Hydrogeological conditions in the deeper horizon are little known at present.

Knowledge of the geological characteristics of crystalline rock environment in the Tribec locality was extended by the project output. The main geological methods applied were: geo-electrical methods, seismic research, thermal conductivity rock research, hydrogeological and engineering geology investigation, optical porosimetry, and atmogeochemistry. Results confirmed the suitability of the area (Site 1 in Figure 16–1) as a prospect for siting the deep geological repository.

16.3.2 Sedimentary rocks

An example of geological research activities in the sedimentary formations of the Cerova vrchovina Upland and the Rimavska kotlina Basin (sites 5 and 4, respectively, in Figure 16–1) is shown in Figure 16–6. Focusing on this potential host rock environment, the geological project "Assessment of the geological factors for the deep geological repository site selection process – part 2 Sedimentary rocks – Methodology" was conducted in the period 2007-2012 (Slaninka et al., 2012). The project followed the results of previous research (e.g. Kováčik et al., 2001) and was supported also through international cooperation (mainly IAEA and Slovak-Belgian cooperation). The main objective was to solve methodological issues of geological research specific for sedimentary environment in the context of deep geological disposal. The main research area was prospective reconnaissance locality Cerova vrchovina Upland.

From a lithological, structural, and spatial point of view, the most prospective host rock layers in the area appeared to be two lithostratigraphic units: the Szecseny Schlier (member of Lucenec Formation) and Lenartovce Beds (member of Ciz Formation). These lithostratigraphic units form the principal mass of the Neogene basin filling. The predominant lithologic type in both formations is a mixture of siltstones and claystones. Maximum thickness of the Ciz Formation in the territory of Slovakia is 400-500 m, while the maximum thickness of the Lucenec Formation in Cerova vrchovina Upland is 1300 m (1100 m in the Rimavska Kotlina Basin). The thickness of both formations increases from the northern margin toward the south. The cumulative thickness of both formations varies between 1400 and 1600 m.

The Ciz Formation originated in a marine environment. It unconformably overlies pre-Cenozoic crystalline basement and is unconformably overlain by Lucenec Formation (Egerian). The age of the Ciz is Kiscellian (Oligocene), as proved by marine micro- and macrofossils and calcareous nanoplankton. This formation continues into the territory of Hungary. The predominant lithotype of the unit is grey, green-grey, friable calcareous claystone and siltstone. In the lower part of the member, sandy lamination or lenticular bedding occurs.

The Lucenec Formation was also created in marine environment. In the Cerova vrchovina Upland, the formation is overlain by the Filakovo Formation (Eggerburgian). The age of the formation is Egerian (Late Oligocene – Early Miocene). Szecseny Schlier is the dominant member of the Lucenec Formation. It is a friable calcareous siltstone of grey or green-grey color. Some layers of sandstone can occur within the lower part of the member. In the Slovak territory the maximum thickness verified by boreholes is about 700 m (Figure 16–7), but one can estimate a greater thickness in the area of the Slovakian – Hungarian state border (as much as 1300 m). The member is rich in marine fauna (molluscs, foraminifers), calcareous nanoflora and the sporomorphs (Figure 16–8).

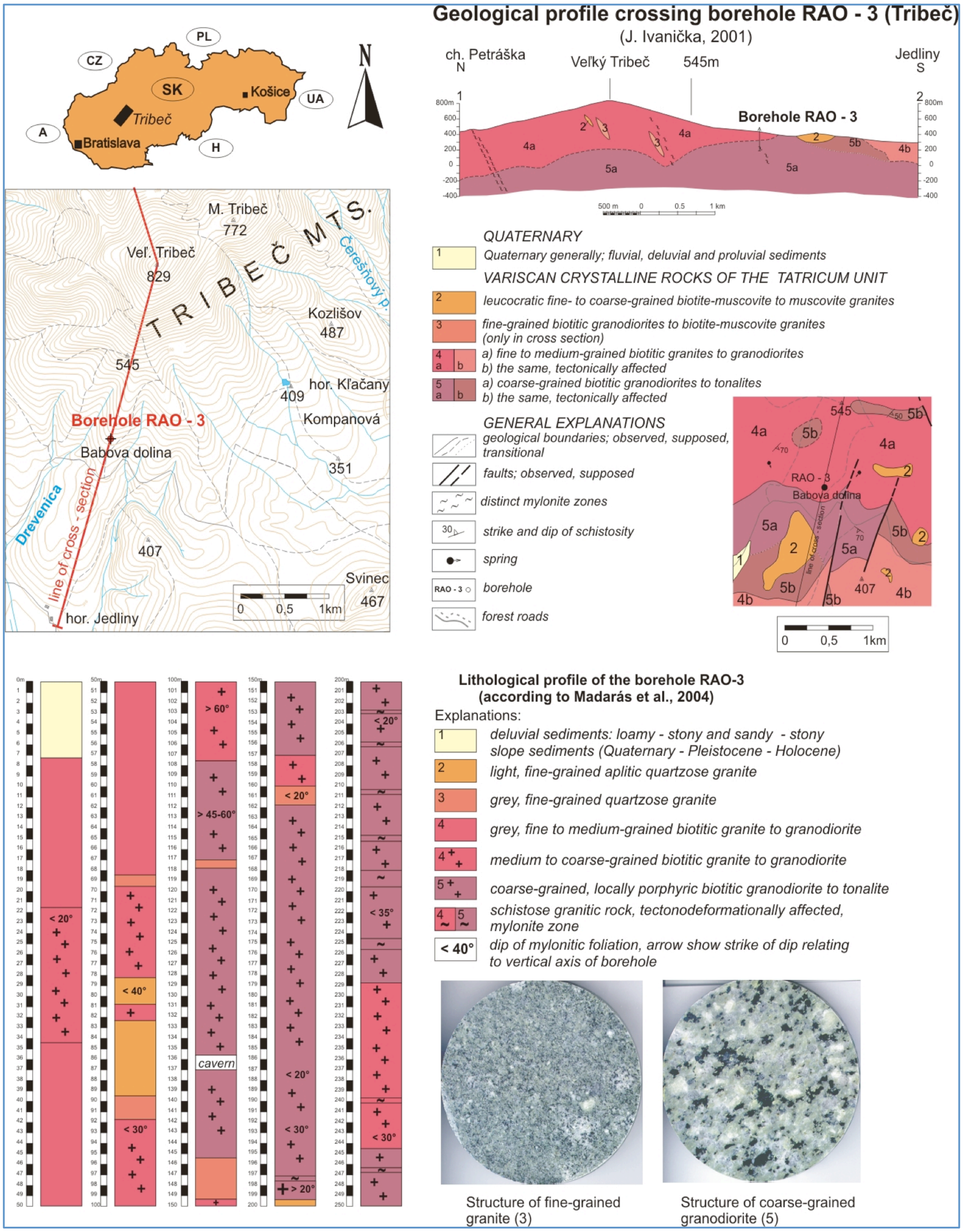

Figure 16–4. Site layout, geological cross-section, and geological structure of prospective area Tribec (Madaras et al., 2004)

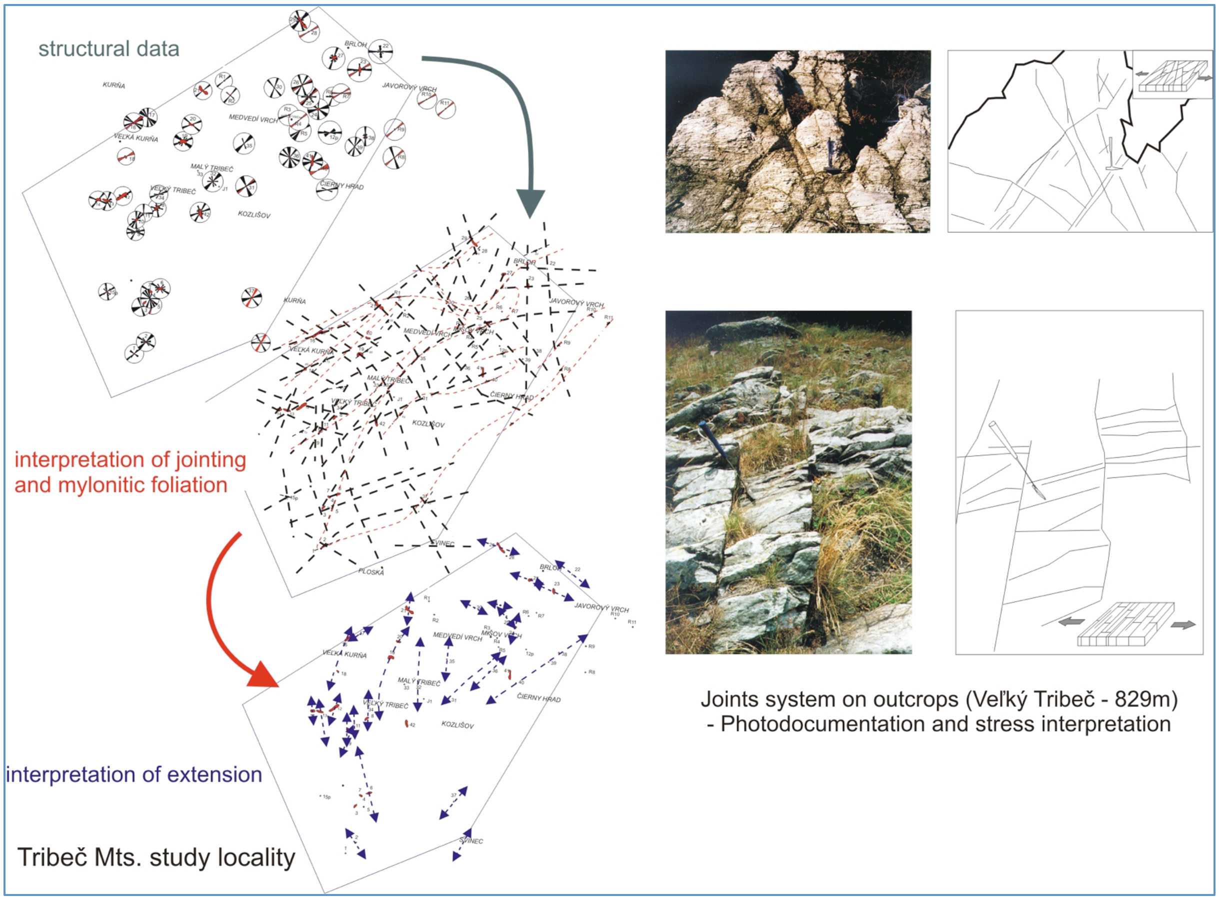

Figure 16–5. Interpretation of the neotectonic joints in granitoid rocks. The Tribec locality case study (Hók et al., 2001)

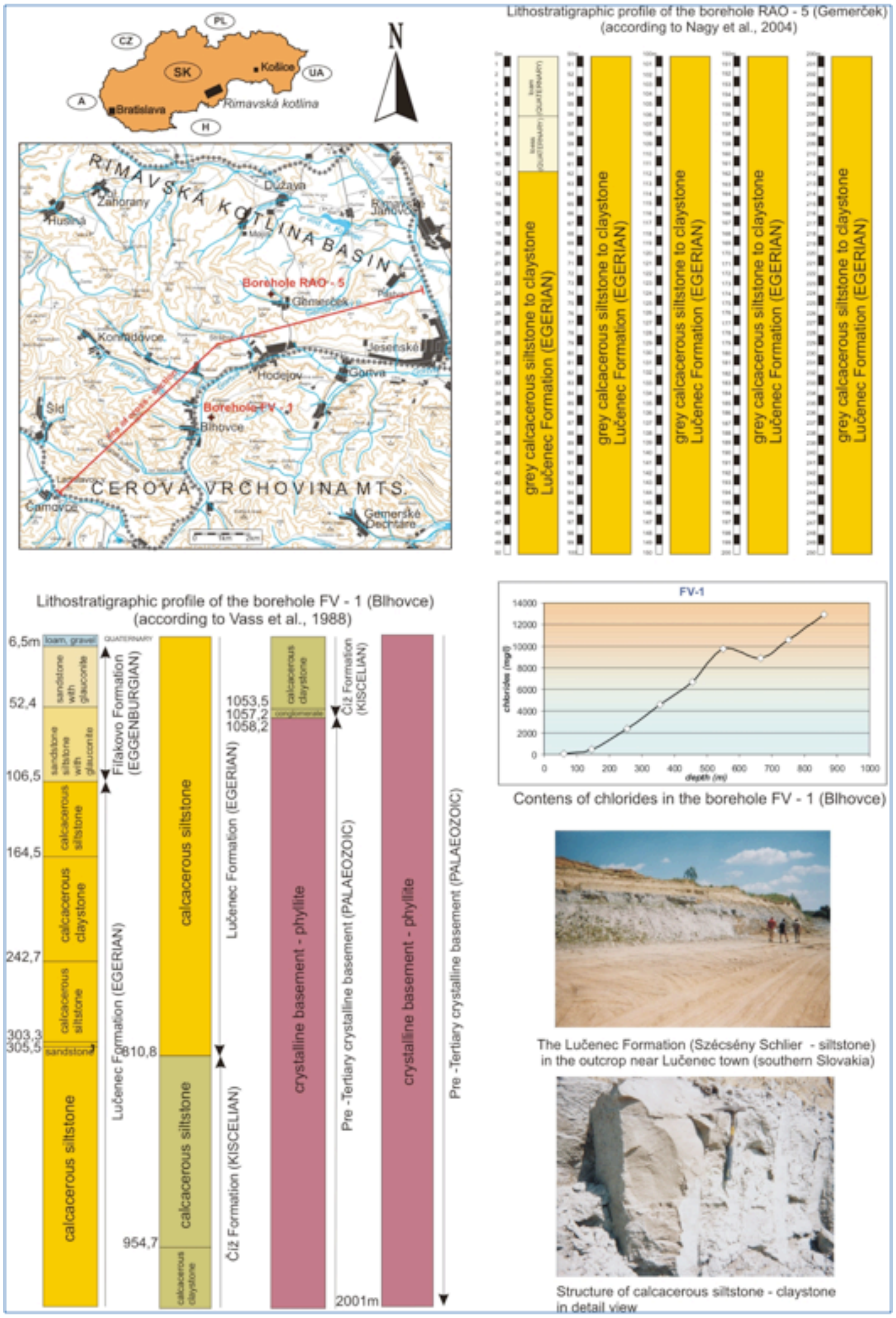

Figure 16–6. Site layout, geological cross-section, and geological structure of prospective area of the Cerova vrchovina Upland and Rimavska kotlina Basin – boreholes RAO-5 and FV- 1 (Nagy et al., 2004, Slaninka et al., 2012)

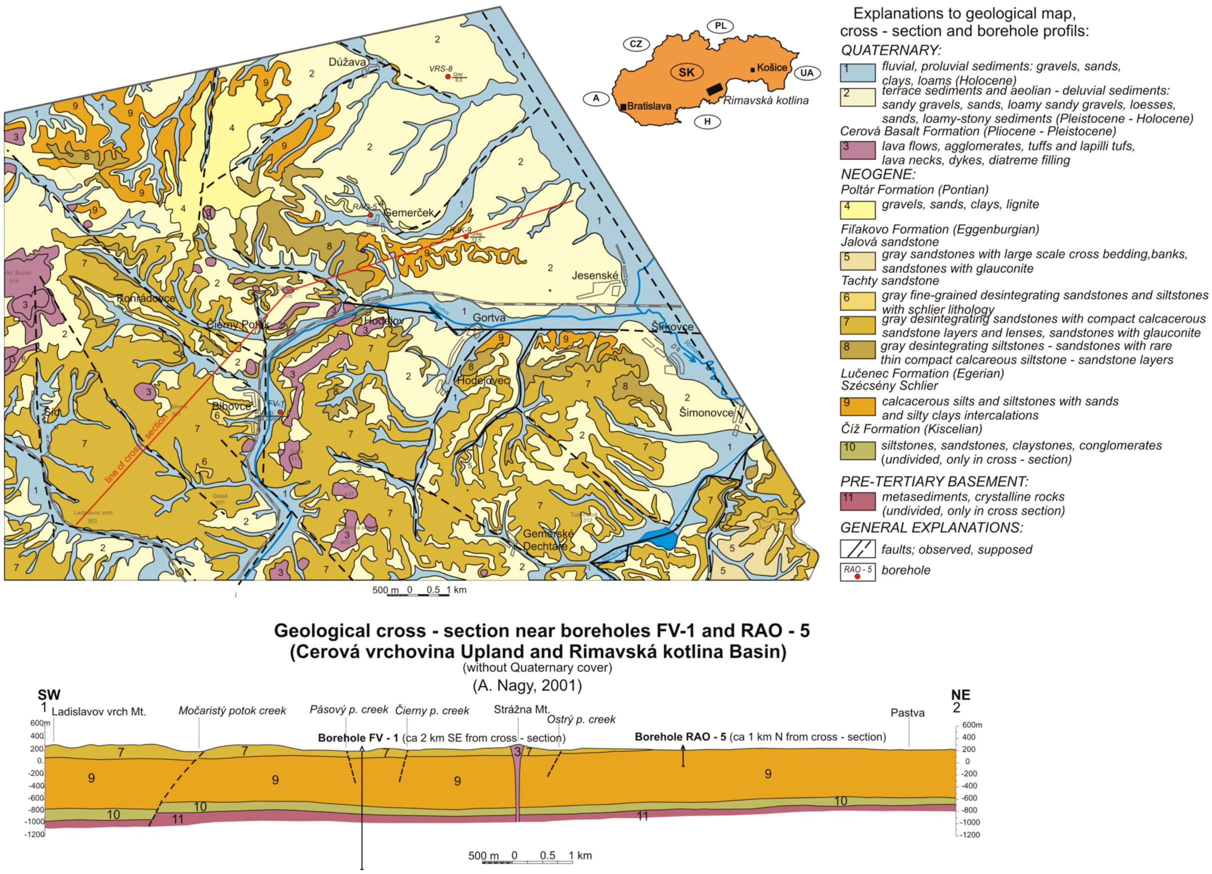

Figure 16–7. Site layout, geological cross-section, and geological structure of prospective area Cerova vrchovina Upland (Nagy, 2001)

Figure 16–8. Biostratigraphical findings in samples from Szecseny Schlier (borehole GOR-1) (Slaninka et al., 2012)

The Lucenec Formation mineralogy is a complex mixture of clay minerals, mainly smectite, illite, and kaolinite, with pyrite and calcite as other important minerals (16–9). A large fraction of quartz is also present. The content of fine pyrite is relatively large. In the presence of oxygen the pyrite can be oxidized to form a sulfate and iron oxo-hydroxides (Figure 16–10). The dry bulk density is very high (about 2.2 g/cm^3 with only a small increase in density with depth), which is probably an indication of a high overburden in geological history. A high dry density means a low porosity (from 14 to 20 %) and, as a consequence, low hydraulic conductivity and low diffusion coefficient. The permeability of the Szecseny Schlier siltstones itself and also the permeability along main regional faults, with sporadic occurrence of acidulous water springs, must be also taken into account and carefully investigated.

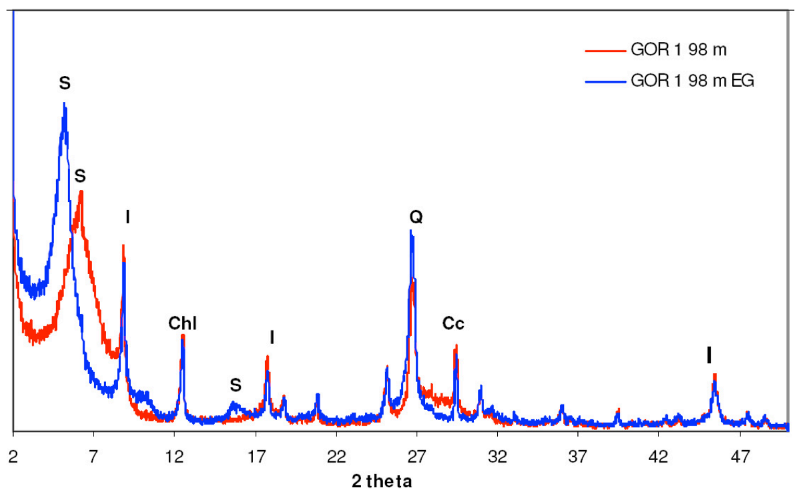

Figure 16–9. Example of X-ray record of clay fraction (<2 microns), Szecseny Schlier samples (borehole GOR 1), natural and saturated by ethylene glycol (EG) (Slaninka et al., 2012)

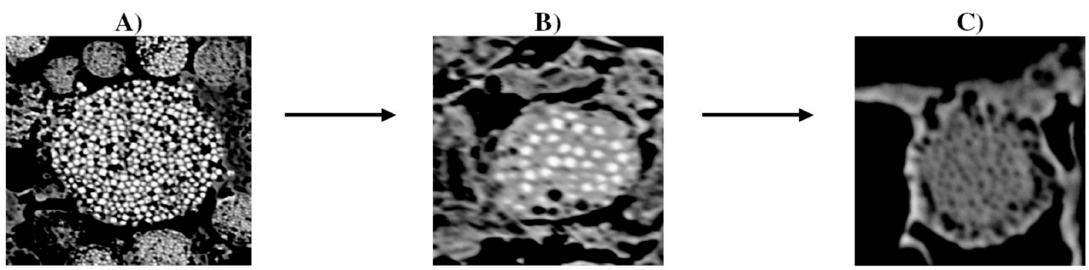

Figure 16–10. The process of oxidation of pyrite grains (white spherical formations): A) non-oxidized pyrite, B) partially oxidized pyrite grains, C) the pseudo fully oxidized pyrite (dark spherical formations) (Slaninka et al., 2012)

Preliminary hydraulic conductivity measurements performed on a limited number of samples, and also field pumping tests, have given values as low as 10^{-10} m/s and even 10^{-11} m/s, consistent with the high density and low porosity of the Szecseny Schlier. According to the preliminary data set, diffusion appears to be the dominant transport mechanism. This observation is particularly important for the site performance assessment, and is a very advantageous characteristic. This is further supported on the basis of well-preserved high NaCl chemical type of pore water in the formation, originating in marine sedimentation. The example of chemical composition of pore water vertical profile with high salinity from FV-1 borehole cores is presented in Figure 6. The maximal content of chlorides in pore water was more than 12,500 ppm, and similar results were also obtained from other boreholes, so this characteristic seems to be regional.

The Szecseny Schlier is a geological formation with favorable properties as a potential host rock for geological disposal of radioactive waste. The main advantages are: lithological homogeneity, high density and low porosity (Figure 11), suitable geomechanical properties, low hydraulic conductivity, and high thermal conductivity. However, all these characteristics have been based on a relatively low number of observations or derived from secondary information. Therefore, they must be confirmed by further detailed site investigations.

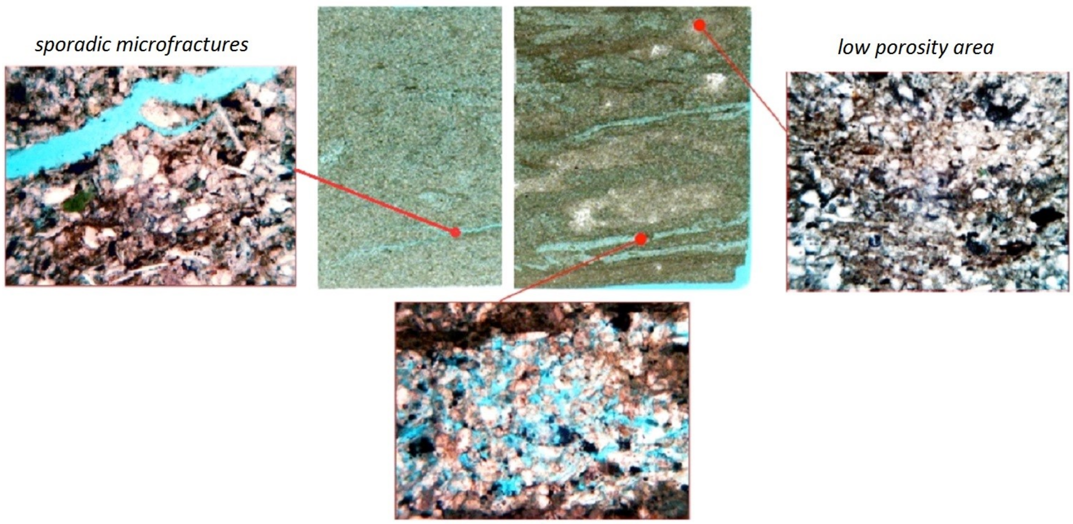

Figure 16–11. Photographs of the rocks used for the porosity analysis of Szecseny Schlier samples (borehole GOR 1) (Slaninka et al., 2012)

16.3.3 Academic research

Additionally, some university research related to the issue of deep geological disposal was also performed at the same time. Properties of clay minerals from Slovak deposits and their suitability for utilization as sealing materials were studied in the R&D project "Study of the properties and use of selected groups of minerals: physical and mechanical properties of sealing materials for radioactive waste repository" (Šucha et al., 2005). The project resulted mainly in demonstrating the potential of Slovak bentonite for use as a buffer and backfill material suitable for a deep geological repository. The study "Adsorption of cesium on domestic bentonites" (Galambos et al., 2009) dealt with adsorption properties of Slovak bentonites potentially suitable for a deep repository, in accordance with other studies elaborated at the same time (Frankovská et al., 2008; Slaninka et al., 2007).

Another study from that period characterized the Lucenec sedimentary formation from an engineering geology point of view (Adamcová et al., 2009).

16.3.4 Participation in international activities

The State Geological Institute of Dionyz Stur participated on the IAEA project "Site Selection Process for a Deep Geological Repository in Sedimentary Rock Formations, Phase I (2003-2004) and Phase II (2005-2006)" (Slaninka et al., 2006). The overall objectives were: identification of key geological problems in the study areas, definition of scientific techniques for optimal solution of identified problems, identification of a pilot area in the South Slovakian Neogene basin, scientific solution of problems in the pilot area, and preliminary characterization of selected prospective sites.

Specific tasks of the project can be summarized as assessment and selection of key geological parameters for their use in performance assessment studies, progress made towards identification of potential candidate sites for geological repository, optimal use and interpretation of the results during the experimental phase, implementation of expert missions, training in the site selection, and characterization of geological repositories.

The project was considered as an integral part of the overall process to select a suitable site for a deep geological repository for high level waste (HLW) in Slovakia. Since the natural characteristics of a site play an important role in the development of a safe geological repository, site characterization and site selection studies represent an essential activity of any national DGR program. The IAEA supported the Slovak expert with technical assistance and services to strengthen the Slovak capacity in site characterization methodologies and site selection strategy for the deep repository.

Training in and Demonstration of Waste Disposal Technologies in Underground Research Facilities, IAEA Network of Centers of Excellence, provided another type of international cooperation. The Network project was formally initiated in 2001 to address the geological disposal needs of member states involved in the development of civilian nuclear technologies. Transfer of knowledge between counterparts with different levels of development of deep geological repository programs, exchange of experience, and training courses for experts using accessible underground research facilities capacities were a main objective here. It was specifically intended to share understanding of the requirements (safety and safeguards) and methodologies (technologies) for radioactive waste disposal, with particular emphasis on the underground disposal of high-level, long-lived, and spent nuclear fuel wastes.

Within the project, the Slovak representative declared his concern related to:

- Development of the appropriate methodology for the geological investigation in the context of the DGR program for the specific geological conditions (sedimentary and crystalline rock environment),
- Development of the siting criteria for the Slovak national geological disposal program.
- Application of the geological research and investigation for site characterization – mainly extension to depth (via deep drilling and geophysical methods, appropriate hydrogeological, mineralogical and geochemical investigation, laboratory experiments and modeling, etc.),
- Geological, hydrogeological, and geochemical monitoring and development of relevant models.

The third type of international cooperation that the State Geological Institute participated in, was the bilateral cooperation Belgium (Tractebel and SCK-CEN) – Slovakia (represented officially by the Slovak Electric, Inc., and later the Nuclear and Decommissioning Company, Plc./JAVYS a.s.). Several expert meetings and exchanges of knowledge between counterparts took place within the cooperation frame and field measurements were also carried out. Cooperation was focused mainly on geological investigation related to the siting of geological repository, site characterization, especially in the study of clay-environment geochemistry, scenario development, soil gas CO_2 measurement, and hydrogeological modeling in porous media.

Extensive results have been achieved with the use of in-situ soil gas measurements (mainly CO_2, but also rare gases). These gases were tested as indicators of permeability of geological discontinuous structures in sedimentary environments, which is an important feature for possible migration. On the basis of the field research results it was concluded that in-situ CO_2 profiling/monitoring in soil gas, together with analysis of carbon isotopes (^{14}C and $\delta^{13}C$) and noble gases (mainly ^{3}He and ^{4}He, Kr, Ar) proved to be a potentially powerful method to detect deep geogenic CO_2 leakage associated with geostructural anomalies, mantle, or crustal activities in specific sedimentary conditions (Drimmer and Slaninka, 2014). Applied methodology could provide an additional helpful safety tool in the decision-making process regarding the repository siting in an area such as the Szecseny Schlier Formation, where CO_2 is produced at a depth and locally comes to surface through previously unidentified fault zones.

Besides co-organization of annual bilateral (later regional) seminars on disposal of radioactive waste and spent fuel mentioned above, DECOM, Ltd. participated in research projects financed by European Commission within its research Framework Programs – Euratom, such as:

- Comparison of Waste Management Alternative Strategies for Long-lived Radioactive Wastes (COMPAS) – 2003-4. The principal goal of the project was to compare and evaluate the different approaches and strategies, as developed in EU Member States and Switzerland, for the management of radioactive wastes. Nearly all the countries that have participated in the project have adopted deep geological disposal as the preferred long-term management option for spent nuclear fuel (that is not considered for reprocessing), high-level waste, and other long-lived waste. However, there were different perceptions of the urgency for implementing deep geological disposal, depending on national policies and social, logistical, and economic issues.
- Support Action: Pilot Initiative for European Regional Repositories (SAPIERR) – 2003-5. Three main deliverables represented the outputs of the project: analysis of the inventory of radioactive waste and spent fuel in 16 European states to identify a potential need for shared repositories, analysis of the legal aspects (in general, international legislation, national legislation), establishing and analyzing technical options for scenario, where several countries agree to develop a shared repository in the territory of one of them.
- Co-ordination Action on Education and Training in Radiation Protection and Radioactive Waste Management (CETRAD) – 2003-2005. The basic aim of the CETRAD project was to assess the current situation of education and training in radioactive waste management, focusing on geological disposal across Europe. The project activities were carried out in two phases. Phase 1 involved national review of both the needs for education and training and the existing infrastructure and resources at the present time and in the future. Phase 2 developed proposals and options for education and training based on the needs identified in Phase 1. Within the project, DECOM, Ltd fulfilled Slovakia's national correspondent role.
- Strategic Action Plan for Implementation of European Regional Repositories: Stage 2 (SAPIERR II) – 2006-2008. The objectives of the project were: define options for organizational frameworks and project plans for a modestly sized, self-sufficient European Repository Development Organization (ERDO), clarify legal, economic, safety and security, and societal aspects of shared regional solutions, and, finally, present the results and recommendation at a seminar for interested countries. The project was transformed into establishment of the ERDO working group, with active participation of the Slovak representatives.
- Arenas for Risk Governance (ARGONA) – 2006-2009. The project investigated how approaches of transparency and deliberation relate to each other and also how they relate to the political system in which decisions (for example on the final disposal of nuclear waste) are ultimately taken. Furthermore, the project investigated how good risk communication can be organized, taking cultural aspects and different arenas into account. In a central part of the project major efforts were made to test and apply approaches to transparency and participation by making explicit what it would mean to use the RISCOM model and other approaches within different cultural and organizational settings.
- Implementing Public Participation Approaches in Radioactive Waste Disposal (IPPA) – 2011-2013. In short, the project was about enhancing the quality of decision making processes in nuclear waste management by clarity, awareness, fairness, and trust, how to implement processes of participation and transparency and how stakeholders should be involved in a "safe space", practical organization of

safe spaces in national programs and exploration of how this can be done also in the multi-national context.

16.4 Recent Developments

The Slovak Republic adopted a new law regarding the National Nuclear Fund in 2006. Fulfilling its particular provisions, the Fund's Board of Governors elaborated the first national complex policy and strategy document (Strategy for the Nuclear Energy Back-End), which was adopted by the Slovak Government in May 2008. The Strategy was later updated (Strategy for the Peaceful Use of Nuclear Energy) and in January 2014 the Government adopted the updated document. Both these strategies assumed resuming activities leading to implementation of the national deep geological repository.

16.4.1 Current Activities

In response to the adoption of the updated strategy, the Slovak program of deep geological repository development was re-opened in 2012. Nuclear and Decommissioning Company (JAVYS), Plc. contracted Consortium led by Institute of nuclear fuels, Plc., Prague (Czech Republic), with principal participation of ÚJV Řež, a.s. (Czech Republic; former Nuclear Research Institute, Plc.) and the Slovak company PC&G, Ltd. supporting the Consortium in area of geological investigation and research. The first stage of reopened project has been planned for 2012-2016 and consists of:

- Complex reviewing of the Slovak deep geological repository development activities, including summary of the results achieved within international activities concerning this topic.
- National program of radioactive waste and spent fuel management according the European Council Directive 2011/70/Euratom of 19 July 2011 establishing a Community framework for the responsible and safe management of spent fuel and radioactive waste (Art. 11 and 12) – preparation of proposal
- Updating the latest (2001) feasibility study.
- Updating the latest (2001) version of the siting criteria.
- Public involvement issues:
    - Preparation of the public involvement strategy within the geological repository implementation area.
    - Analysis of economic and non-economic tools supporting the public acceptance of the geological repository implementation.
    - Proposals for the legislative framing of incentives designed for communities concerned for repository siting and post-siting activities.
    - Proposals for information and publicity matters.
    - Support to JAVYS in organization of initial meetings with concerned communities.
- Elaboration of a detailed plan of the second stage of geological repository development (2017-2023).

A complete review of past activities was summarized in the study "The complex reevaluation of works performed heretofore within the deep repository development project and evaluation of their results" (Havlová et al., 2013). The study also summarized the results achieved in international activities. The main task of the study was to determine how the eleven years vacancy (or performance of isolated activities mainly in the area of geological investigation and research) could influence actual continuation of the past Slovak program, namely:

- Determination which of the past activities and outputs we could immediately continue,
- Determination what of the past activities and outputs could be at leastwise partially utilized,

- Determination what of the past outputs are practically of no value today, i.e., the corresponding activities will be needed to be performed completely again.

The review has identified that outputs of the siting studies (i.e., studies concerned with the geological investigation and research) are the only ones that remain useful to Slovakia today. The past activities dealing with the safety assessment and demonstration methodology and parameters, formally divided on the design approach to disposal container assembly, the source term, near and far fields as well as the safety assessment methodology itself (see above), being research and desk studies, must be repeated.

The next output was the first proposal of a new national program for radioactive waste and spent fuel management. A document was written to meet the provisions of Article 12 of the European Council Directive 2011/70/Euratom of 19 July 2011, establishing a Community framework for the responsible and safe management of spent fuel and radioactive waste. Objectively, the proposed national program is consistent with the Strategy of the back-end of peaceful use of nuclear energy approved by the Government in January 2014. After approval by the top management of JAVYS, the proposal was submitted to the Board of Governors of National Nuclear Fund, which is responsible for elaboration of strategies within the given area. Finally, after the involved state bodies comments, the Governmental Decision approved the national program in July 2015.

Works related to updating the old feasibility study were also launched in 2013. In the first step, the former document (2001) was reviewed and judged to be obsolete and insufficiently comprehensive; complete revision was therefore recommended. The Czech output from the similar project, "Updating of the Czech geological repository reference design" (2011), was recommended as a possible model for such revision. The final version of the revised feasibility study will be finished soon.

For updating the site selection criteria, a new set of criteria has been defined for both crystalline and sedimentary rock conditions; however it is still based on the old one, on international standards and experiences.

At the same time, a proposal for the Strategy for public involvement in the field of geological repository development in the Slovak Republic, including analyses for economical and non-economical tools supporting concerned communities, was developed. Informative and promotional materials about the repository development in Slovakia were elaborated and proposed to the repository implementer. At the same time, legislation was proposed to stimulate municipalities and communities affected by development and operation of the repository, as an amendment to the Act on the National Nuclear Fund.

16.4.2 Intentions And Plans

The approved "Strategy of the back-end of peaceful use of nuclear energy in Slovakia" and the official proposal "National Policy and National Program of spent nuclear fuel and radioactive waste management in Slovakia" deliberated two options for the final step of spent fuel (and radioactive waste not disposable in the existing near surface repository) management (so called "dual track" solution):

- Development of the Slovak deep geological repository,
- Continuing participation in the activities, potentially leading to implementation of a shared, international repository.

In order to implement geological disposal in the Slovak Republic, strategic documents have determined the following milestones:

- Elaboration of the research and development framework program in area of deep geological disposal and creation of internal conditions for its implementation – 2018,
- Creation and implementation a system of economic incentives for municipalities affected by the development and, later, by operation of the repository – 2018,
- Adoption the resolution on continuation or cessation of the double-track approach to geological disposal development, i.e. complex evaluation of the shared, international repository idea – 2020,
- In case of cessation the double-track approach:
    - decision on the deep geological repository site – 2030,
    - commissioning of the repository – 2065.

A detailed plan for the subsequent stage of the repository development (2017-23) is currently under preparation.

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16.6 Acronyms

ARGONA—Arenas for Risk Governance

CETRAD—Co-ordination Action on Education and Training in Radiation Protection and Radioactive Waste Management

COMPAS—Comparison of Waste Management Alternative Strategies for Long-lived Radioactive Wastes

DGR—Deep Geologic Repository

FEP—Features, Events, and Processes

HLW—High Level Waste

IPPA—Implementing Public Participation Approaches

JAVYS—Nuclear and Decommissioning Company

NPP—Nuclear Power Plant

PWR—Pressurized water reactors

R&D—Research and Development

RQD—Rock-Quality Designation

SAPIERR—Support Action: Pilot Initiative for European Regional Repositories

SAPIERR II—Strategic Action Plan for Implementation of European Regional Repositories: Stage 2

WWER—Water-cooled water-moderated power reactor

Chapter 17

Slovenian Approach to Strategy and Planning for High-level Waste and Spent Fuel Deep Geological Disposal

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ABSTRACT: Slovenia, like some other countries, operates a small nuclear fleet and can be expected to generate relatively small amounts of spent fuel and high-level radioactive waste. However, according to European Union Council Directive 2011/70/Euratom for the responsible and safe management of spent fuel and radioactive waste, each European Union member state shall ensure the implementation of its national programme for the management of spent fuel and radioactive waste. For a small nuclear program, the financial and human resources required to develop a national geological disposal facility may not be feasible or economically practical within the framework of the open and connected energy markets. This is why the idea of regional and international cooperation regarding radioactive waste management is interesting, especially for countries with small or very small nuclear programs, and has been advancing since the previous century. However, due to uncertainties over implementation of regional/multinational repository, the national spent fuel and radioactive waste management programme cannot rely only on international cooperation. Policy on national repository option must be kept open in national programme. Both plans can be implemented in parallel in national programmes in so called "dual track" approach. This dual tack approach is presented in more details in this chapter, which gives also a short overview of Slovenian national plan and strategy for managing radioactive waste and spent fuel with the focus on the high-level radioactive waste and spent fuel management via dual track approach policy.

17.1 Introduction

Slovenia has a very small nuclear program: it owns one nuclear power plant in co-ownership with Croatia in 50:50 share located in Krško, Slovenia (Krško NPP). In addition to the operating nuclear power plant there is also one research reactor (TRIGA) and central interim storage facility for radioactive waste from small producers, both near the capital Ljubljana. In Slovenia all spent fuel is located only at reactor sites. Irradiated fuel in the Krško NPP is either loaded in the reactor or stored in spent fuel pool. All irradiated fuel in TRIGA is currently loaded in the reactor itself (URSJV, 2014; SNSA, 2014).

Krško NPP is a 696 MWe Westinghouse two-loop pressurized water reactor in commercial operation since 1983. It was originally designed to operate for 40 years; however its lifetime has been extended to 60 years (until 2043). Solid low- and intermediate-level radioactive waste is treated and then packed into steel drums, which are stored in the on-site solid waste storage building. Spent nuclear fuel is stored in the on-site spent fuel pool.

The research reactor TRIGA Mark II is a 250 kWth pool reactor, manufactured by General Atomic, and situated in the vicinity of Ljubljana. The research reactor began operation in 1966, and is operated by Jožef Stefan Institute.

Slovenia is full member state of the European Union (EU) and is required to implement its national radioactive waste management program according to EU waste management directive (EURATOM, 2011). Slovenia has implemented this through our national legislative system, including the Nuclear Safety Act (Official Gazette of the Republic of Slovenia, 2015) and Resolutions on nuclear safety strategy and on waste management strategy (Official Gazette of the Republic of Slovenia, 2016).

This chapter gives a short overview of Slovenia's national nuclear plans and strategy, focusing on the strategy and planning for high-level waste and spent fuel management with deep geological disposal as the final solution for spent fuel and high-level waste management.

17.2 Strategic Planning for Spent Fuel and High-Level Waste Disposal

Current international and national regulatory and legal frameworks require in all EU countries to maintain a national program for managing radioactive waste and spent fuel. The main goal of these programs is not only to fulfil legal requirements but to ensure safe and efficient management of radioactive waste and spent fuel.

In Slovenia, the national program is adopted in the form of a resolution prepared by the Government and approved by the national Parliament. The resolution must be revised every 10 years. The requirements for the resolution are based on EU waste directive (EURATOM, 2011).

But in fact, Slovenia prepared and approved its first national strategic documents well before the EU waste directive. The first strategic document related to radioactive waste and spent fuel management was approved in 1996, only 5 years after our country became independent. This document was the 1996 Strategy on spent fuel management, and included general direction regarding how to manage all spent fuel in Slovenia. In 2004 this strategy was upgraded to include the strategy for NPP Krško fuel management with the Bilateral Agreement on the status and other legal issues related to the investment, exploitation, and decommissioning of the NEK (Official Gazette of the Republic of Slovenia, 2003). After this, in 2006, Slovenia approved the first revision of the national strategy: Resolution on radioactive waste and spent fuel management for the 2006–2015 period. This resolution included all relevant topics for the management of the radioactive waste and spent fuel, from the legislation and identification of different waste streams in Slovenia, to the management of radioactive waste and spent fuel, the decommissioning of nuclear facilities, and the management of naturally occurring radioactive materials (SNSA 2014, 2015).

In the 2006 resolution it was foreseen that spent fuel will be transferred from wet to dry storage between 2024 and 2030, and stored in casks until 2065. After this, the disposal in a deep geological repository (DGR) was planned between 2070 and 2075.

In 2016, Slovenia adopted the second decennial revision of the national strategy: Resolution on radioactive waste and spent fuel management for the 2016–2025 period. This document incorporates several relevant changes affecting spent fuel management plans which have taken place since 2006. It was adopted by the Slovenian government in March 2016 and was approved by Slovenian parliament in May 2016 (Official Gazette of the Republic of Slovenia, 2016). Regarding the spent fuel management and disposal, the main change is that in the 2016 Resolution the dry storage is foreseen to start approximately 6 years earlier, mainly for safety reasons.

17.3 Dual-Track Approach

Experience shows that in every national program, the route to an operational disposal facility is long and uncertain, which exposes each plan to potential failure. The idea of regional and international cooperation regarding radioactive waste management is not new and has its roots in the previous century. Over 50 countries currently have spent fuel in temporary storage awaiting reprocessing or disposal. Not all countries have the appropriate geological conditions for a DGR. In addition to this the financial and human resources required for the construction and operation of a geological disposal facility are daunting, especially for countries with small or very small nuclear programs (El Baradei, 2003; ERDO-WG, 2011).

However, until a shared or international repository for the final disposal of spent fuel and other radioactive waste is realized, each country must also pursue its own national program. National solutions will remain a first priority in many countries. This is the only approach for countries with large nuclear programs with many nuclear power plants in operation or in past operation. For others with smaller civilian nuclear programs, this so called dual-track approach is needed in which both national and international solutions are pursued, such that a country cooperates in a regional/multinational initiative and in parallel develops its own national program for disposal of spent fuel and high-level waste. Small countries following the dual-track approach should therefore keep several options open and must maintain a minimum national technical competence necessary to act in an international context (World Nuclear Association, 2016). This dual-track approach or multinational option for disposal is included also formally in Slovenian national strategy corresponding to the Waste Directive (World Nuclear Association, 2016).

17.4 Multinational Repository References and Planning

In the last five years the economic, environmental, strategic, safety, and security advantages of multinational high-level waste repositories approach (especially for several countries with small programs) are being slowly recognized also by international organizations. The IAEA acknowledges the potential benefits of multinational disposal and, at a legal level, and the EU recognises that this approach can be valuable (EURATOM, 2011). This interest represents a significant change compared to previous decades. Currently there are several active initiatives addressing the issue of multinational disposal, and some time even coordination of these efforts has itself become a challenge (McCombie et al., 2016; World Nuclear Association 2016).

In 2004, the IAEA summarized early work on multinational concepts going back to the 1970s. In 2005 a high-level expert group produced a report on multilateral approaches to the nuclear fuel cycle, covering enrichment, reprocessing and disposal. Further reports addressed the viability of multinational repositories (IAEA, 2011) and framework and challenges for initiating multinational cooperation for the development of a radioactive waste repository (IAEA, 2016). The latter examines the benefits and all risks of a technical, financial, institutional, or socio-political nature.

A major step forward in evaluating all aspects of multinational repository, or shared geological disposal facility, was taken between 2005 and 2009 through the 14-country (all member states of the EU) SAPIERR project, financed by the EU. The project reported on legal and business options, responsibilities, financial liabilities, economics including compensation to hosting communities, security, and public and political attitudes. SAPIERR showed that the most obvious advantages, apart from the credibility of a concrete plan accepted by all parties, are the economic benefits to countries and to the EU as a whole. Partner countries could save billions of euros by sharing development and disposal costs. SAPIERR found that most of the challenges in developing a shared regional repository are analogous to those of a national facility. Finding suitable sites remains the biggest challenge and SAPIERR helped formulate a possible siting strategy (McCombie et al., 2016). The first step in the strategy was the establishment of a self-financing European Repository Development Organisation Working Group (ERDO-WG) of interested countries to prepare a consensus model using the SAPIERR findings as a starting point.

Outside Europe, international entities are studying the potential impacts of multinational storage or disposal, in particular of spent fuel. The Arius Association − which provides also the secretariat for the ERDO-WG − has contributed to Nuclear Threat Initiative and is involved in an International Framework for Nuclear Energy Cooperation (IFNEC) in the fields of multinational repository options and dual-track approaches. Within the Slovenian dual-track approach for final disposal options, ARAO, as the Slovenian waste management organisation, participates at the EU level in programs to address the possibility of building a multinational/regional repository for spent fuel and high-level waste (ERDO-WG and IGD-TP), and is involved also in work of IFNEC.

17.5 National Repository Planning

In 2004 for the first time, disposal of spent fuel from NPP Krško was addressed more thoroughly as part of the first revision of the NPP Krško decommissioning and waste disposal plan (ARAO, 2004). The main purpose of the study was to elaborate key technical solutions and to assess the costs of basic elements of long-term spent fuel management. The disposal reference scenario was developed with the assistance of Swedish and IAEA experts, based on the Swedish concept of disposal in hard rock. This plan for deep a geological repository consists of both underground and aboveground facilities, including the encapsulation plant. Packaging of spent fuel follows the Swedish model, with fuel elements sealed in massive copper canisters with cast iron inserts.

The first plan was improved with the second revision of the reference scenario of national geological disposal in 2009 (IBE, 2009). The second revision was based on same basic concept with improved and more detailed technical solutions and more reliable cost estimates.

17.5.1 Spent Fuel Inventory

Currently all spent fuel in Slovenia is located in wet storage in the spent fuel pool at the Krško NPP. Spent fuel assemblies are kept under water in the spent fuel pool, where they are cooled. The physical capacity of the pool was increased in 2003 from the 828 to 1709 fuel assemblies (SNSA, 2014), (Official Gazette of the Republic of Slovenia, 2016). But for administrative safety reasons the capacity is limited to 1383 locations. At the end of 2015 there were 1154 spent fuel assemblies stored in the spent fuel pool.

Future spent fuel inventory in the country was assessed until the end of the extended lifetime of the Krško NPP (until 2043). It is expected that the number of spent fuel assemblies will total to 2282 fuel assemblies (or 927 tonnes HM). The uncertainty is estimated to be ± 20 assemblies (NPP Krško). The responsibility for spent fuel from Krško NPP is shared between Republic of Slovenia and Republic of Croatia according to Bilateral Agreement (Official Gazette of the Republic of Slovenia, 2003).

Besides spent nuclear fuel from the Krško NPP, other HLW originating from Krško NPP decommissioning and from operation and decommissioning of other nuclear facilities (research reactor TRIGA) will be disposed of in the geological repository. The expected quantities of those are estimated to be (NIS, 2010), (ARAO, 2010) 74 spent TRIGA fuel assemblies (TRIGA fuel assembly has only 160 g of uranium, this totals to less than 14 kgHM) and 236 m^3 of high-level waste including decommissioning HLW from the Krško NPP, institutional HLW, and HLW from TRIGA and potential encapsulation plant.

17.5.2 Conceptual Design of National Geological Disposal Facility for Spent Fuel and High-level Waste

The reference conceptual design is based on the best available current knowledge about future inventories and operation of both nuclear facilities in Slovenia (IBE, 2009). The national concept scenario should include the overall geological disposal program, including research, development, and implementation activities for siting, construction, operation, and closure of a geological repository.

The conceptual reference scenario for spent fuel disposal was developed with the following assumptions: direct disposal of spent nuclear fuel (no reprocessing), the repository will be constructed in hard rock environment at the depth of 500 m. The repository concept and entire disposal system is based on the Swedish KBS-3V concept, developed by Swedish Spent Fuel Management Agency (SKB). The repository development includes also the construction and operation of an underground testing facility at the site of future repository, and assumes that a sufficient cooling period is available prior the disposal to allow full utilization of canisters' capacity (4 out of 4 spent fuel elements per canister).

17.5.2.1 Host rock properties

No site investigations for a deep geological repository have been carried out in Slovenia, and no specific data for geological disposal are available. The reference scenario is made for a generic location in hard rock media. For some specific aspects assumptions and estimates based on expert judgment were used. In addition, while hard rock is foreseen as the basic geological environment for the reference repository, it is not widespread in Slovenia. There is only one region in the north-eastern part of the country where crystalline rock of potentially favourable rock mass and characteristics could be expected.

The revised reference scenario (IBE, 2009) was therefore made for a generic location in a hard rock media in Slovenia. Where required, area-specific crystalline rock properties are applied, where "crystalline rock" refers to igneous and metamorphic "hard" rock. The hypothetical repository location is set in the depth of 500 m beneath the surface.

Hydrogeological properties of Slovenian igneous and metamorphic rocks are quite comparable with hydrogeological conditions in Sweden. It is estimated that the permeability coefficient of this unfractured magmatic and metamorphic rocks ranges from 10^{-8} m/s to 10^{-12} m/s, which is similar to hydrogeological conditions in Sweden. Fractured and weathered rock zones have higher permeability and should be avoided.

Thermal properties of the rock show that the temperatures at depth in areas with igneous and metamorphic rocks can be as much 10 degrees higher compared to what is observed at Sweden sites (11 °C to 14 °C). This should be considered for canister and tunnel spacing calculations. The heat conductivity of magmatic and metamorphic rocks in Slovenia varies between 2.5 and 3.4 W/m-K, which is not essentially different from the Swedish case.

Hydrologic parameters for the site are not available, but from data on analogous rock it may be concluded that the rock mineralogical composition is similar to that from the Kristallin–I (Chapman, 2007).

Geotechnical rock quality is expected to allow for classification of the largest part of repository to be situated within a "good" to a "very good" rock class according to the Bieniawsky Geomechanics classification system (RMR>70); and from "very good to "exceptionally good" rock class according to the

Q-system (Q>20), where generally no support is required for a span of 10 m of the underground structure.

17.5.1.2 Disposal Concept

The conceptual disposal concept for national repository follows the SKB KBS-3V model of disposal. Where necessary, some modifications have been introduced, either because of changes in the original Swedish model, or to adjust technical solutions to Slovenian conditions.

17.5.1.3 Encapsulation of spent fuel

The encapsulation plant, located at the repository site, is a part of the conceptual national disposal concept. As an alternative, encapsulation elsewhere, or even a shared encapsulation process could also be an option, but due to higher expected transport costs it was not investigated for this revision.

The planned plant will contain units for acceptance of transport containers, for encapsulation, for dispatching and transportation of canisters to underground disposal facilities, for treatment and packaging of long lived waste, office building, store, and auxiliary facilities and systems. The planned encapsulation plant in the conceptual model has an annual production capacity of 40 copper canisters per year. Its operation ceases simultaneously with the repository operation. At the end of operation, the encapsulation plant will be dismantled and its contaminated parts processed as radioactive waste. The dismantling should be finished within five years.

17.5.1.4 Canisters

In the conceptual national model the spent fuel is planned to be encapsulated according to the Swedish KBS-3 concept. Fuel assemblies will be inserted and sealed into massive copper canisters. The canister is a 1 m diameter and 5 m high cylinder with 5 cm thick anticorrosion overpack of copper. From the inside it is reinforced by a cast iron insert which can accept four Westinghouse-type pressurized water reactor fuel assemblies. The weight of the full canister is about 25 tonnes. The conceptual design of the canister is presented in Figure 17–1.

17.5.1.5 Repository description

The conceptual design of a deep geological repository consists of underground facilities and a number of above ground facilities necessary for normal underground repository operation. The surface part is connected with the underground part through access shafts and a waste transportation ramp. Illustration of the conceptual deep geological repository lay-out is presented in Figure 17–2.

In terms of operating requirements and requirements for physical protection, the entire repository area is divided into four areas: unfenced area, accessible also to visitors, fenced industrial area under industrial security regime, fenced technological above-ground area under nuclear security and radiological control and underground area.

It is assumed that the construction of the repository will start with construction of an underground test facility (UTF) as part of the site investigation and characterization process.

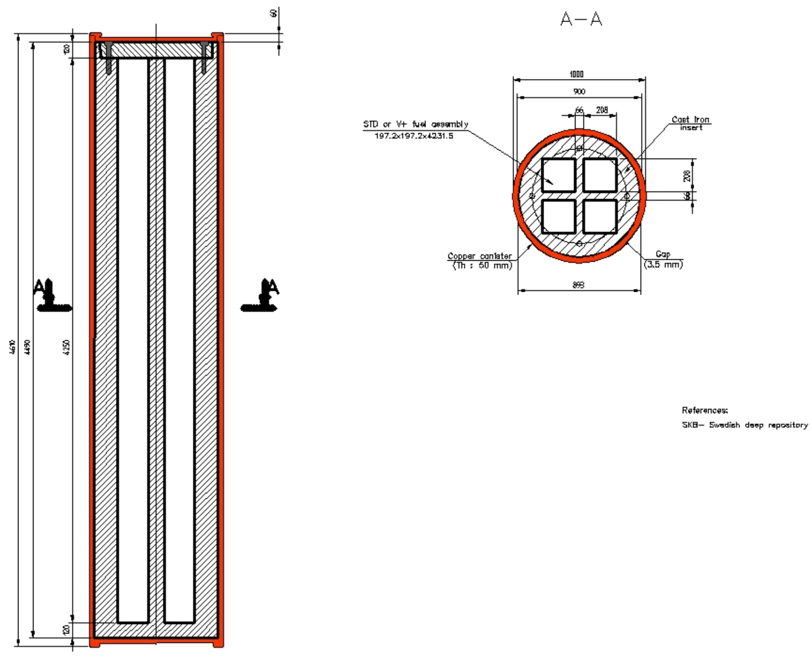

Figure 17–1. Conceptual Copper Canister with Cast Iron Insert (IBE, 2009)

Underground Test Facility: The conceptual design calls for construction of an underground test facility. The facility includes some above-ground structures (utility networks, research/geology building, service building, descent building, etc.) and excavation of underground compartments and connecting structures at depth of 500 m and 800 m alternatively.

Facilities above ground: The aboveground part of the repository includes in the unfenced area: visitor center, utility networks, and rock stockpiles. In the industrial area there are: office building, research/geology building, production building, utility building, store and workshop building. The technological aboveground area includes: service building, descent building, wastewater management facilities, and ventilation building above exhaust ventilation shaft.

In the reference scenario the encapsulation plant is also located at the site of the repository but due to its importance and complexity it is rather considered as separate facility than as part of the above ground repository facilities.

Facilities under ground: The underground part of the repository is situated at a depth of 500 m below the ground surface. Alternatively a depth of 800 m is also considered. It consists of two areas: central service area and disposal area. The underground level can be reached in several ways: for personnel through service shaft, for waste and other cargo through a spiral ramp or alternatively through access vertical shaft with 8.0 m clear diameter. The service shaft is also used as part of ventilation system (air intake). The repository is supplied with a 3 m wide ventilation shaft which can serve as an emergency exit as well.

The horizontal tunnel development is designed as a double loop system. The first loop consists of a large haulage tunnel, designed to provide good accessibility to waste transport vehicles. This loop serves the disposal tunnels and is connected to the access ramp station. The second loop surrounds the service area and is connected to the service shaft and access ramp stations.

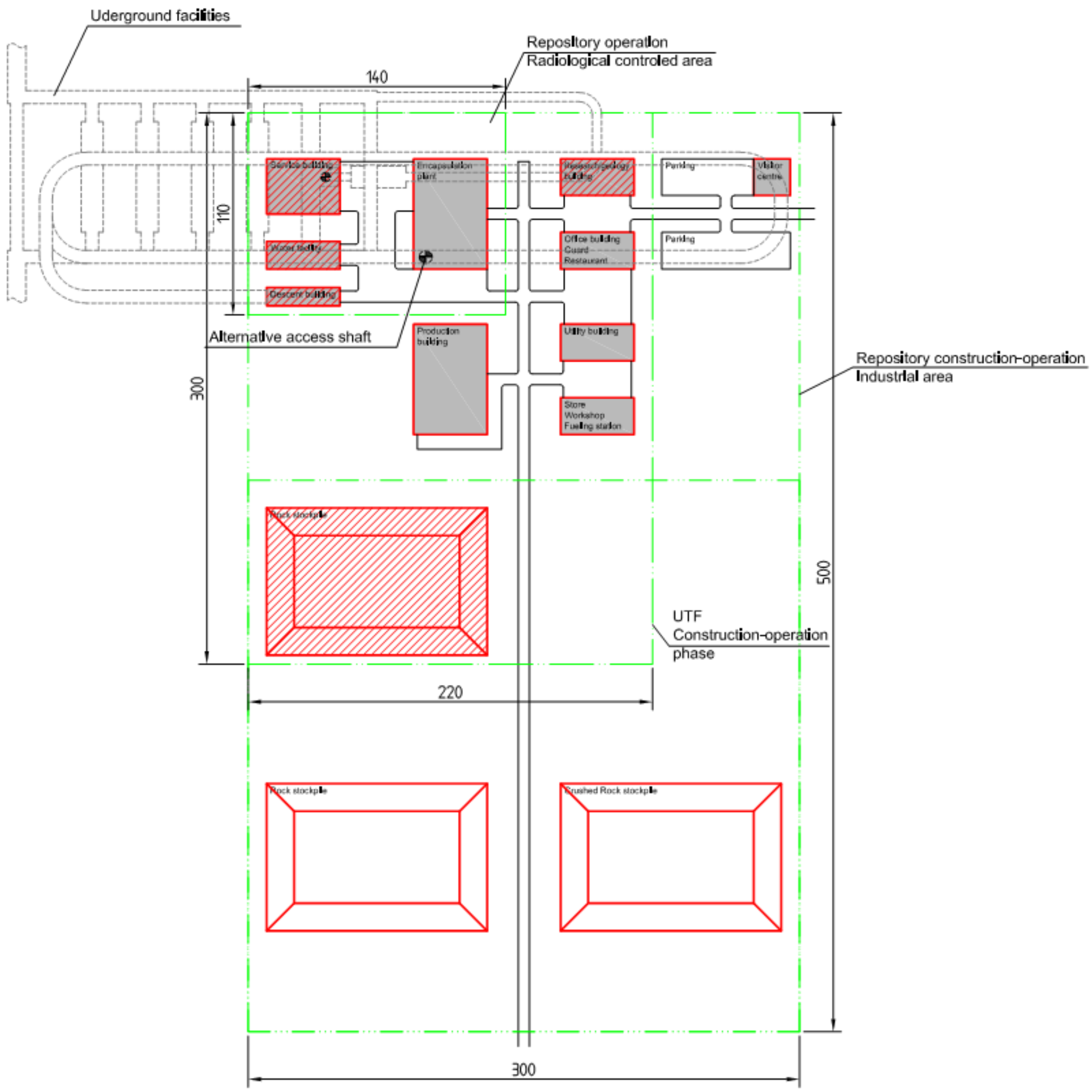

Figure 17–2. Layout of Conceptual Design of National Deep Geological Repository with Indicated UTF Area and Final Repository Area (IBE, 2009).

The central service area is connected to the disposal area which consists of two sections of parallel disposal tunnels. Each tunnel is 207 m long, and the tunnel spacing is 40 m. Canisters are deposited into disposal boreholes at the bottom of each tunnel and surrounded by compacted bentonite. The disposal concept is shown in Figure 17–3. Each tunnel has 21 boreholes spaced 9 m apart.

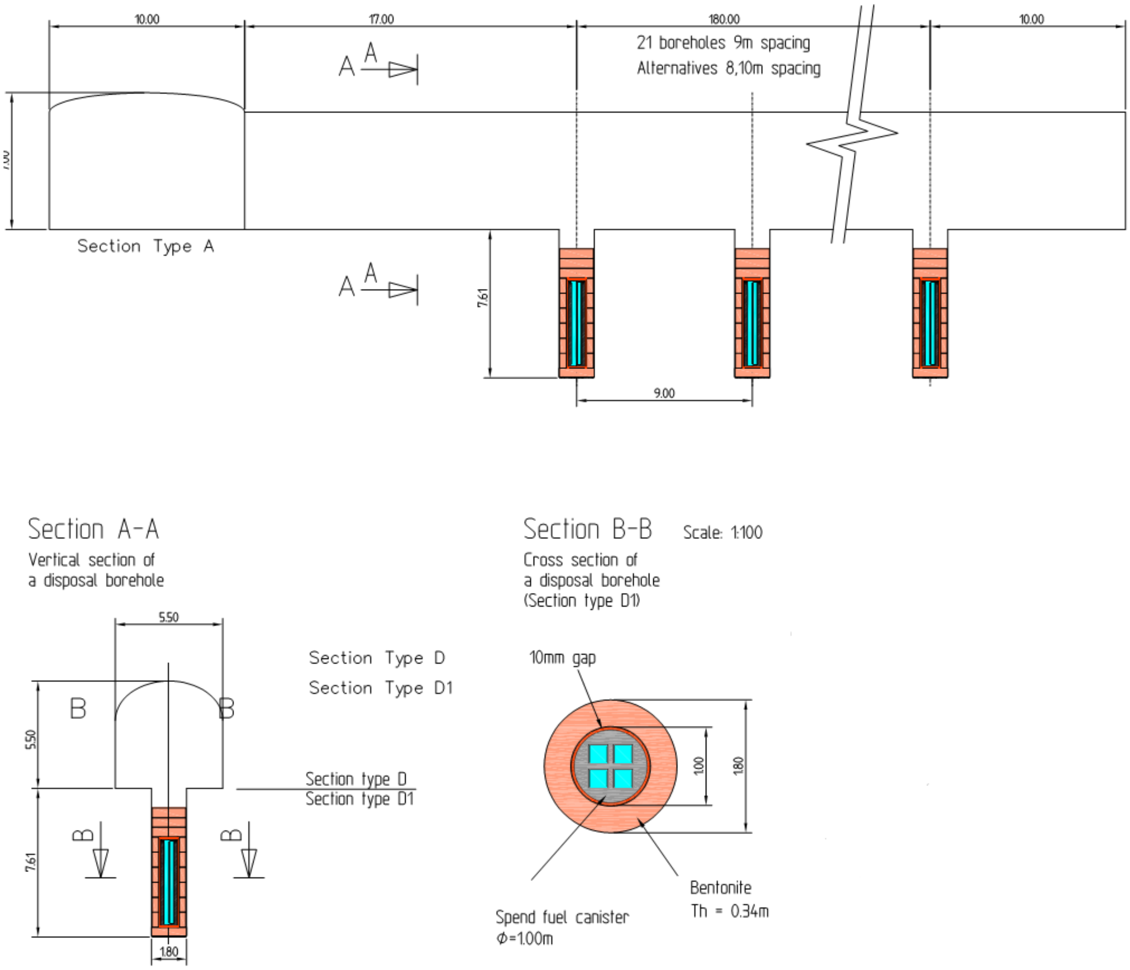

Figure 17–3. Conceptual Design Of Canister Disposal In Tunnel Boreholes With Horizontal And Vertical Cross Sections (IBE, 2009).

17.5.1.6 Time Schedule of the Conceptual National Repository

For the purpose of economical evaluation and with the planned end of operation of the Krško NPP in 2043 and 45 years of dry storage, the conceptual time schedule for the national repository was developed (by the IBE), as shown in Figure 17–4. The timing of events for this scenario are as follows:

The site selection process commences in 2050. Site investigation and characterization includes the construction of an underground test facility. Underground test facility construction is planned to start in 2074. The final site confirmation is assumed in 2081. Planned repository operation is 15 years, starting in 2088; approximately 40 canisters will be disposed of annually. The end of operations is foreseen in 2103-2105.

Decommissioning of technology facilities and closing activities will start after all spent fuel has been disposed of. However, part of decommissioning activities, i.e., backfilling and sealing of disposal tunnels and vaults will begin during the operation of the repository. It is assumed that these activities will last for 7 years, 5 years for decommissioning and 2 years for closure, respectively. Active long-term surveillance is planned to start after closure of the repository. It is assumed that the active institutional control will last for 50 years. Then it will be transferred to passive institutional control of the repository, including mainly activities for record keeping on the repository and the waste.

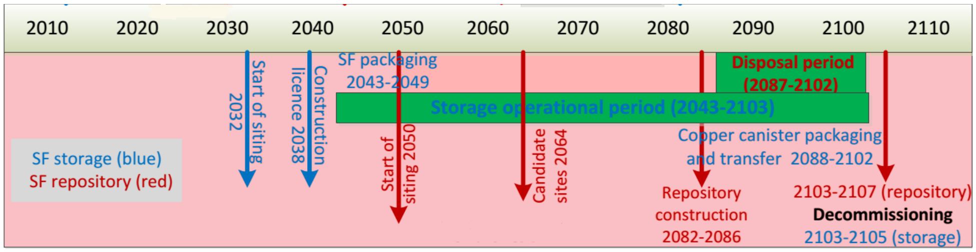

Figure 17–4. Time Schedule for the Spent Fuel Geological Disposal Program Implementation According to the Conceptual Design for Planned End of Operation of the Krško NPP in 2043 and 45 Years of Dry Storage (IBE, 2009).

In the conceptual model the required time from the beginning of siting to operation is 38 years. This could be compared to existing practice in countries implementing deep geological repositories today. On the other hand, the operation phase of repository facility will be shorter, compared to larger programs, mainly due to smaller amount of spent fuel to be disposed of. Still the combined effect of relatively long siting, construction, and operation compared to very small amounts of spent fuel to be disposed off makes this national disposal concept very unattractive from the economic and human resource point of view.

17.6 Future Plans and Conclusion

The Resolution on the 2006–2015 National Program for Managing Radioactive Waste and Spent Nuclear Fuel expired at the end of 2015. However, the process of drafting for a new revision had already been started in 2014, when the national waste management organisation ARAO prepared technical inputs for the draft of a new resolution. The final document Resolution on the 2016–2025 National Program for Managing Radioactive Waste and Spent Nuclear Fuel (ReNPRRO16-25) successfully passed a public consultation process and was approved by the Slovenian Government (March 2016) and Slovenian Parliament in April 2016 (Official Gazette of the Republic of Slovenia, 2016) .

In this Resolution the conceptual idea of shared facilities and regional cooperation in waste management, including the dual-track approach, was clearly implemented. For long-term spent fuel management, a dual-track strategy has been adopted as a reasonable solution in the present situation. The dual-track approach in Slovenian strategy includes both first, the option of multinational disposal is kept open, and second, the basic reference conceptual scenario for national geological disposal is included. Conceptual national disposal concept is based on above described approach. However the revised national program (Official Gazette of the Republic of Slovenia, 2016) opened the possibility for the beginning of disposal of spent fuel in national repository slightly earlier (compared to IBE 2009). In the revised national program the beginning of national disposal of spent fuel is assumed in or after 2065. In parallel to national disposal program the multinational disposal option is possible. Both options go in parallel until the beginning of construction of national repository between 2055 and 2065. This dual-track approach to disposal program for spent fuel and high-level waste is schematically represented in Figure 17–5.

First, shared responsibility and the opportunity to safely and sustainably resolve issues related to radioactive waste and spent fuel management together with other countries was implemented in the general policy on management of radioactive waste and spent fuel, which included the principle of international cooperation regarding radioactive waste and spent fuel. This principle states that actions of Slovenia regarding shared responsibility and the opportunity for safe and sustainable solutions related to radioactive waste and spent fuel management should take into account the principles adopted in the Resolution and policy and regional and international agreements. National responsibility for radioactive waste and spent fuel management is considered in parallel with active participation in international, regional efforts to make progress in connection with joint regional programs on disposal (Official Gazette of the Republic of Slovenia, 2016).

The Resolution requires that spent fuel owners evaluate reprocessing as an option that could reduce the volume and radiotoxicity of waste for final disposal (JRC 2014). However at the end all spent fuel (in direct – un-reprocessed form) or/and high-level waste (reprocessing remains and direct high-level waste) require a disposal in the form of deep geological disposal of spent fuel and high-level waste. The construction of deep geological disposal (national, regional, or multinational) is necessary regardless of the selected option for storage, processing, and other forms of spent fuel management (Official Gazette of the Republic of Slovenia, 2016), (JRC, 2014).

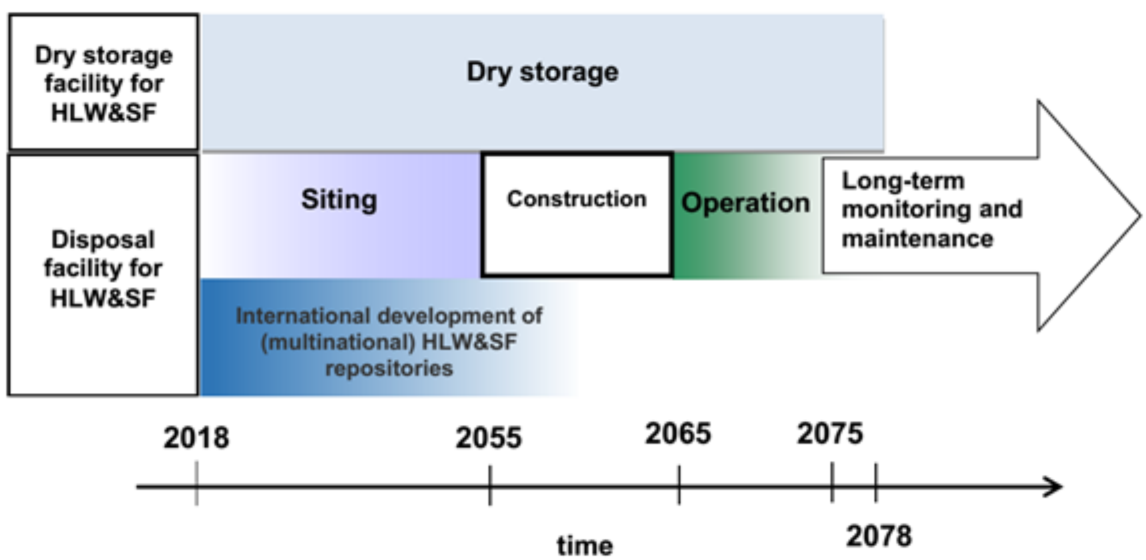

Figure 17–5. SF and HLW Management Diagram According to Adopted Long-term Strategy from ReNPRRO16-25. Both National and Multinational Options are Presented in Parallel in the Diagram (Official Gazette of the Republic of Slovenia, 2016).

One of the significant challenges before Slovenia is the shared responsibility for the spent fuel and waste from shared Krško NPP. Both countries have to respect the obligations of the Bilateral Agreement to share the responsibility for waste management. An important step towards well defined shared responsibility is a well-developed Krško NPP Decommissioning Program and Krško NPP Radioactive Waste and Spent Fuel Disposal Program. Both programs have to be prepared by responsible organizations in both countries and approved by Bilateral Commission supervising the implementation of the Bilateral Agreement. The process of new revision of both programs was started recently according to Bilateral Agreement between Croatia and Slovenia with drafting the necessary Terms of Reference for both Programs. According to the current development, a new revision of the technical study for a geological disposal facility in hard rock with cost estimation (IBE, 2009) will be prepared in 2016 under the umbrella of revised Krško NPP Radioactive Waste and Spent Fuel Disposal Program. This revision will include new technical and technological developments and, on the basis of changes, a new cost estimate for a repository at generic location anywhere in Republic of Slovenia or Republic of Croatia. The document will contain a timeline of activities scheduled according to both National Programmes for radioactive waste and spent fuel (Official Gazette of the Republic of Slovenia,2016), (National Program, 2014) and to the Boundary Conditions for the Revisions of the Krško NPP Decommissioning Program and the Krško NPP Radioactive Waste and Spent Fuel Disposal Program.

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17.8 Acronyms

ARAO—Slovenian waste management organization

DGR—Deep Geological Repository

ERDO-WG—European Repository Development Organisation Working Group

EU—European Union

IAEA—International Atomic Energy Agency

IFNEC—International Framework for Nuclear Energy Cooperation

IGD-TP —Implementing Geological Disposal of Radioactive Waste Technology Platform

KBS—Abbreviation of kärnbränslesäkerhet, nuclear fuel safety

NEK—Nuklearna elektrarna Krško

NPP—Nuclear Power Plant

SAPIERR—Support Action on a Pilot Initiative for European Regional Repositories

SKB—Swedish Spent Fuel Management Agency

TRIGA—Research Reactor

UTF—Underground Test Facility

Chapter 18

Radioactive Waste Disposal in South Africa in 2015: Status and Research and Development Strategies

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ABSTRACT: The decade covered by this Review represents a transitional, pivotal period in the way research and development has been conducted for strategies to manage the radioactive waste produced in South Africa. In future, these activities will be intensified, being the mandated responsibility of the National Radioactive Waste Disposal Institute that became operational toward the end of the Review period. Relative to previous Reviews, there are no significant, projected increases in radioactive waste generated for existing nuclear reactors even though the future construction of additional ones for electricity and isotope production remains a possibility. During the past decade the Necsa generated waste continues to be stored in monitored interim facilities, while the performance of the Vaalputs National Radioactive Waste Disposal Facility for LLW was comprehensively re-evaluated, and found compliant with the best international practices. In the same period, R& D activities were intensified and diversified. On the research side a number of postgraduate projects and dissertations were successfully completed, and a number of benchmark papers were published in high profile international journals. Such investigations dealt with aspects of soil sciences, radioactive elements geochemistry, tectonics, regional stress, geohydrology and climate/paleoclimate, often providing new finds that challenged old ideas. On the Development side, much collaborative work with international organizations and research laboratories went into the successful development of novel technologies for the disposal/treatment of disused, sealed radioactive sources, and of post reactor materials. The methodology for the former is ideal for countries without an established nuclear infrastructure and is referred to as Borehole Disposal of Spent Sources (BOSS). Accordingly, the waste is encapsulated in specially designed boreholes to be constructed on sites whose characteristics in terms of geology, environment and logistics are also carefully prescribed. This Review also describes the selection of a BOSS site at Vaalputs.

18.1 Introduction

Since the Fourth Worldwide Review of 2006, the South African Government established the National Radioactive Waste Disposal Institute (NRWDI) as a state-owned institution through the NRWDI Act (Act 53 of 2008), which also describes the functions of the NRWDI and the relevant powers to perform such functions. The NRWDI was formally launched by the Minister of Energy in March 2014, and currently is in the process of being operationalized. The strategic mandate of the NRWDI is to manage radioactive waste on a national basis, independently of the producers of such waste. NRWDI's role and functional scope comprises the management of a national inventory of all radioactive waste to be disposed, including long-lived waste, high level waste, used nuclear fuel, and disused, sealed radioactive sources. This means that the NRWDI must initiate an appropriate research and development process to investigate alternative disposal concepts for obtaining, characterizing and constructing disposal sites.

While the NRWDI is in the process of being operationalized, the South African Nuclear Energy Corporation (Necsa) continues to discharge the functions of the NRWDI for day-to-day disposal operations, to conduct low-key geological investigations, and to develop innovative technologies for immobilization and disposal of radioactive materials. In particular, significant effort was invested in planning to dispose disused (spent) and sealed radioactive sources in boreholes at Vaalputs, which is the only South African Low-Level radioactive waste-disposal facility. The Vaalputs Radioactive Waste Disposal Facility is operated by Necsa. The facility is located about 100 km southeast of Springbok, in the Northern Province. It covers approximately 10,000 hectare, measuring 16.5 km from east to west, and 6.5 km from north to south at its narrowest point (from https://en.wikipedia.org/wiki/Vaalputs). The borehole disposal has been identified as being the optimal disposal solution for the country's spent radioactive sources, currently scattered among a variety of unregulated sites. In this current review, the authors highlight the most significant, recent achievements in the characterization of the Vaalputs and Pelindaba sites, and in the disposal and immobilization of radioactive materials.

18.2 Sources of Radioactive Waste in South Africa

The main generators of Used Nuclear Fuel (UNF) and other long-lived waste potentially destined for a geological disposal facility in South Africa are Necsa, with its SAFARI-1 reactor for isotope production and research, and Eskom's Koeberg Nuclear Power Station (KNPS). Necsa's location is ca. 20 km west of Pretoria, and Koeberg ~30 km north of Cape Town (see Figure 18–1 inset). Current projections show that SAFARI-1 will produce about 5 m^3 of UNF and the KNPS about 3,000 USF assemblies (including ~1,500 tons of uranium) during their lifetimes. About 10,000 m^3 of long-lived bulk waste and an unknown quantity from industrial and medical industries are also potentially earmarked for geological disposal. In the Fourth Worldwide Review (Andreoli et al. 2006), the authors mentioned a possible pebble bed modular reactor program being implemented in South Africa. Since then the program has been discontinued for various technical and political reasons. However, a program by the South African Government to build additional nuclear power stations, including the successor to the Safari-1 reactor (which recently celebrated its 50th anniversary), remains a possibility.

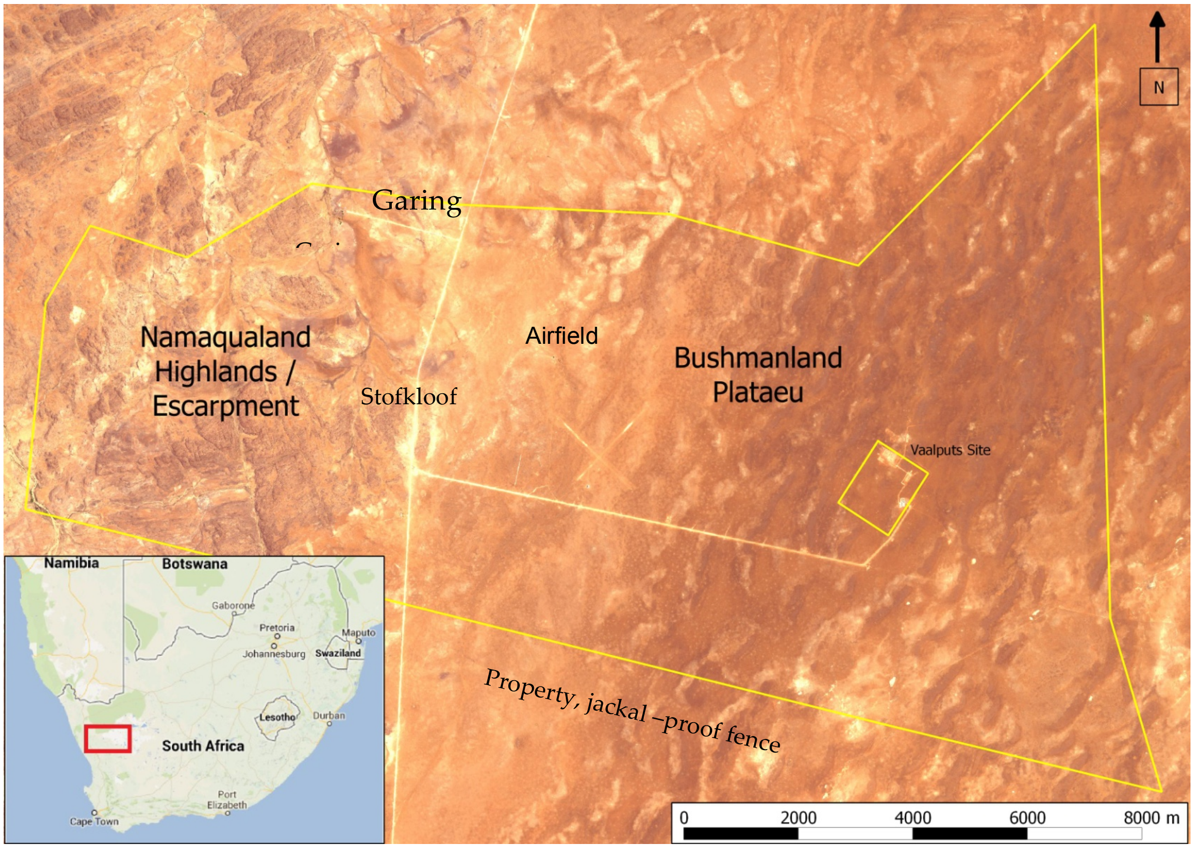

Figure 18–1. Satellite-derived outline of the Vaalputs property and major infrastructure locations. The disposal site lies 100 km south of Springbok and to the east of the Namaqualand highlands which form part of the escarpment (left of the Stofkloof and Garing facilities) running along the western coast of South Africa.

18.3 Necsa-managed Radioactive Waste Storage Facilities in South Africa

Currently, radioactive waste (Used/Spent Nuclear Fuel (USF)), "hot cell" waste, and other historical waste) produced by Necsa and its Nuclear Technology Products (NTP) subsidiary, are stored at several *interim* storage facilities within the Necsa site. More specifically, the USF from the Safari-1 reactor is currently stored in the authorized Thabana dry-storage facility at the Necsa's Pelindaba site, which is currently being extended to increase its storage capacity. Low Level Waste (LLW) generated by the Koeberg Power Station and by Necsa is disposed_at the Vaalputs National Radioactive Waste Disposal Facility in the Northern Cape. Vaalputs complies with the general guidelines deemed suitable for the safe disposal of LLW as considered in typical Generic Post-Closure Safety Assessments (van Blerk et al. 2007). In addition, the Vaalputs facility, in operation since 1984, also presents a logistical advantage that makes it the logical solution for disposal of disused sources (Figure 18–2).

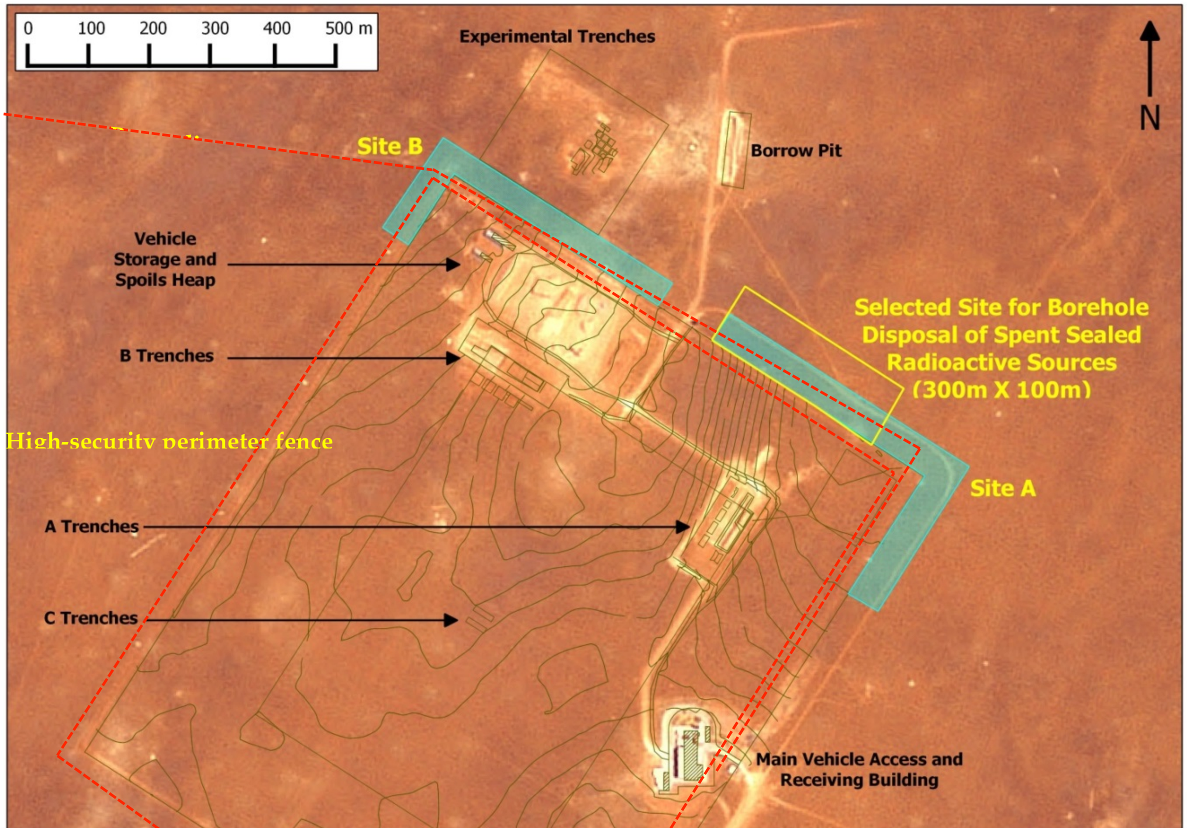

Figure 18–2. The site infrastructures within and around the security fence. Red dashed lines are the main electrical infrastructures (power line, high-security perimeter fence). Green blocks (Sites A and B) are areas considered for the potential boreholes disposal of radioactive sources (see text for discussion).

18.4 Research and Development at Vaalputs

18.4.1 General Approach

The limited resources available to characterize the geology of the Vaalputs site continue to be offered to researchers from South African and overseas institutions who are interested in pursuing areas of specific academic interest. Efforts are also made to insure that the knowledge base accumulated in past decades is transferred to a new generation of students and academics, so that it may benefit future initiatives by the NRWDI. In general, the focus of current research has now shifted from the study of the crystalline basement to the sedimentary and soil cover, regional geology, tectonics, seismicity, groundwater characteristics, rock stress, climate change, and natural radioactivity. The immediate aim of such investigations is to contribute to the Safety Case for the currently disposed low level radioactive waste.

On the engineering side, Necsa scientists played a major role in an IAEA/AFRA project to develop a BDF (Borehole Disposal Facility) system, this being the final step of a technological process called BOSS (Borehole Disposal of Spent Sources). The BOSS/BDF concepts are designed to provide a detailed, engineered level system which allows for: a) pre-disposal activities, such as characterization and conditioning of disused sources, and b) the safe and permanent disposal of disused sealed radioactive sources (DSRS's) in specifically built boreholes.

18.4.2 Earth and Environmental Sciences

Mineral Potential of the Precambrian Geology. An important issue for Precambrian geology is the search for pathfinders to economic mineral deposits that may be present in the ~1,000 my old Namaqualand belt basement under the blanket of Cenozoic deposits (Maier et al. 2012). Of particular interest are the (Th-) monazite deposits of the so-called Steenkampskraal type (Andreoli et al. 2006, and references therein; Clay et al. 2014). Because granite veins with monazite, and thorium anomalies occur in and around the Vaalputs site, the possibility of a 10 m thick lode of Th-rich under the site, albeit unlikely, may not be excluded until more is understood about the geology of the site. The Namaqualand monazite occurrences are now being re-investigated by Dr. Daniel Harlov, monazite specialist at the German GeoForschungsZentrum (GFZ)-Potsdam.

Cenozoic Geology: Clues to Climate Change and Soil Disturbances. Much progress has been made since the last review in characterizing Cenozoic sediments and palaeo-sols exposed in the increasing number of trenches cut at the Vaalputs site (Figure 18–2). These sediments are now recognized as precious archives recording climatic change going back in time to the end of the Cretaceous. The University of the Witwatersrand, Johannesburg, is currently using infra-red stimulated luminescence to date the deposition of the red sand that covers the Vaalputs site is currently underway, while the University of Johannesburg is using the ^{40}Ar-^{39}Ar method to determine the age of the illite of the underlying fluvial, clay-bearing arenaceous sediments. These latter rocks are critical to understanding the site performance in terms of dispersion of the radioactive waste into the environment (Figure 18–3). Among the various multidisciplinary investigations of the trenches (the Near Field; Andreoli et al. 2014, 2015), we are also studying the origin and evolution of the silicified ferruginous layer that typically lies between the red sandy soil and the clay-bearing sediments of the trenches (Figure 18–4; Clarke et al. 2016). This rock, often cut by vertical soil tongues and low angle shear fractures, probably defines a major episode of climatic/geohydrological change, and stress release between ~70 and ~60 ka BP (Evans et al. 2015; Andreoli et al. 2014). The processes leading to widespread pockets of oxidized sediments (see the "oxidized "bowl" in Figure 18–3) are also being investigated because the latter suggest still undetermined phenomena of stratigraphic disturbance and open-system behavior by near-surface water.

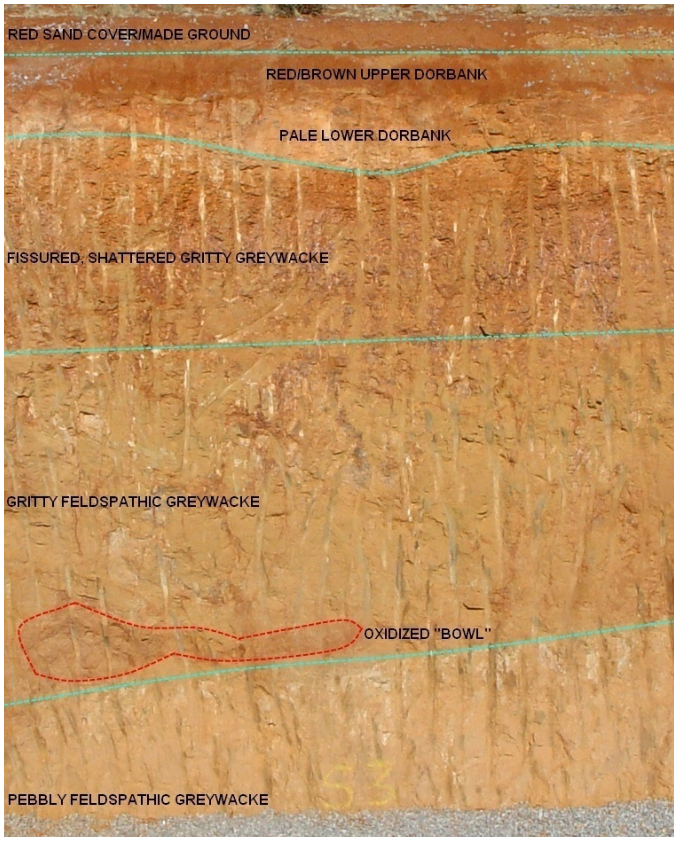

Figure 18–3. Representative stratigraphic profile in Trench B (1, 2).

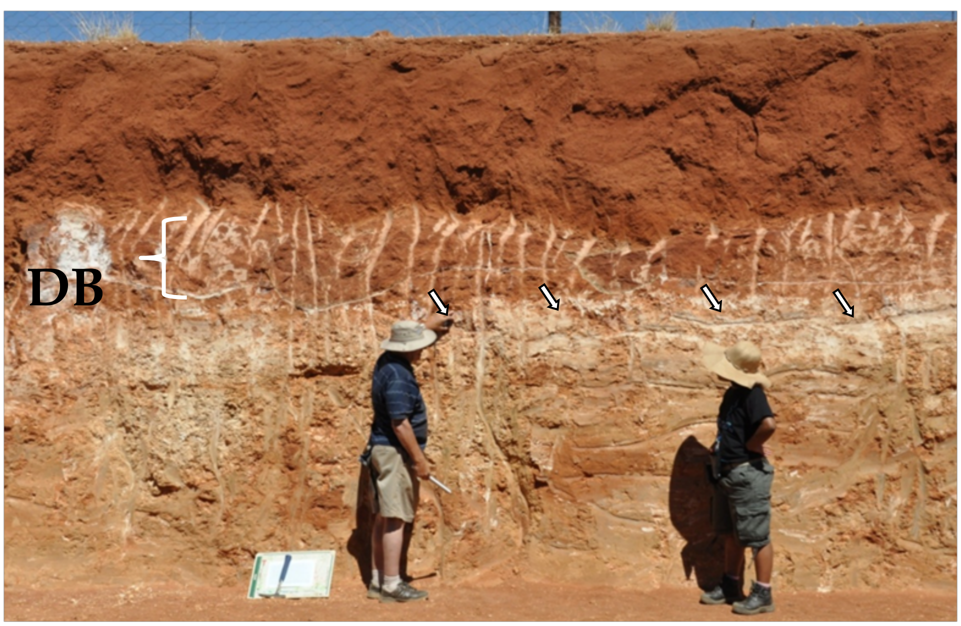

Figure 18–4. Detailed stratigraphy of Late Pleistocene palaeosols in Trench A (0,5), Vaalputs: arrows mark band of sepiocrete nodules at the contact between calcretized, pale-coloured greywackwe (age of calcrete: ≥70 ka) and light brown dorbank (DB). Top: red sand (≤60 ka; Evans et al. 2015).

Site Seismology and Neotectonics. The monitoring of the sporadic seismicity of the broader region around Vaalputs (mentioned in our contribution to the 4th Worldwide Review) has continued in the following years until 2011, when the two Necsa seismic stations installed in 1989 finally ceased functioning. The seismic activity is now monitored by a new network of three seismic stations (one being broadband) forming a triangular array positioned to include the largest number of expected future epicenters. Substantial effort is currently placed on interdisciplinary research to identify the origin of the tectonic force responsible for these events. The available data suggest that the seismicity in the Vaalputs area derives from the interplay of several forces, the weakest being predominantly extensional and induced by the southern propagating East African Rift. The main force, named the Wegener stress anomaly, is horizontal and strike slip in character with a NNW-SSE orientation--see ellipse in Figure 18–5 (Bird et al. 2006). The cause of the Wegener compressive, coast-parallel stress remains unidentified (Andreoli et al. 1996; Viola et al. 2012; Andreoli et al. 2013). As part of the effort to explain the seismic activity in the greater Vaalputs area, Necsa continues to invest in a number of initiatives. These include support for a) a Master of Science (M.Sc.) project to obtain neotectonic stress orientation data in South Africa from off-shore borehole break-out measurements (successfully completed; University of Cape Town), b) a Ph.D. project on the seismicity of the Northern Cape (University of the Witwatersrand), c) a M.Sc. project to establish a GIS database of neotectonic faults in South Africa (University of Pretoria). Additional research initiatives on the subject of neotectonics focus on issues of surface development and land stability (Evans et al. 2015; Andreoli et al. 2015) and morphotectonics. The latter study is greatly aided by recent developments in Digital Terrain Model (DTM) analysis (Figure 18–6) to assess the reactivation potential of the basement faults in the Vaalputs area (Viola et al. 2012). The evolution of the landscape, and, in particular, the age of the 1-km uplift of the Vaalputs plateau are also the subjects of careful investigation, as they are controversial (Kounov et al. 2009).

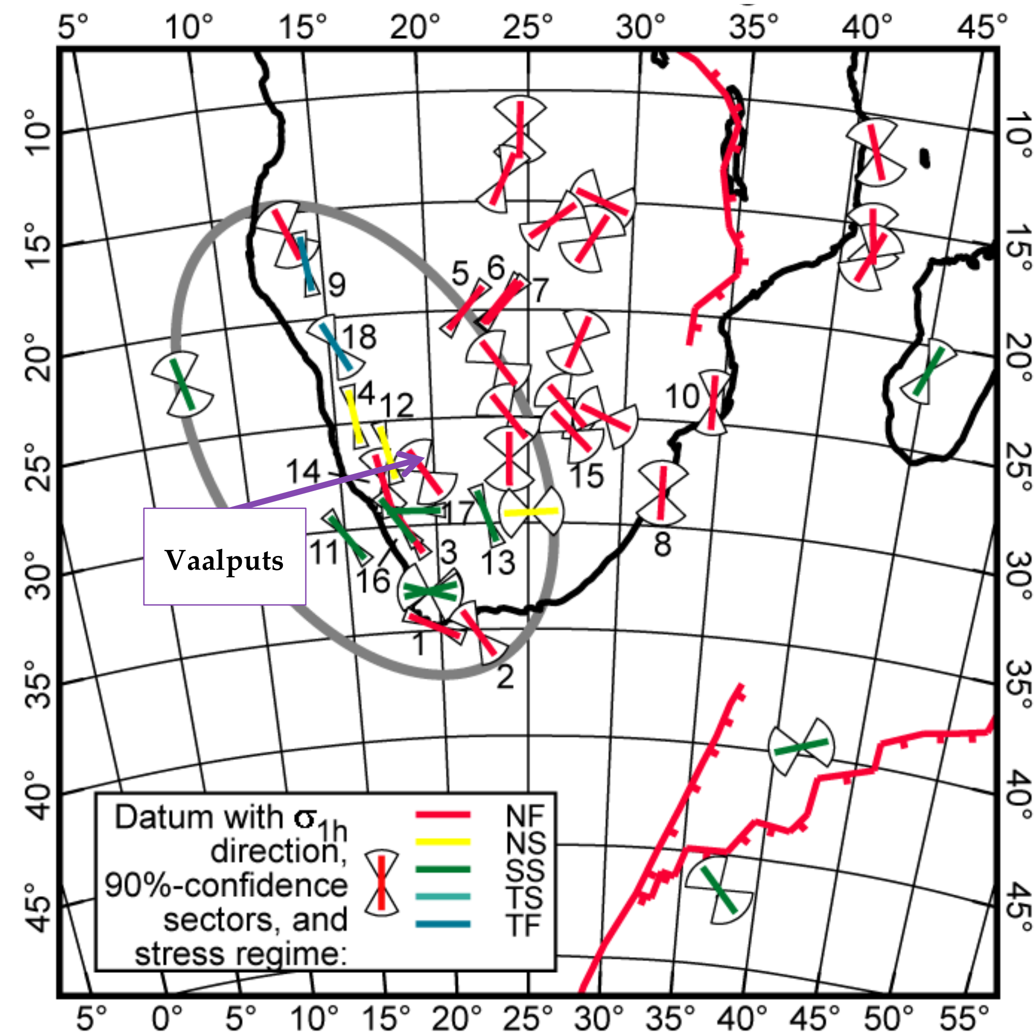

Figure 18–5. Intraplate indicators of stress regime (indicated by colour) and azimuth of the most compressive horizontal principal stress. Ellipse: approximate region of the Wegener stress anomaly (Bird et al. 2006).

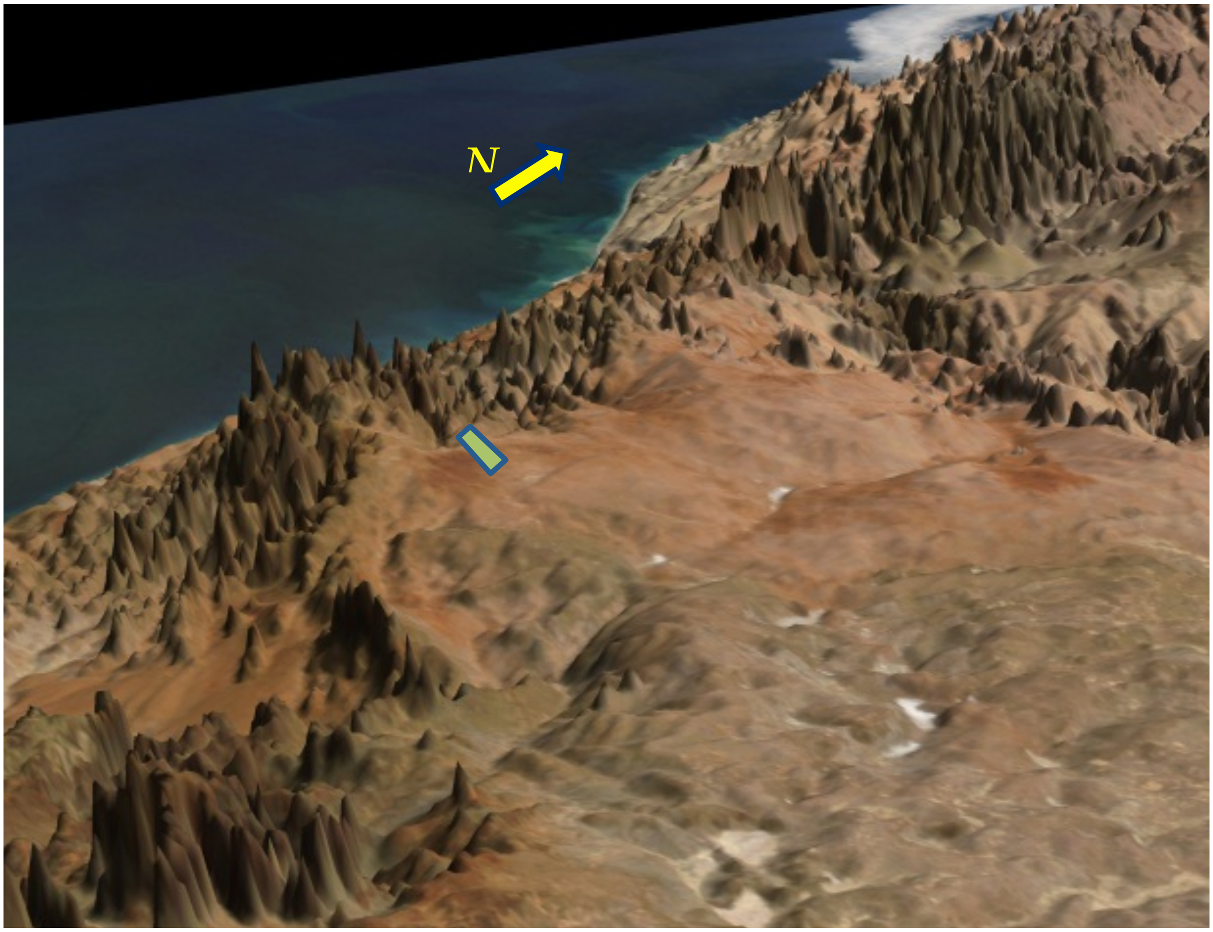

Figure 18–6. Digital terrain model, with vertical exaggeration of the Vaalputs (green polygon) and adjacent areas of the Northern Cape (credit: Dr. F. Eckardt, University of Cape Town).

Alexandre et al. (2006; 2007) studied the regional geology and tectonic stability of the area around the Pelindaba site. An additional investigation was also conducted at the Pelindaba site as part of the discontinued pebble bed modular reactor project. This latter investigation included a study of the regional Brits graben fault that underlies the Pelindaba site. Trenching across the fault indicates that it lies buried under an undisturbed Cenozoic soil cover and, as such, it is inactive under the foreseeable geodynamic conditions.

Geohydrology: Natural Radioactivity and Pathways Modeling. The study of groundwater, its age, movement/recharge dynamics and interaction with the naturally occurring radioactive minerals of the granitic country rocks has taken centre place in recent years by supporting two M.Sc. projects (University of the Free State). It must be stressed that these studies are based on water sampled from fractures in granitic host rocks at a depth of about 54 m. The first project involved statistical processing of more than 20 years of groundwater analyses for the period preceding the disposal of uranium bearing waste at Vaalputs. As such, the study by Pretorius (2012) provides a baseline insight of the behavior of uranium and its decay chain radionuclides as they have partitioned through time between host rock minerals and groundwater. The available data show evidence of occasional, statistically significant releases of radionuclides, mainly ^{234}U, into the groundwater for reasons that are currently under investigation (Figure 18–7). The second M.Sc. project is based on the reconceptualization of the groundwater regime in the Vaalputs site and surrounding farms, and aims to address uncertainties that were raised in the 2007 post-closure radiological safety case assessment (van Blerk, 2007), to gain understanding of the groundwater regime, namely hydrogeochemistry and recharge. The major and minor element data (Figure 18–8) confirm a Na-Mg-Cl rich groundwater under stagnant conditions (Levin, 1988).

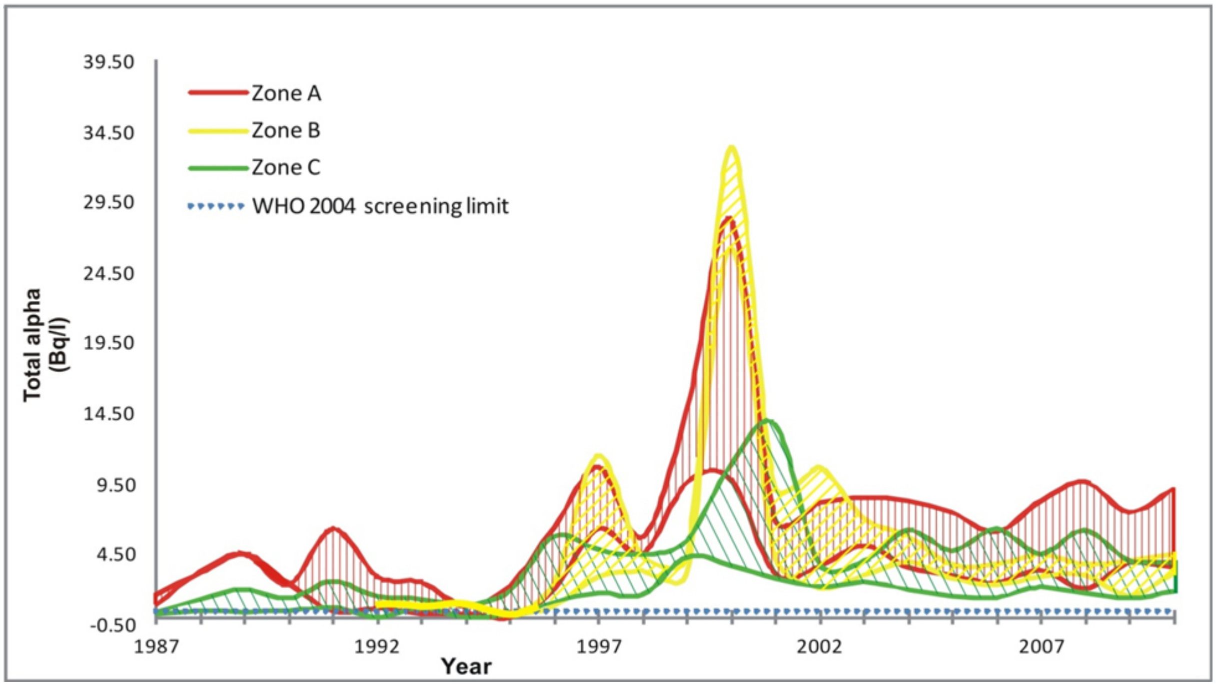

Figure 18–7. Radioanalytical data of groundwater from boreholes in the Vaalputs site and nearby areas: Timeline of total α radiation for boreholes from zones of increasing distance from the trench area (Zones A, B and C: boreholes <2 km, <6 km, and >6 km from disposal trenches respectively).

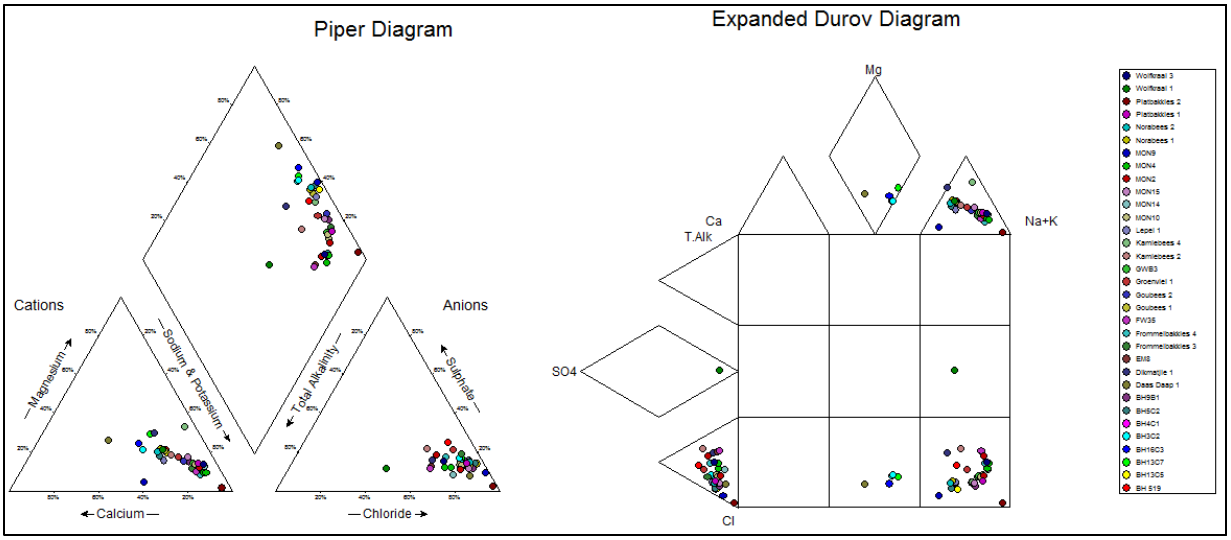

Figure 18–8. Piper and Piper Expanded plots of groundwater from boreholes at Vaalputs and surrounding farms. The concentration of points in the last two 8 and 9 segments suggests old water (unpublished Necsa data by M. Mandaba).

18.4.3 System Engineering

BDF in a nutshell. This technological development provides a safe, economic, simple and permanent solution to spent sources management, and is particularly suitable to countries that do not have an established nuclear infrastructure for radioactive waste management and disposal. The concept, which consists of a system of multiple, physical and chemical barriers_(near-field engineering), is widely applicable as it is versatile and suitable for a range of geosphere, biosphere and climatic conditions. It provides for:

- Selection and characterization of a disposal site
- Construction of a dedicated disposal borehole
- Transport of sources (conditioned in stainless steel capsules) from storage to disposal site
- Containerization of sources into high integrity, stainless steel disposal containers
- Emplacement of the disposal containers in the borehole
- Sealing and closure of the borehole, including restoration of the site
- Site maintenance as per operational and post closure safety assessment requirements

A schematic illustration of the disposal of the sources in a borehole according to the BDF is shown in Figure 18–9. As indicated in the checklist above, a key component of BDF/BOSS system is the selection of an appropriate disposal site. Lessons learned from decades of investigations for the nuclear industry have taught that “the perfect site” is like a safe, high yield investment: impossible to find, given the inherent complexity of every natural system, irrespective of how simple it may appear at first. Table 18–1 lists a number of prominent guidelines (Quintessa 2003, 2005) that will nevertheless assist in the site selection process in order to find a suitable site to implement the BDF concept. For the reason indicated above, it is the aim of a selector to find, if not necessarily the ”perfect site” at least one which is suited to meet all key requirements for safety in a way that can be appropriately justified.

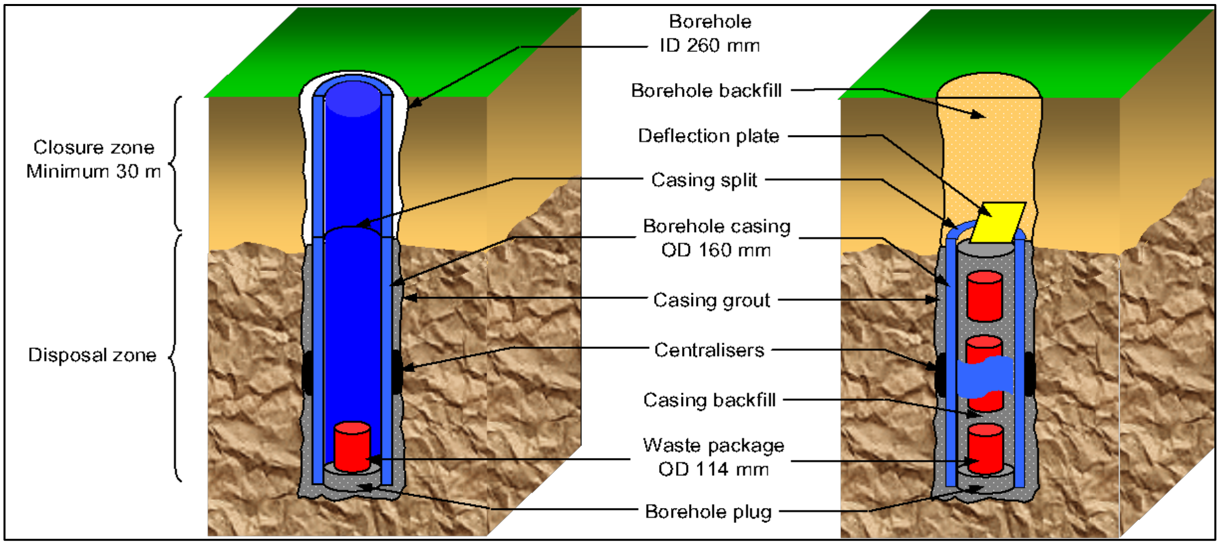

Figure 18–9. Schematic representation of the emplacement borehole (Chaplow and Degnan 2010).

Table 18–1. A Checklist of Desirable Characteristics in Disposal Systems (Quintessa 2004)

Geology	• Low tectonic and seismic activity • Absence of geological complexity • Distance from present or potential mineral exploitation (open cast/underground mining) • Low geothermal heat and gas resources potential
Geomorphology	• Limited geomorphological activity • Borehole disposal zone below local erosion base level
Hydrogeology	• Simple hydrological system • Stability of water table (limited fluctuation of saturate/unsaturated zone interface) • Slow moving ground water (long travel times from borehole to biosphere) • Appropriate dilution along the geosphere to biosphere pathway
Geochemistry	• Significant sorption of radionuclides (e.g., through reducing conditions) • Avoidance of conditions impairing the longevity of engineered barriers (e.g., high SO_4^{2-}, Cl^-)
Climate	• Dry conditions (low rainwater percolation, weathering rates)
Society	• Avoidance of urban areas

In the case of an application of the BDF concept at Vaalputs, this facility was originally selected as it satisfies and incorporates most of the criteria set out in Table 18–1 as best suited for LLW disposal sites. In particular, the low rainfall/low recharge and the great depth of the water table make the Vaalputs site particularly suitable for LLW and spent sources disposal. Similarly, the outcomes of the neotectonic and seismological investigations satisfy the requirements for tectonic stability at the disposal site (low ground acceleration and seismic hazard probability). For the BDF concept, however, there are additional constraints and criteria to consider (Raubenheimer 2012), these being of a rather practical and operational nature. More specifically, the following considerations are envisaged:

- **Placing** the BDF area for spent sources adjacent to the current security fence of the LLW disposal site. Following the emplacement phase, when a temporary security fence might be built to protect the waste, an easier-to-control permanent fence will enclose the perimeter of all security areas.
- **Siting** the BDF beyond the existing security fence will provide a >200 m buffer space from the trenches of the LLW disposal area. This will prevent the co-impact of the different facilities and limit the maximum source term without affecting the long-term safe performance of the Vaalputs site.
- **Avoiding** the LLW infrastructures (e.g., roads, experimental areas) will minimize operational disruption and risks to existing nuclear licenses and operations.
- **Avoiding** areas of deeper, clay-bearing overburden, as these may be needed should the current LLW disposal site require future expansion.
- **Avoiding** electrical infrastructures to minimize any risk of electrocution or damage to wiring. Given the current position of the power line, the latter will need to shift northeast by establishing a 30 m exclusion zone. A 15 m exclusion zone will also run parallel to the potentially electrified security fence (see Figure 18–2).
- **Allowing** generous dimensions for the BDF site (~100 m × 300 m) to provide disposal space to last at least a few decades, and unhindered expansion should more space be needed in the far future.

18.4.4 BDF Site Screening and Selection

Using the above criteria, two candidate areas (see Sites A and B, Figure 18–2) were selected from which a smaller final site will be chosen for borehole disposal. Sites A and B are both feasible, considering both geotechnical/geological and non-geotechnical aspects. Should a specific area be selected on the basis of specific considerations (see below), the next step will involve the acquisition of site-specific parameters for a rigorous Post Closure Radiological Safety Assessment.

Logistic/geotechnical considerations. Both sites are adjacent to the current security perimeter and accessible by the current road network (Figure 18–2). They satisfy most of the abovementioned considerations. If a distance of 10 m between the disposal boreholes is maintained, a large number of disposal boreholes can be accommodated for many years within Site A south of the power line, even without shifting the adjacent road,.

Environmental and geological considerations. In a previous section mention was made of several types of structural//sedimentary features within the overburden. However, such features (low angle shear fractures, penetrative soil tongues, etc.; Andreoli et al. 2014, 2015) are confined to the top few metres and will not be considered further in terms of potential hazard to the sources disposed in the boreholes. Sufficient casing and grouting would minimize the risks associated to water percolation along such features. Drilling and geophysical data further indicate that the basement geology underlying Sites A and B consists of high grade, >1.0 Ga old granitic rocks largely free of mafic/anorthosite intrusions, potentially prone to deeper weathering. High resolution (airborne, ground) magnetic survey data indicate, however, that both Sites A and B are in close proximity to NW-SE trending, probably normal faults which should be avoided not because of their unlikely reactivation potential, but rather because they may represent potential nuclide migration pathways.

Geohydrological considerations. The groundwater levels are frequently monitored throughout the Vaalputs site, especially around the security area, and contribute to a uniquely detailed, 30-year database of fluctuating water levels. It is therefore with a high level of confidence that the water table beneath Zones

A and B is assumed to be at a depth of 50 m, which is 35 m below the basement/cover unconformity (see Figure 18–10). This configuration allows up to ~15 m of stacked sources below the required 30 m intrusion limit, and even provides a 5 m safety margin against the underlying water table. The sulphate and chloride content in the groundwater measured in boreholes close to Sites A and B shows values of about 1000 mg/l and 300 mg/l for chloride and sulphate respectively. However, in the unsaturated zone, these concentrations are likely to be much lower, and close to the X-Ray Fluorescence detection limits for granite-hosted groundwater.

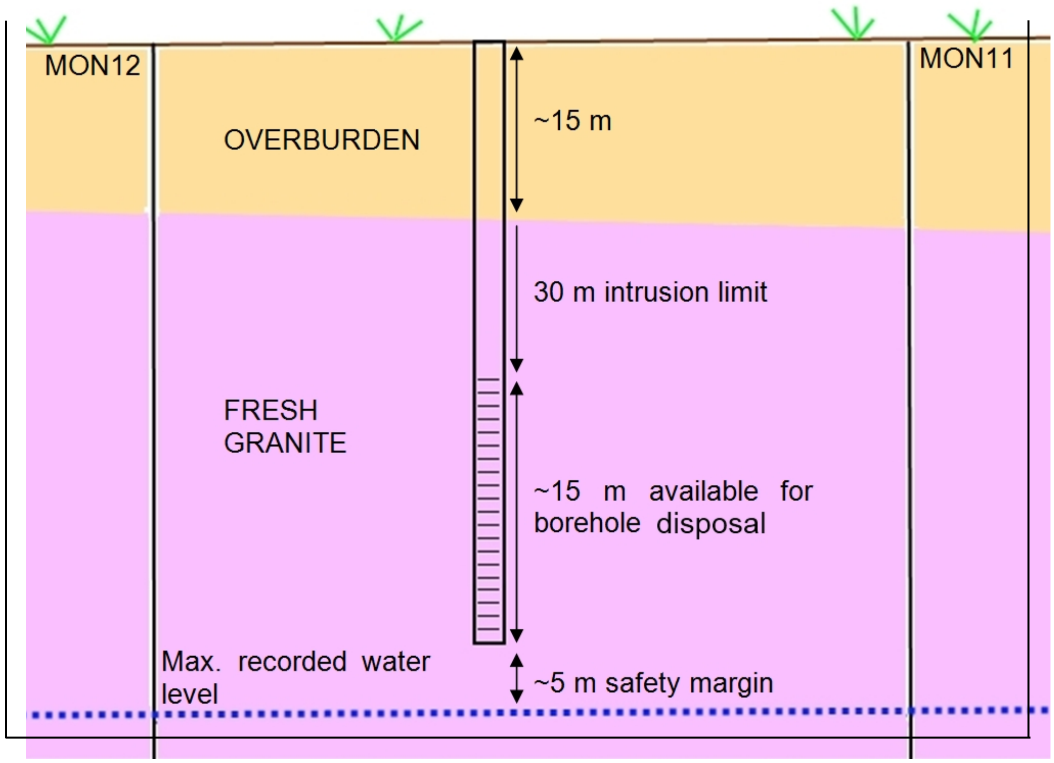

Figure 18–10. Illustration of the BOSS concept based on the available geological and hydrological data. MON 11/12 represents nearby groundwater monitoring boreholes.

18.4.5 Material Sciences: Ceramics as Waste Immobilizers

Research at Necsa is not only concerned with siting new and ongoing assessment of existing nuclear waste disposal facilities, but also with the development of innovative techniques of radioactive waste isolation. This program is the responsibility of the Nuclear Waste Research (NWR) Group, a team within Necsa that carries out research and development on post reactor materials; more specifically, to supply consulting service to different clients (locally and internationally) regarding the chemical treatment and disposal of post reactor waste. This includes in-house research projects as well as international contract research. Contract research completed in 2015 for Argonne National Laboratories (ANL) presented scalable, innovative solutions for the recovery of high-and low-enriched uranium from post-reactor material using proliferation-friendly alkaline technology. Aim of this project is to enable the reuse of uranium for medical target plates or nuclear fuel, and of target plates for ^{99}Mo manufacturing (Carsten et al. 2015). In a second contract for the ANL, and in partnership with Australia's Ansto, the Group is investigating how to encapsulate and dispose different radioactive waste streams produced while purifying uranium from used nuclear fuel using alkaline technology

18.5 Acknowledgments

We thank our colleagues Mbuthokazi Mandaba and Willie CMH Meyer for valuable data provided on the Vaalputs groundwater chemistry and on Materials Sciences research respectively and the management of Necsa for permission to publish this paper. Frank Eckardt (University of Cape Town) kindly provided Figure 18–6.

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18.7 Acronyms

ANL—Argonne National Laboratories

BDF—Borehole Disposal Facility

BOSS—Borehole Disposal of Spent Sources

DTM—Digital Terrain Model

GFZ—GeoForschungsZentrum, Zentrum-Potsdam

KNPS—Koeberg Nuclear Power Station

LLW—Low Level Waste

Necsa—South Africa Nuclear Energy Corporation

NRWDI—National Radioactive Waste Disposal Institute

NTP—Nuclear Technology Products

NWR—Nuclear Waste Research

UNF—Used Nuclear Fuel

USF—Used/Spent Nuclear Fuel

Chapter 19

Geological Disposal of Spent-Fuel and High-Level Waste in Spain

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ABSTRACT: The spent fuel managed in Spain derives from the eight operating nuclear reactors located in the country. There are also other nuclear facilities in operation: the Juzbado fuel manufacturing facility in Salamanca and the El Cabril Low Level Radioactive Waste disposal facility in the province of Córdoba. Spain has the necessary infrastructure in place to manage spent fuel and radioactive waste, from institutional, administrative, technical, and economic and financial standpoints. It has also established appropriate measures to ensure the public's rights of access to information and participation. The Sixth General Radioactive Waste Plan (GRWP) approved by the Government in June 2006 planned a substantial modification to the management strategy to develop a Centralized Temporary Storage (CTS) facility as its top and most urgent priority. The permanent geological disposal of spent nuclear fuel (SNF) occupies since then a secondary place. Among other things, this new plan has implied the deceleration of all activities relating to deep geological disposal. ENRESA (Empresa Nacional de Residuos Radiactivos, S.A), the public company that manages all radioactive waste produced in Spain, has essentially concentrated efforts in R&D activities, adapted to the current Spanish SNF/HLW management strategy. These activities have allowed technical knowledge to be updated and national working teams to be trained in the development of a permanent disposal option, participating in international research projects and in demonstration projects in overseas underground laboratories.

19.1 Introduction

The Empresa Nacional de Residuos Radiactivos, S.A. (ENRESA), a public company founded in 1984, manages all radioactive waste produced in Spain. The Spanish Government through the General Radioactive Waste Plan (GRWP) approves all ENRESA activities. The Sixth GRWP was approved by the Government in June 2006 and remains in effect. This GRWP is planning a substantial modification to the management strategy to develop a Centralized Temporary Storage (CTS) facility as its top and most urgent priority. Specifically, in light of the analyses performed from the technical, strategic, and economic points of view, the proposed solution is based on the availability of a vault-type CTS facility by around 2018 the operating period of which would be some 60 years. From the point of view of economic calculation and planning, it has been assumed that a permanent disposal facility could be put into operation around the year 2050, which would house spent nuclear fuel (SNF), other high-level waste (HLW) and such intermediate-activity wastes that cannot be sent to the already existing El Cabril low-level waste (LLW) facility.

Consequently, the permanent geological disposal of spent nuclear fuel (SNF) will occupy a secondary place since, with a CTS facility in operation, there will be less urgency for a permanent solution. Among other things, this new plan has implied the deceleration of all activities relating to deep geological disposal. The activities contemplated in previous plans have been significantly reduced, these being limited fundamentally to consolidation and updating of the knowledge acquired, taking advantage of international developments in the field.

In this respect, the activities for the forthcoming years will be as follows:

- Documents will be drawn up summarizing the information acquired to date, with site selection activities being discontinued.
- Generic repository designs will be developed for each potential host rock type
- The corresponding safety assessment exercises will be revised and updated on the basis of progress made in R&D programs, in keeping with the revised designs and international projects.

During the period 2006–2014, ENRESA has essentially concentrated efforts in R&D activities, adapted to the current Spanish SNF/HLW management strategy. These activities have allowed technical knowledge to be updated and national working teams to be trained in the development of the permanent disposal option, participating in international research projects and in demonstration projects in overseas underground laboratories. The present discussion is based upon a series of publications listed at the end of this article.

19.2 R&D Activities Performed In 2006-2014

Since 2006, two R&D National Programs conducted in 2004-2008 and 2009-2013 have framed the activities in the area of deep geological disposal during. These programs have been closely related to the 6th and 7th EU Framework Programs. These activities are briefly described below.

19.2.1 Repository technology components

The R&D activities related to waste forms, waste containers, engineered clay barriers, and plugs and seals are described in the following sections.

19.2.1.1 Physico-chemistry of actinides and spent nuclear fuel technology

ENRESA has continued research in this field, in cooperation with Spanish research groups (Centro de Investigaciones Energéticas y Medioambientales (CIEMAT) and Universidad Politécnica Catalunya (UPC), and with the Karlsruhe Joint Research Centre (ITU), to address activities relating to the physico-chemistry of actinides and the behavior of fuel. The main activities have been:

- International collaboration in the development of thermodynamics databases: Participation in the OECD/NEA projects Thermodynamic Data Bases and NEA-SORB.
- Understanding and development of models for radionuclide retention and release mechanisms.
- Analysis of interaction between Th (IV) and Pu (IV) and iron oxy-hydroxides (magnetite and goethite) as cask corrosion products, and bentonite, including the development of surface complexation models. The results have identified different sorption mechanisms in granite, depending on the presence or absence of bentonite.

19.2.1.2 *Waste containers*

These activities constitute the systematic course of action that ENRESA has been performing, mainly in collaboration with INASMET (Investigación en el Area de los Materiales) and to a lesser extent with CIEMAT and CENIM (National Centre for Metallurgical Research—Scientific Research Council), since the early 1990s.

During the reporting period, activities in this field have focused on the study of corrosion of metals and alloys under the conditions of salinity existing in granite and (especially) clays. The studies for granite concluded with satisfactory results for carbon steel, stainless steel, and alloys of titanium Gr7. Electron-beam welding provides the best resistance to corrosion. At present, passivation studies are being undertaken, and the optimization of manufacturing processes is being analyzed. These activities are the basis for the performance of the future CTS facility.

19.2.1.3 *Engineered clay barriers*

The activities in this area have been mainly addressed with the Full-scale Engineered Barriers EXperiment (FEBEX) project performed at the Grimsel Test Site (Switzerland) and the Engineered Barrier Emplacement Experiment (EB) project performed at the Mont Terri Rock Laboratory (Switzerland).

Thanks to the performance of these projects, a multidisciplinary team of researchers and engineers (including UPC, AITEMIN, CIEMAT, UDC, UPM-CSIC, and DMIberia) has been set up in Spain, ensuring the availability of capacities in the fields of design, characterization, monitoring, and modeling of the THMC (thermo-hydro-mechanical and chemical) behavior of compacted clay barriers for both granite and clay host rocks.

After the completion of the initial phase of FEBEX dismantling in the year 2002, a concrete plug was installed, and new instrumentation was emplaced via boreholes for the monitoring of THM properties, along with a system specifically set up to collect samples of bentonite saturation water. As a result, the experiment has been re-initiated and has become a part of the NF-PRO project, co-financed by the EU, with different goals and scope from those in the original FEBEX project. The experiment and its corresponding mock-up are currently in a latent monitoring phase and continue to generate data available to the modeling teams. A final phase of dismantling of the *in situ* test is scheduled for 2015.

The Engineered Barrier Emplacement Experiment (EB) was dismantled in 2013 (as part of the PEBS project, co-financed by the EU in the 7th FP), after almost eleven years of operation, and has been a long, well monitored, and full-scale demonstration of the use of a Granular Bentonite Material (GBM) as a clay barrier. The experiment was carried out, with a dummy canister resting of bentonite blocks, in a 6 m long gallery section (EB niche) excavated in the Opalinus clay of the Mont Terri Rock Laboratory (Figure 19–1).

The main objective of the controlled dismantling of this experiment has been to evaluate the actual state and properties (especially the hydraulic conductivity) of the emplaced bentonite barrier after its complete isothermal saturation (using an artificial hydration system). Therefore, the dismantling operations have been carefully coordinated with an extensive sampling program: more than 500 samples have been taken for on-site and laboratory analyses; mostly of the bentonite materials (GBM and blocks) of the barrier, but also includes the concrete plug, concrete-bentonite and rock-bentonite interfaces, rock massif, water, monitoring sensors, and elements of the hydration system.

The dismantling work of the EB experiment (visual observation, monitoring, extensive post-mortem analyses, and geophysical investigations) have clearly confirmed the following significant information about a bentonite barrier (emplaced using bentonite blocks and GBM) hydrated under isothermal conditions:

- The barrier was practically fully saturated.
- The hydraulic conductivity of the saturated GBM is low enough (less than 5×10^{-12} m/s), even if emplaced with a relatively low average dry density (1.36 g/cm^3 in this experiment). Therefore it has been shown that this key safety indicator falls within the acceptable limits considered in the Performance Assessment of the repository concepts.
- Homogenization between the two types of bentonite emplaced (blocks and GBM) has taken place. Nevertheless, throughout the bentonite mass, still (after the experiment life of more than ten years) some heterogeneities persist: the moisture content tends to increase (and the dry density to decrease) towards the bottom of the experiment niche (Figure 19–2). This is probably due to the fact that the GBM emplacement was difficult in this case due to the existing hydration tubes.
- The measured values of the thermal conductivity of the saturated bentonite (from 0.90 to 1.35 W/m·K) are high enough.
- Self-sealing of the EDZ in the Opalinus clay had been observed during the experiment, due to the swelling pressure developed in the barrier. As expected, after the dismantling the seismic data do suggest the gradual regeneration of the EDZ.
- The dismantling has provided the opportunity to perform microbial analyses of the bentonite emplaced more than ten years before. Samples analyzed had water activities higher than 0.96 (note: water activity is defined as the ratio of the partial pressure of water vapor in a soil to the equilibrium vapor pressure of water at a given temperature); relatively high culturability levels for heterotrophic aerobes; and low culturable levels of sulfate reducing bacteria.
- In general, the obtained gas permeability values of the saturated bentonite are low and homogeneous (from 1 to 6×10^{-22} m^2).

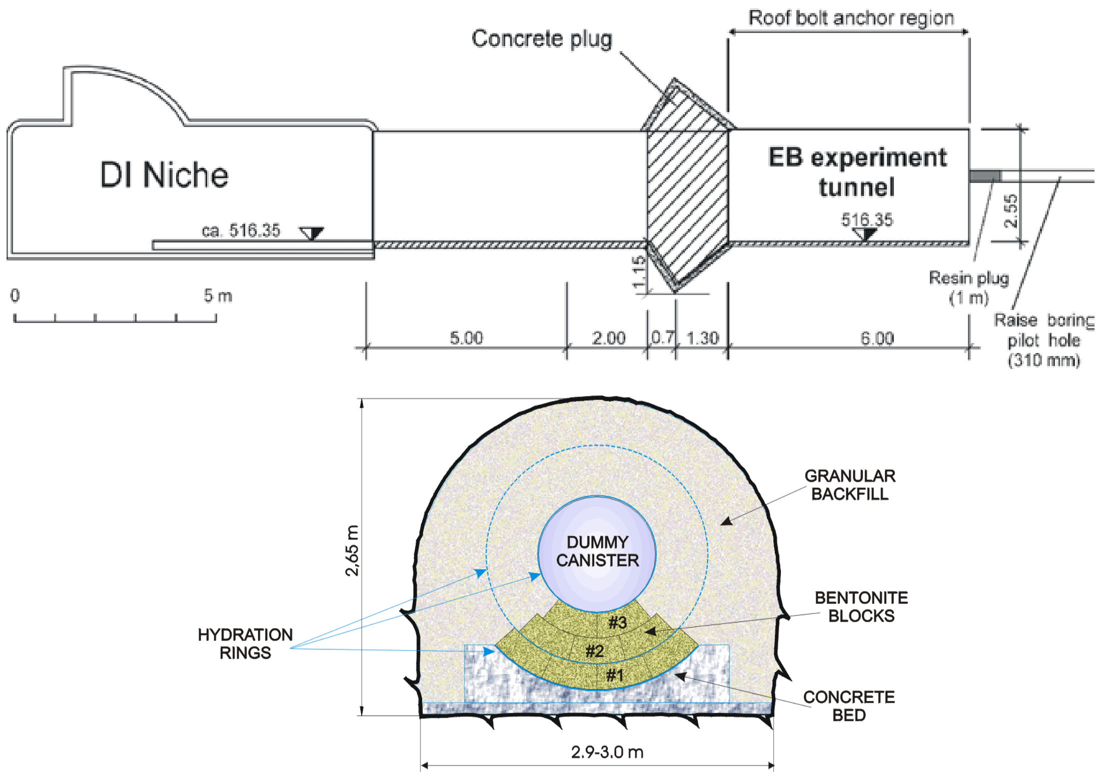

Figure 19–1. EB niche at Mont Terri URL, longitudinal and cross sections

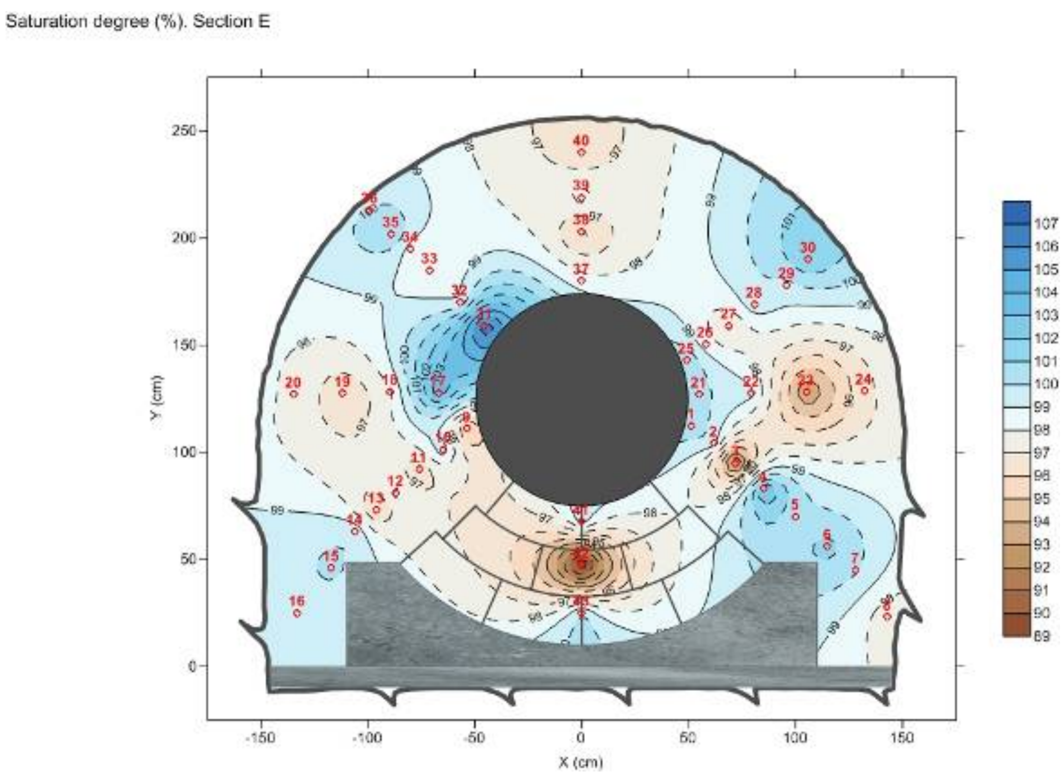

Figure 19–2. Isolines of the degree of saturation (Sampling section E)

The controlled dismantling of the EB experiment has allowed us to complement and improve upon the previously gained knowledge (through the available monitoring data) of the isothermal saturation process of a full-scale bentonite barrier. It has been fully confirmed that the use of a GBM is a good option to construct bentonite barriers.

Modeling results using the CODE_BRIGHT code concerning the hydration process and the final state of the EB bentonite barrier are in reasonable agreement with the actual findings after the dismantling. Moreover, the real data obtained in this operation is a sound basis for a better formulation of the numerical models; reducing their uncertainties and providing clearer criteria to be conservatively applied in the Performance Assessment of engineered barriers.

19.2.1.4 Plugs and seals

A full-scale low-pH shotcrete plug was constructed in February 2007 at the Grimsel Test Site, as part of the ESDRED project, co-financed by the EC. The aim of the test was to demonstrate the supporting capacity of the plug under realistic conditions, that is, with the swelling pressure of a bentonite buffer installed on one side of the plug.

The basic layout of the full-scale demonstration test consists of a 4 m long parallel low-pH shotcrete plug constructed at the back end of a 3.5 meter diameter horizontal gallery, excavated in granite with a tunnel boring machine and sealed with 1 m of highly compacted bentonite blocks. The bentonite is provided with an artificial hydration system to accelerate the saturation process and, if required, to impose a pore water pressure in the buffer. In addition several sensors and a data acquisition unit, display and control system were also installed to follow the evolution of the test (Figure 19–3).

From the results and interpretation of the long plug test, the main conclusions that can be drawn are the following.

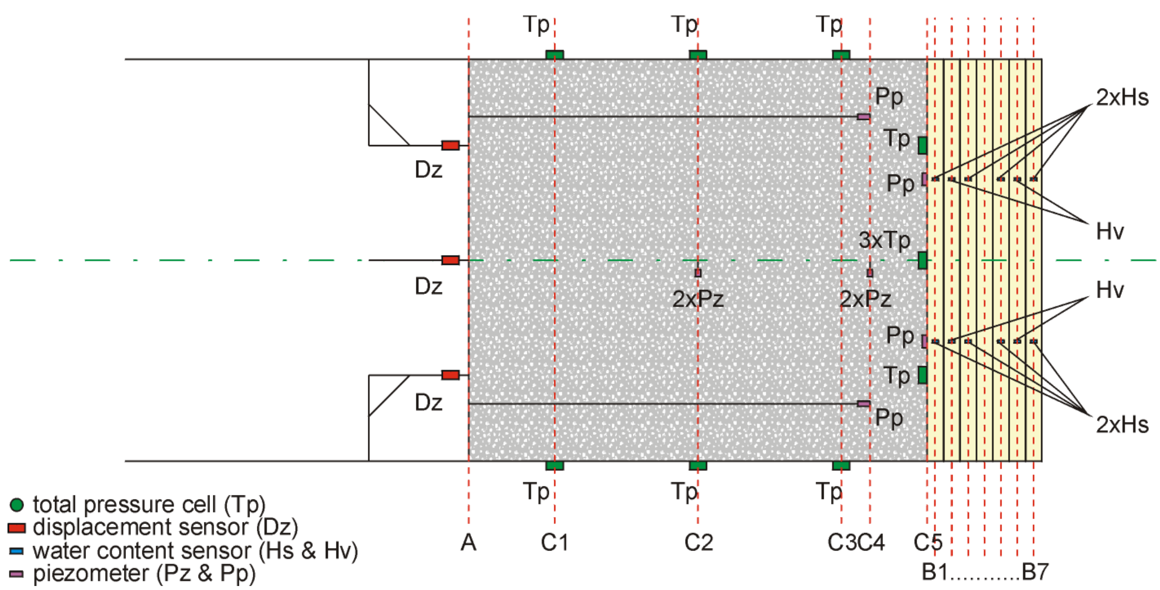

Figure 19–3. Basic layout with instrumentation

- The concrete plug construction methodology applied has proved to be efficient even in the most restrictive conditions, as those of the Grimsel test site during winter. It made possible the construction of a large diameter shotcrete plug of 38.5 m^3 in four days, with a construction rate of 1 meter per day and in a semi-automated mode, using a shotcrete robot. This is important in application cases where time is a key factor and where operators must stay clear of the working face.

- The function of the plug, which is to mechanically support the bentonite buffer, has not been completely proven yet, at least not up to the end of 2014, when the bentonite swelling pressure was still well below the maximum expected swelling pressure of 4.15 MPa, although it continues increasing at a good rate. The test will continue in a latent monitoring phase, so eventually it will be possible to fully check the supporting capacity of the plug.
- Although water tightness is not a requirement for this plug, it should be mentioned that water leaks appeared in the bottom of the plug, as they did during the failure test of the short plug in Äspö Laboratory. These leaks could be due to potential heterogeneities in the bottom part of the plug caused by shotcrete rebounds. If future plugs were required to be watertight, improvements should be introduced both in the design and in the construction procedure to avoid this. In any case, the leaks in this plug were eventually sealed by the swelling bentonite, so the system has behaved as it was intended to.

In addition, in the framework of the EC co-financed project MoDeRn (Monitoring Developments for safe Repository operation and staged closure), ENRESA in collaboration with AITEMIN (Asociación para la Investigación y Desarrollo Industrial de los Recursos Naturales) demonstrated the potential of using wireless sensors and transmission networks in underground confined environments, part of which would be immersed in solid material (buffer, concrete, rock, etc), and to manage the corresponding energy needs for inaccessible immersed network sensors (nodes). Development of wireless data transmission methods would allow for data measured by sensors emplaced within the EBS to be relayed to receiving stations, and would represent a method for monitoring of the EBS without affecting the passive safety of key design components (e.g. buffer, plugs and seals) by short-circuiting their containment function through the introduction of wires.

A High Frequency Wireless (HFW) system consisting of five sensing units, each one containing the needed instrumentation, a battery, and the HF emitter (electronics and aerial), was installed in the different plug experiment elements (bentonite, concrete, and granitic rock) in Grimsel. For the installation of the sensing units, five boreholes with an inner diameter of 86 mm in order to house the HFW nodes were drilled in the plug and the rock (November 2011). The geotechnical sensors of the wireless monitoring system were pore pressure transducers, total pressure cells, and water content probes.

The transmission distance for HF signals at four frequencies was tested in the laboratory and in the field. The frequencies tested were 2.4 GHz, 868 MHz, 433 MHz, and 169 MHz. In laboratory tests, signals at 868 MHz and 433 MHz were capable of passing through 50 cm of bentonite, 25 cm of salty water and 40 cm of argillite rock; transmission distances at 2.4 GHz were lower. Field tests demonstrated that transmission distances at 169 MHz were greater, about 3.5 m in clay-based rocks and greater than 5 m in saturated bentonite, and this frequency was adopted for the demonstration tests in the MoDeRn Project. The wireless nodes developed used a $Li\text{-}SOCl_2$ battery combined with some high performance capacitors that support high current demands and have a low leakage current (of the order of a few nA). The final design for the nodes used is for a small self-contained device with an expected lifetime up to 20 or 25 years. The node is illustrated in Figure 19–4.

Comparison of the data transmitted using wireless nodes with that from conventional hard-wired monitoring systems previously installed at the test site confirmed the feasibility of using wireless nodes.

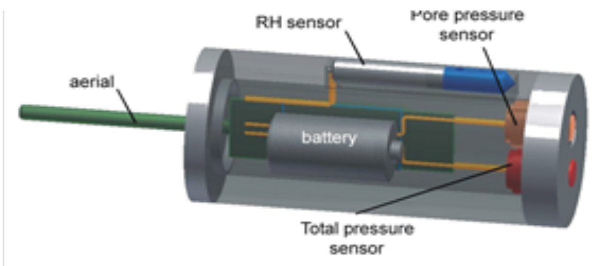

Figure 19–4. Design of the node (left) and installation in the Grimsel Test Site (right)

19.2.2 Geological barrier

Activities related to the geological barrier have focused on progressing in process understanding and with techniques for characterizing the performance of compacted clay media and, to a lesser extent, of granitic media.

Clay Media

The projects carried out at Mont Terri Rock Laboratory have focused on the clay rock performance under hydraulic and thermal loads.

The desaturation effects due to normal tunnel ventilation were studied by the Ventilation Experiment (VE), co-financed by the EC through the NF-PRO integrated project. This is a demonstration test performed at relatively large-scale and of long duration, and has allowed us to evaluate “in situ” and better understand the desaturation process of a drift excavated in a hard clay (Opalinus Clay), when ventilated during several months with dry air. The experiment included generating a flow of dry air (isothermal conditions, T ≈ 15-16 ºC) along a 10 m long section of an unlined microtunnel (diameter = 1.3 m). It was excavated (by the raise-boring technique) in 1999, and oriented perpendicular to the bedding strike direction of the shaly facies of the Opalinus Clay.

The controlled ventilation during two desaturation phases has been accomplished using a blowing device, located outside the test section, and inflow and outflow pipes, monitored with flowmeters and hygrometers. The blowing device generates a flow of air (Qin) with specified values of the relative humidity (RHin) and temperature (Tin); which is sent to the test section through the inflow pipe, flows along the test section, and then is evacuated with the outflow pipe. Values of Qout, RHout and Tout are also measured in the outflow pipe. Inside the test section, the air relative humidity and its temperature are recorded with four hygrometers; the air pressure is recorded with two sensors; and the evaporation of free liquid water is measured in two water pans. Water balances (rock outflow and inflow of vapor) are calculated using the data provided by the hygrometers and flowmeters installed in the inflow and outflow pipes.

The VE test section was monitored in a rock thickness of approximately two meters around the test section with 24 mini-piezometers, 32 hygrometers, 8 mini-extensometers, 10 time-domain- reflectometry probes and 5 electrode chains (geoelectric survey), which were installed in the eleven cross-sections indicated in Figure 19–5. Also, an evapometer was installed on the rock surface of the test section.

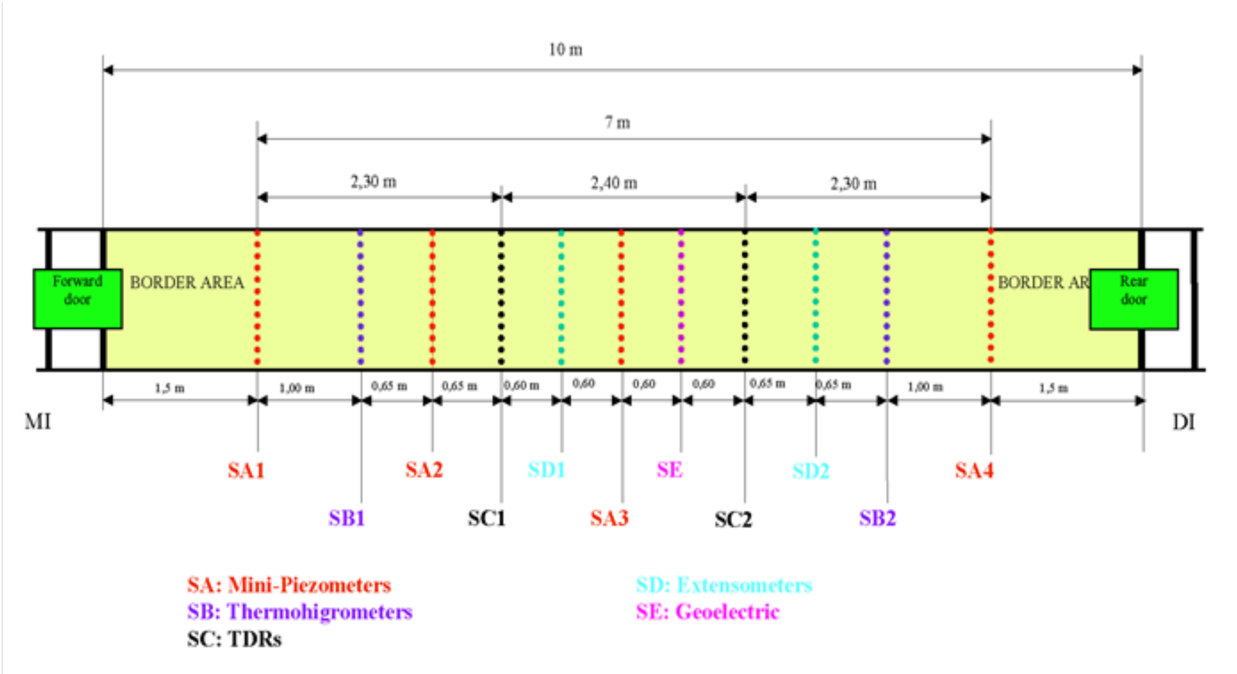

Figure 19–5. Location of cross-sections with different types of instrumentation.

The most important and general finding obtained from the actual data and the modeling of the VE experiment is the following:

The desaturation of clayey rocks of low hydraulic conductivity ($K < 10^{-12}$ m/s) due to ventilation is very small. Under real repository conditions, relative humidity of the air much higher than in the desaturation period of the VE test, the thermal and hydraulic rock characteristics will not be practically affected by the ventilation.

Other complementary and site-specific conclusions are the following:

- The estimated hydraulic conductivity of the rock at the VE test site is equal to 2.5×10^{-13} m/s.
- The mechanical effects of the ventilation with dry air can be considered as almost negligible.
- The installed monitoring sensors have in general worked properly, demonstrating their adequate selection and installation procedures in a shaly rock.

The performance of the clay rock under thermal loading was studied with the Heater Experiment HE-E conducted at the Mont Terri Rock Laboratory, in the framework of the EC-co-financed project PEBS (Long Term Performance of Engineered Barrier Systems). The HE-E is a 1:2 scale heating experiment considering natural resaturation of the EBS at a maximum heater surface temperature of 140ºC. The experiment is located in the VE test section. The heating started in June 2011, and the maximum temperature was reached in June 2012. Since then, the temperature has been held constant.

The aims of the HE-E experiment are elucidating the early non-isothermal resaturation period and its impact on the thermo-hydro-mechanical behavior, namely: (a) to provide the experimental database required for the calibration and validation of existing THM models of the early resaturation phase; and (b) to upscale thermal conductivity of the partially saturated EBS from laboratory to field scale (pure bentonite and bentonite-sand mixtures).

The experiment consists of two independently heated sections, each of 4 m long, with the heaters are placed in a steel liner supported by MX80 bentonite blocks (dry density 1.8 g/cm^3, water content 11%). The two sections are identical apart from the granular filling material. Section 1 is filled with a 65/35 granular sand– bentonite mixture, and section 2 is filled with pure MX80 bentonite pellets (Figure 19–6). This allows comparison of the thermo-hydraulic behavior of the two EBS materials under almost identical conditions. Both materials have been characterized in the laboratory with respect to their thermo-hydraulic properties.

A total of 18 humidity/temperature sensors are emplaced at the Opalinus Clay/EBS interface and another 60 in the EBS materials (24 sensors in the blocks and 36 in the granular materials). Instrumentation rings were designed with insulating material to avoid thermal bridges. The high sensor density ensures an accurate spatial characterization of the thermal behavior and how it is affected by the saturation in the Opalinus Clay and vapor formation owing to the heating. Rock instrumentation was essentially the same as used in the preceding VE experiment.

The observed temperature increases in EBS and rock are in line with those predicted by the design calculations (slight variations are attributed to differences in model setup and conceptualization). The EBS is characterized by a very strong temperature gradient owing to its low thermal conductivity in the current dry state. Drying in the inner part of the EBS continues while a complex development of the humidity profiles takes place. The latter is strongly determined by the different water contents and densities of the materials at installation, high sensitivity to changing two-phase flow parameters, and the impact of vapor diffusion in a changing porous matrix.

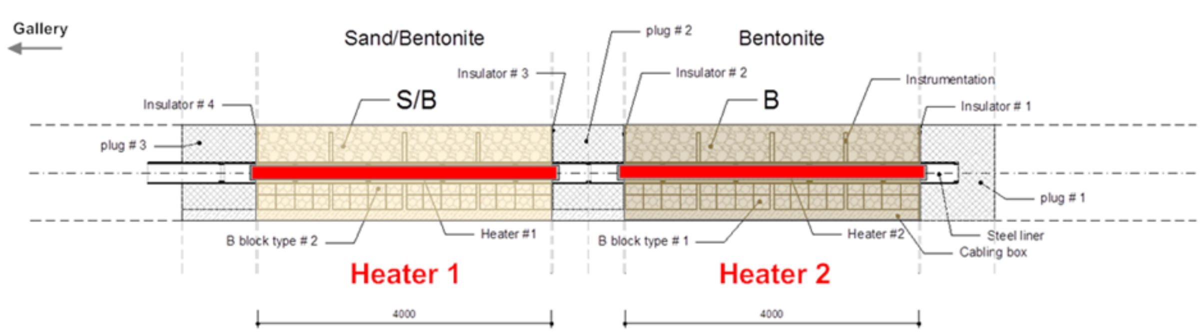

Figure 19–6. Schematic layout of the HE-E experiment

The Opalinus Clay is at relatively low temperature while still partially unsaturated close to the microtunnel. The point at which hydraulic pressures are registered, influenced by the anisotropy of the Opalinus Clay, is converging slowly toward the tunnel wall. After 15 months, it is still located more than 1.5 m from the microtunnel wall. As predicted by the models, a hydraulic pressure increase, associated with the differential thermal expansion of the Opalinus Clay and the porewater, is observed in the saturated Opalinus Clay at a larger distance from the tunnel. The porewater pressure increase, which started developing shortly after switching on the heaters, is still continuing and the maximum value has not yet been reached.

Granitic Media

Aspects of hydrogeological and geochemical characterization and modeling have been developed in association with the FEBEX project in the Grimsel Test Site. The objective was to analyze the evolution and behavior of the granite surrounding the experiment during the heating and saturation of clay barrier. The different models have been notably improved.

Basic Deep Geological Disposal (DGD)

With a view to providing support for future decision making, ENRESA has included the available information on geological disposal in granites and clays in the corresponding strategic documents. These documents (which are not published) consider the available information in reference to the following:

- Reference concept: characteristics of components, basic design, and cost evaluation.
- DGD components: research and development activities relating to waste, engineered barrier, geological barrier, and biosphere
- Safety assessment
- Conclusions and recommendations: design and R&D.

These documents will be updated based on progress made in relation to both R&D and safety assessment or design. They are living documents, the objectives of which are to facilitate decision making regarding the definitive management of fuel. They also constitute a good tool for forecasting in relation to activities linked to deep geological disposal.

19.3 Conclusions

The Sixth General Radioactive Waste Plan, in effect since 2006, identifies the development of a centralized temporary storage facility as top priority, with eventual deep geological disposal to begin about 2050. Activities relating to deep geological disposal in Spain have therefore been aimed at developing the technologies required for its implementation, improving key areas of knowledge required for safety assessment, and integrating this knowledge within basic reference documents supporting future decision making. All the activities have been performed within a framework of close and efficient international collaboration. The current result of the above is the availability of equipment and capacities that will allow for implementation of solutions when they are considered appropriate. The transition from generic activities to specific activities is an issue that has yet to be decided and is not a short-term priority in the new ENRESA strategy.

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Note: All the ENRESA technical publications can be accessed at www.enresa.es.

19.5 Acronyms

AITEMIN—Asociación para la Investigación y Desarrollo Industrial de los Recursos Naturales

CENIM—National Centre for Metallurgical Research—Scientific Research Council

CIEMAT—Centro de Investigaciones Energéticas y Medioambientales

CTS—Centralized Temporary Storage

DGD—Basic Deep Geological Disposal

EB—Engineered Barrier Emplacement Experiment

FEBEX—Full-scale Engineered Barriers EXperiment

GBM—Granular Bentonite Material

GRWP—General Radioactive Waste Plan

HFW—High Frequency Wireless

INASMET—Investigación en el Area de los Materiales

ITU—Karlsruhe Joint Research Centre

MoDeRn—Monitoring Developments for safe Repository operation and staged closure

PEBS—Long Term Performance of Engineered Barrier Systems

SNF—Spent Nuclear Fuel

THMC—Thermo-Hydro-Mechanical and Chemical

UPC—Universidad Politécnica Catalunya

VE—Ventilation Experiment

Chapter 20

Licensing of the KBS-3 Concept for Spent Nuclear Fuel in Sweden

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ABSTRACT: In Sweden SKB (Swedish Nuclear Fuel and Waste Management Company) has been given the responsibility for management and disposal of all radioactive waste from the nuclear power plants. The waste is going to be disposed of in three different geological final repositories:

- The Final Repository for Spent Fuel
- The Final Repository for Short-lived Waste (SFR; this acronym comes from the Swedish term "Slutförvaret för kortlivat radioaktivt avfall")
- The Final Repository for Long-lived Waste (SFL; this acronym comes from the Swedish term "Slutförvaret för longlivat låg och medelaktivt avfall")

The principal alternative for a final geological repository for spent nuclear fuel is called the KBS-3 method. It involves encapsulating the fuel in copper canisters with cast iron inserts and embedding each canister surrounded by bentonite clay at a depth of 500 meters into the bedrock. An application to build such a final repository in Forsmark in the municipality of Östhammar was sent in 2011, and is currently being reviewed by The Swedish Radiation Safety Authority the authorities.

Since 1988, short-lived waste (generally this is low and intermediate level activity waste) has been deposited in SFR in Forsmark. The repository consists of a silo and four rock vaults. SKB now plans to extend the capacity, and in December 2014 an application to build another six vaults was submitted to the authorities.

Among SKB's repositories, SFL is planned to be the last one to be put into operation. However, several milestones must be passed. A concept study was completed in 2014, and a safety evaluation is underway.

20.1 Background

Sweden has been generating electricity using nuclear power for more than 40 years. The first reactor was put into operation in 1972 and the latest in 1985. A referendum in 1980 limited the nuclear program to twelve reactors. Two of these, Barsebäck 1 and Barsebäck 2, were closed down in 2005 and in 1999 respectively, for political reasons. The ten remaining reactors are operated by three utilities and supplied about 41 per cent of the nation's electricity in 2014, amounting to 62.2 TWh. Seven of these reactors are boiling water reactors (BWRs) and three are pressurized-water reactors (PWRs).

Back in the 1970s, legislation charged the nuclear power industry with the responsibility for managing and disposing of all the radioactive waste from its installations in a safe manner. In response, the owners of the nuclear power plants formed SKB (Swedish Nuclear Fuel and Waste Management Company) for this purpose. SKB's assignment is to plan, construct, own, and operate the systems and facilities necessary for transportation, interim storage, and final disposal. A fund to finance the activities was set up a few years later.

The plans for disposal of the radioactive waste and the spent nuclear fuel are to a great extent determined by the properties of the waste. According to its level of radioactivity (short- or long-lived waste), the waste is divided into categories. A final repository for short-lived waste is already in operation, and final repositories for spent fuel and long-lived waste are respectively under way. Which barriers or barrier functions are needed in a final repository is largely dependent on the content of radioactive substances and their half-lives. Accordingly, the requirements on the barriers in a final repository for short-lived radioactive waste are less stringent than those on the repositories for spent fuel and for long-lived waste.

The designs of the three repositories are based on the following principles:

- The repositories shall be located in a long-term stable geological environment.
- The repositories shall be situated in bedrock that can be assumed to be of no economic value to future generations.
- The safety of the repositories shall be based on multiple barriers.
- Engineered barriers shall primarily consist of naturally occurring materials that are long-term stable in the repository environment.
- The barriers shall work passively, i.e. without human intervention and without input of energy or materials.

The repositories shall be designed in such a manner that they do not need to be monitored after closure.

20.2 Spent Nuclear Fuel

The spent nuclear fuel is long-lived, and contains by far most of the radioactivity, both short- and long-lived. It comprises a small fraction of the total quantity of waste to be disposed of, but is high-level and requires radiation shielding in conjunction with all handling, storage and final disposal.

Up to December 2014, approximately 5,900 tonnes of spent nuclear fuel have been used in power production and are at present stored in Clab, the central interim storage facility outside Oskarshamn in southeast Sweden. According to the current plans, the total quantity of spent fuel to be disposed of in a final repository comprises about 12,100 tonnes, expressed as the quantity of uranium that was originally present in the fuel.

In addition, the spent nuclear fuel to be deposited also includes fuel from the Ågesta reactor, fuel residues from testing programmes at Studsvik and MOX-fuel. However, these fuel types comprise a very small fraction of the total quantity.

After more than three decades of research, technology development and investigations, in March 2011 SKB applied to the regulatory authorities for licences to build and operate a final repository for spent nuclear fuel in Forsmark, as well as an encapsulation plant adjacent to the central interim storage facility in Clab.

20.2.1 The Final Repository for Spent Fuel

SKB's principal alternative for spent fuel disposal is called the KBS-3 method. The disposal concept involves encapsulating the spent fuel in copper canisters and embedding each canister in bentonite clay in the bedrock, as shown in Figure 20–1. In brief, this method ensures that the spent fuel is protected by three different barriers.

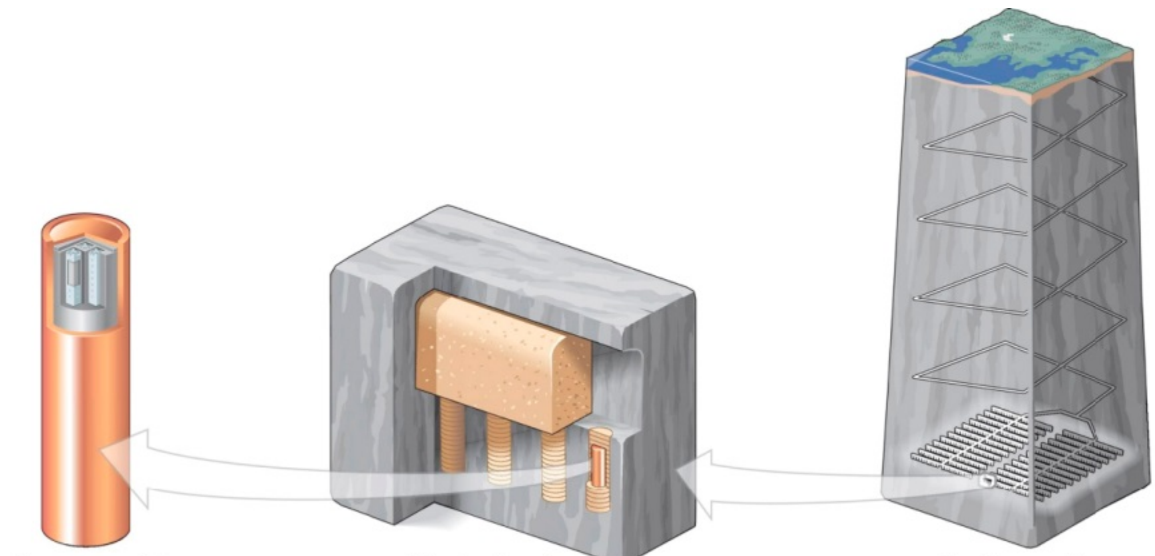

Figure 20–1. The spent nuclear fuel will be encapsulated in copper, embedded in compacted bentonite clay, and emplaced at a depth of approximately 470 meters in the Forsmark bedrock.

The key safety-related features of the KBS-3 disposal system can be summarized in the overall safety functions *containment* and *retardation*. Containment is the primary safety function of the barriers in a KBS3 repository to contain the fuel over one million years. The copper canister serves this function.

Should the canister be damaged, the canister itself, together with the buffer and the host rock, will contribute to retarding any potential release of radionuclides. Hence, retardation of radionuclides is defined as the second safety function of the repository. Many radionuclides have a strong propensity for diffusion into and adsorption in the buffer as well as the host rock.

The Final Repository for Spent Fuel will consist of a surface part and an underground part, as shown in Figure 20–2. The underground part consists of a central area and a number of deposition areas plus connections to the surface part in the form of a ramp for vehicle transport and shafts for elevators and ventilation. About 470 meters below ground level, deposition areas comprising a large number of deposition tunnels are found. Each deposition tunnel contains a number of vertical holes in which the canisters with spent nuclear fuel will be deposited. After the canisters have been placed in the deposition holes, surrounded by tightly packed bentonite, the tunnel is filled with clay and crushed rock. A description of the underground design can be found in (Hansson et al. 2009).

20.2.2 Site Investigations and Site Descriptions

The work of finding a suitable site for a final repository for spent nuclear fuel has spanned several decades. At the end of the site selection process, the choice stood between Forsmark in Östhammar municipality and Laxemar in Oskarshamn municipality. A decisive factor in the selection of Forsmark was that the prospects of achieving long-term safe disposal were judged to be better there (SKB 2011a).

The Forsmark region is a part of the sub-Cambrian peneplain belonging to the Fennoscandian shield, and is relatively flat with a gentle slope to the northeast. The coastal area that has been selected is situated southeast of the Forsmark Nuclear Power Plant. Quaternary deposits are more dominant than exposed bedrock or bedrock with only a thin Quaternary cover. The predominant rock type in the region is a grey-to-red equigranular metagranitoids.

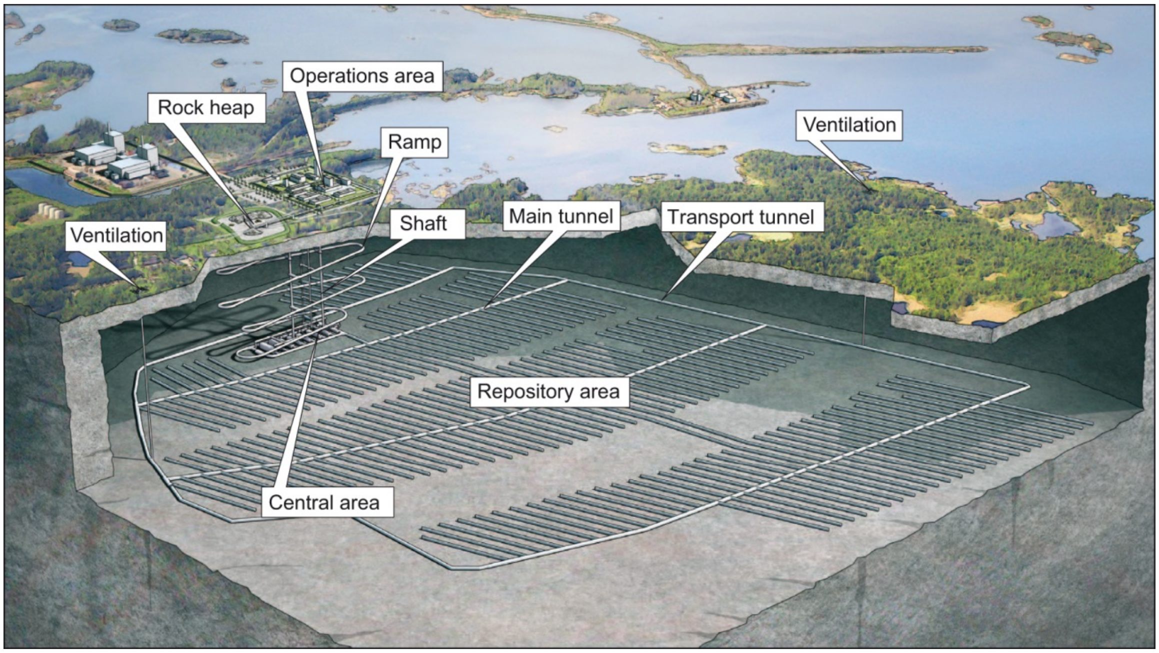

Figure 20–2. Schematic of the Final Repository for Spent Fuel to be built in Forsmark.

The major part of the bedrock was formed 1.9 billion years ago, and it has been affected by both ductile and brittle deformation. The ductile deformation has resulted in large-scale ductile high-strain zones, and the brittle deformation has given rise to large-scale faults and fracture zones. Tectonic lenses, within which the bedrock is much less affected by ductile deformation, are enclosed between the ductile high-strain zones. The Repository for Spent Fuel is going to be built in one of these tectonic lenses. At depth the rock is relatively dry and fracture-poor, but more hydraulically conductive near the surface.

An important component of the characterization work is the development of a site conceptual model that constitutes an integrated description of the site and its regional setting. This model is an input to the safety analysis and covers the current state of the geosphere and the biosphere, as well as those ongoing natural processes that affect the long-term evolution (SKB 2008). A three dimensional geometric fracture model of the rock volume is shown in Figure 20–3. Figure 20–4 shows conceptual models for fracture domains, connected open fractures, and distribution of end-members.

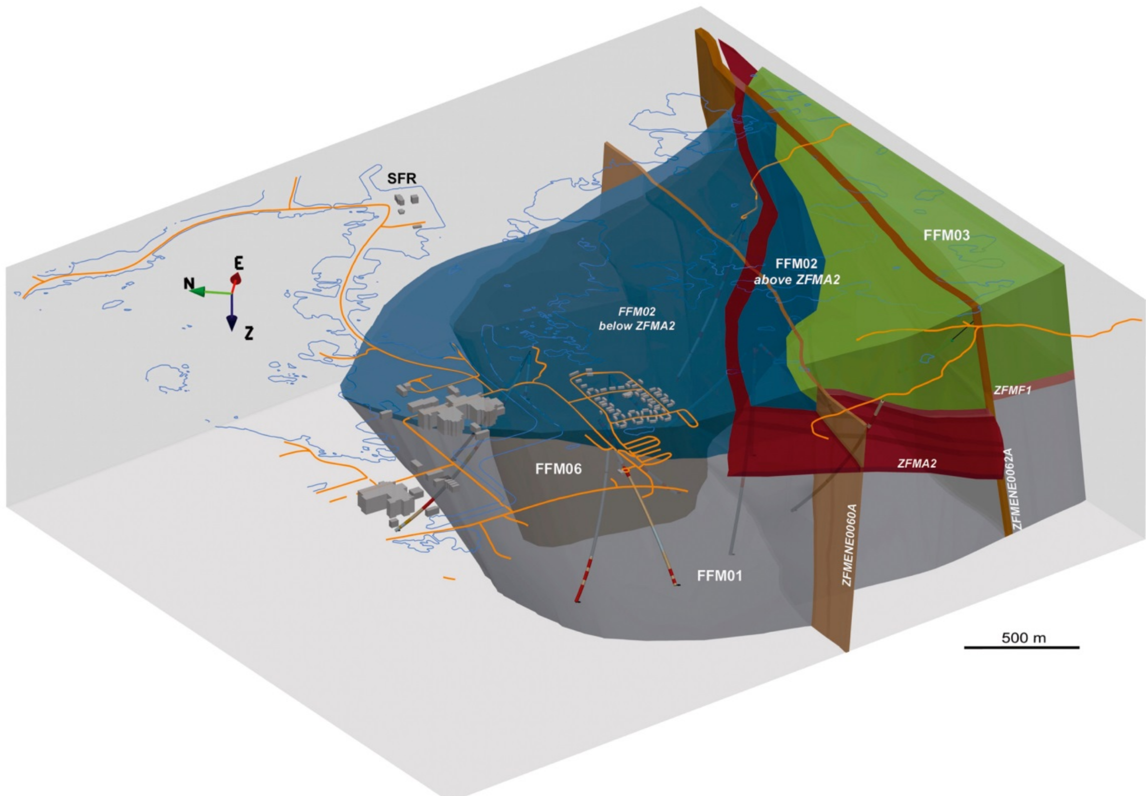

Figure 20–3. Three-dimensional geometric model for fracture domains FFM01, FFM02, FFM03 and FFM06 in the northwestern part of the Forsmark tectonic lens, viewed toward the east-north-east. The local model block for the Forsmark site is shown in pale grey. The gently dipping and sub-horizontal zones ZFMA2 and ZFMF1 as well as the steeply dipping deformation ZFMENE0060A and ZFMENE0062A are also shown.

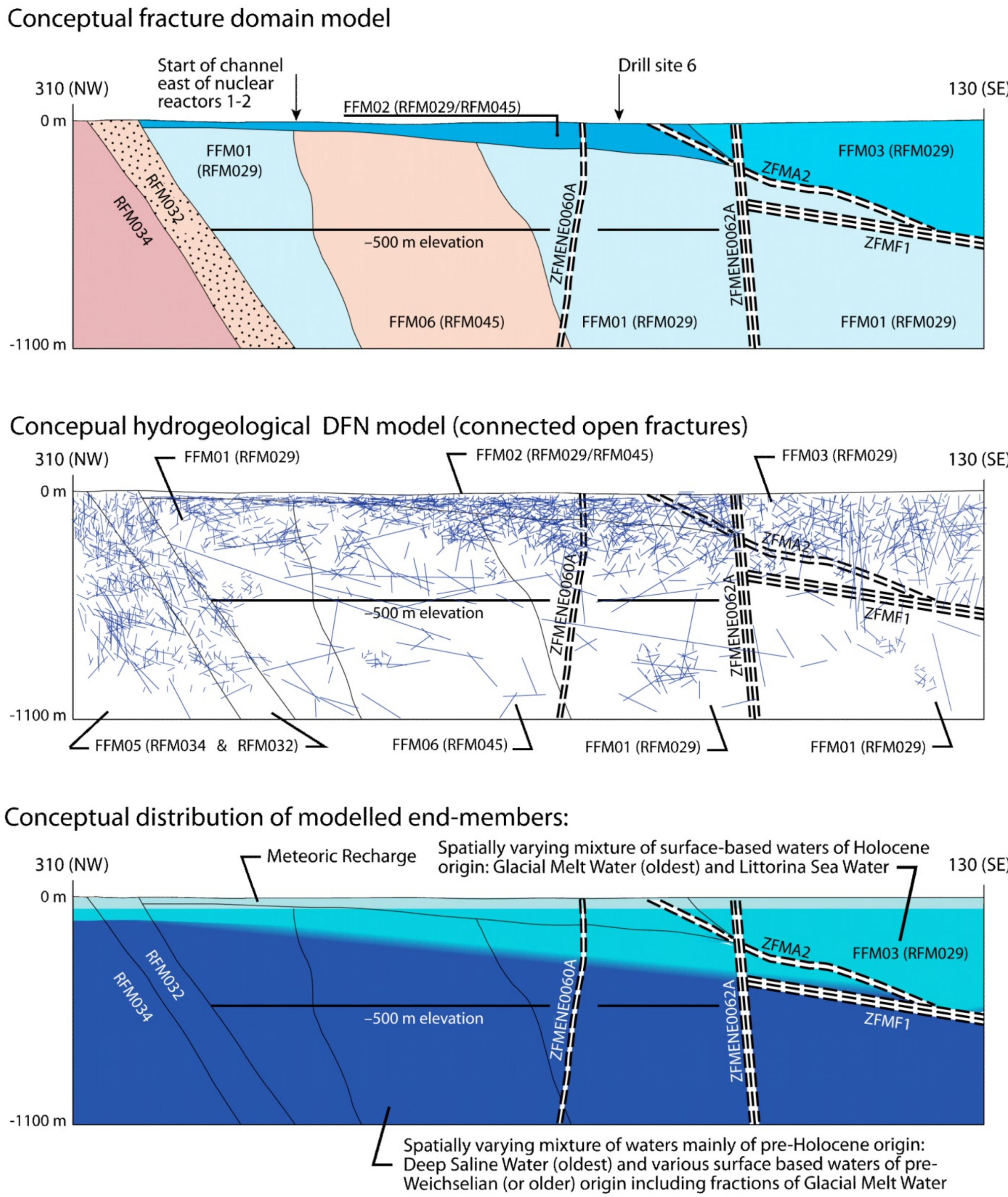

Figure 20–4. Conceptual models of fracture domains, connected open fractures, and distribution of end-members in Forsmark.

20.2.3 Safety Analysis

Over time, the repository and its nearby environment will evolve as they are subjected to processes such as heating, cooling, groundwater flow, and geochemical reactions including corrosion. The long-term performance of the final repository is evaluated in safety assessments, the main purpose of which is to investigate whether the repository can be considered radiologically safe over time. In principle, this is established by comparing the estimated releases of repository-derived radionuclides and associated radiation doses with regulatory criteria (SSM 2008). SKB's most recent safety report for the final repository for spent nuclear fuel was called SR-Site (SKB 2011b), and consisted of 11 main steps; see Figure 20–5. SR-Site supported the licence application by demonstrating that the repository SKB intends to build at the Forsmark site complies with the regulatory risk criterion. The Swedish Radiation Safety Authority's Regulations concerning the Protection of Human Health and the Environment in connection with the Final Management of Spent Nuclear Fuel or Nuclear Waste" (SSMFS 2008:37)

Figure 20–5. An outline of the eleven main steps of the SR-Site safety assessment of the final repository for spent nuclear fuel. The boxes at the top above the dashed line are inputs to the assessment.

The initial state is the starting point for the safety analysis. Future states of the system will depend on its initial state and on internal processes, as well as on external factors acting on the system, which may have an impact on internal processes as well. Comprehensive sets of long term processes relevant to repository safety were considered for the different components of the repository system (SKB 2010a; SKB 2010b; SKB 2010c). External factors included effects of future climate and climate-related processes, such as glaciations and land uplift (SKB 2010d). Also, future human actions may affect the repository (SKB 2010e).

A key point in the analysis of future states is the evaluation of reference external conditions. These external reference conditions postulated a repetition of the last 120,000-year glacial cycle, the Wechselian glaciation. An alternative simulation of future states was also conducted, including global warming.

The further selection of scenarios focused on addressing the safety-relevant aspects of evolution. The selected scenarios should cover all reasonable evolutions. Furthermore, it should be possible to calculate the risk associated with that connected to the presence of the repository as a sum of contributions from the set of scenarios. The following scenarios were selected for the SR-Site:

- A main scenario, corresponding to the reference evolution split into two variants
- A buffer advection scenario exploring the routes to and quantitative extent of advection conditions in the deposition hole
- A buffer freezing scenario exploring the routes to buffer freezing
- A buffer transformation scenario exploring the routes to buffer transformation
- A scenario exploring the routes to and quantitative extent of canister failure due to corrosion
- A scenario exploring the routes to canister failure due to shear load
- A scenario exploring the routes to canister failures due to isostatic load
- Hypothetical, residual scenarios to illustrate loss of barrier functions
- Scenarios related to future human action

The central conclusion for the SR-Site was that a KBS-3 repository that would fulfil long-term safety requirements can be built on the Forsmark site, because the favorable properties of the Forsmark site ensure the required long-term durability of the repository's barriers. Based on the results of modeling, canister failures in a 1-million-year perspective would be rare. Even with a number of pessimistic assumptions regarding detrimental phenomena affecting the buffer and the canister, the occurrence would be sufficiently rare to limit their cautiously modeled radiological consequences to well below 1 per cent of the natural background radiation; see Figure 20–6.

Erosion of the bentonite buffer is the single process that could lead to the highest risk contribution in a million-year perspective. In the presence of water with low ionic strength, bentonite changes from a gel to a sol, and is more easily transported by flowing water. Loss of buffer may occur from exposure to low ionic strength waters but the extent is uncertain. The Forsmark site has a large potential to maintain a sufficient ionic strength at repository depth over a glacial cycle. Loss of buffer mass, to the extent that advective conditions arise in the deposition hole may, however, occur in a million-year perspective for typically less than 10 (out of 6,000) deposition holes with high flow rates.

Canister corrosion requires transport of dissolved oxygen to, and corrosion products away from, the copper canister. Therefore advective flow in a deposition hole would enhance the canister corrosion rate. In the 1-million-year perspective, this may lead to failures of a few canisters when applying the most pessimistic of the hydraulic predictions made for the Forsmark site, with cautious assumptions regarding concentrations of corrosion agents and deposition hole acceptance criteria.

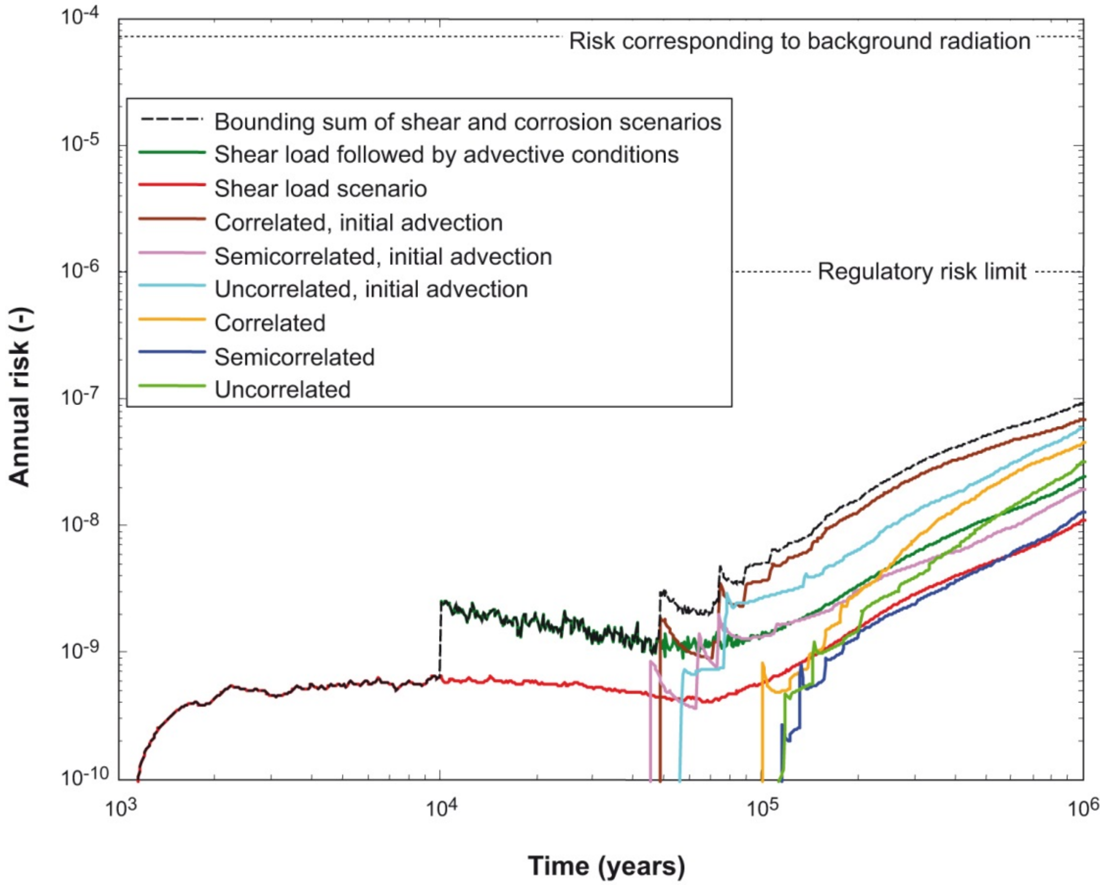

Figure 20–6. Risk curves expressed as annual individual risk. Several alternatives for the corrosion scenario are shown, and two for the shear load scenario. The bounding dashed curve is the sum of the curve for the shear load failure followed by advective conditions (dark green) and the curve from the variant of the corrosion scenario yielding the highest risk (brown). Three correlation models between fracture size and transmissivity have been used, where the fully correlated and uncorrelated models are assumed to be bounding models, whereas the semi-correlated model represents a more realistic description of the likely correlation between fracture size and transmissivity (SKB 2011d).

With pessimistic assumptions regarding buffer erosion, copper corrosion, and radionuclide transport condition, the radiological risk from canister failures caused by buffer erosion is pessimistically calculated to be around 1/100 of the regulatory risk limit in a 100,000-year perspective and around 1/10 of the regulatory risk limit in a 1-million-year time perspective.

There is also a small risk contribution for canister failures due to earthquakes. Larger earthquakes cannot be ruled out in a long-term perspective. However, probabilistic analyses imply that, on average, it would take considerably more than 1 million years for even one such canister failure to occur. The contribution of radiological risk from earthquakes is pessimistically calculated to be less than 1/100 of the regulatory limit in a 100,000-year perspective and less than 1/10 of the regulatory limit in a 1-million-year perspective.

Both risk contributors are related to the occurrence of large and/or highly conductive fractures intersecting the deposition holes. This applies to the buffer colloid release process and the impact of major earthquakes in the vicinity of the repository. Accordingly, such fractures must be identified, and deposition holes must be built to avoid intersecting these fractures.

20.2.4 Geoscientific Research

SKB's geoscientific research embraces four disciplines: geology, hydrogeology, hydrogeochemistry, and transport properties of the rock. Future research in these areas will be aimed at broadening the knowledge base concerning rock conditions of great importance for the outcome of the safety assessment. SR-Site clearly shows which properties and processes are crucial for the Repository for Spent Fuel. As described in the last RD&D program (SKB 2013a), the aim is to obtain greater knowledge of how the properties vary in the rock volume and as a function of the rock types that are present.

Research efforts in the discipline of geology is focusing on gaining a better understanding of spalling caused by stresses in the rock and by high temperature, methodology for identification of large fractures, further studies of glacially induced faults, seismic measurements to support earthquake modeling, and general increased knowledge regarding seismicity in the Swedish bedrock. Discrete fracture network models, which serve as a basis for analysis of rock movements and solute transport in fractures, is a major field of research with a bearing on hydrogeology, geology, and rock mechanics.

Data on hydrology on surface systems is mainly being gathered in investigations in Greenland (SKB 2014a) of how the flow changes and varies during a period dominated by permafrost. Hydrogeological research on deep groundwater systems involves integrating data and models with other disciplines (geochemistry, rock mechanics, and transport) and maintaining and improving the codes that are used for flow and transport calculations. Flow conditions during a glaciation with extensive permafrost have consequences for the situation at repository depth. Investigation data from the project in Greenland are being used both to understand the flow process and to optimize the modeling tools.

Research in the field of geochemistry continues to focus on reactions between water and rock and effect of movement of the water in fracture systems. Efforts are being focused on models where geochemical conditions are integrated with hydrogeological conditions and the transport properties of the rock. Microbial processes are also an increasingly important topic, e.g., the importance of acetogens, the interaction between microbes and viruses in the rock, biofilms on fracture surfaces, and microbial processes in the presence of hydrogen, methane and sulfide.

20.2.5 Technology development

The development work for the barriers for long-term safety is being pursued in production lines for fuel (SKB 2010f), canister (SKB 2010g), buffer (SKB 2010h), backfill (SKB 2010i), closure (SKB 2010j) and underground openings (SKB 2010k). The production lines for buffer, backfill, and closure share a number of common issues, so the development work for these production lines is integrated. Furthermore, technical systems are being developed for, e.g., logistics and machines that are unique for the final disposal facility.

When the applications were submitted in 2011, development of buffer and backfill had passed the concept phase. Since then, technology development has been further pursued in preparation for system design for buffer, backfill, and plugs in deposition tunnels. Development of the closure has not come as far, but efforts are being made to finish the work in the concept phase.

Technology development for rock includes detailed characterization, design, construction and maintenance of the Repository for Spent Fuel's underground openings. The development work also includes methods for investigations and modeling. It also includes rock construction, including sealing and rock support measures, as well as development of special equipment with a focus on rock condition prevailing in Forsmark.

SKB has developed a framework program for detailed characterization (SKB 2011c). SKB's underground laboratory, the Äspö Hard Rock Laboratory, has been extended with new tunnels, which has made it possible to integrate detailed characterization and design. The methodology for needs to be further developed. This applies mainly to strategies for detailed adaption of deposition areas and coordination of detailed characterization and building production.

Method and equipment for detailed characterization with associated modeling will also be developed. This development work will be carried out in a first step prior to construction of the Repository for Spent Fuel's access tunnels and in a second step prior to construction of the deposition areas.

Further information regarding SKB's work on technological development can be found in the latest RD&D Programme (SKB 2013a) and in the Äspö HRL Annual Report 2013 (SKB 2014b).

20.3 Short-lived Waste

Since 1988, the short-lived low- and intermediate-level waste has been deposited in SFR, the Repository for Short-lived Waste. The facility is located at Forsmark in Östhammar municipality, below sea level and covered by about 60 meters of granitoid rock. This repository was the first of its kind in the world, and contains different types of radioactive waste from nuclear power plants as well as from industrial and medical activities. Waste to be deposited in SFR must meet special waste acceptance criteria.The radioactivity of the radionuclides in the waste to be disposed of in SFR is dominated by short-lived radionuclides. This means that a large fraction of the activity will decay substantially during the operational phase. The total activity content at 100 years after closure is less than half of its original value, and 2 per cent remains after 1,000 years. Initially, ^{63}Ni dominates the activity, but after about 1,000 years this radionuclide will decay substantially, leaving Ni-59 and C-14 dominant.

20.3.1 The existing SFR facility

The underground part of the existing facility consists of four 160 meter long waste vaults, plus a 50-meter high vault with a concrete silo. Two parallel kilometer-long access tunnels connect the facility to the ground surface.

Low-level waste from the nuclear power plants is deposited in one of the four rock vaults. This waste consists of such items as used protective clothing. Since the radioactivity is very low, this waste type can be handled without any radiation shielding. It is transported from the nuclear plants in ordinary freight containers, and then directly deposited in the vault using forklifts.

The three remaining rock vaults are used to dispose of the intermediate-level waste. The radioactivity is by definition so high that radiation shielding is required. Dewatered filter resins are kept in two of these vaults, while the last rock vault contains more hard-to-handle waste.

The silo is a cylindrical vault, in which a free-standing concrete cylinder, with an inside diameter of 26 meters, has been built. Founded on a bed of sand and bentonite, the 18,000 m^3 cylinder is divided internally into a number of vertical shafts, as shown in Figure 20–7. The intermediate-level waste deposited here consists primarily of solidified filter resins used for purification of water from the reactors, and contains most of the radioactivity in the facility. Conditioned intermediate-level waste is deposited in the silo in concrete and steel molds as well as in steel drums. Grouting of waste packages in the shafts is done progressively.

Figure 20–7. Solidified filter resins are deposited in the silo.

SFR's current storage capacity is 63,000 m^3. As of December 2014, 36,000 m^3 of operational waste from the nuclear power plants and other nuclear activities have been deposited. Almost 1,000 m^3 of waste is added every year. However, this volume is not sufficient to accommodate future waste from the dismantling of the twelve nuclear power plants, nor is the current facility licensed for decommissioning waste. A need for additional disposal capacity in SFR has been further accentuated by the closure of the two reactors in Barsebäck. Additional disposal capacity is also needed for the operational waste from nuclear power plants in operation, since their operating life-times have been extended from 40 to up to 60 years. Hence, SKB plans to extend the facility with a new section, directly adjoining the existing facility.

20.3.2 The extended SFR facility

The extension of the SFR facility requires two licence applications: One under the Nuclear Activities Act and one under the Environmental Code. The licensing documentation was submitted to the authorities in December 2014, and consists of an application document and a set of appendices, among them the first preliminary Safety Analysis Report for the extended SFR, SR-PSU (SKB 2014c. Note: PSU stands for "project for SFR extension" in Swedish).

The planned extension entails an increase of the facility's storage capacity by an estimated 110,000 m^3 plus storage space for nine BWR reactor pressure vessels. In addition, the extended part of SFR will be used for interim storage of long-lived low- and intermediate-level waste awaiting final disposal in a future repository for long-lived waste, the SFL repository. The SFR extension will be built with a rock cover of about 120 meters, i.e. about the same level as the bottom of the silo. The underground part will consist of six new waste vaults. In Figure 20–8, the layout of the entire extended facility is shown.

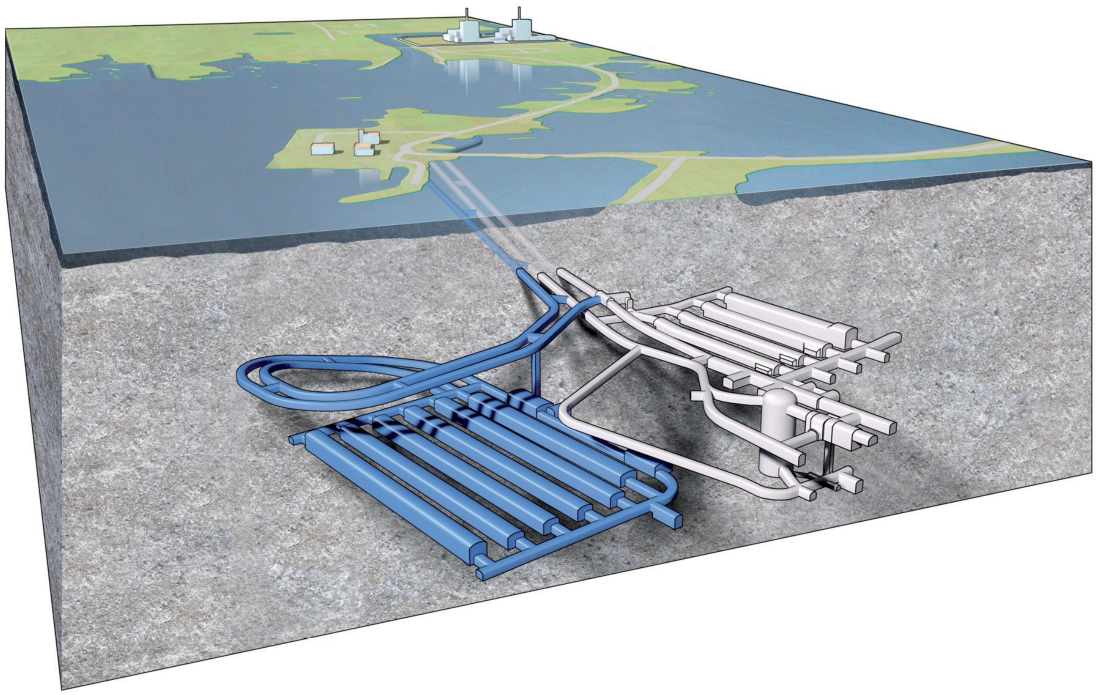

Figure 20–8. The existing SFR (light grey) and the extension (blue) with access tunnels

To achieve post-closure safety, two safety principles have been defined: limitation of activity of long-lived radionuclides and retention of radionuclides. The SFR is not designed to prevent releases indefinitely, but to control release within acceptable levels. Accordingly, SFR is designed to prevent, limit, and delay radionuclide release from the waste by means of both engineered and geological barriers. The properties of the waste, together with the properties of the waste containers and of the engineered barriers in the waste vaults, contribute to the safety by providing low water flow through the waste and a suitable chemical environment to reduce the mobility of the radionuclides.

The location of the repository below the Baltic Sea constitutes a barrier to future human intrusion and ensures a low hydraulic gradient and good retention during the first 1,000 years, during which much of the radioactive inventory decays. For longer time spans, degradation of the repository system must be considered. The amount of longer-lived radionuclides must therefore be sufficiently limited so that the radiological consequence of these longer-lived radionuclides does not pose a risk.

20.3.3 Site investigations

To find a suitable location for the planned extension, SKB has undertaken site characterization in the SFR area in Forsmark. To carry out the site investigation, the work was first divided into field investigations and modeling. The role of the field investigations was to collect primary data and to store them in a primary database, while the role of modeling was to produce the discipline-specific models and prepare the overall site description.

The site investigation was conducted mainly from the surface. Altogether four percussion boreholes and eight cored boreholes were drilled and investigated. Approximately 3,000 meters of drill-core length were analysed, and data indicate good rock quality in the area. The bedrock consists for the most part of metagranite and pegmatite with an average frequency of three to four open fractures per meter. The investigations in the boreholes included BIPS (Borehole Image Processing System), borehole radar, geophysical borehole logging, difference flow logging, water chemistry, and pressure measurements.

Integrated modeling has been carried out for the overall purpose of developing a site descriptive model, SDM-PSU (SKB 2013b). The site descriptive model forms a basis for repository engineering aimed at designing the underground extension facility and developing a repository layout adapted to the site. The model is also essential for safety assessment, since it is a vital source of site-specific input. Another important use of the site specific model is in the environmental impact assessment.

In general, the modeling project can be said to have comprised quality control of data, evaluation, analyses of primary data, three-dimensional modeling and reporting. The final result is a site and integrated geoscientific account of the properties of the investigated rock volume and its relation the regional environs. The principal remaining uncertainty concerns the occurrence, size, nature and transmissivity of sub-horizontal to gently dipping structures in the uppermost part of the bedrock.

20.3.4 Safety analysis

The SR-PSU report is a main component in SKB's licence application to extend SFR. Its role in the application is to demonstrate the long-term (post-closure) safety of the extended SFR repository over the 100,000 year analysis. In comparison, the analysis for the Repository for Spent Fuel is performed for the time period of 1,000,000 years. This is done by conducting a detailed safety analysis and evaluating the compliance with the Swedish Radiation Safety Authority's regulations (SSM 2008) concerning safety and protection of human health and the environment in the long-term perspective for the extended SFR repository. In addition to demonstrating long-term safety, the purpose of the present report is also to identify areas where further research and technology development are needed. The report complements SKB's RD&D Program 2013 (SKB 2013a) and helps in prioritizing further research work.

The methodology applied consists of ten main steps, and has been developed since the last safety assessment for SFR. It is now reasonably consistent with the methodology applied in the safety assessment SR-Site shown in Figure 20–5. A graphic illustration of the ten steps is shown in Figure 20–9.

However, certain repository-specific modifications are necessary. For example, the description of the future climate evolution has been handled differently. This is due to the fact that the time period analysed differs between a repository for short-lived waste and a repository for long-lived waste. The relative importance of different issues is also different, for example the time when a future glaciation/permafrost occurs is of crucial importance for a repository for short-lived waste, while in the case of the Repository for Spent Fuel it is more important to analyze how thick a future ice sheet may get, or how deep future permafrost may reach.

The evolution of the barrier systems will differ in safety assessment for an extended SFR repository compared with the Repository for Spent Fuel, partly because the waste and the barrier system are different. However, the future evolution of surface systems/biosphere is also based on the models used in SR-Site.

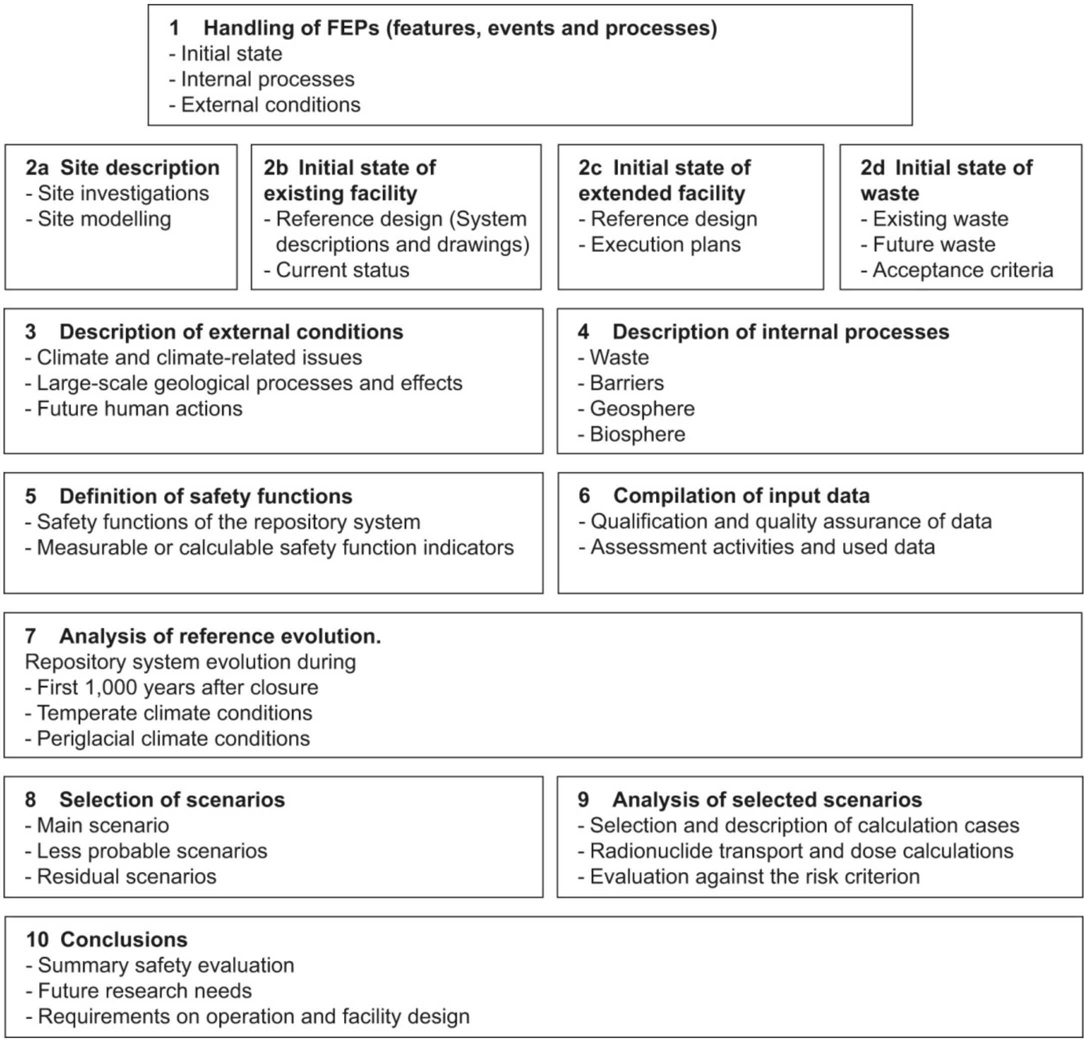

Figure 20-9. Overview of the ten steps in the methodology used for the long-term safety assessment (SR-PSU) of the repository for short-lived waste.

The initial state in SR-PSU is defined as the expected state of the repository and its environs immediately after closure in 2075. The description of the initial state is divided into two major parts, a first part describing the waste and the repository and a second part describing its environs. Information and assumptions that form the basis for the analysis were compiled (SKB 2014d), including general descriptions together with initial state values of variables applied to assess the safety of the repository.

Both external conditions and internal processes will affect the evolution of the repository system, which encompasses the repository and its environs. External conditions include climate and climate-related processes; for example, permafrost and shoreline displacement and the current process of global warming (SKB 2013c). Future human actions may also affect the future state of the repository (SKB 2014e). Internal processes include thermal, hydraulic, mechanical, and chemical processes, for example groundwater flow and chemical degradation affecting the engineered barriers. All processes identified to be of potential importance for the long-term safety of the repository system are described in three process reports for waste (SKB 2014f), barriers (SKB 2014g) and geosphere (SKB 2014h) respectively.

A key point in the handling of external conditions was the establishment of reference external conditions for the subsequent analysis. Most important are the future evolutions of the climate and climate-related processes. In earlier safety assessments for low- and intermediate-level waste a reconstruction of the last glacial cycle was used, along with a span of other climate cases, as was done for the repository for spent nuclear fuel. Given the shallow repository depth and the properties of the barriers in SFR, the present assessment has focused on determining the potential time of onset of the first period with permafrost in the Forsmark area. The current state of knowledge (SKB 2013c) suggests that due to human activities, in combination with small variations in future insolation, the evolution of the global climate during the coming 100,000 years will not resemble the last glacial cycle. Rather, the coming 100,000 years are expected to be characterized by a prolonged interglacial. Based on the external conditions and internal processes, a reference evolution, consisting of three distinct periods have been analysed:

- First 1,000 years after repository closure when the repository is below the groundwater table. During this time period, the low hydraulic gradient in the bedrock ensures close to stagnant hydraulic conditions and a low water flow through the repository. The climate is expected to remain temperate and the engineered barriers are expected to retain their properties. During this period, short-lived radionuclides will decay substantially. Less than 10^{-9} of the initial inventory will remain after 1,000 years. Also some long-lived radionuclides, such as Ni-63 with a half-life of 100 years, will decay substantially. During the following ~50,000 years periods of temperate climate will prevail. The shoreline passes over the repository and the hydraulic gradient increases. The engineered barriers will degrade, and the inventory of Am-241, C-14, and Mo-93 will decay substantially.
- During the last ~50,000 years periods of temperate and periglacial conditions will prevail. Beyond 50, 000 years only long-lived radionuclides like Ni-59, Cl-36, and U-238 and its progeny will remain. The evolution of the repository system is more uncertain than before and is described with a simplified model.

As shown in Figure 20–10, three climate cases are included in the reference evolution: the extended global warming, the global warming, and the early periglacial climate cases.

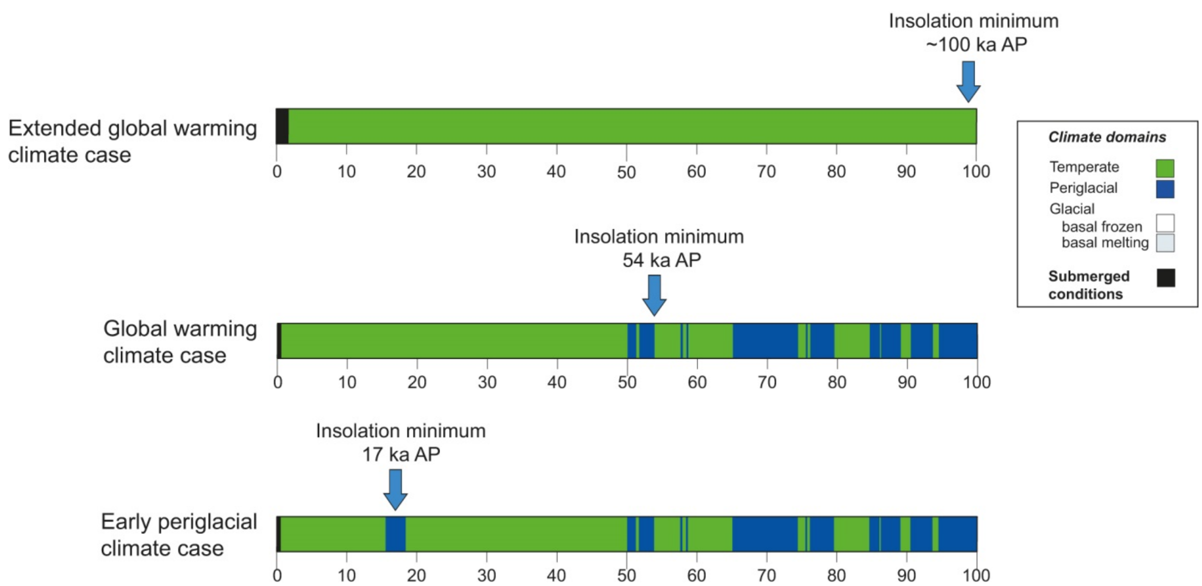

Figure 20–10. Evolution of climate-related conditions at Forsmark as a succession of climate domains and submerged periods for the climate cases included in the reference evolution. Note that the reference evolution does not include any glacial periods.

A main scenario split in two variants (global warming and early periglacial climate cases) is defined, based on the reference evolution, the initial state, and the internal processes that are found to be of importance for the long-term evolution and safety of the repository. The further selection of scenarios focused on addressing the safety-relevant aspects of evolution. These less probable scenarios are selected by going through all possible routes to violation of each safety function, as shown in Table 20–1. A set of residual, hypothetical scenarios is also defined. These consist of scenarios in order to illustrate the significance of individual barriers and barrier functions, human intrusion and the consequences of an unclosed repository.

Table 20–1. Safety functions and selected less probable scenarios.

Safety function					Scenarios
Limited quantity of activity	Low flow in bedrock	Low flow in waste vaults	Good retention	Avoid wells in the direct vicinity of the repository	
x					High inventory scenario
	x				High flow in bedrock scenario
		x			Accelerated concrete degradation scenario
		x			Bentonite degradation scenario
	x	x			Earthquake scenario
			x		High concentrations of complexing agents scenario
				x	Wells downstream of the repository scenario
				x	Intrusion wells scenario

The central conclusion of the safety assessment SR-PSU is that the extended SFR repository meets regulatory criteria with respect to long-term safety. Figure 20–11 shows the total radiological risk obtained from the combination of the maximum of the main scenario variants and all less probable scenarios.

The highest maximum annual radiological risk (4.0×10^{-7}) is obtained for the main scenario. For most of the scenarios, except for the earthquake scenario, initially the radiological risk increases with time, and then decreases or remain nearly constant during the rest of the assessment. The calculated risk for the main scenario and for the less probable scenarios are summed, taking the probability of each less probable scenario into account, to obtain a total risk over time for the repository. The maximum total risk 7.7×10^{-7} is obtained at 5000 AD.

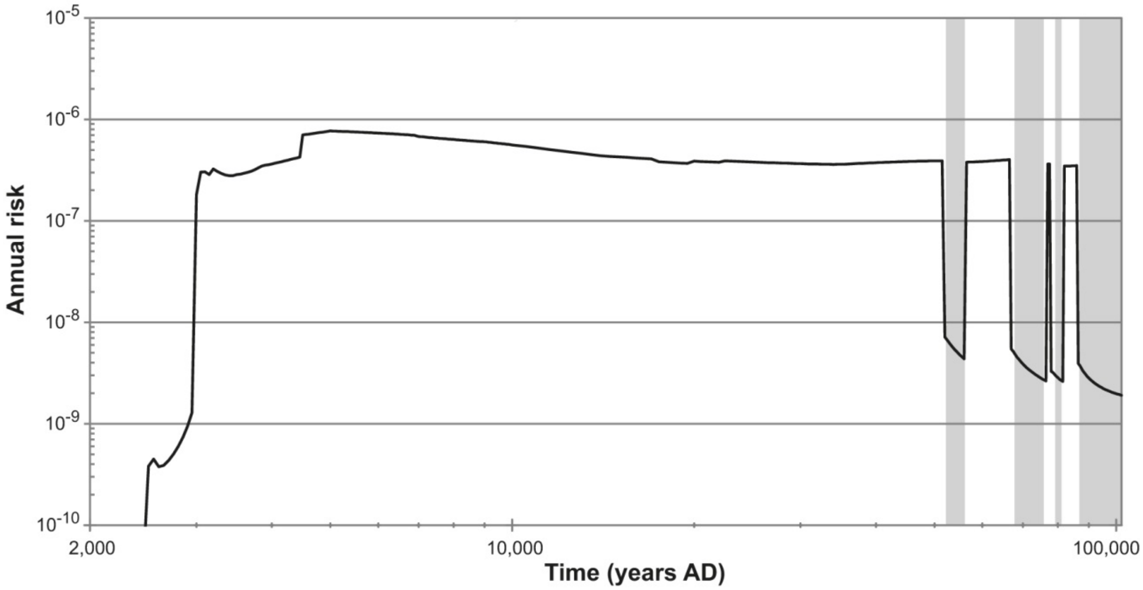

Figure 20–11. Total radiological risk obtained from the combination of the maximum of the main scenario variants and all less probable scenarios. Grey shaded areas represent periglacial conditions with continuous permafrost.

20.3.5 Research

The safety assessment has revealed areas that need to be explored to reduce the uncertainties in future long-term safety assessments. Some of these areas are specific to the SFR repository, whereas others can be relevant for both SFR and the planned repository for long-lived waste. Some areas, especially questions related to the bedrock and the biosphere, are also of importance to the repository for spent nuclear fuel. The previous planned future work related to the long-term safety of SFR is described in SKB's RD&D Program (SKB 2013a).

A number of areas for which additional research efforts might contribute to reduce uncertainties in future safety assessments have been identified in SR-PSU and will be considered in the coming RD&D Program 2016.

20.4 Long-lived Waste

The long-lived waste comprises of four main categories:

1. Neutron-irradiated components such as reactor internals, core components, and PWR pressure vessels.

2. Control rods from boiling water reactors.

3. Legacy waste from early research in the Swedish nuclear programs. This waste fraction is currently managed by AB Svafo.

4. Waste from other sources, such as industry, hospitals, and research facilities. This waste fraction in currently managed by Studsvik AB.

Among SKB's repositories the Final Repository for Long-lived Waste, SFL, is planned to be the last to be put into operation. However, several important milestones must be passed, such as choice of reference design, site selection, assessment of long-term safety, preparation of applications, constructions, etc. SFL will be the smallest of SKB's repositories. The total quantity of long-lived waste is estimated at about 16,000 m^3, about one-third of which belongs to the first two categories, and two-thirds to categories number three and four.

20.4.1 Previous work

A preliminary safety assessment for a repository for long-lived waste was presented in 1999 (SKB 1999) with the purpose of investigating the capacity of the facility to act as a barrier to the release of radionuclides, and to shed light in the importance of the location of the repository site. Three hypothetical sites represented fairly different conditions in terms of hydrogeology, hydrochemistry, and ecosystems. The primary conclusions from that study were:

- The highly mobile and long-lived radionuclides are the ones of greatest importance for assessing the safety, and hence the barriers and the ecosystems must be regarded on a very long timescale.
- To reduce the uncertainty in calculated environmental impact, it is important to reduce the uncertainties in the estimates of the dose-dominant radionuclides Cl-36 and Mo-93.
- The water flow at repository depth, and the nature of the ecosystem on the surface where releases may occur in the future are important.
- An unfavorable high water flow in the rock around the repository can be compensated for by better barriers in the near field, but their function must be sustained for a very long time.

This preliminary safety assessment was reviewed by the authorities and by an international team of experts. Both the regulatory authorities and the expert team draw the conclusion that a great deal of research and development work remains to be done before the level of knowledge in this field is comparable with that associated with the spent nuclear fuel, and called for a new repository design that can be considered sufficiently independent of site-specific factors and their long-term evolution.

20.4.2 A new concept study

Learning from the experiences from the preliminary safety analysis, SKB launched a concept study (Elfwing et al. 2013) to select a strategy and a generic repository design that would have the potential to meet the long-term safety requirements, regardless of previous design solutions. A number of repository concepts were evaluated, and a repository design consisting of a two-section layout was singled out as being the most functional. In this repository, one section is designated for metallic waste and the other one for other waste types.

The repository section for the metallic waste, the so called concrete repository, contains more than 99 percent of the estimated activity of the full SFL inventory in the form of neutron-irradiated components such as BWR control rods, core components, and PWR pressure vessels. The nuclide inventory also contains activation products contributing significantly to the dose burden of the critical group, e.g. Cl-36 and Mo-37. The total metallic waste volume amounts to approximately 3,000 tonnes of steel, equivalent to 5,000 m^3 of disposal volume.

The concrete repository section can be built using well-known materials and methods, and constructed either as a cavern or as a silo, based on local conditions. There is also an option of having one large silo or several small ones. An overhead crane, running on tracks mounted on separate pillars, may be used for deposition of the containers. The cross-sectional layout for a concrete repository designed as a cavern is shown in Figure 20–12. The length of the repository is approximately 75 meters.

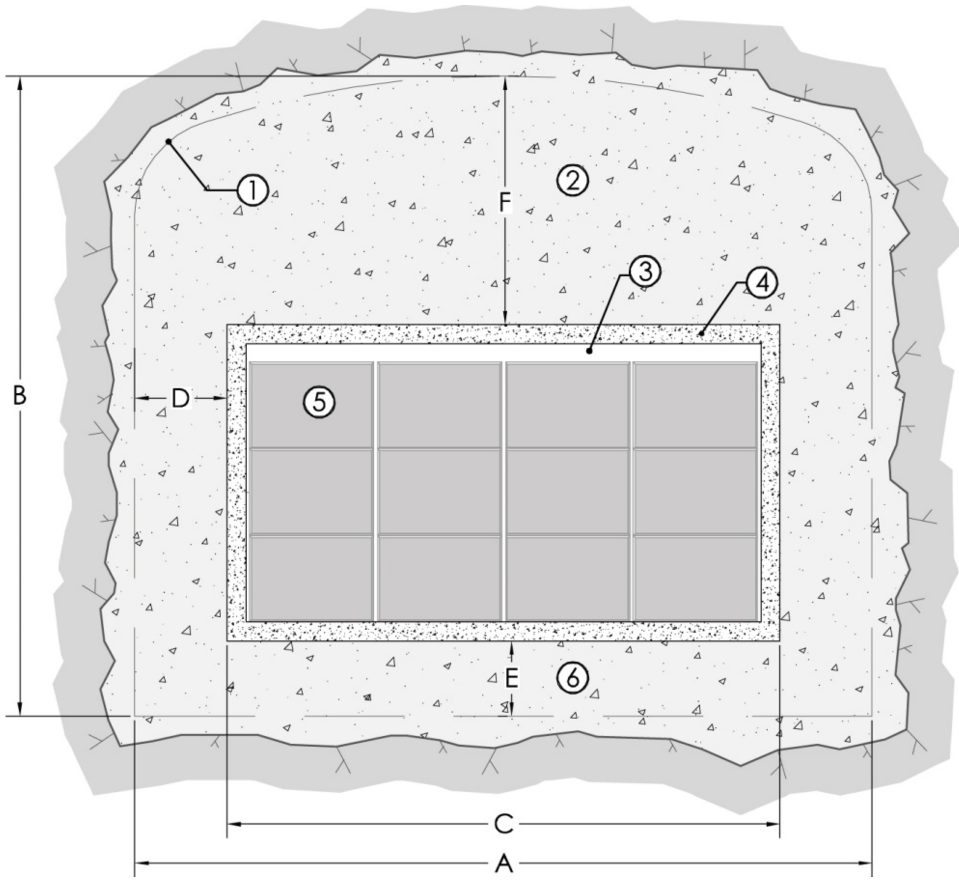

Figure 20–12. Schematic cross-sectional layout of the repository section for the waste from AB Svafo and Studsvik Nuclear AB. Legend: (1) Theoretical tunnel contour. (2) Bentonite pellets. (3) Grout. (4) Concrete structure for the operating phase (0.5 m). (5) Granite pillars. (6) Waste container. (7) Bentonite blocks. Approximate dimensions: A = 20 m, B = 17 m, C = 16 m, D = 2 m, E = 2 m, F = 3–4 m, G = 2–3 m.

Since most of the activity is bound in the metal, it is possible to postulate a dissolution rate equal to the corrosion rate as the source term in the assessment of long-term safety. Considering the large amounts of concrete present in the repository section, the pH can be expected to be buffered for a long period of time, providing a high pH environment.

This section of the SFL will be fully backfilled with concrete and plugged at the end of the operating phase. By using concrete in both structure and backfill, the environment in the repository section will be alkaline, which will create a passivating layer on the surface of the steel component and reduce the corrosion rate of steel. In turn, the low corrosion rate results in a low gas production rate, which is considered beneficial for the ability of the barriers to transport gas without negative effects on them.

Although not for the anions formed by chlorine and molybdenum, the concrete has a high sorption rate for many other radionuclides and the diffusion rate through concrete is low. In pristine concrete, the hydraulic conductivity is also very low, which makes diffusion the dominant transport mechanism.

The repository section for the legacy waste from early research activities in the Swedish nuclear programs, the so called clay repository, contains less than 1 per cent of the estimated activity of the full SFL inventory. This part of the inventory also contains chemotoxic waste comprising cadmium, lead, beryllium, and mercury.

This section of SFL has a total disposal volume of 11,000 m³, and will be fully backfilled with bentonite at the end of the operating phase, as shown in Figure 20–13. Since the bentonite permits only a small active flow, diffusion will be the dominant transport mechanism for radionuclides trough the bentonite. The initial state of the barrier is relatively easily described, in part because it is an established research field, in part due to the planned late installation of the bentonite.

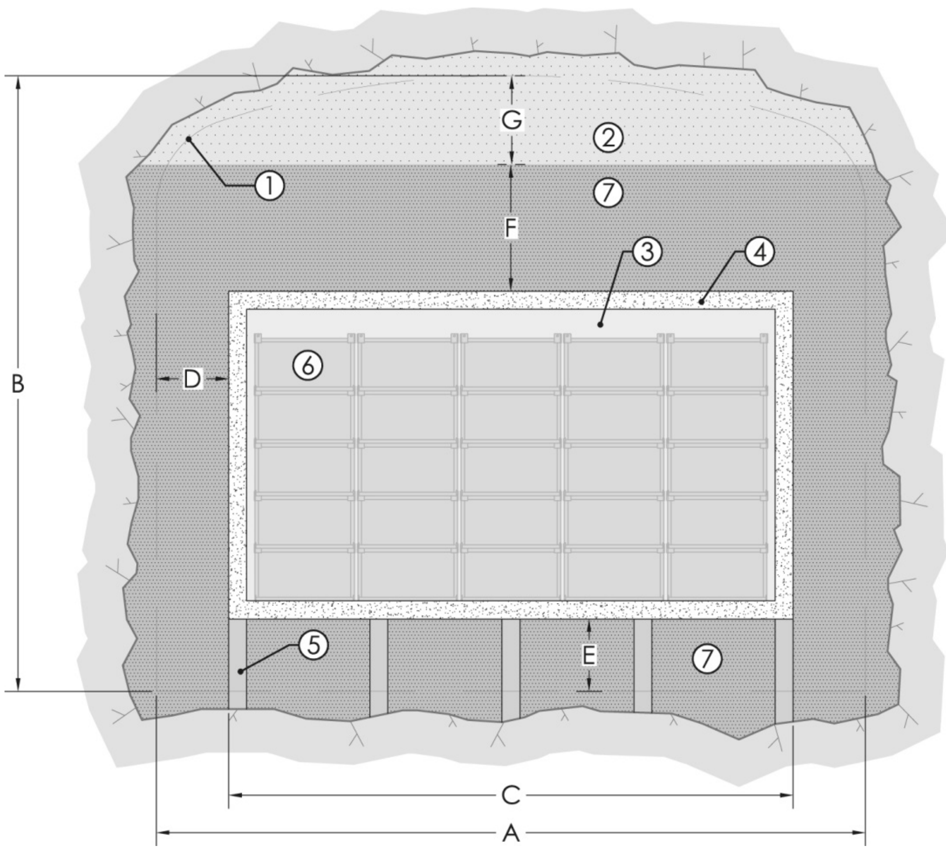

Figure 20–13. Schematic cross-sectional layout of the repository section for the waste from AB Svafo and Studsvik Nuclear AB. Legend: 1) Theoretical tunnel contour. 2) Bentonite pellets. 3) Grout. 4) Concrete structure for the operating phase (0.5 m). 5) Granite pillars. 6) Waste container. 7) Bentonite blocks. Approximate dimensions: A = 20 m, B = 17 m, C = 16 m, D = 2 m, E = 2 m, F = 3–4 m, G = 2–3 m.

Actinides can, together with organic material, form different types of actinide-carrying colloids that could be transported out of the repository. However, bentonite clay has the ability to effectively filter colloids. This is an advantage to this type of waste, since it contains more actinides compared to the metallic waste.

20.4.3 Safety evaluation

SKB's method for developing the SFL is an iterative process where technology development and research are followed by evaluation of the long-term safety of the repository. After having finished the concept study, an evaluation of long-term safety started in 2015. This evaluation is, however, not a complete safety assessment, since important components for making such an assessment are missing, for example site-specific data.

The purpose of the safety evaluation is to evaluate whether the chosen repository concept has the potential to meet the requirements and, if so, under what circumstances. Moreover, the safety evaluation will develop the set of requirements for the waste, the engineered barriers and the rock in order to support the work with waste acceptance criteria, continued technology development, and the site selection process. To reduce the uncertainties in future safety analyses, the safety evaluation also constitutes a basis for identifying areas in need of further research and serves as support in selecting and prioritizing those areas where the greatest development needs exist.

The concept study and the updated reference inventory for the metallic waste (Herschend 2013) comprise important background material for the safety evaluation, along with results from completed research projects for SFL and other repositories. A similar update for the legacy waste is underway.

Site data will be taken from the site investigation for the final repository for spent nuclear fuel in Laxemar, outside Oskarshamn. Other background material consists of international databases for features, events, and processes in the final repository and SKB's own databases from previous safety assessments.

20.4.4 Research and Development

In the repository, the waste will be emplaced in a central structure and surrounded by one or more engineered barriers contributing to safety. The research program will therefore be concerned with the long-term performance of the waste package and the engineered barriers (concrete, bentonite, and possibly crushed rock). In addition, research may be required to improve the waste form by conditioning. The research program aimed at studying the long-term performance is described in SKB's RD&D program 2013 (SKB 2013a) and is also a vital part of the planned extension of SFR, the Repository for Short-lived Waste.

The main safety function of the engineered barrier system is retardation of released radionuclides. This requires that all features, events, and processes which may affect the barriers with respect to this main safety function be identified. The main concern revolves around the interaction between the released radionuclides and the barrier material over the time period covered by the safety assessment. Thus, research efforts are needed to establish and model the chemical behavior of the SFL specific key radionuclides in the expected repository environment.

Another area that is of great importance for the safety assessment is to develop a detailed understanding of the mechanisms and time scales associated with the chemical and mechanical degradation of the concrete engineered barriers. Individual chemical processes such as leaching, carbonation, and sulfate attack are well known today, but there is a need for further knowledge of the effects of these processes, inidividually and in combination.

In the project "Chemical and mechanical properties of aged cement", specimens are being prepared by means of an electrochemical leaching method. Portlandite and some of the CHS gel can be leached out of the specimen, enabling the mechanical properties to be studied as a function of the degree of leaching. Other studies concern the influence of interactions between the concrete and species resulting from the degradation of the different waste forms. Experiments focusing on these issues were installed in the Äspö Hard Rock Laboratory as part of the project "Concrete and Clay", and will be retrieved and analysed at regular intervals to chart the progress of the degradation process over time.

A number of technical challenges related to choice of material, design, and construction have also been addressed (Grahm et al. 2013). Factors of importance are, for example, the composition of the concrete in the engineered barriers in the concrete repository, and the bentonite installation method in the clay repository.

20.5 Conclusions

The Swedish program for nuclear waste is well under way. Applications for a final repository for spent nuclear fuel and an extension of the existing repository for low-level waste were submitted to the authorities in 2011 and 2014 respectively. Both repositories will be situated in Forsmark, about 150 kilometers north of Stockholm. The planning for a third repository for long-lived waste has also started, but many important milestones still remain.

Long-term safety has been analysed for both the Forsmark repositories. The central conclusion in the safety analysis for the Repository for Spent Fuel was that a KBS-3 repository that fulfils the requirements can be built at Forsmark site, since the favourable properties of this site ensure the required long-term durability of the barriers. The conclusion is the same for the SFR repository, where the estimated risk is below the risk criterion of 10^{-6} during the assessment period.

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20.7 Acronyms

BWRs—boiling water reactors

PWRs—pressurized-water reactors

SFL—The Final Repository for Long-lived Waste (this acronym comes from the Swedish term "Slutförvaret för longlivat låg och medelaktivt avfall")

SFR—The Final Repository for Short-lived Waste (this acronym comes from the Swedish term "Slutförvaret för kortlivat radioaktivt avfall")

SKB—Swedish Nuclear Fuel and Waste Management Company

Chapter 21

Swiss Geological Studies to Support Implementation of Repository Projects: Status 2015 and Outlook

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ABSTRACT: The Swiss Nuclear Energy Law and the corresponding Ordinance, both in force since February 1, 2005, define deep geological disposal as the way forward for the long-term management for all types of radioactive waste and spent fuel declared as waste. The concept of "monitored, long-term geological disposal" is based on combining passive safety after emplacement of the waste with a monitoring period and retrievability "without undue effort" before repository closure. A new licensing process for site selection was approved and implemented by the federal government in 2008. This so-called "Sectoral Plan" process is driven by the long-term safety and feasibility of the geological repositories, and is based on a three-stage, stepwise, decision-making approach, with a strong participatory component from the affected communities and regions. Two repositories, one for low and intermediate level waste (LLW), and one for high-level waste and spent fuel (HLW), respectively, are proposed. In Stage 1 of the Sectoral Plan process, Nagra, on behalf of the waste producers, proposed six potential sites with favorable properties for construction, operation, closure and long-term safety of a deep geological repository. After extensive review by the authorities, followed by a broad public consultation, Nagra's siting proposals were approved by the Federal Government in November 2011. In Stage 2, the process of site selection continued with the goal of selecting at least two potential sites for each type of repository within the siting regions identified in Stage 1. In addition to the emplacement of the underground facilities, areas for surface facilities were identified as a result of an intense interaction with regional interest groups. In January 2015, following a safety-based comparison of the potential sites, Nagra submitted a proposal for two sites for Stage 3, where more detailed investigations will be performed, and the proposal for one site for each type of repository (or a combined repository) should be made. The host rock proposed for both repositories is the Opalinus Clay. Nagra's proposals are currently being reviewed by the authorities.

Chapter 21

21.1 Background and Historical Evolution of Radioactive Waste Disposal in Switzerland

Switzerland is a small country with limited natural resources and must import about 80% of its primary energy needs (predominantly petroleum products). Electricity supplies about 20% of the energy demand. About 40 % of the electrical energy is currently supplied by nuclear power plants (NPPs). Most of the remaining 60% is generated by hydropower. Commercial use of nuclear power in Switzerland began in 1969. By 1984, five nuclear power units were generating around 3300 MWe to the electricity grid. For the last six decades, Swiss electricity consumption has increased almost continuously, from about 10 TWh in 1950 to approximately 62 TWh in 2014.

Following the March 2011 Fukushima Dai-ichi accident, in May 2011, the Federal Government decided to make a fundamental change in the energy policy, which was approved by Parliament in June 2011. The revised strategy ("2050 Energy Strategy") is currently being discussed in Parliament, together with the corresponding modifications of the nuclear energy legislation. In particular, a gradual phase-out of nuclear power is expected by about 2035. No new nuclear plants will be built. In the next 20 years, energy saving measures and development of renewable energy resources should replace the 40% nuclear electricity currently produced.

Nuclear power production is the main source of Swiss radioactive waste, although a significant amount of waste also results from medical, industrial, and research applications. The responsibility for safe and sustainable management of all waste categories lies with the producers. 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) founded the "National Cooperative for the Disposal of Radioactive Waste" (Nagra) in 1972. Nagra is responsible for geological disposal and for advising on conditioning of all types of waste. The utilities remain responsible for transport, conditioning, and interim storage.

An overview of the historical evolution of the Swiss radioactive waste disposal program in 1945–2006 can be found in Hadermann et al. (2014). As early as 1978, the nuclear energy legislation set a very specific requirement to demonstrate that safe disposal of radioactive waste in Switzerland is feasible, as a prerequisite to the continued operation of existing NPP and the construction of any new NPP. The demonstration of disposal feasibility includes three main aspects: (i) engineering feasibility, based on existing engineering technologies; (ii) long-term safety to man and the environment; and (iii) site feasibility to fulfill the aspects (i) and (ii) above, based on the evaluation of geological formations in Switzerland.

An extensive regional field investigation program in northern Switzerland, including seven deep boreholes and two seismic surveys, as well as field investigations on several sites in the Alps, was launched in the early 1980s. In parallel, an extensive technical program was initiated with activities in areas such as development, testing, and application of safety assessment models; waste inventory characterization; development of geochemical databases; and characterization of engineered barriers. In addition, the underground rock laboratories (URLs)—Grimsel Test Site (managed by Nagra, in operation since 1984) and Mont Terri (managed by Swisstopo, in operation since 1996)—were established as centerpieces of Nagra's research and development program.

With a view to demonstrating disposal feasibility, the results of these activities were compiled in Nagra's "Project Gewaehr" reports, submitted to the Government in 1985 (Nagra 1985). For low and intermediate level waste (L/ILW), the demonstration was formally accepted by the Federal Council in 1988. Concerning high-level waste (HLW), the Federal Government judged that only two of the three conditions were satisfied: (i) construction feasibility, as safe construction and operation are possible with current technologies; and (ii) long-term safety is achievable, provided that the database used in the analyses is applicable to a sufficiently extensive, potential disposal area. However, the highly localized nature of the geological field data did not allow one to say with confidence that sufficiently large areas of crystalline rock with required properties could be found in Switzerland. A specific, suitable site had not been identified, and thus siting feasibility was not fully demonstrated. The Government required that the siting feasibility be more convincingly demonstrated, and also that sediments be investigated more intensively as alternative potential host rock for the disposal of HLW. Nagra subsequently expanded the geological investigations, construction feasibility studies, and safety studies of sedimentary host-rock options in parallel with its work on crystalline rock. The regional investigations of the crystalline basement were concluded and documented in 1994 (Thury et al. 1994, Nagra 1994a, b). Due to the restricted explorability of the crystalline basement, which in Switzerland is covered by sedimentary rocks, the Swiss Federal Nuclear Safety Inspectorate expressed strong reservations concerning the chance of finding sufficiently large blocks of suitable crystalline rocks (HSK 2004).

The Opalinus Clay was identified as the priority host rock by Nagra for the HLW repository, in consensus with the regulatory authorities and their experts. A field program in the potential siting area "Zürcher Weinland" was performed, including a deep borehole in the community of Benken in northern Switzerland and a 3D seismic campaign. To demonstrate the feasibility of disposing of HLW, in 2002 Nagra submitted the "Opalinus Clay Project" reports (Nagra 2002a, b, c) to the Federal Government. A comprehensive review and a positive evaluation of the project by the federal authorities and international experts (OECD/NEA 2004) in 2006 led to the final approval of the demonstration by the Federal Government.

The current legal framework for nuclear energy, including the management of radioactive waste associated with nuclear energy production, is defined in the Nuclear Energy Law (KEG 2003) and in the corresponding Ordinance (KEV 2004). These two documents came into force on February 1, 2005. The following provisions are related to the radioactive waste disposal issues:

- The concept of "monitored long-term geological disposal" (EKRA 2000), based on the idea of using passive measures to maintain the safety with a monitoring period and retrievability "without undue effort," before repository closure, which is applied to all types of radioactive waste—low and intermediate-level waste (L/ILW), high-level waste (HLW), and spent fuel (SF) when declared as waste.
- A step-by-step licensing process, under the responsibility of the Federal Government, which includes a general license for the selected site, followed by licenses for repository construction, operation, and closure.
- A Waste Management Program (Nagra 2008a) to be compiled every five years and approved by the Federal Government.
- The implementation of geological disposal in Switzerland. The option for the disposal of radioactive waste within the framework of a bilateral or multilateral project is kept open under very strict conditions, but not actively pursued.
- Site selection to be based on a so-called "Sectoral Plan" under the leadership of the Federal Government. A Sectoral Plan is a land-use planning tool commonly applied to large-scale infrastructure projects such as the construction of a motorway or of an airport (see below).
- Site selection supported by a broad consultation process. The government decision on a general license is subject to the approval of Parliament and to an optional national referendum.
- A 10-year moratorium on spent fuel reprocessing starting on July 1, 2006.

The legal framework also includes a set of ordinances and regulatory guidelines set forth by the Swiss Federal Nuclear Safety Inspectorate (ENSI).

21.2 The Repository Concepts

Since the founding of Nagra in 1972, work has been carried out on the development of disposal concepts and identification of potential sites for deep geological repositories. Based on the multi-barrier principle, the requirements for packaging, engineered structures, and geological isolation were derived for the various categories of waste. Two separate geological repositories are being planned: one for L/ILW, and the other for spent fuel (SF), vitrified high-level waste (HLW), and, if necessary, long-lived intermediate-level waste (ILW). (Note: for the SF/HLW/ILW repository, the term "high-level waste repository" (HLW) is generally used.) The option of co-locating these facilities at a single site, however, is being kept open.

The various elements of a multi-barrier safety system can be summarized as follows:

- Waste matrix: Glass for high-level waste, UO_2 and MOX pellets within their cladding for spent fuel, and immobilization of the waste along with various low- and intermediate-level wastes.
- Disposal canisters/containers: Corrosion-resistant canisters with a lifetime of at least 1000 years for the vitrified HLW and SF; concrete containers for L/ILW and ILW.
- Backfilling of the disposal tunnels: Using pre-compacted granulated bentonite for the HLW and SF, and cement-based mortars for L/ILW and ILW.
- Geological barrier: Host rock and adjacent low-permeability confining units.
- Geosphere and geological situation at large.

In the case of the HLW repository, the underground facility will be constructed in Opalinus Clay at depths of 400–900 m below the ground surface (Figure 21–1). The thickness of the Opalinus Clay layer varies between 100 m and 130 m. The host rock has a very low hydraulic conductivity ($\leq 10^{-13}$ m/s), and its dominating solute transport mechanism is diffusion.

The underground facility for SF and HLW includes a series of dead-end emplacement drifts, with excavation diameter of about 2.5–3 m and lengths ranging from 300 m to 1000 m. According to the design concept, the disposal canisters will be emplaced horizontally and co-axially with respect to the drift direction on pedestals of compacted bentonite blocks. Immediately after emplacement, the respective tunnel sections will be backfilled with highly compacted granulated bentonite. The current reference concept also includes a liner along the SF and HLW emplacement drifts. After approximately every 10th canister, the liner is interrupted by an intermediate seal, which ensures a direct physical contact between the bentonite backfill and the Opalinus Clay host rock, thus interrupting any potential preferential water flow, and solute transport pathways along the liner (see Nagra 2010).

Long-lived, intermediate-level waste (ILW) will be packaged in concrete disposal containers. They will be stacked in dead-end emplacement caverns about 7 m wide, supported by concrete lining. The remaining void spaces will be backfilled with a specifically designed mortar. The current reference concept foresees a gas-permeable seal at the end of each cavern.

The emplacement concept for the L/ILW is more or less identical to that of the ILW part in the HLW repository. The underground facilities of both the HLW repository and the L/ILW repository consist of:

- The main facility, i.e., the drifts in which the radioactive waste will be emplaced
- The pilot facility with representative amounts of the radioactive waste
- A test area, also called the underground research laboratory (URL)
- A central area

- Various types of seals at different locations within the underground structures

The emplacement drifts will be located in disposal areas, the final size and shape of which will be designed later based on in situ geological conditions. During construction and operation, access to both repositories will be provided by a ramp and shafts, or by their combinations.

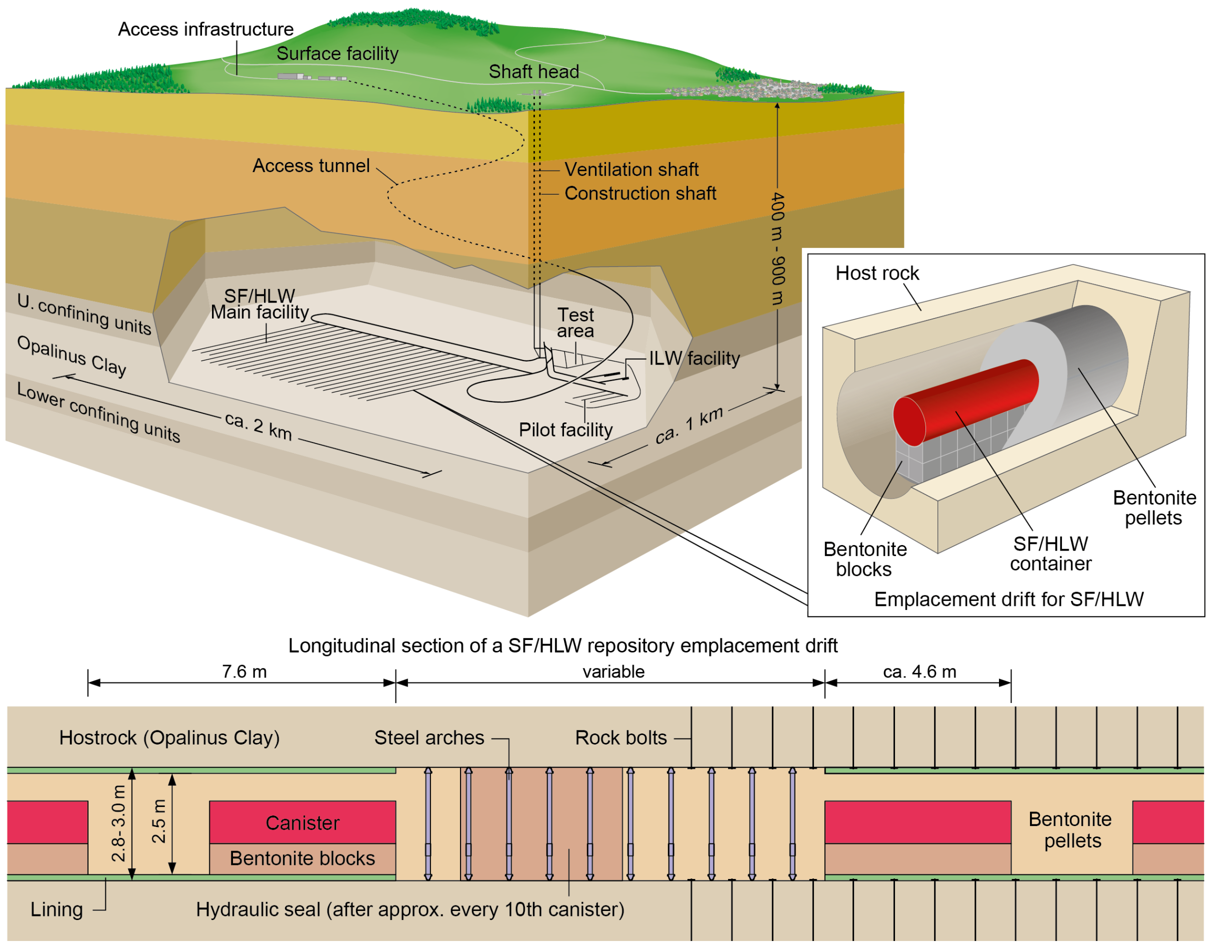

Figure 21–1. Schematic layout of the HLW repository with its main features (not to scale).

21.3 The Site Selection Process and a Geological Basis for Nagra's Proposals

The Federal Government's approval of Projekt Gewaehr for L/ILW repository in 1985, and of the Opalinus Clay Feasibility project for the HLW repository in 2006, addressed the issue of how to safely dispose of radioactive waste in Switzerland. The next step was to focus on the question of "where," for which a new site selection process, as required in the Nuclear Energy Ordinance (KEV 2004), had to be defined. This stepwise approach was defined as the "Sectoral Plan for Deep Geological Repositories" (SFOE 2008, also referred to as "Sectoral Plan" below). It was prepared by the federal authorities with input from different stakeholders under the lead of the Swiss Federal Office of Energy (SFOE), and, following a broad consultation, was approved by the Federal Government on April 2, 2008. The so-called "conceptual part" of the Sectoral Plan defines the site selection criteria, the role and responsibilities of the various stakeholders, and the three stages of the process that should lead to the identification of suitable sites. The Sectoral Plan also defines in detail the inputs needed for decision-making in each of the three stages. Throughout the site selection process, highest priority is given to safety.

Stage 1 of the site selection process starts with a "white map" of Switzerland (which implies that no area is a priori excluded), resulting in the selection of several broad geological siting regions for each repository type—L/ILW and SF/HLW. **Stage 2** leads to the identification of at least two potential siting regions for each repository type, and potential sites for the surface facilities. **Stage 3** includes investigations of the remaining geological siting regions in more detail. This leads to a safety-oriented selection of a single region for each repository type, and preparation of the corresponding general license applications.

At the end of each stage, the Federal Government approves the proposals based on the results of a detailed review by the authorities and on the outcome of a broad consultation involving all stakeholders. These three stages of the Sectoral Plan, in particular with respect to the geological description of the regions, are described in more detail in the remainder of Section 21.3.

21.3.1 Stage 1 of the Sectoral Plan

The focus of Stage 1, which started in 2008, was on a scientific screening process leading to the identification of broad, geological siting regions. Highest priority was given to safety, while ensuring the technical feasibility of the repository construction. Societal aspects were not part of the evaluation at this stage. To assess safety and technical feasibility, the Sectoral Plan defines 13 criteria, grouped into four broad areas, namely "Properties of host rock," "Long-term stability," "Reliability of geological information," and "Suitability for construction" (Table 21–1). These 13 criteria are supported by 49 indicators (derived by Nagra in Step 2 of Stage 1; see below). Each includes the evaluation of minimum (indispensable) requirements (for example, formation thickness) to assure a favorable impact on the long-term safety and feasibility.

Table 21–1. Criteria for site evaluation from the viewpoint of safety and engineering feasibility, as defined in the Sectoral Plan

Criteria group	Criteria
1. Properties of the host rock and of the formations contributing to the waste isolation	• Spatial extent • Hydraulic barrier effectiveness • Geochemical conditions • Release pathways
2. Long-term stability	• Geologic / tectonic stability • Erosion • Repository-induced effects • Resource conflicts
3. Reliability of geological database and statements	• Ability to characterize the formations • Explorability of the spatial conditions • Predictability of the long-term changes
4. Engineering suitability	• Geomechanical properties and conditions • Underground access and management of inflowing water

The identification of suitable geological siting regions was conducted in five steps. In a first step, the waste inventory was defined (including reserves for future developments), and various waste types (approximately 120) were allocated to either the HLW or the L/ILW repository. In a second step, the barrier and safety concepts for the two repositories were defined based on the evaluation of geological siting possibilities; at this stage, quantitative and qualitative requirements on geology were derived. These related to identification of the spatial (i.e., lateral extent and depth of intact rock blocks) requirements of the host rocks, the barrier properties of the host rock (i.e., thickness and hydraulic conductivity), long-term stability (uplift, erosion, differential movements, etc., for the timescale of concern), the reliability of geological findings (spatial explorability and temporal predictability), and engineering feasibility (e.g., rock strength).

Steps three to five covered the evaluation of the geological siting options. The geological information basis available in Switzerland is extensive, and includes data and information from investigations performed by Nagra over a period of 30 years, as part of its geological disposal program, as well as on the analysis and interpretation of data gathered by other parties (Figure 21–2). The latter include, for example, deep boreholes and seismic campaigns for oil and gas prospection and geothermal energy, shallower boreholes, surface geological and tunnel mapping, high-level precision geodetic monitoring, etc.

In step three, the large-scale, geological-tectonic conditions were assessed, and potentially suitable, large-scale areas were identified from the viewpoint of long-term stability (uplift and erosion, differential movements) and spatial conditions (size of not significantly disturbed blocks of rock, explorability of spatial conditions). The evaluation showed that all large-scale geological-tectonic areas in Switzerland could, in principle, be taken into consideration for the L/ILW repository, whereas for the HLW repository, the Alps, the Folded Jura, the western Tabular Jura, and a small part of the Molasse Basin (western sub-Jurassic zone) had to be excluded.

Step four involved selecting the preferred host rock formations within the large-scale areas still under consideration. This was done in several sub-steps and led to the following results: (a) for the L/ILW repository: the Opalinus Clay with its confining units, the claystone-dominated unit 'Brauner Dogger' with its confining units, the marly-calcareous Effingen Beds, and the marl formations of the Helveticum (a tectonic accumulation of low-permeability marly rocks in the Alps) were proposed; (b) for the HLW repository: the Opalinus Clay with its confining units was proposed as the preferred host formation.

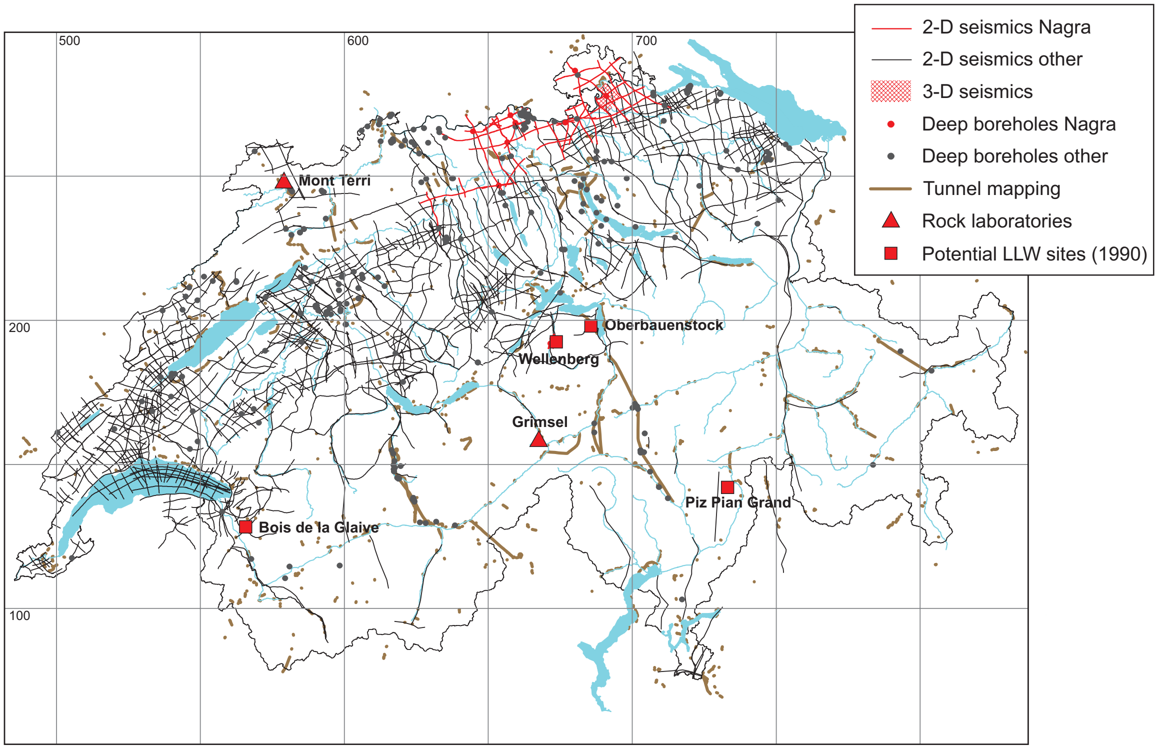

Figure 21–2. An overview of the geoscientific sources of information used in Stage 1 of the Sectoral Plan.

In step five, the configurations of the preferred host rocks within the large-scale areas under consideration were evaluated. Taking into account the presence of regional geological features (regional fault zones, over-deepened valleys resulting from glacial erosion, zones with indications of small-scale tectonic dissection, other zones to be avoided for reasons of neotectonics), preferred areas were identified within which the preferred host rocks could be found at suitable depth and with sufficient thickness and lateral extent. The preferred areas were used as the basis for delimiting the geological siting regions. Some siting regions contain several preferred areas, with sometimes more than one host rock type for the L/ILW disposal.

This systematic approach was developed to ensure that the identification and selection of the proposals for the geological siting regions were performed in a fully transparent manner. The detailed documentation was elaborated to deliver a clear answer to the question, *"why here and not there?"* from the point of view of safety. This is also considered to be important in view of gaining acceptance and support.

In October 2008, Nagra (Nagra 2008b) proposed three geological siting regions for the HLW repository, and six for the L/ILW repository (Figure 21–3). The three siting regions for the latter are almost identical with those for the HLW repository. The geological basis and rationale for site selection is documented in Nagra 2008c.

While Nagra's proposals were being reviewed by the authorities, under the auspices of the SFOE, the potential geological siting regions were identified, and the development of the organization for the participatory process in Stage 2 was initiated. The communities that needed to be involved in the

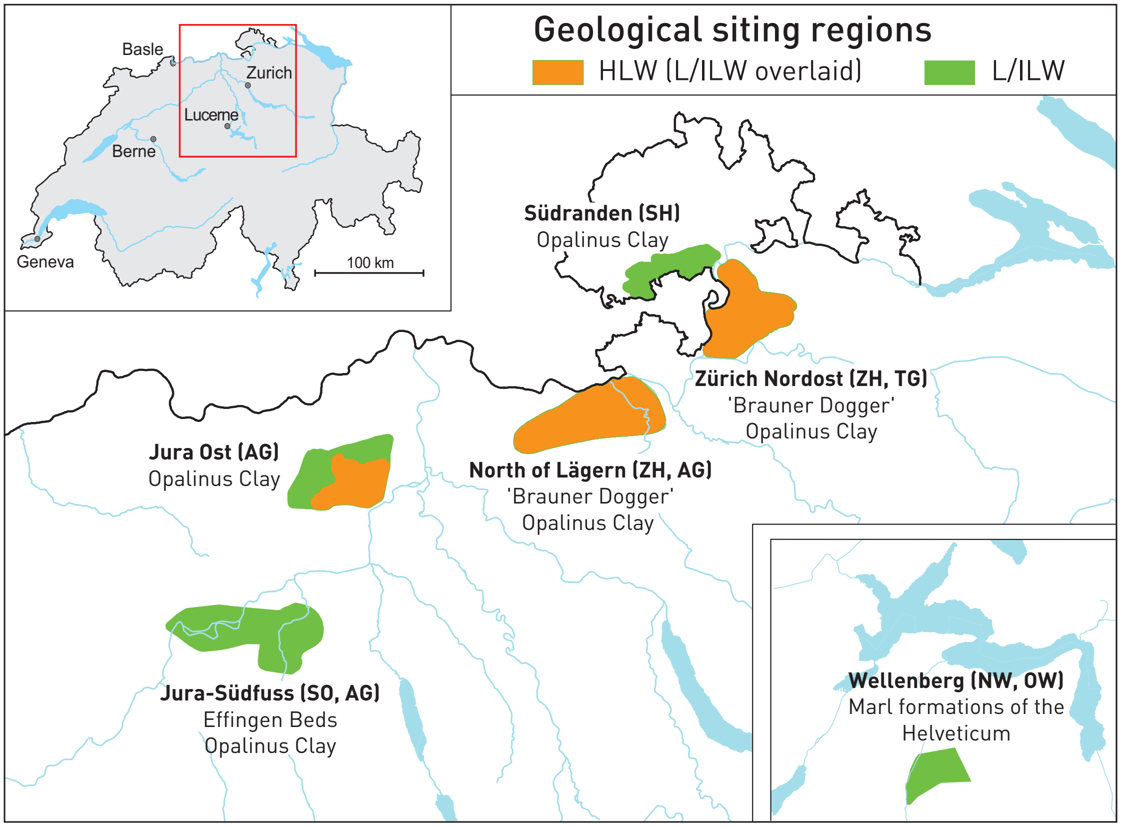

Figure 21–3. The proposed siting regions (Cantons are given in brackets) and the respective host rocks (Nagra 2008)

participatory process in each of the regions had to be determined. For this purpose, the areas that would be, in principle, available for siting the surface facilities were assessed, i.e., geological siting regions with an added 5 km to account for the possible offset of the surface facility from the geological siting region when using a ramp as access from the surface site to the underground facilities. Clear "no go" areas from the point of view of land-use planning were excluded. A regional participation group (so-called "regional conference") was proposed for each of the siting regions. For the regions in the north, communities from Germany also participated in the regional conferences.

After an extensive review by the authorities and a public consultation, Nagra's proposals for geological siting regions were approved by the Federal Government in November 2011. This completed Stage 1 of the Sectoral Plan process.

21.3.2 Stage 2 of the Sectoral Plan

Stage 2 of the Sectoral Plan has two main goals: first, to select at least one site for the surface facilities in each of the geological siting regions, and, second, to reduce the number of siting regions to at least two for each repository type.

For the selection of sites for the surface facilities, Nagra, as a first step in Stage 2, submitted proposals as a starting point for the discussions with the siting regions. The proposals were based on a conceptual design of the surface facilities, and took into account the following primary objectives proposed by Nagra: (i) ensure safety and engineering feasibility of the surface facilities and of the connection to the underground facilities of the geological repository; (ii) ensure compatibility with spatial and environmental planning in order to minimize environmental impact; and (iii) ensure an optimal integration of the facilities into the region. Nagra developed criteria for each objective, using more detailed indicators. The development of the proposals for sites for the surface facilities showed that, due to the high population density and the intense land-use in the regions considered, there was hardly any location without conflict with at least one of the criteria/indicators. Thus, the weighing and setting of priorities for the different criteria/indicators became a critical issue.

In January 2012, Nagra's twenty proposals were published, starting an intense phase of interaction with the regional conference and their sub-groups within each of the siting regions. By applying a different weighting on the criteria and indicators, the regional conferences identified additional sites for the surface facilities, which they requested Nagra to evaluate. A total of 32 potential sites were considered. By the end of January 2014 all siting regions had issued their opinion about potential sites for the surface facilities, and the development of the corresponding planning studies was completed by the end of May 2014. As a second goal of Stage 2, Nagra had to narrow down the number of geological siting regions to at least two for each repository type. At an earlier date, in preparation for Stage 2, Nagra had to evaluate the available geological information and to assess the impact of uncertainties with respect to decision-making (Nagra, 2010). This evaluation led to the decision to collect additional geological data in some of the siting regions (e.g., through 2D seismics, analyses of data from third-party boreholes, and borehole geophysical logging in geothermal boreholes).

The proposals for the regions to be further investigated in Stage 3 were prepared in five steps. In step one, the methodology used in Stage 1 was adapted to address the more specific boundary conditions and take into account requirements and suggestions set forth. In a second step, for those siting regions with more than one host rock that were proposed for the L/ILW repository, a safety-based comparison of host rocks led to identification of Opalinus Clay as the “priority host rock” for the following steps. (This step was not required for the HLW repository, as only the Opalinus Clay was identified as a host rock in Stage 1.) When comparing the abandoned sedimentary host rocks to the Opalinus Clay, clear disadvantages appeared, such as intercalations of potentially water-conducting sandy-calcareous or calcareous beds, reduced self-sealing capacity due to lower clay content, and limitations with respect to exploration and characterization of safety-relevant properties.

Hydraulic tests in Opalinus Clay in boreholes of northern Switzerland indicated horizontal hydraulic conductivities $\leq 5 \times 10^{-13}$ m/s for the depth larger than 200 m (Figure 21–4). The in situ packer test data generally showed good agreement with hydraulic testing of the rock matrix in drill cores, which supports the assumption of a (anisotropic) homogeneous medium. Various observations highlight the good self-sealing properties of Opalinus Clay (e.g., Bock et al. 2010), and natural tracer profiles provide independent evidence for diffusion-dominated radionuclide transport (e.g., Mazurek et al. 2009, 2011).

The third step led to the selection of an optimized spatial configuration of the priority host rock within the siting regions identified in Stage 1. In the fourth step, the suitability of the geological siting regions and the associated disposal perimeters was reviewed for safety, on the one hand using dose calculations (so-called “characteristic dose intervals,” see Nagra 2014c) and, on the other hand, through a qualitative assessment using the criteria relating to safety and technical feasibility set out in the Sectoral Plan. Finally, the fifth step involved a safety-based comparison and the overall comparative assessment of the

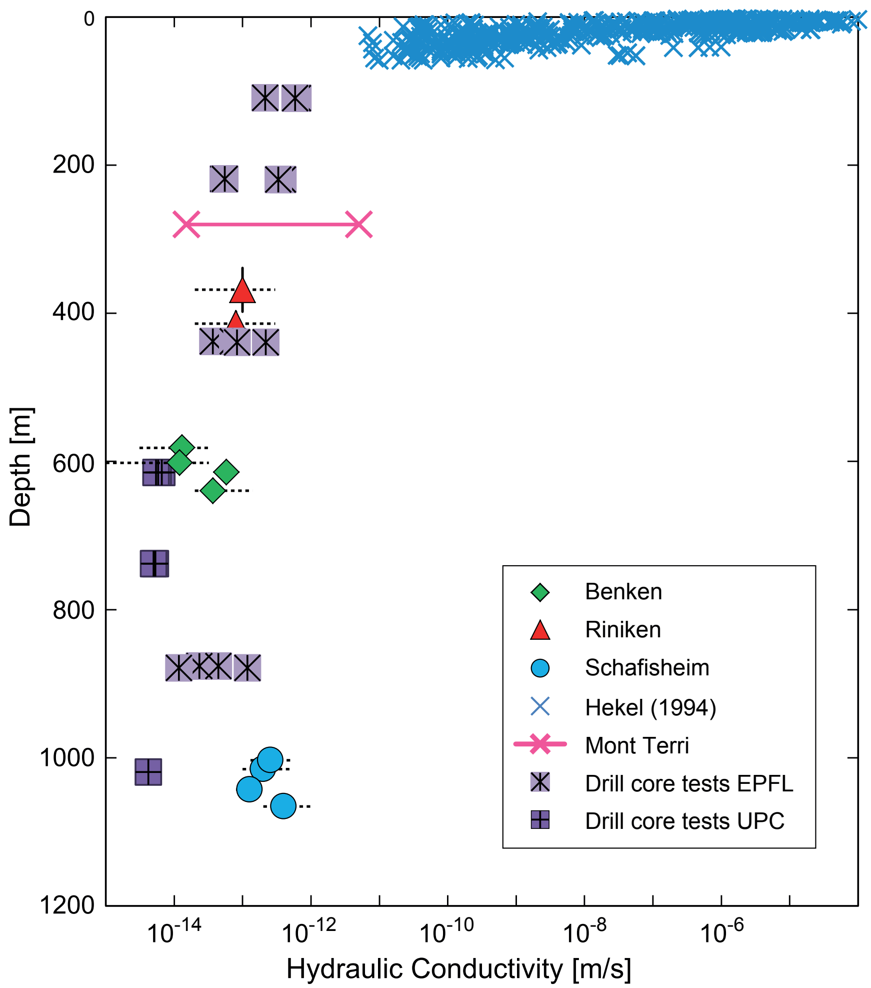

Figure 21–4. Hydraulic conductivity of Opalinus Clay as a function of depth. Packer test data indicate best estimates and ranges. Data from drill cores recalculated to depth based on average confining pressure (after Nagra 2014b, slightly modified).

Table 21–2. Safety-based comparison of the siting regions: Evaluation of decision-relevant features and indicators (after Nagra 2014a)

	HLW repository			L/ILW repository					
Decision-relevant features / Decision-relevant indicators	Zürich Nordost	Nördlich Lägern	Jura Ost	Südranden	Zürich Nordost	Nördlich Lägern	Jura Ost	Jura-Südfuss	Wellen-berg
Effectiveness of the geological barrier (E)	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Suitable
Hydraulic conductivity	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Suitable
Type of transport pathways and structure of the pore space	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Suitable
Transmissivity of preferential release pathways	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Suitable
Self-sealing capacity	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Limited suitability
Homogeneity of the rock structure	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Limited suitability
Thickness	Very suitable	Very suitable	Suitable	Suitable	Very suitable	Very suitable	Suitable	Limited suitability	Very suitable
Length of critical release pathways	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable
Colloids	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Limited suitability
Long-term stability of the geological barrier (S)	Suitable	Suitable	Suitable	Limited suitability	Suitable	Suitable	Suitable	Suitable	Limited suitability
Conceptual models of long-term evolution (geodynamics and neotectonics; other processes)	Very suitable	Suitable	Suitable	Limited suitability	Very suitable	Very suitable	Very suitable	Suitable	Limited suitability
Self-sealing capacity	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Limited suitability
Potential for formation of new water flowpaths (karstification)	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable
Erosion during the time period under consideration	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Suitable
Depth below the local erosion base level as relevant for formation of new ice-marginal drainage channels	Very suitable	Very suitable	Suitable	Suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable
Depth below terrain as relevant for rock decompaction	Very suitable	Very suitable	Suitable	Limited suitability	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable
Depth below top bedrock as relevant for glacial overdeepening	Suitable	Very suitable	Very suitable	Limited suitability	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable
Seismicity	Suitable	Suitable	Suitable	Suitable	Suitable	Suitable	Suitable	Suitable	Limited suitability
Explorability and ease of characterisation of the geological barrier in the siting region (C)	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Limited suitability
Variability of the rock properties as relevant for their ease of characterisation	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Limited suitability
Exploration conditions in the geological underground	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Very suitable	Less suitable
Engineering feasibility (F)	Suitable	Less suitable	Suitable	Suitable	Suitable	Less suitable	Suitable	Limited suitability	Limited suitability
Depth with respect to engineering feasibility (considering rock strength and deformation properties)	Suitable	Limited suitability	Very suitable	Very suitable	Suitable	Less suitable	Suitable	Suitable	Limited suitability
Geotechnical and hydrogeological conditions in overlying rock formations	Suitable	Suitable	Very suitable	Suitable	Suitable	Suitable	Very suitable	Limited suitability	Very suitable
Available space underground	Suitable	Less suitable	Suitable	Suitable	Very suitable	Less suitable	Very suitable	Suitable	Suitable

Very suitable · Suitable · Limited suitability · Less suitable geological siting regions and the associated disposal perimeters based on the relevant features specified by ENSI (identification of siting regions with "clear disadvantages" in comparison with other siting regions) with corresponding indicators proposed by Nagra (see Table 21–2).

Nagra's proposals were submitted and published by the SFOE in January 2015 (Nagra 2014 a, b, c). Of the six L/ILW regions and three HLW regions identified in Stage 1, Nagra proposed the two regions, Jura Ost and Zürich Nordost, for further investigation (Figure 21–3). The other four siting regions would also meet the safety requirements, but they were placed in reserve because in relative terms they showed clear disadvantages from a safety viewpoint. Such a disadvantage would be, for example, construction of a repository at greater depth in the Opalinus Clay and the associated additional challenges. In December 2016 ENSI announced its conclusions of the review (the full review will be available at the beginning of 2017). ENSI approved Nagra's proposals for: a) the choice of Opalinus Clay as the host rock for LLW repository; b) the two regions for further investigation in Stage 3; c) placing three of the remaining four regions in reserve; but d) recommended to include in addition North of Lägern in Stage 3. The review continues with a review by federal commissions and a broad public consultation before the Swiss Federal Council decides on the result of Stage 2 (expected in 2018).

21.3.3 Stage 3 of the Sectoral Plan

In Stage 3, the siting regions designated in Stage 2 will be investigated in detail, based on the 3D seismics and use of deep boreholes with extensive testing programs, accompanied by regional geoscientific studies addressing sedimentary facies distribution of host rock and confining units, long-term geological evolution (neotectonics, erosion), and hydrogeology. Finally, general license applications will be submitted for the L/ILW, the HLW, or combined repositories. The decision of the Federal Council on a general license is not expected before 2027. It will be followed by a debate in Parliament, and an optional referendum at the national level.

After the construction and operation of an *in situ* rock laboratory, applications for a construction license, and ultimately for an operating license for each repository, will be submitted to the Federal Government for approval. According to the current schedule, the L/ILW repository should be operational around 2050, and the HLW repository around 2060.

21.4 Geological studies at the URLs—the Grimsel Test Site and the Mont Terri Rock Laboratory in Switzerland

Two operating underground research laboratories (URLs)—the Grimsel Test Site (GTS) and the Mont Terri Project—are located in two different host rocks. GTS, owned and operated by Nagra, is situated in crystalline rocks in the Swiss Alps and has been in operation since 1984 (www.grimsel.com). Mont Terri, which is owned by the Republic and Canton of Jura and operated by the Swiss Geological Survey of the Swiss Federal Office of Topography (swisstopo), is situated in Opalinus Clay in the Jura Fold and Thrust Belt and has been in operation since 1996 (www.mont-terri.ch).

Nagra is actively engaged in both URLs with activities derived from the overall Nagra RD&D plan. In addition to contributing to Nagra's national geologic disposal program, both URLs offer a platform for cooperation with international partners, for training, and for interaction with interested stakeholders.

As a result of the selection of potential sites for geological disposal in clay-rich host rocks, the work performed by Nagra at the Mont Terri laboratory focuses on understanding host rock properties and repository-induced effects in the nearfield of a repository. Thermal, hydraulic, mechanical, chemical (THMC) phenomena and their impact on the host rock properties, repository-generated gas migration, interaction of host-rock with engineered barrier materials, and radionuclide transport through the host rock are issues being studied. Large-scale and/or full-scale long-term projects to demonstrate the transient behavior of the repository components have also been initiated with a planning horizon on the order of decades. Table 21–3 (after Vomvoris at al. 2015) provides an overview of the experiments in which Nagra currently participates. Nagra's current focus at GTS lies on the experimental investigation of engineered barrier system components. These experiments are considered complementary to those at the Mont Terri laboratory, taking advantage of the geologic boundary conditions offered by the GTS. An overview of the on-going projects at GTS is shown in Table 21–4 (modified after Vomvoris at al. 2015).

Table 21–3. Mont Terri field experiments of Nagra active participation: Phase 20 (6/2014–6/2015)

Title of experiment	**Partners(*)**	**End Phase**
Cement-Clay Interaction	A/C/I/N O S	37
Deep inclined borehole through the OPA	B G I ,N, T W	22
Porewater characterization—benchmarking and investigation of interface to adjacent aquifer	N T W	20
Diffusion, retention and perturbations	D N W	20
Long term diffusion	N W	23
THM part of the full scale emplacement experiment	A B D G N W	35
Emplacement part of the full scale emplacement experiment FE-D: co-financed by EC (LUCOEX Euratom project)	N	25
EDZ-characterization in the vicinity of the FE Gallery	B N T W	20
Long-term monitoring of the full scale emplacement experiment	D N	35
Analysis of geochemical data	A E N S	22
Hydrogeological analyses	B N	21
In situ heater test in VE microtunnel	B E G N	22
Gas path through host rock and seals	A B N W	22
Reactive gas transport in Opalinus Clay	A N	22
Iron corrosion of Opalinus Clay	A J N W	22
Corrosion of iron in bentonite	A N W	22
Long-term monitoring of the measured pore parameters	A B I N T V W	25
Properties analysis in lab tests	B G N	20
Microbial activity	A B N W	20
Modular platform for microbial studies	N T	21
Rock mechanics analyses	B N	20
Borehole sealing experiment	B G N	21
Palynology of Opalinus Clay	N T	21
Investigation of spatial variability within Opalinua Clay	B N	20
Investigation of wet spots	B N	20

*) A: ANDRA; B: BGR; C: CRIEPI; E: ENRESA; G: GRS; H: ENSI (HSK); V: CHEVRON; I: IRSN; J: AEA; N: Nagra; O: OBAYASHI; S: SCK-CEN; T: swisstopo; W: NWMO

In parallel to their main roles as research facilities, the URLs are important platforms for active interaction with the interested public, contributing to the understanding and acceptance of the scientific and engineering work performed in the area of geological disposal.

Two large-scale experiments, Full-scale High-level Waste Engineered Barriers experiment (FEBEX) at GTS and the full-scale emplacement (FE) experiment at Mont Terri are highlighted below, as examples of the activities performed in the two URLs and their contribution to Nagra's RD&D activities.

FEBEX is a full-scale, in situ EBS test for the disposal of high-level waste, performed under natural conditions, in which dummy canisters are placed horizontally in drifts and are surrounded by a clay barrier constructed of highly compacted bentonite blocks. The emplacement tunnel and the EBS system were constructed in 1996 with two heaters simulating the thermal input (Figure 21–5). Heating started in 1997 and the first heater was dismantled five years later, in 2002. The second heater has been maintained at a constant temperature of 100°C since 1997.

The partial dismantling showed that the hydration pattern around the canister is relatively symmetric, with no major differences along the x-axis. The resaturation process is driven by the suction of the bentonite rather than by the water availability in the host rock, especially in the early phase. As of the end of 2013, the water content in the buffer close to the heater still continued to increase, but very slowly, whereas the hydraulic pore pressures in the buffer and the geosphere had practically stabilized. The total pressure showed an almost flat trend with pressures in some parts of the buffer of over 6 MPa.

At this point, options included continuing the experiment far beyond the manufacturer's suggested lifetime of the sensors with very slow changes, if any, and a controlled excavation and characterization of the bentonite. The decision fell on the latter, and the FEBEX-DP project was initiated, with a broad, partner participation (in total 11 organizations). In March 2015, after 18 years of operation, the excavation of the experiment was initiated (Figure 21–6). Over 1000 samples have been collected as of the time of this report for (i) the determination of the distribution of key physical properties, such as density and water content; (ii) the characterization of corrosion and microbiological processes on the heater, instruments and coupons resulting from evolving redox conditions and saturation states, including gas analysis; and (iii) characterization of mineralogical interactions at material interfaces (e.g. cement-bentonite or iron-bentonite, rock-bentonite) by macro- and micro-level studies. The focus of the analysis will be the understanding of the thermo-hydro-mechanical (THM) and thermo-hydro-chemical (THC) processes in clay-based engineered barriers through integration of monitoring and in situ measurements.

The focus of FE is to provide a demonstration of the disposal concept and an opportunity to further develop and test THM models of the disposal system (engineered barrier and geosphere) at full scale. It is based on the Swiss disposal concept for SF/HLW shown in Figure 21–1. The construction of the 50 m long experimental tunnel with a diameter of approximately 3 m was completed in September 2012. At the far end of the tunnel the so-called "interjacent sealing section" (ISS) was built using only steel arches for rock support, whereas the rest of the tunnel is supported by shotcrete. In the FE tunnel, three heaters with dimensions similar to those of waste canisters were

Table 21–4. Projects at GTS—Phase VI (status: June 2015)

Project	Partner*	Schedule
Engineered Barrier Systems (Processes and Long-term behaviour)		
FEBEX Full-scale engineered barriers experiment extension	CIEMAT, Nagra, POSIVA, SKB, KAERI	(1995)–2015
FEBEX-DP Dismantling project	CIEMAT, Nagra, POSIVA, SKB, KAERI; ANDRA, BGR, Obayashi, RWM, SURAO, USDOE (LBNL)	2014–2016
GAST (Gas Permeable Seal Test)	Nagra, ANDRA, NWMO, KORAD	2010–2018
MACOTE (Material Corrosion Test)	Nagra, NWMO, RWM, SURAO.	2013–2018
BentLab (Bentonite laboratory for testing and characterization of buffer performance mid-scale)—in preparation	Nagra	2015–
Engineering & Operational Aspects of Repository Implementation		
LASMO (Large Scale Monitoring Project)	RWM , SURAO, Nagra	2012–2018
TEM (Test & Evaluation of Monitoring Techniques—initiated as part of MoDeRn, an EC-funded project)	ANDRA, NDA (RWM), Nagra	2006–
LSP (Low-pH Shotcrete Plug experiment—initiated as part of ESDRED Module IV, an EC-funded project)	EC (Task Leader ENRESA), Nagra	2006–
BELLT (Bentonite Large Scale Test)	Nagra	2014–2015
Waste—EBS—Host-rock Interaction		
CFM (Colloid Formation and Migration)	KIT-INE*, JAEA, KAERI, Nagra, POSIVA, RWM	2004–2018
LCS (Long-term Cement Studies)	JAEA, POSIVA, RWM, Nagra	2005–2016
Geological Barrier Processes & Characterization		
LTD (Long Term Diffusion)	NRI, JAEA, HYRL*, Nagra	2004–2018
Training & general scientific projects		
IAEA—courses	IAEA/NCE*, Nagra	2003–
Master courses / PhD	University of Bern	2006–
BBS (Broad-Band Seismometer project)	SED*	2010–
ISC (In situ Stimulation and Circulation Test; part of the project Small-Scale Experiments in Deep Underground Laboratories—Swiss Competence Center on Supply of Electricity/Deep Geothermal Energy)	ETH*, Nagra	2015–2018

*ETH-Z: Federal Institute of Technology, Zurich, Switzerland; HYRL: Hydraulic Research Laboratory, University of Helsinki; IAEA/NCE: IAEA Network of Centers of Excellence in Training and. Demonstration in Underground. Research Facilities; KIT-INE: Karlsruhe Institute of Technology, Institut für Nukleare Entsorgungstechnik, Germany; SED: Swiss Seismological Service.

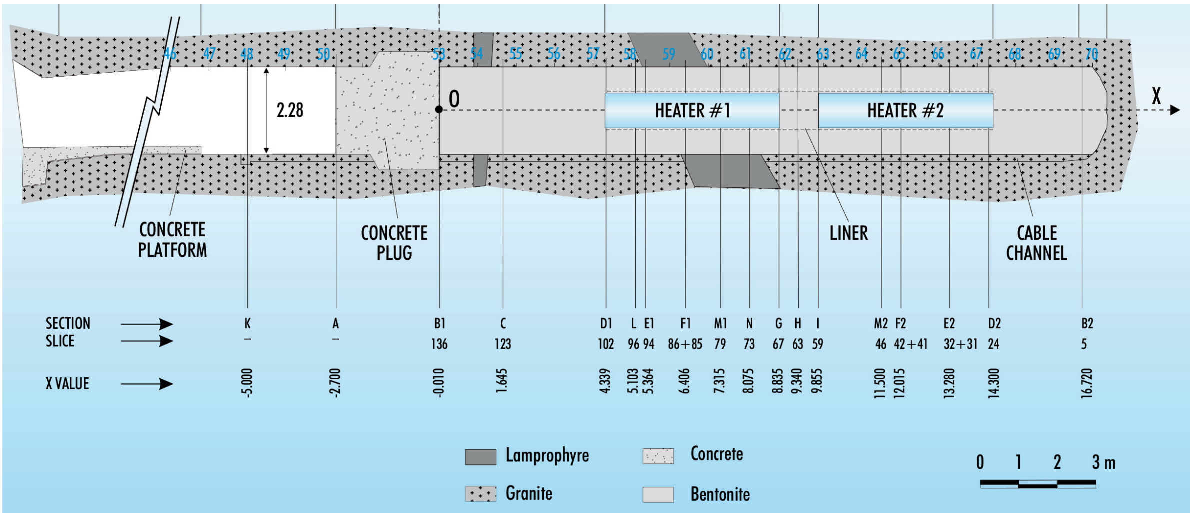

Figure 21–5. The FEBEX in situ test layout for the second operational phase.

Figure 21–6. Section 67 (approximately 2 m away from Heater 2) at the time of emplacement (2006) and excavation (2015)

emplaced on top of pedestals built of bentonite blocks (see Figure 21–7). The remaining space was backfilled with granular bentonite. Early in 2015, the experiment was sealed off toward the FE cavern with a concrete plug holding the buffer in place and reducing air and water flux.

The entire experiment implementation, as well as the post-closure THM(C) evolution, is monitored using several hundred sensors. The rock in the 'far-field' was instrumented with up to 45 m long boreholes drilled from the FE cavern; this instrumentation was completed before the FE tunnel was built and therefore allowed a 'mine-by' observation of the tunnel construction effects.

The backfill material is sodium bentonite from Wyoming, USA. Compacted blocks (bulk dry density of 1.8 g/cm^3 and water content of 20%) were used for filling in the ISS as well as the pedestals below the three heaters, and a granulated mixture with highly compacted bentonite granules ("pellets" with an average bulk dry density of 2.18 g/cm^3) were used for the buffer. The overall bulk dry emplacement density of the buffer was targeted to at least 1.45 g/m^3, in line with the Swiss concept. A prototype of the machine for backfilling the horizontal tunnels was constructed (Figure 21–8) and was successfully deployed for emplacing the backfill material.

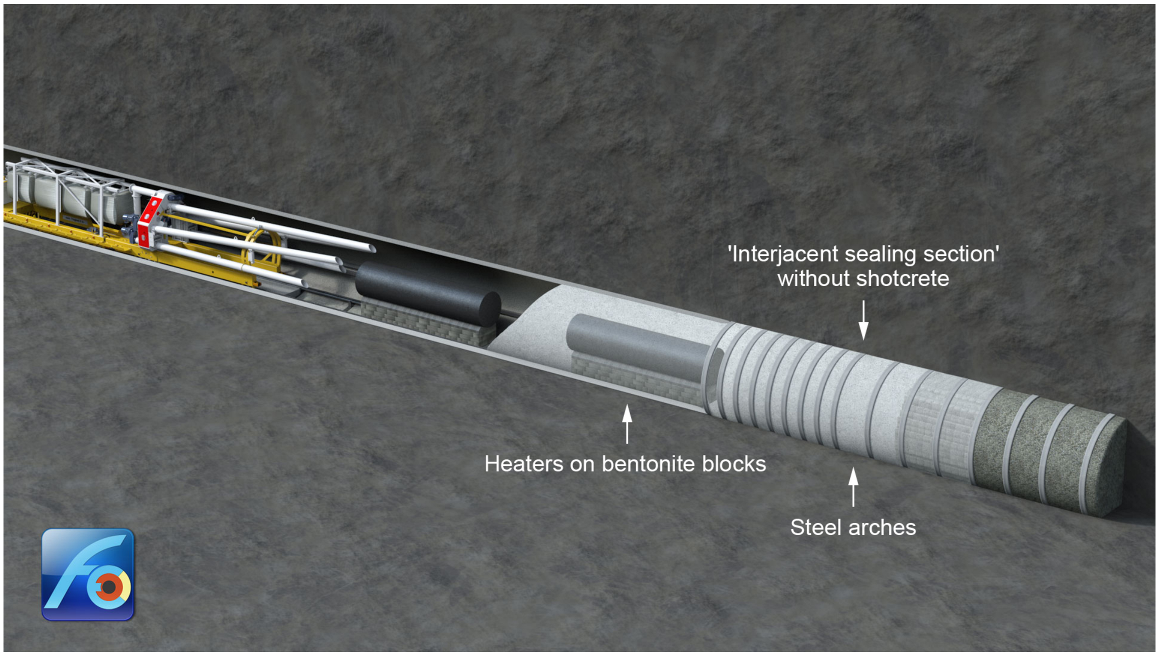

Figure 21–7. Visualization of the general experimental layout of and backfilling procedure of FE (diameter 3 m), as well as, the developed prototype machine with five auger conveyors used for the backfilling. The longest auger is 8.5 m long.

Figure 21–8. Left—The backfilling machine in the FE tunnel. Right—A detail of the emplaced granular bentonite and the contact with the liner.

The heating phase started in late 2014. With a target output of 1350 W per heater, temperature of approximately 120–150°C at the heater surface and 60–80°C at the rock surface is expected. According to current planning, the heating and monitoring phase of the FE experiment is envisaged to last at least 10 to 15 years. Co-financing of part of the project was provided by the EU within the framework of the LUCOEX program.

21.5 International Collaboration

The Swiss nuclear waste management program, although relatively small in terms of the budget and manpower, has been wide-ranging in scope in evaluating suitable host rocks for different waste types. This program greatly benefits if, in addition to its own resources, extensive use is made of international collaboration and information exchange with other national programs to allow effort to focus on specific key areas.

Apart from active participation in the International Atomic Energy Agency (IAEA) and the Nuclear Energy Agency (NEA) of the Office for Economic Co-operation and Development (OECD), Nagra has formal agreements with the European Comission and numerous organizations in various countries—Belgium, Canada, the Czech Republic, Finland, France, Japan, South Korea, Spain, Sweden and the United Kingdom. Informal collaborations extend the list further. The underground rock laboratories offer an ideal platform for scientific and technical exchanges and collaborations.

In the European context, Nagra is a member of the executive group of the Technology Platform for the Implementation of Geological Disposal (IGD-TP; http://www.igdtp.eu/), launched in 2009 by a group of European radioactive waste management organizations and other bodies involved in the long-term European research effort in this field, with the support of the EU. The secretariat of IGD-TP coordinates R&D activities following a strategic research agenda and a deployment plan through a number of joint actions and organization of exchange forums. In recent years, the IGD-TP has had a significant influence on the nature of projects selected within the EU Framework Program.

Since 1997, Nagra has also expanded the provision of technical support services to other countries and organizations and to applications outside the nuclear waste management field. This development has clear advantages for both Switzerland and foreign partners: the experience accumulated at considerable investment within the Swiss national program can be made available for other purposes, while the Nagra staff has the opportunity to further broaden their experience, which would be essential for the Swiss program.

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21.7 Acronyms

ENSI—Swiss Federal Nuclear Safety Inspectorate

EU—European Union

FE—Full-scale emplacement

FEBEX—Full-scale High-Level Waste Engineered Barriers

GTS—Grimsel Test Site

HLW—High-level waste

IAEA—International Atomic Energy Agency

IGD-TP—Technology Platform for the Implementation of Geological Disposal

ILW—Intermediate-level waste

ISS—Interjacent sealing section

L/ILW—Low and intermediate level waste

Nagra—National Cooperative for the Disposal of Radioactive Waste

NEA—Nuclear Energy Agency of the OECD

NPP—Nuclear power plants

OECD—Office for Economic Co-operation and Development

R&D—Research and development

SF—Spent fuel

SFOE—Swiss Federal Office of Energy

THC—Thermo-hydro-chemical

THM—Thermo-hydro-mechanical

THMC—Thermal, hydraulic, mechanical, chemical

URL—Underground rock OR research laboratories

Chapter 22

The Status of Geological Disposal in the United Kingdom

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22.1 Introduction

The United Kingdom (UK) has been a nuclear nation since the late 1940s, when research in support of the nuclear weapons program began following the Second World War. Shortly afterwards a civil nuclear power program commenced, and the first commercial nuclear power station was opened in 1956 at Calder Hall in Cumbria. By 1972 the UK had 11 operational power stations, all of which have now come to the end of their operational lifetime. Construction continued, and there are now eight operational power stations across the UK.

At the time these stations were built, radioactive waste management and disposal were not a priority, and as a result the UK has accumulated a legacy of higher activity wastes and material from electricity generation and defense activities. Waste has also arisen from other industrial, medical and research activities. Most of this waste is stored on an interim basis at nuclear sites across the UK. More will arise in the future from the continued operation and decommissioning of existing facilities and the operation and subsequent decommissioning of future nuclear power stations.

Some waste is being disposed safely in facilities on the surface, but a long-term solution is still needed for higher activity waste, some of which will remain hazardous for hundreds of thousands of years.

Following an independent assessment (CoRWM 2006a) of a wide range of options for dealing with the UK's higher activity waste, the UK Government adopted a policy of geological disposal, coupled with safe and secure interim storage as the best available approach for the long-term management of the UK's legacy of higher activity wastes (CoRWM 2006b).

An illustration of a possible structure of a geological disposal facility (GDF) in the UK is shown in Figure 22–1. To identify potentially suitable sites for a GDF, the Government has developed an approach based on working with interested communities that are willing to participate in the siting process. A framework for implementing geological disposal is set out in a Government Policy White Paper (UK Government 2014). This sets out an initial program of preparatory work to be completed before formal discussions with communities can begin. A high-level program showing the stages of implementation, the activities at each stage and indicative timescales is given in Figure 22–2. Formal discussions with communities can begin once this program of preparatory work is complete.

The Nuclear Decommissioning Authority (NDA) is responsible for implementing Government Policy on the long-term management of radioactive waste. Radioactive Waste Management Ltd (RWM) is a wholly owned subsidiary of the NDA, established as a delivery body to work with the waste producers and Government to develop waste management solutions and deliver geological disposal.

This paper describes RWM's progress in implementing geological disposal. It starts with an explanation of the inventory for disposal, and then describes the progress made in the three main strands of RWM's current work: preparing for siting, working with the waste producers, and developing the underpinning science and technology. Following that, there is a description of RWM's work with the regulators, key challenges in implementing disposal, and a summary of the next steps.

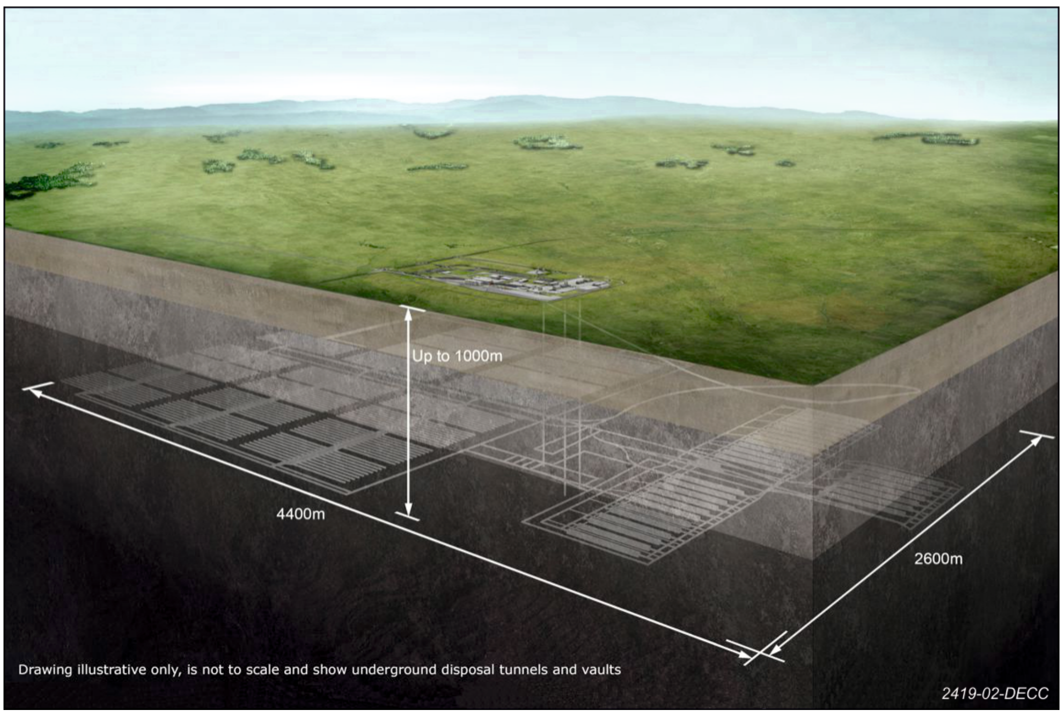

Figure 22–1. Illustrative example of a geological disposal facility

Figure 22–2. Implementation of geological disposal

22.2 Inventory for Disposal

The UK's higher activity waste is made up of:

- High level waste (HLW), which arises initially as a liquid during the reprocessing of spent fuel and is vitrified into a solid glass form. Spent fuel and other materials may also be declared as waste if they are not reprocessed.
- Intermediate level waste (ILW) from a range of nuclear activities including the operation, maintenance and decommissioning of nuclear facilities and the reprocessing of spent fuel.
- A small amount of low level waste (LLW), which is not suitable for disposal at the existing UK facilities for low level waste disposal.

For the purpose of planning and designing the GDF, and to give local communities clarity about the wastes that could be disposed there, the 2014 White Paper (UK Government 2014) defines an "inventory for disposal". In addition to the higher activity wastes and expected waste from existing power stations, the inventory for disposal includes wastes from potential new nuclear facilities and nuclear materials that could be declared as waste, such as spent fuel that will not be reprocessed, and uranium and plutonium that may be surplus to requirements.

Scottish wastes are not included in the inventory for disposal because the Scottish Government's policy is for the higher activity waste arising in Scotland to be managed in near-surface facilities (Scottish Government 2011).

For the purpose of design and safety studies, the inventory for disposal is separated into wastes that generate only low amounts of heat, that is, LLW, ILW and uranium (excluding highly enriched uranium); and high heat generating wastes, which comprises HLW, spent fuel, plutonium and highly enriched uranium. This classification reflects the different packaging and disposal processes involved, with low and high heat generating wastes being disposed of in separate disposal areas of the GDF. Highly enriched uranium is included with the high heat generating waste even though it does not generate significant heat because it has a similar disposal concept.

The total packaged volume of the inventory for disposal is approximately 650,000m^3. Figure 22–3 (NDA 2015) shows the packaged volume and activity, attributable to each category. This also demonstrates the significant contribution of the ILW to the overall volume. The dominance of the contribution to total activity from the spent fuels is also clear. The LLW, plutonium and uranium between them contribute only approximately 0.2% of the total activity.

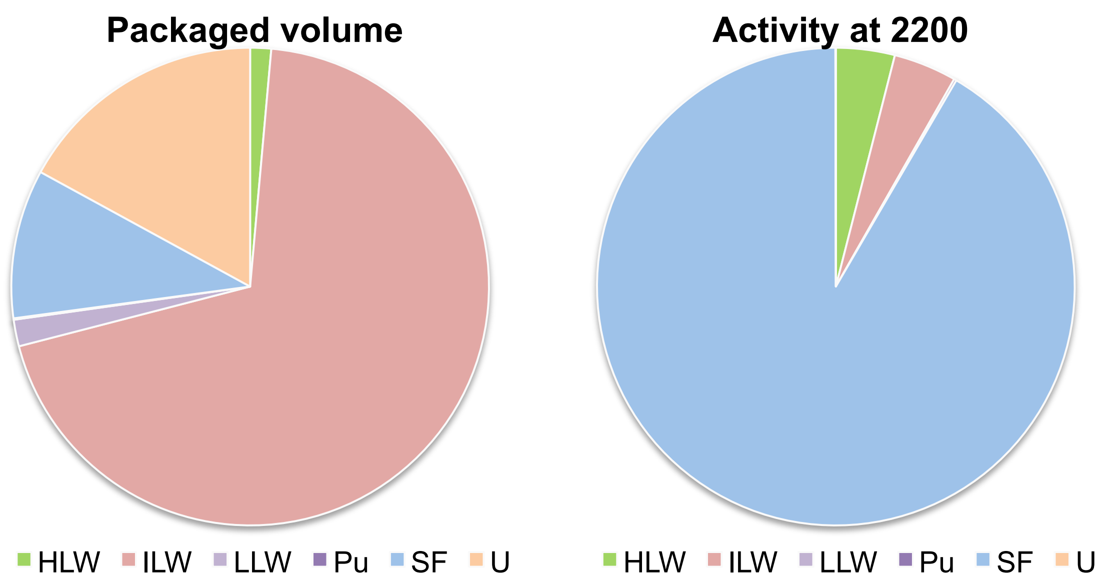

Figure 22–3. The packaged volume and activity broken down by waste category.

The Government has a strong preference to manage the inventory for disposal in a single GDF, on the basis that lower environmental impacts and major cost savings could be realized by developing a single site. However, the feasibility of a single GDF depends on whether a large enough volume of suitable rock exists in an area with a community willing to host a GDF. Hence, the Government has not ruled out developing more than one site.

22.3 Preparations for Siting

22.3.1 The History of the Siting Process

In 2008, the UK Government with the Welsh and Northern Ireland devolved administrations[1)] published a White Paper (DEFRA et al. 2008.) that set out the intention to manage higher activity waste in the long-term through geological disposal, with interim storage and ongoing research and development. It set out a siting process for identifying a GDF site that was based on local communities' willingness to participate in the process. That siting process operated for five years. However, by February 2013 there were no longer any communities involved.

In July 2014, following a call for evidence and public consultation, the Government and Northern Ireland Executive produced the revised 2014 White Paper (UK Government 2014), confirming Government policy for the management of higher activity wastes and setting out a revised siting process and policy framework. There were over 700 responses to the consultation, and the revised siting process has been shaped by the lessons learnt through these responses.

Useful lessons were learnt from the call for evidence on the earlier siting process set out in the 2008 White paper about how an approach based on working with communities can be made more effective in the future. In particular, the importance of providing information in advance on issues such as geology, socio-economic impacts and community investment became apparent. Another important lesson was that clear, evidence-based information on both technical issues and the siting process, available much earlier and in advance of formal discussions, would enable communities to engage with more confidence.

In response to these lessons, the revised siting process comprises a number of initial actions that the Government and RWM as developer will carry out to deliver clear, evidence-based information at the national level, to inform local discussions. Following these initial actions, formal discussions between interested communities and the developer will begin. The next two sections explain the developments in these areas.

22.3.2 Initial Actions

These initial actions of the siting process are:

- Development and implementation of a National Geological Screening exercise to provide information to help answer questions about potential geological suitability for GDF development across England, Wales and Northern Ireland.
- Development of a policy framework for planning decisions in England.
- Development of a process for working with communities, including community representation, community investment, and a means of obtaining independent views.

RWM is responsible for the first action, the National Geological Screening exercise, and the Government is responsible for the remaining two initial actions, which are developing a policy framework for planning decisions in England and developing a process for working with communities.

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1) Scottish Government does not support deep geological disposal and has published its own policy for long-term management of higher activity wastes arising in Scotland in near surface facilities.

22.3.2.1 National Geological Screening

National Geological Screening is an exercise that brings together existing information about aspects of geology that are relevant to the long-term safety of a GDF and makes it available in an accessible form. It will provide authoritative information on England, Wales and Northern Ireland that can be used in early discussions with communities about their geological potential to host a GDF.

The exercise is not intended to definitively rule areas as either suitable or unsuitable, however it is possible that it may lead to some areas being identified as unsuitable for hosting a GDF.

The screening exercise will present an overview of relevant existing information on geology to a depth of about 1,000 meters beneath England, Wales and Northern Ireland. It will focus on aspects of the geological environment that are relevant to the long-term safety of the GDF. In particular, it will set out the available information on the distribution of suitable rocks with low groundwater flow in which a GDF could potentially be built. It will identify geological features which may influence the movement of groundwater from GDF depths to the surface environment. Information will be included on the likely impact of future geological changes, such as sea level rise and future ice ages, and the distribution of minerals, hydrocarbons and other resources which may affect the likelihood of future civilizations inadvertently drilling into or mining the waste.

The National Geological Screening exercise has two parts. The first part involved developing Guidance (RWM 2016a), which sets out how information will be assembled and presented. The second part involves applying the Guidance. The Guidance has been reviewed by an independent panel established by the Geological Society of London, and subsequently was presented for public consultation (RWM 2016b).

In applying the Guidance, RWM, as experts in the science and engineering of geological disposal, is working with the British Geological Survey (BGS), which holds much of the authoritative and existing information on British geology, to apply the Guidance and develop the outputs. RWM is developing detailed technical instructions for the BGS to collate the existing information. The BGS will produce a series of Technical Information Reports and maps for each region. These will provide the geological basis for RWM to develop outputs describing the key characteristics of the geological environment and their relevance to safety in a way that will be accessible to a non-technical audience.

22.3.2.2 Policy framework for planning decisions

The policy framework developed by the Government for planning decisions will take the form of a National Policy Statement. The aim of the National Policy Statement is provide guidance to the Secretary of State and the Planning Inspectorate in their consideration of an application for a Development Consent Order for the use of boreholes to characterize potential sites and for the development of a GDF.

22.3.2.3 The process for working with communities

The process of working with communities is being developed by the Government, and involves:

- Deciding on an approach to community representation, informed by a community representation working group.
- Providing high level information on community investment, including the process for deciding how and when this money will be invested, in relation to communities engaging in the siting process and the community or communities that decide to host the GDF.

- Establishing a mechanism by which communities, the developer and the Government can openly access independent, third party advice on key technical issues during the siting process.

22.3.3 Preparing for Discussions with Communities

To provide a basis for formal discussions, RWM will first undertake activities to ensure that the public in general has good access to information on topics such as the science and engineering of geological disposal, the arguments on which the safety cases are built, and the environmental effects of GDF implementation, operation and closure. This information needs to be in an appropriate form to be informative and engage the interest of communities. Information will include that provided by the initial actions, high-level public-focused summary information, and the output of RWM's generic studies, such as the generic Disposal System Safety Case (generic DSSC) (NDA 2010a). The output from the National Geological Screening exercise is expected to be of particular interest.

In preparing for formal discussions RWM is also developing its approach to engagement, ensuring that its organization has the appropriate skills, culture and processes, and that appropriate messaging and materials are readily accessible.

22.4 Working with Waste Producers

The NDA has responsibility for the safe and efficient clean-up of the UK's nuclear legacy wastes. This includes developing strategies to be implemented by the site license companies for the retrieval, characterization, conditioning and packaging of waste. It is necessary to ensure that waste packages produced now for interim storage meet the anticipated needs of the geological disposal system.

To that end, RWM works closely with the waste producers to advise on their plans for waste packaging and interim storage prior to transport to and emplacement in the GDF.

RWM interacts with waste producers in four distinct areas:

- Provision of standards and specifications for waste packaging, based on the requirements of geological disposal
- Assessment of individual packaging proposals for disposability and, where appropriate, formal endorsement
- Assessment of waste producers' quality management system assessment and audit of the process for development and manufacture of waste packages
- Optimization of waste packaging strategy, supported by informal advice

These are described below.

22.4.1 Provision of Standards and Specifications for Waste Packaging

As implementer and future operator of the GDF, and therefore as the ultimate receiver of waste for disposal, RWM will be responsible for defining waste acceptance criteria. While plans for the construction of the GDF remain at an early stage, the information necessary to define firm and precise waste acceptance criteria is not available. In the meantime, as a precursor to waste acceptance criteria, RWM produces packaging specifications so that wastes can be converted into passively safe and disposable forms as soon as is reasonably practicable. These specifications define the bounding features and performance requirements for waste packages that would be compatible with the anticipated needs for transport to, and disposal in the GDF.

22.4.2 Disposability Assessment of Waste Packages

The RWM Disposability Assessment process has been established as a means of supporting the UK nuclear industry's ongoing work on the conditioning and packaging of higher activity wastes for disposal. The process has been extensively developed over a period of more than 20 years in cooperation with the site operators and industry regulators, and in a manner that aligns with regulatory expectations for the long-term management of higher activity wastes (HSE/EA/SEPA 2015).

The principal aim of the Disposability Assessment process is to minimize the risk that the conditioning and packaging of radioactive wastes results in packages incompatible with geological disposal and the associated transport system, as far as possible in advance of the availability of waste acceptance criteria for a GDF. As such, it is an enabler for early hazard reduction on UK nuclear sites.

Evaluation and assessments undertaken during a disposability assessment include the comparison of waste package performance against the package specifications and the generic DSSC. Where a disposability assessment identifies no significant uncertainties in the ability of the proposed packaging approach to produce disposable waste packages, RWM issues a Letter of Compliance to endorse the approach and the resulting waste packages. The Letter of Compliance demonstrates RWM's satisfaction that the resulting waste packages will be compliant with published RWM packaging specifications.

RWM interacts periodically with the waste producer during the continuing development and implementation of the proposed waste packaging process. This is of considerable benefit to both the waste producer and RWM; it provides the waste producer with the opportunity to submit information pertinent to the state of development of the proposals, and allows the information to be accumulated in consultation with RWM. The interaction is aligned with the waste producer's staged decision-making within the packaging process, and therefore reduces the waste producer's risk at each step of the process.

Figure 22–4 shows RWM's progress at March 2015 (RWM 2016a) in the disposability assessments of ILW packages, as a percentage of total conditioned volume to be produced. This illustrates the split between wastes actually packaged, wastes with a final Letter of Compliance, wastes within the Disposability Assessment process, and wastes that have not yet been assessed. Waste 'not covered by advice' is further split into 'commercial reactor final state decommissioning wastes' (which have not yet arisen and therefore are not currently available for packaging), and 'other ILW' waste.

Similarly, Figure 22–5 shows the status of HLW that is packaged and within the Disposability Assessment process, in the process but not packaged, and yet to be addressed.

22.4.3 Quality Management System Assessment and Audit

RWM also undertakes on-site technical audit of waste producers' management systems and package records. The purpose of these audits is to assess compliance with RWM's requirements as the eventual custodian of the waste packages, with the emphasis on the technical aspects of waste packaging and record keeping.

Waste produces are required by RWM to establish a data recording system for acquiring, recording and subsequently managing information for each waste package such that it can be used to establish, infer or predict package properties and performance under all relevant circumstances. Ultimately, this information may be used to demonstrate conformance with future requirements for transport and GDF waste acceptance criteria.

Waste packages require an associated waste package record that should:

- Describe the physical, chemical and radionuclide content of the waste package
- Identify and define the properties and performance of the waste package that are relevant to its ongoing management
- Provide a comprehensive radionuclide inventory and
- Provide sufficient data to predict the evolution of the waste package with time, and of the effect of interactions with other packages and GDF components

The audits enable verification that appropriate management systems are in place to control activities that could impact the quality of the packaged waste, confirm that waste packages have been manufactured within the requirements of the Letter of Compliance, and give RWM an opportunity to focus on specific requirements, such as conditions of the Letter of Compliance.

22.4.4 Optimization of Waste Packaging Strategy

RWM also works with waste producers to seek opportunities to improve the existing plans for waste packaging and interim storage in the light of new technology or innovation. This work considers all waste management activities up to and including geological disposal and specifically the implication of activities and plans for waste conditioning and packaging that could affect disposability. This work allows the waste producers to implement near-term solutions that allow progress to be made in their site lifetime plans, reduce risk across the waste lifecycle, and enable cost savings to both the waste producers and the NDA.

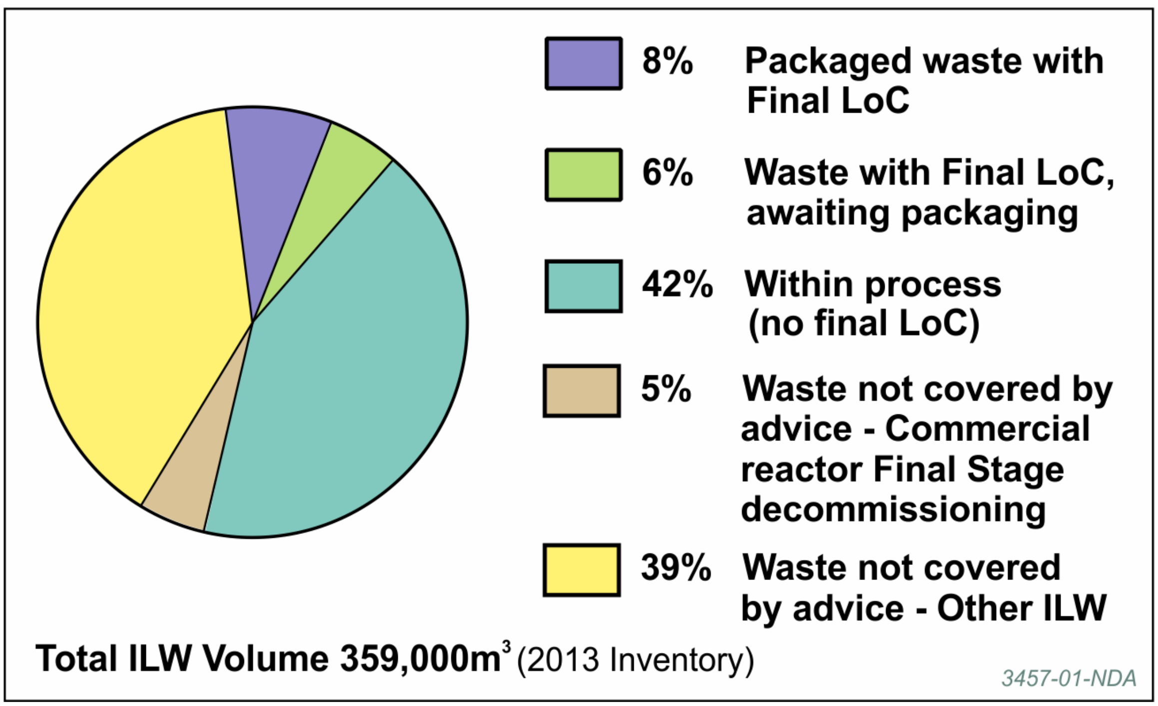

Figure 22–4. Packaging status of ILW (2013 UK Radioactive Waste Inventory, conditioned waste volumes)

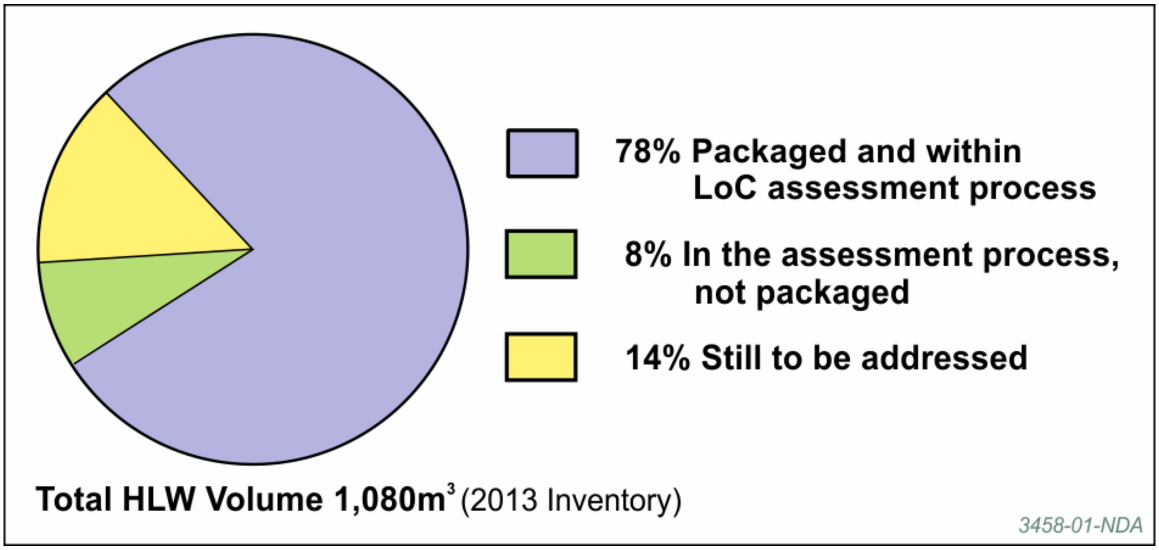

Figure 22–5. Packaging status of HLW (2013 UK Radioactive Waste Inventory, conditioned waste volumes)

22.5 Science and Technology

RWM is currently developing the geological disposal system on a generic basis. That is, not specific to any site. These studies include maintaining the disposal system specification, illustrative designs and safety assessments, and developing the underpinning knowledge base.

22.5.1 The Iterative Approach to Developing a Geological Disposal System

The development of a geological disposal system is an iterative process. A specification for a disposal system is derived from external considerations such as regulatory and stakeholder requirements as well as the nature, characteristics and quantities of the inventory for disposal. For the purpose of the generic DSSC, illustrative disposal concepts based on international examples have been developed for a range of possible host rocks. Illustrative designs have been developed based on these concepts, to meet the requirements in the specification, and assessed for their safety and environmental impacts. The findings of the assessments iteratively inform subsequent development of the disposal system specification and illustrative designs and to identify where further research and development is required. This approach is illustrated in Figure 22–6.

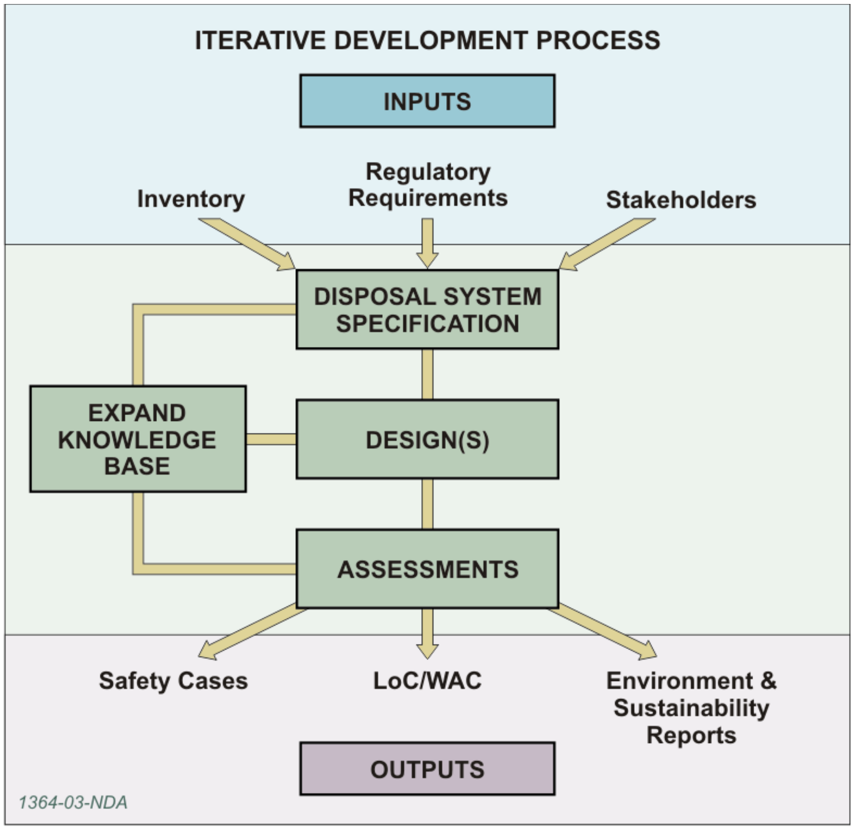

Figure 22–6. Iterative disposal system development

22.5.1.1 *The Science and Technology plan*

A fundamental part of implementing geological disposal is the development of the knowledge base. The knowledge base is the primary means of addressing the needs of the disposal system specification, design development and safety assessments. A Science and Technology Program (RWM 2015a) sets out the needs-driven approach and describes the evolution of the work program as the siting process progresses. It describes a number of key deliverables.

The Science and Technology Plan, S&T Plan; (RWM 2015b) presents an analysis of the nature and timing of RWM's future generic research and development activities. Specific packages of technical work required for the key deliverables are described. Although primarily an internal document used for planning research priorities, it is intended to also provide opportunities for dialogue and involvement of interested parties in the development of RWM's knowledge base. A horizon of a decade is covered by the S&T Plan, during which the vast majority of planned generic research will be completed, with the associated reduction in knowledge gaps. Without an identified site for the GDF, its scope is constrained to those activities that can be conducted at this generic stage. However, the identification of a specific geological environment will provide opportunities for optimization of the S&T plan.

22.5.1.2 *The generic DSSC*

The generic DSSC is part of RWM's work to implement geological disposal. It brings together the disposal system specification, illustrative designs, safety assessments and underpinning knowledge base into a single suite of documents. Its purpose is to describe and assess the safety and environmental implications associated with all aspects of geological disposal of higher activity wastes in the UK. At this early stage, the generic DSSC covers a range of possible host geological environments and illustrative designs. The intent is to develop the DSSC progressively as the plans and designs for the GDF are developed. The first version was produced in 2010 and an update is in preparation for publication in time to support the siting process.

The generic DSSC addresses three main areas:

- The safety of radioactive waste transport to the GDF—the generic Transport Safety Case (NDA 2010b)
- The safety of the design, construction and operation of the GDF—the generic Operational Safety Case (NDA 2010c)
- The protection of the surrounding environment during construction and operation of the GDF and in the very long-term, after the GDF has been sealed and closed—the generic Environmental Safety Case (NDA 2010d)

22.5.2 Building Confidence in Safety

As no site has yet been identified for the GDF, the host geological environment is not known. In addition, the wide range of potentially suitable geologies in the UK means that there is significant uncertainty in this respect. In order to prepare for GDF implementation, RWM has therefore selected and examined a wide range of potentially relevant disposal concepts from published disposal concepts developed internationally. In doing so, a well-informed assessment of options can be carried out at appropriate decision points in the implementation program. For the purpose of the current design and assessment work, a smaller number of illustrative disposal concepts (NDA 2010e). has been defined for three broad types of host rocks present in the UK.

All of the disposal concepts developed by waste management organizations around the world are based on a multi-barrier approach to isolate and contain the wastes, the nature of the barriers depending on the geological environment and the type of wastes to be disposed of.

A disposal concept is specific to a waste category (either low or high heat generating waste) and host rock, and hence the GDF may incorporate more than one disposal concept.

The three broad types of host rocks considered are:

- **Higher strength rocks**—may be igneous, metamorphic or older sedimentary rocks, have a low matrix porosity and low permeability, with the majority of any groundwater movement confined to fractures within the rock mass.
- **Lower strength sedimentary rocks**—fine-grained, sedimentary rocks with a high content of clay minerals that provide low permeability and are mechanically weak, so that open fractures are not sustained. They will be interlayered with other sedimentary rock types.
- **Evaporite rocks**—these formed when water evaporated from ancient seas and lakes, and often contain bodies of halite (rock salt) that provide a suitably dry environment. They are weak and creep easily so that open cracks are not sustained.

The illustrative disposal concepts provide an assessment basis for the generic DSSC and RWM's studies of the environmental, social and economic impacts of a GDF, and for disposability assessments of waste packages proposed by waste producers.

The international concepts in Table 22–1 have been selected as the basis of RWM's illustrative disposal concepts for both low and high heat generating waste, for each of the three host rocks considered.

Table 22–1. International disposal concepts selected as the basis for RWM's illustrative disposal concepts

Host Rock	International Disposal Concept *(Developer, Country)*	
	LHGW	**HHGW**
Higher strength rock	UK LHGW Concept *(RWM, UK)*	KBS-3V Concept *(SKB, Sweden)*
Lower strength sedimentary rock	Opalinus Clay Concept *(Nagra, Switzerland)*	Opalinus Clay Concept *(Nagra, Switzerland)*
Evaporite rock	WIPP Bedded Salt Concept *(US DOE, USA)*	Gorleben Salt Dome Concept *(DBE Technology, Germany)*

The illustrative disposal concepts for a higher strength rock are depicted schematically in Figures 22–7 and 22–8. In this concept, low heat generating waste packages are stacked in vaults that are backfilled on closure. For high heat generating waste, disposal units are lowered into deposition holes and surrounded by bentonite, and backfilled following deposition. Illustrative disposal concepts for a lower strength sedimentary rock and for evaporite rock are given in NDA (2010f).

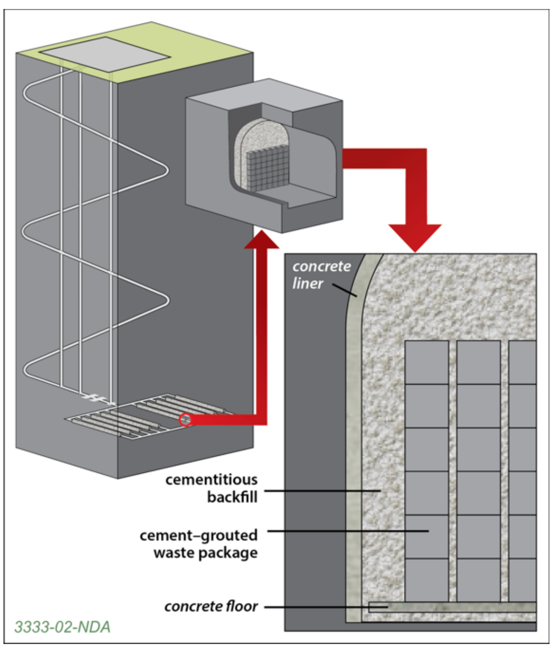

Figure 1 Illustrative disposal concept for low heat generating waste in a higher strength rock

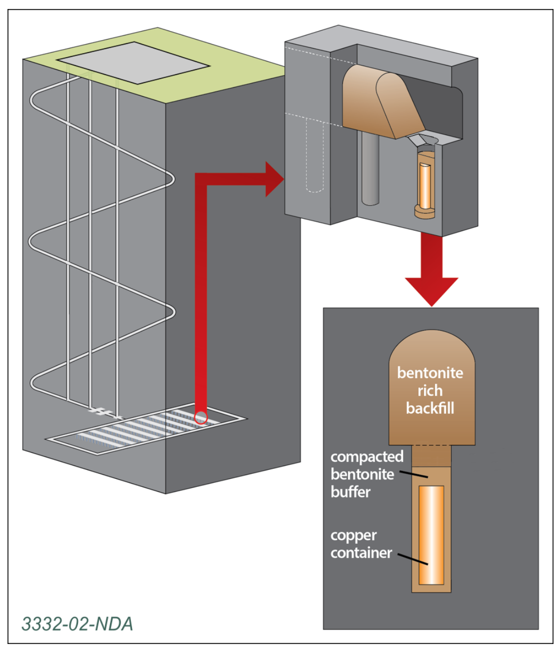

Figure 2 Illustrative disposal concept for high heat generating waste in a higher strength rock

environment environment

Through the iterative development of the disposal system, the illustrative disposal concepts also enable RWM to further develop its understanding of the requirements on the disposal system, develop and prioritize its research program and underpin analyses of the potential cost of geological disposal.

22.5.3 Strategy for Moving from Generic to Site-specific Stage

As the siting process moves forward, details of site-specific geological environments and other characteristics will become available and RWM will start to develop site-specific safety cases. These will be based on specific designs proposed for specific sites. Safety cases for licensing will follow the established process for nuclear safety case development. The Environmental Safety Case will be developed based on site-specific understanding from site characterization activities and in an iterative way that is informed by, and informs, the evolving site characterization process.

The site-specific safety cases will be developed as a separate and parallel work stream to the generic safety cases. This strategy ensures that RWM has accepted, benchmark safety cases in place, whilst developing the site-specific ones.

One specific purpose of the site-specific safety cases, which differs from the purposes of the current, generic DSSC, will be to support the development of the regulatory submissions required at various stages of the implementation program. Figure 22–9 shows at a high level how future safety cases will support permitting in the generic and site-specific parallel work streams.

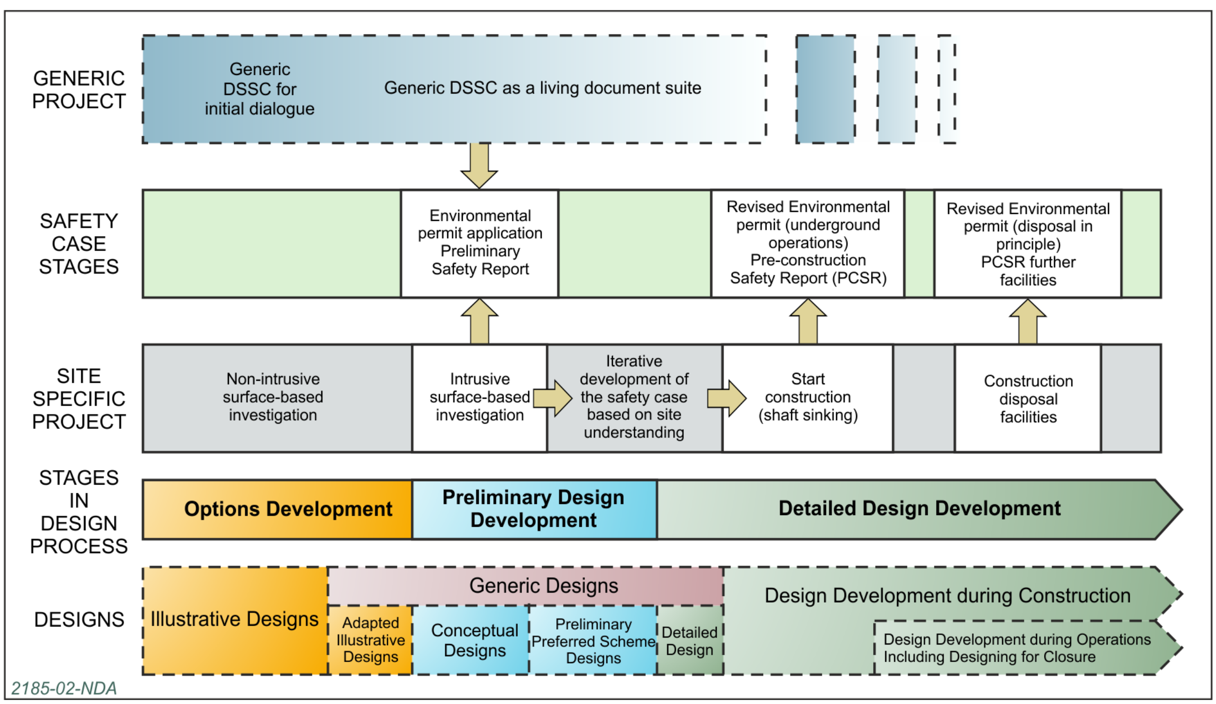

Figure 22–9. Safety case stages.

22.6 Working with the Regulators

RWM will need to demonstrate to regulators that the inventory for disposal can be safely disposed of in the GDF and that the GDF will protect people and the environment at the time of disposal and in the future, following closure. The generic DSSC serves that purpose, as far as appropriate to the current generic stage.

RWM will implement management arrangements for the control of key activities and demonstrate to the nuclear regulators, principally the Office for Nuclear Regulation (ONR), the Environment Agency (EA) and National Resources Wales, that it has the necessary capability, organization, resources and arrangements to undertake these activities safely and securely.

A future GDF is expected to be a nuclear installation under the Nuclear Installations Act 1965 (as amended; Stationery Office 1965) and as such it will be ONR's role, if it is satisfied with the safety of what is proposed, to grant a license for the site and then to enforce the requirements of that license. ONR will also be responsible for regulating security arrangements and granting approval for transport of packages of higher hazard radioactive materials.

The EA will, if satisfied, grant environmental permits to carry out site investigation and to construct the GDF. The EA will require RWM to demonstrate that the location and design of the GDF ensures environmental safety of the facility.

ONR and EA have no formal regulatory role in selecting a site for geological disposal, but they will help the process by advising and commenting on safety, transport and environmental matters that will become important when their formal roles begin; that is, when RWM applies for licenses or permits. The regulators also provide advice and comment to Government, the NDA, RWM, local authorities and other stakeholders on geological disposal.

Both ONR and EA have established agreements with RWM to review and scrutinize its work towards implementing geological disposal and developing the GDF. In 2011, the ONR and EA performed a joint scrutiny of the 2010 generic DSSC, and subsequently stated that they believed it to be fit for purpose. The scrutiny helps RWM progress implementation of the GDF and the associated safety cases, and will inform the preparation of any necessary applications for licenses or permits. The work will also inform the regulators' decision-making throughout this process.

22.7 Key Challenges

The science and technology underpinning geological disposal and its implementation in the UK is well understood and research is at a mature stage. The generic DSSC has demonstrated that a GDF can be constructed and operated safety and in accordance with regulatory requirements, in a range of geological environments found within the UK.

The main challenge now is for RWM to successfully deliver the siting process. Taking a proactive lead in the process is a new role for RWM. There is however considerable overseas experience and RWM is engaging with other waste management organizations worldwide that have successful experience of leading a volunteer siting process.

RWM is developing its organization to face this challenge and has created new siting and communications teams within its structure. It has an organizational and business development program to equip it to be ready for its new role in 2017.

22.8 Next Steps

RWM as an organization will continue to evolve towards holding environmental permits and a nuclear site license as the operator of the GDF. For a major project such as this, the requirements for the organization change as the project progresses through distinct phases. The general principle however is that RWM will remain a lean, intelligent customer organization using the supply chain to provide the different skills and resources required as the nature of the project changes.

During the preparation phase the technically focused activities will be integrated with external activities focused on engaging with potential host communities. As potential sites are identified, RWM will enlist a management contractor that will instigate surface based investigations. Further organizational changes will be necessary as GDF construction proceeds and again when operations commence.

The iterative maintenance of the generic DSSC will continue; findings from the safety assessments will feed back into the design, and inform the science and technology work program. Changing requirements, for example from stakeholder or regulators, will be captured in the disposal system specification, and designs updated accordingly and reassessed. RWM will continue to interact with the regulators, give disposability advice on waste packaging proposals and engage with other stakeholders.

As potential sites are identified, site-specific safety cases will be developed to inform site assessment, optimization studies and regulatory submissions. The site-specific safety cases will be developed as a separate and parallel work stream to the generic safety cases rather than evolve the generic safety cases into site-specific ones. Eventually site-specific development and the safety cases will reach a sufficiently advanced state and be wide enough in scope for the generic safety cases to be no longer necessary and all ongoing safety work will be site-specific.

22.9 Acknowledgments

The author would like to acknowledge the significant contribution from Ann Rostern of AREVA Risk Management Consulting Ltd. in producing this paper.

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22.11 Acronyms

BGS—British Geological Survey

CoRWM—Committee on Radioactive Waste Management

DSSC—Disposal System Safety Case

EA—Environment Agency

GDF—Geological Disposal Facility

HAW—Higher Activity Waste

HHGW—High Heat Generating Waste

HLW—High Level Waste

ILW—Intermediate Level Waste

LHGW—Low Heat Generating Waste

LLW—Low Level Waste

NDA—Nuclear Decommissioning Authority

ONR—Office of Nuclear Regulation

RWM—Radioactive Waste Management Ltd

S&T Plan—Science and Technology Plan

Chapter 23

Geological Disposal of Radioactive Waste in Ukraine: Background, Status, and Future Steps

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ABSTRACT: A description of the current state of a geological repository development in Ukraine is given in this chapter. The description covers the following areas: national background, progress in site selection, development of the concept for geological disposal and its safety case development. Three promising areas (Zhovtneva, Veresnia and Novosilky) for waste geological disposal were identified within crystalline rocks of the Chernobyl Exclusion Zone and in adjacent territories. The advantages of a new waste classification along with the implementation of concepts of disposal of intermediate level waste in a mined repository at the intermediate depth, and the disposal of spent fuel and vitrified high-level waste in a deep geological repository of borehole type are presented in this Chapter.

23.1 Introduction

This Chapter describes the progress made in Ukraine in the field of radioactive waste geological disposal since the Third World Wide Review published in 2001. This report summarizes the following types of data:

- Legislative and institutional framework of activities for radioactive waste disposal
- Recent legislative and regulatory actions, which include the plans for creation of a geological repository, safety requirements and waste classification, and decisions about site selection, etc.
- Radioactive waste inventory lists subject to geological disposal
- History of developing the concepts for geological disposal in Ukraine
- Current status of site selection, development of the concept for geological disposal and its safety case development

23.2 Legislative and Institutional Framework

23.2.1 State policy of waste management

The major principles of state policy in the field of radioactive waste (RAW) management are stated in the laws of Ukraine (Law of Ukraine 1995a; Law of Ukraine 1995b). They are as follows:

- Priority to protection of personnel, population, and environment against the impact of RAW
- Regulatory control of RAW management
- Separation of the regulatory functions and RAW management functions
- Separation of the public administration in the field of nuclear energy use and RAW disposal
- Responsibility of radioactive waste producers for safety of RAW
- Decisions on siting of new radioactive waste management facilities, taking public opinion into account
- Prohibition of RAW transfer to Ukraine for storage or disposal
- Prohibition of RAW disposal by its producer

23.2.2 Main players and their responsibilities

In Ukraine the competencies and responsibilities for RAW management (including geological disposal) are distributed at the national level as follows:

- *Ukrainian Parliament (Verkhovna Rada):* adoption of laws in the area of nuclear energy utilization and waste management including siting for nuclear facilities and RAW management facilities of a national importance.
- *Cabinet Ministers:* Decision making on construction, designing, and siting of RAW management facilities, except those under the competence of Parliament.
- *State Nuclear Regulatory Inspectorate (SNRI):* Establishing normative criteria for radiation safety, issuing licenses for RAW management activity on, supervision over compliance with regulatory requirements and conditions of license, including coercive actions.
- *Ministry of Energy and Coal Industry (MECI):* National administration of RAW management at Ukrainian nuclear power plants (NPPs). Practical management of waste arising during NPP operation is implemented by the state-owned National Energy Generating Company "Energoatom" (NEGC "Energoatom").
- *State Agency for Management of the Exclusion Zone (SAMEZ):* National administration of RAW management, including of activity on construction and operation of RAW disposal facilities.

Practical siting or design activities are realized by the State Specialized Companies (SSCs), which are charged with ensuring radiation safety and bear responsibility for any nuclear damage. Currently, the responsibility for siting, design and construction, on the one hand, and the responsibility for operating and closing of a radioactive waste disposal facility, on the other hand, are divided between various SSCs.

Research and development activities (R&D) are mainly performed by the National Academy of Sciences of Ukraine, Ukrainian Geological Survey, and Scientific Research Centers of MECI and SAMEZ.

23.2.3 Main strategic documents

Action plans to create a deep geological repository (DGR) in Ukraine are defined in two documents: *State Goal-Oriented Ecological Program for Radioactive Waste Management* (Law of Ukraine 2008) and the *Strategy for Radioactive Waste Management in Ukraine* (Strategy 2009).

The *State Strategy* defines the milestones and tasks to create a complete and self-sufficient Ukraine infrastructure for disposal of all RAW arising from nuclear power use. The strategy covers all existing accumulated waste, as well as the streams that will arise in the future, taking into account existing reactors operation life extension, reactors decommissioning, plans for construction of new units, and future use of ionizing radiation sources in industry, medicine, and science.

According to Strategy (2009), a geological repository in Ukraine should be commissioned by 2048. Figure 23–1 shows the tentative schedule for development of a geological repository in Ukraine, which was the basis for the *State Strategy*.

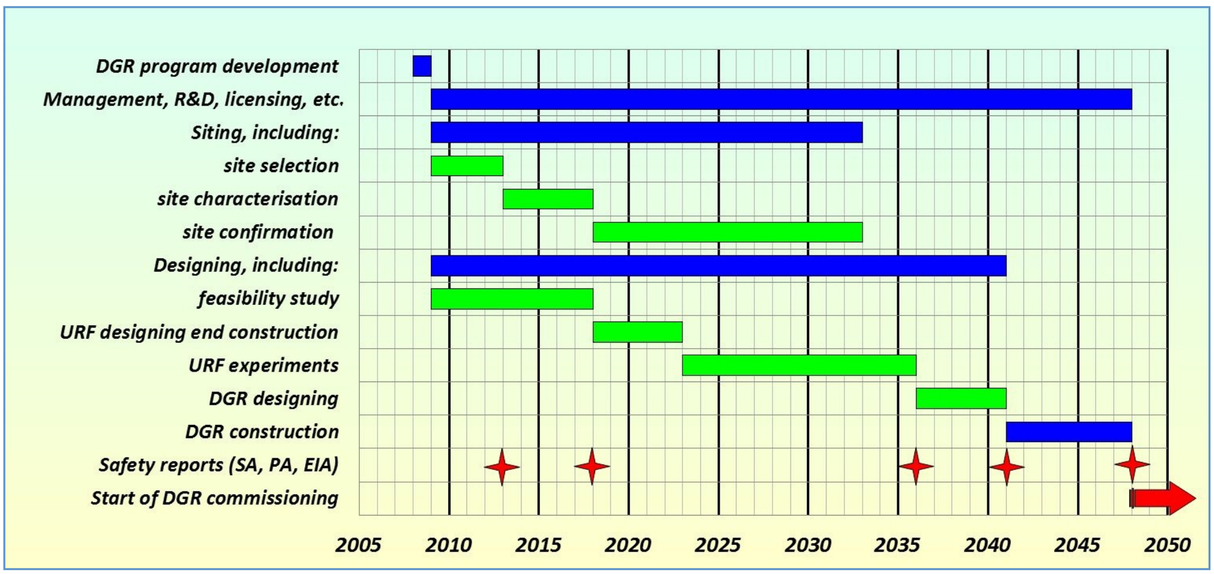

Figure 23–1. Tentative schedule for creation of a geological repository in Ukraine.

The *State Program* details the activities for the first phase of *State Strategy* and defines the technical policy in the field of RAW management for 2008-2017. The key task of this program for development of a geological repository is the implementation of research to evaluate and recommend a site for a deep geological repository. This research program was developed according to Requirements (2008), and includes development and implementation of:

- An R&D program to develop and select methods of field geological research, technologies for creation of engineering barriers and waste packaging, and methodology and software tools to perform safety assessments
- Development of criteria for site selection
- Conceptual design of the geological repository
- Databases for management and archiving of the data obtained
- Preliminary geophysical, geological, and hydrogeological research on a regional scale to select at least three sites for detailed study

- Detailed field studies and drilling program at three sites to obtain data to evaluate the safety of geological disposal system
- Safety assessments and development of safety case for the three sites and for the developed concept of geological repository
- A special Law of Ukraine on the geological repository placement

It should be noted that the present *State Program* is the fourth program on RAW management, which has operated in Ukraine since independence. The Second (1999) and the Third (2002) *State Programs* explicitly stated that the site for the geological repository must be selected within the Chernobyl Exclusion Zone (ChEZ), or in adjacent areas, but the present State Program does not include such a statement.

23.2.4 Financing of radioactive waste management

According to Law (1995b), activities on RAW management, which are defined in the *State Program* and include geological repository development, are financed by the State Fund. The mechanism for accumulating costs for the State Fund is based on the principle of "who contaminates must pay." In accordance with the adopted regulations, all enterprises and organizations in the territory of Ukraine whose activities result in, or may result in, RAW must pay fees to the Fund and receive guaranties that the State will ensure further safe management of the RAW generated, including its final permanent disposal.

23.2.5 General safety requirements

General safety requirements for the geological disposal and requirements for site selection for a repository are defined by two regulations: *General Provisions for Ensuring Safety of Radioactive Waste Disposal into Geological Disposal Facilities* (General Provisions 2007) and *Requirements for Site Selection for the Facility for Radioactive Waste Disposal* (Requirements 2008).

The *General Provisions* establish the basic criteria and requirements for safety of RAW geological disposal at all stages of the life cycle of a geological repository: concept development, siting, design, construction, operation, closure, and post-closure. The *General Provisions* describe the process as follows:

Selecting a site for the geological disposal facility is carried out in stages. These stages are: repository concept development, regional studies or screening; site characterization, and confirmation. The selection is made by comparing the conditions of at least three sites.

The geological repository safety evaluation is performed at different stages of the repository life cycle and is the basis for issuing a license for operation, making decisions regarding siting, construction, closure of the repository, etc.

In the course of performing the repository safety assessments for the distant future (10,000 years after its closure), the use of indirect safety indicators is allowed (for example, concentration of radionuclides, or the speed of their migration).

The *Requirements* establish the technical and organizational requirements for site selection for the surface and deep geological repositories for radioactive waste. These requirements also apply to safety re-assessment of the existing repositories. The *Requirements* comprise:

- Criteria for the site eligibility for repository placement
- Requirements for establishing the site selection criteria

- Preliminary list of site characteristics (geological, hydrogeological, geochemical, socio-demographic, etc.) which need to be studied for the site selection, as well as recommendations for changing this list
- Recommendations for the development of the safety case

23.2.6 Waste classification

All radioactive waste is classified according to several classification criteria: types, categories, and kinds (BSRU 2005; SUP 2000). Table 23–1 presents criteria for short- and long-lived waste types, and Table 23–2 presents the classification of the solid waste categories (LLW, ILW and HLW) according to the specific activity or dose rates. The aggregate state for kinds: solid and liquid waste

Table 23–1. Current waste classification by the criteria of allowed disposal option

Waste type	Potential radiation dose rate in 300 years after the waste disposal	Type of clearance from regulatory control in 300 years after the waste disposal	Allowed waste disposal facility
Short-lived	Below the radiation zone 1 $mSv \cdot yr^{-1}$	Complete, limited	Surface or near-surface
Long-lived	Above the level of 50 $mSv \cdot yr^{-1}$	Not considered	In stable deep geological formations

Table 23–2. Current classification of solid waste by dose rate

Waste category	Dose rate, $\mu Gy \cdot h^{-1}$
Low level waste (LLW)	1 – 100
Intermediate level waste (ILW)	100 – 10,000
High-level waste (HLW)	> 10,000

The current Ukrainian waste classification focuses mainly on ensuring radiation safety for workers and the population during waste management (collection, sorting, transport and storage) during the pre-closure phase of nuclear facilities, and the current classification is not suited for the disposal of all streams of existing waste. It allows the use of only two types of repositories: (1) near-surface repositories for short-lived waste, and (2) deep geological repositories for long-lived waste. According to the current classification, most of the ‘accident waste’of Chernobyl origin is to be disposed of in a DGR (see also Section 3).

A new classification scheme (NCS) for RAW was developed in 2011–2012 within the framework of the EC project INSC U.04.01/08-C, “Improvement of the Radwaste Classification System in Ukraine.” The project was carried out by a consortium consisting of DBE TECHNOLOGY GmbH (Germany, consortium leader), SKB International AB (Sweden), ANDRA (France), COVRA (Netherlands), and ENRESA (Spain). The consortium was assisted by Radioenvironmental Centre of National Academy of Sciences of Ukraine.

The new classification scheme (Table 23–3) is based on final disposal options for different waste classes and complies completely with the IAEA recommendations (IAEA 2009). The new scheme intends to use four types of repositories for RAW disposal. This will significantly reduce the cost of RAW disposal in Ukraine.

Despite the large amounts of 'accident waste', it is not appropriate to define it as a special class. Sorting such waste into classes should be determined only by the waste characteristics, not by the origin. 'Accident waste' contains higher amounts of long-lived nuclides than 'ordinary' waste. This constitutes the main problem associated with its disposal.

The problem can be solved by applying special, less restrictive Waste Acceptance Criteria (WACs) for repositories for VLLW and LLW disposal of the 'accident waste' in the Chernobyl Exclusion Zone (ChEZ). These less restrictive WACs for waste of Chernobyl origin, which is already located in the ChEZ, are based on the limited access to the ChEZ. These special WAC should be applied only for the 'accident waste.' The general WACs, by contrast, are based on estimates of the radiation exposure to critical groups of population living outside of the Exclusion Zone.

23.3 Ukrainian Inventory of RAW Considered for Geological Disposal

Currently, 15 water-cooled, water-moderated (WWER) reactors (2 of WWER-440 and 13 of WWER-1000) are being operated at four Ukrainian NPPs. Four RBMK reactors were operated at the Chernobyl NPP, one of which was destroyed during the accident in 1986. The total capacity of the operating units is 13.8 GW. They generate approximately $90 \cdot 10^9$ kW·h of electricity annually or up to 50% of the total electricity in Ukraine. Construction of new reactors is planned, as well as extending the life of existing ones. It is expected that more than 15,000 tHM of spent nuclear fuel (SNF) and almost 200,000 m^3 of waste will be generated by the existing NPPs, and that these quantities will increase in the future, depending on construction of new reactors.

Currently in Ukraine a part of SNF (reactor WWER-440 and WWER-1000) is sent for reprocessing to Russian Federation. In the near future, vitrified HLW from the reprocessing of SNF from WWER-440 reactors will arrive in Ukraine from the Russian Federation. A part of SNF, which remains in Ukraine, is not reprocessed and is not considered as radioactive waste, pending approval of a final decision on either reprocessing or disposal.

Mining and processing of uranium ore are performed in Ukraine, and as a result approximately 65 million m^3 of waste has accumulated. This waste is not declared as radioactive waste according to the Joint Convention on Safety of Spent Fuel Management and on Safety of Radioactive Waste Management (JC 2008).

Six plants of the Ukrainian State Association, "Radon" (UkrSA "Radon"), deal with the collection, transport, storage, and disposal of RAW from Ukrainian enterprises and medical and research institutions (including sealed radiation sources). There are two research reactors in Ukraine.

The accident at the Chernobyl NPP contributed approximately 3.3 million m^3 to the total amount of waste. The accident waste is located mostly within the Chernobyl Exclusion Zone (ChEZ).

Data on the volume of RAW in Ukraine are summarized in Figure 23–2 on the basis of JC (2008) and Shestopalov (2013). These estimates assume that the lifetimes of existing reactors will be increased from 30 to 50 years, but they do not take into account the possible construction of new reactors. It should be noted that the data on the volume of RAW in Ukraine are periodically updated and refined in the course of preparation of national reports for the Joint Convention.

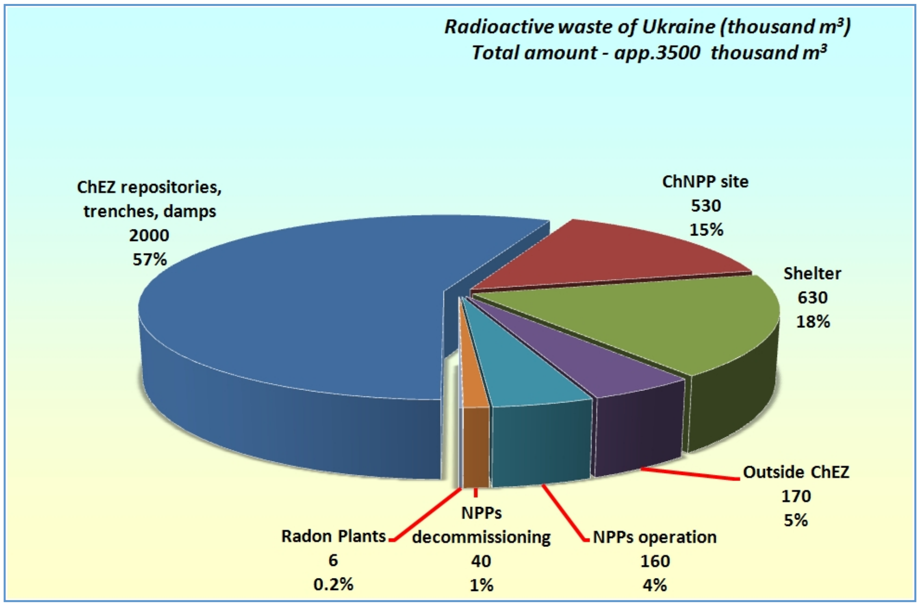

Figure 23–2. The total amount of radioactive waste of Ukraine.

Identification of RAW streams and volumes to be disposed in a deep geological repository taking into consideration the proposals on the new classification scheme (Table 23–3) were made within the framework of the European Commission Projects: TACIS U4.03/04 (TACIS 2008) and INSC U.04.01/08-C. The resulting estimates are shown in Figure 23–3. Wastes that are subject to obligatory geological disposal are classified as HLW according to the current classification. According to the new classification scheme (Table 23–3) they are classified as HLW and ILW. It should be noted that according to our estimates, the strict application of the current classification norms means that an additional 720,000 m^3 of long-lived waste, mostly of Chernobyl origin, should be disposed in the deep geological repository.

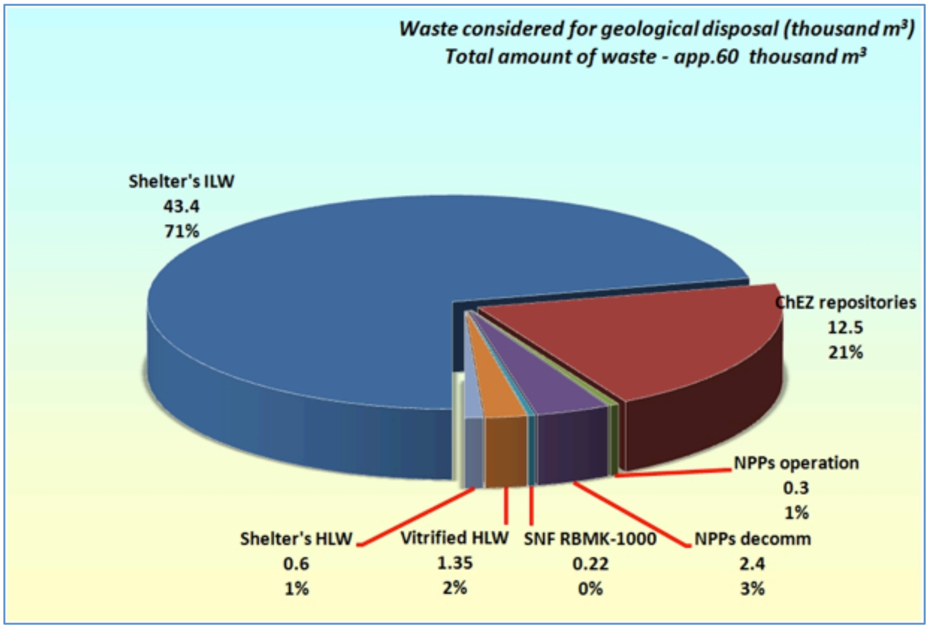

Figure 23–3. Preliminary estimate of HLW volume for geological disposal.

Table 23–3. The new classification scheme proposed for Ukraine

Waste class	Description	Disposal option
Exempt waste (EW)	Exists in Ukraine and the clearance levels are established.	Exempt waste does not need to be managed as radioactive waste
NORM waste	Not determined in Ukraine. Contains only natural nuclides. It is not considered as radioactive waste.	Disposal outside the nuclear legislation (except for separate streams of TE-NORM)
Very Low Level Waste (VLLW)	Not determined in Ukraine. There are large volumes of waste in Ukraine that can be classified as VLLW.	Landfill repositories
Low Level Waste (LLW)	This corresponds to existing short-lived waste.	Surface (near-surface) repositories
Intermediate Level Waste (ILW)	This corresponds to existing long-lived waste.	Intermediate depth repository or deep geological repository
High-level Waste (HLW)	This corresponds to existing heat-generating HLW.	Deep geological repository
Disused Sealed Radiation Sources (DSRS)	Not determined in Ukraine. Due to high risks associated with DSRS, a special class was created.	Separate DSRS may be disposed as LLW or ILW

It should be also noted that at present in Ukraine the above-mentioned consortium is implementing the European Commission Project INSC U.04.01/09-B, “Disposal Concepts for Radioactive Waste in Ukraine.” It is assumed that one of this project’s results will specify the distribution of Ukrainian waste according to the possibilities of their disposal in four types of repositories, as contemplated by the new classification scheme.

23.4 Background of the Activities and Concepts

23.4.1 R&D activities

In Ukraine the activities to develop a deep geological repository have been carried out since 1993. They are performed by the National Academy of Sciences of Ukraine (NASU), universities, State Specialized Enterprises of SAMEZ, the enterprises of the State Geological Survey, and by international organizations and consortia. These activities are funded from the state budget of Ukraine, at the expense of the European Commission and IAEA, and by international organizations' grants (mainly by the Scientific and Technological Center of Ukraine [STCU]). Table 23–4 includes the information on some of the most important (for a geological repository) projects implemented in Ukraine since 2000.

The main focus of current activities continues to be site selection. However, the development of methodology for such a selection is restricted by the fact that the data on the geological evaluation of the

Table 23–4. Some projects executed in Ukraine since 2000

Project	Client	Contractor, year	Type of work*)	Reference
TACIS U4.03/04 *'Strategy of radioactive waste management in Ukraine'*	EC	International Consortium, leader DBE Technology (ICD), 2006-2008	NPM	TACIS, 2008
INSC U.04.01/08-C *'New radwaste classification in Ukraine'*	EC	ICD, 2011-2013	NPM	
INSC U.04.01/09-B *'Disposal concepts for Radwaste in Ukraine'*	EC	ICD, 2013- present	D, SA	
UKR 13374 (the part of IAEA's T21024 project '*The use of numerical models in support of … performance assessment studies of geologic repositories')*	IAEA	Radioecological Center NAS Ukraine (REC), 2006- 2010	SA	IAEA, 2013
UKR/9/026 '*Comprehensive Safety Assessment of Radwaste in Ukraine'*	IAEA	SNRI , SAMEZ, MECI 2007- 2011	SA	
STCU 1396 '*Grounds for Radwaste Disposal in Deep Boreholes'*	STCU	REC, 2001- 2003	S, D, SA	Shestopalov, 2006
STCU 3187 '*Grounds for Radwaste Disposal within Korosten Crystalline Massif'*	STCU	REC, 2005 - 2006	S, SA	
26/19nk-04 '*Development of the concept for geological disposal of long lived waste'*	SAMEZ	SSC 'Technocentre', 2004	NPM	
34/1k-08 '*The investigations program for siting of deep geological repository'*	SAMEZ	REC, 2008	S	
122/13 *'The Development of Screening Methodology for the Siting of Geological Repository'*	SAMEZ	Kiev University, 2013-2014	S	
1050 *'Investigation of Korosten Pluton Concerning its Prospects for Geological Disposal of Radwaste'*	NASU	REC, 2007-2011	S, SA	
'Studying of possibility for radwaste disposal in the mines of Ukraine'	NASU	Institute of Environmental Geochemistry of NASU, 1998-2003	S	Skvortsov, 2003

*) NPM – development of normative, programmatic, and methodological documents

S – Siting, including an assessment of the potential geological formations and locations for placement of a deep geological repository

D – Pre-conceptual design to select an optimal concept of the geological repository

SA – Development of methodology and selection of software for safety assessments site candidates for the geological repository are largely outdated, as these data were collected a few decades ago.

Insufficient attention was paid to developing the concept of a geological repository and its engineered barriers (containers, buffer, and backfill materials, etc.). Therefore, the assessments of the prospects of various regions and areas were carried out on the basis of obsolete geological data, often without taking into account the waste properties and the disposal system concept. At the same time, significant progress in Ukraine has been made in development and application of modern techniques for safety assessments, performance assessment, and development of a safety case for disposal facilities.

In recent years, the main source of funding for R&D activities has been international technical assistance. Funding from the state budget of Ukraine has significantly decreased, and there is no funding at all from the State Fund.

23.4.2 Sites and concepts

Activities on siting the geological repository in Ukraine started in 1993. During 1993–1996 the suitability of geological formations and regions for RAW disposal over the entire Ukraine territory was assessed. Promising formations, and possible areas for repository placement within these formations, were identified.

Since the late 1990s, study has focused on the crystalline formations of the Chernobyl Exclusion Zone (ChEZ) and its environs, as required by the *State Program on Radioactive Waste Management*. In 1997–2000, the regional study of granitoid formations in the borders of Korosten pluton and ChEZ was performed to assess the suitability of this territory for RAW disposal in a mined geological repository. The research was carried out by a desk study of the results of geophysical surveys at scale of 1:200,000, as well as by the results of geological and hydrogeological mapping at scale of 1:200,000 done in the 1960s. One result of this work was selection of two sites within the ChEZ, Veresnia and Tolsty Les, as the most promising for the further study. At the same time, the feasibility of disposal of long-lived and high-level RAW in Ukrainian mines was also studied.

In 2001–2006, within the framework of STCU Projects (see Table 23–4), the scientific grounds were proposed concerning the possibility of creating a borehole-type geological repository in this region. This work led to the preliminary conclusion that Archaean and Proterozoic crystalline rocks of the ChEZ and its vicinity are suitable for allocation of both types of geological repository—mined and deep borehole. These activities were accompanied by certain safety assessments.

Since the end of 2000s and to date an additional limited study of the areas within the Exclusion Zone has been performed, using geophysical data (magnetic and gravity surveys at a scale of 1:50,000), satellite images decoding data, results of hydrogeological modeling, and refinement of new regulatory requirements. The result was the identification of a new area in the ChEZ, Zhovtneva, and replacement of the vast Tolsty Les area by its southwestern part with the new name, Novosilky area (the details are given in the next section). The results of these studies are illustrated in Figure 23–4, and are summarized in Table 23–5.

Figure 23–4. Map of the main prospective formations, areas and sites within the territory of Ukraine. Numbers on the map: within the Ukrainian Crystalline Shield: 1 - Korosten pluton (Proterozoic granitoids); 2 - The Chernobyl Exclusion Zone (crystalline formations of Proterozoic and Archaean); 3 - Iron ore mine Saksagan (Archaean granitoids); 4 - operating uranium mines (Proterozoic crystalline formations); outside the Ukrainian Crystalline Shield: 5 – salt-dome structures of the Dnieper-Donets depression; 6 - bedded salts of Donbas folded structure; 7 - clay formations of the Black Sea depression; 8 - potash salts and clays within the Eastern Carpathian Depression.

Table 23–5. Prospective areas and sites over the territory of Ukraine

Number on map (Figure 23–4)	Description	Period of study	Repository concepts	Safety assessments	Reference
Within Ukrainian Shield					
1	Separate massifs within the Korosten Pluton (Proterozoic granitoids)	1993-2000	Mined	no	Khrushchov, 1996, 2001
2	ChEZ areas within Korosten pluton and the shield slopes (Proterozoic granitoids and Archaean gneisses)	1997 – up to now	Mined and borehole	yes (indirect)	Shestopalov, 2006 IAEA, 2013
3	Iron ore mine Saksagan (Archaean granitoids)	1996-1998	Abandoned mine	yes (direct)	Röhlig, 1998
4	Uranium mines (Proterozoic crystalline formations)	1997-2003	no	no	Skvortsov, 2003
Outside Ukrainian Shield					
5	Salt-dome structures of the Dnieper-Donets depression	1993-2000	Mined	no	Khrushchov, 1996, 2001
6	Permian bedded salts of Donbas folded structure	1993-2000	Existing mine	no	Khrushchov, 1996, 2001
7	Black Sea depression (Paleogene and Neogene clays)	1993-1996	no	no	Khrushchov, 1996
8	Eastern Carpathian depression (potash salts and Neogene clays)	1993-1996	no	no	

In Ukraine, in recent years, the siting activities have been conducted on sites in and near the ChEZ, because the Ukrainians, suffering "radiophobia" after the Chernobyl disaster, does not accept the idea of placing a geological repository anywhere except within the ChEZ. Moreover, the Ukrainian legislative mandates to involve public organizations in decision-making regarding placing a geological repository. Research investigations have shown the feasibility of placing a geological repository in ChEZ crystalline formations:

- Most of the RAW destined for geological disposal are already located in ChEZ; the infrastructure for RAW management has been developed.
- The spent nuclear fuel from RBMK-1000 reactors is stored in the ChEZ. It is planned also to build in the ChEZ a centralized facility for interim storage of spent fuel from WWER-440 and WWER-1000 reactors of Rovno, South-Ukrainian and Khmelnitsky nuclear power plants. This means that if spent fuel is determined to be waste, and a geological repository is located in the ChEZ, the problems of transportation of spent nuclear fuel from storage facilities to the repository will be minimized.

The main results related to siting of geological repository were obtained on the basis of available geological and hydrogeological data, based on the information obtained in the 1960s through 1980s. Surveys and drilling to adequately characterize the subsurface in these areas still need to be completed. Only a very limited field research (magnetic survey, gravitational prospecting and seismic profiling) was carried out in 2005–2006 on Veresnia area within the Project STCU 3187.

23.5 Current Status of the National Geological Disposal Program

23.5.1 Progress in site selection

Research on siting during the last decade in Ukraine has concentrated on the ChEZ (Figure 23–5). Most of the information about the regional geological conditions of the ChEZ, its crystalline basement structures, and its sedimentary cover have been summarized in Shestopalov (2006) and IAEA (2013). The main features of the ChEZ geological structure are described below.

Three major geological structures are distinguished within ChEZ borders. They are: (1) the Korosten Pluton, (2) the east fringe area of the Korosten Pluton, and (3) the Dnieper-Donets Depression. They stretch from SW to NE. The crystalline basement of the ChEZ is dissected by three large fault zones: Pre-Cambrian Teteriv Fault Zone of NE strike, the Prypiat Fault Zone of latitudinal strike, and the Phanerozoic Kyiv Fault zone of NW strike parallel to SW slope of Dnieper-Donets Depression.

The western part of the crystalline basement of the ChEZ is occupied by the Korosten Pluton. Most of the Korosten Pluton is composed by rapakivi and rapakivi-like granites. The intensively dislocated Early Proterozoic metamorphic rocks (Teteriv series), penetrated by granitoids of the Zhytomyr and Osnytsa complex, serve as enclosing strata for Korosten Pluton (composed by Korosten complex). The ages of these complexes are: Zhytomyr: 2020–2080 Ma; Osnytsa: 1980–2010 Ma; and Korosten: 1760–1600 Ma.

Sediments of Paleozoic, Mesozoic, and Cenozoic ages overlay the crystalline rocks of the ChEZ basement (see Figure 23–6). Their thickness increases from west to east, and from 50 to 450 (or more) meters in the vicinity of the Chernobyl nuclear power plant location. Clays and argillaceous rocks make up 40–50% of the total thickness of these sedimentary rocks. The thickness of separate clay layers within the ChEZ does not exceed 30–40 meters.

Ten individual aquifers existing in the sedimentary formation of the ChEZ merge into the following aquifer complexes (depending on the geological age of the water-bearing sediments): Quaternary, Eocene, Cenomanian and Callovian. Low-permeability layers separate them. The groundwater chemical composition is calcium bicarbonate, magnesium-sodium-calcium bicarbonate, sodium-chloride-bicarbonate, and, less frequently, calcium-magnesium sulfate. The total dissolved solids range from 0.2 to 1 $g \cdot l^{-1}$.

Three hydrodynamic zones hypothetically are distinguished in the crystalline basement: (1) zone of fresh groundwater, where the advective transport of contaminants is prevailing; (2) zone of brackish and saline water or transition zone; (3) zone of brine, where contaminant migration is controlled by diffusion. In the Ukrainian shield the last zone is situated below than 2000-2500 m depths.

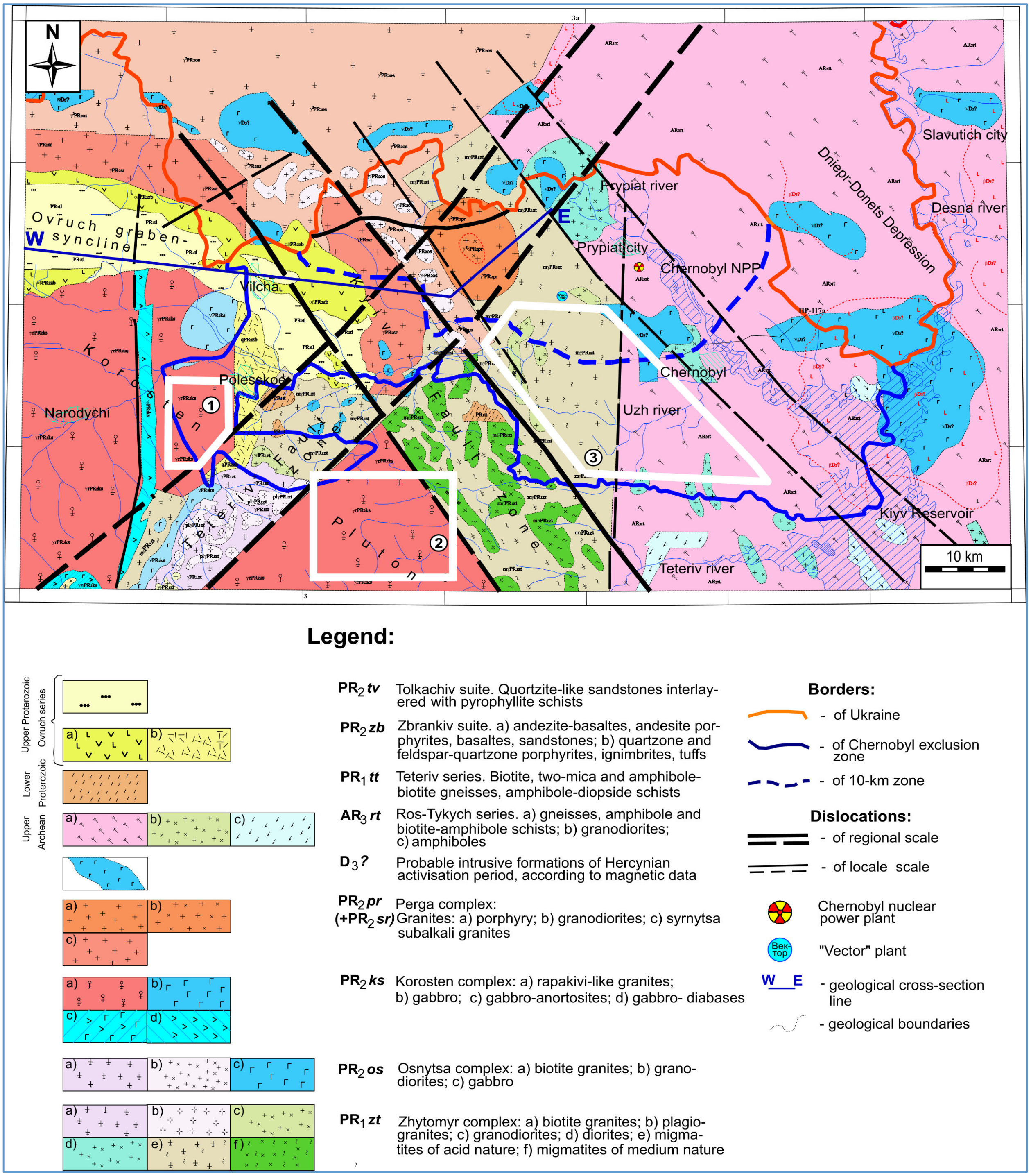

Figure 23–5. The regional geological schematic map of the ChEZ and adjacent territories showing locations of prospective areas: 1 – Zhovtneva, 2 – Veresnia, and 3 – Novosilky.

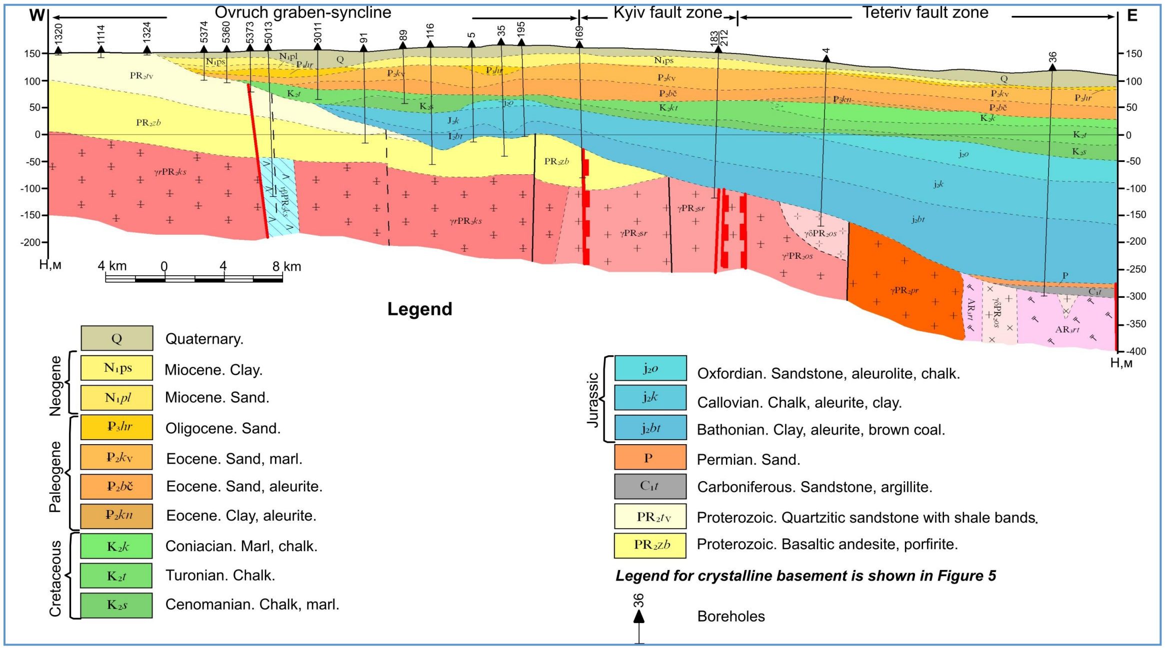

Figure 23–6. Geological cross-section through the ChEZ along the line W-E.

Currently three areas, Zhovtneva, Veresnia, and Novosilky, are identified within the ChEZ and the adjacent territory as the most promising for placing a geological repository (see Table 23–6). These areas were identified based on the following criteria:

- In watershed areas outside the floodplains of major rivers
- Outside the fracture zones and their intersections
- In the areas of minimum density of local faults
- In areas with low gradient geophysical fields

Table 23–6. Typical characteristics of prospective areas

Name of area	Area, km^2	Scale of investigation	Rocks	Sediments thickness, m
Zhovtneva	80	Magnetic and gravimetric survey of scale 1:10,000 - 1:25,000. More than 10 boreholes to the basement Geological mapping scale 1:50,000	Granitoids of Korosten complex	50-100
Veresnia	210	Magnetic and gravimetric survey of scale 1:25,000 - 1:50,000. 2 seismic profiles. Several boreholes to the basement. Geological mapping scale 1: 200,000	Granitoids of Korosten complex	160-200
Novosilky	400	Magnetic and gravimetric survey of scale 1:50,000 - 1: 200,000. Geological mapping scale 1: 200,000	Granitoids of Zhytomyr complex and gneisses of Ros-Tykych series	350-500

The geological and surface characteristics of the Ukrainian sites are similar to Scandinavian sites (Forsmark, Olkiluoto). This relates to age, rock composition (petrographic, mineralogical), degree of tectonic transformation, hydrologic properties of crystalline rocks, and some ecosystem characteristics.

The differences between the Ukrainian and Scandinavian sites are:

- Climatic characteristics (the Ukrainian sites are located in warmer climatic conditions with higher evaporation)
- Presence in the Ukrainian sites of developed sedimentary cover with several water-bearing horizons (Ukrainian sites join the safety function of crystalline and terrigenous formations)
- Groundwater sources and their history (Scandinavian sites are close to the sea coast, in contrast to the continental Ukrainian sites. This determines differences in groundwater salinity and chemical composition)
- Scale of manifestation of the Ice Age effects (Ukrainian areas are located on the periphery of the glacier influence zone).

23.5.2 Choice of repository design

Two possible options of the repository concepts are considered for geological disposal of RAW in Ukrainian conditions: mined and deep borehole. More details can be found in Shestopalov (2005, 2006).

The mined concept assumes the disposal of SNF and long-lived radioactive waste in crystalline rocks at depths of about 500 m. The engineered barriers (thick containers, bentonite buffer and backfill of underground openings) play a key role in the safety of the disposal system, as the waste is placed within the zone, where advective transport of nuclides is dominant. The deep borehole concept assumes placing waste in large diameter boreholes at depths of 3 to 5 km. The safety case is based on the natural barrier functions. At such depths, radionuclide migration is controlled by diffusion processes, and there is less need to rely on engineered barriers.

Figure 23–7 shows that the advantages of the deep borehole concept of DGR are not only lower construction costs, but also shorter construction times and lower risks of human intrusion, etc. However, its disadvantages, such as limited canister dimensions, will not allow for the disposal of all streams of intermediate level and high-level radioactive wastes produced in Ukraine. This pertains to both streams of intermediate level waste contained in Shelter Object and ChEZ repositories, and to those wastes arising from nuclear reactor dismantling. This can be a reason for developing two types of geological repositories in Ukraine: borehole DGR (for disposal of vitrified HLW and SNF) and mined DGR (for disposal of ILW).

It was shown also (Proskura 2015) that the new classification scheme (see Table 23–3) provides significant savings by assigning wastes to optimal repository types. The implementation of the NCS, deciding to dispose of 'accident waste' as VLLW or LLW in the ChEZ, using three repository types (landfill, near-surface, and mined geological repositories) for RAW disposal, and co-disposal of ILW and HLW in a deep geological repository, all will allow a tenfold decrease in total disposal costs (in comparison to disposal of waste using the current classification). If ILW and HLW are disposed of separately in different repositories (mined and borehole), the costs will decrease by 30–40 times (Figure 23–8).

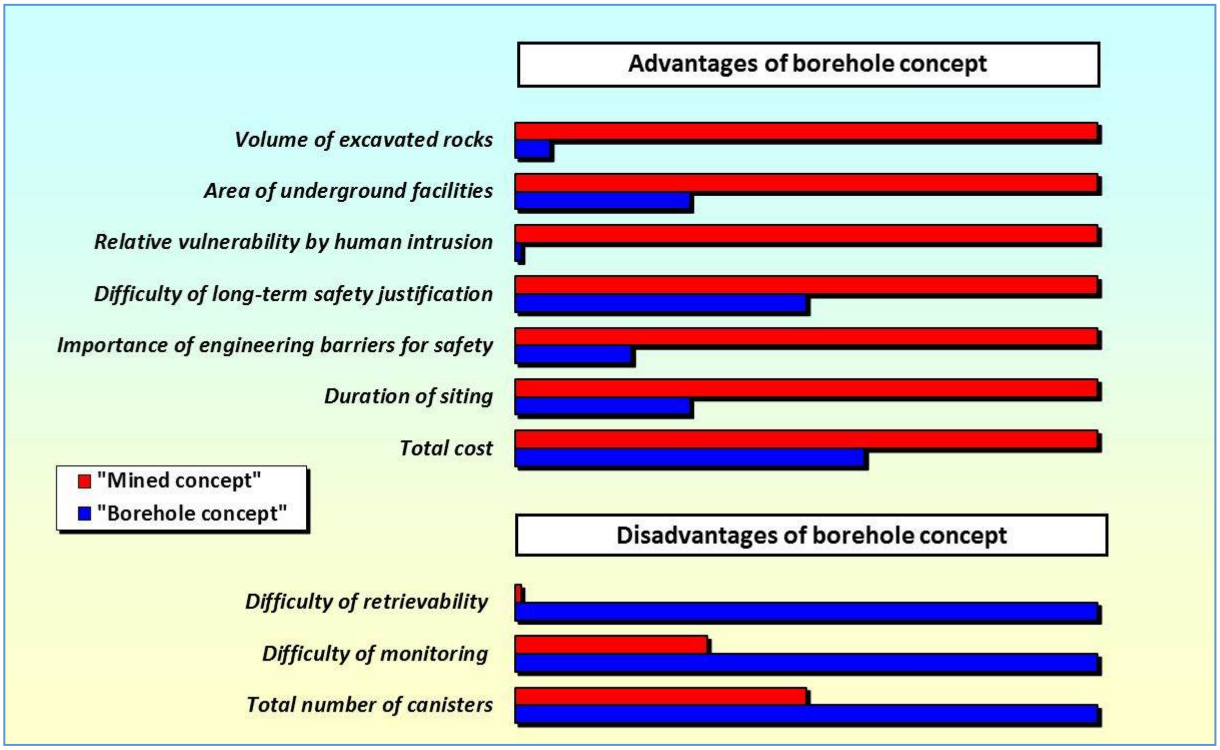

Figure 23–7. Comparison of mined and borehole concepts of deep geological repository.

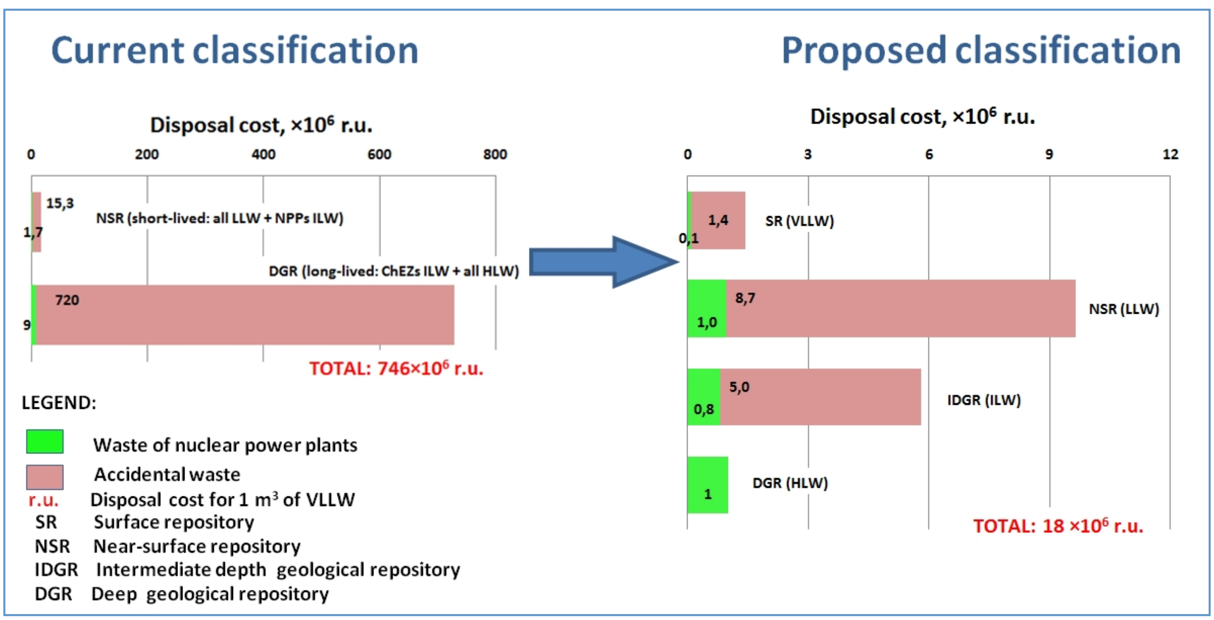

Figure 23–8. Savings which can be provided by implementation of new classification scheme.

It should be stressed that implementation of this approach requires an obligatory preservation of the ChEZ within its present borders. It was shown (Shestopalov 2002) that the main absorber of radionuclides in the ChEZ is surface sandy-clayey deposits of its geological section. Analysis of field data and modeling results showed that the annual amount of radioactivity absorbed by sandy-clayey deposits is much bigger than that carried out by the Pripyat river outside of the ChEZ. As a result, the quality of water at subterranean groundwater intakes in Chernobyl town (located 18 km from the accident site) currently meets drinking water standards.

Based on these considerations, we propose the approach for separate disposal of high-level and intermediate-level (long-lived) radioactive waste in Ukraine, which includes the disposal of the waste streams in both mined and borehole geological repositories (as shown in Figure 23–9). Identified mined repository has capacity of approximately 59,000 m³ of ILW, while identified borehole repository has capacity of approximately 2,200 m³ of HLW and SNF.

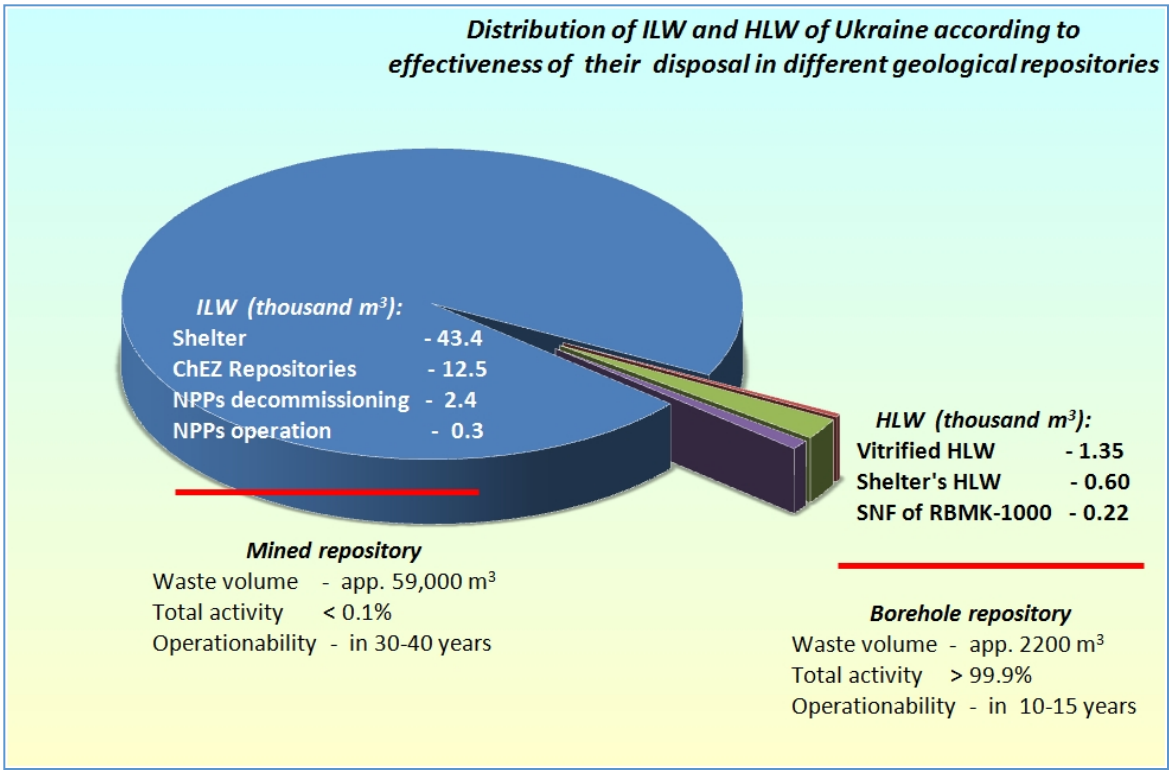

Figure 23–9. Cost effective distribution of Ukrainian high-level and intermediate level (long lived) waste streams between two type geological repositories.

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Preliminary analysis shows that the prospects of using various DGR concepts differ at different sites (Shestopalov 2014):

- The Novosilky area has the thickest sedimentary cover and is the least studied, so this area is more suitable for a deep borehole repository
- The Zhovtneva and Veresnia areas are more suitable for mined repository construction
- The Zhovtneva area is the most suitable for constructing a repository at intermediate depths (here the thickness of the sediments is the smallest)
- The Veresnia area is situated outside the ChEZ, so it should be a secondary choice for the disposal site (in case it is not possible to place geological repositories in ChEZ).

23.5.3 Experience in safety case development

A preliminary 2-D numerical model for predictions of flow and transport within the Veresnia area, accounting for advection, dispersion, and equilibrium sorption processes, was developed based on the PMWIN groundwater modeling system (Chiang and Kinzelbach 2001). Modeling was used to characterize the influence of hydrogeological conditions in upper aquifers on radionuclide migration from a deep geological repository (Shestopalov 2006; IAEA 2013).

The sensitivity study showed that changes in hydraulic conductivity affect both the shallow and deep flow and transport patterns, leading to significant changes in predicted transport behavior due to the effect of a spatial distribution of hydraulic conductivity. Changes in fracture hydraulic conductivity and sorption coefficient, as expected, affect radionuclide concentrations in the river and wells.

The presence of shallow discharge points (such as drainage wells and small rivers), downward infiltration through the unsaturated/vadose zone and groundwater recharge are all favourable factors for the repository's safety. In such conditions, the infiltration flux is mainly intercepted in the upper aquifer layer, and the deeper geological medium remains intact, i.e., independent of slow groundwater flow and advective contaminant transport.

In addition to large variations in the calculated travel times as a result of uncertainty and variability in hydrogeological input parameters, predicted radionuclide fluxes as well as peak and cumulative concentrations also depend on the simulated release scenario, resulting in many-orders-of-magnitude differences in the predicted doses. The uncertainty of predictions results from:

- The lack of clear concepts of waste disposal, in particular, the repository design and the type of a container,
- Limited information on the waste inventory and possible radioactive release scenarios, and
- Limited data about the geological and hydrogeological characteristics of the prospective sites

For these reasons, the results of radionuclide transport modeling cannot be directly used for geological repository safety assessment. Modeling results clearly show the critical importance and the necessity of detailed field investigations of the promising site candidates, and further development of the national concept of geological repository.

23.6 Main Challenges and Conclusions

During the last 15 years in Ukraine, significant progress has been made toward the development of a geological repository. This progress is due to the analysis of f geological data, siting of the promising areas, development of the safety case methodology, and performance of preliminary safety assessments.

The main difficulties impeding the creation of a geological repository in Ukraine can be divided into organizational-financial and technical ones. Largely unresolved organizational-financial problems give rise to technical problems and complicate their solution. Unfortunately, there is no single balanced R&D program which would cover the issues of site selection, disposal technology development, and safety case development. Information about the deep geological structure of the studied area is based on results of surface geophysical studies (seismic, magnetic, and gravimetric surveys), with no drilling through the crystalline rock formation within the candidate sites. The remaining, critical tasks in support of geological repository development are to:

- Implement the new classification scheme (Table 23–3), which allows an economically acceptable solution to the problem of 'accident waste' disposal and geological disposal of high-level and long-lived waste
- Finish the organizational changes and create an Organization for RAW Management, which is responsible for all stages of a geological repository life cycle, and ensure the required level of its funding
- Accomplish the measures foreseen by the *State Program* in the part of geological repository creation.

Current activities in the field of the deep geological repository development in Ukraine are in the early stages of site selection and of pre-conceptual designing.

Three promising areas for waste geological disposal were determined within Chernobyl Exclusion Zone and in adjacent territories. These are the Zhovtneva, Veresnia and Novosilky areas, marked 1, 2 and 3, respectively, in Figure 23–5. Preliminary safety assessments have demonstrated the suitability of crystalline rocks of the ChEZ for geological disposal of RAW, as well as the necessity for detailed field investigations, including drilling, within promising areas.

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23.8 Acronyms

ChEZ—Chernobyl Exclusion Zone

DGR—Deep Geological Repository

HLW—High-level Waste

ICD—International Consortium, leader DBE Technology

ILW—Intermediate Level Waste

LLW—Low Level Waste

MECI—Ministry of Energy and Coal Industry

NASU—National Academy of Sciences of Ukraine

NCS—New Classification Scheme

NPM—Development of normative, programmatic, and methodological documents

R&D—Research & Development

RAW—Radioactive Waste

REC—Radioecological Center NAS Ukraine

SA—Safety Assessment

SAMEZ—State Agency for Management of the Exclusion Zone

SSCs—State Specialized Companies

SNF—Spent Nuclear Fuel

SNRI—State Nuclear Regulatory Inspectorate

STCU—Scientific and Technological Center of Ukraine

VLLW—Very Low Level Waste

WAC—Waste Acceptance Criteria

WWER—Water-cooled, water-moderated energy generating reactor

Chapter 24

Research & Development Program for the Used Nuclear Fuel and High-Level Radioactive Waste Disposition in the United States

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ABSTRACT: This Chapter describes the generic work performed by the United States (U.S.) during the last several years in understanding the nature of performance issues related to various geologic environments for the disposal of used nuclear fuel and high-level radioactive waste materials. In particular, the disposal concepts of mined repositories in salt, clay/shale and granitic rocks, and in deep boreholes in crystalline rock are addressed. The research and development activities evolved from the potential key features, events and processes associated with the respective geologic media and that are important to developing a technically sound safety case. The scope of research activities include participation in selected international repository projects and the publication of key research results from the U.S. program.

24.1 Introduction

The Nuclear Waste Policy Act, as amended (1987), required the U. S. Department of Energy (U.S. DOE) to characterize a site at Yucca Mountain, Nevada, for its potential use as a deep geologic repository. The earlier reports, Third Worldwide Review (Dyer and Voegele 2001) and the Fourth World Review (Arthur and Voegele 2006) addressed the regulatory requirements and the progress toward a license application for the potential geologic repository. However, in 2010, it was determined that developing a repository at Yucca Mountain was not a workable option and the U.S. needed a different solution for spent nuclear fuel (SNF) and high-level waste (HLW) disposal. A Blue Ribbon Commission (BRC) on America's Nuclear Future evaluated alternative approaches for managing spent fuel and HLW from commercial and defense activities. The BRC report contained recommendations for legislative and administrative action to develop a new strategy to manage nuclear waste. Based on the BRC recommendations, DOE published DOE's *Strategy for the Management and Disposal of Used Nuclear Fuel and High-Level Radioactive Waste (January 2013),* containing policy on addressing the disposition of used nuclear fuel (UNF) and HLW. Note that the term "used nuclear fuel" is synonymous to "spent nuclear fuel"; either terminology refers to fuel discharged from a nuclear reactor.

The DOE Office of Nuclear Energy (NE) is responsible for UNF activities. Within NE, the Office of Used Nuclear Fuel Disposition Research and Development (UFD) has developed and is executing a research and development (R&D) program that will address critical scientific and technical issues associated with the long-term management of UNF. This Chapter provides the status of spent fuel and radioactive waste management activities conducted by the UFD campaign, and in particular, disposal R&D activities for deep geologic repositories. These activities include R&D related to the safety implications of extended storage, transport, and disposal of UNF and HLW.

The principal focus of DOE's R&D activities is to develop a suite of options that will enable future decision makers to make informed choices about how best to manage the SNF from reactors and HLW. An additional objective is the demonstration of technologies necessary to allow deployment of solutions for the sustainable management of spent fuel that is safe, economic, and secure.

24.1.1 Storage and Transportation R&D

Most commercial SNF in the U.S. is currently stored at the nuclear power plant (NPP) site where generated. In the absence of a geologic repository for disposal, spent fuel will continue to be stored either in spent fuel pools (SFP) or in dry cask systems, in facilities licensed by U.S. Nuclear Regulatory Commission (NRC). NRC's regulatory framework provides information on the storage of spent fuel for long periods and NRC licensees are responsible for maintaining the continued safe storage of spent fuel during the period of the license.

Licenses for storage of commercial spent fuel are issued by NRC for periods of up to 40 years, and can be renewed for additional terms, under regulations in 10 CFR Part 72. NRC review of applications for storage license renewal includes consideration of time-limited aging management analyses to address potential material degradation processes that could affect performance of the storage system.

Science-based information is being collected to support extended storage and eventual large-scale transport of SNF and HLW for future disposal. Current R&D activities examine topics to identify those technical gaps related to extended storage of spent fuel. Significant testing, modeling, and demonstration activities are being conducted to enhance the technical basis for safe storage and disposal of spent fuel, particularly as spent fuels discharge, burn-up is increased and time of storage extends beyond what was originally licensed. Collaboration with the private sector, international and other organizations are important to develop and implement concepts that ensure safe, secure, and timely storage and disposal of spent fuel and HLW.

24.1.2 Disposal R&D for Spent Fuel and HLW

Additional technical bases for multiple viable deep geologic disposal options enhance confidence in the robustness of generic disposal concepts and facilitate developing the science and engineering tools needed to support disposal concept implementation.

Disposal R&D focuses on (a) identifying multiple viable geologic disposal options and (b) addressing technical challenges for generic disposal concepts in various host media (e.g., mined repositories in salt, clay/shale, and granitic rocks, and deep borehole disposal in crystalline rock). The generic disposal R&D will transition to site-specific challenges as the implementation of the strategy advances. Disposal R&D goals at this stage are to reduce generic sources of uncertainty that may affect the viability of disposal concepts, to increase confidence in the robustness of generic disposal concepts, and to develop the science and engineering tools needed to select, characterize, and ultimately license a repository.

As part of disposal R&D, DOE is actively involved in international and bilateral R&D activities, in order to provide the U.S. with an understanding of the fuel cycle activities of other countries. This involvement allows the participants to leverage their expertise and gain cost benefits in conducting technical assessments for different geologic media and waste forms. The U.S. also collaborates with other countries to conduct joint experiments or data exchanges associated with underground research laboratories (URLs). Active collaboration with international programs, initiatives, or projects is beneficial to the U.S. disposal research program, providing access to decades of experience gained in various disposal environments.

24.1.3 Disposal R&D in the U.S.—Used Fuel Disposition Campaign (UFDC)

The stated mission of the UFDC is

> *To identify alternatives and conduct scientific research and technology development to enable storage, transportation and disposal of used nuclear fuel and wastes generated by existing and future nuclear fuel cycles.*

A comprehensive R&D program investigating a variety of geologic media has not been a part of the U.S. waste management program since the mid-1980s, but now is being executed in the UFDC. DOE is investigating various geologic media and concepts for the disposal of UNF and HLW that exists today and that could be generated under future fuel cycles. The disposal of UNF and HLW in a range of geologic media has also been investigated internationally. Considerable progress has been made in the U.S. and other nations, with further technical bases being established in current programs.

24.2 Goals of the UFDC R&D Program

The UFDC R&D program is designed to ensure that the technical information needed to implement new national policy for managing the back end of the nuclear fuel cycle is available for decision-making. Initially, UFDC focuses on generic research and development work undertaken today that will support future site-specific work. The research and development is focused on finding solutions and building confidence, and reducing uncertainties in the technical basis information related to challenges in nuclear waste repository siting. The UFDC conducts its R&D in collaboration with national laboratories, university, industrial, and international collaborators.

The UFDC approach focuses on identifying knowledge gaps and opportunities where research and development have the greatest potential to contribute to advancing the understanding of technical issues regarding deep geologic disposal of nuclear waste. The research will also help to maintain U.S. expertise in repository sciences, both within the U.S. national laboratories and university system (through the Nuclear Energy University Program). The UFDC collaborates, where appropriate, with other countries that are pursuing the geologic disposal of spent nuclear fuel and HLW. The work is of value in communicating information about repository sciences, building confidence in the technical basis used for repository siting and supporting broad confidence building and education efforts with stakeholders and the public. The research may provide information to address misperceptions about repository science and the siting of nuclear waste repositories. Given the time period involved with geologic disposal, uncertainties may play an important role in public and stakeholder decisions and confidence in moving the nuclear waste repository program forward.

In particular, there is a push to quantify the levels of uncertainty, and to understand how that uncertainty propagates through the technical analyses supporting the safety case for a nuclear waste repository. The information that would be collected through research and development provides a basis to communicate the safety case for a generic repository, or a repository in a particular geologic medium. The UFDC conducts broad R&D related to the entire waste management system to obtain perspective regarding the implementation of future waste management strategies.

24.3 Disposal Environment Options

The UFDC is currently evaluating the viability of mined repositories in three geologic media (salt, clay and crystalline rock), and, in addition, the use of deep boreholes in crystalline rock. For each of these disposal options, the rock type is identified at a broad level. Thus salt includes both bedded and domal salt; clay includes a broad range of fine-grained sedimentary rock types, such as shales, argillites and claystones, as well as soft clays; crystalline rock includes a range of lithological formations, including granite, gneiss, and a variety of other igneous rock types. These disposal options are not presented as a final list of the best possible alternatives, and the DOE recognizes that other options have been identified in the past that also have the potential to provide safe long-term isolation. If other disposal concepts are identified that warrant further investigation, they will be evaluated. There are multiple reasons for focusing on these main concepts at this stage of the program.

First, the U.S. went through an extensive review of all available options for disposal and management during the 1970s, culminating in the 1980 *Environmental Impact Statement on Management and Disposal of Commercially Generated Radioactive Wastes* (DOE/EIS-0046). This review considered a full range of alternatives to mined geologic repositories, including deep boreholes, sub-seabed disposal, space disposal, and ice sheet disposal. Mined repositories were the favored option, but sub-seabed disposal and deep boreholes were retained for further consideration. Technically, sub-seabed disposal remained a promising option, but it was precluded by international treaty in the 1990s. Deep boreholes were considered to require further technological advances, and disposal programs in both the U.S. and other nations focused on mined repositories beginning in early 1970s. The U.S. program evaluated salt, granite, shale, basalt, and volcanic tuff before focusing exclusively on volcanic tuff at Yucca Mountain because of the 1987 Nuclear Waste Policy Amendments Act.

Second, conclusions drawn in the U.S. program in the early and mid-1980s about the potential viability of salt, granite, and clay as disposal media have since been confirmed by extensive work. Crystalline rock is a general term used here to refer to large bodies of igneous intrusive rock and high-grade metamorphic rock regardless of its protolithic type. Examples are predominantly granitic in composition, but other metamorphic and igneous lithology may also be suitable.

Crystalline (granite) repository concepts have been evaluated in Sweden, Finland, Switzerland, and Japan. Clay disposal concepts have been evaluated in France, Belgium, and Switzerland. Salt has been shown to be a viable medium for disposal of non-heat generating transuranic waste at the Waste Isolation Pilot Plant in the U.S., and research in Germany continues to show promise for the disposal of heat-generating waste in salt. Other geologic media are under consideration for specific purposes (e.g., Canada is investigating the use of a mined repository in carbonate rock to dispose of intermediate level waste, and the U.S. has disposed of low-level and transuranic waste in near-surface alluvium).

Third, deep borehole disposal continues to be the primary viable alternative to mined repositories. DOE investigated this concept in the 1990s for the potential disposal of surplus plutonium, and studies have continued at Sheffield University in the United Kingdom, at Massachusetts Institute of Technology (MIT), and at Sandia National Laboratories (SNL) in the U.S.

Scope of UFDC Activities. The UFDC activities performed for the evaluation of deep geologic repositories include work performed under the following R&D topics:

- Crystalline Disposal R&D
- Argillite Disposal R&D
- Salt Disposal R&D
- Deep Borehole R&D
- Generic Disposal System Analysis
- International R&D
- Dual-Purpose Canisters

The design of a generic disposal system includes the evaluation of common features, events and processes (FEPs) that provide the bases for many of the fields of study as illustrated in Figure 24–1. In addition, the participation in international underground research laboratory (URL) programs enhances the overall research value for the UFDC.

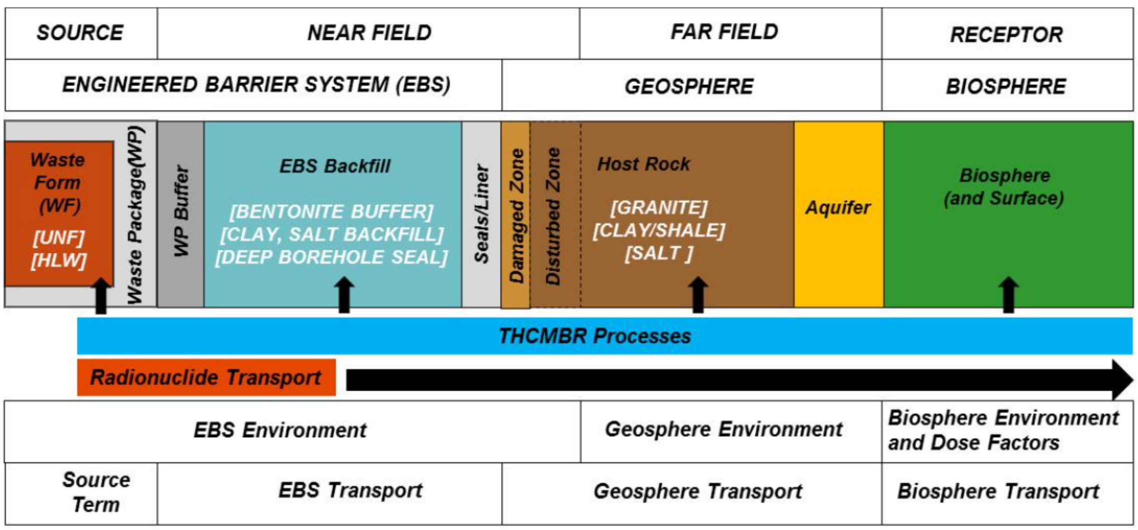

Figure 24–1. Schematic Illustration of generic disposal system components and phenomena (Freeze and Vaughn 2012). Note: THCMBR represents Thermal-Hydrological-Chemical-Mechanical-Biological-Radiological processes

24.3.1 Crystalline Disposal R&D

The objective of Crystalline Disposal R&D is to advance our understanding of long-term disposal of used fuel in crystalline rocks and to develop necessary experimental and computational capabilities to evaluate various disposal concepts in such media. Significant progress has been made in the last few years in both experimental and modeling arenas in evaluation of used fuel disposal in crystalline rocks (Wang et al. 2015). The work covers a wide range of research topics identified in the R&D plan. The major accomplishments include:

- A generic reference case for crystalline disposal media has been established. The reference case specifies the emplacement concept, waste inventory, waste form, waste package, backfill/buffer properties, EBS failure scenarios, host rock properties, and biosphere. This is an important step in developing a baseline for total system model development.

- Developed and applied THMC models to the analysis of coupled EBS processes in bentonite-backfilled repositories. This development was based on the extension of the Barcelona Basic Model (BBM) to a dual-structure model for expansive clay, such as bentonite. This model improvement reduces the modeling uncertainty for THM processes in a bentonite buffer.
- Discrete fracture network (DFN) generation, parallel flow solutions, and particle tracking were demonstrated at application-relevant field scales using fracture parameters from a well-characterized Forsmark SKB site. UFDC has developed and demonstrated the capability to simulate intra-fracture variability within a full-scale DFN network. The ability to produce high-quality computational DFN meshes suitable for state-of-the-art, parallel, subsurface flow codes is a capability that will allow large DFNs to be considered in applications.
- Significant progress has been made to improve reactive transport predictions over long time and distance scales. The methodology includes the use of credible slow desorption rates. The uranium sorption and transport in Grimsel Test Site granodiorite was chosen as a study case, but the method could be applied to any moderately sorbing radionuclide in any hydrogeologic system.
- A study of the dissolution of intrinsic plutonium colloids in the presence of montmorillonite at different temperatures was successfully accomplished using a novel experimental setup containing a dialysis device. This device enables the separation of two solid phases, but let them interact with each other via aqueous plutonium ions.

24.3.2 Argillite Disposal R&D

Radioactive waste disposal in shale/argillite rock formations is attractive because of its desirable isolation properties such as low permeability, geochemically reduced conditions, variations in groundwater pressures, and widespread geologic occurrence. Also, clay/shale rock formations are rich in clay minerals such as smectites and illites that have high capacity to sorb radionuclides, so that transport is mainly by diffusion. Shale's high sorption and low permeability are the key attributes that impede radionuclide mobility.

Shale host-media has been comprehensively studied in international nuclear waste repository programs as part of URL programs in Switzerland, France, Belgium, and Japan. These investigations, in some cases a decade or more long, have produced a large body of fundamental information spanning from site characterization data (geological, hydrogeological, geochemical, geo-mechanical) to controlled experiments on the engineered barrier system (EBS) (barrier clay and seals materials). Evaluation of nuclear waste disposal in shale formations in the U.S. was conducted in the late 70s and mid 80s. Most of these studies evaluated the potential for shale to host a nuclear waste repository but not at the programmatic level of URLs in international repository programs. The R&D focuses work and capabilities relevant to disposal of heat-generating nuclear waste in shale/argillite media. Integration and cross-fertilization of these capabilities are utilized in the development and implementation of the shale/argillite reference case.

Disposal R&D activities under the UFDC in the past few years have produced state-of-the-art modeling capabilities for coupled Thermal-Hydrological-Mechanical-Chemical (THMC) processes, used fuel degradation (source term), and thermodynamic modeling and database development to evaluate generic disposal concepts (Jove-Colon et al. 2014). The THMC models have been developed for shale repository, leveraging in large part on the information garnered in URLs (international programs) and laboratory data to test and demonstrate model prediction capability and to accurately represent behavior of the EBS and the natural [barrier] system (NS). In addition, experimental work to improve our understanding of clay barrier interactions and thermal-mechanical couplings at high temperatures are key to evaluate thermal effects because of relatively high heat loads from waste and the extent of *sacrificial zones* in the EBS. To assess the latter, experiments and modeling approaches have provided important information on the stability and fate of barrier materials under high heat loads. This information is central to the assessment of thermal limits and the implementation of the reference case when constraining EBS properties and the repository layout (e.g., waste package and drift spacing).

R&D activities applicable to shale/argillite media are varied. For example, progress made in modeling and experimental approaches to analyze physical and chemical interactions affecting clay in the EBS, NS, and used nuclear fuel (source term) in support of R&D objectives. Accomplishments include:

- Developed a reference case for shale/argillite to support safety case evaluations
- Investigated reactive transport and coupled THM processes in EBS:
    - Developed and validated constitutive relationships for permeability, porosity and effective stress
    - Developed discrete fracture network (DFN) approach for fractures in argillaceous rock—using rigid-body-spring network (RBSN) approach to model geo-mechanical behavior including fracturing
    - Conducted THM modeling of underground heater experiments from URLs
    - Investigated the maximum allowable temperature and detailed impacts of a high temperature on repository performance
    - Developed transport in clay and clay rock including different modeling approaches and conducting benchmark studies for software codes
- Update on experimental activities on buffer/backfill interactions at elevated pressure and temperature
- Thermodynamic database development: evaluation strategy, modeling tools, first-principles modeling of clay, and sorption database assessment
- Mixed potential model for used fuel degradation: application to argillite and crystalline rock environments

24.3.3 Salt Disposal R&D

Salt R&D comprises several completed and ongoing laboratory studies, testing, and modeling activities relating to thermal, mechanical, hydrologic, and chemical laboratory processes (Kuhlman and Sevougian 2013). The Salt R&D program takes advantage of past and current U.S. and international activities in this area. In addition, U.S. DOE and German scientists work under a memorandum of understanding (MOU) and enable field, laboratory and model benchmarking research.

24.3.3.1 Laboratory studies

Laboratory studies are being performed to enhance the technical bases for disposal of heat-generating waste in salt. Improved data will be developed by experiment for thermal-mechanical behavior of granular salt including; the determination of capillary pressures and transport properties in reconsolidated salt, thermal-mechanical testing of intact salt, and thermodynamic properties of brines, salt, and corrosion products. These laboratory-based experiments are aimed at reducing uncertainties that remain in the technical bases for supporting disposal of heat-generating waste in salt. This scope includes:

- Hot granular salt consolidation, constitutive modeling and micromechanics: this work improves the understanding of the role of elevated temperature and pressure regimes on transport properties of reconsolidated crushed salt.

- Laboratory thermal-mechanical testing: this work includes salt cores in an unconfined condition at a constant axial strain rate and elevated temperatures.
- Study of thermodynamics of brines, minerals and corrosion products at high temperatures: this work studied the of elevated temperature and pressure regimes on chemical properties and chemical interactions of brines, minerals and corrosion products for a salt repository. Data derived from this experimental program will supplement an existing database used to model chemical equilibrium in high ionic strength systems.

24.3.3.2 Modeling studies related to salt

- Brine migration modeling: testing and application of computational models and tools to simulate coupled THM processes and brine migration in salt have been conducted. Incorporated new constitutive models for the damage evolution of salt and new capabilities for simulating large deformations.
- Evaluation of modeling capabilities, constitutive models, and validation against benchmark tests were conducted.

24.3.3.3 Brine migration experimental and modeling studies in run-of-mine salt backfill

- This laboratory testing activity compliments ongoing experimental studies that focus on brine migration in intact salt and has the goal of supporting the planning/design of thermal tests at intermediate and field scales. Existing computational models of THC processes were used to support possible designs of a large-scale thermal test.
- Transport properties of run-of-mine salt and damaged intact salt: this activity will conduct laboratory tests to collect data used to characterize flow and transport parameters in unconsolidated granular salt and damaged intact salt. This activity is a continuation of an ongoing activity. Scoping laboratory tests have begun on reconsolidated run-of-mine samples left over from previous testing. Numerical modeling and model development will be conducted to interpret and support the laboratory testing results.

24.3.3.4 Field test planning

- Framework for underground research: the framework describes a field-testing research strategy that can be successfully developed and deployed, leading to actionable test plans. The Framework document sets up a protocol for selecting activities in a transparent objective forum, and to ensure that infrastructure requirements are adequately established to achieve testing objectives and to avoid interference between testing and other activities.
- Thermal testing: a phased set of in situ thermal tests in bedded salt can support a number of generic safety case objectives. The most important of these is gaining quantitative insight into the larger scale response of brine in the intact, disturbed, and backfill salt to a relatively large horizontal heat source. To gain this quantitative insight the tests must be properly phased and designed to assure that attendant thermal and brine-related parameters and processes can be adequately measured and monitored.
- Mechanical and hydrological behavior of the near-field host rock surrounding excavations: excavations created in a salt present an opportunity to measure and characterize in situ development of the excavation disturbed zone (EDZ). This test will provide data to characterize and quantify the time-dependent mechanical behavior and hydrologic response of

test-room near-field host rock and establish boundary conditions for any test that might be conducted in an excavation.

24.3.3.5 Deep Borehole R&D

The purpose of the Deep Borehole Field Test (DBFT) is to evaluate the feasibility of the Deep Borehole Disposal (DBD) concept via characterizing the salient geologic conditions of a representative site, demonstrating handling and emplacement activities, and evaluating the safety of the DBD concept using data collected in the test. This activity is also associated with the DOE's Subsurface Technology and Engineering Research (SubTER), Development, and Demonstration crosscut research program in subsurface research. Multiple factors have indicated that the deep borehole disposal concept may provide a technically feasible and cost-effective alternative for safe disposal of selected DOE-managed radioactive waste forms. There exist widely available locations with favorable geological and hydrological characteristics for isolation of nuclear material at depth within the geosphere basement complex (Arnold et al. 2013). The implementation of deep borehole disposal, with a simple reference design and operations, could be a feasible and cost effective mode of disposal.

A deep borehole field test would not include emplacement of radioactive wastes but would provide valuable information and data related to deep borehole disposal as well as provide an opportunity to gain insight into crosscutting subsurface challenges (e.g. drilling techniques, wellbore stability, sealing, etc.) across such diverse areas as geothermal energy production, fossil energy production, and carbon sequestration. In addition, there are potential economic and scientific benefits of a deep borehole field test for local, state, and regional stakeholders. In October 2015, the DOE participated in a Nuclear Waste Technical Review Board (NWTRB) International Workshop on Deep Borehole Disposal (http://www.nwtrb.gov/meetings/meetings.html) to discuss the overall DBD concept and present the high-level goals of the DBFT.

24.3.3.6 Deep borehole concept

The deep borehole disposal concept consists of drilling a borehole, or an array of boreholes, into crystalline basement rock to about 5,000 m depth with a bottom-hole diameter of approximately 43 cm (17 inches) and at least 3,000 m of crystalline basement rock in the bottom of the borehole:

- Approximately 400 waste packages would be emplaced in the lower 2,000 m of the borehole
- Upper borehole would be sealed with compacted bentonite clay and cement plugs
- Several factors suggest the disposal concept is viable and safe:
    - Crystalline basement rocks are common in many stable continental regions
    - Existing drilling technology permits dependable construction at acceptable cost

Low permeability and long residence time of high-salinity groundwater in deep continental crystalline basement at many locations suggests very limited interaction with shallow fresh groundwater resources. Figure 24–2 describes the components of a conceptual deep borehole design.

24.3.3.7 Completed activities

2009—Completed the conceptual evaluation and the preliminary performance assessment analyses for deep borehole disposal of high-level waste.

2011—Completed a detailed reference design and operations for the deep borehole disposal concept.

2012—Completed a site characterization methodology for deep borehole disposal with analysis of features, events, and processes relevant to disposal safety, and review of borehole testing and logging methods.

2012—Completed a research, development, and demonstration (RD&D) roadmap for deep borehole disposal. This includes planning for scientific investigations and engineering demonstration for a deep borehole field test, cost estimates and schedule.

2013—Completed technical and logistical guidelines for selecting a field test site, borehole seals research, RD&D needs, and updated the performance assessment analyses.

2014—Completed a DBFT plan. Completed Deep Borehole Disposal Research Geological Data Evaluation, Alternative Waste Forms, and Borehole Seals.

2015—Completed conceptual design for the characterization borehole (CB) for the DBFT and issued a Request for Proposals for a Site and Site management team to conduct DBFT site management, and drilling and testing activities for the CB:

https://www.fbo.gov/index?s=opportunity&mode=form&id=c26f0b3b3e670d0fd610d3c4a6514bb7&tab=core&tabmode=list&=.

2016—Awarded contract for site management team to conduct DBFT site management, and drilling and testing activities for the DBFT CB.

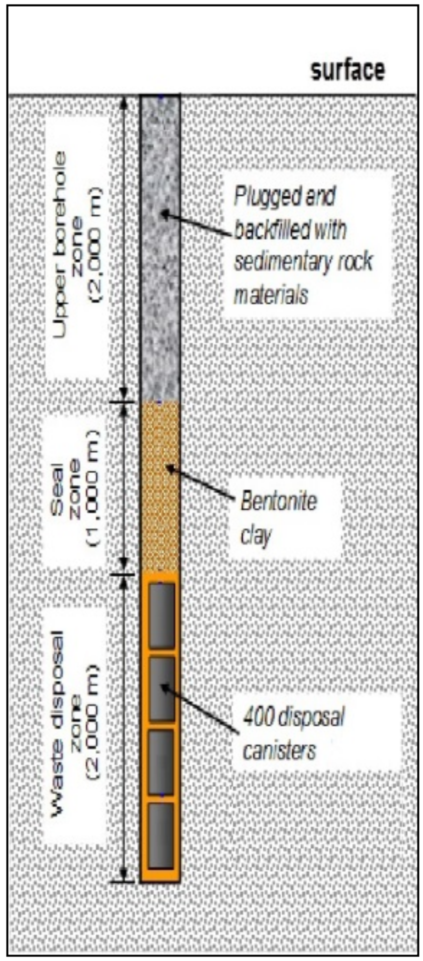

Figure 24–2. Schematic of a Deep Borehole Concept

24.3.3.8 Planned activities

The key future milestones include; design and fabrication of borehole canister, borehole construction, conduct canister emplacement test, and carry out science and engineering demonstrations.

24.3.4 Generic Disposal System Analysis (GDSA)

A key element in developing a safety case for any deep geologic repository is to quantitatively evaluate the response of various engineered and natural barriers limiting radionuclide transport to an accessible environment. This evaluation is provided by performance assessment methodologies such as Total System Performance Assessment (TSPA). In the U.S., the approach taken has been to construct a logical network for incorporating FEPs into quantitative methodologies that include appropriate representation of models and associated uncertainties. This approach is illustrated schematically in Figure 24–3, showing some potential process model codes that are used for simulations of thermo-, hydraulic, mechanical and chemical coupling processes in various types of geologic media and for deep borehole disposal.

Integration of the various FEPs into a system performance assessment (PA) model is accomplished by logically and systematically incorporating sub-level models into a computational framework. Such a system model was developed and presented in Figure 24–4.

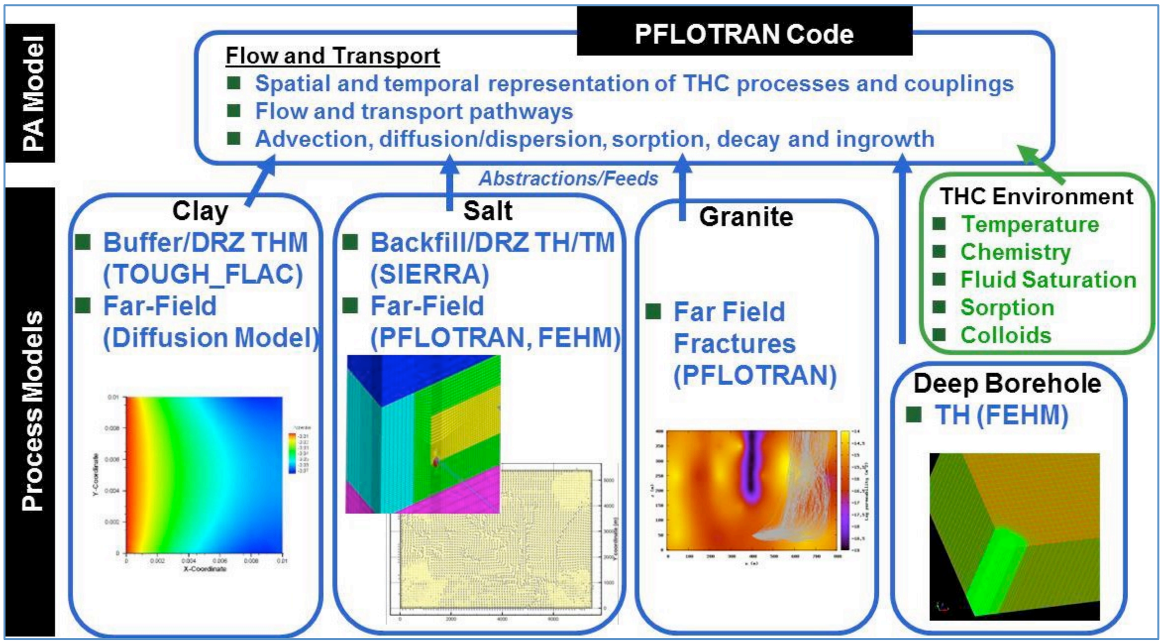

Figure 24–3. Integration of PA Model and Process Models (DOE 2014)

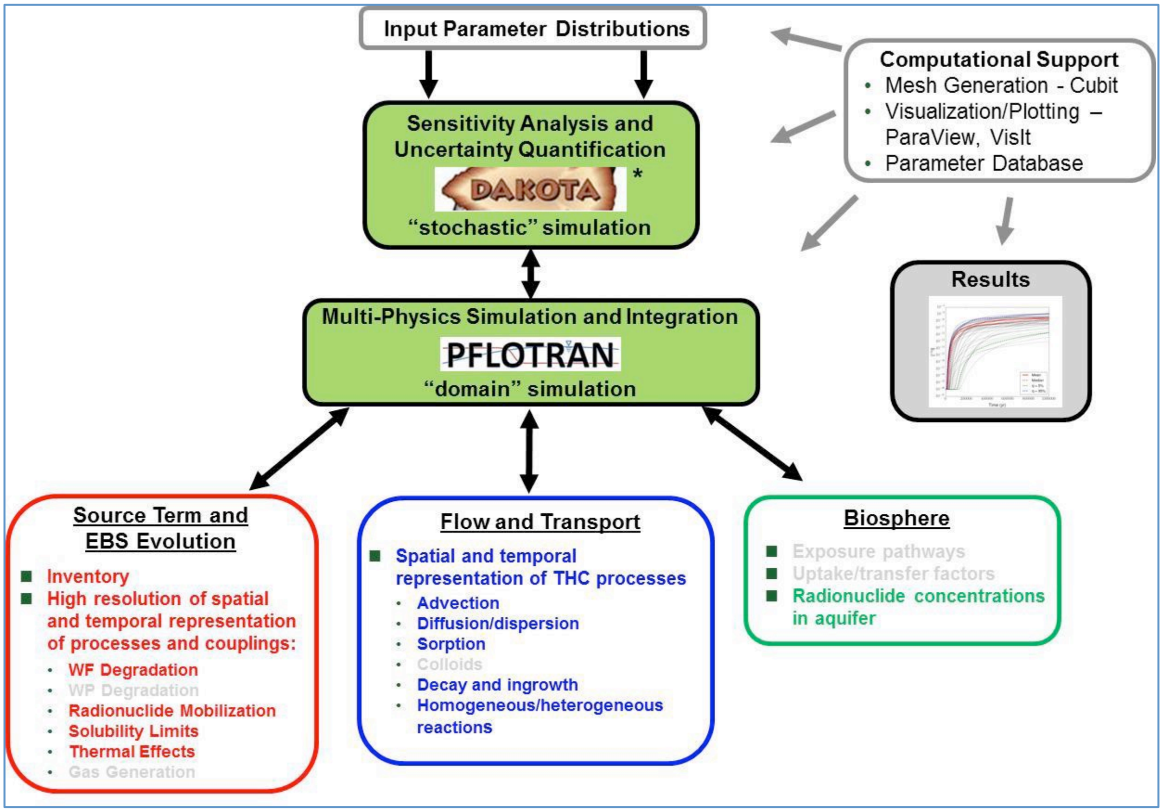

Figure 24–4. Computational Framework (DOE 2014)

Note the acronyms DAKOTA, PFLOTRAN, TOUGH-FLAC, SIERRA and FEHM represent computational tools that support process model calculations.

The focus over the past few years has been to perform generic disposal system analyses for the various geologic media and for deep borehole disposal. The generic disposal system model has been used to:

- Evaluate and test various platforms/software for advanced system modeling, including down-selection and implementation of the models (e.g., COMSOL, ANSYS, ASCEM, DAKOTA, and MOOSE).
- Define and describe advanced modeling "requirements" that address programmatic objectives, goals, constraints and enable flexibility for future modifications.
- Facilitate integration workshops to inform planning, testing, and modeling activities that is supportive of advanced modeling.
- Develop documents that present the evaluation and selection of codes, platforms and tools for use in the analysis.
- Incorporate waste form (WF) degradation tools into advanced system model. Experience gained and lessons learned here will support application to future work.
- Develop the equivalent of a generic disposal system models for salt argillite and granite to follow repository media.
- Launch prototype version / advanced modeling system implementation and testing, but still with limited EBS and coupled process capabilities.

- Develop preliminary generic reference cases for each geological environment but may need further refinement.

24.3.5 International R&D

Active collaboration with international programs, initiatives, or projects is considered very beneficial to the disposal research program, providing access to the decades of experience that some international programs have gained in various disposal options and geologic environments. Consistent with the UFDC priorities, and recognizing the benefits of international collaboration in the common goal of safely and efficiently managing the back end of the nuclear fuel cycle, Office of Used Nuclear Fuel Disposition Research and Development (UNFD) has developed a strategic plan to advance cooperation in disposal research with international partners.

UNFD's strategic plan lays out two interdependent areas of international collaboration (Birkholzer 2014). The first area is cooperation with the international nuclear community through participation in international organizations, working groups, committees, and expert panels. Such participation typically involves conference and workshop visits, information exchanges, reviews, training and education. The second area of international collaboration laid out in the strategic plan involves active R&D participation of U.S. researchers in international projects or programs.

UNFD considers the second area, active international R&D, to be very beneficial to efficiently achieve the program's long-term goals. Advancing opportunities for active international collaboration with respect to geologic disposal has therefore been the primary focus of UFD's international activities. An effort was made to collect information on international opportunities that complement ongoing disposal R&D within the UNFD, help identify those activities that provide the greatest potential for substantive technical advances, interact with international organizations and programs to help advance specific collaborations, and to initiate specific R&D activities in cooperation with international partners.

Following an evaluation of the international collaboration opportunities, DOE joined several international cooperation initiatives as a formal partner/participant/member. These include the following:

- The DEvelopment of COupled models and their VALidation against EXperiments (DECOVALEX) Project is an international research collaboration activity for advancing the understanding and mathematical modeling of coupled thermo-hydro-mechanical (THM) and thermo-hydro-chemical (THC) processes in geological systems.
- The Mont Terri Project is an international research project for the hydrogeological, geochemical, and geotechnical characterization of a clay/shale formation suitable for geologic disposal of radioactive waste.
- The Colloid Formation and Migration (CFM) Project is an international research project at the Grimsel Test Site (GTS) for the investigation of colloid formation/bentonite erosion, colloid migration, and colloid-associated radionuclide transport. In addition to the CFM project, DOE joined the ongoing Full-scale Engineered Barriers (FEBEX) experiment at GTS.
- KURT is a generic underground research laboratory located in a shallow tunnel in a granite host rock, located in a mountainous area near Daejeon, Republic of Korea. This work is conducted under the Joint Fuel Cycle Study with the Republic of Korea. UFD/Sandia National Laboratories (SNL) and KAERI agreement was signed in FY 2013.
- UFD campaign is also a member of the OECD/Nuclear Energy Agency Thermochemical Database Project.

- The DOE joined two task activities with SKB, the Swedish Nuclear Fuel and Waste Management Company. The task activities include conceptual and numerical modeling of performance-relevant processes in ground water transport and engineered systems.

The UFD campaign is also exploring direct collaboration with international partners where formal agreements may not be necessary.

24.3.6 Dual-Purpose Canisters (DPCs)

An aspect of the current inventory of stored commercial spent nuclear fuel is that there are approximately 10,000 large sized DPCs containing about 140,000 MT of SNF that is designed for both storage and transportation. A study was undertaken to evaluate the effects of disposing the DPCs directly in a mined repository (Hardin et al. 2015). The study concluded that direct disposal of commercial SNF in DPCs is technically feasible at least for some disposal concepts (salt; unsaturated, un-backfilled hard rock). Thermal management and post-closure criticality control are two important aspects of disposability, and both of these could be relatively simple to manage for a salt repository. Other media such as crystalline rock exist with sufficient heat dissipation for repository closure in the desired timeframe (e.g., 150-year fuel age out-of-reactor). Even in fresh groundwater, post-closure criticality control could be demonstrated for a majority of as-loaded DPCs (generally not burnup-credit designs) using uncredited margin. The proportion of DPCs that would remain subcritical increases with the salinity of repository groundwater (e.g., chloride content at least that of seawater and up to that of concentrated chloride brine). In addition, site-specific aspects must be considered when a repository site is identified for waste disposal.

24.4 Summary—Used Nuclear Fuel Management

The U.S. nuclear power industry has generated 71,700 metric ton of heavy metal (MTHM) of spent fuel as of the end of 2013. Of this, approximately 22,000 MTHM is in dry storage at nuclear power plant (NPP) sites. Most U.S. commercial spent fuel will remain stored at NPPs until a disposition path is identified. Some spent fuel is also being stored away from NPPs. The DOE spent fuel storage facilities include those used to store foreign research reactor and U.S. research reactor spent fuel transferred to DOE. Eventually the accumulated DOE, defense, and civilian waste will require disposal in a safe and secure environment. The SNF and HLW are expected to be disposed in deep and stable geologic formations. The work currently undertaken by DOE takes a systematic and science-based approach to evaluating different geologic media (crystalline, argillite and salt) to determine the generic system characteristics and performance of each media for the secure and safe disposal of the nuclear waste for long periods.

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24.7 Acronyms

BBM—Barcelona Basic Model

BRC—Blue Ribbon Commission

CB—Characterization Borehole

CFM—Colloid Formations and Migration

DBD—Deep Borehole Disposal

DBFT—Deep Borehole Field Test

DECOVALEX—DEvelopment of COupled models and their VALidation against EXperiments

DFN—Discrete Fracture Network

DOE—Department of Energy

DPC—Dual-Purpose Canisters

EBS—Engineered Barrier System

EDZ—Excavation Disturbed Zone

FEBEX— Full-Scale Engineered Barriers Experiment

FEP—Features, Events and Process

GDSA—Generic Disposal System Analysis

GTS—Grimsel Test Site

HLW—High-Level Waste

KAERI—Korea Atomic Energy Research Institute

KURT—KAERI Underground Research Tunnel

MIT—Massachusetts Institute of Technology

MOU—Memorandum of Understanding

MTHM—Metric Ton of Heavy Metal

NE—Nuclear Energy

NPP—Nuclear Power Plant

NRC—Nuclear Regulatory Commission

NS—Natural [barrier] System

NWTRB—Nuclear Waste Technical Review Board

OECD—Organisation for Economic Co-operation and Development

PA—Performance Assessment

R&D—Research and Development

RBSN—Rigid-Body-Spring-Network

RD&D—Research, Development, and Demonstration

SFP—Spent Fuel Pool

SKB—Swedish Nuclear Fuel and Waste Management Company

SNF—Spent Nuclear Fuel

SNL—Sandia National Laboratories

SP—Spent Fuel Pools

SubTER—DOE's Subsurface Technology and Engineering Research

THC—Thermo-Hydro-Chemical

THM—Thermo-Hydro-Mechanical

THMC—Thermal-Hydrological-Mechanical-Chemical

THCMBR—Thermal-Hydrological-Chemical-Mechanical-Biological-Radiological processes

TSPA—Total System Performance Assessment

UFD—Office of Used Nuclear Fuel Disposition Research and Development

UFDC—Used Fuel Disposition Campaign

UNF—Used Nuclear Fuel

UNFD—Office of Used Nuclear Fuel Disposition Research and Development

URL—Underground Research Laboratories

WF—Waste Form

Note: DAKOTA, PFLOTRAN, TOUGH-FLAC, SIERRA and FEHM represent computational tools that support process model calculations. DECOVALEX, COMSOL, ANSYS, ASCEM, DAKOTA, and MOOSE are all modeling tools.

Chapter 25

Multinational Geological Disposal Facilities

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ABSTRACT: The paper looks back at the development of concepts and initiatives for multinational geological disposal facilities (MGDF) since the origins of the idea in the 1970s and examines the increasing acceptability since the first World Wide Review in 1989. It highlights the constructive role taken by the IAEA and the direct support given by the European Commission (EC). The most comprehensive studies to date have been carried out in Europe where the Arius Association co-organised the major SAPIERR projects funded by the EC and has also been instrumental in establishing and running the European Repository Development Organisation Working Group (ERDO-WG), which has been supported by 8 Governments interested in progressing the MGDF concept. Arius has also been involved with related studies in other regions of the world including the Middle East, North Africa and Asia. A recent important development in which a State Government (South Australia) has launched a study of the business case for offering a multinational disposal service is also described. The paper draws the conclusion that multinational RWM solutions are now firmly embedded within the international nuclear power development landscape and, with the considerable security and economic advantages they offer, will be seen as the norm for many countries in the future.

25.1. Introduction

We have contributed to all of the series of review meetings and reports on "Geological Challenges in Radioactive Waste Isolation" since the original took place in Washington DC in 1989, as authors of either national or multinational disposal programs. It is interesting to look back over the publications from 1991, 1996, 2001 and 2006. In the national program descriptions, there are several ups and several downs; some of the downs have led to complete restructuring of programs (e.g., in Switzerland, France, the UK, the USA and Canada), some of the ups show reassuringly continuous progress, albeit often with longer delays than was foreseen (e.g., Sweden and Finland). In the present paper, however, our focus is on the issue of multinational disposal concepts. Here also there have been periods of marked progress, but also times when multinational initiatives faced stronger headwinds.

Interestingly, at the first meeting in 1989 the IAEA paper included a section on potential future international cooperation. Here it was stated that concept of international repositories has been "strongly advocated by many of the member states, that IAEA groups have worked on this, and that many countries believe that it is feasible to establish an international waste repository system." There is then a caveat expressing the view that, "due to the political nature, many international legal, environmental and technical concerns must be addressed prior to further consideration." In fact, around the turn of the century, there was indeed sometimes heated debate on the topic of international repositories, with some participants claiming that the concept could damage purely national programs, or even that the concept itself was not ethical. Nevertheless, in the 1996 report, the introductory section on highlights points out that a number of studies had been completed by that time by, for example, the NEA and the IAEA, and cites a paper by the head of the waste management organization in a small nuclear country, Switzerland, which directly pointed out the potential advantages of the multinational repository.

In the 2001 report (which was very upbeat about nuclear power being on the "threshold of a second frontier") international repositories took a very prominent place. There were three papers presented: one on the general potential benefits, one on the specific non-proliferation advantages, and one from the USA setting out the boundary conditions that would be necessary. In the 2006 report, there was less positive news about some national programs (e.g., Switzerland had abandoned the Wellenberg program, the UK had given up its aspirations at the Sellafield site, and Japan had started a brave new voluntary siting program but with no visible results in its first years). In the multinational area, however, significant progress was reported. The IAEA had published its seminal report, TECDOC 1413, on multinational repositories; the EC was supporting the SAPIERR projects directly, studying regional/multinational concepts within the EU framework and was preparing drafts of its Waste Directive, which would eventually be produced with wording confirming the legality of shared repositories within the EU; the Arius Association, established in 2002, was directly involved in managing the SAPIERR projects and was making preparations for the establishment of the ERDO working group which has now been in existence for some years.

At the end of our review in 2006, we observed that: *"We need bold initiatives for global solutions if we are to achieve the global improvements in safety, security and economics that multinational repositories can bring."* As will be seen in the following review, it seems as though this point may now have been reached.

25.2. The Bigger Picture

The concept of developing a multinational geological disposal facility (MGDF) has been discussed for more than 40 years, and has generally been closely linked with wider consideration of multinational fuel cycle activities, often via initiatives taken by the IAEA. Most observers can readily see the advantages that an MGDF would afford, but everyone is cautious about the possibility of delivering such a solution, owing to the cross-national political challenges that clearly exist. Some commentators have seen the political challenges to be so strong, especially in presenting possible difficulties to national RWM programs within their own country, that they have opposed any form of initiative. This active opposition has sometimes meant that making progress on MGDFs has been an uphill struggle.

Our understanding of the advantages and challenges has changed little since the idea was first mooted in the 1970s, although some of them have become more focused on the specific needs of certain types of national RWM program. We look again at the advantages and challenges below.

25.2.1 Advantages that could be provided by an MGDF

Global nuclear security: Especially in collecting and disposing in a central safe and secure location sensitive nuclear materials from widespread RWM programs with different paces of development; many with no realistic disposal solution in sight, or with only vague time schedules.

Economy of scale: An MGDF would, in principle, allow disposal of wastes at lower cost than in any single national GDF—this is particularly attractive for a group of countries working together towards a shared, regional solution "at cost."

Commercial realism: A commercially-based MGDF operating within a strong, national political and regulatory framework is likely to be driven more efficiently than any national program (regardless of scale) and achieve more timely and cost-effective results for its users.

Access for small users: Many countries with no nuclear power program still have radioactive wastes that require geological disposal, yet it is unlikely that they would develop their own national GDF, owing to the costs and societal difficulties in siting one: access to an MGDF provides an ideal solution.

Credibility of nuclear power: The ability for any nuclear power nation to show that it has a closure solution for its RWM underpins all arguments for the continuation or growth of nuclear power.

25.2.2. Challenges that have been experienced to date include

Concerns about disruption of national RWM programs: Inability to counter perceived public concerns about possible "dumping of other nations' nuclear waste in our country" has been a strong counter against progress: several national programs have presented active opposition to multinational initiatives, particularly in Europe.

Unwillingness to face the political challenge: With a national program in place, no matter how slow, there has been no political imperative for most countries to get together and address the possibility of a multinational approach: national politicians do not want to address arguments from the electorate that an MGDF 'might be here.'

Accusations of 'wait and see': Opponents of developing an MGDF have emphasized that it is improper for a national RWM program to cite an MGDF as its eventual solution if no active project exists, as this is simply putting off finding a credible solution.

Finding the right host: Although a number of possible projects have been advanced around the world over the last decades, it is essential that any country offering possible long-term storage or MDGF services has the strongest non-proliferation, nuclear security and environmental protection credentials. Few nations would be universally accepted in this respect.

Lack of resources: Because multinational solutions have tended to be regarded as something peripheral, and because no country has regarded it as their own responsibility, or (until very recently) as an interesting challenge with potential, resources to progress MGDF initiatives have been meagre. The limited funding that has been available has been provided by the European Commission, some US Foundations and the small nations involved with the ERDO working group.

Reviewing these pros and cons, perhaps the three most significant strategic developments that have emerged from the discussions over the last decade have been the growing consensus that multinational repositories will be a valuable or even a necessary complement to national facilities, the adoption of the concept of a 'dual track approach' which keeps options open, and the recognition that all developed industrial countries that use nuclear technologies will need access to geological disposal.

The **'dual track' concept** emerged from European projects, discussed later in this review, during the last few years. The basis of the concept is that any nation pursuing a multinational solution must also have an operational GDF program that could lead to a national solution. Two drivers were influential in establishing this approach, which is now cited specifically as the national RWM model by several countries. The first was to counter any accusation of 'wait and see.' It is certainly not credible for a national RWM program to rely entirely on a solution that is not totally within its control and is not assured. The second driver was the requirement to satisfy international legal obligations. The earliest of these was for signatories to the IAEA Joint Convention, which requires regular and formal reporting, open to international scrutiny, which must present a credible national strategy and timeline for RWM. For the European Union countries, the need to have a clear and credible strategy was formalized in European Council Directive 2011/70/EURATOM, which sets out a legal timetable for EU Member States to establish a RWM program. As referred to above, this Directive also, for the first time, very explicitly acknowledges the legality of transferring radioactive wastes between countries for the purposes of disposal. The current version of the Directive restricts this freedom to transfers to other Member States. For transfers outside the EU, the host country would have to have a deep geological repository actually in operation, and this will not be the case for a long time for any potential foreign host state.

The weight of discussions on multinational solutions has focused on countries with growing stocks of spent fuel to manage, as these present the greatest security issues and would involve the largest and most costly GDF development projects. Relatively little attention in multinational discussions was given to **non-nuclear power countries** until recently. Most of the advanced industrial countries have long-lived or high activity wastes in storage from medical, research or industrial applications. Although much of this material is under agreement to be returned to the supplier (e.g., radiation sources and research reactor fuel), some countries possess materials that they currently have no final solution for, other than permanent storage. European examples include Denmark, Ireland and Norway. A parallel example is Italy, which exited early from nuclear power in 1984 but, from its relatively brief nuclear era, still has material in store which must be disposed of geologically. Access to an MGDF would provide the most obvious solution to these problems.

25.3. Strengthening the Documentary Basis

Routes to and constraints on MGDF development, along with the surrounding context of multinational co-operation on nuclear fuel cycle activities, have been under more-or-less continuous review over the last decade. The IAEA has been foremost in capturing these discussions and establishing a solid basis of documentation on possible ways forward.

In 2004, the IAEA summarized (IAEA 2004) early work on multinational concepts, going back to the 1970s. In 2005, at the request of the Director General, a high-level expert group produced a comprehensive report (IAEA 2005) on multilateral approaches to the nuclear fuel cycle, covering enrichment, reprocessing and disposal, with external support to the disposal study being provided by Arius. Further IAEA reports followed, addressing the viability of multinational repositories (IAEA 2011) and discussing a staged approach for partnering in the implementation of such facilities (IAEA 2016). The latter report examines not only the benefits of multinational concepts but also all risks of a technical, financial, institutional or socio-political nature. The report took around three years to be cleared at higher levels in the IAEA, indicating that the topic of multinational disposal is, nevertheless, still sensitive at the Agency. Currently, the INPRO (International Project on Innovative Nuclear Reactors and Fuel Cycles) section at the IAEA is starting a project aimed at studying drivers and impediments to multinational back-end cooperation.

25.4. Progress in Europe

Over the last decade, Europe has taken the lead on MGDF concept development. The most comprehensive project to assess technical, economic, legal, security, political and societal aspects of developing an MGDF took place through the SAPIERR projects, which ran from 2005–2009. The work was financed by the European Commission and carried out by a consortium of specialists from fourteen EU countries.

The results of the projects highlighted (Verhoef et al. 2009) that, apart from the credibility imparted by having a concrete and common plan, the most obvious advantages are the economic benefits to partner countries and to the EU as a whole. Partner countries could each save billions of EUR by sharing development and disposal costs rather than each having to implement a national GDF, with over half the savings being in shared RD&D. Working together on a common concept, design and, eventually, site can have tremendous economic and political benefits. For the models analyzed, the saving to the EU as a whole was estimated at 15 to 25 billion EUR. If a regional MGDF were able to offer disposal as a commercial service to other European countries once it becomes operational, the original partner countries may be able to manage their own current and future wastes with further significant cost reductions. There would also be specific economic benefits to the host country and community.

SAPIERR found that most of the challenges involved in developing a shared regional GDF are closely analogous to those of a national facility. In both national and multinational programs, finding suitable sites remains the biggest challenge. SAPIERR was influential in formulating a possible staged siting strategy that was published in 2008 (Chapman and McCombie 2008).

SAPIERR concluded by making proposals for a staged, adaptive implementation strategy leading to a shared European GDF. A smaller group of potential partners was formed: the European Repository Development Organisation Working Group (ERDO-WG). Governments from eight Member States have provided funding and delegated representatives since 2009. The activities of the group have involved consideration of organisational forms and financing models for a European Repository Organisation (ERO) that would initially function as a small sister to existing national organisations. Discussion documents (ERDO-WG web 2016) cover siting strategies for a shared GDF, the size and form of an ERO, outreach activities, operating guidelines and a model constitution. An important part of ERDO-WG activities has been analysing the impact on small radioactive waste programs of European Council Directive 2011/70/EURATOM, discussed above. A key aspect of this is the need to pursue the dual track approach, with partner countries in a sharing project also maintaining a strong national program until significant progress has been made on the shared solution.

It is important to understand that the ERDO developments have taken place from the bottom up. Although the seed funding came from the EC (for SAPIERR), there has been no strategic push or leadership from the EU political institution and its agencies to encourage the idea of a shared European MGDF. In practice, financial support in the radioactive wastes management area has been provided by the EC almost exclusively for cooperative projects focused very directly on technical R&D. The Commission has less flexibility in funding initiatives of a more strategic nature. The position of the EC and the European Parliament is that such developments must come from like-minded countries working together. It can be seen that the step between national organisations discussing the possibility of a shared facility and political endorsement within the EU (with all that this entails inside the European bureaucracy and EU national political interests) is inevitably the highest hurdle that ERDO must overcome.

25.5. 2016: Australia Takes the International Lead

While Europe moves slowly forward, by far the most important development to have occurred recently is the initiative by the Government of South Australia to consider the pros and cons of establishing nuclear fuel cycle facilities in the State. South Australia is among the world's major suppliers of uranium and the government wished to consider whether it should develop a whole range of other services, as well as its own nuclear power program. The Royal Commission established by the Government reported in May 2016. One of its central findings (Government South Australia 2016) was that establishment of commercially based storage and disposal facilities for international clients, for spent fuel, HLW and long-lived ILW, would be feasible and of great benefit to South Australia. The sheer economic scale of the storage and MGDF project that was modeled can be astonishing to those outside the nuclear industry. For a waste inventory representative of a significant number of the world's smaller nuclear power programs, the resource turnover was estimated to be some hundreds of billions of dollars.

The report has created considerable interest globally, and active discussion within South Australia. The Government will respond to the findings by the end of 2016. The overall tenor of comment is generally positive and an opinion poll carried out while the Commission was at work reported an almost even split between those in favor and those against. This contrasts considerably with public opinion in most national programs, even some of the more advanced ones, and is certainly very different to the reaction to the Pangea proposal in Australia some 15 years ago. The proposed commercial Pangea project (with similar working assumptions to the current South Australian study) originated within the nuclear industry outside Australia and was initially carried out in secrecy. When it was leaked, just prior to its intended public launch, the political and public reaction was predictable and was terminal for the project. This time, the difference is that the initiative comes from within government and the assessment has been carried out openly and transparently. In practice, the sound scientific and technical studies carried out in the Pangea project (McCombie et al. 2000) demonstrated the potential benefits of disposal in the stable geological environment to be found in large areas of Australia. These studies may well be of value in the current South Australian initiative.

The response of the Government to the Royal Commission report will be a critical barometer for multinational initiatives. If a state-sponsored multinational initiative in a country with the high global political status and credentials of Australia were to be available, it would change the worldwide paradigm of radioactive waste management forever and for all RWM programs in almost every country.

25.6. Elsewhere in the World

International entities and think tanks continue to study the potential impacts of multinational storage or disposal, in particular of spent fuel. The Arius Association, with the support of the Sloan and Hewlett Foundations, has examined how the European ERDO model might be extended to countries in the MENA regions (Middle East and North Africa) or in Asia. Workshops focussing on common or regional waste management issues have been run in Tunisia, with participation of MENA nations, and in the UAE, with participation of Gulf Co-ordination Council countries. Some meetings have been held in cooperation with both the IAEA and the Arab Atomic Energy Agency (AAEA).

The Nuclear Threat Initiative (NTI) runs the *'Developing Spent Fuel Strategies'* project, supported by the MacArthur and Hewlett Foundations, which also looks at how multinational facilities might impact on the nuclear fuel cycle. The American Academy of Arts and Sciences (AAAS) also examines this question in its *'Global Nuclear Futures'* project. A current International Framework for Nuclear Energy Cooperation (IFNEC) project organised by the Working Group on Reliable Nuclear Fuel Services is a multi-year study aimed at assessing what small nuclear programs might do to progress most effectively and efficiently the back-end planning using a dual track approach.

We have focused above mainly on the general contextual discussions that reverberate among international organisations, on Europe and on Australia. But other initiatives exist, connected with the possibility of leasing and taking back nuclear fuel. There have been several such concepts mooted by Russia over the years, the most recent being in early 2016. In this concept, fuel is leased from and then returned to Russia for reprocessing, fresh REMIX fuel is returned to the user (in an indefinite cycle) and the HLW from reprocessing at each iteration may be disposed of in Russia. It is this latter part that makes the potential offer novel. According to WNA reports (World Nuclear News 2016), the model might be extended to link with the findings of the South Australian Royal Commission: a user acquires uranium from Australia and sends its HLW for disposal in Australia, with an Australian entity owning the uranium throughout the whole REMIX cycle, and also the eventual vitrified high-level wastes. In this model, Russia (or France, the UK or Japan) handles reprocessing, enrichment of fresh uranium and fuel fabrication.

25.7. Arius: Moving the Concept Forwards

A linking thread of almost all the initiatives described has been the Arius Association, which is actively involved as participant or adviser in all the projects described above. Arius (Association for Regional and International Underground Storage) is a non-profit body based in Switzerland, which was founded in 2002 with the mission of promoting concepts for multinational storage and disposal facilities. The Association is funded by a combination of Organisational and Individual Members and by project work. For example, Arius acts as the secretariat for the ERDO-WG and has managed the SAPIERR projects. The activities of Arius have led to the Association building a worldwide reputation as a source of information and ideas related to the important question of how cooperation at the back-end of the fuel cycle can enhance global and national nuclear safety and security. In addition to involvement in virtually all of the initiatives mentioned above, Arius is continually involved in responding to enquiries from the media on multinational issues and is also involved in various university studies on different aspects of the issue. One interesting example here is the involvement of Arius and the ERDO-WG in an ongoing study by the University of Delft into the ethical issues associated with transfer of radioactive wastes between nations (Taebi 2015).

Arius has been at the forefront of developments since 2002, and is gratified to observe that the initial, predominant resistance, even hostility, to the MGDF concept, has turned into widespread acknowledgement of its advantages and realization that its development will benefit many industrialised countries.

25.8. The Next Ten Years

Ten years ago, we were optimistic in our outlook for the future—justifiably, we believe, based on the current level of international interest and activity in multinational RWM possibilities. All RWM programs take time and all GDF projects have experienced delays and restarts, if not outright cancellation and returns to square one. The slow development of multinational RWM projects has been no different. However, the situation in national programs is improving. The world's first spent fuel GDF in Finland received its construction license in 2015 and should be operational in the next five years. In parallel, the challenges to MGDFs now look less problematic than they did a decade ago, and the balance between challenges and advantages has swung heavily towards appreciation of what such facilities have to offer globally.

When we look forward to the next decade, we expect to see strengthening of the regional European MGDF activities of ERDO, as two or three national GDFs become operational and the smaller nuclear power nations begin to work more confidently together. It remains a point for speculation as to whether any of the national European GDF projects will begin to offer services to other European countries, especially those non-nuclear power countries with just a few cubic metres of waste to dispose. In other regions of the world, such as the Gulf States, the MENA and ASEAN regions, we expect progress to be more hesitant, with a continuing eye to developments elsewhere and examples set by other countries. We certainly expect the Australian initiative to be the main test of the political practicalities of establishing an MDGF. If the South Australian Government decides to move forward and endorse the establishment of a commercial project that offers a broad range of RWM services internationally, then this could become the only MGDF that is needed globally for the foreseeable future.

In all respects, we believe that multinational RWM solutions are now firmly embedded within the international nuclear power development landscape and, with the considerable security and economic advantages that they offer, they will be seen as the norm for many countries in the future.

25.9. Acknowledgements

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25.11. Acronyms

Arius— Association for Regional and International Underground Storage

AAAS—American Academy of Arts and Sciences

AAEA—Arab Atomic Energy Agency

EC—European Commission

ERDO—European Repository Development Organization

ERDO-WG—European Repository Organization Working Group

ERO—European Repository Organization

EU—European Union

GDF—Geological Disposal Facility

HLW—High-Level Waste

IAEA—International Atomic Energy Agency

IFNEC—International Framework for Nuclear Energy Cooperation

ILW—Intermediate Level Waste

MDGF—Multinational Geological Disposal Facility

MENA—Middle East and North Africa

MGDF—Multinational GDF

NTI—Nuclear Threat Initiative

REMIX—REgenerated MIXture (REMIX)

RWM—Radioactive Waste Management

SAPIERR—Support Action: Pilot Initiative for European Regional Repositories

TECDOC—Technical Document

UAE—United Arab Emirates

WNA—World Nuclear Association