ELECTRIC POWER RESEARCH INSTITUTE

EPRI Review of Geologic Disposal for Used Fuel and High Level Radioactive Waste

Volume III—Review of National Repository Programs

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1021614

Final Report, December 2010

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NOTE

ACKNOWLEDGMENTS

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This report describes research sponsored by EPRI.

This report is the result of contributions from a number of individuals and organizations. Principal contributions to the preparation of this report were made by the following individuals: Michael Stenhouse, Michael Apted, and Wei Zhou of INTERA, Inc., and Andrew Sowder and John Kessler of EPRI. Reviews of specific chapters were provided by Anne Claudel of Nagra (Switzerland chapter) and anonymous Atomic Energy of Canada, Ltd., personnel (Canada chapter). In addition, a number of appendices were prepared by experts familiar with the respective subject matter. Appendix A on the nominative site selection process in Switzerland was prepared by Stratis Vomvoris of Nagra. Appendix B on geologic disposal concepts was prepared by Neil Chapman of Chapman & Company Consulting with contributions from the UK Nuclear Decommissioning Authority - including permission to adapt and incorporate portions of a report previously prepared for the NDA. Appendix C on alternative disposal concepts was prepared with the assistance of Ian McKinley of McKinley Consulting; support from the Obayashi Corporation of Japan for preparation of some of the graphics used in Appendix C is acknowledged.

This publication is a corporate document that should be cited in the literature in the following manner:

EPRI Review of Geologic Disposal for Used Fuel and High Level Radioactive Waste: Volume III—Review of National Repository Programs. EPRI, Palo Alto, CA: 2010. 1021614.

REPORT SUMMARY

The effective termination of the Yucca Mountain program by the U.S. Administration in 2009 has left the U.S. program for management of used fuel and high level radioactive waste (HLW) in a state of uncertainty. In concert with this major policy reset and in response to the resulting policy vacuum, the President directed the Energy Secretary to establish the Blue Ribbon Commission on America's Nuclear Future (BRC) "…to conduct a comprehensive review of policies for managing the back end of the nuclear fuel cycle and to provide recommendations for developing a safe, long-term solution to managing the Nation's used nuclear fuel and nuclear waste" (http://brc.gov/pdfFiles/BRC_Charter.pdf). The BRC has been given until January 2012 to complete its review and deliver a final report to the Energy Secretary. Most nations pursuing a permanent disposal path for used fuel and HLW are now considering or implementing a deep geologic disposal program. These programs represent a collective source of experience and knowledge that may prove useful for informing the development of a post-Yucca-Mountain disposal strategy and program in the United States.

Background
U.S. efforts to site and construct a deep geologic repository for used fuel and HLW proceeded sporadically over a three-decade period from the late 1950s until 1982, when the U.S Congress enacted the Nuclear Waste Policy Act (NWPA) codifying a national approach for developing a deep geologic repository. Amendment of the NWPA in 1987 resulted in a number of dramatic changes in direction for the U.S. program, most notably selection of Yucca Mountain as the only site of the three remaining candidates for continued investigation. The Energy Policy Act of 1992 further defined the direction of the U.S. repository program, mandating development of Yucca-Mountain-specific standards and regulations, documented in 40 CFR 197 and 10 CFR 63. In June 2008, the U.S. Department of Energy (DOE) submitted a license application to the U.S. Nuclear Regulatory Commission (NRC) for the construction of a geologic repository at Yucca Mountain, Nevada. The license application was accepted for formal NRC review in September 2008. The United States is not alone, however, in the pursuit of deep geologic disposal for used fuel and HLW.

Objective
To review and summarize the approach, developments, and status of national deep geologic disposal programs in a number of countries.

Approach
From 1989 until 2008, EPRI conducted independent assessments of a proposed deep geologic repository for the disposal of used fuel and HLW at Yucca Mountain, Nevada—closely following the development of repository programs in the United States and abroad. In this report, EPRI provides a review and summary of geologic disposal programs in a number of countries

pursuing geologic disposal of used fuel and HLW, including Belgium, Canada, China, Finland, France, Germany, Japan, Spain, Sweden, Switzerland, Taiwan, and the United Kingdom. These countries were selected as a representative sample spanning a range of characteristics such as size, population, geological diversity, role of nuclear generation, and program maturity.

Results

Key observations from this review relevant to a new U.S. program include the following:

- All countries seriously pursuing permanent disposal for used fuel and HLW consider the development of a deep geologic repository as the preferred option, even when faced with limits on territorial area and geologic diversity.
- Based on collective international experience, there is no single best approach or combination of attributes for an effective geologic disposal program.
- Despite variation among countries, there has been a general convergence in many elements of their repository programs, such as the virtually universal adoption of the multi-barrier concept in which the engineered system can be matched with the natural host environment to optimize performance. Convergence is also seen in the adoption of risk-based performance metrics and the use of performance assessment methods for demonstrating repository performance over unprecedented timeframes.
- In most countries, the fate of national disposal programs is strongly tied to establishment and sustainment of public and stakeholder support over the extended time period required for repository development.
- Successful programs in Sweden and Finland provide positive cases in which construction and operation of a repository for used fuel and HLW appear likely in the near future.

EPRI Perspective

EPRI pioneered application of the total system performance approach (TSPA) for evaluating the performance of geologic repository systems on a probabilistic basis. Over the two decades of research in this area, EPRI has released and maintained a substantial body of work in the public domain to inform its member organizations, the public, government, independent technical review bodies, and other stakeholders. EPRI is leveraging its experience and expertise to provide an independent, third-party scientific and technical perspective. The goal is to inform ongoing debate in the United States and internationally on the merits of various nuclear fuel cycle options and their impacts on the management of used fuel and HLW.

Keywords

Used Fuel
Spent Nuclear Fuel (SNF)
Deep Geologic Disposal
High Level Radioactive Waste (HLW)
HLW Repository
International HLW Repository Programs

FOREWORD

From 1989 until 2008, EPRI conducted independent assessments of a proposed deep geologic repository for the disposal of used fuel and high level radioactive waste (HLW) at Yucca Mountain, Nevada. Over this two-decade time period, EPRI's expert team has followed the development of the US repository program closely and integrated its pre- and post-Nuclear Waste Policy Act era experience and unique electric utility perspective to provide an independent, third party perspective on this national undertaking with respect to technical issues, including site suitability and regulatory compliance.

EPRI pioneered application of the total system performance assessment (TSPA) approach for evaluating performance of geologic repository systems on a probabilistic basis. Along the way, EPRI developed and updated analytical tools for TSPA-based evaluations. Over the two decades of research in this area, EPRI has released and maintained a substantial body of work in the public domain to inform its member organizations, the public, government, independent technical review bodies, and other stakeholders.

In June 2008, the US Department of Energy (DOE) submitted a license application to the US Nuclear Regulatory Commission (NRC) for the construction of a geologic repository at Yucca Mountain, Nevada. The license application was accepted for formal NRC review in September 2008. Once docketed, the US disposal program entered a new and fundamentally different phase – one involving formal regulatory review and administrative law proceedings via the Atomic Safety Licensing Board. In keeping with its role to operate in the public domain, EPRI ended its Yucca Mountain specific research program at that time. Subsequently, EPRI has refocused its used fuel and HLW management research program on more generic topics related to the evaluation of nuclear fuel cycle options. In parallel, EPRI has also begun to ramp up a research program on the performance of used fuel storage systems over extended periods, i.e., beyond the current 60 year licensing timeframe for dry storage and reactor operations.

The effective termination of the Yucca Mountain program by the US Administration in 2009 has once again pushed the operation of a permanent disposal facility for used fuel and HLW into an uncertain future. In concert with this major policy reset, the President directed the Energy Secretary to establish the Blue Ribbon Commission on America's Nuclear Future (BRC) "…to conduct a comprehensive review of policies for managing the back end of the nuclear fuel cycle and to provide recommendations for developing a safe, long-term solution to managing the Nation's used nuclear fuel and nuclear waste." The BRC has been given 24 months from January 2010 to complete its review and deliver a final report to the Energy Secretary, which is to include: (1) "Consideration of a wide range of technological and policy alternatives, and should analyze the scientific, environmental, budgetary, financial, and management issues, among others, surrounding each alternative it considers;" (2) " …a set of recommendations regarding policy and management, and any advisable changes in law;" and (3) "Recommendations on the fees currently being charged to nuclear energy ratepayers and the recommended disposition of the available balances consistent with the recommendations of the Commission regarding the management of used nuclear fuel."[1]

EPRI is uniquely positioned to provide a scientific and technical perspective for the ensuing national (and international) conversation on the future of nuclear fuel cycles and their impacts on the management of used nuclear fuel and HLW. This perspective is informed by a combination of experience, expertise, and collaboration in the relevant areas of geologic disposal, used fuel storage and transportation, and advanced nuclear fuel cycle and by EPRI's role as an international focus for collaborative research supporting electric power generation. To this end, EPRI is preparing a set of technical reports on storage, disposal, and advanced fuel cycle issues related to the management of used fuel and HLW in 2010.

There are numerous options for managing the wastes associated with the nuclear fuel cycle, but all nuclear fuel cycles eventually require permanent disposal for some form and some amount of long-lived radioactive material. This report is one of a multi-volume series entitled **"EPRI Review of Geologic Disposal for Used Fuel and High-Level Radioactive Waste"** that surveys and evaluates past, present and planned disposal options gleaned from five-decades of geologic disposal science and regulation in the US and abroad. This effort is not intended to provide a comprehensive review of the US program or of those in other countries. Instead, EPRI has prioritized its efforts to focus on a handful of key areas and topics to avoid unnecessary duplication of other efforts and to direct attention and resources on those areas where EPRI could make the greatest contribution to the ongoing technical debate on the management of the back-end of the fuel cycle.

The first three volumes of the EPRI geologic disposal review series are intended to provide basic descriptions and observations on key repository siting, regulatory and disposal program experiences. A fourth volume is planned to capture and distill key lessons learned from US and international experience. These respective reports are outlined below:

Volume I - The U.S. Site Selection Process Prior to the 1987 Nuclear Waste Policy Amendments Act:[2] The screening and siting processes for a HLW repository in the US, including many stops and starts along the way, spanned from the 1950's to 1987 when amendment of the 1982 Nuclear Waste Policy Act restricted further site characterization to the Yucca Mountain site in Nevada.

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[1]Blue Ribbon Commission on America's Nuclear Future, Advisory Committee Charter. <http://brc.gov/pdfFiles/BRC_Charter.pdf> U.S. Department of Energy, Washington, D.C., March 2010; accessed 14 May 2010.

[2]EPRI Review of Geologic Disposal for Used Fuel and High-Level Radioactive Waste: Volume I - The U.S. Site Selection Process Prior to the 1987 Nuclear Waste Policy Amendments Act. EPRI, Palo Alto, CA: 2010. 1021056.

<u>Volume II - U.S. Regulations for Geologic Disposal</u>:[3] The evolution of US standards and regulations for geologic disposal of used fuel and high-level waste illustrates the critical role that the regulatory framework for and context of a national geologic disposal program plays in the development and design of repositories.

<u>Volume III - Review of National Repository Programs</u>:[4] Most nations pursuing a permanent disposition path for used fuel and HLW are considering or implementing a deep geologic disposal program. These programs provide examples of geologic repository siting approaches and design for a range of candidate geologies and other important technical factors.

<u>Volume IV - Lessons Learned</u>:[5] Over five decades of scientific study and peer-review, technical and regulatory developments, and site selection and characterization have led to an international consensus that geologic disposal is an appropriate method for isolating used fuel and HLW from the biosphere. Likewise, many important lessons can be gleaned from a half-century of experience.

[3] EPRI Review of Geologic Disposal for Used Fuel and High-Level Radioactive Waste: Volume II - U.S. Regulations for Geologic Disposal. EPRI, Palo Alto, CA: 2010. 1021384.

[4] This Report.

[5] EPRI Review of Geologic Disposal for Used Fuel and High-Level Radioactive Waste: Volume IV – Lessons Learned. EPRI, Palo Alto, CA: 2010. 1021057.

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1 INTRODUCTION

All countries seriously pursuing programs for the management of used fuel and/or HLW have chosen or are considering deep geologic disposal as a feasible means to provide the safe, secure isolation from the biosphere. The nature and status of these programs vary among countries, but there are also noted examples of similarities and convergence of thinking on the challenging task of siting a deep geologic repository intended to isolate long-lived radioactive waste over tens of thousands to hundreds of thousands of years. The collective international experience offers a wealth of information for informing planning and execution of geologic disposal programs, including in the US where the precedent setting Yucca Mountain program has been effectively terminated by policy changes and decisions within the Executive Branch.

This report seeks to capture and summarize the developments and status of national repository programs from a representative sample of countries:

- Belgium
- Canada
- China
- Finland
- France
- Germany
- Japan
- Spain
- Sweden
- Switzerland
- Taiwan
- United Kingdom

This list is not intended to be complete or inclusive, and there are clearly many additional countries that could have been included. Instead, the intent here is to provide a diverse collection of countries of different sizes, population, geologies, and cultures to provide a reasonably complete sampling of approaches to geologic disposal. Disposal programs for used fuel and HLW in these twelve countries are reviewed and summarized in Chapters 2 – 13 with an emphasis on the following key elements:

- General nuclear profile and waste inventory;
- Institutional arrangements - including legal, regulatory, and funding frameworks;
- Site screening, selection, and characterization;
- Repository design concepts;
- Transparency and stakeholder involvement;
- Performance metrics, safety assessments, and licensing; and
- Program status and maturity.

At the end of each chapter is a summary of key observations, and a global summary is provided in Chapter 14 to facilitate comparisons by issue or subject among national programs.

A detailed description of the nomative site selection process in Switzerland is presented in Appendix A. Appendix B reviews geologic disposal concepts under consideration in the UK to complement a range of geologic environments. Appendix C provides a state-of-the-art review of alternative disposal concepts. Appendix D describes the Safety Case in terms of history, main elements, and uses including examples from its application in the UK in both nuclear and non-nuclear contexts.

2
BELGIUM

2.1 Introduction

2.1.1 General Nuclear Profile

Belgium's nuclear fleet comprises 7 operational reactors and 6 shutdown units. In 2009, nuclear generation totaled 45 TWh, representing 52% of the country's total electricity supply (WNA, 2010). The projected total inventory of irradiated nuclear fuel from nominal 40-year lifetimes for all plants is approximately 5,000 MTHM[6] (FANC, 2009).

Until the mid 1990's, Belgium's principal waste management strategy for the back-end of the nuclear cycle had been to adopt a fuel cycle that included reprocessing and single recycling of Pu as mixed-oxide (Pu-U) fuel (MOX) in light water reactors (LWRs). Approximately every 10 years, a government-appointed National Commission reviews the country's nuclear energy program and makes recommendations for future policy. Based on the recommendations of this Commission in 2000 (AMPERE, 2000), the government decided to phase out nuclear power by legislative act (Law of January 31 2003). This phase out includes a ban on new NPP construction and a 40-year operating life for operating NPPs.

Belgium has studied the feasibility of geologic disposal of HLW and used nuclear fuel for over 30 years, and waste producers have been paying into a dedicated long-term waste management fund since 1985. However, in spite of its established disposal program, the country has not formally committed to the geologic disposal option. Most studies to date have focused on Boom Clay as a candidate host formation. In the absence of a decision on long-term waste management, irradiated nuclear fuel is being stored at nuclear plant sites in both wet and dry configurations.

2.1.2 Used Fuel / HLW Inventory and Projected Disposal Requirements

Belgium currently has in storage a total of 195 m^3 vitrified HLW arising from domestic reprocessing operations at the former Eurochemic (now Belgoprocess) plant between 1985 and 1991. In addition, a total of ~670 MTHM of irradiated UO_2 fuel was reprocessed at AREVA's (formerly COGEMA's) La Hague facility in France, which generated 59 m^3 of vitrified HLW in

[6] Metric tonnes of heavy metal.

387 canisters; this inventory was returned to Belgium by the end of 2007 for interim storage at Belgoprocess's Dessel site (FANC, 2009). Additional waste associated with Belgium's used fuel reprocessing includes:

- 528 canisters of compacted technological and hulls and end pieces;
- 1100 drums bitumenized waste (Belgian Category B, or LL-L/ILW).

Some of this incidental waste remains in storage outside Belgium but all is slated for eventual return (FANC, 2009).

The national policy shift away from nuclear power halted reprocessing of irradiated nuclear fuel reprocessing in 2001, and any change in this status awaits further government action on waste management policy. A 2007 government study (Bossier et al., 2008) recommended a fundamental review of energy policy, particularly in the context of national CO_2 reduction commitments, including reconsideration of the nuclear phase out and extension of nuclear plant operating lifetimes. A subsequent report (GEMIX, 2009) recommended 10-year extensions to the lifetimes of the three oldest reactors by 10 years and 20-year extensions for the other four. While 10-year delay to the nuclear phase out was initially approved, a subsequent a change in government resulted in no change to the standing policy. National waste management strategy remains undecided, which means that used nuclear fuel Belgium is not classified as a waste. In the absence of active reprocessing, storage remains the exclusive option for managing used fuel inventories. Both wet and dry storage configurations are currently being used in Belgium (De Valkeneer and Dierick, 2001):

- Wet storage in the original reactor spent fuel pools;
- Dry storage in dual-purpose metal cask systems at the Doel nuclear plant (since 1995);
- Auxiliary wet storage at the Tihange nuclear plant site (since 1997); this auxiliary pool capacity extends that of the original facility.

ONDRAF/NIRAS's waste classification system, compatible with IAEA and EU international classification systems, was adopted to direct the long-term management of conditioned waste as well as to determine waste processing routes for unconditioned waste. The three categories (A, B, and C) are based on radiological (radionuclide activities in Bq and Bq/m^3) and heat-producing criteria, with a thermal output limit of 20 W/m^3 separating Categories B and C. Belgium's Category B is equivalent to IAEA's LL-ILW and Category C is equivalent to HLW.

The Belgian government has yet to decide formally on permanent disposal of HLW (and LL-LILW) in a geologic repository as the national policy for managing the back end of the fuel cycle. As a basis for a safety assessment associated with the disposal of these types of waste, the SAFIR-2 assessment report (ONDRAF/NIRAS, 2001) provides an estimate of 4,860 MTHM for the total amount of conventional (UO_2) irradiated nuclear fuel at the end of 40 years, assuming a reference burn-up of 45 GWd/MTHM for each reactor, together with 70 MTHM of irradiated MOX fuel.

Thereafter, two management options were considered:

- *Complete reprocessing of all conventional irradiated nuclear fuel*: This option results in 3,920 containers of HLW and 6,410 containers of structural waste (hulls, and end pieces) from irradiated nuclear fuel assemblies. 70 MTHM of irradiated MOX fuel must be added to this inventory.
- *Direct disposal*: For this option, reprocessing ceases at the end of the current contractual processing period covering 630 MTHM of used fuel. This option yields an estimated 420 containers of HLW, 820 containers of structural waste, 4,230 MTHM of irradiated nuclear fuel, and 70 MTHM of irradiated MOX nuclear fuel.

The above estimates correspond to a significant reduction in overall inventory and associated volume of wastes for geologic disposal compared with the estimates provided in the original 1989 SAFIR report (ONDRAF/NIRAS, 1989). Even though the latter assumed complete reprocessing of all types of irradiated nuclear fuel, the overall revised waste volume for geologic disposal dropped from 27,000 m^3 (ONDRAF/NIRAS, 1989) to either 10,000 m^3 for the complete conventional reprocessing option or 12,500 m^3 for the direct disposal option (ONDRAF/NIRAS, 2001). This decrease is attributed to better estimates and improved conditioning methods for the structural waste.

2.2 Institutional Arrangements

2.2.1 Institutional Framework

Figure 2-1 shows the key players and the current organizational structure in Belgium concerning radioactive waste management. Electrabel, a subsidiary of GDF Suez, is the country's principal nuclear reactor owner and operator. The Electrabel subsidiary Synatom manages fuel cycle operations for the Belgian nuclear fleet. While Synatom is a commercial (i.e., non-governmental) entity, the Belgium government does retain certain rights in governance of the organization. Principal roles of organizations are listed below.

POLICY and OVERSIGHT - The Ministry of Energy, as well as providing oversight of the implementer, establishes energy policy subject to ratification by parliament. The Ministry of the Interior provides oversight on the regulatory side.

IMPLEMENTER - The implementing entity responsible for all aspects of radioactive waste management (including conditioning, storage, transportation, and disposal) is the Belgian National Agency for Radioactive Waste and Fissile Materials (ONDRAF/ NIRAS), established by law in 1980. ONDRAF/NIRAS is supervised by the federal Minister of Energy and reports annually to Parliament The public nature of the Belgian implementer is intended to ensure that public interest is integral to waste management decisions.[7]

[7] In 1991, the name of the organization changed to the Belgian National Agency for Radioactive Waste and *Enriched* Fissile Materials, but ONDRAF/NIRAS remains in use as the organization's acronym.

ONDRAF/NIRAS subcontracts industrial waste management activities to its wholly-owned subsidiary Belgoprocess, also shown in Figure 2-1. Belgoprocess operates radioactive waste conditioning and storage facilities at Belgium's major nuclear industrial installations: Mol and Dessel.

REGULATOR – The Federal Agency for Nuclear Control (FANC) is the nuclear regulatory authority in Belgium and operates under the Ministry of Interior. Within FANC, the Facilities and Waste Division, oversees the review of nuclear facility license applications, including disposal facilities. FANC establishes the operational requirements of nuclear facilities in the licenses, which are issued formally by the government.

Prior to 2007, Association Vinçotte Nuclear (AVN) served as the authorized inspection arm of FANC. Following FANC restructuring in 2007, this activity is now carried out by the 'Bel V' foundation, a subsidiary of FANC. As shown in Figure 2-1, two independent organizations now support FANC, Bel V and AVC (health physics support).

ADVISORY and SUPPORT - Of the research organizations that support ONDRAF/NIRAS, the Belgian Nuclear Research Center (SCK•CEN) is the principal research entity addressing long-term management and geologic disposal of HLW and LL-ILW, since 1974.

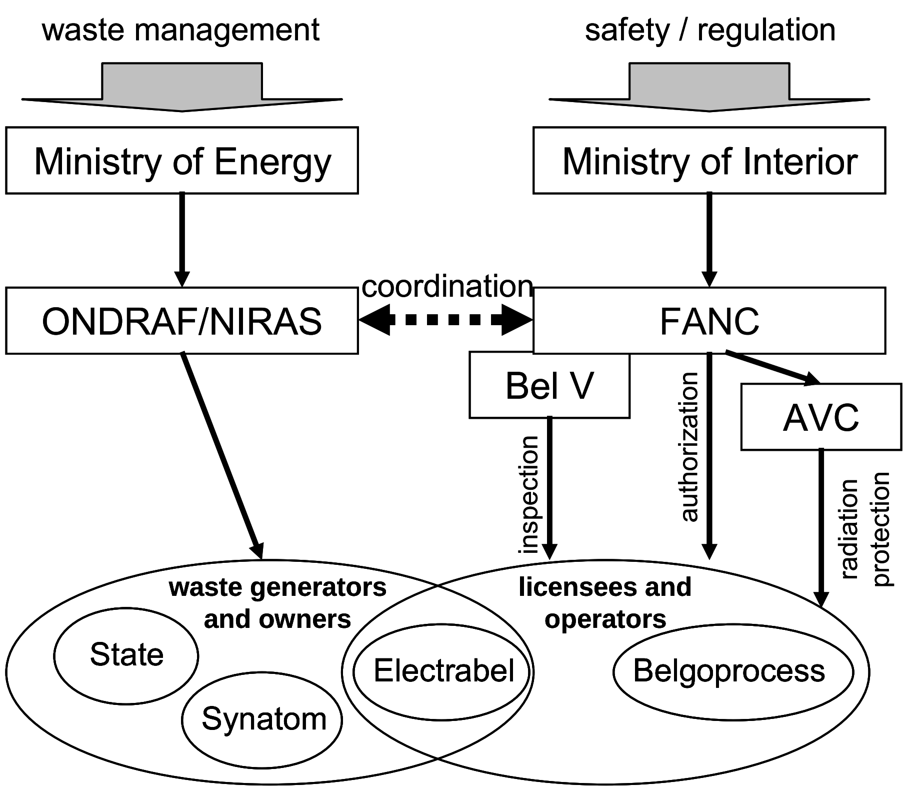

Figure 2-1
Organizational structure associated with radioactive waste management in Belgium (adapted from FANC, 2009).

2.2.2 Legal and Regulatory Framework

The principal governing regulation in Belgium for HLW management is the Royal Decree of July 20, 2001 (GRR-2001). This regulation has broad application to all nuclear activities in Belgium and specifies an annual dose limit of 1 mSv/yr to members of the public from all controllable practices and sources. No timeframe is specified in GRR-2001 with respect to radiological assessments and facility performance evaluations. New regulations governing radioactive waste disposal activities are reported to be under development (NEA, 2010).

2.2.3 Waste Classification

ONDRAF/NIRAS has adopted a waste classification system for conditioned waste, aimed at directing the long-term management of radioactive waste. The three main cateories are defined according to radiological and thermal power criteria:

- Category A: Radioactive waste that contains activities of radionuclides sufficiently low and with half lives sufficiently short to be compatible with near-surface disposal.
- Category B: Radioactive waste that does not meet the criteria for Category A but also does nt generate sufficient heat to be Category C.
- Category C: Radioactive waste that contain large amounts of alpha- and beta-emitting radionuclides and generates a thermal output > 20 W/m^3.

2.2.4 Funding

The law establishing ONDRAF/ NIRAS also stipulated that costs associated with radioactive waste management in Belgium are covered by the waste producers. In the case of nuclear power generation, utilities pay into a dedicated decommissioning and waste management fund administered by Electrabel's fuel cycle management subsidiary, Synatom – a commercial entity with some limited governmental oversight (WNA, 2010).[8]

2.3 Site Screening, Selection, and Characterization

2.3.1 Early Studies

Geological studies were initiated by SCK•CEN over 35 years ago into the suitability of formations within Belgium for the disposal of Categories B and C wastes (Volckaert and Ignatov, 2006). From an inventory of possible host formations based predominantly on expert review and opinion, an Evaluation Commission for Nuclear Energy concluded in 1976 that the deep argillaceous layers offered the most promising candidate host geology in Belgium for deep disposal and recommended more detailed research.

[8] As mentioned above, Synatom is a wholly-owned subsidiary of Electrabel, which is itself a subsidiary of GDF-Suez. However, in spite of this non-governmental pedigree, the Belgium government is reported to have special rights in the governance of the organization.

Independently in 1978, the European Commission started to compile an inventory of geological formations in Europe that might have suitable characteristics. The basis for this inventory of suitable formations was via exclusion criteria based solely on physical type of rock (clay, salt or granite), depth and thickness of formations. Within this EC program, in agreement with the Belgian study, argillaceous rocks were identified as being potentially suitable in Belgium, from which two main groups were recognized:

- Formations comprising hard rocks (shales) of the Paleozoic; and
- Formations comprising poorly consolidated, plastic formations of the Cenozoic, notably Boom Clay and Ypresian Clays.

Because only sparse information was available at the time on the characteristics of the hard rocks at depth, the preliminary results from the Boom Clay made it a strong candidate formation within the Belgian program.

Thereafter, the choice of the Mol-Dessel area was reinforced by practical considerations, in particular the ready access to the subsurface for detailed geological investigations due to the area being designated as a "nuclear zone,"[9] and the existence of a well-developed technical infrastructure covering a range of scientific disciplines. Soon after, in 1978, a 3-D seismic survey was carried out in the Mol-Dessel nuclear zone to complement the results from the geological review studies. The results helped to confirm the suitability of the site.

One of the primary concerns regarding the Boom Clay was the feasibility of excavating and developing underground infrastructures, especially at depths of 200 m in such clays. Thus, the decision was made relatively early to construct an underground R&D facility (High-Activity Disposal Experimental Site, or HADES). This underground research laboratory (URL) was constructed on and under the premises of SCK•CEN at Mol-Dessel. Construction began in 1980, and the URL was completed in 1983.

2.3.2 Detailed Geological and R&D Studies – Underground Laboratory

The more detailed studies associated with Belgium's deep disposal program included investigations into structural discontinuities and heterogeneities within the Boom Clay. Primary techniques of investigation were geophysical methods (wireline logging in boreholes) or surface 2-D seismic reflection (~65 km of profile data), with additional testing of core material. Textural heterogeneities were studied using a variety of borehole logging tools providing high vertical resolution, including resistivity and nuclear magnetic resonance measurements. Such techniques were able to show the banded layers and different grain textures typical of the Boom Clay (Figure 2-2).

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[9] Nuclear facilities in Belgium are constructed within a nuclear zone, which is associated with special planning and development requirements, including an exclusion zone.

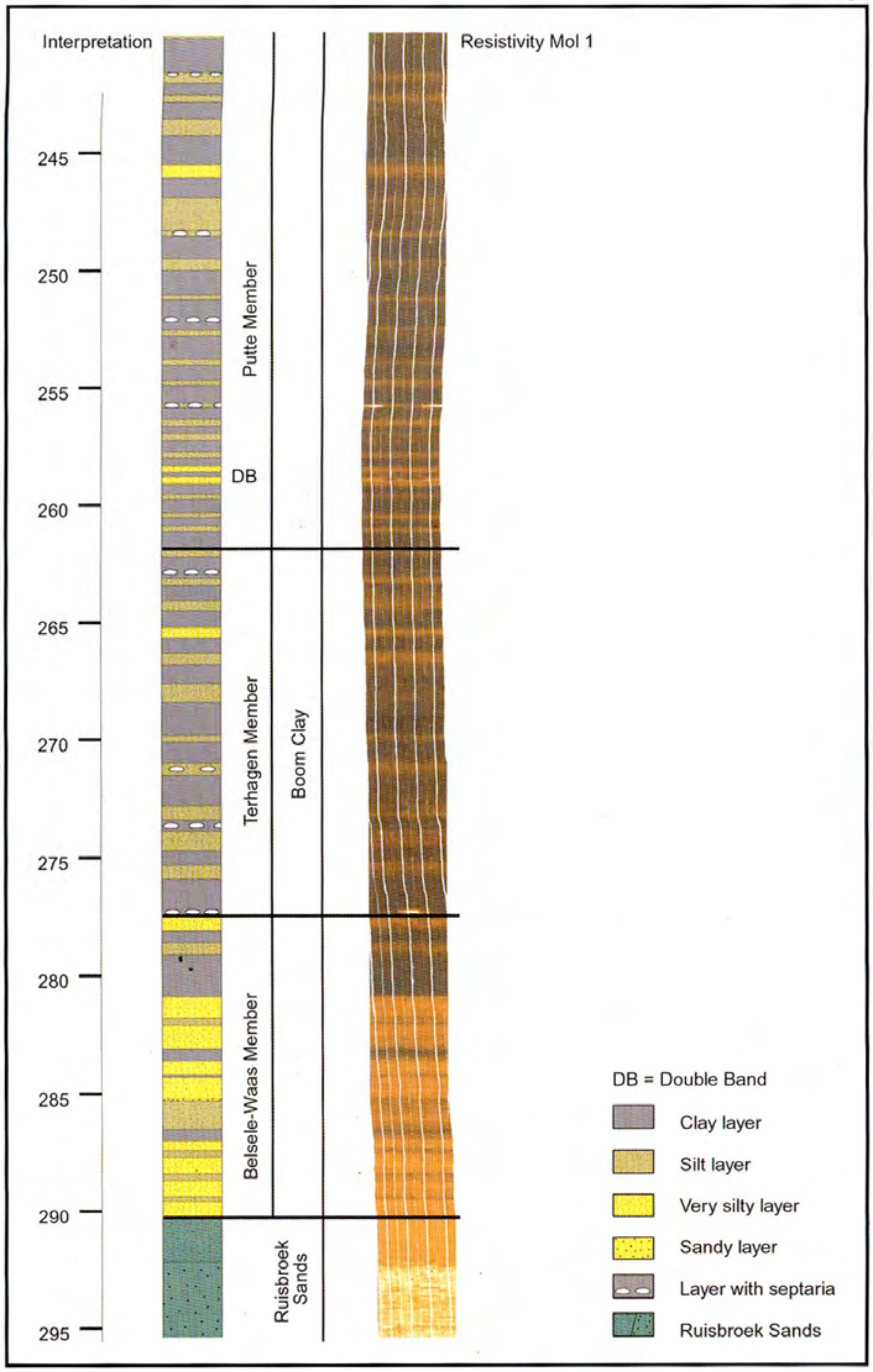

Figure 2-2
Lithostratigraphic profile of Boom Clay based on high vertical resolution logging (ONDRAF/NIRAS, 2001). Used with permission of ONDRAF/ NIRAS.

The first borehole (Dessel-1) was drilled to 613 m and the second (Mol-1) to 572 m. The Mol-1 borehole was also cored between 150 m and 332 m in order to examine the entire Boom Clay column and surrounding formations. Vertical seismic profiles were carried out in both boreholes to correlate seismic data, expressed as a function of time, with borehole data, expressed as a function of depth.

The Boom Clay has also been studied outside the geologic disposal program, in outcrop clay pits to the south and west of Dessel, as well as taking advantage of excavations associated with an underground tramway and road tunnel.

Since construction of the underground facility, the HADES URL has been the site of a range of detailed *in situ* investigations. Further stages of construction of the URL involved (Volckaert and Ignatov, 2006):

- 1983-1984: Experimental shaft and gallery;
- 1987: Extension of the URL - Test Drift;
- 1997-1999: Second shaft;
- 2001-2002: Connecting gallery between two shafts.
- 2005-2006: PRACLAY (PReliminaAry demonstration test for CLAY disposal) gallery, site of the feasibility demonstration.

This early work culminated in the submission of SAFIR (Phase 1), the first safety assessment and feasibility report (ONDRAF/ NIRAS, 1989; assessments are discussed in detail in Section 6.1), on the reference solution of deep geologic disposal of Categories B and C wastes in Boom Clay. As stated previously, disposal of such wastes has not yet been accepted as national policy in Belgium.

The R&D work under Phase 2 of the disposal program evolved towards large-scale integrated demonstration *in situ* tests. Activities since 1995 have been focused on the PRACLAY project, aimed at demonstrating the concept for the disposal of HLW in clay. In 1995, SCK•CEN and ONDRAF/NIRAS set up an Economic Interest Group (EIG PRACLAY). Since 2000, the EIG has been responsible for management of all activities taking place in the integrated underground infrastructure of HADES. EIG's name was subsequently changed to EURIDICE (European Underground Research Infrastructure for the Disposal of waste in a Clay Environment), to reflect the international nature of the collaboration.

Elsewhere, in support of the Belgian HLW waste management program, SCK•CEN continues its partnership within the Mt. Terri project (Switzerland) with active contributions to the diffusion and gas migration experiments of the Opalinus Clay, a potential host formation in Switzerland.

2.3.3 Key Features of Boom Clay Formation

Key properties of clay as a potential host medium for geologic disposal include:

- Low hydraulic conductivity;
- Plastic character, thereby enabling self-healing properties;
- High retention capacity (for radionuclides).

The Boom Clay is a silty clay or argillaceous silt with relatively high pyrite and glauconite (mica group, typical of slow rates of accumulation) content in its siltiest bands. The Clay is characterized by its banded structure indicating differences in the sedimentation environment as well as carbonate and organic matter contents. The thickness of the formation varies from ~30 m in the outcrop zone to ~100 m in the north-east of the country. The formation is relatively flat, with a 1% to 2% dip towards the northeast. In the Mol-Dessel region, the depth is between 190 m and 290 m beneath the surface (see Figure 2-3).

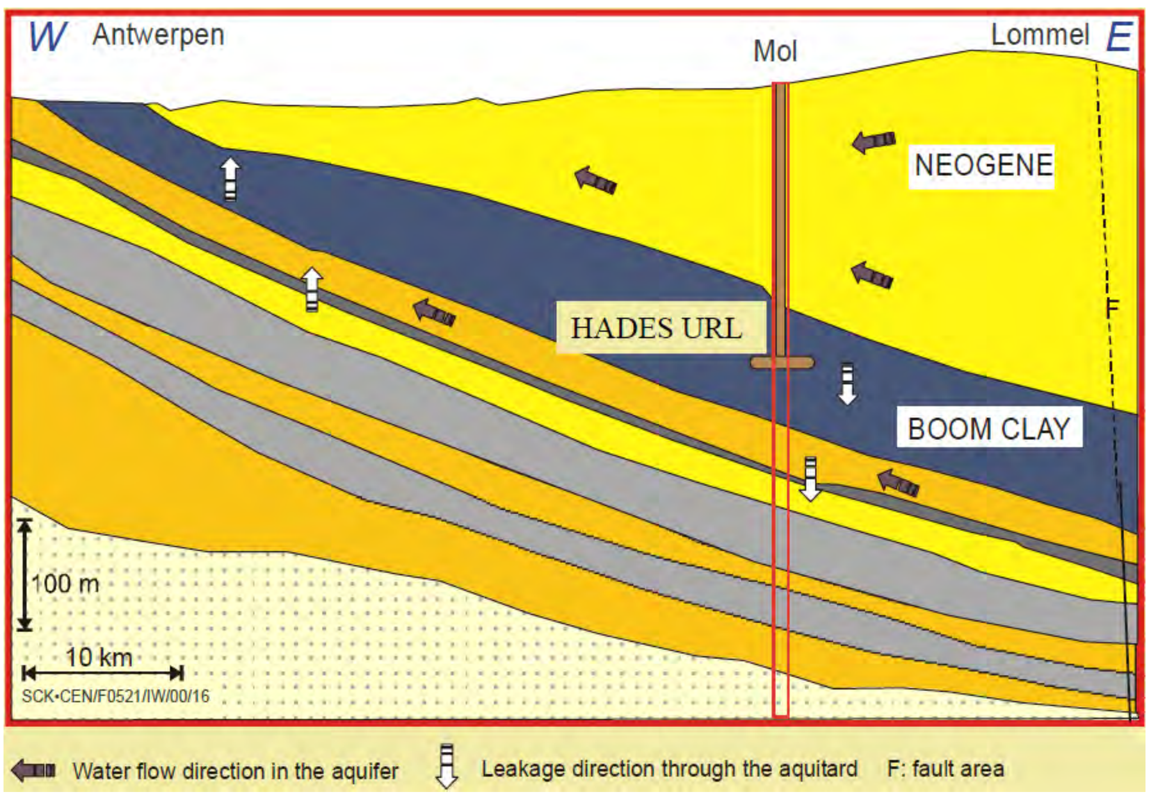

Figure 2-3
Boom Clay at the Mol site, between 190 m and 290 m. The location of the HADES URL is also shown in the diagram (Bernier and Demarche, 2006). Used with permission of ONDRAF/NIRAS.

2.3.3.1 Mineralogy of Boom Clay

Boom Clay is characterized by a fairly constant chemical and mineralogical composition (Vandenberghe, 1978). The clay content varies between ~30% and 70%, with an average of 55%. The predominant clay mineral is illite, followed by lesser amounts of smectite and kaolinite. Thereafter, quartz is the main mineral, with other minerals containing varying amounts (1% to 5%) of feldspars, carbonates and pyrite. The organic content varies from 1% to 3%. Total porosity is in the range 30% to 40%, but most of the water content is tightly bound to the clay minerals.

2.3.3.2 Geochemistry

Geochemical conditions in the Boom Clay are favorable for retention of most radionuclides, with mildly alkaline (pH = 8.2) and reducing (Eh = -0.25 V) conditions. With respect to natural radioactivity, thorium and uranium nuclides are essentially in equilibrium, suggesting restricted mobility over a time period of about one million years. Radium nuclides indicate variable mobility but where it appears to be mobile, its mobility is associated with the presence of silty layers.

2.3.3.3 Tectonic Characterization

The Basin in which the Boom Clay is located shows virtually no large-scale structural features, although the north-east part of the basin is disturbed by a complex system of faults – the Roermond Graben. Subsidence rates in the middle of the graben region have averaged 0.25-0.5 mm per year over the last 2 million years. The two major faults are located 5 km and 7 km from the Mol-Dessel zone.

Historical records indicate that the largest earthquake in the area over the past 200 years, the Roermond quake, occurred in 1992, with a magnitude of 5.8 on the Richter scale, and with an estimated origin at 17 km depth. Studies during the detailed investigations of the 1990’s into the occurrence of earthquakes suggested the possibility of greater magnitude earthquakes occurring and with higher frequency, and a reassessment by ONDRAF/NIRAS of the seismic risk for the region has been carried out (Wouters, 2006).

2.3.4 Phase 2 Research and Development Program

Following submission and review of SAFIR, the main objectives for the second phase of R&D associated with disposal established by ONDRAF/ NIRAS addressed the recommendations of the Nuclear Safety Commission in a number of areas, based on its review of SAFIR; in particular:

- *Waste characterization*, in particular the establishment of a national inventory taking into account the non-reprocessing option for irradiated nuclear fuel;

- *In-depth characterization of host formation*, including faults and heterogeneities in the Boom Clay, understanding of the thermal-hydrogeological-mechanical (THM) behavior of the clay, and improved understanding of the regional and local hydrogeology in and around the Mol-Dessel zone. In addition, a preliminary characterization of Ypresian Clays below the Doel nuclear zone.
- *Development of repository design*, including layout to maximize the thickness of undisturbed clay and separation of different categories of radioactive waste, an excavation demonstration, and specific sealing methods. Also implement the PRACLAY experiment – a full-scale demonstration of HLW emplacement and study of thermal impact at large scale.
- *Develop better understanding of key interactions within repository*, in particular gas generation and migration, and the behavior of different waste types.
- *Assessment of long-term safety*, both radiological and chemical toxicity impacts, and the impact of organic matter and chemical plumes from the EBS (cement/concrete) on the behavior of safety-relevant radionuclides.
- *More accurate cost analysis*.

Figure 2-4 shows a schematic of the various PRACLAY experiments underway to study the thermal impact of HLW.

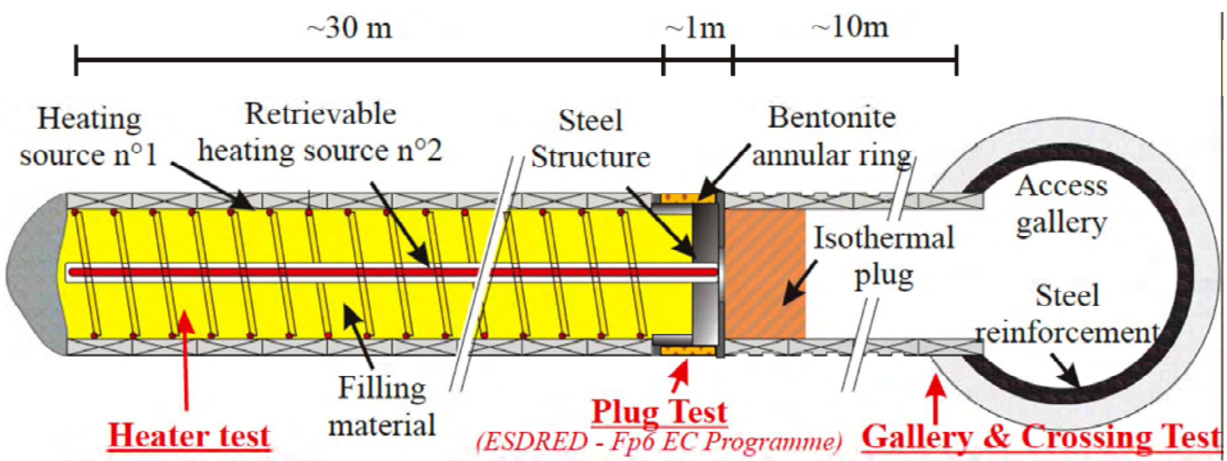

Figure 2-4
The in situ PRACLAY thermal experiments coordinated by EURIDICE (Bernier and Demarche, 2006). Used with permission of ONDRAF/NIRAS.

The main objectives of the 10-year heater test are to:

- Test a configuration similar to that in a real repository;
- Determine the combined effects of the excavated damaged zone (EDZ) and the thermally-damaged zone (TDZ);
- Verify / confirm understanding of the THMC behavior of Boom Clay;
- Demonstrate that the specified thermal load does not impact the performance of the Boom Clay.

2.4 Disposal Concept

2.4.1 General Philosophy

Figure 2-5 shows schematically the primary long-term safety functions recognized by ONDRAF/NIRAS as being essential for safe deep disposal:

- Physical containment – isolation from the surrounding environment, in particular water;
- Controlled (delayed) release of radionuclides in the near-field combined with dispersion;
- Dilution and dispersion in the geosphere and biosphere – reduction in radionuclide concentrations primarily by mixing with groundwater and/or surface waters;
- Restricting access to minimize the possibility and consequences of human intrusion.

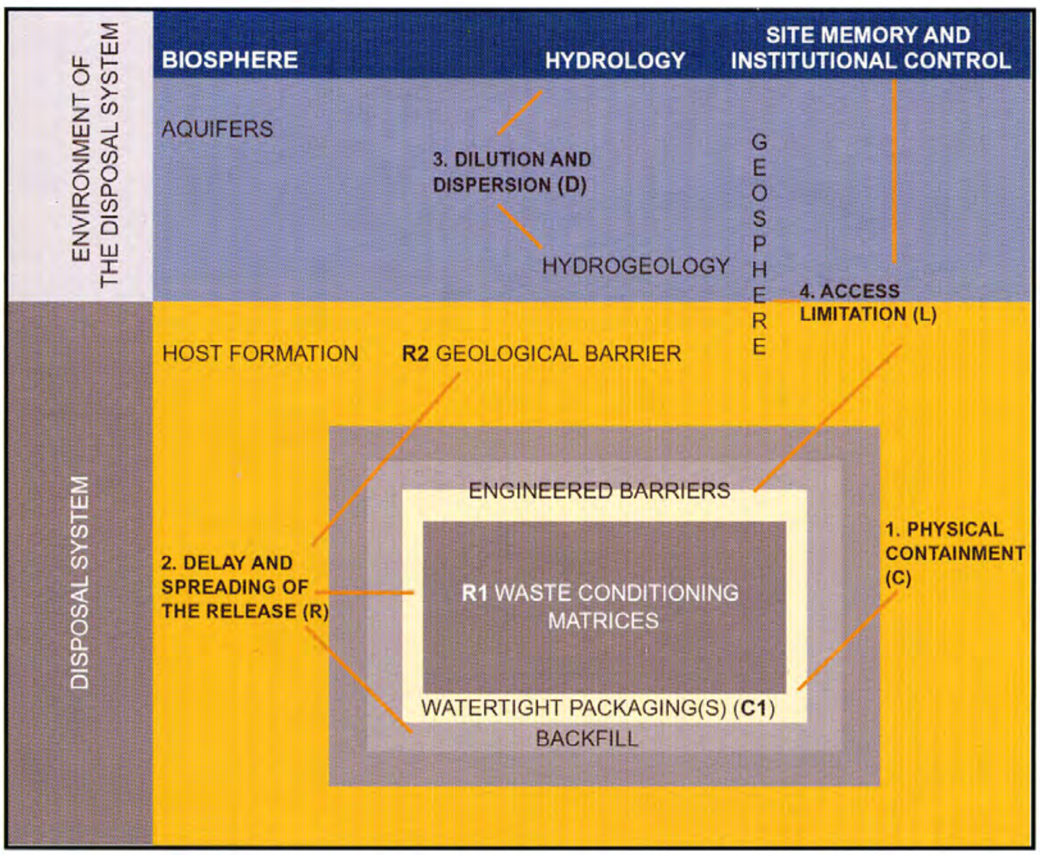

Figure 2-5
Long-term safety functions of the deep disposal system considered in SAFIR-2. These functions were addressed in the assessment of the normal-evolution scenario (ONDRAF/NIRAS, 2001). Used with permission of ONDRAF/NIRAS.

In terms of timeframes over which the safety functions are expected to operate, Figure 2-6 shows the relevant relationships within the Belgian reference concept.

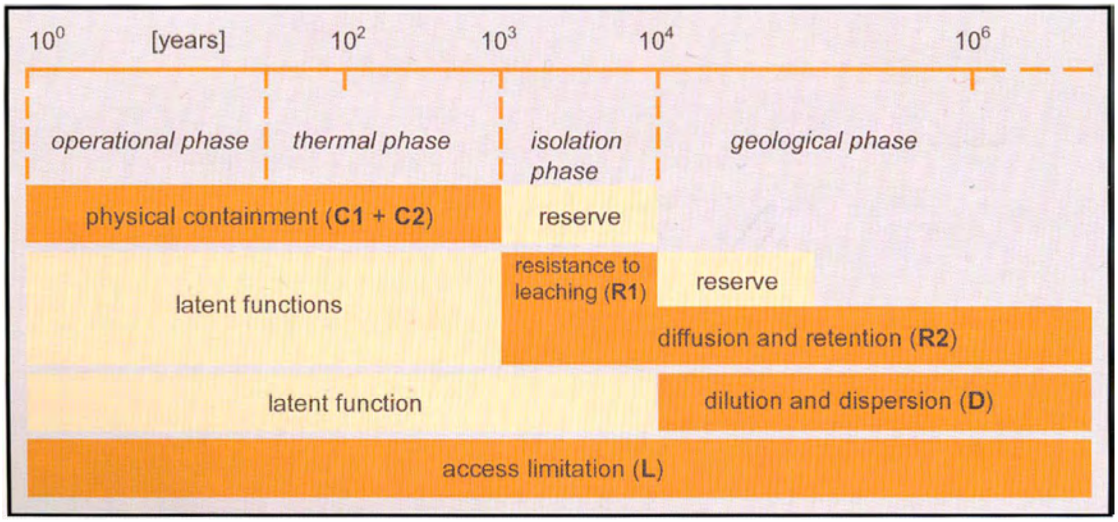

Figure 2-6
Timeframe associated with the different safety functions of the reference geologic repository (ONDRF/NIRAS, 2001). Used with permission of ONDRAF/NIRAS.

While emphasizing that disposal implies no intention to retrieve waste, ONDRAF/NIRAS discussed the possibility of retrievability, identifying certain aspects of the repository design that allow retrievability over a limited period of time, e.g. the nature of the HLW overpack and maintain open access routes to the disposal drifts.

With regard to the Boom Clay formation, the overall repository design seeks to at maximize the thickness of clay surrounding the repository and minimize perturbations to the surrounding clay. These perturbations include heat, radiation (radiolysis), gas generation from corrosion and the pressure increase associated with this gas, geochemistry and the effect of the alkaline plume, and mechanical disturbance from excavations.

2.4.2 Repository Layout

Two reference designs formed the basis of the SAFIR-2 assessment: one for HLW and one for irradiated nuclear fuel. Figure 2-7 shows the reference design for the repository layout for HLW (ONDRAF/NIRAS, 2001). To accommodate the HLW waste packages two main galleries service eight disposal galleries, each 800 m long and 2 m in diameter, at right angles to the main galleries. Each disposal gallery is divided into three segments:

- Two segments of 200 m outside the main galleries;
- One 400 m segment between the other two segments.

The first disposal gallery is 100 m from the connecting gallery and adjacent galleries are spaced at 40 m intervals.

Figure 2-8 shows the corresponding design for the repository for irradiated nuclear fuel, adapted from the design for HLW. The main differences (reduced angle between main gallery and disposal galleries) are intended to accommodate the longer length of waste package (5 m for irradiated nuclear fuel vs. 1.6 m for HLW) as well as the different thermal output. Thereafter, a cross-section of the EBS design (Figure 2-9) demonstrates the multi-barrier principle of the Belgian disposal concept.

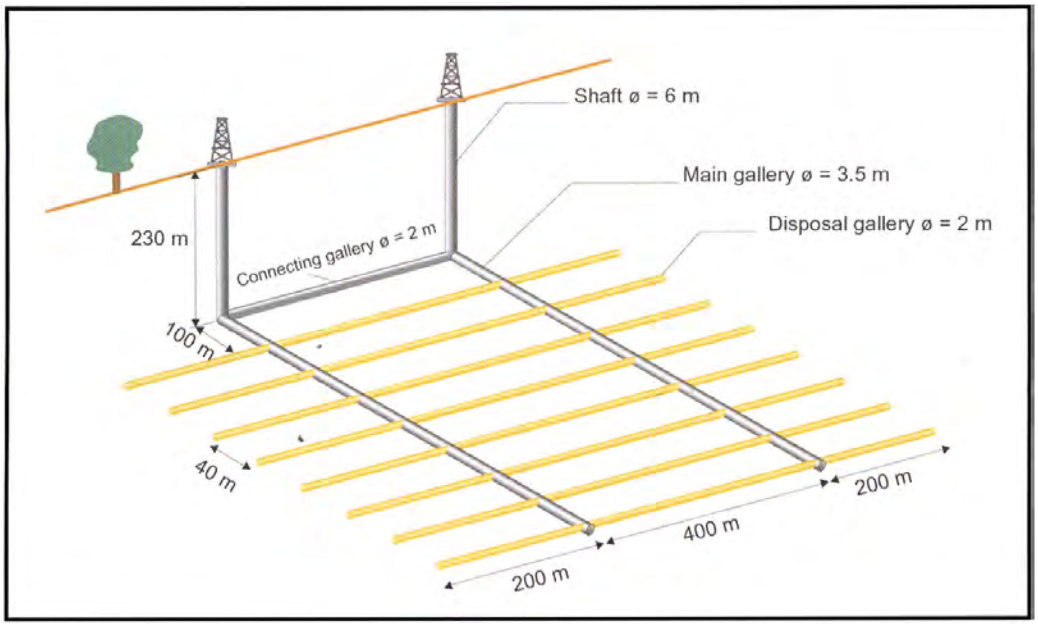

Figure 2-7
Reference repository layout for geological repository for HLW (ONDRAF/NIRAS, 2001). Used with permission of ONDRAF/NIRAS.

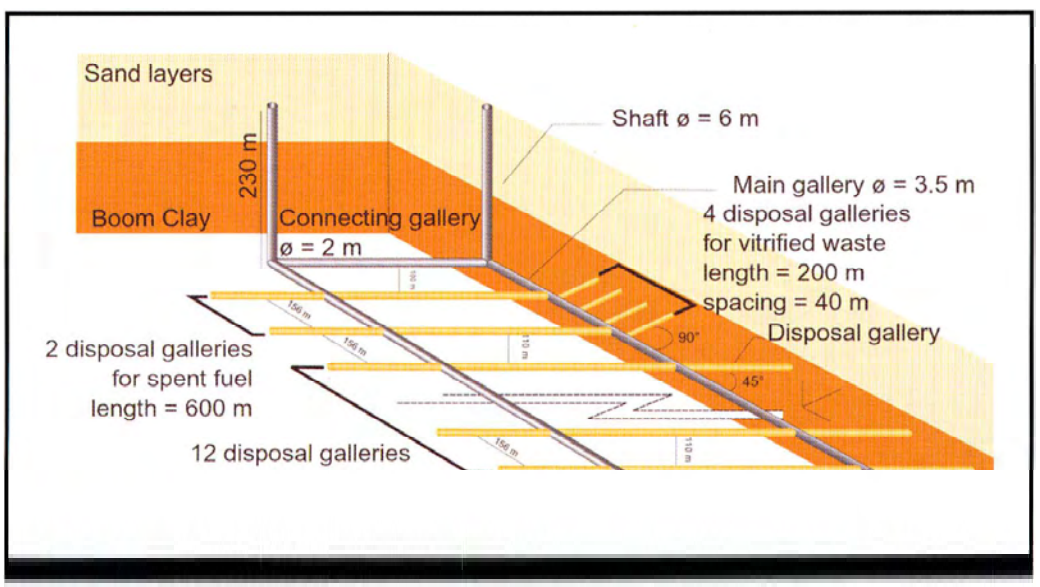

Figure 2-8
Reference layout for repository for irradiated nuclear fuel (ONDRAF/NIRAS, 2001). Used with permission of ONDRAF/NIRAS.

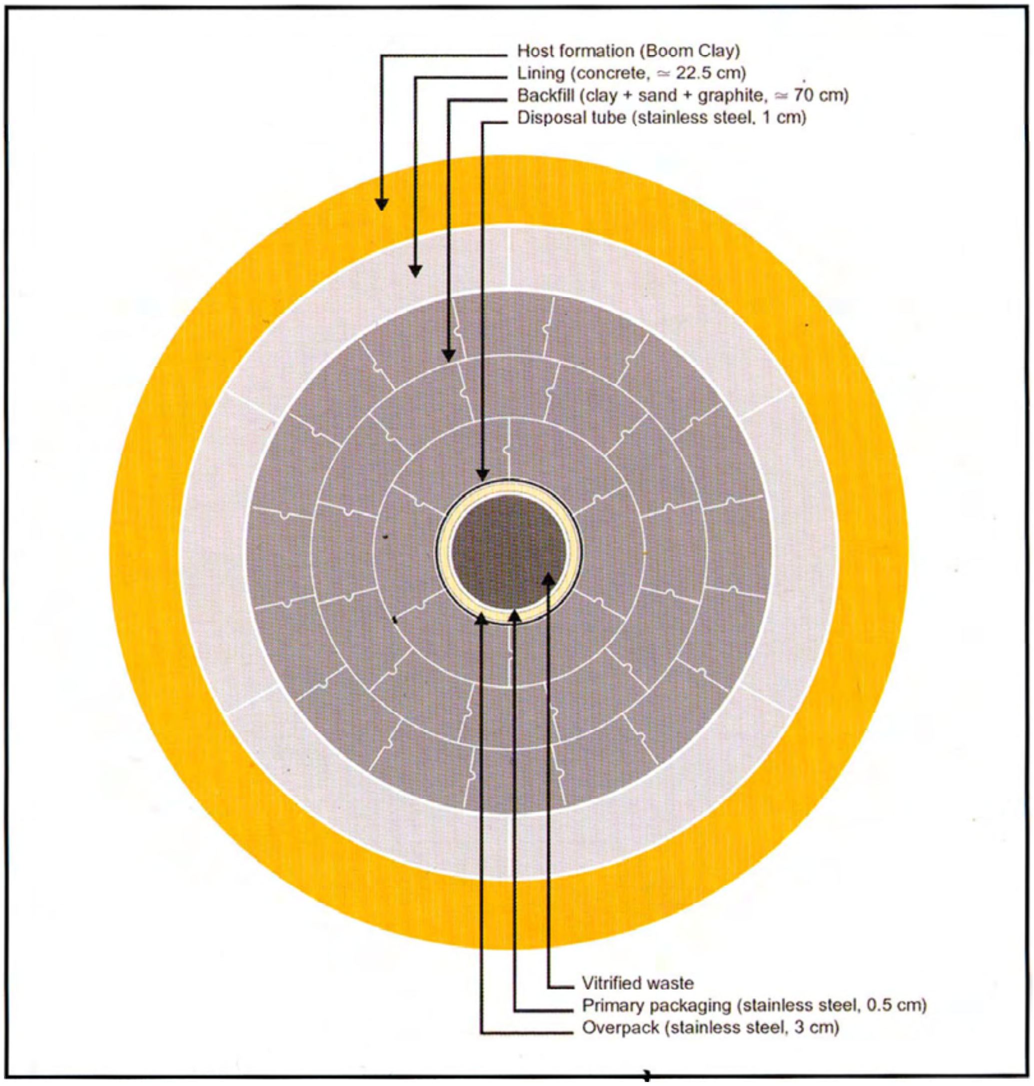

Figure 2-9
Cross section through disposal gallery for HLW, showing the multi-barrier principle (ONDRAF/NIRAS, 2001). Used with permission of ONDRAF/NIRAS.

2.5 Transparency and Stakeholder Involvement

2.5.1 Public Involvement

The draft National Waste Plan of 2010 is expected to focus on the long-term management of LL-ILW and HLW. A decision on the Plan is necessary to allow ONDRAF/NIRAS to continue with its waste management responsibilities in this area. According to the current program, public consultations using selected participation methods were scheduled to start in January 2009 (FANC, 2009), allowing opportunities for consultations with the general public. Furthermore, any license application (see Section 2.6.2) involves the preparation of an environmental impact statement (EIS), which also requires extensive communication and engagement with the public.

2.5.2 International Involvement

Belgium's Minister of the Interior released the SAFIR-2 report for international peer review by an NEA Expert Group. Both this external group and ONDRAF/NIRAS's own R&D work did not find any potential problem that might impact or prevent the disposal of HLW in Boom Clay.

The capabilities of both ONDRAF/NIRAS and SCK•CEN were integrated towards the end of the 1990's, resulting in the creation of EURIDICE. The mission of this consortium is to manage the present extension of the URL and the underground experimental program. The latter includes the PRACLAY project (Section 2.3.4) aimed at demonstrating the overall safety and feasibility of deep geological disposal of HLW in a clay formation. The emphasis of EURIDICE is on large-scale engineering and demonstration tests, e.g., shafts and access galleries, disposal galleries, the THMC behavior of Boom Clay, and construction, handling and emplacement of engineered barriers. Many countries and national waste management agencies have participated in research studies at the HADES URL including in particular France (ANDRA), Spain (ENRESA) and Japan (JAEA).

2.6 Safety Assessment and Licensing

2.6.1 Safety Assessment

Two safety reports discuss the philosophy and disposal concept for the reference HLW management solution in Belgium:

- Safety Assessment and Feasibility Interim Report (SAFIR); and
- Safety Assessment and Feasibility Interim Report 2 (SAFIR-2).

The first report (SAFIR) only considered the long-term safety of a geologic repository for HLW, whereas the subsequent assessment took into account other types of waste, i.e. Categories B and C, that were likely to be disposed of in a deep repository. The emphasis, however, was on HLW, irradiated nuclear fuel and the highly-active hulls and end pieces.

2.6.1.1 Scenario Development

ONDRAF/NIRAS's scenario development work generated two groups of scenarios:

- *Normal-evolution scenario*, or reference scenario, which takes into account all FEPs that are certain to occur; therefore, describing the expected evolution of the repository following closure, including future events that would result in radiological exposures.
- *Altered-evolution scenarios*, which address possible perturbations that are unlikely but capable of affecting the disposal system if they occur, resulting in radiological exposures.

In carrying out its assessments, ONDRAF/NIRAS acknowledge three types of uncertainty: scenario uncertainty, conceptual model uncertainty, and uncertainty in the values of model parameters.

2.6.1.1.1 Normal-evolution Scenario

The key elements of the normal-evolution scenario are shown in Figure 2-10.

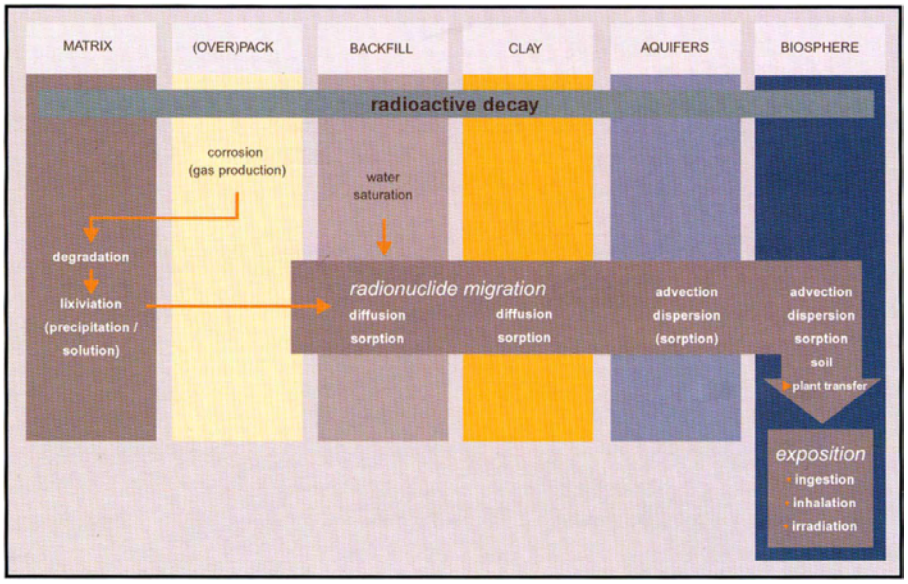

Figure 2-10
Key components of the repository system and key processes taken into account in the normal-evolution scenario (ONDRAF/NIRAS, 2001). Used with permission of ONDRAF/NIRAS.

2.6.1.1.2 Altered-evolution Scenarios

The altered-evolution scenarios addressed perturbations resulting from:

- Human intrusion – resource exploitation drilling;
- Anthropogenic climate change;
- Fault activation;
- Severe glaciation;
- Inadequate sealing;
- Premature failure of an engineered barrier.

2.6.1.2 Results for Normal-Evolution Scenario

Only those radionuclides with sufficiently long half-lives (including relevant decay chains) were selected for assessment calculations. Doses to an individual of the reference group (likely to receive the maximum dose) were calculated in three steps:

- Radionuclide migration simulations were carried out to determine he activity fluxes of radionuclides at the interface between Boom Clay and the aquifer above (Neogene Aquifer) were calculated based on different source term models for vitrified waste, used nuclear fuel, and hulls and end pieces;
- Simulations of radionuclide migration within the Aquifer were carried out to determine radionuclide activities in water pumped from a deep well immediately above the disposal facility as well as radionuclide activity fluxes to nearby rivers.
- The concentrations of radionuclides in water collected from the well were multiplied by the corresponding biosphere conversion factors.

The calculated total annual dose (Sv/year) from activation and fission products in HLW is shown in Figure 2-11. This figure shows that the maximum dose is ~10^{-5} Sv/year, attributable to Se-79. Other key (dose-contributing) radionuclides were found to be I-129, Sn-126 and Tc-99. The total peak dose from the actinides alone (not shown) is over 3 orders of magnitude less.

The corresponding results from irradiated nuclear fuel are shown in Figure 2-12. In this case the maximum dose is just below 10^{-5} Sv/year, again attributable mainly to Se-79. Other key (dose-contributing) radionuclides were found to be I-129, Cl-36, Sn-126 and Tc-99. Again, the total peak dose from actinides (not shown) is over 3 orders of magnitude less.

For hulls and end pieces (not shown), the peak dose is a factor of~30 less than that of HLW, mainly due to a lower Se-79 inventory.

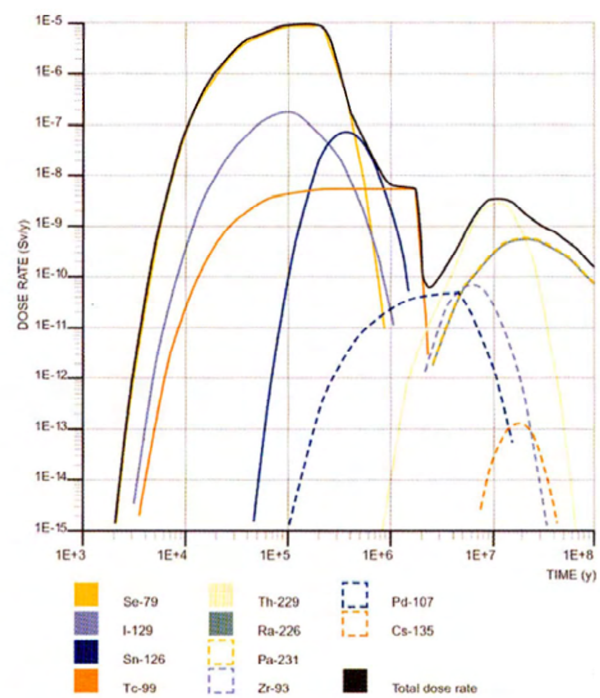

Figure 2-11
Total annual dose to an individual of the reference group via deep well from activation products and fission products in HLW (ONDRAF/NIRAS, 2001). Used with permission of ONDRAF/NIRAS.

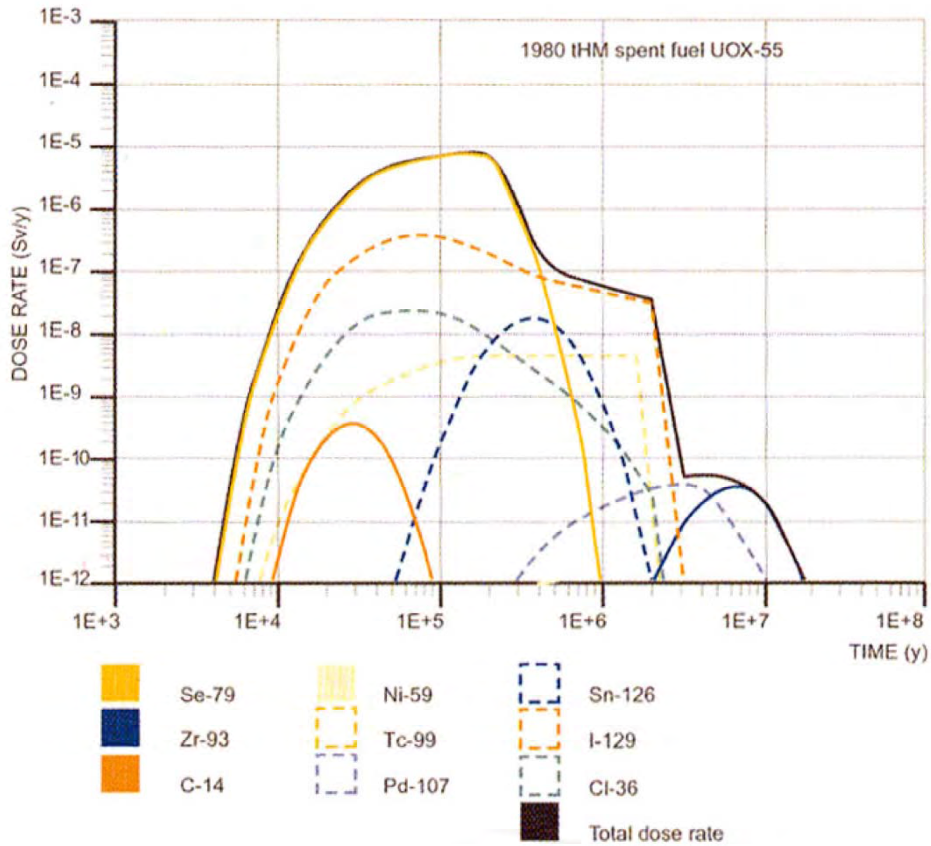

Figure 2-12
Total annual dose to an individual of the reference group via deep well from activation products and fission products in irradiated nuclear UO_2 fuel (ONDRAF/NIRAS, 2001). Used with permission of ONDRAF/NIRAS.

2.6.1.3 Overview of Assessment Work

ONDRAF/NIRAS's program has been centered around the characterization of argillaceous formations with two key formations / sites identified:

- *Boom Clay* (reference host formation) and the *Mol-Dessel* nuclear zone (reference site);
- *Ypresian Clays* (alternative host formation) and *Doel* nuclear zone (alternative site).

FANC (2009) emphasizes that the word 'site' used above does not imply disposal, in keeping with a feasibility study.

The key stages in ONDRAF/NIRAS's assessment work may be summarized as follows:

- *1980's*: Early in its disposal program (1984), ONDRAF/NIRAS announced the decision to prepare and submit a report documenting the systematic analysis of all R&D studies carried out between 1974 and 1989, following the recommendations of the Evaluation Commission for Nuclear Energy. This Commission had recommended that HLW must remain accessible and under control until a final management solution has been reached. Furthermore, an assessment of the risk associated with geologic disposal should be carried out every 10 years.

Thereafter, the important stages in the timeline of ONDRAF/NIRAS' assessment work was:

- *1989*: Five years after its decision to prepare a report, ONDRAF/NIRAS submitted the Safety Report (SAFIR) to the Secretary of State for Energy. The Secretary established a Safety Commission comprising Belgian and foreign experts to evaluate this report. ONDRAF/NIRAS also submitted a 5-year R&D program plan for review by the Safety Commission.
- *1990*: Based on its review, the Safety Commission recommended an expansion of ONDRAF/NIRAS's R&D program. Government gave its approval to start a 10-year 'methodological' R&D program (Phase 2), to establish the technical and economic feasibility of constructing a geologic repository for HLW and LL-ILW.
- *2001*: ONDRAF/NIRAS submitted SAFIR-2 Report, describing the technical and scientific advances that had been made during the preceding 10-year program, focusing on an assessment of the confidence in safety, feasibility and robustness of the reference system. No societal aspects were included in this report.
- *2002*: Review of SAFIR-2 by international peer group under the oversight of NEA.

The Evaluation Commission reviewed the 2001 assessment and feasibility report and concluded that the selection of the Boom Clay in the Mol-Dessel area is a good one.

It should be noted that ONDRAF/NIRAS does not regard the SAFIR assessments, especially SAFIR-2, as a true (formal) safety assessment, but rather the basis for in-depth discussions with FANC on what would be required for the eventual licensing of a facility.

2.6.2 Licensing Process

No decision has yet been made concerning geologic disposal of HLW and LL-ILW in Belgium. Accordingly, there is no specific licensing application or action to discuss. In general, the license application for Class I facilities, which would include a geologic repository, has to be accompanied by an environmental impact assessment (EIA), prepared according to European legislation[10]. For Class I facilities, safety must be re-evaluated every 10 years.

[10] European Directive 1985/337/EEG, as modified, and the Recommendation of the European Commission 1999/829/Euratom concerning the application of article 37 of the Euratom Treaty.

In its safety assessment work, ONDRAF/NIRAS took into account the effective dose limit laid down by the European Directive 96/29/EURATOM, i.e., an annual dose limit of 1 mSv/year to a member of the public (also specified in the GRR-2001 regulation), and also the recommendation of ICRP of 0.3 mSv/year as a dose constraint. In its assessment reports, ONDRAF/NIRAS notes that the average background dose in Belgium is 3.6 mSv/year, due mainly to natural sources.

2.7 Current Status

2.7.1 General Situation

Table 2-1 describes the current liabilities in Belgium associated with HLW and LL-ILW (FANC, 2009). Since the decision on future reprocessing is still pending, studies on the direct disposal of irradiated nuclear fuel in a clay formation have gained increasing interest. As discussed in Section 2.1.2, the volume of radioactive waste for geologic disposal was estimated at 10,000 m^3 or 12,500 m^3 depending on waste management option.

Table 2-1
Current responsibilities in Belgium for LL-ILW and HLW (adapted from FANC, 2009).

Type of Responsibility	Current Practice / Facilities	Long-term Management Policy	Funding of Responsibility	Planned Facilities
Irradiated Fuel *HLW*	On-site wet or dry storage Storage by Belgoprocess	Moratorium on reprocessing (NPPs only); long-term management policy (reprocessing or direct disposal) still being developed	NPPs contribute to fund managed by Synatom; various funds for historical responsibilities fed by state	Under investigation as reference solution: geologic repository in Boom Clay
Nuclear fuel cycle waste	Interim storage at Belgoprocess site	LL-ILW: No decision yet	Producer pays contribution to ONDRAF/NIRAS long-term fund.	Under investigation as reference solution: geologic repository in Boom Clay

In the present HLW management reference scenario, the operation of a geologic repository could start around 2040/2050 and be operational until 2070/2080 (closure phase). This timeline requires, in the case of HLW and irradiated nuclear fuel, a temporary surface storage of about 60 years, in order that the heat output is reduced significantly. Such a facility is now operational.

Given the existing policy situation in Belgium, work continuing in the area of geologic disposal is still regarded as research, development and demonstration (RD&D) of the feasibility of disposal in Boom Clay, including both technical and economic feasibilities, and "*without prejudging the site where such a solution would actually be implemented*" (FANC, 2009).

2.7.2 Current EBS Concept for HLW and Irradiated Nuclear Fuel

After publication of the SAFIR-2 (ONDRAF/NIRAS, 2001) and during the preparation of the PRACLAY *in situ* demonstration experiment, ONDRAF/NIRAS acknowledged the need for a more carefully-justified and fully-integrated repository design. As a result, the implementer developed three different repository designs specifically for HLW (Figure 2-13):

- Borehole;
- Supercontainer; and
- Sleeve.

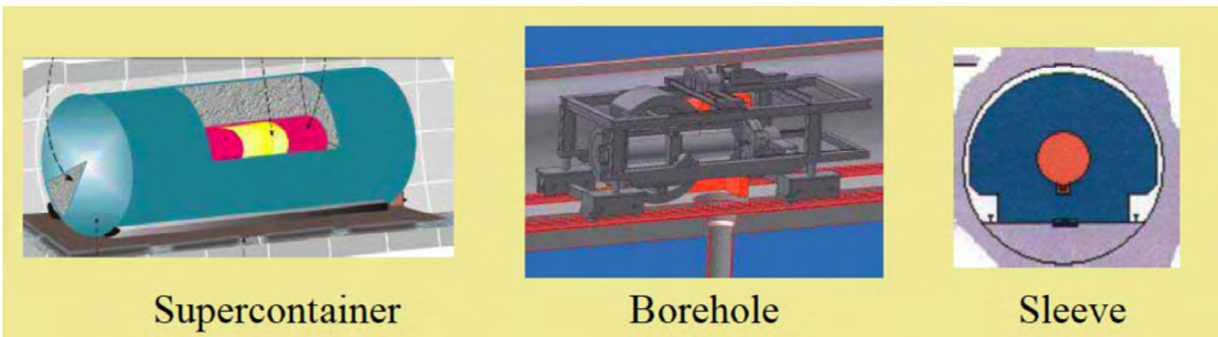

Figure 2-13
Three disposal design alternatives under consideration by ONDRAF/NIRAS (Bernier and Demarche, 2006). Used with permission of ONDRAF/NIRAS.

Of these, the Supercontainer concept was selected as the most promising design, although in parallel, a preliminary design for LL-ILW (Category B waste) was also being developed, leading to the integrated concrete *Monolith* (Bel *et al.*, 2005).

2.7.2.1 Supercontainer Concept

The primary motivation for the Supercontainer was logistics and the ability to contain HLW in an integrated waste disposal package constructed completely above ground.

The Supercontainer concept (Figure 2-14) comprises:

- Vitrified HLW canister or irradiated nuclear fuel assemblies placed inside a ~30 mm thick carbon steel overpack with end pieces 60 mm thick.
- Overpack surrounded by a high alkaline environment of concrete buffer.

The container has a Portland cement buffer for both HLW and irradiated nuclear fuel, primarily to maintain an alkaline chemical environment, modeled to last for thousands of years. The buffer also has a low hydraulic conductivity and provides radiation shielding. The design has the surface of the overpack passivated to inhibit corrosion. The carbon steel is considered to have predictable corrosion properties, thereby providing the necessary minimum overpack lifetime of 500 years for vitrified HLW and 2,000 years for irradiated nuclear fuel.

Current R&D efforts are aimed at addressing some of the uncertainties in the Supercontainer disposal concept (Bel *et al.*, 2006). In addition, part of ONDRAF/NIRAS's work is directed towards helping to influence government policy by informing and gaining society acceptance of the reference HLW management solution.

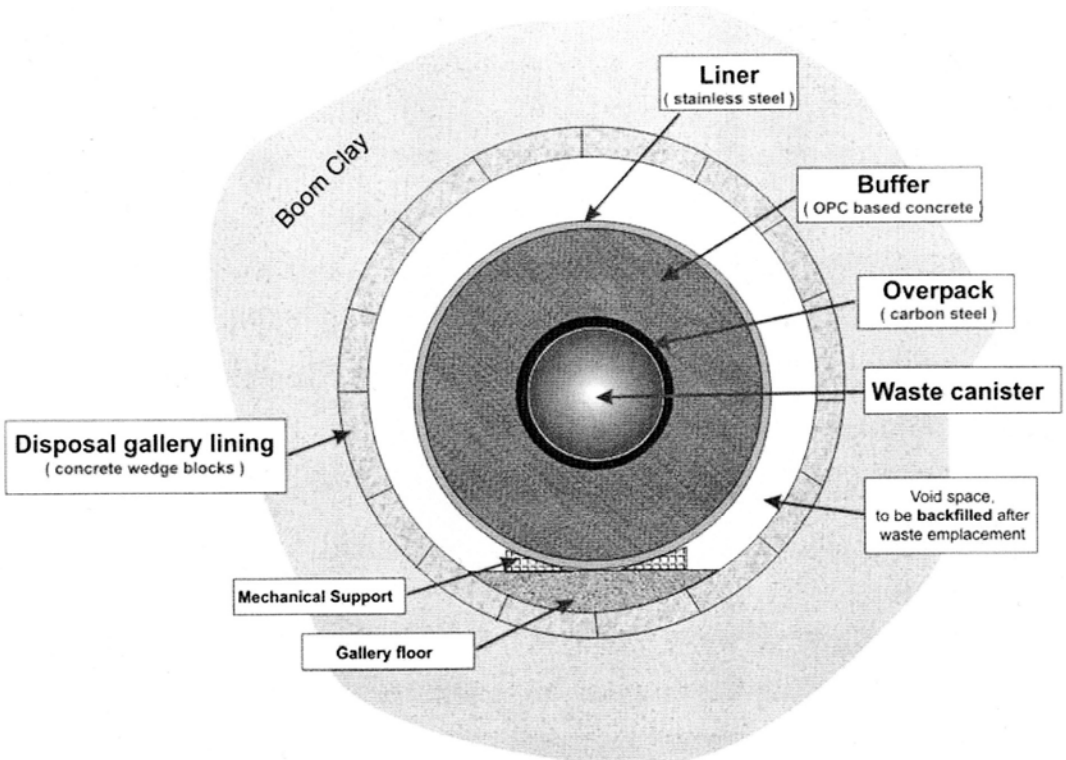

Figure 2-14
Cross-section of Supercontainer in Boom Clay (Bel *et al.*, 2006). Used with permission of ONDRAF/NIRAS.

2.8 Summary and Key Observations

- <u>*Policy on Geologic Disposal*</u>: Although the decision to dispose of HLW has not yet been made, the Belgian R&D program on evaluating the feasibility of geologic disposal is well established. Belgian policy is reviewed every 10 years. The government made the decision in 2000 to eventually abandon the use of nuclear power. A government decision on waste management of used nuclear fuel, including the option of reprocessing and the decision on HLW disposal, is expected in 2011.
- <u>*Institutional Arrangements*</u>: ONDRAF/ NIRAS, the implementer responsible for radioactive waste management, is under public control and supervised by the Minister of Energy. FANC is the regulatory body in Belgium. The Agency, under the supervision of the Minister of the Interior, was re-organized in 2007 and one of its four divisions, Facilities & Waste, is responsible for reviewing the license applications of nuclear facilities a well as the licensing of new disposal facilities. With regard to funding, utilities pay into a dedicated fund for decommissioning and waste management. The fund is managed by Synatom, a commercial subsidiary of Electrobel, but with some government oversight.

- *Key Laws and Regulations*: The key regulation is GRR-2001, the Royal Decree of July 20, 2001. This regulation specifies an annual dose limit of 1 mSv/year to members of the public from all controllable practices and sources. No timeframe is specified in GRR-2001 in the context of the period over which radiological assessments should be performed. No regulations have been prepared specifically addressing geologic disposal.
- *Site Screening and Selection*: Studies within a late 1970's EC program as well as an independent Belgian study identified only argillaceous rocks as being potentially suitable in Belgium, from which two main groups were recognized: hard rocks (shales) and poorly consolidated, plastic clay. Preliminary results from studies of the latter identified Boom Clay as being suitable, its key features being low hydraulic conductivity, reducing conditions (Eh = -0.25V), and high retention properties for most radionuclides. A decision was made to construct an underground R&D facility (HADES) on and under the premises of SCK•CEN at Mol-Dessel. The initial construction phase started in 1980 and resulted in the completion of the URL in 1983.
- *Repository Design Concepts*: Two reference designs exist, one for HLW disposal and one for used nuclear fuel, with a multi-barrier disposal concept for each – stainless steel waste container, clay-sand-graphite buffer, and concrete lining surrounded by the host clay formation. The repository layouts are on a single level. While emphasizing that disposal implies no intention to retrieve waste, ONDRAF/NIRAS has discussed and identified certain aspects of the repository design that allow retrievability over a limited period of time.
- *Performance Metrics and Assessments*: In the absence of regulations to guide the long-term safety evaluation, ONDRAF/NIRAS have prepared and submitted two 'feasibility' assessments (SAFIR and SAFIR2). ONDRAF/NIRAS considered timeframes out to a million years and presented results in terms of annual dose (Sv/year). Peak dose was ~10^{-5} Sv/year for both HLW disposal and used fuel disposal.
- *Independent Peer-Review and Advisory Bodies*: After submission of the first safety evaluation (SAFIR), the Secretary of Energy established a Safety Commission comprising Belgian and foreign experts to evaluate this report, as well as review ONDRAF/NIRAS' 5-year R&D program plan. An international peer group under the oversight of NEA reviewed the second safety evaluation SAFIR2, in addition to the internal Evaluation Commission. The conclusion was that the selection of the Boom Clay in the Mol-Dessel area is a good one. These assessments are not considered as safety assessments in support of licensing but more a basis for discussions between implementer and regulator.
- *Stakeholder and Public Involvement*: Public involvement, in terms of construction of the HADES R&D facility, has been relatively small. According to the current program, public consultations on the long-term management plan for LL-ILW and HLW started in January 2009, allowing opportunities for consultations with the general public. Any license application for the construction of disposal facilities involves the preparation of an EIS, which requires informing the public and public involvement in the form of public hearings.
- *Program Maturity*: Although the country has not committed to this option, the Belgian R&D program on the feasibility of geologic disposal has been active for over 30 years, working with a reference design and repository layout for both HLW and used nuclear fuel. Ongoing work in this area has focused on constructing, and carrying out experiments and tests in the underground facility HADES in clay underneath one of the main nuclear sites in Belgium.

- *Additional Observations*: Despite the country's siting challenges in terms of size and population density, the implementer has so far been able to demonstrate the feasibility of a geologic repository for either HLW or used nuclear fuel within the territory of Belgium.

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3
CANADA

3.1 Introduction

3.1.1 General Nuclear Profile

The Canadian nuclear fleet comprises 18 operating nuclear reactors all of the domestic CANDU pressurized heavy water reactor (PHWR) design, generating 85 TWh of electricity in 2009 and contributing approximately 15 % of the total electric power generation in the coutnry (WNA, 2010a). Sixteen of the 18 operating units are located in Ontario and provide over 50% of that province's power. A number of older units are undergoing or are being considered for refurbishment. Canada is firmly committed to nuclear energy as one component of its energy policy, with proposals in place to construct new reactors over the next 10 years. Three reactors have been shut down and are being decommissioned (WNA, 2010b).

After a period of almost 20 years of developing a disposal concept for irradiated nuclear fuel, the government concluded that there was insufficient public support to implement a geologic repository program. Since then, a newly formed government waste management organization has engaged in an extensive dialogue with stakeholders to develop an acceptable waste management approach that includes a period of interim storage followed by geologic disposal. Currently, all irradiated nuclear fuel from NPPs is held in interim wet or dry storage, the latter in steel-lined concrete storage casks at the plant sites where the irradiated fuel is generated.

Currently, Ontario Power Generation (OPG), the major power generator in Canada, is developing a deep geologic repository (DGR), ~650 m depth, at its Bruce NPP site, for OPG's L/ILW (see also Section 3.3.2). In addition, Atomic Energy of Canada Limited (AECL) is assessing the feasibility of siting a Geologic Waste Management Facility (GWMF) for AECL's L/ILW at a depth of 500-700 m at one of its nuclear sites.

3.1.2 Used Fuel / HLW Inventory and Projected Disposal Requirements

The Canadian Nuclear Safety Commission (CNSC, 2008) reports the quantities of irradiated nuclear fuel in wet storage at each reactor site, and dry storage, at each waste management site as follows:

- Wet storage (reactor sites): 29,559 MTHM (1,535,412 fuel bundles);
- Dry storage (waste management sites): 8,820 MTMH (465,966 fuel bundles).

CNSC has stressed the need for continuous institutional controls such as security measures, monitoring, and maintenance for used fuel storage. Therefore, while considered as a safe practice, Canada, as other countries, regards wet or dry storage of used fuel as safe practice but only an effective interim measure, and not a permanent solution.

3.2 Institutional Arrangements

3.2.1 Implementation Framework

Figure 3-1 shows the specific responsibilities for the management of the different types of radioactive waste identified in the previous section. Figure 3-2 shows a schematic of the institutional framework for nuclear safety and radioactive waste management in Canada. From these diagrams, the following key organizations and governmental agencies are identified:

POLICY and OVERSIGHT - *Natural Resources Canada* (NRCan): responsible for developing Canadian policy concerning energy sources. NRCan is also responsible for ensuring the safety of long-term management of radioactive waste.

IMPLEMENTER - *Nuclear Waste Management Organization* (NWMO); established in 2002 by the nuclear utilities via the Nuclear Fuel Waste Act (NFWA; see Section 2.2.2), NWMO is responsible for the long-term management of Canada's irradiated nuclear fuel. NWMO makes recommendations to, and is overseen by, the federal government (Natural Resources Canada).

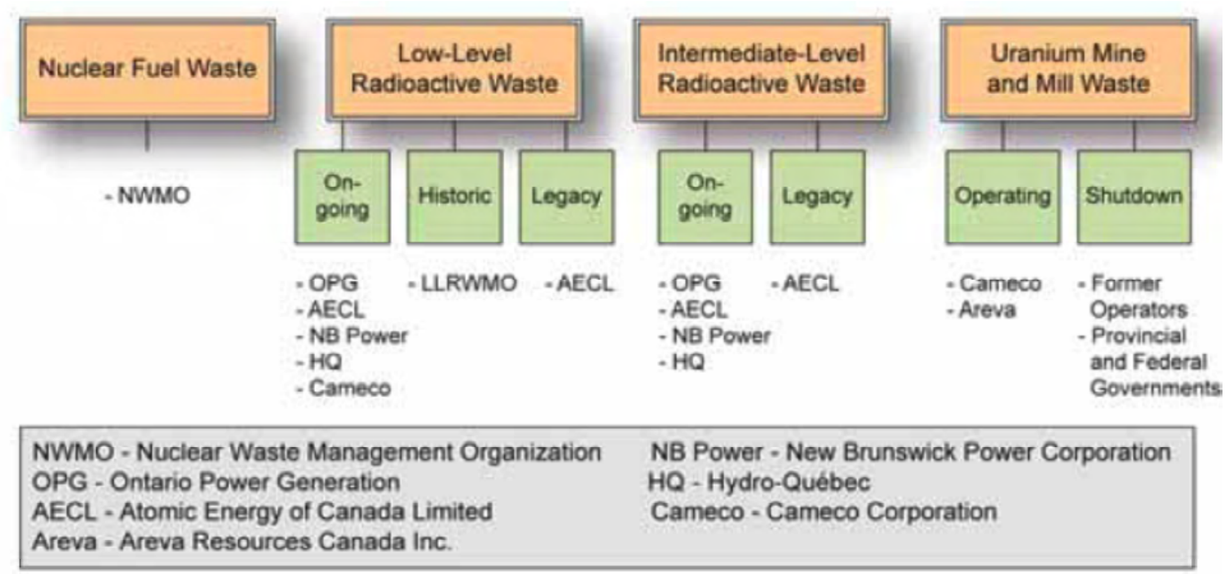

Figure 3-1
Organizational responsibilities for the long-term management of radioactive waste in Canada (CNSC, 2008). Used with permission of the Canadian Nuclear Safety Commission.

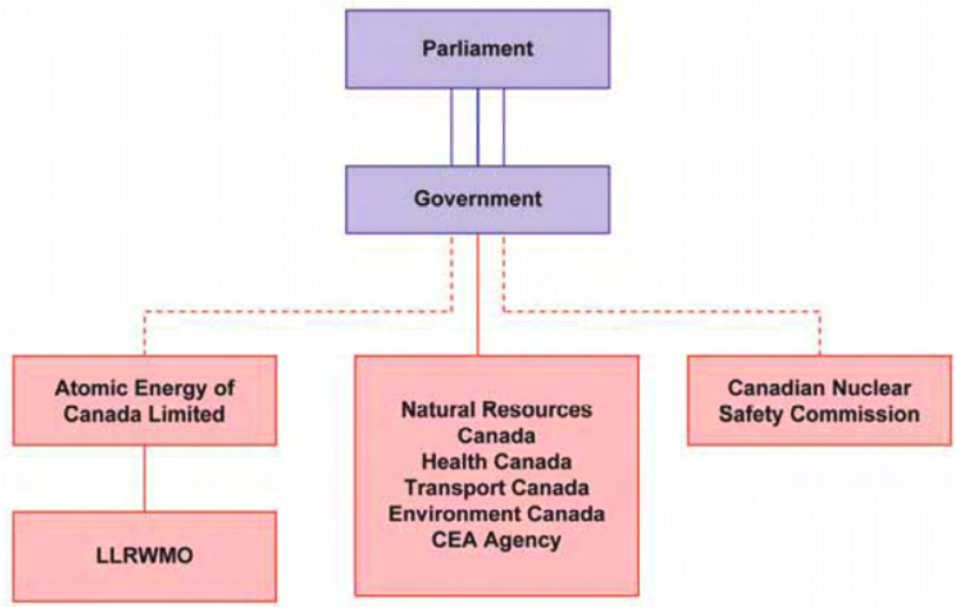

Figure 3-2
Institutional framework for nuclear activities in Canada (CNSC, 2008). Used with permission of the Canadian Nuclear Safety Commission.

REGULATOR - *Canadian Nuclear Safety Commission* (CNSC): Canada's nuclear regulatory body, created by the Governor-in-Council under the National Safety and Control Act (NSCA). CNSC (formerly the Atomic Energy Control Board, AECB) reports to the Canadian Parliament through the Minister of Natural Resources, but is not part of the Department of Natural Resources. CNSC is an independent federal regulatory agency and holds the authority to regulate the emplacement of radioactive waste in a deep geologic repository in Canada.

In addition to the CNSC, the Canadian Environmental Assessment Agency and Health Canada (via the EIA process) and Environment Canada (in the broader context of regulating environmental quality) have supporting roles in the safe long-term management of HLW radioactive waste.

ADVISORY and SUPPORT – Other organizations within Canada that have notable supporting roles in HLW management include *Atomic Energy of Canada Limited* (AECL): AECL is wholly owned by the Government of Canada. AECL designs, markets, sells and builds Canadian-designed CANDU power reactors and generally provides technical support to nuclear utilities. AECL also has a significant role in decommissioning and waste management in Canada. Peer-review has previously been provided by ad hoc advisory panels. For example, AECL's 1994 assessment was reviewed by a special Review Panel of the EIA, which concluded that the disposal concept was technically safe and met current regulatory requirements.

In 1978, the governments of Canada and Ontario established the Canadian Nuclear Fuel Waste Management Program (CNFWP) towards establishing the safe and permanent management of HLW. At that time, the two government bodies announced the inception of a program "*to dispose of radioactive wastes from nuclear power reactors in a deep, underground repository in* *intrusive igneous rock*." (AECB, 1985). AECL and Ontario Hydro (now OPG) were given primary responsibility for developing storage, transportation, and deep repository technology. In 1981, the governments of Canada and Ontario announced that site selection for a HLW repository would not be undertaken until after the disposal concept had been accepted. A disposal concept was developed by AECL and deemed to be technically acceptable, but as stated previously, the government did not consider there was sufficient public support for the disposal concept.

The Canadian Government's 1996 *Policy Framework for Radioactive Waste* established the institutional and financial arrangements that were necessary to manage radioactive waste in a safe, comprehensive, environmentally sound, and cost-effective manner. Major elements of this *Policy Framework* stipulate that:

- The federal government has the responsibility to develop policy and regulate and oversee radioactive waste producers and owners so that they meet their operational and funding responsibilities according to approved long-term waste management plans, and
- Waste producers and owners are responsible, in accordance with the "polluter pays principle", for the funding, organization, management and operation of long-term waste management facilities and other facilities required for their waste. Also in accordance with the NWFA, waste producers are responsible for final disposal including siting, licensing and construction of disposal facilities, and established Trust Funds in 2002 for this purpose. In 2008, NWMO submitted for government approval a proposed funding formula for long-term waste management as well as the deposits to be made annually by waste producers, and the Minister of Natural resources approved this proposal in 2009.

3.2.2 Legal and Regulatory Framework

The key pieces of existing legislation governing radioactive waste management in Canada are:

- *Nuclear Energy Act (NEA)*: umbrella legislation governing nuclear energy related activities; supersedes the Atomic Energy Control Act of 1946.
- *Nuclear Safety and Control Act (NSCA)*: aimed at ensuring the safety of the nuclear industry and radioactive waste management in Canada.
- *Nuclear Fuel Waste Act (NFWA)*: establishes clear responsibilities on the part of owners of irradiated nuclear fuel concerning long-term waste management approaches. This Act was passed in response to the recommendations of a Review Panel of the EIA associated with the technical disposal concept for the geologic disposal of irradiated nuclear fuel (discussed in Section 2.6.1).
- *Canadian Environmental Assessment Act (EAA)*: addresses responsibilities and procedures associated with major federal projects such as a HLW storage or disposal facility.

Formerly, the Regulatory Document that provided the basis for AECL's 1994 assessment (AECL, 1994) was R-104 in which the regulator required that the risk be summed over "*all significant scenarios*" and that the estimated annual risk be less than 10^{-6} per year, specified as being equivalent to an annual dose of 0.05 mSv per year. With regard to timescale of concern, R-104 stated:

"The period for demonstrating compliance with the individual risk requirements using predictive mathematical models need not exceed 10,000 years. Where predicted risks do not peak before 10,000 years, there must be reasoned arguments that beyond 10,000 years the rate of radionuclide release to the environment will not suddenly and dramatically increase, and acute radiological risks will not be encountered by individuals."

Following publication of the federal government's 1996 Policy Framework, and after extensive consultation with the public and industry stakeholders, CNSC issued *Regulatory Policy P-290, Managing Radioactive Waste,* in 2004. The policy outlines CNSC's philosophy and six principles governing its regulation of radioactive waste. Three of these are directly relevant to HLW disposal:

- The assessment of future health, safety and environmental (HSE) impacts covers the time period when the maximum impact is predicted to occur.
- The predicted HSE impacts from the management of radioactive waste are no greater than the impacts that are permissible in Canada at the time of the regulatory decision.
- The measures needed to prevent unreasonable risk to present and future generations from the hazards of radioactive waste are developed, funded and implemented as soon as reasonably practicable.

CNSC's Regulatory Guide G-320, which superseded Regulatory Document R-104, was issued in 2006 to help licensees and applicants assess the long-term HSE impacts from radioactive waste storage and disposal methods. Among a range of issues, the Guide addresses:

- Assessment approaches, structures and methodologies,
- Level of detail in assessments and confidence in assessment results,
- Application of radiological and non-radiological criteria,
- Definition of critical groups for impact assessments, and
- Selection of timeframes for impact assessments – no time limit is specified but similar to R-104, the assessment timeframe is expected to include the time of maximum impact. The assessment itself is expected to provide a rationale for timeframe.

The Guide does not address social acceptability or economic feasibility of long-term management methods.

3.2.3 Waste Classification

The Canadian Standards Association (CSA) recently updated Canada's waste classification scheme taking into account IAEA's waste classification scheme (IAEA, 2007). CSA's standard identified four categories of radioactive waste (CNSC, 2008):

- *HLW*: heat-generating, with heat output nominally >2 kW/m^3;
- *ILW*: normally requires shielding and contains alpha nuclides; may be subdivided into short-lived and long-lived;

- *LLRW*; waste containing radioactive material above clearance and exemption levels and limited amounts of long-lived radionuclides, requiring isolation for periods up to a few hundred years.
- *Uranium Mine and Mill Waste*: especially important in Canada due to the substantial uranium resources; Canada long held the number one spot in worldwide uranium production until recently being overtaken by Kazakhstan in 2009[11].

This waste classification scheme is designed to reflect the degree of containment and isolation necessary to ensure both short- and long-term safety. The first category, HLW, combined with the long-lived component of ILW are directly relevant to this review.

3.2.4 Funding

Financial resources for ensuring the long-term management of used nuclear fuel in Canada are mandated by the Nuclear Safety and Control Act. Both utilities and AECL make annual contributions to dedicated trust funds administered exclusively by NWMO (NEA, 2005; WNA, 2010b).

3.3 Geological Studies for Deep Disposal

3.3.1 Early Geological Studies

After reviewing the options for the long-term management of used nuclear fuel, the Hare Commission had recommended emplacement of used fuel in a deep underground repository within the Canadian Shield (Aiken et al., 1977). Charged with the responsibility of developing the disposal concept, AECL recognized at an early stage the need for an underground facility to conduct large-scale multi-disciplinary experiments and demonstration tests. Consequently, early geological investigations were focused on an Underground Research Laboratory (URL) program. In this context, the recent site selection initiative is discussed in Sections 3.3.4 and 3.4.2.

In its original conception, the URL program in Canada consisted of site evaluation and underground experiments as two independent programs, but following peer review of these plans, the two programs were fully integrated (Simmons and Soonawala, 1982). The revised URL program was planned as three overlapping phases:

- Phase 1 (1980-2011): Site Evaluation, comprising surface exploration, pre-construction selection activities, and post-excavation activities. The last stage of Phase 1 consisted primarily of hydrogeological and hydrochemical investigations including fracture fill and bulk rock geochemistry, and groundwater composition.
- Phase 2: (1983-1990): Construction.
- Phase 3 (1989-2011): Operating Phase.

[11] Source: World Nuclear Association profile – Uranium in Canada: http://www.world-nuclear.org/info/inf49.html.

The overall objectives of the URL program (Simmons and Soonawala, 1982) were to improve an understanding in:

- Geology, geophysics and hydrogeology related to site screening, selection and evaluation;
- Rock mechanics, hydrogeology and geochemistry, at both ambient and elevated temperatures;
- Vault sealing techniques;
- Instrumentation for monitoring parameters in an underground excavation and in the geosphere; and
- Mathematical modeling of events in the URL and the surrounding geosphere.

Site characterization focused on detailed hydrogeological and hydrochemical investigations of four plutons[12] across the Canadian Shield, one of which was the Lac du Bonnet batholith[7], in order to characterize groundwater flowpaths. The characterization findings suggested that subsurface groundwater movement is controlled largely by major sub-horizontal and/or sub-vertical fracture zones. This initial characterization work was completed in 1984.

Based on a series of reconnaissance studies and surveys, a site in the Lac du Bonnet batholith in Eastern Manitoba was selected. The site was chosen primarily because it was located well within the western portion of the intrusive granite formation, the bedrock exposure was good, and the site was close to the Whiteshell Nuclear Research Establishment facilities (Simmons and Soonawala, 1982).

The Lac du Bonnet igneous intrusion is ~2,400 Ma old and is considered to be representative of many large granite intrusions of the Canadian Shield. Records of private wells in the area indicated the presence of three sub-horizontal fracture zones in the granite as major sources of groundwater.

3.3.2 Site Selection for L/ILW Repository

Although not a HLW disposal site, the current repository being constructed for L/ILW is relevant both in terms of its depth (geologic) and the public initiatives that were carried out as a basis for achieving success in terms of siting.

In 2002, OPG signed a Memo of Understanding (MOU) with the Municipality of Kincardine, which established the terms by which OPG, in consultation with the Municipality of Kincardine, would develop a plan to review options for the long-term management of L/ILW at its Western Waste Management Facility (WWMF), Bruce site, Ontario. Members of the Municipality of Kincardine and OPG subsequently formed a Steering Committee to guide activities under the MOU (OPG, 2004).

[12] A pluton is an intrusive body of igneous rock underneath the earth's surface. A batholith is a massive dome-shaped formation of intrusive igneous rock created from a number of plutons converging.

The plan called for Kincardine and OPG to conduct an independent assessment of possible long-tem management options for L/ILW at the WMMF. The Independent Assessment Study (IAS), carried out by an external consultant, addressed issues of technical feasibility, safety, environmental protection feasibility, and social and economic impacts, as well as the benefits of the different long-term management options.

The options assessed were:

- Enhanced processing, treatment and long-term storage: super-compaction followed by secure long-term storage.
- Disposal in a repository: Two options were identified as being technically feasible:
  - Covered Above-Ground Concrete Vault (CAGCV): similar to the LLW disposal site at Centre de l'Aube in France;
  - Deep Rock Cavern Vault (DRCV): construction of a vault at an anticipated depth of ~ 425 to 750 m below ground surface.

The Study results concluded that each of the three options considered:

- Is capable of safely managing some or all of the L/ILW;
- Is geo-technically feasible;
- Would have no significant residual environmental impacts;
- Would have no adverse impact on tourism or agriculture;
- Would provide economic benefits for the community.

Results from a preliminary assessment of the repository concepts supported the conclusion that both the CAGCV and the DRCV options could be designed and constructed to meet the '*international dose criterion*[13] *(0.3 mSv per year)*'. However, while this criterion would be achievable for all LLW and a *range of ILW* in the case of the CAGCV, it would be achievable for all LLW and *all ILW* in the case of the DRCV.

Following submission of the IAS report in early 2004, the Municipality of Kincardine passed a Council Resolution affirming:

> *"….the Deep Rock Vault" option as the preferred course of study in regards to the management of low and intermediate level radioactive waste."*

OPG is now moving forward with the process for construction of a Deep Geologic Repository (DGR current acronym) on the Bruce Nuclear Site within the Municipality of Kincardine, Ontario. The project is in the middle of the regulatory process and, pending approvals and licensing by the authorities, OPG will begin construction in 2012, with operation of the facility planned for 2017. An Environmental Assessment (EA) is required for this project and the Canadian Environmental Assessment Agency will be the Federal EA Coordinator for this assessment.

[13] This 'international dose criterion' presumably refers to ICRP [1993].

OPG's technical program involving a deep geological repository for used fuel has been summarized in a series of annual reports (for example, Hobbs et al., 2005). NWMO has now taken over responsibility from OPG for pursuing the DGR and satisfying regulatory requirements for its construction and operation.

3.3.3 Underground Research Laboratory Program

AECL constructed the URL near Lac du Bonnet, Manitoba in a large, previously undisturbed granitic pluton. A 21-year URL lease, allowing underground excavation, drilling, and testing for research purposes, was obtained in 1980 but was later extended to 2014. The surface facilities were constructed in 1984, and the underground excavations to the 420 m level were completed by 1989.

AECL's URL provided the Canadian repository program with a unique research facility for developing site-characterization methods and *in situ* stress measurement techniques, demonstrating full-scale container emplacement, evaluating rock excavation methods and rock opening stability, modeling groundwater flow and contaminant transport, and conducting grouting and tunnel sealing experiments. A schematic of the URL and location of experiments is shown in Figure 3-3.

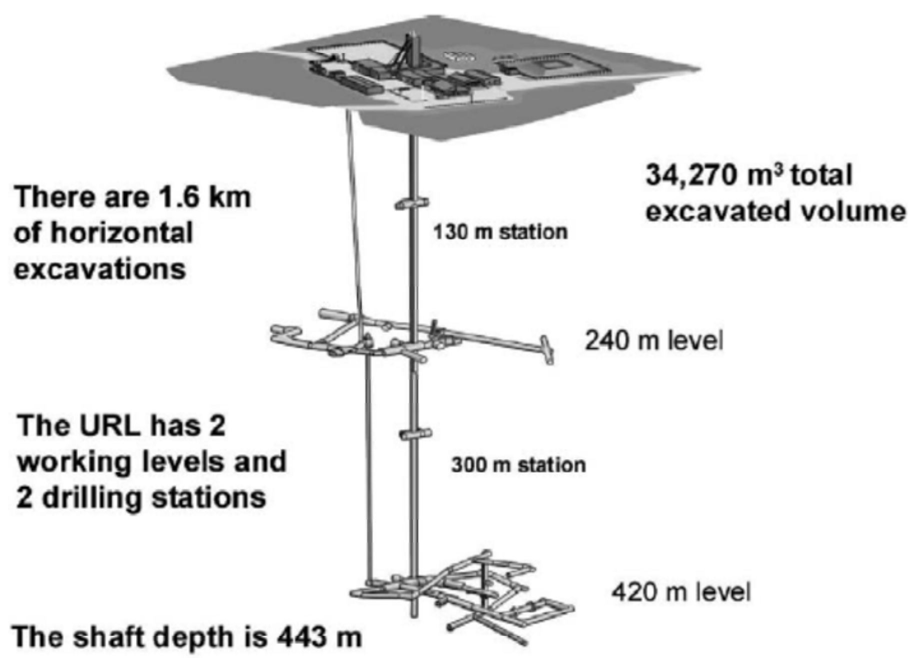

Figure 3-3
Test drifts at AECL's URL, Lac du Bonnet batholith, Manitoba (AECL, 2004). Copyright © Atomic Energy of Canada Limited, 1994. Used with permission.

The URL hosted R&D organizations from many countries over the course of its operation as well as a number of co-operative international projects with Sweden, Japan, France, and the U.S.A. After a funding reduction by OPG, AECL was forced to decommission the URL in 2003.

3.3.4 Current Site Investigation Approach

In keeping with the new, phased approach to the long-term management of used nuclear fuel, as discussed in Section 3.5.1, there is no immediate pressure to identify potential sites for geologic disposal. Rather, rock types are being considered in a generic way, with sedimentary rocks now being considered as well as crystalline rock, based on the extensive studies and favorable results noted from international experience.

3.4 Disposal Concept

At an early stage in Canada's HLW disposal program, (1980), the federal government and the provincial government of Ontario both stated that selection of a site for HLW disposal would not proceed until the concept for disposal had been reviewed and assessed. Thus, a generic, rather than site-specific, disposal concept was developed (Allan and Simmons, 1996).

3.4.1 Reference Concept

AECL's original reference disposal concept and the basis for its assessment in the EIA (AECL, 1994) is shown in Figure 3-4.

More recently, in connection with latest initiative concerning site selection, a reference conceptual design and cost estimate for the in-room placement of irradiated nuclear fuel in a deep geological repository in Canadian Shield host rock was developed by CTECH (2002). Alternative conceptual placement configurations have been considered, including an in-floor borehole concept and a long horizontal borehole concept (Gierszewski et al., 2004; Russell and Simmons, 2004). Ultimately, the most appropriate placement method will depend on site-specific conditions and demonstration of appropriate engineering technology.

Conceptual designs have been prepared for the irradiated nuclear fuel packaging facility as well as the underground repository, expected to be constructed within a depth range of 500 m to 1,000 m (CTECH, 2002). Figure 3-5 shows a schematic of the overall design of the deep repository. The reference design for an irradiated fuel canister and overpack is shown in Figure 3-6, with an inner carbon steel lining (80-100 mm thick) and an outer oxygen-free phosphorus-doped (OFP) copper shell (25 mm thick). The interior of the steel vessel would be filled with an inert gas, e.g., helium. The design life for such a reference container is 100,000 years (NWMO, 2005), comparable with the design lifetimes of Swedish and Finnish canisters.

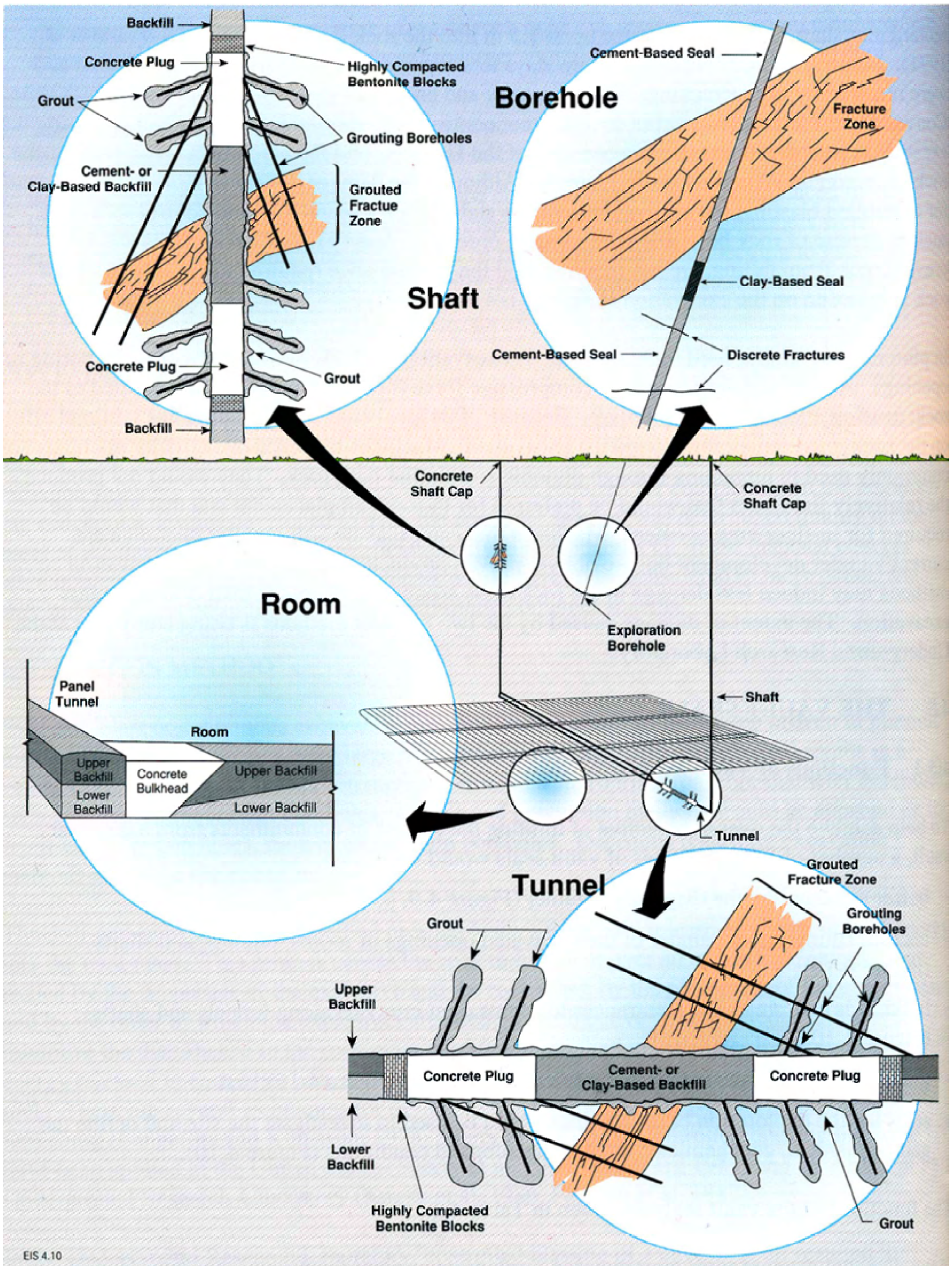

Figure 3-4
Original Canadian program's reference disposal concept as envisioned by AECL in its assessment, with a strong emphasis on sealing concepts (AECL, 1994). Copyright © Atomic Energy of Canada Limited, 1994. Used with permission.

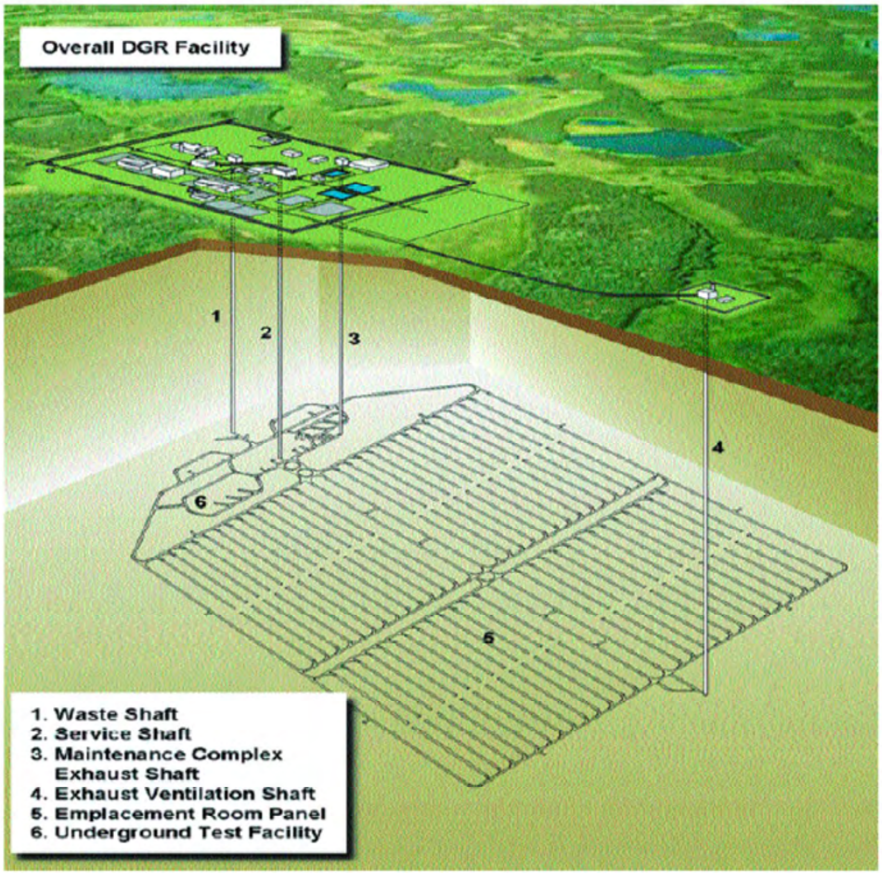

Figure 3-5
Reference conceptual design (Russell and Facella, 2006). Used with permission of Lawrency Berkeley National Laboratory.

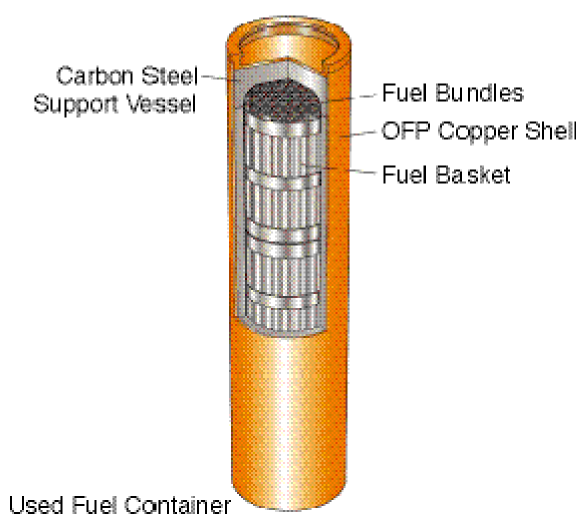

Figure 3-6
Canister design for HLW disposal (Russell and Facella, 2009). Used with permission of Lawrence Berkeley National Laboratory.

3.5 Transparency and Stakeholder Involvement

3.5.1 Public Involvement

Because of lack of progress in identifying a site for HLW disposal, and following the government's 1996 *Policy Framework for Radioactive Waste*, NWMO examined a number of waste management possibilities for used nuclear fuel. Under the NWFA, the implementing organization was responsible for studying at least three options:

- Deep Geological Disposal in the Canadian Shield;
- Storage at nuclear reactor sites; and
- Centralized Storage, either above ground or below ground.

NWMO was required to submit its results and recommendations to government before the end of 2005. Accordingly, the implementing organization engaged in widespread dialogue with Canadian stakeholders in an effort to develop a management approach that was socially acceptable, technically sound, environmentally responsible, and economically feasible (Russell and Pacella, 2006). NWMO took part in almost three years of discussions with aboriginal peoples, the public, and specialists. The Canadian public generally agreed that the focus should be on the three methods specified above.

Based on its extended dialogue with different stakeholders, NWMO published two discussion documents (NWMO, 2003, 2004), culminating in its final recommendation – a fourth option, the Adaptive Phased Management (APM) approach, after additional consultation with stakeholders (NWMO, 2005). NWMO believed that this APM approach, which represented a combination of the above three options, addressed the necessary objectives and influencing factors:

- Fairness, public health and safety, worker health and safety, community well-being, security, environmental integrity, economic viability and adaptability.

The three phases of implementation are (NWMO, 2005):

- *Phase 1*: Preparation for centralized storage, covering a period of ~30 years, including development of a transportation system.
- *Phase 2*: Centralized storage and demonstration of long-term containment and isolation technology at the underground characterization facility at the central site, expected to take ~30 years.
- *Phase 3*: Focus on long-term containment, isolation, and monitoring.

The key attributes of the APM approach are (NWMO, 2005):

- Centralized containment and isolation of the irradiated nuclear fuel in a deep geological repository within a suitable rock formation (crystalline rock of the Canadian Shield or Ordovician sedimentary rock).
- Flexibility in implementation through a phased decision-making process covering a period of over 60 years, allowing the opportunity for continuous learning;
- An optional interim step in the implementation process of shallow underground storage of irradiated nuclear fuel at the centralized interim storage site, prior to final placement in a deep repository;
- Continuous monitoring of the irradiated nuclear fuel to allow for data collection and confirmation of the safety and performance of the repository;
- Potential for retrievability of the irradiated nuclear fuel over an extended period, until future society makes the determination on final closure.

As part of the technical support underlying the APM, conceptual designs and cost estimates were developed for extended irradiated nuclear-fuel storage at nuclear reactor sites, centralized storage and deep geological disposal (CTECH 2002; 2003a; 2003b), as well as the transportation system to a central facility (COGEMA, 2003).

3.5.2 International Involvement

As discussed previously, AECL's involvement in the Canadian Nuclear Fuel Waste Management program including the URL program provided many opportunities for international collaboration.

3.6 Safety Assessment and Licensing

3.6.1 Safety Assessment

The first assessment, carried out by AECL as a basis for its EIS submission (AECL, 1994), also called the "EIS Case Study," evaluated the performance of a vault design consisting of titanium containers placed in boreholes drilled in the floor of the deposition rooms at a depth of 500 m, in a hypothetical geosphere. This geosphere was based on site-characterization data obtained during development of AECL's URL near Lac du Bonnet, Manitoba (Goodwin et al., 1994). The second assessment, referred to as the "Second Case Study," evaluated the performance of copper containers placed in an in-room configuration at a similar depth, using a more permeable geosphere (Goodwin et al., 1996). These two repository concepts are illustrated in Figure 3-7.

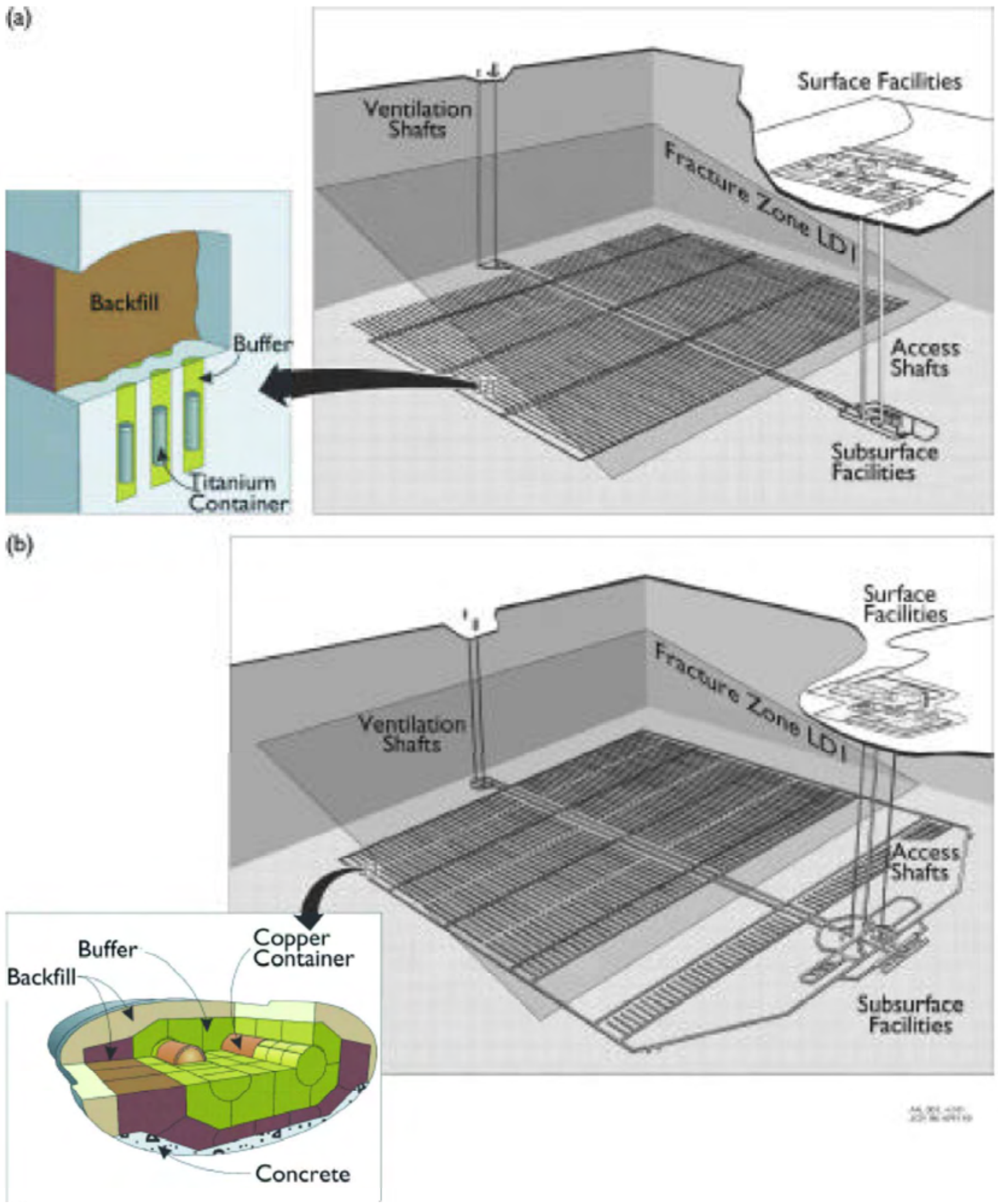

Figure 3-7
Illustration of two hypothetical used-fuel repositories (Goodwin et al., 1996). Copyright © Atomic Energy of Canada Limited, 1996. Used with permission.

The upper diagram (a) depicts the repository that was evaluated in the EIS case study (borehole emplacement of titanium containers). The lower diagram (b) shows the repository that was evaluated in the Second Case Study (in-room emplacement of copper containers). The insets show details of the repository deposition rooms in each case.

3.6.1.1 Scenarios

The Regulatory Document that formed the basis of AECL's 1994 assessment was R-104 in which the regulator required that the risk be summed over "all significant scenarios" and that the estimated risk be less than 10^{-6} per year.

The scenario approach is shown in Figure 3-8 (Goodwin et al., 1994). AECL's scenario development process made use of qualitative and quantitative arguments, excluding certain factors, used to construct scenarios, if:

- The estimated probability of an individual incurring a health impact if that factor occurred, was $<10^{-8}$ per year; or
- The estimated probability of the event occurring was $<10^{-8}$ per year; or
- It was judged by AECL researchers that the event would not contribute significantly to the risk.

The output from this analysis was the identification of a:

- Group of *central scenarios* in which the primary pathway involved normal or expected groundwater-mediated processes, release of waste materials from the vault, radionuclide migration through the geosphere to the biosphere, and potential exposure to members of the critical group leading to impacts.
- Group of *alternative scenarios* involving an *open borehole*.
- Group of *disruption scenarios* revolving around *inadvertent human intrusion caused by drilling*.

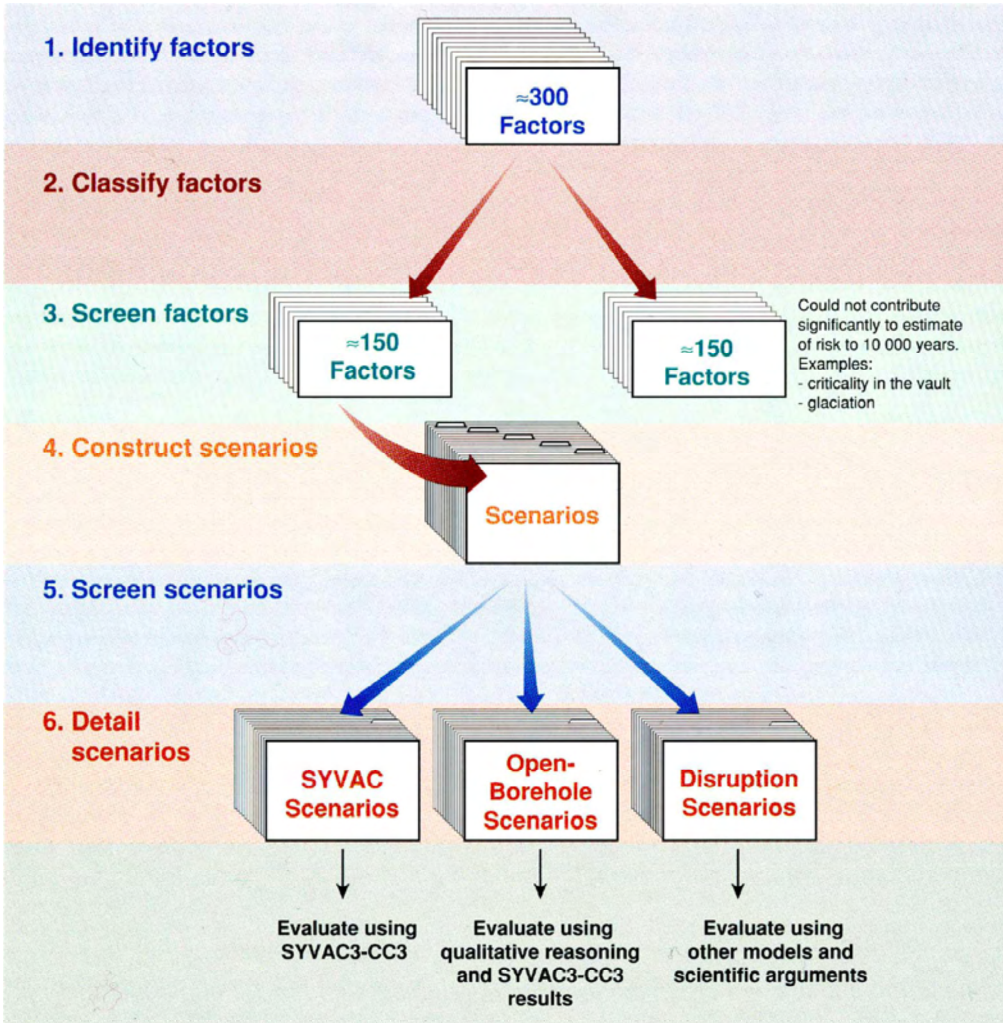

Figure 3-8
Scenario approach adopted by AECL (Goodwin et al., 1994). Copyright © Atomic Energy of Canada Limited, 1994. Used with permission.

3.6.1.2 Assessment Results

The key results from the assessment were presented as the estimated mean dose rate and risk to an individual of the critical group as a function of time, presented out to 100,000 years. Figure 3-9 shows these results in which mean dose rate (vertical axis) is presented (i) on a logarithmic scale and (ii) on a linear scale (AECL, 1994).

The most recent post-closure safety assessment of the deep geological repository concept is documented in the Third Case Study prepare by OPG (Gierszechowski et al., 2004). This study built on the two previous safety analyses by AECL (Goodwin et al., 1994; 1996) and used the most recent conceptual design (horizontal emplacement) of a deep geological repository at a generic location in the Canadian Shield. In particular, the Third Case Study took into account some methodology issues identified by reviewers of the previous assessments, e.g. the use of regional groundwater modeling studies to help identify the repository location, coupling of assessment models to the site characterization models, three-dimensional modeling of vault and geosphere, and explicit analyses of "what-if" type scenarios with relatively high consequences (high doses).

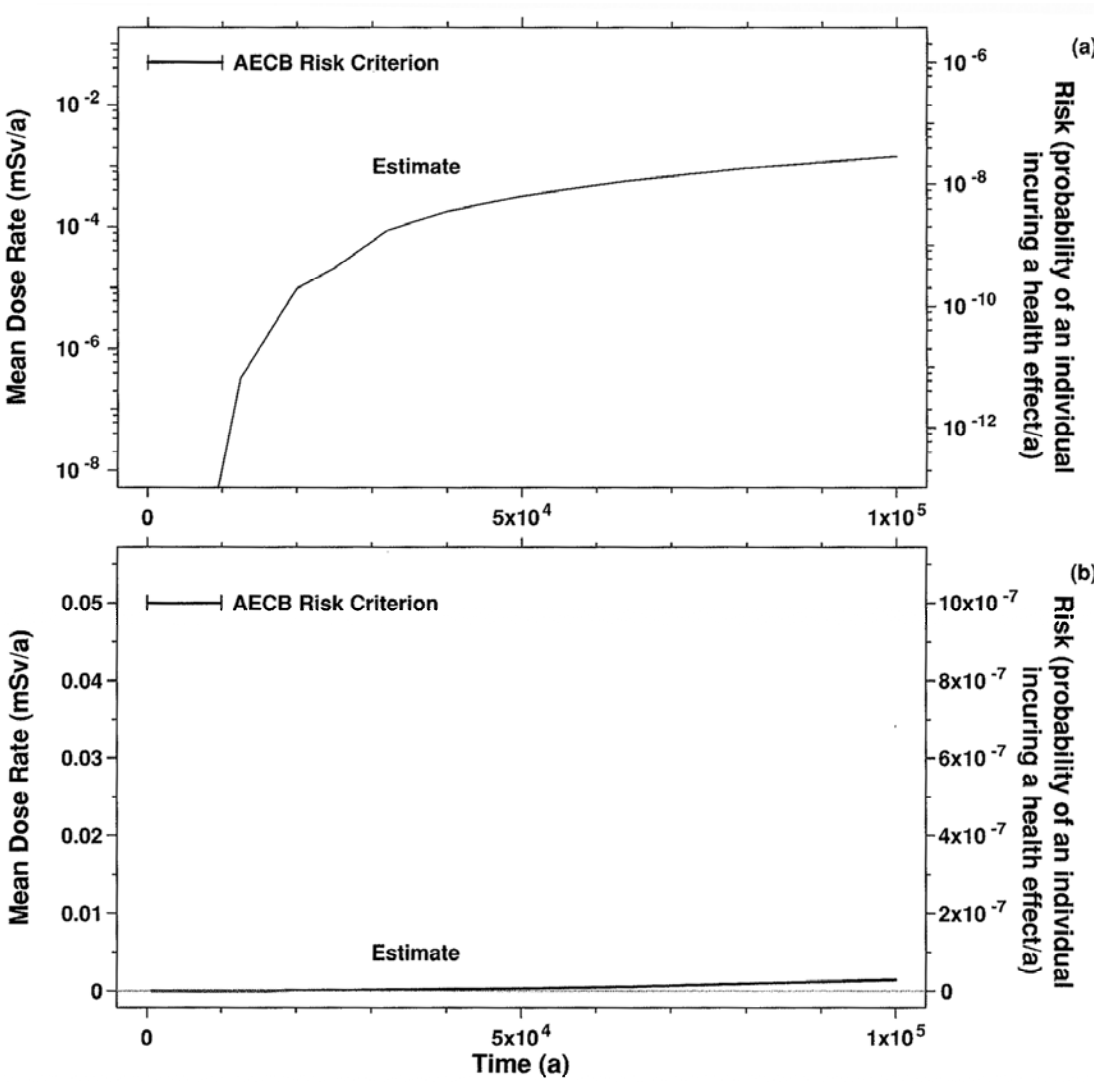

Figure 3-9
Estimated mean dose rate and risk to an individual member of the critical group as a function of time, 0-100,000 years (AECL, 1994). Copyright © Atomic Energy of Canada Limited, 1994. Used with permission.

The calculated annual dose to a member of the public from the assumed undetected early failure of several irradiated nuclear fuel containers in the repository for the reference scenario is illustrated in Figure 3-10. The post-closure annual dose was modeled for the most exposed member of the critical group, a self-sufficient farmer residing near the surface discharge location. The peak dose of about 10^{-7} Sv/year occurs after nearly 500,000 years and is dominated by the long-lived and relatively mobile radionuclide iodine-129.

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

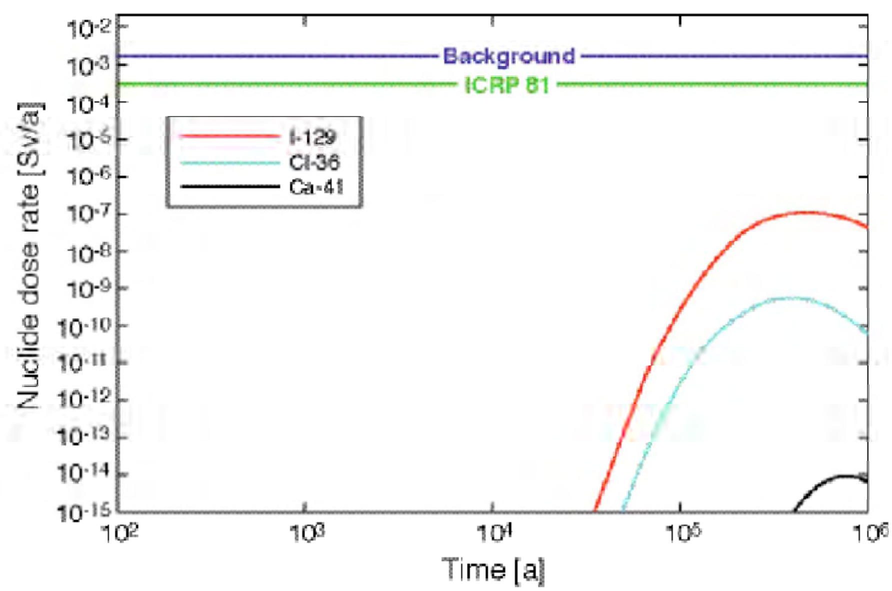

Figure 3-10
Annual dose results for Third Case Study assessment (Russell and Pacella, 2006). Used with permission of Lawrence Berkeley National Laboratory.

3.6.2 Licensing Process

According to the regulations, a license is required from CNSC for all phases of a project (i.e., site preparation, construction, operation, decommissioning, and abandonment). A provision in the Nuclear Safety and Control Act also requires a financial guarantee from the licensee for all phases of a project once a license has been issued.

In 1994, AECL submitted its Environmental Impact Statement (EIS) (AECL, 1994) on the disposal concept for review by a federal Environmental Assessment Panel. This review included input from government agencies, non-governmental organizations, and the general public. Public hearings associated with the review took place during 1996 and 1997. The submission included two post-closure safety assessments associated with AECL’s disposal concept.

The EIA Panel's report, which contained a number of recommendations, was submitted to the Canadian federal government in March 1998 (CEAA, 1998). The federal government responded to the Panel’s report by the end of the same year (NRCan, 1998). In its response, the federal government concluded that the AECL concept for a deep geologic repository of nuclear fuel waste was technically safe and meets current regulatory requirements, in that it can provide passive safety in the long term. However, the federal review also concluded that there was insufficient public support at the time to implement a repository siting program.

Figure 3-11 shows a schematic diagram of Canada's current licensing approach, specifically CNSC. Each phase requires a license. Typical terms of such licenses are for five to ten years. By way of an example, CNSC issued an operating license for OPG's Darlington Waste Management facility, following two public hearings three months apart. The license allows OPG to operate a spent fuel dry storage waste management facility at the Darlington NPP site, near Toronto. The license was issued for a five-year term and requires OPG to submit a mid-term status report on operational performance.

Licensing for the construction and operation of a HLW disposal facility, when it occurs, will involve the EIA process and, hence, input from all stakeholders, including the general public.

Figure 3-11
CNSC's stepwise approach to licensing (CNSC, 2008). Used with permission of CNSC.

3.7 Current Status

For 18 years, AECL worked on assessing the concept for HLW emplacement in a deep repository excavated in the plutonic rock of the Canadian Shield, culminating in the submission of an EIS (AECL, 1994). Following public hearings, the government found the concept to be technically safe, and in compliance with current regulatory requirements, it also concluded that there was insufficient public support to move forward and implement a repository-siting program.

Since then, NWMO carried out an extensive dialogue with stakeholders on the key elements for a fair process to identify an informed and willing community to host facilities for the management of Canada's used nuclear fuel for the long term. Guided by the public input, the implementer developed a Proposed Process for Selecting a Site for review and comment in 2008.

NWMO indicates that such a facility will be cost between $16 - $24 billion. The repository system will be designed so that the waste will be continually monitored and retrievable for an extended period of time. The project will also involve creation of a centre of expertise for technical, environmental and community studies related to the design and operation of deep geological repositories.

Construction of the facility will proceed only after the NWMO demonstrates that all safety, health and environmental protection standards set by regulatory authorities can be met or exceeded.

In May 2009, NWMO initiated further dialogue with stakeholders inviting comments and views that will help to generate an open, transparent, fair and inclusive site selection process. A variety of mechanisms were used to seek feedback, including regional information sessions, multi-party dialogues, aboriginal dialogues and citizen dialogues. NWMO has also created a ‘Learn More’ program that makes available information and funding to assist communities, organizations and individuals to learn more about APM approach.

The NWMO emphasizes that it is not currently looking for a site, and that site selection will begin only after the process for site selection has been confirmed and finalized.

NWMO recently issued a new draft document *Implementing Adaptive Phased Management 2010 to 2014* that represents the implementer’s annual five-year strategic plan for long-term management of irradiated fuel. The Plan presents highlights of NWMO’s work program in seven key areas. The organization expects to refine the draft Plan and publish the final version early in 2010. Progress in 2010 against this Plan will then be described in its Annual Report to be published in March 2011.

3.8 Summary and Key Observations

- *Policy on Geologic Disposal*: Canada is committed to the geologic disposal of used nuclear fuel, although its first attempt to progress towards finding a suitable site ended in the mid-1990’s when the government decided there was no public support for a geologic repository. Since then, the implementer has been proceeding cautiously towards gaining a consensus for geologic disposal.
- *Institutional Arrangements*: The NWMO is the implementer responsible for the long-term management of Canada’s irradiated nuclear fuel. NWMO is overseen by the federal department NRC. AECL had previously been charged by the government to develop a disposal concept. The relevant regulator is CNSC, an independent federal agency that regulates the emplacement of radioactive waste in a deep geologic repository. CNSC (formerly AECB), reports to the Canadian Parliament through the Minister of Natural Resources, but is not part of the Department of Natural Resources. Financial resources for

ensuring the long-term management of used nuclear fuel in Canada are mandated by law. The utilities and Atomic Energy of Canada Limited (AECL) make annual contributions to dedicated trust funds administered exclusively by NWMO.

- *Key Laws and Regulations*: The NSCA, which came into force in 2000, was aimed at establishing clear responsibilities on the part of owners of irradiated nuclear fuel concerning long-term waste management approaches, and led to the formation of NWMO. With regard to regulations addressing geologic disposal, the regulatory document R-104 issued by AECB the basis for AECL's 1994 assessment. CNSC's equivalent document to R-104 is Regulatory Guide G-320, which was issued in 2006.
- *Site Screening and Selection*: A 1970's Commission recommended emplacement of used fuel in a deep underground repository within the Canadian Shield (i.e., crystalline rock). AECL then moved towards constructing an underground facility for detailed *in situ* geological studies, also incorporating in its URL program experiments and testing of emplacement techniques. Reconnaissance studies and surveys were used to identify an appropriate site for the URL in a granite intrusion. Recent site characterization of the low-permeability sedimentary formations at the Bruce reactor site in Ontario for the disposal of L/ILW operational wastes may lead to an expansion in the rock types considered for future disposal of Canadian used fuel.
- *Repository Design Concepts*: The federal government and the provincial government of Ontario both stated (1980) that selection of a site for HLW disposal would not proceed until the concept for disposal had been reviewed and assessed. Thus, a generic, rather than site-specific, disposal concept was developed and AECL was given the responsibility to develop a disposal concept compatible with disposal in the Canadian Shield. This multi-barrier disposal concept consisted of a titanium overpack with inner steel container as waste package, and bentonite buffer. The most recent repository conceptual design involves horizontal emplacement in a repository at a generic location in the Canadian Shield. The current reference waste package design has an inner carbon steel lining (80-100 mm thick) and an outer oxygen-free phosphorus-doped (OFP) copper shell (25 mm thick), the interior of the steel vessel being filled with an inert gas such as helium. The design life for such a reference container is 100,000 years, comparable with the design lifetimes of Swedish and Finnish canisters. The stepwise management approach now being adopted by NWMO allows for retrievability of the irradiated nuclear fuel over an extended period, until future society makes the determination on final closure.
- *Performance Metrics and Assessments*: The regulation used as performance metric for AECL's performance assessments was the AECB's regulatory document R-104 in which the regulator required that the risk be summed over "*all significant scenarios*" and that the estimated annual risk be less than 10^{-6} per year, specified as being equivalent to an annual dose of 0.05 mSv per year. With regard to timescale of concern, R-104 stated: "*The period for demonstrating compliance with the individual risk requirements using predictive mathematical models need not exceed 10,000 years. Where predicted risks do not peak before 10,000 years, there must be reasoned arguments that beyond 10,000 years the rate of radionuclide release to the environment will not suddenly and dramatically increase, and acute radiological risks will not be encountered by individuals.*" While no time limit is specified in the current document, Regulatory Guide G-320, similar to R-104, the assessment timeframe is expected to include the time of maximum impact. In addition, the assessment itself is expected to provide a rationale for timeframe. AECL's assessment results, submitted

as part of an EIS, presented as the estimated mean dose rate and risk to an individual of the critical group out to 100,000, yielded a maximum risk more than an order of magnitude less than the regulatory criterion at the time. The most recent assessment calculated the predicted annual dose for the most exposed member of the critical group, a self-sufficient farmer residing near the surface discharge location. The resultant peak dose of ~10^{-7} Sv/year occurred after nearly 500,000 years.

- *Independent Peer-Review and Advisory Bodies*: AECL's 1994 assessment was reviewed by a special Review Panel of the EIA, which concluded that the disposal concept was technically safe and met current regulatory requirements.
- *Stakeholder and Public Involvement*: After almost two decades of research towards the geologic disposal of used nuclear fuel, the government acknowledged in 1998 that there was insufficient public support to move forward and implement a repository-siting program. Since that time NWMO, the implementing organization, has engaged in widespread dialogue with Canadian stakeholders, taking part in almost three years of discussions with aboriginal peoples, the public, and specialists. The result of these interactions indicated that Canadian public generally agreed that the national waste management focus should be on (i) geologic disposal and interim storage, both (ii) on site and (iii) at a dedicated centralized storage facility. NWMO has since settled on a fourth option, the Adaptive Phased Management approach, which is a combination of the three options originally proposed.
- *Program Maturity*: AECL had worked on assessing the concept for HLW emplacement in a deep repository excavated in the plutonic rock of the Canadian Shield, culminating in the submission of an EIS in 1994. Following public hearings, the government found the concept to be technically safe and in compliance with current regulatory requirements, but it also concluded that there was insufficient public support to move forward and implement a repository siting program. Since then, the new implementer NWMO has carried out an extensive dialogue with stakeholders on a fair process to identify a community willing to host facilities for the management of Canada's used nuclear fuel for the long term. Guided by the public input, NWMO developed a Proposed Process for Selecting a Site for review and comment in 2008.
- *Additional Observations*: In its discussions with stakeholders, NWMO emphasizes that it is not currently looking for a site, and that site selection will begin only after the process for site selection has been confirmed and finalized.

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4
CHINA

4.1 Introduction

4.1.1 General Nuclear Profile

The Chinese nuclear power industry started in the late 1970's as part of the country's economic reform and development policy issued by the State Council. Since then, China has invested substantial resources to develop nuclear power in order to meet sharply increasing electricity demand. Currently, China has 12 operating commercial nuclear power reactors located at four individual sites. The Chinese fleet generated 66 TWh in 2009, corresponding to just under 2% of the nation's total electric power generation (State Power Information Network, 2009; WNA, 2010).

China's plans for expansion of nuclear power generation is ambitious, with 24 new reactors being constructed and additional ones planned. China seeks to increase its installed capacity to 80 GWe in 2020; to 200 GWe by 2030, and to 400 GWe by 2050 (WNA, 2010a). Along with its aggressive new plant construction plans, China's intends to reprocess spent fuel from light water reactors for recycle of plutonium as MOX in the short-term, with a view toward closure of the fuel cycle via deployment of fast reactors. Accordingly, the predominant waste form for from nuclear generation for a deep geologic repository would be vitrified HLW.

The State Council's Nuclear Power Mid- to Long-Term Development Plan calls for the development of a permanent nuclear waste repository at the "same pace" as nuclear power plant construction (NDRC 2007).

4.1.2 Used Fuel / HLW Inventory and Projected Disposal Requirements

The consequence of rapid development of nuclear power is the increasing rate at which radioactive waste is generated. According to Guoqing (2010), the estimated cumulative used fuel inventory is 1,300 MTHM as of 2010, with projections for a 40 GWe fleet of 7,500 MTHM in 2020 and 13,000 MTHM in 2025. China's national policy calls for reprocessing of the used fuel and recycling it into existing nuclear power reactors. China is also pursuing the development of fast breeder reactors for greatly increasing utilization of natural uranium resources for energy production (WNA, 2010b). Therefore, most of the waste from irradiation of nuclear fuel will be in the form of vitrified HLW.

Currently, irradiated fuel in China is stored and most is in pools at reactor sites. A limited amount has been transferred from Daya Bay to the temporary holding pool at the Lanzhou pilot reprocessing facility, and construction of a dry storage facility at Qinshan Phase III is reported (Zhou and Zhang, 2010).

4.2 Institutional Arrangements

4.2.1 Institutional Framework

Three organizations were involved in drafting a key document governing the geologic disposal of HLW (see Section 4.2.2):

- The Ministry of Science and Technology (MOST),
- The Chinese Atomic Energy Authority (CAEA), and
- The Ministry of Environmental Protection (MEP).

These organizations / agencies represent the top governmental decision makers.

POLICY and OVERSIGHT - CAEA is the entity responsible for planning and funding of HLW nuclear waste management in China.

IMPLEMENTER - The implementing body responsible for geologic disposal is the Chinese National Nuclear Corporation (CNNC). CNNC is a state-owned organization and the owner of most of the operating nuclear reactors.

REGULATOR - The independent regulatory bodies that are relevant to the safety of irradiated fuel management are the Ministry of Environmental Protection (MEP), the Ministry of Health and the Ministry of Public Security (PRC, 2008). The Department of Nuclear and Ionizing Radiation Safety Management under MEP, in particular, the Division of Radioactive Waste Management, will establish regulations and issue permits, i.e., the regulatory authority.

ADVISORY and SUPPORT - Numerous government research institutes and universities participate in R&D activities associated with the long-term program towards an HLW repository. These organizations regularly hold meetings, workshops, and seminars and have close ties with their international counterparts. The leading organizations within this group are:

- CNNC’s Beijing Research Institute of Uranium Geology (BRIUG),
- Chinese Institute of Atomic Energy (CIAE),
- CNNC’s China Nuclear Power Engineering Co., Ltd. (CNPE), and
- Chinese Institute for Radiation Protection (CIRP).

BRIUG is responsible for site characterization, R&D activities involving bentonite, and performance assessment. Radionuclide migration research is primarily carried out by CIAE. CNPE will be responsible for the engineering design and construction of the repository, while CIRP will provide safety assessment, primarily, biosphere assessment expertise.

4.2.2 Legal and Regulatory Framework

In 2003, China passed a Radioactive Pollution Prevention and Treatment Law, which requires the disposal of high-level radioactive waste (HLW) from all sources in a central repository that is located in a deep geological formation (Article 43). Other important items in this law include:

- Site-specific regulations will be generated based on an environmental impact assessment (EIA) carried out by the Administration of Nuclear Facilities in conjunction with the Ministry of Environmental Protection. This requirement must be ratified by the State Council before taking effect (Article 44).
- Waste generators shall treat and condition waste before sending the conditioned waste to disposal as well as bearing the disposal cost (Article 45). The price for disposal will be determined by the government's financial agencies under the State Council in conjunction with the environmental protection administration.
- Establishment of a special governmental agency responsible for nuclear waste disposal (Article 46).

Beyond the above Law, China does not have specific regulations and technical standards for the geologic disposal of HLW. General laws that are applicable to nuclear safety are the Law on Environmental Protection, the Law on Prevention and Control of Radioactive Pollution, and the Law on Environmental Impact Assessment (PRC, 2008).

In 2006, three government organizations jointly issued a "Guideline to China HLW Geological Disposal R&D". This guideline planned a three-phase R&D program for the Chinese nuclear waste management:

- *Phase I*: From present to 2020, China will complete site selection and construction of an underground research lab (URL);
- *Phase II*: From 2020 to 2040, China will carry out *in-situ* underground experiments using the URL;
- *Phase III*: China will build the repository and start receiving wastes and operation of the repository by 2050.

Overall, the guideline is a milestone in the Chinese program and covers all the important technical and safety issues. Furthermore, the guideline lays the foundation for establishing a regulatory framework.

4.2.3 Waste Classification

China's waste classification system recognizes the following categories (PRC, 2008):

- *LLW*: waste with specific activity < 4 MBq/kg;
- *ILW*: divided into three sub-categories according to half-life:
  - Half-life > 60 days but ≤ 5 years – specific activity > 4 MBq/kg;
  - Half-life > 5 years but ≤ 30 years – specific activity < 4.10^5 MBq/kg;
  - Half-life > 30 years – specific activity > 4 MBq/kg and heat output < 2 kW/m^3;
- *HLW*: either
  - Half-life > 5 years but ≤ 30 years – heat output > 2 kW/m^3 or specific activity > 4.10^5 MBq/kg;
  - Half-life > 30 years – specific activity > 4.10^4 MBq/kg or heat output > 2 kW/m^3;
- *Alpha radioactive waste*: waste containing alpha-emitting radionuclides with half-life > 30 years, specific activity > 4 MBq/kg in a single container.

4.2.4 Funding

The Nuclear Power Mid- to Long-Term Development Plan asks for the establishment of an irradiated nuclear fuel post-processing fund. Until now, however, no details about the funding have been established. Currently, the primary investment is allocated towards the construction of nuclear power plants.

4.3 Site Screening, Selection and Characterization

China started scientific research on nuclear waste geologic disposal in 1985. Since then, the Chinese program has been slowly but steadily making progress with limited financial resources.

4.3.1 Site Screening and Selection

China initially selected six locations, distributed in various parts of the country, as the basis for initiating site selection:

- Southern, eastern, western, and northwest China, Xinjiang, and Inner Mongolia.

In the absence of specific regulations, the Chinese siting program follows, in principle, the IAEA guidelines (IAEA, 1981) together with the Chinese Regulations for Radioactive Waste Management issued by the then Chinese government agency on technologies and quality administration (Wang *et al.*, 2000). Currently, the R&D work is all focused on a long-term feasibility study of the Beishan site that is located in northwest China as shown in Figure 4-1.

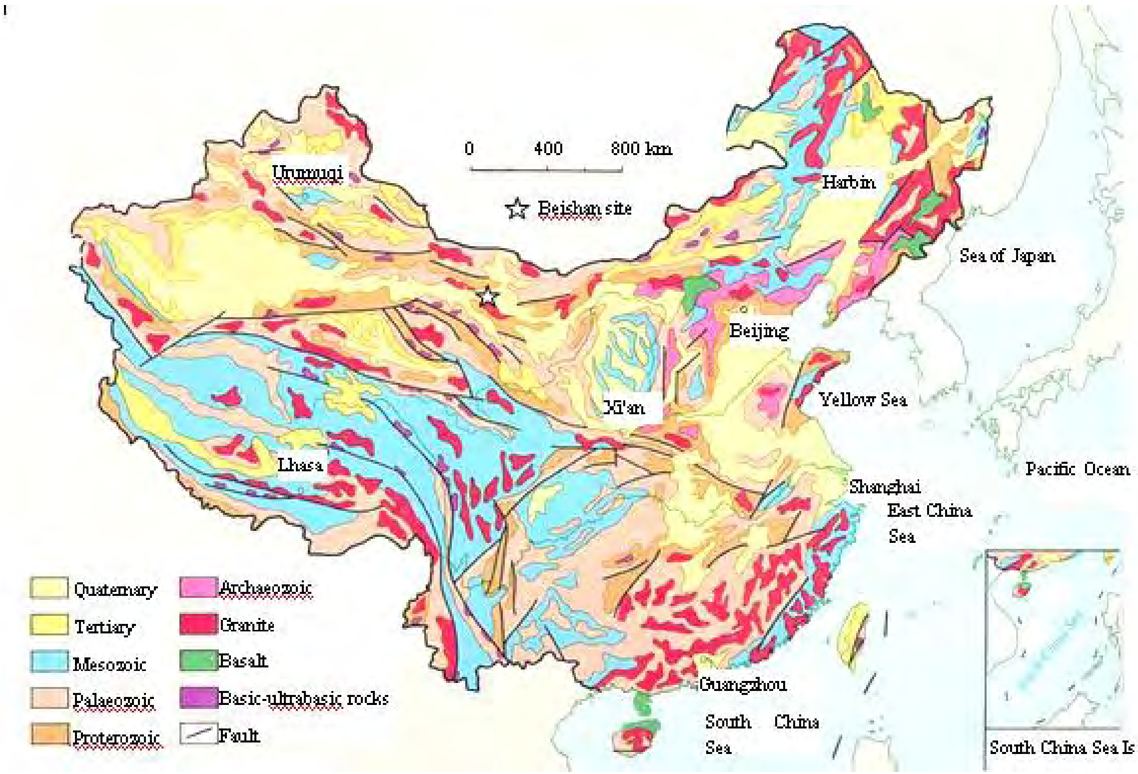

Figure 4-1
The location of the candidate repository site, Beishan, in northwest China (Wang, 2010a). Copyright 2010 Journal of Rock Mechanics and Geotechnical Engineering. Used with permission.

Beishan is located in the Gobi desert, in the Gansu province. The area has no groundwater resources, no important mineral resources, a relatively low population density, and no economic prosperity. The region hosting the Beishan site, an area of ~60,000 km^2 has a population of only about 12,000 people, almost all of them living in two towns. Only sporadic nomads roam the desert area.

4.3.2 Detailed Characterization of Beishan Site

Fourteen boreholes have so far been drilled at the Beishan site and substantial site data have been acquired.

Geological characterization shows that the crust in the area has a block structure with a thickness of 47 to 50 km and is stable (Wang *et al.*, 2004). The host rock is primarily granite with low permeability, low porosity and small linear thermal expansion coefficient. The hydrogeological characterization shows low water outflow in the area, with annual precipitation in the range 60 to 80 mm and a high evaporation rate, up to 2,900 to 3,200 mm. The deep groundwater is almost stagnant with a long circulation time, over a timescale of thousands of years. The groundwater chemistry analysis shows the deep groundwater to be of high salinity, slight-alkaline, and reducing conditions. All these properties make the Beishan site favorable for HLW disposal.

4.4 Disposal Concept

4.4.1 Overview

As stated in Section 4.1.2, China will process used fuel from light water reactors and the intended waste form is mainly vitrified HLW. A small amount of CANDU irradiated fuel is also slated for direct disposal in the repository.

According to China's plans, the final repository will be located in a deep geological formation, in water-saturated granite. The engineered barrier design will be similar to the KBS-3 type concept developed originally in Sweden, with similar tunnels and shafts. Bentonite is planned for use as the buffer material, emplaced between waste packages and the host rock.

Figure 4-2 shows schematically the Chinese disposal conceptual design. The vitrified waste form has a radius 0.3 m and length 1.1 m. The waste is contained in a 0.07 m thick canister surrounded by 0.8 m buffer material.

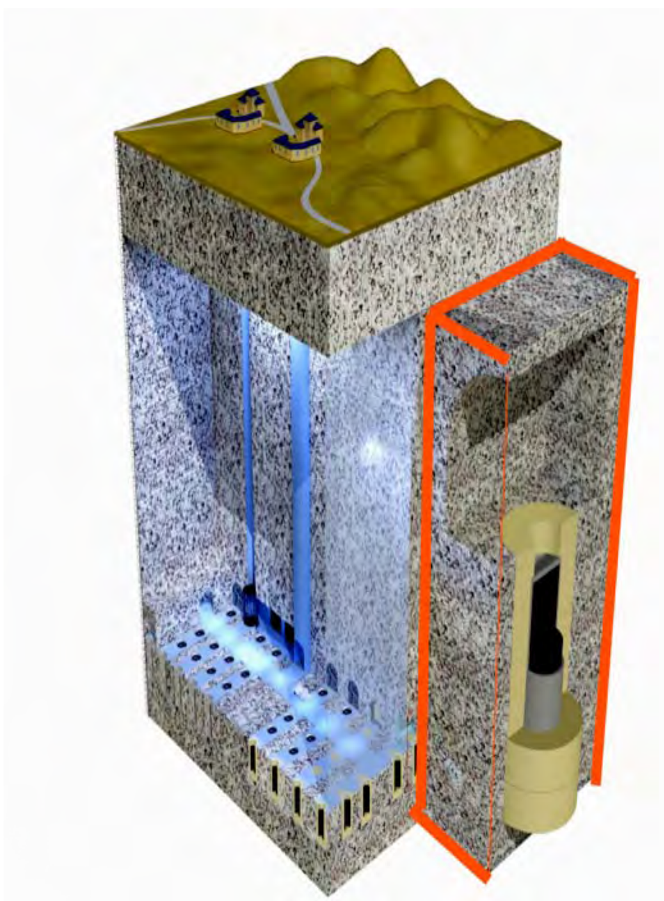

Figure 4-2
Schematic illustration of the Chinese disposal concept. The EBS contains the vitrified waste form, canister, and buffer (Wang, 2010a). Copyright 2010 Journal of Rock Mechanics and Geotechnical Engineering. Used with permission.

4.4.2 R&D Activities

During the program's initial period from the mid 80's to mid 90's, the focus was placed on acquiring site data. The objective was to determine if the site is tectonically stable, as well as obtain the groundwater flow rate and direction, geochemical conditions of the potential repository area. Boreholes were drilled to obtain rock and corresponding groundwater samples. Various geophysical methods were used, such as electromagnetic measurement, high-resolution magnetic survey, magneto-telluric survey, vertical seismic profile, borehole optical camera, acoustic televiewer, and radar survey. Researchers also carried out isotope analysis of water samples in order to derive and understand the sources of shallow and deep groundwaters.

Modeling was also carried out to simulate flow and stress fields (Wang *et al.*, 2000) including the regional flow field.

Over the years, the Chinese program has gradually expanded, and the study of bentonite has become an important part of R&D activities. The potential buffer material is derived from Gaomiaozi in Inner-Mongolia where the bentonite clay is relatively undisturbed. Most of the early research relied on laboratory experiments to measure the physical properties and sorption coefficients of the clay. More recently, research activities have expanded to include "mock-up" experiments. Numerical simulations of coupled processes in laboratory bentonite samples have been carried out and the results compared with field test observations.

4.5 Transparency and Stakeholder Involvement

4.5.1 Public Involvement

Since its inception, the Chinese program on geologic disposal of HLW has been open to both the domestic public and international communities, although domestically, the program has not attracted significant public attention. There are several reasons for this situation. Firstly, China has been undergoing significant economic reform and social development, in which an improvement in living standards is the first priority for the majority of ordinary people. Secondly, China does not have as long a history of nuclear power generation as that of developed countries and hence has not accumulated significant amounts of nuclear waste. Hence, the public does not feel an urgency to achieve the level of knowledge and information needed to understand the geologic disposal issue. Thirdly, other environmental and social issues have caused more direct and visible impact on the population than the nuclear waste issue, and hence have occupied public opinion. In 2007, the Chinese MEP issued a decree "Environmental Information Disclosure Methods", which requires that the ministry allows the public access to some form of record-keeping information provided by the government and private sectors. This decree provides a legal basis for public involvement in the geologic disposal of HLW both now and in the future.

4.5.2 International Involvement

On the international side, the Chinese HLW disposal program has made a direct appeal for collaboration and the exchange of information via a wide range of channels including IAEA, international conferences, inter-governmental collaborations, private visiting, training programs, and the exchange of students. Representatives of Chinese academia have offered to host international conferences in order to attract more attention from abroad. In addition, foreign visitors can easily obtain a permit to visit the Beishan site. As a result, there have been numerous visits by scientists from North America, Asia, and Europe. Meanwhile, Chinese scientists have been active and visible in various international meetings where the Chinese disposal program has been introduced (Wang, 2010b).

4.6 Safety Assessments and Licensing

4.6.1 Safety Assessment

Although China has not formally carried out a performance assessment (PA) associated with geologic disposal, scientists have been paying close attention to this topic, starting with relatively simple models to simulate radionuclide release and transport. Seminars and workshops have been held regularly to familiarize scientists with various aspects of PA. Given the large amount of data collected during site characterization and R&D activities, China is expected to conduct a formal PA in the near future (Wang, 2910b).

4.6.2 Licensing Process

The general licensing process for nuclear facilities in China is the same for nuclear power plants, other types of nuclear reactor site, fuel fabrication and used fuel reprocessing, waste storage, and disposal facilities. Each site requires a:

- Nuclear safety license – initially for facility construction, then for facility operations;
- Radiation safety license - as specified by the Regulations on the Safety and Protection of Radioisotope and Ionizing Ray Installations; and
- Qualification Certificates – qualifying the facility according to the requirements of the Law on the Prevention and Control of Radioactive Pollution. Approval is granted by the competent environmental authority of the State (MEP).

Site-specific regulations will be generated based on an EIA carried out by the Administration of Nuclear Facilities in conjunction with the MEP and ratified by the State Council before taking effect.

4.7 Current Status

The country is currently heavily involved in expanding its nuclear capacity by completing the construction of 24 new commercial nuclear reactors.

Although site investigations have focused on the Beishan site, national policy now requires comparative site selection before making the final decision on a suitable site. It is likely that a site in southern or eastern part of China will be selected. These alternative sites, however, all have a higher population density compared to Beishan. The alternative site will be compared with the Beishan site as a basis for supporting the ultimate decision on site selection.

As noted in Section 4.2.2, China plans to have a URL constructed by 2020. Thereafter, the long-term plan is to carry out detailed *in situ* investigations of the geology and hydrogeology, as well as perform tests and demonstrations of disposal technology.

Given an estimated completion date of 2050 for construction of the HLW repository, a detailed repository design is expected by 2030, following several years of conceptual and preliminary design stages. Research and development studies will focus on in situ tests, radionuclide migration, and development of engineered barrier construction technologies (Wang et al., 2006).

4.8 Summary and Key Observations

- *Policy on Geologic Disposal*: The Chinese government's Nuclear Power Mid- to Long-Term Development Plan calls for the development of a geologic repository but at a pace that is compatible with the development of nuclear power and the construction of nuclear reactors.
- *Institutional Arrangements*: The implementing body will be CNNC, the country's major nuclear utility. The regulatory authority is the Division of Radioactive Waste Management within the Department of Nuclear and Ionizing Radiation Safety Management under the supervision of the Ministry of Environmental Protection. Government plans call for the establishment of a dedicated irradiated nuclear fuel post-processing fund.
- *Key Laws and Regulations*: The Radioactive Pollution Prevention and Treatment Law, passed in 2003, requires HLW disposal from all sources in a central repository that is located in a deep geologic formation. While China does not have specific regulations and technical standards addressing the geologic disposal of HLW, three government organizations were involved in drafting a recent document covering key technical and safety issues associated with geologic disposal.
- *Site Screening and Selection*: In the absence of specific regulations, China's siting program follows, in principle, the IAEA guidelines. Six locations, distributed throughout the country, were selected initially as the basis for initiating site selection. Currently, the R&D work is focused only on a long-term feasibility study of a site located in northwest China, Beishan, in the Gobi desert. The area has no groundwater resources, no important mineral resources, a relatively low population density, and no economic prosperity. The host rock is primarily granite with low permeability and porosity with low water outflow in the area. The deep groundwater is almost stagnant with a circulation time of thousands of years. Chemical analysis indicates high salinity, slight-alkaline, and reducing conditions. While site

investigation studies are continuing at Beishan in the Gobi desert, existing national policy requires comparative site selection before the final choice is made

- *Repository Design Concepts*: The wasteform is HLW. According to current plans, the final repository will be located in a deep granite formation. The intended EBS is similar to the Swedish KBS-3 type concept, with similar tunnels and shafts, and bentonite as the buffer material, 0.8m thick. The vitrified wasteform will be contained in a 0.07 m thick (inner) canister. The possibility of retrievability has not been discussed.
- *Performance Metrics and Assessments*: No information is available. While no assessments have yet been carried out, Chinese scientists are acquiring knowledge and experience in assessment tools. A formal PA is expected in the near future.
- *Independent Peer-Review and Advisory Bodies*: Numerous government research institutes and universities participate in R&D activities associated with the long-term program towards an HLW repository. These organizations regularly hold meetings, workshops, and seminars and have close ties with their international counterparts. The leading organizations within this group are BRIUG, CIAE, CNPE, and CIRP.
- *Stakeholder and Public Involvement*: The HLW disposal program has not attracted significant public attention, primarily because the commercial nuclear power program is at the development stage. There is no impediment, however, to public scrutiny as a recent (2007) decree “Environmental Information Disclosure Methods” provides a legal basis for public access to documents and involvement in the geologic disposal of HLW both now and in the future.
- *Program Maturity*: The relatively recent radioactive waste management program reflects the country’s developing nuclear profile in terms of nuclear power plants and commercial power generation. Rapid growth in both the construction and commissioning of commercial nuclear reactors and the pursuit of a geologic repository is expected.
- *Additional Observations*: Although site investigation studies are continuing at Beishan in the Gobi desert, north-west China, existing national policy requires comparative site selection before the final choice is made.

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5
FINLAND

5.1 Introduction

5.1.1 General Nuclear Profile

Finland generated 23 TWh (33%) of electricity in 2009 with its fleet of four commercial power reactors (WNA, 2010a). Teollisuuden Voima Oy (TVO) and Fortum Power and Heat Oy are the owners of the two corresponding nuclear power plant sites: Olkiluoto with two reactors (OL-1 and OL-2) and Loviisa with two reactors (Lo-1 and Lo-2), respectively. A third reactor (OL-3) is under construction at the Olkiluoto site. The Olkiluoto and the Loviisa nuclear power plants began operations between 1977 and 1979, respectively. The used fuel produced in Olkiluoto and Loviisa is being stored at the plant sites in used fuel pools. Used fuel from the Loviisa power plant was formerly returned to Russia, with the latest transfer to Russia in 1996. The currently operating reactors at Olkiluoto (OL-1 and OL-2) are Swedish-designed boiling water reactors (BWR), the Loviisa reactors are Russian designed-pressurized water reactors (VVER-440) and the third reactor in construction at Olkiluoto (OL-3) is a Generation III+ European Pressurized Reactor (EPR) design from AREVA (WNA, 2010).

According to Finnish law, all used fuel from Finnish reactors must be stored and permanently disposed of in Finland. TVO and Fortum have established their own on-site wet storage facilities. Finland has had an active geological repository program since 1983. TVO and Fortum formed Posiva Oy in 1995 as a private joint venture to manage used fuel and implement the geologic disposal program (Figure 5-1). With a similar host geology and environment, geologic disposal in Finland has benefited from the technical developments in the Swedish program, including the adoption of the KBS-3 disposal concept. Overall, the Finnish program is at an advanced stage, with site selection complete and progress towards the application for a construction license, which is expected to be submitted in 2012 (NEA, 2009; WNA, 2010b).

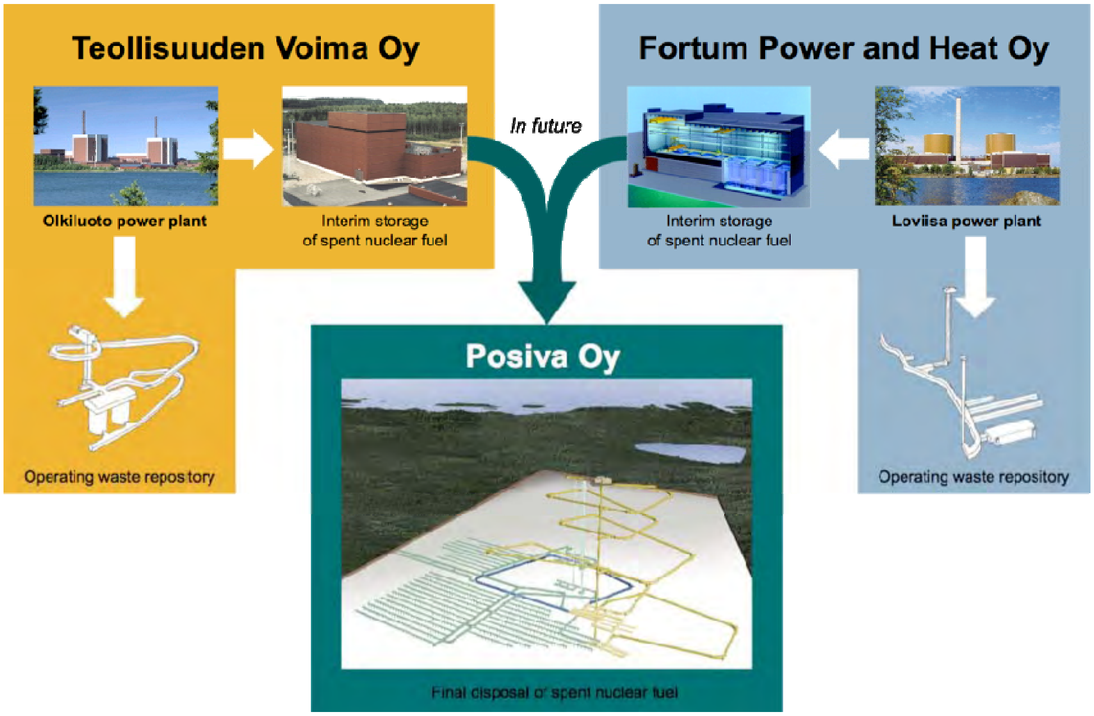

Figure 5-1
Basic relationship of nuclear power plant operators to the Posiva Oy organization instituted for final geological disposal of used fuel in Finland. Used with permission of Posiva.

5.1.2 Used Fuel / HLW Inventory and Projected Disposal Requirements

TVO and Fortum have estimated the amount of used nuclear fuel to be produced over the life of each reactor class and corresponding number of disposal canisters as follows (Posiva, 2007a):

- Loviisa 1-2: 700 canisters containing 1020 MTHM of used fuel;
- Olkiluoto 1-2: 1210 canisters containing 2530 MTHM of used fuel;
- Olkiluoto 3: 930 canisters containing 1980 MTHM of used fuel;

The above used fuel inventory estimates assume a maximum discharge burnup of 45 to 50 GWd/MTHM, and such the projections are subject to change with future changes in reactor operation. The total inventory for disposal amounts to 5,530 MTHM of used fuel in 2,840 canisters (Posiva, 2007a). Used fuel inventories are stored exclusively at the reactor sites in pools. At Loviisa, end-of-life wet storage capacity was added in 2000 in response to the end of return of used VVER fuel to Russia in 1996. Current used fuel storage capacity at Olkiluoto is reported to be sufficient up to 2014 (NEA, 2009; WNA, 2010b).

5.2 Institutional Arrangements

In 1983, the Finnish Government established guidelines and schedules for direct disposal of used fuel in Finland. The Nuclear Energy Act of 1987 (NEA, 1987) and its 1994 amendment established the overall process and schedule, establishing that nuclear waste could be neither exported nor imported, and excluded reprocessing of used fuel in practice.

As indicated in Figure 5-2, a Decision-in-Principle (DiP) issued by the Government is required before a Construction License for any significant nuclear facility can be applied for. An Environmental Impact Assessment (EIA) has to be conducted prior to the application of the DiP and the resultant EIA report annexed to the DiP application. In order to grant a DiP, one of the requirements is that the construction of the facility should be compatible with the overall good of society. Besides a requirement on safety, the municipality in which the facility is to be located must be in favor of constructing the facility.

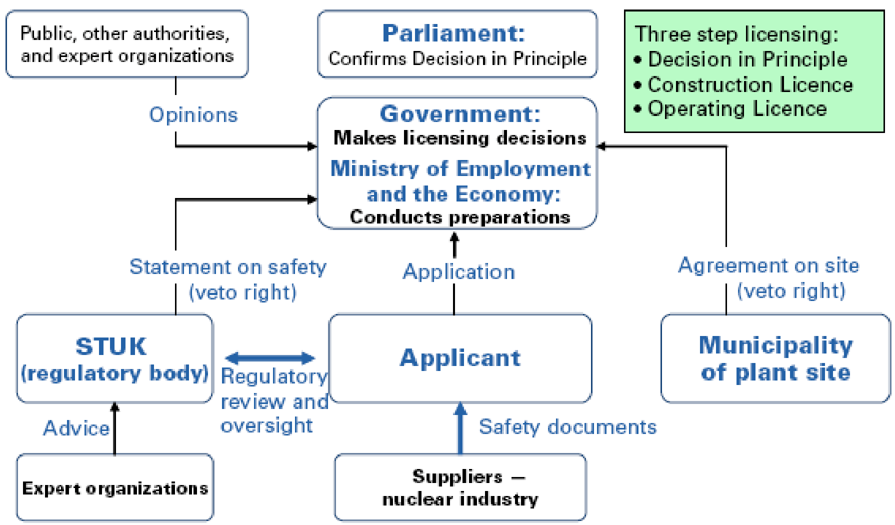

Figure 5-2
Implementation framework for licensing nuclear facilities in Finland (STUK, 2008). Used with permission of STUK.

5.2.1 Institutional Framework

POLICY and OVERSIGHT – The Ministry of Employment and the Economy (formerly the Ministry of Trade and Industry) has overall authority for nuclear energy related activities and is responsible for formulating national energy policy. The Minstry is also responsible for the licensing of nuclear waste management facilities.

IMPLEMENTER - For the purpose of final disposal, TVO and Fortum, owners of the nuclear plants, joined together in 1995 to form Posiva Oy as an independent private company to plan, construct, and operate a geological repository for final disposal of used fuel, although lower-activity wastes are being disposed of at the separate reactor sites for each of the companies in this consortium (Figure 5-1).

Figure 5-2 shows the institutional framework for licensing of nuclear facilities, including geologic disposal facilities, in Finland. In the case of the geologic repository for used fuel, the "Applicant" or licensee is Posiva Oy.

REGULATOR - The Radiation and Nuclear Safety Authority (STUK) is the independent nuclear regulatory authority in Finland. The relationship between STUK and other government organizations is shown in Figure 5-3.

ADVISORY and SUPPORT - STUK has an Advisory Committee on Nuclear Safety (by decree), while the Ministry of Employment and Economy has an Advisory Committee on Nuclear Energy and one for Radiation Protection.

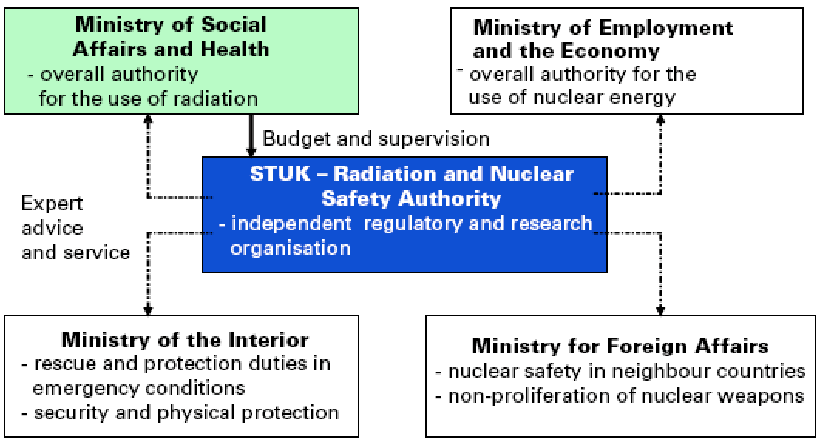

Figure 5-3
Relationship between STUK and Ministries and other government organizations (STUK 2008). Used with permission of STUK.

5.2.2 Legal and Regulatory Framework

The evolution of Finnish regulations traces back to the so-called Nordic Flag Book entitled, *Disposal of High Level Radioactive Waste; Consideration of Some Basic Criteria,* developed by the combined Nordic Radiation Protection and Nuclear Safety Authorities, A consultative version was published in 1989 and a final version in 1993. From 1993, STUK developed several versions of its regulations based on the Flag Book, with accompanying memoranda on reasoning and explanations.

To provide STUK with further guidance, the Finnish Government Decision 1999/478 defined the regulatory compliance period as "*an assessment period that is adequately predictable with respect to assessments of human exposure but that shall be extended to at least several thousands of years.* […] *Beyond the assessment period referred to above, the average quantities of radioactive substances over long time periods, released from the disposed waste and migrated to the environment, shall remain below the nuclide specific constraints defined by the Radiation and Nuclear Safety Authority.*"

The formal regulatory requirements and assessment time frames for the disposal are set forth in this Government DiP on the safety of the disposal of spent nuclear fuel (STUK 1999) and, in more detail, in the Regulatory Guide YVL 8.4 (STUK 2001), recently revised to YVL E.5 (STUK, 2009). A detailed discussion of regulatory requirements related to its planned safety case is given in Posiva (2006).

STUK's (2001; 2009) regulations make several important observations:

> *"Primarily due to the environmental uncertainties … it is not well-founded to extend individual dose or risk predictions …into the very far future. In general, dose assessments beyond about ten thousand years are very uncertain."*
>
> *"Dose assessments, in the relative sense, can be made for longer time periods…In this case the resulting doses or risks should be interpreted as safety indicators…not as predictions of really occurring doses. Such analyses should be extended to time periods until the calculated doses no longer increase or the related uncertainties are too high."*
>
> *"Realistic scenarios, models and input data should be applied … with the aim of comparing several different disposal systems, so as to facilitate a comparison between the various alternatives. The analysis of radiological impact can be truncated at the time when the subsequent impact is the same for all the alternatives or when it is no longer possible to distinguish between options due to large uncertainties associated with the calculations."*
>
> *"In the very far future, beyond millions of years, the total activity inventory of a high-level waste repository (around 100 TBq) will be less than that of a typical uranium ore deposit. … Thus, a repository of high-level waste can be regarded as a part of nature after millions of years and assessments of radiological impact need not extend beyond that time."*

Based on these reasons, regulatory requirements by STUK mandate that calculations for the quantitative safety assessment be extended to at least "several thousand years" after the closure of the repository (the so-called 'environmentally foreseeable future") to ensure that:

- The annual effective dose to the most exposed members of the public shall remain below 0.1 mSv, and
- The average annual effective doses to other members of the public shall remain insignificant.

The acceptability of these doses depends on the number of exposed people, but they shall not be more than 1/100 to 1/10 of the constraint for the most exposed individuals, i.e. no more than 0.001 to 0.01 mSv/year.

STUK's regulations also provide guidance on the potential exposure environments and pathways that shall be considered in the safety assessment. Disposal of used fuel shall not affect detrimentally species of fauna and flora. This shall be demonstrated by assessing the typical radiation exposures of terrestrial and aquatic populations in the disposal site environment, assuming the present kind of living populations. These exposures are to remain below the levels that, on the basis of the best available scientific knowledge, would cause a decline in biodiversity or other detrimental impacts to any living population. Moreover, rare animals and plants as well as domestic animals shall not be exposed detrimentally (YVL 8.4).

In the long term, after several thousand years, the quantitative regulatory criteria are based on constraints on the release rates of long-lived radionuclides from the geosphere into the biosphere. The nuclide-specific constraints for this period are defined in the Guide YVL E.5, and shown in Table 5-1. These radioactivity constraints apply to releases that arise from the expected evolution scenarios, and ratios of calculated release activities from performance assessments are normalized by the Table 5-1 biosphere activity limits. In the assessment calculations, the calculated release activities can be averaged over 1,000 years at most. The sum of the ratios between the nuclide-specific release activities and the respective regulatory constraints on release activities into the biosphere shall be less than one. The long-term safety relevance of unlikely disruptive events must also be assessed and, whenever practicable, the acceptability of the consequences and expectancies of radiation impacts caused by such events shall be evaluated in relation to the dose and release rate constraints.

Table 5-1
Activity Constraints on Radionuclide Releases into the Biosphere for 10,000 to several 100,000's of Years after Repository Closure (STUK, 2009).

Radionuclides	Release Activity Constraint [GBq/yr]
Long-lived alpha-emitting Ra, Th, Pa, Pu, Am and Cm isotopes	0.03
Se-79; I-129; Np-237	0.1
C-14; Cl-36; Cs-135; long-lived uranium isotopes	0.3
Nb-94; Sn-126	1
Tc-99; (*Mo-93*)	3
Zr-93	10
Ni-59	30
Pd-107; Sm-151	100

The unlikely disruptive events considered must include at least:

- Boring a deep water well at the disposal site;
- Core drilling hitting a waste canister;
- Substantial rock movement occurring in the environs of the repository.

In the very long term, after several hundred thousand or one million years, no rigorous quantitative safety assessment is required but the judgment of safety can be based on more qualitative considerations, such as bounding analyses with simplified methods, comparisons with natural analogues and observations of the geological history of the site (Ruokola, 2002).

The safety regulations imply that, in the safety case, the main emphasis shall be put on the isolation and containment capacity of the disposal system. For quantitative safety assessment, the STUK requirements lead to the identification of three time periods as follows:

- 0 to ~10,000 years: this is the "environmentally foreseeable future" and is the period that STUK's defined dose rate constraints apply. Biosphere transport and dose assessments need to be performed for those radionuclides that might be released into the biosphere during this period.
- 10,000 to several 100,000's of years: this is the period of "large-scale climate changes" when episodes of permafrost and glaciations are expected, and radiation protection criteria are based on STUK's geo-bio flux constraints (Table 5-1). No biosphere analyses are needed and dilution plays no role in the fulfillment of the regulatory constraints. Doses can still be used as safety indicators to gain additional insight to repository performance during this period.
- Several 100,000's to 1,000,000 years: no rigorous quantitative safety assessment is required for this "very far future" period but the judgment of safety can be based on more qualitative considerations, such as bounding analyses with simplified methods, comparisons with natural analogues and natural fluxes of radioactivity, and observations from the geological history of the site (Ruokola, 2002).

5.2.3 Waste Classification

The Finnish waste classification system (STUK, 2008) has two categories for waste:

- Nuclear waste: and
- Radioactive waste not originating from nuclear energy and the nuclear fuel cycle.

Thereafter, nuclear waste is categorized according to disposal route, *viz.*

- Irradiated nuclear fuel (deep repository)
- Low and Intermediate-Level Waste (rock caverns at intermediate depth).

5.2.4 Funding

Licensees of nuclear facilities pay into a special fund for waste management called the State Nuclear Waste Management Fund, independent of the State Budget but controlled and administered by the Ministry of Employment and the Economy, previously the Ministry of Trade and Industry. The charges are established annually by the government based on assessments for each company. By the end of 2008, approximately 1.7 billion Euros had been accumulated in the Fund from charges on electricity generated, which account for about 10% of nuclear electricity production costs. The waste management obligation of the licensees expires when STUK has confirmed that the nuclear waste is permanently disposed of in an approved manner (NEA, 2009; WNA, 2010b).

5.3 Site Screening, Selection, and Characterization

From 1983 onward TVO (superseded by Posiva in 1996) conducted a step-wise program for screening and selecting candidate sites for a final geological repository. This process is illustrated in Figure 5-4.

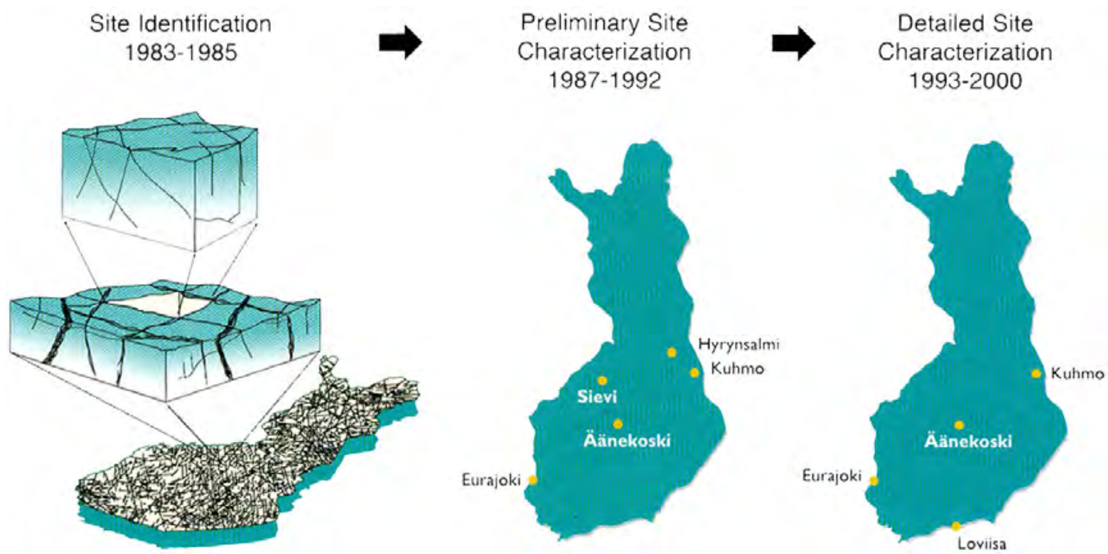

Figure 5-4
Step-wise Siting Process conducted in Finland from 1983 to 2000 (Posiva, 1999c). Used with permission of Posiva.

5.3.1 Site Screening and Identification: 1983 to 1985

As a first step in screening of sites, TVO evaluated the bedrock of all of Finland during 1983-1985 in order to locate potentially suitable candidate sites for further characterization. This step allowed TVO to obtain information about typical deep bedrock conditions in Finland, as well as insights into comparative features and characteristics among different sites. Most of Finland is composed of hard, so-called 'crystalline" igneous and metamorphic rock of variable compositions and containing a certain degree of pre-existing fractures. A key aim was to identify both major fracture (or 'crush') zones within Finland that would be excluded from further consideration and identification of relatively intact rock blocks of suitable size for construction of a repository.

TVO developed a range of screening criteria, including:

- The homogeneity, size and depth of geological formations,
- The expected groundwater flow characteristics on the basis of topography,
- The impact of fracturing on the water conductivity of the bedrock and on constructability of a repository,

- The stability of the bedrock with respect to potential movement on pre-existing fracture zones, and
- The occurrence of natural resources.

In addition to geoscientific criteria, TVO also assessed factors such as:

- Population centers,
- Land ownership,
- Transportation routes,
- Conservation and groundwater-resource areas, and
- Land-use plans.

Summary reports[14] on 102 sites were sent to STUK and Government authorities in late 1985 as the completion of the first step in siting (e.g., Äikäs, 1985). At the same time TVO established the Swedish KBS-3 repository concept as their reference in order to begin relating future siting activities with future research, development and design activities on the repository. This decision for the Finnish and Swedish programs to adopt basically the same disposal concept and design reflects several factors, including geographical proximity, advantages in shared R&D funding, historical backgrounds, and similarity in fractured crystalline bedrock as the host rock for disposal.

STUK reviewed this initial screening, and noted that the areas considered featured similar rock formations (nominally crystalline granitic/ gneissic formations, also called 'felsic' [high-SiO_2 content] rocks), but accepted TVO's basic approach to selecting areas for preliminary site characterization. The Finnish Government also reviewed the TVO initial siting report and eliminated several sites and areas from further consideration based on alternative pending usages.

5.3.2 Preliminary Site Characterization: 1987-1992

In 1987 TVO selected five areas for preliminary site characterization, which included many of the principal candidate rock types within the category of crystalline rock. The actual choice of sites was also influenced by land ownership and discussions with local municipalities with regard to their acceptance for further characterization as a potential repository site. These five areas and their municipalities included:

- Veitsivaara in Hyrynsalmi,
- Kivetty in Äänekoski,
- Romuvaara in Kuhmo,
- Syyry in Sievi, and
- Olkiluoto in Eurajoki.

[14] Published in Finnish.

The areas selected represented several of the main rock types within the basic category of crystalline rock. The actual choice of sites was also influenced by land ownership and discussions with local municipalities with regard to their acceptance for further characterization as a potential repository site. Posiva issued a report on its preliminary site characterization activities, selection criteria, and related safety assessments for different sites in 1992 (TVO, 1992; Vieno et al, 1992).

5.3.3 Detailed Site Characterization: 1993-2000

Beginning in 1993 and through the time up to 2000, more detailed site characterization was conducted on four sites composed of fractured crystalline rocks:

- Kivetty,
- Romuvaara,
- Olkiluoto, and
- Hästholmen.

The Hästholmen site was added in 1997, in part because it was the site of Fortum's Loviisa reactors, and in 1996 Fortum had joined with TVO to form Posiva as the joint implementer for geological disposal of used fuel in Finland.

In order to evaluate deep geological conditions, a number of deep exploratory boreholes (between 8 and 13) were drilled at each site, with some exceeding 1 km in depth. Extensive mapping and geophysical and geochemical surveys were made of the four sites. In 1996, Posiva made a report on the key findings for each site (Posiva, 1996, in Finnish). In addition Posiva was conducting regional studies across Finland, not specific to any one site, regarding impacts of rock movements and prevailing conditions during past ice ages.

TVO/ Posiva also was requested by the Finnish Government to consider rock types composed of 'mafic' (low-SiO_2 content) volcanic and extrusive rocks in Finland. Although such rocks were speculated to have more advantageous characteristics than felsic granitic / gneissic rocks, investigations could not confirm this. Accordingly, Posiva's site characterization program remained focused on the originally selected four felsic rock sites.

5.3.4 Selection and Exploration of Olkiluoto Site (2000-today)

Following its step-wise program of preliminary site characterization and safety assessment comparisons, Posiva (Posiva, 1999b; Vieno and Nordman, 1999) identified Olkiluoto as the site for the Finnish used nuclear fuel repository and applied for a Decision-in-Principle by the Finnish Government (Posiva, 1999a). This process is discussed further in Section 5.6.2.

The Olkiluoto site is located on a small island (approximately 10 km^2 in area) on the southwestern coast of Finland and is separated from the mainland by a narrow strait. The western part of the island hosts the Olkiluoto nuclear power plant, with two reactors in operation and a third under construction, as well as a low and intermediate-level waste (VLJ) repository. Construction of the used fuel repository is planned near the centre of the island.

5.3.5 Underground Research Facility

Investigations on the suitability of Olkiluoto as location for a used fuel repository have been ongoing for over 23 years by means of air- and ground-based methods, including shallow and deep boreholes. Construction of an underground rock characterization facility, called ONKALO, began in 2004 (Posiva 2003a-c, Vieno *et al.* 2003). The ONKALO facility will be used to further characterize the properties of the bedrock, to test and develop repository construction, operation, and closure technologies, and to find the most suitable locations for the used fuel waste packages (Posiva 2003b).

A total of three shafts are planned in ONKALO: a personnel shaft (4.5 m diameter), a supply air shaft, and an exhaust air shaft (both 3.5 m diameter). Posiva notes that the ONKALO facility, as well as providing research premises for *in situ* investigations, has also been designed to serve as an access route to the repository when constructed.

Figure 5-5 shows a schematic diagram of the planned ONKALO facility in its finished state, as well as an outline of the repository itself (Posiva, 2010a). As of October 22, 2010, construction had reached a depth of 426 m[15].

5.4 Disposal Concepts

5.4.1 Reference Design

The reference design for the repository is the vertical KBS-3V concept (Autio *et al.*, 1996); the expected performance is described in detail in Pastina and Hellä (2006). The reference design consists of a one-level underground facility with disposal tunnels at a depth of approximately 420 m, as shown in Figure 5-5. Specific details and functions of the engineered barrier system of the vertical KBS-3 repository are shown in Figure 5-6.

Used nuclear fuel, in the form of whole fuel assemblies, will be sealed inside a canister structure that consists of a massive cast iron insert covered by a 50 mm thick copper overpack. The term 'canister' is often used in Finland to refer to the entire waste package comprising the used fuel, cast iron insert and copper overpack. Within the repository, canisters will be located individually inside deposition holes spaced at intervals along the floor of long, horizontal deposition tunnels. The void spaces between the canister and the rock in the deposition hole will be filled with rings and blocks of compacted bentonite clay. Each deposition tunnel will be backfilled with a clay-rich material after all deposition holes have had waste packages and buffer emplaced. At the end of the operational phase, all open regions in the repository, as well as most direct access points to the surface (e.g. boreholes) will be backfilled and sealed to limit access of water to the repository area, although some boreholes may be kept open to allow for remote monitoring of the repository.

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[15] http://www.posiva.fi/.

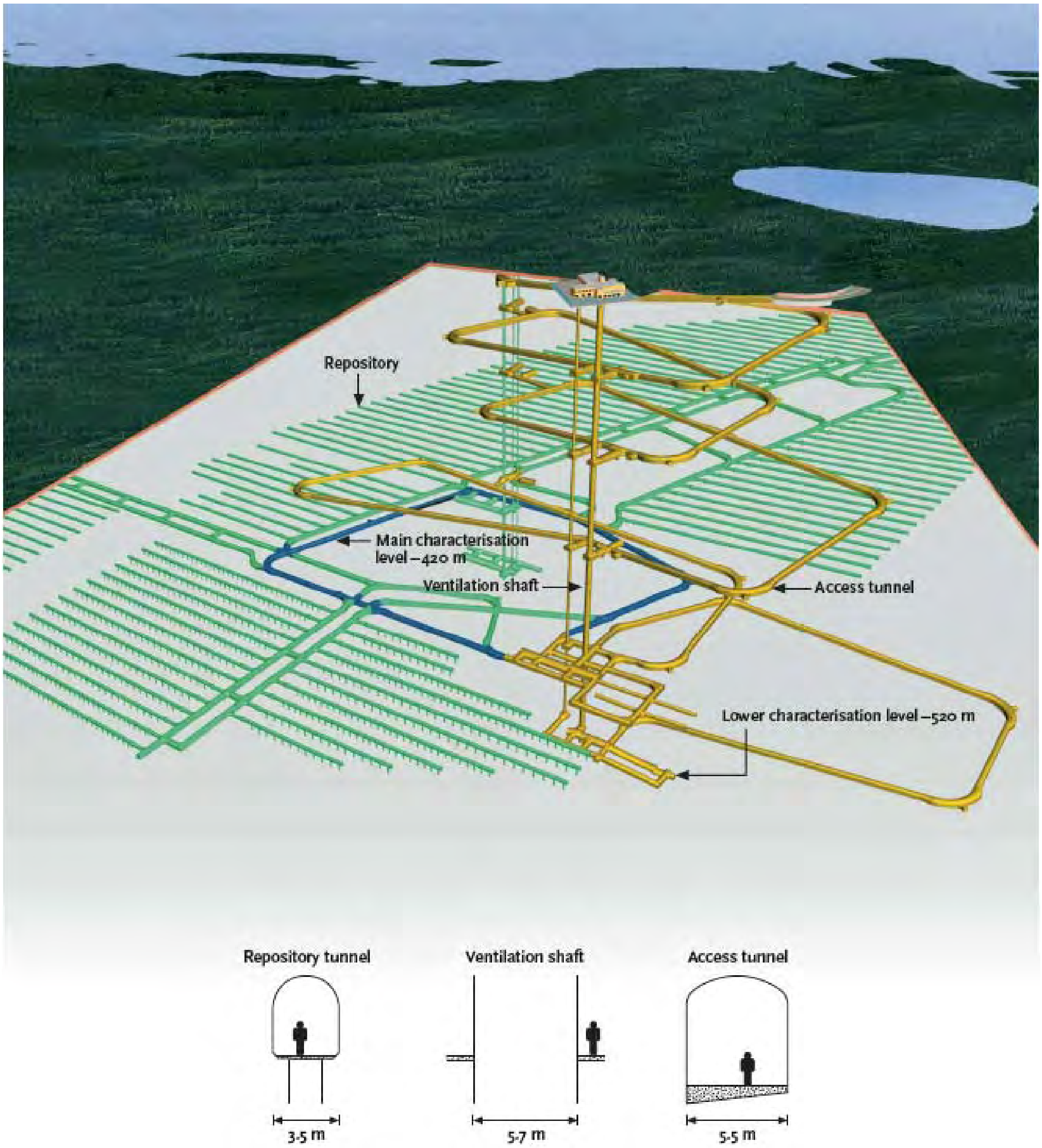

Figure 5-5
Schematic diagram of ONKALO characterization facility and the repository (Posiva, 2010a). Used with permission of Posiva.

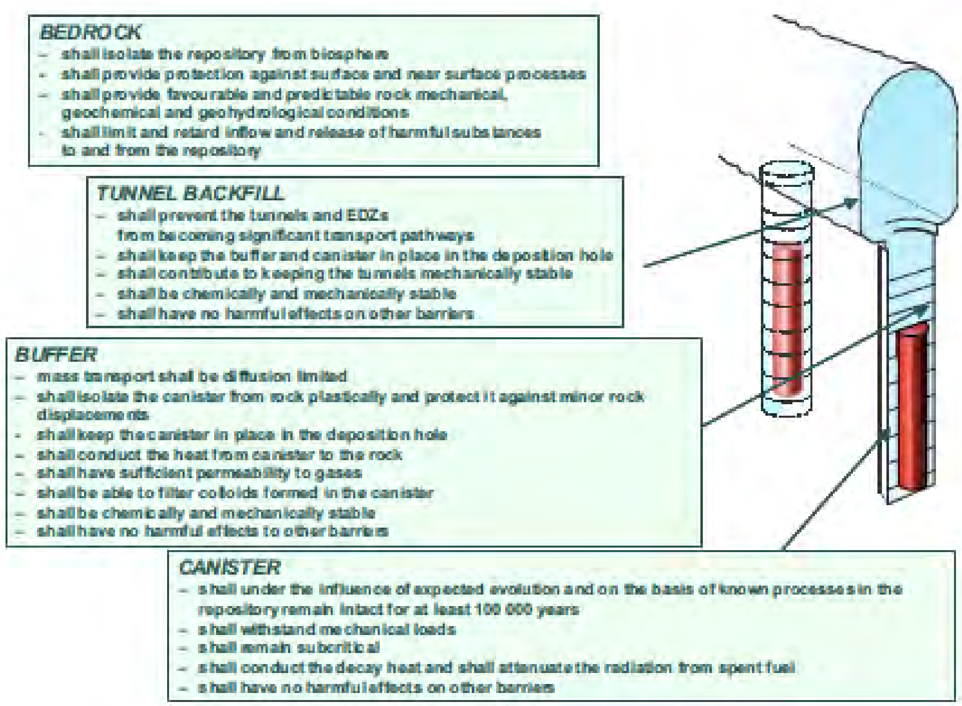

Figure 5-6
Schematic Illustration of the Engineered Barrier System for the Vertical KBS-3 Concept, and Associated Safety Functions (Posiva, 1999b). Used with permission of Posiva.

With respect to the KBS-3V concept, certain issues have recently been addressed regarding the full-scale constructability of this design. Handling and transportation of compacted bentonite buffer in humid underground mines have proven challenging. High rates of groundwater inflow in parts of certain sites have posed difficulties in achieving a sufficiently high rate of backfilling of access tunnels above vertical deposition holes. Emplacement of used fuel canisters up to 5-m long into vertical deposition holes has required creative handling methods, and is complicated by the potential for slightly higher stress anisotropy that may require excavated tunnels with flattened-oval cross section (i.e., a huge horizontal dimension in order to accommodate a vertical ceiling height of nearly 15-m) in both Finnish and Swedish bedrock.

Because of these uncertainties in constructability, a large number of studies on a horizontal emplacement concept, called “KBS-3H” (Figure 5-7) have been conducted recently by Posiva and SKB jointly (Posiva, 2007 a-d; 2008). While this alternative design may overcome some of the issues associated with the KBS-3V vertical design, it is not yet clear if there may be new constructability issues associated with the KBS-3H configuration.

The repository depth, tunnel geometry and backfill material compositions will be finalized once ongoing site characterization, and research and development programs are completed. The underground characterization and research program, and the program for research, development and technical design for 2007–2009 are described in Andersson *et al.* (2007) and in Posiva (2006), respectively.

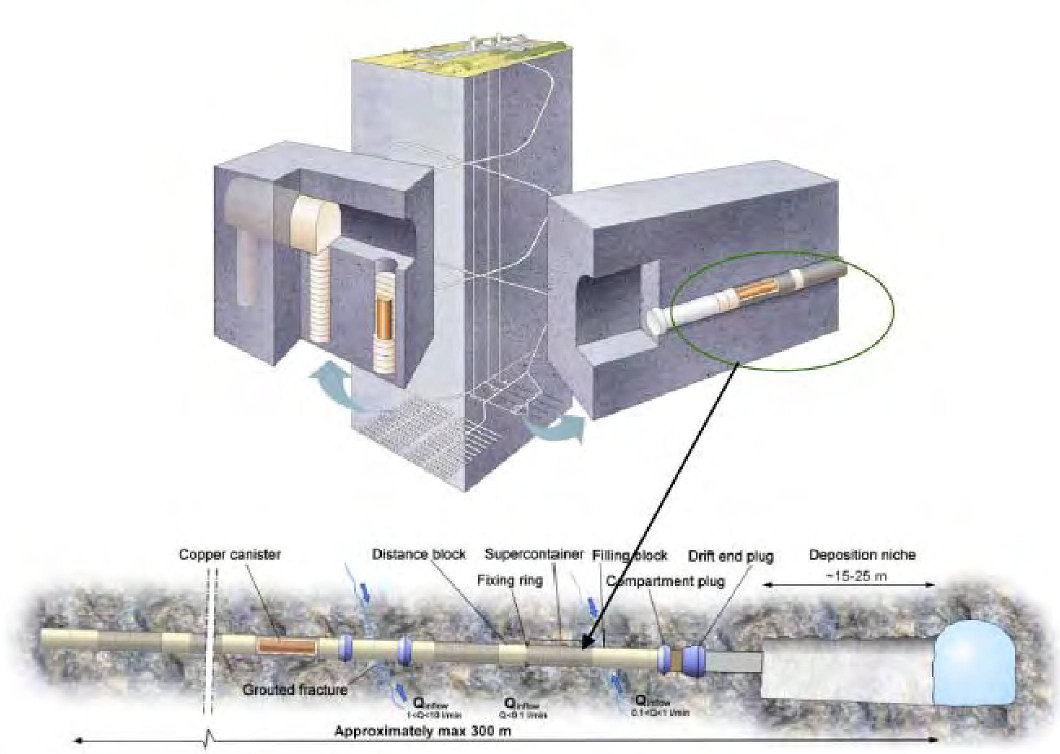

Figure 5-7
The KBS-3V design concept shown on the left and the alternative KBS-3H design concept shown at the right and below (Posiva, 2008). Used with permission of Posiva.

Alternative designs have considered a two-level design with disposal tunnels located vertically above each other separated by some tens of meters of rock (Saanio *et al.* 2006).

5.5 Transparency and Stakeholder Involvement

5.5.1 Public Involvement

By way of background information, in January 2010, a TNS Gallup survey (N=1000) commissioned by Finnish Energy Industries (*Energiateollisuus*) showed that 48% of Finns had a positive view of nuclear power, and only 17% were negative[16].

As discussed in Section 5.2.2.1, the EIA process is a key component of the application for a DiP and this process involves the general public during the public hearings part of the process.

Besides the EIA process, Posiva's proposal had strong local community support, with the Eurajoki Council, which had the right to veto the decision, voting 20 to 7 in favor of constructing the repository.

[16] http://www.world-nuclear-news.org/NP-Finns_more_positive_towards_nuclear-1502107.html.

5.5.2 International Involvement

Posiva has international agreements with a number of international organizations including SKB (Sweden), ANDRA (France), Nagra (Switzerland), OPG (Canada), and NUMO and RWMC (Japan), which allow the mutual exchange of information and research results. Posiva has particularly strong ties with SKB in terms of disposal concept (as discussed in Section 5.4.1) and also ANDRA on research into disposal facility systems and bedrock investigations.

The ONKALO underground research facility is already providing access to research tunnels for international involvement and collaboration.

The latest safety assessment carried out for the final disposal of used nuclear fuel according to the KBS3-V disposal concept was reviewed by an external (international) peer review group on behalf of STUK (Apted et al., 1999).

STUK also had an independent audit of its role in the Finnish repository program conducted through its membership in the European Union. A so-called "EU-27" team of outside regulatory experts met with and reviewed all aspects of STUK. The organization of EU-27 review areas, as defined in the IAEA *Safety Standard GS-R-1*, "Legal and Governmental Infrastructure for Nuclear, Radiation, Radioactive Waste and Transport Safety", included:

- Legislative and governmental responsibilities;
- Management system of the regulatory body;
- Authorization process and requirements;
- Review and Assessment procedures
- Inspection and Enforcement;
- Development and implementation of regulations and guidance;
- Organization of the Regulatory Body;
- Stakeholder relationship, public involvement.

The general conclusion of the EU-27 team was that

> *"STUK has a well established and apparently effective and efficient basis for regulating nuclear waste management in Finland. This is particularly impressive as the development of the Olkiluoto repository is at an early stage without as yet a fully developed safety case."*

5.6 Safety Assessment and Licensing

5.6.1 Safety Assessment

Posiva Oy commissioned a safety assessment (Vieno and Nordman, 1999) to support the submission of an EIA to the Ministry of Trade and Industry (KTM) with regard to a final disposal facility for used nuclear fuel. This safety analysis followed the same principles as those of the two previous safety assessments, TILA-96 (Vieno and Nordman, 1996) and TVO-92 (Vieno et al., 1992). No one specific site is considered but rather four potential host sites.

Treatment of the Reference Scenario is shown schematically in Figure 5-8 (Posiva, 1999c). This scenario assumes that conditions in the bedrock will gradually return to the pre-construction state and that subsequent changes in the bedrock will be relatively minor over a period of tens of thousands or hundreds of thousands of years. Any oxygen present in the bedrock as a result of the operational phase is expected to be consumed within a few hundred years. While sulfide dissolved in groundwater could cause significant corrosion of the copper canister, the levels of sulfide in Finnish groundwaters are exceedingly low, so that an estimate of millions of years would be necessary for canisters to fail by this type of corrosion (Posiva, 1999c).

A detailed description of the natural and repository systems is given for distinct time periods:

- First hundred years;
- 100 to 10,000 years;
- 10,000 to 100,000 years;
- 100,000 to 1,000,000 years; and
- Beyond 1,000,000 years.

Vieno and Nordman (1999) concluded that there were no releases for the Reference Scenario (intact canisters) and note that this conclusion is similar to assessments carried out in Sweden and Canada where similar canister materials (copper-iron) were adopted.

The possibility of one or more canisters failing over a one million year time period could not be discounted. Accordingly, as shown in Figure 5-8, variants of the Reference Scenario included a leaking canister, either as a result of a small pinhole (cross-sectional area ~5 mm^2), or from a larger hole (~1 cm^2). Results of calculations associated with these variants are shown in Figure 5-9 (Posiva 1999c). In addition, the results from the analysis of a scenario involving the total failure of a canister and release of its radioactivity are show in Figure 5-10.

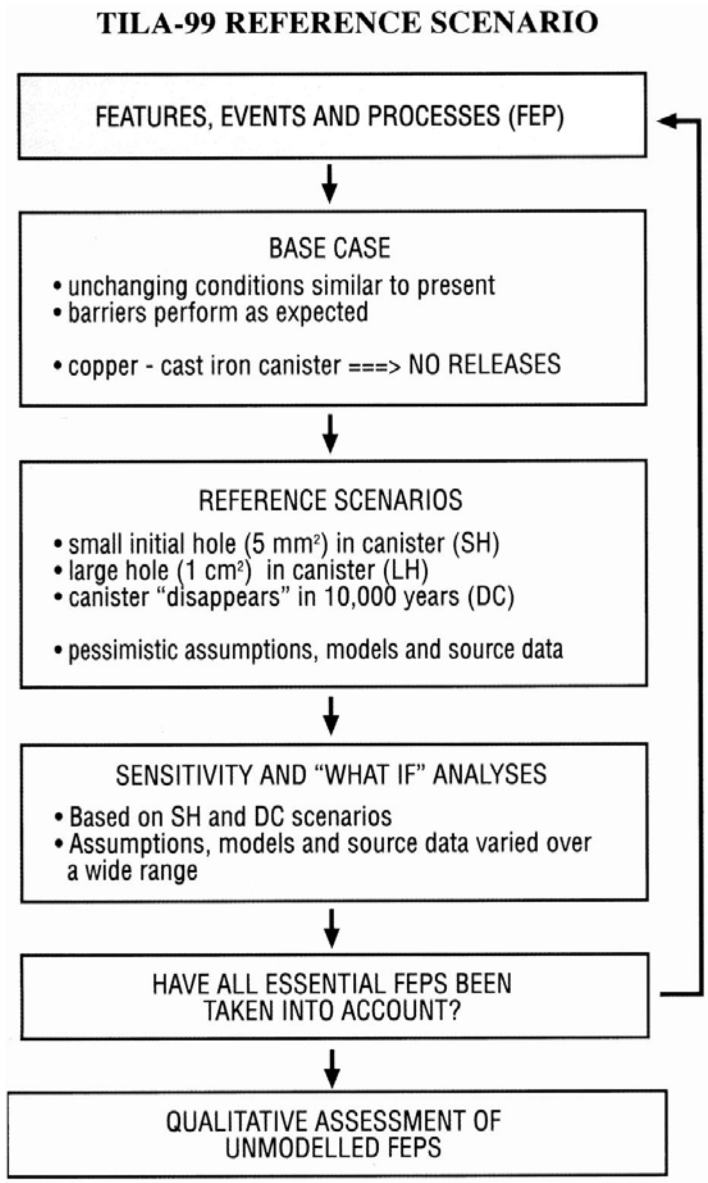

Figure 5-8
Reference Scenario and variants of Reference Scenario for TILA-99 (Posiva, 1999c). Used with permission of Posiva.

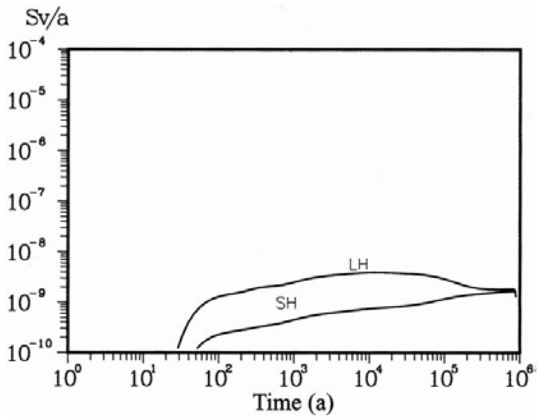

Figure 5-9
Maximum annual individual doses from initially defective canister under median flow and transport conditions (SH = small hole; LH = large hole, see text) (Posiva, 1999). Used with permission of Posiva.

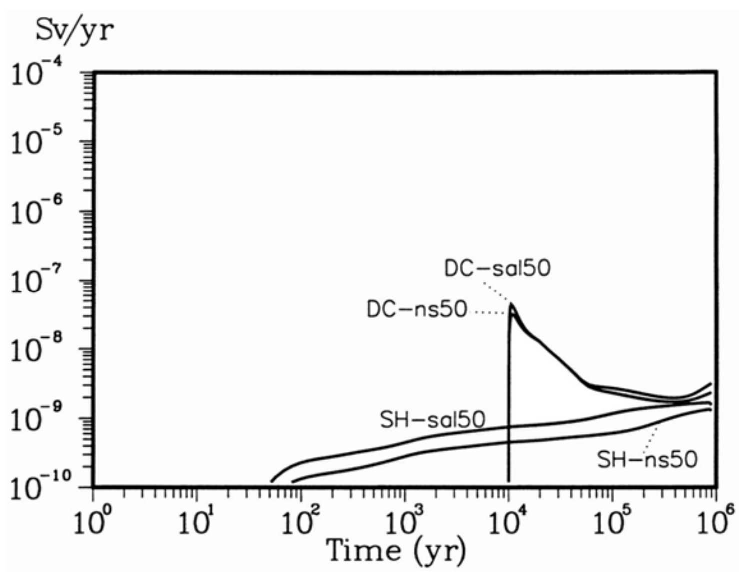

Figure 5-10
Total annual individual doses: Comparison of disappearing canister after 10,000 years (DC, total failure), and initially defective canister, small hole (SH) – under median flow and transport conditions (Vieno and Nordman, 1999). Used with permission of Posiva.

Note: sal50 = saline groundwater; ns50 = non-saline, fresh groundwater.

A more recent safety analysis has been carried out on the KBS-3H disposal concept, in which canisters containing used nuclear fuel are emplaced in a horizontal configuration rather than a vertical one (Smith et al., 2007). In this case, Olkiluoto was the site selected for the assessment, at a depth of about 400 m. Again, the possibility of canister failure is addressed via a range of assessment cases, in this case with the following canister failure modes:

- Initial penetrating defect;
- Canister failure due to corrosion;
- Canister failure due to rock shear.

The assessment relied heavily on data from the Swedish assessment SR-Can (SKB, 2006) except where site-specific data were justified, the latter being based on Olkiluoto. In all cases calculated, the results complied with Finnish regulatory criteria (Smith et al., 2007).

5.6.2 Licensing Process

As discussed in Section 5.2.2.1, Posiva's application for a DiP for the used fuel repository (Posiva, 1999a) was accompanied by an EIA of the Olkiluoto site (Posiva, 1999c). This application was reviewed and endorsed by STUK (STUK, 2000) and forwarded for a Government DiP. In 2000 the Finnish Government approved Posiva's application, and in 2001, the Finnish Parliament ratified the Government's favorable Decision-in-Principle on Posiva's application to locate the repository at the Olkiluoto island site.

Since then, following the requirements and framework time schedules set forth by the Finnish Government in the Nuclear Energy Act (as amended), as well as guidance from the Ministry of Trade and Industry, and the regulations set by STUK, Posiva has been following a step-wise approach to licensing, construction and operations (see Figure 5-11), *viz.*

- Pre-Construction License Application outline was sent to the Government in September 2009 and to STUK in December 2009, to be evaluated by STUK in 2010 who will then issue a Statement to the Finnish Government by the end of 2010;
- Publication of Posiva's forward research, development and design plans in its TKS-2009 report (presently available only in Finnish) to be evaluated by STUK and a Statement to be issued to the Finnish Government by the end of 2010;
- Posiva's Preliminary Safety Analysis Report (PSAR) of the used fuel;
- Disposal facility to be submitted to STUK in early 2010, followed by evaluation, then STUK will issue a Statement to the Finnish Government by the end of 2010,
- Construction License Application (CLA), supported by the full PSAR and Safety Case, to be submitted by 2012, which will consider:
- Authorization to construct deposition tunnels, deposition holes and other underground facilities,
- Authorization to construct the encapsulation plant and EBS components;

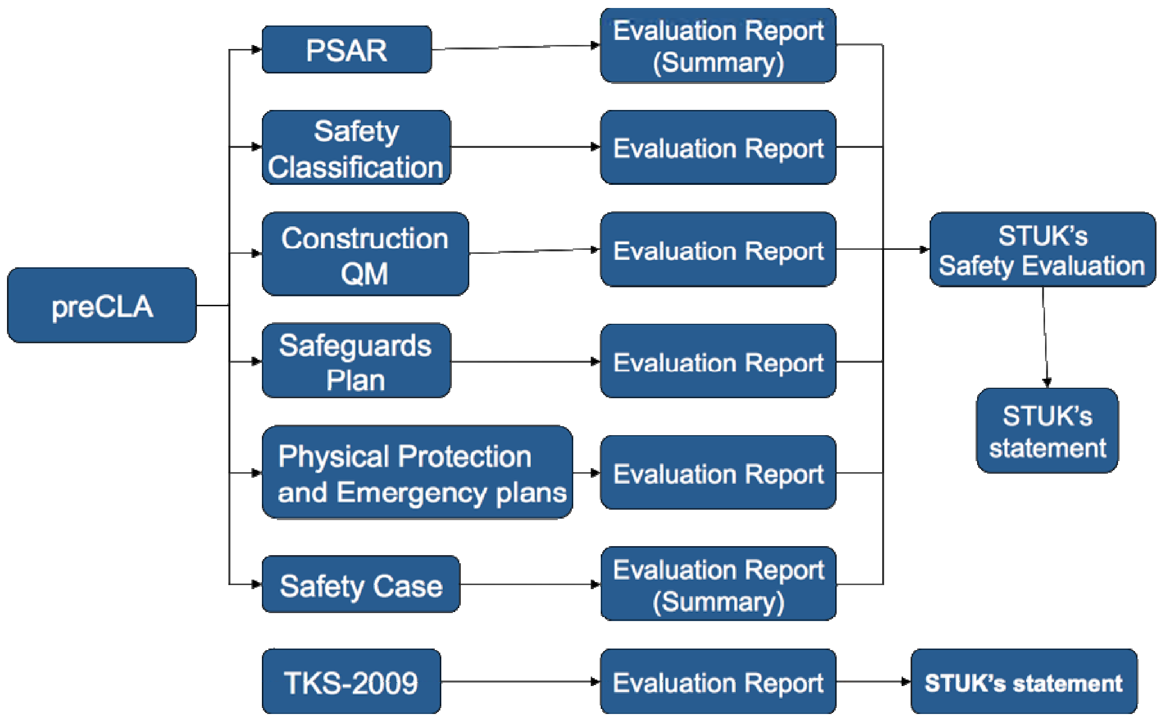

Figure 5-11
Content and Planned STUK Review of Posiva's Pre-Construction License Application (pre-CLA) and Research, Development and Design Plans (Posiva, 2010c). Used with permission of Posiva.

At this stage in the process, no nuclear waste can be introduced into the repository. Thereafter:

- Operating License Application (OLA), supported by the Final Safety Analysis Report (FSAR), to be submitted by Posiva about 2018, which if accepted will allow introduction of nuclear waste into the encapsulation and repository facilities;
- Instituting a fixed period of regulatory oversight, with full safety reviews at 15-year intervals (or as specified in license authorization);
- Begin repository operations in the early 2020's; and
- Construction of the repository in several stages with the total operation period possibly extending to about 100 years.

Details of the phased activities, schedules, and associated reports to support Posiva's overall licensing activities are provided in Posiva (2010b).

5.7 Current Status

Figure 5-12 provides a summary overview of the entire Finnish program towards the final disposal of used nuclear fuel.

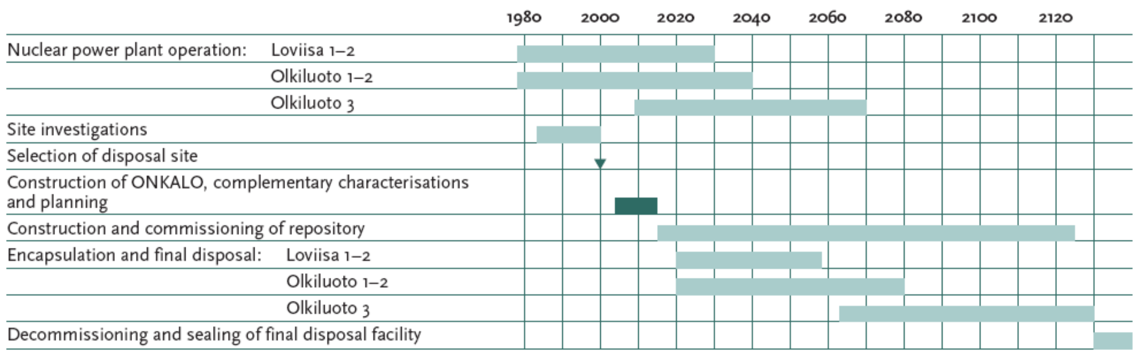

Figure 5-12
Preparation and implementation of the final disposal of used nuclear fuel in Finland (Posiva, 2010a). Used with permission of Posiva.

In 2008, TVO applied for a DiP to construct a 1000-1800 MWe PWR or BWR unit as Olkiluoto 4. This construction license was granted in May 2010.

Meanwhile, Posiva is proceeding systematically towards the licensing phase for the construction of a geologic repository for the final disposal of used nuclear fuel (see Figure 5-13). Posiva plans to submit the license application by about 2012. The operating license application is expected in 2018, with a view to starting the operational phase in 2020. The total period of operation of the geologic repository for irradiated nuclear fuel may extend to about 100 years (Andersson *et al.* 2007).

The repository will be constructed in several stages, with the construction of new disposal tunnels continuing in parallel with operations. Posiva recently proposed an increase in the final size of the repository, from its planned capacity of 6500 tonnes of used fuel to 12,000 tonnes, in order to accommodate waste from Olkiluoto 4 and the proposed Loviisa 3 power plant. STUK supported this proposal, and in July 2010, Parliament voted in favor of an expansion to 9000 tonnes to accommodate the used fuel from Olkiluoto 4[17].

[17] http://world-nuclear.org/info/inf76.html

Figure 5-13
Access tunnel to the disposal facility for irradiated nuclear fuel under construction at Olkiluoto (STUK, 2008). Used with permission of STUK.

STUK is also preparing to receive and evaluate Posiva's CLA, planned for 2012. The regulator has outlined what topics are expected to be included in Posiva's PSAR and eventual license application (see Figure 5-11). As part of its preparations, STUK has established several working groups composed of STUK staff and outside technical experts on the EBS, the natural barrier system and overall safety assessment. Furthermore, STUK convenes regular meetings with Posiva and its technical contractors to review key concerns and outstanding technical topics. STUK has implemented a so-called 'traffic light' approach to evaluate the degree of confidence Posiva has achieved in addressing and resolving key issues related to the constructability and long-term safety of the repository.

5.8 Summary and Key Observations

- *Policy on Geologic Disposal*: Finland is actively pursuing a geologic repository for used nuclear fuel and has reached the licensing stage for the construction of such a repository. A license application is anticipated at the end of 2012.
- *Institutional Arrangements*: The two utility companies in Finland, TVO and Fortum, joined together in 1995 to form Posiva Oy as an independent private company to plan, construct, and operate a geological repository for final disposal of used fuel. STUK is the Finnish regulator. Licensees of nuclear facilities pay into a special fund for waste management called the State Nuclear Waste Management Fund, independent of the State Budget but controlled and administered by the Ministry of Employment and the Economy, previously the Ministry of Trade and Industry. The charges are set annually by the government according to the assessed liabilities for each company, and also cover decommissioning.
- *Key Laws and Regulations*: STUK developed its regulations largely based on a document prepared originally by the combined Nordic Radiation Protection and Nuclear Safety Authorities (the Nordic Flag Book) and published in final version in 1993. The Finnish Government provided STUK with additional guidance in its Decision 1999/478, which defined the regulatory compliance period as "*an assessment period that is adequately predictable with respect to assessments of human exposure but that shall be extended to at least several thousands of years.* […] *Beyond the assessment period referred to above, the average quantities of radioactive substances over long time periods, released from the disposed waste and migrated to the environment, shall remain below the nuclide specific constraints defined by the Radiation and Nuclear Safety Authority."* Current regulatory guidelines by STUK (YVL 8.4 and YVL E.5) mandate that calculations for the quantitative safety assessment be extended to at least "several thousand years" after the closure of the repository (the so-called 'environmentally foreseeable future") to ensure that:
  - The annual effective dose to the most exposed members of the public remains below 0.1 mSv, and
  - The average annual effective doses to other members of the public remains insignificant.

  The acceptability of these doses depends on the number of exposed people, but they must not be more than 1/100 to 1/10 of the constraint for the most exposed individuals, i.e. no more than 0.001 to 0.01 mSv/year.
- *Site Screening and Selection*: Posiva and its predecessor, TVO, have been conducting a step-wise program for screening and selecting candidate sites for a final geological repository from 1983. As a first step in screening of sites, TVO evaluated the bedrock of all of Finland during 1983-1985 in order to locate potentially suitable candidate sites for further characterization. Following the development of a range of screening and geoscientific criteria, TVO provided summary reports on 102 sites to STUK and Government authorities in late 1985. In 1987, following review and recommendations by STUK and the government, TVO selected five areas for preliminary site characterization, which included many of the principal candidate rock types, all within the category of crystalline rock. The actual choice of these sites was influenced by land ownership and discussions with local municipalities with regard to their acceptance for further characterization as a potential repository site. From

1993-2000, detailed characterization was carried out on four sites, including drilling deep boreholes, from which one site was identified for further detailed evaluation, on Olkiluoto, a small island to the southwest of Finland, and already hosting two nuclear power reactors.

- *Repository Design Concepts*: At the same time as TVO submitted its list of 102 potential sites to the government, it also established the Swedish KBS-3 repository concept as its reference in order to begin relating future siting activities with future research, development and design activities on the repository. The reference design is for vertical emplacement in a one-level underground facility with disposal tunnels at a depth of approximately 420 m. With regard to the EBS, whole used fuel assemblies will be sealed inside a canister structure consisting of a massive cast iron insert covered by a 50 mm thick copper overpack. Canisters will be placed individually inside deposition holes spaced at intervals along the floor of long, horizontal deposition tunnels, the void spaces between the canister and rock in the deposition hole being filled with rings and blocks of compacted bentonite. Each deposition tunnel will be backfilled with a clay-rich material after all deposition holes have had waste packages and buffer emplaced. Retrievability does not play a part in Finland's disposal concept.

- *Performance Metrics and Assessments*: Specific performance metrics are provided in STUK's regulatory documents. For quantitative safety assessment, the STUK requirements lead to the identification of three time periods:

  – *0 to ~10,000 years*: the "environmentally foreseeable future" and the period that STUK's defined dose rate constraints apply. Biosphere transport and dose assessments need to be performed for those radionuclides that might be released to the biosphere during this period.

  – *10,000 to several 100,000's of years*: the period of "large-scale climate changes" when episodes of permafrost and glaciations are expected, and radiation protection criteria are based on STUK's geo-bio flux constraints. No biosphere analyses are needed and dilution plays no role in the fulfillment of the regulatory constraints. Doses can still be used as safety indicators to gain additional insight to repository performance during this period.

  – *Several 100,000's to 1,000,000 years*: A "very far future" period for which no rigorous quantitative safety assessment is required but the judgment of safety can be based on more qualitative considerations.

  Assessments were carried out in support of the site selection program and, more recently, to support the EIA as part of the license application submission. In the EIA safety analysis, no one specific site was considered, but the four potential host sites. Detailed descriptions of the natural and repository systems were given for five time periods: first hundred years; 100-10,000 years; 10,000-100,000 years; 100,000-1,000,000 years; and beyond a million years. While no releases were obtained for the Reference Scenario (intact canisters), the possibility of one or more canisters failing over a one million year time period could not be discounted. Thus, variants of the Reference Scenario included a leaking canister, either as a result of a small pinhole (cross-sectional area ~5 mm^2), or from a larger hole (~1 cm^2) as well as a scenario involving the total failure of a canister. The peak dose from the latter scenario, the most constraining, yielded an annual dose < 10^{-7} Sv per year. The most recent assessment, of a KBS-3H (horizontal emplacement of canisters) yielded results that complied with the regulatory criteria.

- *Independent Peer-Review and Advisory Bodies*: An international peer group reviewed the latest safety assessment for the final disposal of used nuclear fuel according to the KBS3-V disposal concept on behalf of STUK. STUK also had an independent audit of its role in the Finnish repository program conducted through its membership in the European Union. While the government has no specific Advisory Group, it is open to comments received from expert groups within Finland.
- *Stakeholder and Public Involvement*: The Finnish public are well informed about nuclear power and the need for waste management, such that Posiva's proposal to construct a repository on Olkiluoto had strong local community support, with the Eurajoki Council, which had the right to veto the decision, voting 20 to 7 in favor of constructing the repository. In addition, the EIA process, a key component of the application for a DiP, involves the general public during the public hearings part of the process, giving them the opportunity to comment. In 2000-2001, the Finnish government reviewed STUK's favorable preliminary safety appraisal of the Posiva repository program and the accompanying municipal acceptance, and issued a formal Decision in Principle (DiP) that represented public acceptance and approval to proceed to the planned construction license application in 2012.
- *Program Maturity*: The Finnish program to identify a site for a geologic repository has developed in a stepwise manner since the first study of potential sites in the early 1980's, culminating in the identification of a specific site at Olkiluoto. Posiva is now preparing a license application for the repository construction phase at this site.
- *Additional Observations*:
  - The program in Finland highlights that significant progress toward successful siting, design, confirmation testing, and licensing can be achieved with extremely limited financial and staffing resources.
  - The Finnish nuclear waste disposal program has common characteristics with, and is strongly connected to, the repository program in Sweden.
  - Posiva recently proposed an increase in the final size of the repository, from its planned capacity of 6500 tonnes of used fuel to 12,000 tonnes, in order to accommodate waste from the new Olkiluoto 4 and the proposed Loviisa 3 power plant. STUK supported this proposal, and in July 2010, Parliament voted in favor of an expansion to 9000 tonnes to accommodate the used fuel from Olkiluoto 4.

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6
FRANCE

6.1 Introduction

6.1.1 General Nuclear Profile

The French nuclear power fleet comprises 58 operating PWRs owned and operated by Electricité de France (EdF) and represents an installed capacity of 63 GWe. In 2009, the French fleet generated 392 TWh of electricity (WNA, 2010a). A 59th reactor, the sodium cooled fast reactor Phenix, ceased electric power generation in 2009. Second only the United States in size and generation capacity of its nuclear fleet, France leads the world in the contribution of nuclear to overall electricity generation, with a share greater than 75%. France is currently building a Generation III+ nuclear reactor and is expected to follow with a second in the near future (IAEA, 2009; WNA, 2010b).

Consistent with its national energy policy, France regards irradiated nuclear fuel as a resource rather than as waste, and the country reprocesses and recycles irradiated fuel on a commercial scale.[18] AREVA NC[19] operates a reprocessing plant at La Hague and a MOX fuel fabrication plant at Marcoule.

Based on the findings of Dossier 2005, France has identified a candidate host geology in the Callovo – Oxfordian layer of the Meuse and Haute Marne Districts. Public consultations are now focusing on acceptable and suitable locations for both the subsurface and surface facilities. A 2015 target is currently envisioned for submittal of a license application for repository construction (ANDRA, 2007).

6.1.2 Used Fuel / HLW Inventory and Projected Disposal Requirements

The anticipated volumes of HLW and LL-ILW expected by 2020 are 3,600 m^3 and 55,000 m^3, respectively (ANDRA, 2006).

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[18] The broader context for French nuclear energy policy is addressed in detail in EPRI report 1020307 (2010).

[19] Formerly COGEMA, which was created in 1976 from the production division operations of the French government's CEA.

Most irradiated nuclear fuel stored in France originates from PWRs, therefore containing either UO_2 or MOX (ASN, 2008). By 2010, La Hague had reprocessed approximately 27,000 tons of LWR fuel from France and other countries, representing over two-thirds of LWR fuel reprocessed worldwide. In addition, over 23,000 tons of fuel from the French fleet of gas-cooled reactors was reprocessed at La Hague and the Marcoule facility, which closed in 1997 (WNA, 2010b).

With reprocessing and Pu recycling in place, storage of most used fuel is short term in nature, with the storage time kept to the minimum period required to allow for transport and handling such that decay losses of the shorter lived ($t_{1/2}$ ~ 15 years) fissile isotope ^{241}Pu (and ingrowth of heat generating and neutron absorbing ^{241}Am), are minimized through to reprocessing and fuel fabrication process (NEA, 2007). With the short time horizons for irradiated UOX fuel management and the greater heat loads from irradiated MOX fuel, all storage of irradiated fuel in France is in pools.

At the end of 2007, the amount of irradiated nuclear fuel being stored in the country was approximately 12,800 MTHM, mainly at La Hague and EDF NPPs. An estimated 155 m^3 HLW and 930 m^3 conditioned LL-ILW are produced annually. The estimated amounts of processed LL-ILW and vitrified HLW in storage at the end of 2004 (ASN, 2008) were 45,518 m^3 and 1,851 m^3, respectively, and at the end of 2007 (NEA, 2009) were 41,757 m^3 and 2,293 m^3 (including 74 m^3 irradiated fuel). The corresponding volumes predicted by the end of 2030 are 51,009 m^3 of IL-LL and 5,060 m^3 of vitrified HLW, the latter also including 74m^3 irradiated fuel (NEA, 2009). LL-ILW is currently stored at three locations: La Hague, Marcoule and Cadarache; HLW stabilized in a vitrified waste form is stored at the facilities where it was produced, i.e., La Hague and Marcoule. (ASN, 2008; NEA, 2009). These inventories do not include wastes arising from foreign origin fuel, which are (by French law and contractual arrangement) to be returned to the country of origin after a suitable storage period.

With regard to disposal, several scenarios were taken into account when evaluating the repository concept discussed in the following sections. Thus, ANDRA defined four scenarios to estimate the repository inventory:

- Continued reprocessing of all commercial irradiated nuclear fuel (main scenario);
- All uranium oxide irradiated nuclear fuel is reprocessed but not MOX fuel (two scenarios), to allow consideration of direct disposal of MOX fuel. MOX makes up less than 10% of the overall reprocessing inventory, in terms of number of assemblies (ANDRA, 2005c, Table 2.1-7);
- Reprocessing of uranium oxide fuel is stopped after 2010, allowing for direct disposal of the remaining irradiated uranium oxide fuel, as well as irradiated MOX fuel.

In some cases, irradiated nuclear fuel may be designated for direct disposal, such as in the case of experimental reactors for which reprocessing of fuel is not economically feasible or becomes exceptionally complex.

6.2 Institutional Arrangements

6.2.1 Institutional Framework

Figure 6-1 shows the institutional framework for radioactive waste management in France. These include the following key roles.

POLICY and OVERSIGHT - Apart from Parliament, the Directorate of Energy and Climate for the Ministry of Ecology, Energy, Sustainable Development and the Sea (NEA, 2009) is responsible for establishing policy.

IMPLEMENTER – The National Radioactive Waste Management Agency (ANDRA) is the organization responsible for managing the different types of radioactive waste in France. Originally formed in 1979 within the Atomic Energy Commission (CEA), ANDRA gained independence in 1991 with the passing of the 1991 Law and supporting Decrees. As of December 1991, the organization became a public body, funded by government and industry, and supervised by the Ministries for Industry, Research and the Environment.

CEA was created in 1945 to perform research and development (R&D) for the implementation of civilian nuclear activities (energy, industry, research and health). In addition, CEA supported the development of national nuclear defense activities. As described in Section 1.1, CEA is currently responsible for R&D on partitioning and transmutation of long-lived radioelements.

ANDRA is funded by a "research tax", which is imposed on every nuclear facility that operates with a quantity of radioactive material above a certain threshold (INB, as defined in Decree No. 2007-830). ANDRA is also funded by research contracts with the various major waste producers.

REGULATOR - The regulatory authority in France responsible for radiation safety at all nuclear facilities is ASN, independent from the government as a result of the 2006 Planning Act (see Section 6.2.2). The duties of the regulator include supervision of ANDRA's overall organization for the design and operation of all disposal facilities. Thus, ASN authorizes the different stages of construction and operation of the URL as well as the eventual repository.

Beyond ASN, the government is responsible for issuing licenses (via Decrees) for the siting, construction, and operation of nuclear facilities, including waste disposal facilities.

ADVISORY and SUPPORT - The National Review Board (French acronym CNE), reviews key safety submissions of ANDRA and provides opinions and recommendations to the Government. In addition, ASN is supported by four Advisory Groups, one of which – the Advisory Group for Radioactive Waste - is relevant to disposal (NEA, 2009).

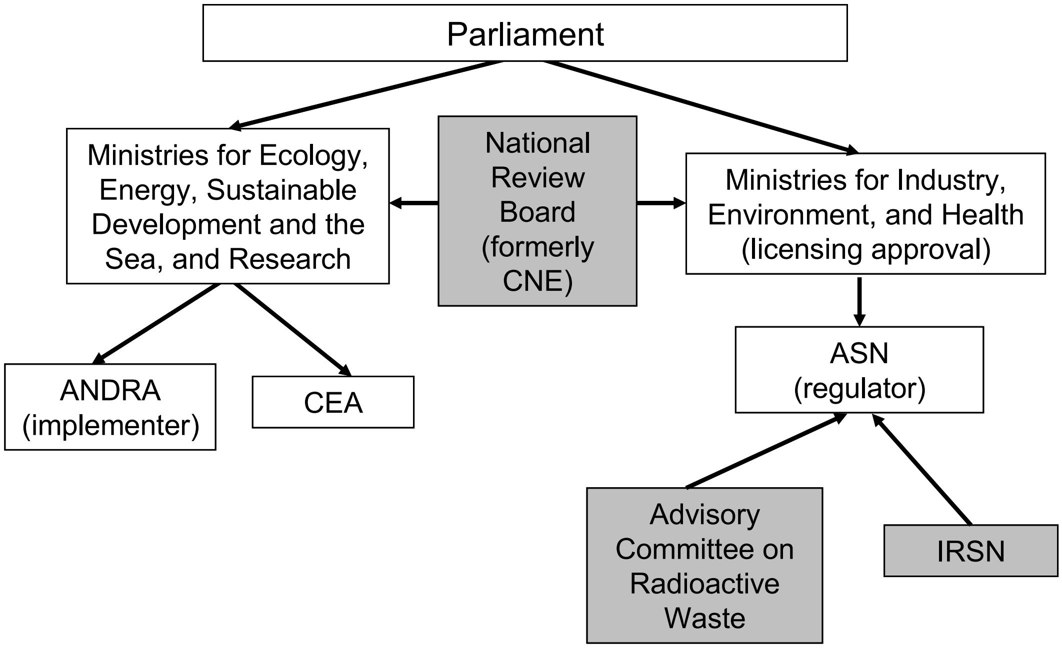

Figure 6-1
Institutional framework for nuclear waste disposal in France (adapted from NEA, 2006).

6.2.2 Legal and Regulatory Framework

French radioactive waste management policy, enacted via the 1991 Law (Loi de 1991), established a 15-year research period into the feasibility of deep geologic disposal, with a stepwise approach to nuclear waste management. The 1991 Law defined three complementary areas of research and development (R&D) concerning HLW and LL-ILW:

- Investigations on the feasibility of geologic disposal, both retrievable and non-retrievable, with a minimum time period of retrievability of 100 years;
- Partitioning and transmutation of long-lived radioelements;
- Conditioning and long-term storage facilities, to accommodate the volumes of waste to be expected by 2015.
- ANDRA (Agence nationale pour la gestion des déchets radioactifs) is responsible for the first area, while CEA (Commissariat à l'Énergie Atomique, the French Atomic Energy Commission) is responsible for the other two.
- As discussed in more detail in Section 6.3.2, the geological preferences for geologic disposal and the siting and operation of a URL were outlined in this Law, which required ANDRA to carry out investigations on clay and granite sites and to submit a safety assessment for each type of rock. The objective, as laid out in the 1991 Law, was to establish two URLs, one in clay and one in granite and to have a more democratic process whereby local communities were consulted and hosting communities were compensated.

The key pieces of existing legislation that govern the disposal of radioactive waste in France are:

- The *2006 Planning Act* concerning the sustainable management of radioactive materials and waste. This Act was adopted after the 15 years of research on radioactive waste management initiated by the 1991 Law, and reinforced ANDRA's mandate in the field of radioactive waste management and all types of radioactive waste. The Planning Act states that the host formation for a deep repository must have been studied previously via an underground research laboratory (URL).
- The *TSN Act* concerning the transparency and security in the area of nuclear activities. This Act provides a legislative basis for controlling nuclear safety and radiation protection. In particular, the TSN Act established the independence of the nuclear regulatory authority ASN (Autorité de Sûreté Nucléaire, French Nuclear Safety Authority).
- The *Environmental Code*. The Environmental Code addresses ANDRA's roles and responsibilities, in particular, the responsibility to initiate and pursue investigations on deep geological disposal, and specifies the process involved.

In addition, the regulation that specifically addresses HLW disposal is the Fundamental Safety Rule (RFS) III.2 issued in 1991 by the Ministry of Industry. This Rule is discussed further in Section 6.3.2.

6.2.3 Waste Classification

France recognizes six categories of waste according to half life and activity levels. Of these, HLW[20] comprises mainly vitrified waste in stainless steel containers as a result of reprocessing. In addition, intermediate-level long-lived radioactive waste (LL-ILW[1]; radionuclides with half life > 31 years) comprises waste generated also during the reprocessing of irradiated nuclear fuel, including structural components from fuel assemblies (hulls, end pieces), as well as 'technological' waste and residues such as solidified (bitumen) sludges from effluent processing (ANDRA, 2005b).

- HLW: typical activities ~ several billions of Bq per gram.
- LL-ILW: typical activities between a million and a billion Bq per gram, with negligible heat emitted.

6.2.4 Funding

Funding for geologic disposal programs is provided by the reactor operators. Each principal nuclear operator in France (i.e., EdF, AREVA, and CEA) administers its own independent, internal fund. However, these funds are subject to conditions set forth in the 2006 Planning Act to ensure adequacy of assets backing the funds' reserves and to independent external review. A national financial commission established under the 2006 Planning Act assesses the funding levels needed to cover the costs of waste management and reviews the methods used by

[20] The French equivalent nomenclature is HA for HLW (formerly Category C) and MAVL for LL-ILW (formerly Category B).

operators to estimate their waste management liabilities and the nature of the assets used to guarantee the funds. Apart from geologic disposal costs, other related costs, such as research and development and economic incentives for host communities, are funded via separate taxes on the nuclear operators (NEA, 2009). EdF is reported to allocate 0.14 cents per kWh of nuclear electricity generated for long-term waste management, which includes both reprocessing and disposal of long-lived (i.e., ILW-LL and HLW) wastes (WNA, 2010b).

6.3 Site Screening, Selection, and Characterization

6.3.1 Early Studies

In the 1970's, European Union-funded studies had been conducted in granite (France and UK), clay (Belgium and Italy), and salt (Germany, Netherlands). Building on the results from this European work, more focused studies were carried out in France in the early1980's to identify general attributes that were preferable for a repository site. As a result of the European work and the French studies, two generic rock types were identified as being suitable in France: clay and granite, resulting in two reports by individual Commissions (Castaing, 1984; Goguel, 1987) formed to identify the appropriate attributes. The Castaing Commission also noted that "*Potential variation in the properties of the formations, ranked by category, are so great that research must focus on specific rocks and sites as soon as preliminary screening is completed.*" (Castaing, 1984). Therefore, this Commission favored progressing towards site-specific investigations. Table 6-1 summarizes the characteristics of argillaceous and granitic rocks recognized as favorable by these Commissions.

In the context of exploitable natural resources, it was acknowledge that salt formations represented a potential resource, which resulted in excluding this generic rock type from further consideration.

Table 6-1
Favorable properties of argillaceous and granitic rocks (Castaing, 1984; Goguel, 1987).

Argillaceous rocks	Granitic rocks
Low permeability	Low permeability in non-fractured regions
Low spatial variability over a scale of kilometers	
Degree of plasticity, allowing deformation without fracturing	High mechanical strength
Ability of clay minerals to fix (sorb) radionuclides	Retardation (sorption) of some radionuclides
Absence of exploitable resources	Absence of exploitable resources

Once potential rocks were identified, the more specific topics for consideration included:

- Large enough volume of rock with favorable and relatively homogeneous characteristics to accommodate a repository;
- Understanding of the long-term natural evolution of a site in terms of the potential for erosion, glaciation, earthquakes,
- Evaluation of potential impacts on a natural rock formation by the construction of a repository;
- Evaluation of the potential for release of radionuclides once they reach the boundaries of the host formation?

Towards the end of the 1980's, CEA's borehole investigations on different types of rock in different parts of the country, as a means to obtain information on the above topics, met with substantial resistance, primarily from the local communities, such that a moratorium was called on these site investigations.

Following intense socio-political opposition to geological investigations in support of identifying potential host sites for geologic disposal, a totally new approach was necessary and the 1991 Law defined a national research policy towards the long-term management of French nuclear waste.

6.3.2 Implications of 1991 Law for Site Selection

The 1991 version of the Safety Rule III.2.f, which complements the 1991 Law, discusses a number of criteria or characteristics required for a disposal site for LL-ILW and HLW, *viz.*

- *Geological stability* for a period of at least 10,000 years;
- *Hydrogeology*: a low permeability in the host formation combined with a low hydraulic gradient. A low regional hydraulic gradient is also considered important and may be used to differentiate between sites.

Thereafter, four characteristics are recognized as being important for site assessment:

- *Mechanical and thermal characteristics*: the ability to design a repository that does not alter the geological barrier significantly.
- *Geochemical characteristics*: acknowledging the importance of geochemical characteristics and, therefore, requiring a quantitative description of the geochemical characteristics of the system.
- *Minimum depth*: sufficient to avoid the effects of erosion, seismic events or 'normal' intrusion.
- *Resources*: a location that avoids areas with a high potential for natural resources.

RFS III.2.f is also specific about faulting:

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

Christian Bataille, Member of Parliament and main proponent of the 1991 Law (also called the "Bataille Act") was appointed in 1993 to develop a dialogue with local authorities and representatives of business industry and social organizations in order to seek out volunteer candidate sites and explain the nature of the URL project.

Initially, four candidate regions were proposed and pre-selected according to their geological characteristics:

- Gard in the south of France for clay (Cretaceous siltstones of the Marcoule area);
- Haute Marne and Meuse in eastern France for clay (Sedimentary Paris Basin); and
- Vienne in western France for granite (La Chappelle-Baton).

Gard was rejected because studies indicated that the area was intersected by two major faults showing evidence of fluid circulation postdating deposition (Rousset *et al.*, 2005). Following geological studies above ground and public inquiries, the government approved the East Paris Basin site at Bure (the potential Meuse and Haute-Marne adjacent sites were combined into one site). The population density at the Bure site is low, with less than 3,000 inhabitants living in 312 small communities.

In 1999, following submission of ANDRA's report on the characterization studies at Vienne, the government abandoned the Vienne site based largely on technical reservations expressed by the CNE. At that time, the government decided to pursue a site in granite, but eventually was forced to abandon its plans owing to strong local opposition to the government's handling of new selected areas (NAS, 2001).

6.3.3 Site-Specific Geological Investigations – Bure Site (Meuse / Heute-Marne)

ANDRA's detailed geological investigations can be divided into four stages:

1994-1996: ANDRA's initial work in the Meuse / Haute-Marne region was aimed at assessing the quality of the region in an area of several hundred km^2 with a view to identifying a smaller area for more detailed investigation via a URL. Studies included an initial characterization of the properties of the different formations, but specifically the Callovo-Oxfordien. This involved surface geological mapping, data from two deep cored boreholes (~ 1 km each) about 15 km apart, continuous logs (resistivity, porosity, density), collection of core samples for laboratory tests, hydrogeological measurements, and the interpretation of seismic data from 68 petroleum exploration boreholes, some 1300 km of 2D seismic profiles.

Based on this work, the Callovo-Oxfordien, a thick clay layer of age about 160-million years, was identified as a potentially suitable formation for a URL.

1995-1996: Work carried out during this period provided the preliminary characterization necessary as a basis for URL construction and to be able to identify what studies needed to be carried out in the URL. The work during this period involved:

- Three 2-D high-resolution seismic profiles (total length 15 km);
- A 3-D seismic profile over a surface area of 4 km^2;
- Drilling of 27 boreholes in the region covering a total length of 5 km, with 4.2 km of core and rock of which 2.3 km was in the Callovo-Oxfordien. 23 boreholes were vertical and 5 deviated, allowing a more comprehensive interpretation of the formation in a horizontal direction as well as an examination of vertical fractures;
- Collection and analysis of 7300 water samples collected at different depths and different formations, to characterize the fluids present in different formations above and below and within the host formation.

1996-2004: In preparation for its own URL, ANDRA joined an international consortium to conduct studies/experiments at the Mont Terri road tunnel in Opalinus Clay, which has comparable characteristics to the Callovo-Oxfordien. By 1998, the consortium comprised 8 partners from 7 countries and focused on the design and set-up of experiments to collect key scientific data governing clay behavior, e.g. diffusion. ANDRA also took part in experiments at the Mol URL in Belgium, which helped identify instrumentation needs.

2004-: URL phase - construction of components of URL in Callovo-Oxfordien Formation involving:

- Excavation of two access shafts to the host formation, allowing an Auxiliary Shaft to ~490 m in October 2004 and the Principal Shaft that extended to 490 m by October 2005.
- Detailed *in situ* characterization of the different formations intersected by the shafts.
- Creation of a horizontal experimental gallery at 445 m off the Principal Shaft in the upper part of the Callovo-Oxfordien, which extends from 417 to 550 m. The experimental gallery, with cumulative length 35 m, allowed a variety of observations and experiments including the collection of fluids from the host rock and migration (diffusion) experiments (see Figure 6-2).
- A (second) migration experiment involving matrix diffusion – a state-of-the-art experiment relying on a borehole drilled from the surface. The results were used to complement those obtained from the initial gallery migration experiment.
- Creation of other experimental galleries at 490 m using two Auxiliary Shafts, with cumulative length of 200 m to pursue direct observation of the rock and further characterization of the excavation damaged zone (EDZ).

The overall experimental URL program was aimed at full-scale model validation based on data collected from the various experiments (diffusion, gas characterization, geochemistry, heating).

The transposition zone[21] is defined as a 200 km^2 area around the site of interest, i.e., the Bure site.

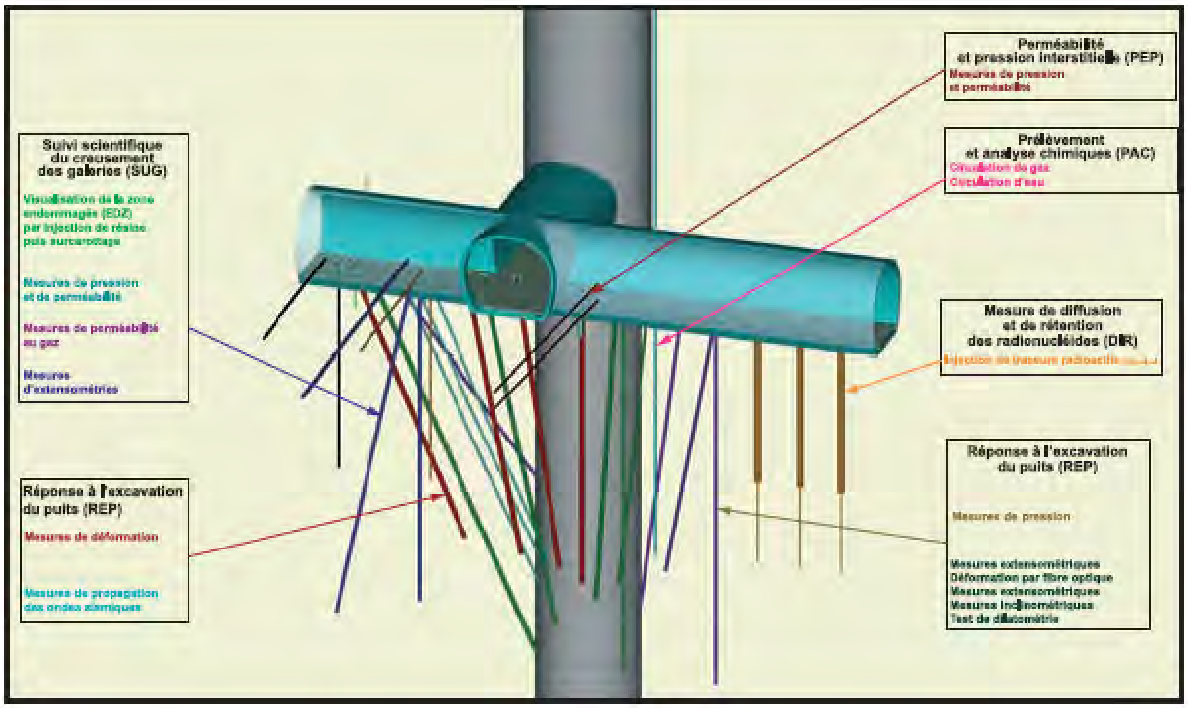

Figure 6-2
Experimental gallery at the Bure URL site (ANDRA, 2005a). Used with permission of ANDRA.

6.3.4 Key Features of Bure Site

The Bure site was shown to consist of a series of almost horizontal layers, with relatively simple geology (see Figure 6-3). Tectonic activity in the region is extremely low, confined to infrequent recurrence, if any, of pre-existing faults (ANDRA, 2005a). In addition, the host formation appeared to be sufficiently homogeneous with few discontinuities in the form of fractures or mineral composition.

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[21] Adopting the transposition approach, a geographical domain is defined so that it can be considered equivalent to the URL in terms of formation confinement properties and disturbance characteristics (ANDRA, 2005a).

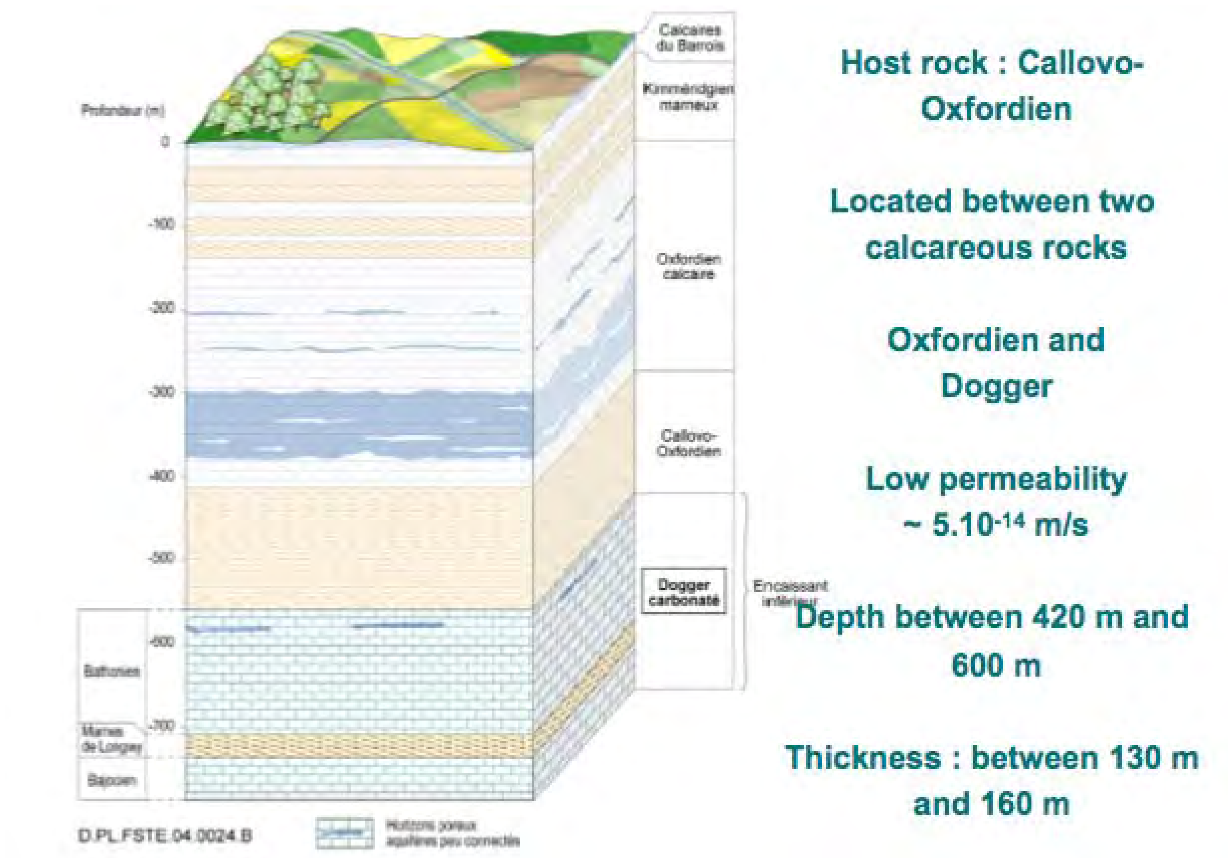


Figure 6-3
Geological cross-section of Meuse / Haute-Marne URL site (ANDRA, 2005a). Used with permission of ANDRA.

The clay formation itself comprises three main types of mineral:

- Sheets of clay minerals, which contribute to radionuclide retention;
- Carbonate minerals, which help to buffer pH and contribute to mechanical strength;
- Quartz, which has good mechanical and thermal properties.

Measurements of permeability indicate permeability in the range 10^{-12} to 10^{-14} m^2/s with most measurements between 5 x 10^{-13} and 5 x 10^{-14} m^2/s. Thus, the main transport mechanism is by diffusion. Isotopic studies on interstitial waters from the host formation and the layers above and below it indicate very slow transfer times.

6.4 Disposal Concept

6.4.1 Background - General Philosophy

The principal disposal concept to be evaluated included the disposal of both vitrified HLW and long-lived ILW. To cover possible evolution in France's nuclear fuel cycle, ANDRA was asked to take into account the possibility of irradiated nuclear fuel disposal. Safety Rule RFS III.2.f specifies a multi-barrier disposal concept.

In addition, the Government indicated that the emphasis should be on the "logic of reversibility", based on a report by the National Review Board (CNE, 1998). The concept of reversible repository is discussed by ANDRA:

> *"A reversible repository is one which, by its design and the quality of understanding available, allows waste management choices to be made at any time, as for an interim storage."*

and

> *"A reversible repository must be robust in the long term, with respect to the basic objectives of personal and environmental protection (it must be possible to close the repository when the decision to do so is taken)."*

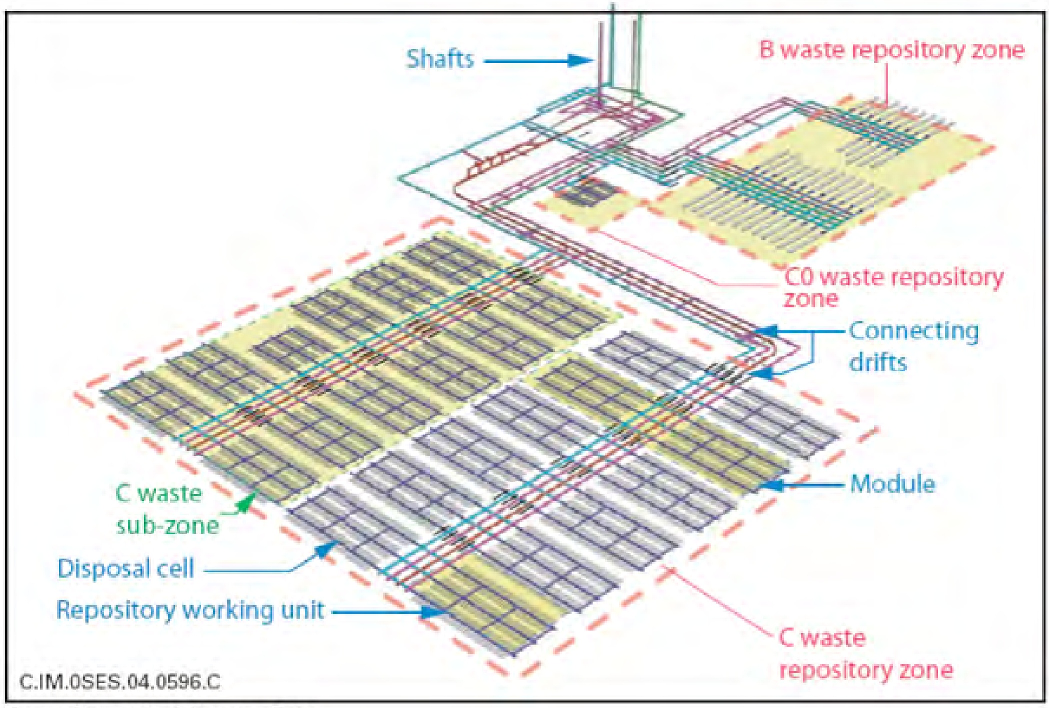
ANDRA's concept of reversibility is the possibility of "gradual and flexible operation of the repository process", and the ability to allow evolution of the repository design during the repository process. The distinction was made between reversibility and retrievability, the latter referring to the ability to remove waste packages that have already been emplaced.

6.4.2 Repository Concept

Figure 6-4 shows the layout of the proposed geologic repository for HLW and LL-ILW.

According to the disposal concept, the entire repository will be laid out on a single level in the middle of the Callovo-Oxfordien. The associated specification is a minimum (natural) 'buffer' thickness of 50 m surrounding the repository.

The repository is compartmentalized, to minimize any impacts from human intrusion. It comprises a number of *zones* for different types of waste, i.e., irradiated nuclear fuel, vitrified HLW, LL-ILW, and *modules*. Each zone is separated by a 250 m thickness of clay and each module by a 50 m thickness. The separation into zones is intended to avoid special interactions between different types of waste. The separation between adjacent disposal cells, five times the diameter of a disposal cell, is to avoid mechanical disturbances. There is a lining ('sleeve') to be placed within the repository excavations for ground support during waste emplacement and to aid in retrievability, with minimum design lifetime of 100 years.



General lay-out of the repository

Figure 6-4
Overview of repository layout for HLW, irradiated nuclear fuel and LL-ILW (ANDRA, 2005a). Used with permission of ANDRA.

NOTE: The designations B and C refer to LL-ILW and HLW, respectively, reflecting France's original categories of radioactive waste.

With regard to temperature in the repository, ANDRA's goal is a maximum temperature at the outer shell of a HLW / irradiated nuclear fuel waste package of 100°C and 90°C at the host rock. ANDRA's approach in terms of these temperature limits was to keep temperatures to a range where heat-related processes and couplings were understood. In addition, in order to minimize irreversible mineral changes, ANDRA's specification is that the temperature must not remain above 70°C for longer than a thousand years.

6.4.2.1 Seals

At closure, all disposal cells, access shafts and drifts will be closed with low-permeability seals comprising swelling clay, once the decision has been taken to close each repository component.

6.4.2.2 Disposal cells

The disposal cell proposed by ANDRA for HLW is a dead-end tunnel (0.7 m diameter, 40 m long) with a low-alloy steel sleeve as ground support. Each cell can hold 6 to 20 waste packages depending on the thermal output of the waste packages. Packages generating most heat are separated by an inert spacer (metal envelope containing a material such as silica that is chemically compatible with vitrified waste) with the heat dissipated by passive conduction within the geological formation. The disposal cell is served by an access drift (see Figure 6-5). Once the decision has been made to close the cell, the cell is sealed, also with swelling clay.

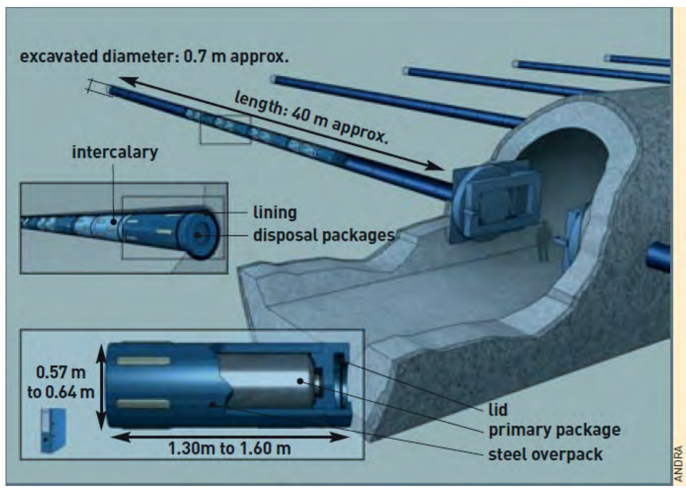


Figure 6-5
Disposal cell for HLW waste packages (ANDRA, 2005b). Used with permission of ANDRA.

6.4.2.3 Canisters

A number of waste package designs and conditioning exist, but for LL-ILW there are two standard designs depending on the type of waste: concrete and steel (Figure 6-6A). The metal standardized container is for compacted metal waste such as hulls.

There is also a standardized overpack for vitrified HLW (Figure 6-6B). The material consists of P235 steel, fitted with ceramic runner pads to facilitate emplacement. The lid, also P235 steel, is electron beam welded to the casing. The thickness of the steel, 55 mm, is chosen conservatively to withstand corrosion over long time periods – the lifetime estimated at several thousand years. Figure 6-7 shows ANDRA's engineering design for emplacement of HLW and LL-ILW.

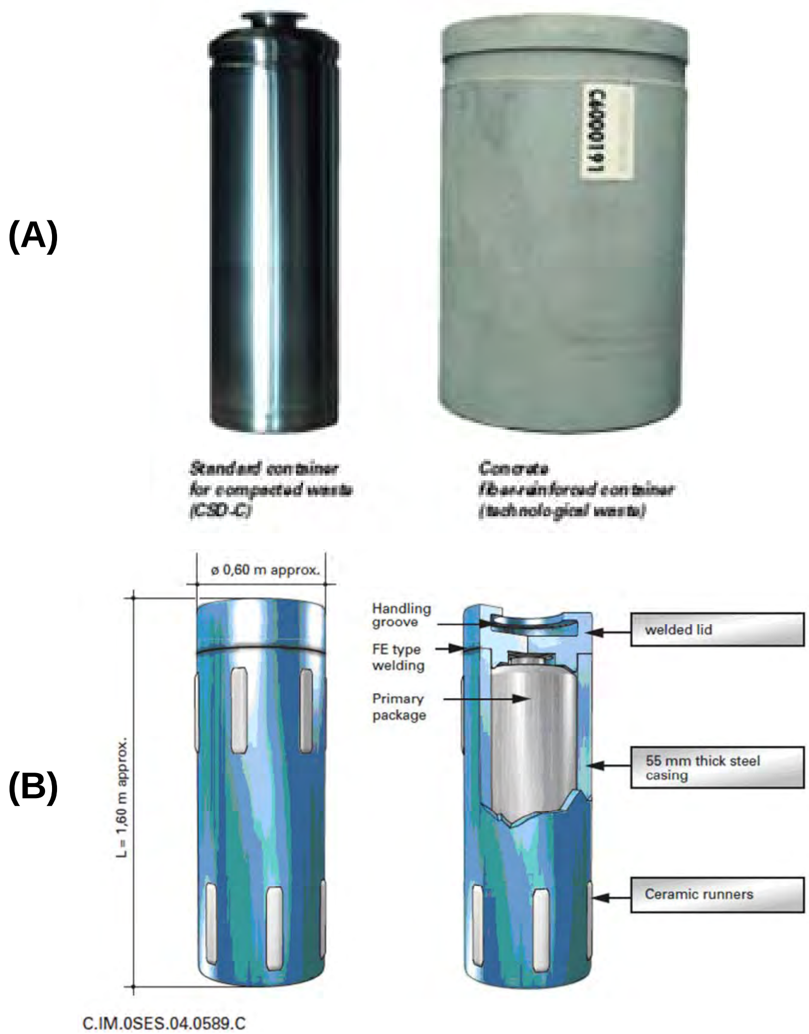


Figure 6-6
Standard radioactive waste package designs for disposal of: (A) LL-ILW and (B) HLW (ANDRA, 2005a). Used with permission of ANDRA.

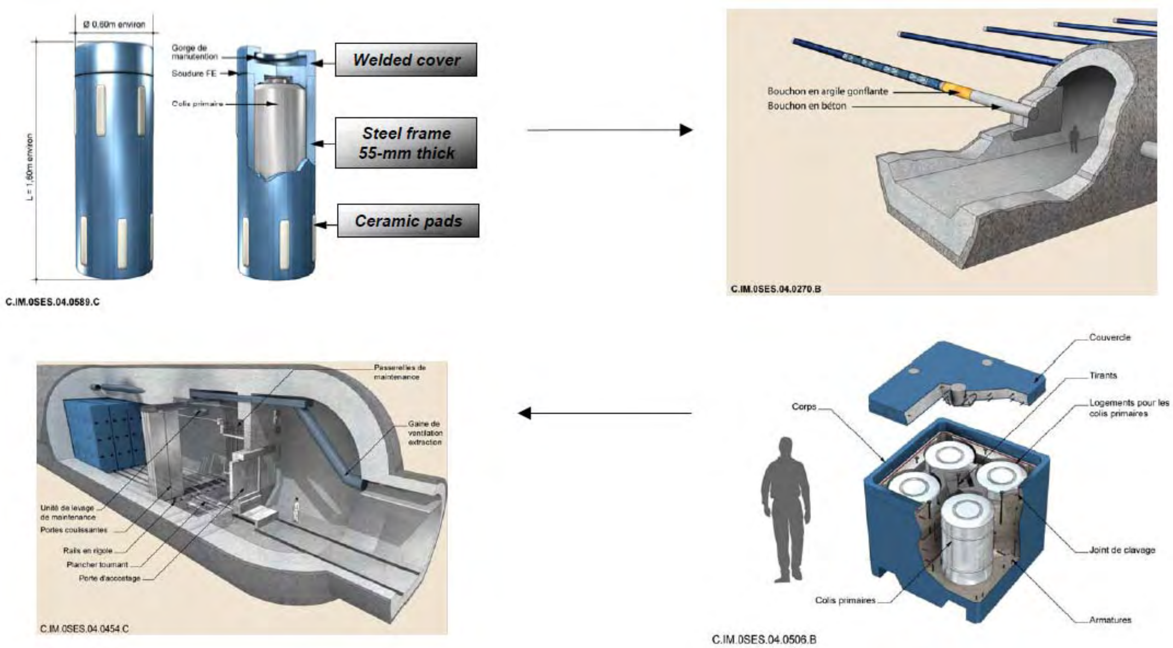


Figure 6-7
Emplacement of waste packages of HLW and LL-ILW (ANDRA, 2005a). Used with permission of ANDRA.

6.4.2.4 Engineered Barrier of Clay

The reference specification for the disposal of HLW waste packages is "without an engineered buffer barrier" (ANDRA, 2005a). This specification assumes the presence of a metal (low-alloy steel) sleeve in direct contact with the ground. While the primary concern is whether a fractured zone might develop in the natural clay around the disposal cell, observations indicate the absence of such zones when at a depth of ~500 m. Taking into account the many uncertainties, ANDRA considered it possible that a small-scale fracture zone may develop, but only a few cm.

Thus, ANDRA evaluated a disposal cell variant in which there is an engineered clay barrier, its main safety function being to provide a diffusive barrier. Such a barrier would only be considered necessary if the hydraulic properties of the clay indicated substantial damage, which is not predicted by existing knowledge. ANDRA provided a specification for this engineered barrier in terms of permeability, thermal conductivity, and swelling pressure, equivalent to a 70% swelling clay (MX80 type) and 30% sand.

6.5 Transparency and Stakeholder Involvement

6.5.1 Public Involvement

Prior to 1990, there was virtually no public consultation on the geological disposal program, in particular, in area where site investigations were taking place. As a result, there was strong opposition to what the government (CEA) was doing, which eventually halted progress and resulted in a moratorium.

Legislation allowed more opportunities for public involvement by requiring the formation of LICs (Local Information and Follow-up Committees) to facilitate dialogue among local stakeholders at each candidate site concerning a deep repository. In addition, the Environmental Code requires a public inquiry associated with an Environmental Impact Assessment.

6.5.2 International Involvement

ANDRA's URL at Bure has been open to participation by a number of organizations from different countries. One large study in particular, the European Ecoclay Project involves 17 organizations from 8 European countries. The study involves an analysis of the phenomena and consequences associated with the propagation of an alkaline plume in the clay structures (cell plug, seals) as well as the Meuse / Heute-Marne clay. The results from this study indicate a small extension to the alkaline plume (from cement structures) into the host formation, but not sufficiently to affect the key transfer properties (diffusion, permeability) of the clay or of the engineered barrier clay (swelling pressure).

With regard to external consultation on its safety evaluation, a draft version ("2001 Argile") of the 2005 assessment was provided to the National Review Board as well as to a group of NEA international experts for peer review. The findings from this peer review were generally positive, and ANDRA acknowledged that recommendations provided by both groups helped focus additional research efforts, particularly in the area of geological characterization and the seals for closing the access shafts.

6.6 Safety Assessment and Licensing

6.6.1 Safety Assessment

The 1991 version of the RFS III.2.f formed the basis for ANDRA's submission of the safety assessment Dossier "Argile 2005" (ANDRA, 2005a). The reference situation for geological disposal is an annual dose limit to an individual of 0.25 mSv/year over a post-closure period of 10,000 years. Beyond 10,000 years, 0.25 mSv/year is treated as a reference rather than a strict limit. The Rule also states that for non-reference situations, the annual dose shall be "*sufficiently limited compared to levels of deterministic effects*".

Guidance is also provided in RFS III.2.f concerning the safety assessment, which specifies that the safety evaluation must consider:

- The justification of any advantageous feature on the performance of each barrier;
- Evaluation of the effects of the disposal on the host rock and the verification of the acceptability of these effects;
- Evaluation of the future behavior of the disposal and the verification that individual doses are acceptable.
- The assessment carried out by ANDRA included both the operational period as well as post-closure.

In order to describe the temporal and spatial evolution of the proposed repository, ANDRA introduced the technique Phenomenological Analysis of Repository Situations (PARS). The method, similar to the FEPs approach, describes the evolution in terms of thermal, mechanical, hydraulic, chemical, and radiological phenomena and their couplings. The time period for ANDRA's evaluation is from the construction phase out to one million years post-closure.

ANDRA identifies different subsystems of the repository and different time periods for which the analysis is carried out, which correspond to different phases of the repository. ANDRA's assessment was intended to identify areas where there were limits to the knowledge and understanding of the phenomena, and, in this way, define the associated uncertainties.

Safety functions of the proposed repository are:

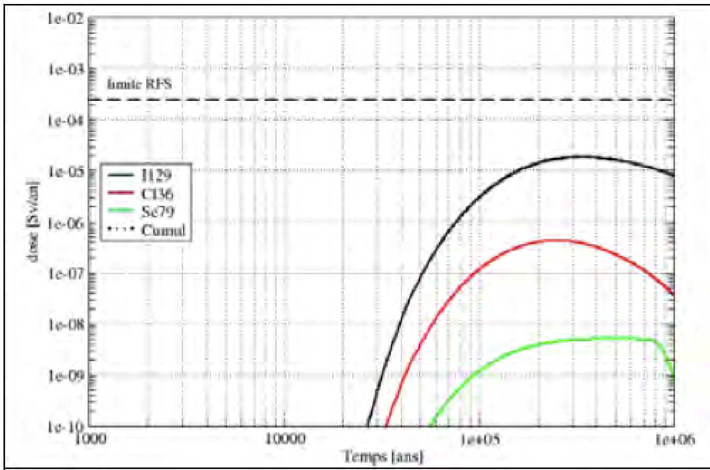
- Protection of waste from surface phenomena and human intrusion;
- Prevention of water circulation;
- Limitation of radionuclide release and immobilization in the near-field;
- Delay and reduction in migration of radionuclides released by waste.

ANDRA's biosphere corresponds to release of radionuclides to an overlying aquifer, with doses obtained from a drinking well (at Saulx outlet of the Oxfordien).

The framework under which the assessment was carried out was the Basic Safety Rule III.2.f, which specified a dose limit of 0.25 mSv/year and a time period of at least 10,000 years. ANDRA adopted a timeframe for its calculations of one million years. Given the difficulties of defining a reference group over such a long time period, ANDRA defined the reference or 'critical' group based on the conservative assumption that small groups of people live close to the repository and derive their food from local production.

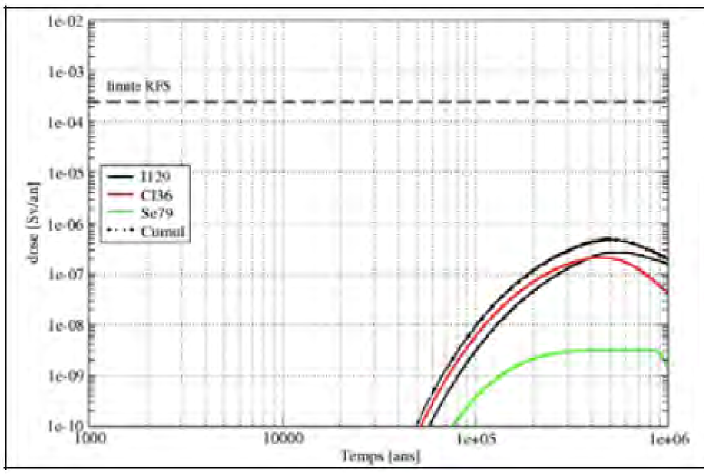
Under the normal evolution scenarios, failure times of 4,000 years and 10,000 years were applied to the carbon steel HLW overpack and used fuel canister, respectively, although a specific number of early failures were assumed to occur (1/10,000 of all HLW canisters after 100 years, and 1/10,000 of all used fuel canisters after 200 years).

Results from ANDRA's assessment of conservative inventory scenarios indicated a peak annual dose of 0.022 mSv/y associated with the disposal of used nuclear fuel (CU1 – CU3 reference packages). The results for the CU1 reference package (highest dose) are shown in Figure 6-8. Actinides had a negligible contribution to this dose, the primary nuclides being I-129 (major contributor to dose), Cl-36, and Se-79. Peak doses from the disposal of vitrified wastes (Glasses C0 – C4 reference packages) were over an order of magnitude lower, 0.0008 mSv/y, due to the lower inventory of the principal dose driver I-129. The results for the C1+C2 reference packages are shown in Figure 6-9. The peak dose in each case occurred at ~330,000 years and ~490,000 years, respectively.



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	1.9E-05	3.3E+05
I129	1.8E-05	3.4E+05
Cl36	4.3E-07	2.5E+05
Se79	5.0E-09	4.7E+05

Figure 6-8
Results for reference calculation of normal evolution scenario – doses at the Saulx outlet of the Oxfordien - for irradiated fuel (CU1 canister) (ANDRA, 2005a). Used with permission of ANDRA.



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	4.7E-07	4.9E+05
I129	2.6E-07	5.5E+05
Cl36	2.1E-07	4.5E+05
Se79	3.2E-09	8.0E+05

Figure 6-9
Results for reference calculation of normal evolution scenario – doses at the Saulx outlet of the Oxfordien - for vitrified HLW (C1+C2 canisters) (ANDRA, 2005a). Used with permission of ANDRA.

ANDRA defined three main scenarios that tested the safety functions:

- Failure of seal(s);
- Failure of waste package / overpack;
- Drilling.

Hypothetical situations that were evaluated included mining, failure of engineered barriers and climatic effect due to global warming. In addition, ANDRA considered a severely degraded scenario that combined a higher permeability of the clay host rock with defects in seals, waste packages and overpacks. This scenario resulted in a peak annual dose of 0.12 mSv/year.

The lessons learned by ANDRA from the assessment were:

- Geochemical properties of the clay limited the migration of key radionuclides except for Cl-36 and I-129;
- Seals play an important role in limiting the migration of radionuclides that are not retarded by solubility constraints, e.g. I-129, Cl-36.
- Engineered barriers play a role on the treatment of uncertainty in thermo-mechanical behavior.

6.6.1.1 Safety Evaluation - Role of Regulator

ASN published its opinion of ANDRA's assessment in 2006. Its findings, based on a review by a group of experts together with review comments by IRSN, were that a potentially repository site within the 200 km^2 area defined by ANDRA seemed reasonable and was likely to yield a site capable of demonstrating safety. In addition, ANDRA's treatment of operational safety was considered appropriate.

ANDRA had concluded that it was probably possible to maintain reversibility for 200-300 years, but ASN questioned this conclusion, stating that additional work would be necessary to support this conclusion. In any case, the Law of 2006 specifies only 100 years as the objective for reversibility.

ASN also identified areas where more information was needed prior to the license to construct a repository:

- With regard to the transposition area:
  - Additional 2D seismic studies and boreholes drilled over the 200 km2 area by the end of 2009;
  - Finalization of a number of issues, including inventory, design options, identification of a smaller area for more precise characterization and experimentation, including 3D seismic studies;
- Additional experiments on the mechanical behavior of the host rock, especially for tunnels where LL-ILW may be emplaced;

- Additional information on gas-induced effects on the engineered barrier system, in particular seals;
- Additional work on the operational phase, focusing, for example, on ventilation;
- Practical experiments to confirm the reversibility of waste package emplacement, by the removal of waste packages from a tunnel.

6.6.2 Licensing Process

Each stage in the process requires authorization by the safety regulator ASN, which in turn is enacted by government decision / decree. ANDRA is required to submit a safety evaluation in support of its application for a construction / operating license. The license to construct the URL at Bure was issued in 1999. Section 6.7 discusses the future licensing steps.

6.7 Current Status

ANDRA submitted the synthesis report on the feasibility of reversible deep geological disposal of HLW and LL-ILW to the public authorities in 2005. Following a debate in Parliament on the various options for the long-term management of radioactive waste, Parliament voted on the Planning Act. The associated decree (April 2008) establishes the general principles for radioactive waste management and prescribes new milestones for ANDRA for its research work and studies, in particular the deep geologic disposal project (referred to as the HA-MAVL project).

The 2006 Planning Act adopted ANDRA's disposal concept presented in "Argile 2005" as the reference solution for the long-term management of HLW and LL-ILW. Currently, research continues into the design and siting of the disposal facility that is planned for commissioning (operational phase) by 2025, provided its license is granted in 2015.

RFS III.2.f has been updated (since 1991) to reflect the role of reversibility in the French concept for geological disposal as well as updates in international thinking represented by new recommendations of IAEA and ICRP.

ASN authorized the continuation of activities at the Bure URL until the end of 2011. Eventually, the government needs to authorize the reversible geological disposal facility as a nuclear installation (INB), once opinions are provided by both CNE and ASN. Parliament will then establish the conditions for reversibility of geological disposal.

The current timeline for the deep repository is summarized in Figure 6-10.

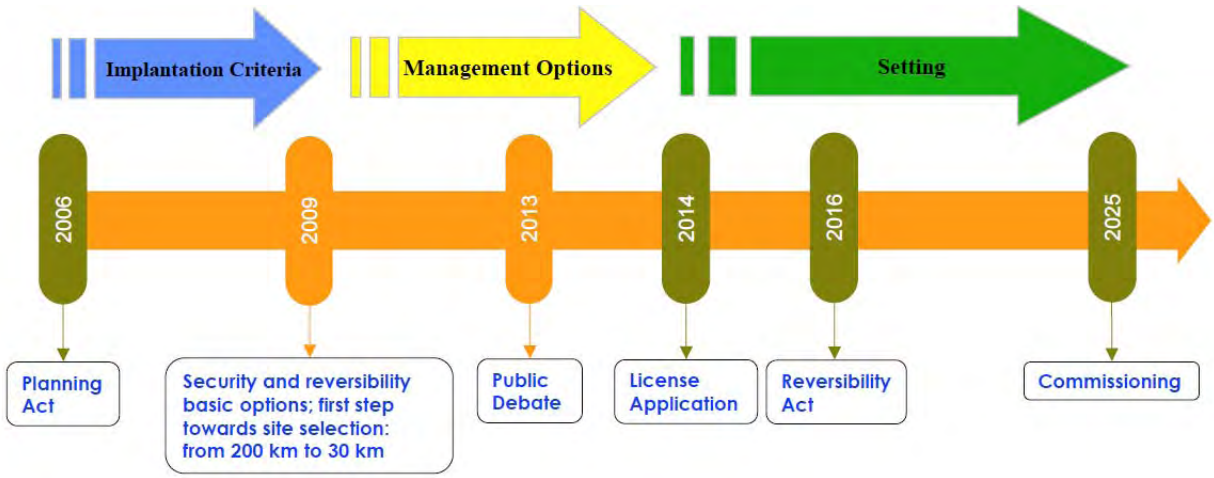


Figure 6-10
Current timeline for France's deep geologic repository project (ANDRA, 2007). Used with permission of ANDRA.

Additional milestones not shown in the above timeline are:

- 2010 / 2011: Filing of license application to pursue the operation of the Meuse / Haute-Marne URL beyond 2011.
- 2017: Start of construction of the disposal facility, subject to the regulatory authority's findings and recommendations.

The French National Assessment Board periodically reviews the country's radioactive waste management program as required by law. In its most recent report (NAB, 2009), the Board commented on the impact of partitioning-transmutation on disposal, in particular the footprint of the geologic repository due to a reduction in heat output of the waste. The Board concluded that there was insufficient information / data available for a quantitative assessment at that time and so CEA and ANDRA needed to urgently carry out a study on the impacts of transmutation on the costs and safety of a geologic repository in order to meet the required deadline of 2012.

6.8 Summary and Key Observations

- *Policy on Geologic Disposal*: France is firmly committed to the geologic disposal of HLW and submitted a feasibility report to the safety authorities in 2005. Following a favorable review, Parliament set in motion the next phase of repository development and research is continuing into the design and siting of the disposal facility. At some stage in the next few years, the government needs to authorize the planne geological disposal facility as a nuclear installation, once opinions are provided by the regulators.
- *Institutional Arrangements*: The implementer in France is ANDRA, an independent public body as of 1991, funded by government and industry, and supervised by the Ministries for Industry, Research and the Environment. Also on the implementation side and reporting to the same Ministry is CEA, originally merged with ANDRA until the separation in 1991. This organization is pursuing research on partitioning and transmutation. The responsible regulatory authority, ASN, authorizes the different stages of construction and operation of the

URL as well as the eventual repository. ASN reports to the Ministries of Industry, Environment and Health. Funding for geologic disposal programs is provided principally by the reactor operators. Each principal nuclear operator in France (i.e., EdF, AREVA, and CEA) administers its own separate, internal fund - subject to conditions set forth in law to ensure adequacy of the assets set aside.

- *Key Laws and Regulations*: The 1991 Law was a key law, changing the direction of radioactive waste management in France and setting in motion a 15-year research period into the feasibility of deep geologic disposal, with a stepwise approach to nuclear waste management. Following ANDRA's successful completion of this 15-year period, Parliament passed the 2006 Planning Act which allowed ANDRA to continue its research using its URL and pursue a license application to start construction of the disposal facility, subject to the regulatory authority's findings and recommendations.
- *Site Screening and Selection*: Early studies identified two types of generic rock, granite and clay, as being suitable for geologic disposal. Subsequent geological investigations towards identifying a site for geologic disposal due to intense socio-political opposition. The 1991 Law required ANDRA to identify at least two sites that would be suitable for geologic disposal, one in clay and one in granite. From ANDRA's initial investigations, four sites were originally proposed for more detailed investigations. Of these, one was rejected on hydrogeological grounds and two clay sites in the East Paris basin were combined, leading to detailed geological investigations on two sites, one in clay (Bure, east Paris Basin) and one in granite (Vienne, western France). Eventually, based on a National Commission's review of ANDRA's submissions for each site, the government approved continued research at the clay site but rejected the granite site based on technical concerns of the review commission. The Bure site consists of a series of almost horizontal layers, with relatively simple geology, low tectonic activity, and low permeability. Homogeneity of the clay appears to be relatively high with few discontinuities. Geochemical conditions are reducing, with pH buffered by carbonate minerals.
- *Repository Design Concepts*: A Safety Rule issued to complement the 1991 Law specifies a multi-barrier concept. The principle of reversibility is a principal requirement to be accomodated by the French repository concept. The disposal concept calls for the entire repository to be laid out on a single level in the middle of the clay formation (Callovo-Oxfordien). The disposal concept also calls for the co-disposal of HLW, a small quantity of irradiated nuclear fuel, and long-lived ILW, with separate *zones* for different types of waste, each zone separated by a substantial thickness (250 m) of clay. Tunnels will have a low-alloy steel lining during waste emplacement, for stability and to aid retrievability. The overpack for HLW consists of 55-mm thick P235 steel, with a lid of similar material electron beam welded to the casing. The overpacks are fitted with ceramic runner pads to facilitate emplacement. ANDRA's disposal concept for HLW has no engineered clay barrier but relies on the hydraulic properties of the natural clay.
- *Performance Metrics and Assessments*: The reference situation for geological disposal is provided in the 1991 Safety Rule that complemented the 1991 Law. The Safety Rule provides an annual dose limit to an individual of 0.25 mSv/year over a post-closure period of 10,000 years. Beyond 10,000 years, 0.25 mSv/year is treated as a reference rather than a strict limit. The Rule also provides guidance on the assessment process itself, ANDRA's assessment (Argile 2005) of the Reference Scenario yielded a peak annual dose of 0.022 mSv/y, due primarily to radionuclide releases from irradiated nuclear fuel. ANDRA also

tested barrier functions with specific scenarios and also considered a relatively extreme, severely degraded scenario that combined a higher permeability of the clay host rock with defects in seals, waste packages and overpacks. This scenario resulted in a peak annual dose of 0.12 mSv/year.

- *Independent Peer-Review and Advisory Bodies*: France has a National Assessment Commission (CNE) with responsibilities for reviewing ANDRA's work and making recommendations to the government. A draft version ("2001 Argile") of the 2005 assessment was provided to this Commission as well as to a group of NEA international experts for peer review. The regulator, ASN, is supported by ISRN (Institute of Radiation Protection and Nuclear Safety) as well as an advisory group of experts.
- *Stakeholder and Public Involvement*: There was minimal public involvement in site investigation / selection activities prior to the 1991 Law. Thereafter, legislation allowed more opportunities for public involvement by requiring the formation of Local Information and Follow-up Committees to promote dialogue among local stakeholders at each candidate site concerning a deep repository. In addition, France's Environmental Code requires a public inquiry associated with an Environmental Impact Assessment.
- *Program Maturity*: Following a failed attempt at repository siting in the late 1980's, the French program has been reinvorgated and is proceeding in a stage-wise manner, with careful review by a National Commission consisting of respected scientists and academics. A site has been selected, and specific siting of repository facilities is underway with public input. Subject to the favorable outcome of a public debate, ANDRA is close to submitting a license application for the construction phase of a geologic repository.

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7
GERMANY

7.1 Introduction

7.1.1 General Nuclear Profile

In 2009, nuclear generation provided 128 TWh of electricity or approximately 26% of Germany's total electricity production (WNA, 2010a). Currently, 17 commercial power reactors, either PWRs or BWRs, are operating, while 19 reactors are in different stages of decommissioning. After the re-unification of Germany in1990, all the Soviet-designed reactors (PWR-VVER or -WWER) in former East Germany were shut down for safety reasons (WNA, 2010b). Nuclear fuel consists of either enriched UO_2 or MOX fuel containing recycled plutonium from the reprocessing of German irradiated nuclear fuel in Britain or France.

Following the phase-out policy of the German government as stipulated in the amended Atomic Energy Act (2002), a limit was defined for the amount of electricity to be generated by nuclear power. Coordinated phase out of operating nuclear power plants was based on an average total operating life of 32 years. As a consequence, the overall volume of radioactive waste generated in Germany is now reduced from previous estimates. However, following concerns about meeting CO_2 emission targets, Germany's energy policy is under review and utilities are preparing for a general extension of the operational life of reactors to at least 40 years, and individual extensions to 60 years[1].

The amended Atomic Energy Act also brought an end to the transportation of irradiated nuclear fuel to reprocessing facilities. As a result, a number of on-site interim storage facilities for dry storage of irradiated nuclear fuel have been created at each commercial nuclear plant, using transport and storage casks.

Previous plans (in the 1970's) for siting of centralized waste management center (WMC) at Gorleben, including storage, conditioning, reprocessing, fuel fabrication, and disposal, did not come to fruition. As a result of this failure, storage of HLW and used nuclear fuel occurs at reactor sites and at several other away-from-reactor locations:

- Dry interim storage at 12 nuclear power plant sites, following an initial period of wet storage, typically 5 years to allow for heat decay. Following wet storage, irradiated nuclear fuel assemblies are placed in storage or transport casks that give the operator some flexibility in on-site management;
- Two centralized facilities at Gorleben and Ahaus; and

- Central interim storage facilities at Greifswald (ZLN) for irradiated nuclear fuel assemblies from Russian-type reactors and Jülich for irradiated nuclear fuel pebbles) from the center's advanced high-temperature pebble-bed reactor.

In addition to the above, there is a pool for interim storage of irradiated nuclear fuel assemblies at the now-closed reactor site (Obrigheim), until a dry storage facility is constructed at the same site.

7.1.2 Used Fuel / HLW Inventory and Projected Disposal Requirements

Heat-generating waste, predominantly irradiate nuclear fuel elements from reactors and vitrified waste from reprocessing, accounts for ~10% of the total predicted waste volumes - approximately 22,000 m^3 by 2040 (BfS, 2007).

The Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU) in its Third National Joint Convention Report describes the planning and funding associated with the management and disposal of different types of radioactive waste (BMU, 2008). Extracts from this table relevant to HLW and LL-ILW are shown in Table 7-1.

Table 7-1
Waste management strategy for heat-generating radioactive waste (adapted from BMU, 2008).

Waste management task	Long-term strategy	Financing	Current practise / installations	Planned installations
Spent fuel assemblies	Interim storage in casks; subsequently conditioning and direct disposal in deep geologic formations	Annual refunding, as adequate for the waste producer, of the costs incurred by the Federation for the planning and construction of repositories according to fixed distribution key (polluter-pays principle)	4 central dry storage facilities; 12 dry storage facilities at nuclear power pant sites; 1 wet storage facility (Obrigheim)	1 on-site interim storage facility (Obrigheim); 1 repository (site yet undecided)
Radioactive waste from the nuclear fuel cycle	Interim storage at the site of origin or centrally with the aim of disposal in deep geologic formations	As above	Conditioning and interim storage (at the site of origin or centrally)	One repository licensed; refitting under preparation; commissioning approximately 2013
Other radioactive waste	Interim storage at central sites with the aim of disposal in deep geologic formations	Waste producers pay fees to the Land collecting faciltiies (polluter-pays principle); Land collecting facilities pay repository cost portion to the Federation	Conditioning and interim storage (*Land* collecting facilities)	One repository licensed; refitting under preparation; commissioning approximately 2013

Table 7-2 provides data on the current inventory of irradiated nuclear fuel in Germany, equivalent to a total of 12,505 Mg HM. In addition to the current inventory, the utilities estimate that another 4,870 Mg HM will arise from irradiated nuclear fuel assemblies over the remaining years of operation of the current reactors, yielding an overall predicted total of 17,200 Mg HM of which 10,500 Mg HM requires conditioning for disposal. The balance has been reprocessed, generating HLW. The major portion of conditioned heat-generating waste is derived from reprocessing and comprises hulls and end pieces.

Table 7-2
Total quantities (rounded) of irradiated fuel from German light-water reactors (LWRs) up to December 31, 2007 (adapted from BMU, 2008).

Storage location	Quantity of used fuel Mg HM
Used LWR fuel assemblies in used fuel pools at NPPs, including Obrigheim	3,541
Dry storage of used Soviet WWER fuel assemblies in casks at ZLN facility	583
Dry storage in casks at NPP sites	1,614
Dry storage at the Ahaus and Gorleben interim storage facilities	92
Shipped to La Hague (France) for reprocessing	5,393
Shipped to Sellafeild (UK) for reprocessing	851
Reprocessed at the WAK reprocessing plant in Karlsruhe	90
Reprocessed at the former EUROCHEMIC reprocessing plant (Belgium)	14
Returned to the former USSR (WWER fuel assembies)	283
Shipped permanently to Sweden (CLAB)	17
Reuse of WWER fuel assemblies at Paks (Hungary)	27
TOTAL	12,505

Vitrified HLW from reprocessing is currently stored in 2,100 stainless steel canisters in 75 overpacks, all but one being of the CASTOR-type cask.

The projected cumulative volume of heat-generating waste to the year 2080 has been estimated at 28,000 m^3 comprising, in addition to a small volume of packaged assemblies from experimental / research reactors (DBE, 2008):

- ~20,800 m^3 of packaged irradiated nuclear fuel assemblies from commercial power plants;
- ~780 m^3 of vitrified HLW from France and the UK;
- ~950 m^3 structural components from irradiated nuclear fuel assemblies, generated during reprocessing;
- ~1,970 m^3 packaged fuel assemblies from non-light water reactors;
- ~3,400 m^3 waste packages containing structural components from fuel assemblies from the Gorleben pilot conditioning plant.

7.2 Institutional Arrangements

7.2.1 Institutional Framework

Figure 7-1 shows the principal entities associated with radioactive waste management in Germany. There are additional government organizations involved in the execution of the German waste management program not shown in the summary diagram. The primary roles are described below.

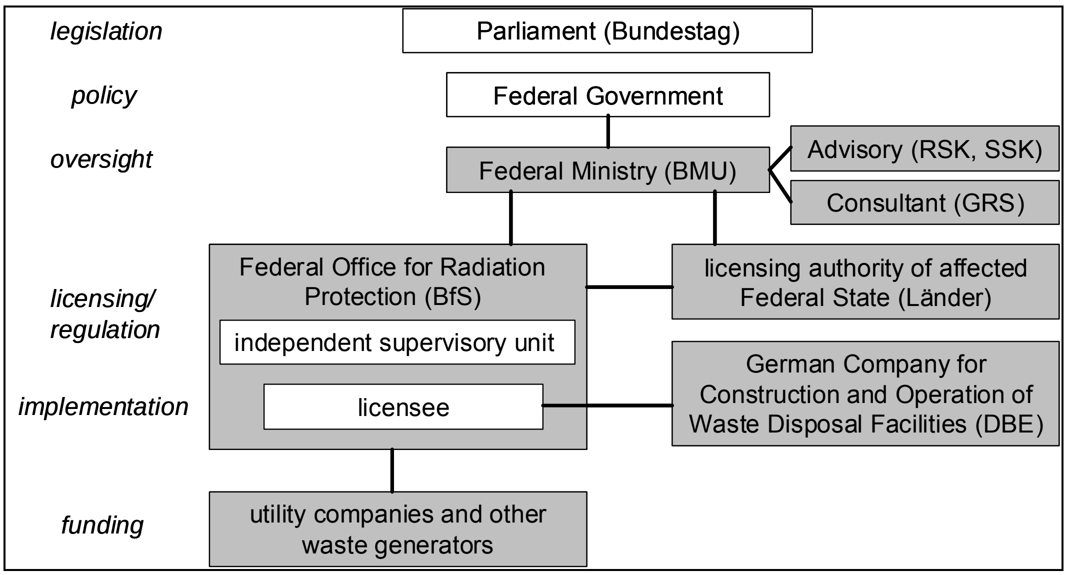

Figure 7-1
Key entities associated with radioactive waste management in Germany (adapted from NEA, 2005).

POLICY and OVERSIGHT - Apart from the federal government and parliament, the *Federal Ministry of the Environment, Nature Conservation and Nuclear Safety* (BMU) is responsible for policy decisions.

IMPLEMENTER - The *Office for Radiation Protection* (BfS), which reports to BMU, provides support in the technical and scientific research associated with the management of radioactive waste including waste disposal. BfS is responsible for the planning, construction and operation of radioactive waste repositories as well as their supervision, and, for these purposes, makes use of external contractors – in particular, the *Company for the Construction and Operation of Waste Repositories* (DBE). DBE is assigned to plan, design, construct, and operate the repository.

REGULATOR - BMU is responsible for nuclear safety and radiation protection and has the oversight to enforce regulations issued by the regulatory authorities in the States. BMU is supported by BfS in the technical and scientific aspects of radiation protection. The affected German Federal States or Länder have legislative authority except where specified in the Basic Law. After federalism reform in 2006, the federal government was given exclusive powers in the regulation of nuclear power, so that the Länder only have legislative authority if explicitly authorized in a federal act.

The federal and Länder (State) governments share responsibility for licensing the construction and operation of all nuclear facilities, with the Länder having close to veto power for both.

ADVISORY and SUPPORT - At the ministerial level, BMU is supported by two other ministries:

- *Ministry of Economics and Labor* (BMWA), responsible for R&D in nuclear waste disposal;
- *Ministry of Economics and Technology* (BMWi), responsible for the nuclear energy industry and repository-related applied basic research. BMWi recently provided an overview of the findings on the Gorleben salt dome (BMWi, 2008). The Institute for Geosciences and Natural Resources (BGR) supports BMWi, dealing primarily with basic geoscientific issues concerning the final disposal of radioactive waste.

In addition, BMU set up a special expert group, AkEnd, to develop repository site selection criteria and a site selection procedure. Other independent advisory bodies to BMU include the Reactor Safety Commission (RSK) and the Commission on Radiological Protection (SSK), both of which make recommendations that feed directly into the licensing process. Of particular note are RSK's guidelines "Safety Guidelines for Dry Interim Storage of Irradiated Fuel Assemblies in Storage Casks".

Finally, the Nuclear Safety Standards Commission (KTA) formulates regulations with detailed technical specifications, and the Company for Plant and Reactor Safety (GRS) also assists the regulatory authority.

7.2.2 Legal and Regulatory Framework

Figure 7-2 shows the hierarchy of regulations (acts, ordinances, and safety rules) and guidelines relating to the nuclear industry in Germany. The diagram also shows the relevant authority and to what extent the different levels of regulations are binding.

At the top level, the Basic Law describes the responsibilities of different authorities within the federal government, including the independence between licensing and supervisory authorities.

The main laws governing radioactive waste disposal in Germany are:

- Atomgesetz – Atomic Energy Act (AtG);
- Strahlenschutzvorsorgegesetz (StrVG) - Precautionary Radiation Protection Act;
- Strahlenschutzverordnung (StrlSchV) - Radiation Protection Ordinance;
- Bundesberggesetz (BBergG) - Federal Mining Act;
- Sicherheitskriterien für die Endlagerung radioaktiver Abfälle in einem Bergwerk - Safety Criteria for the Disposal of Radioactive Wastes in a Mine.

These laws address all aspects of radioactive waste management and disposal, including the licensing of storage and disposal facilities.

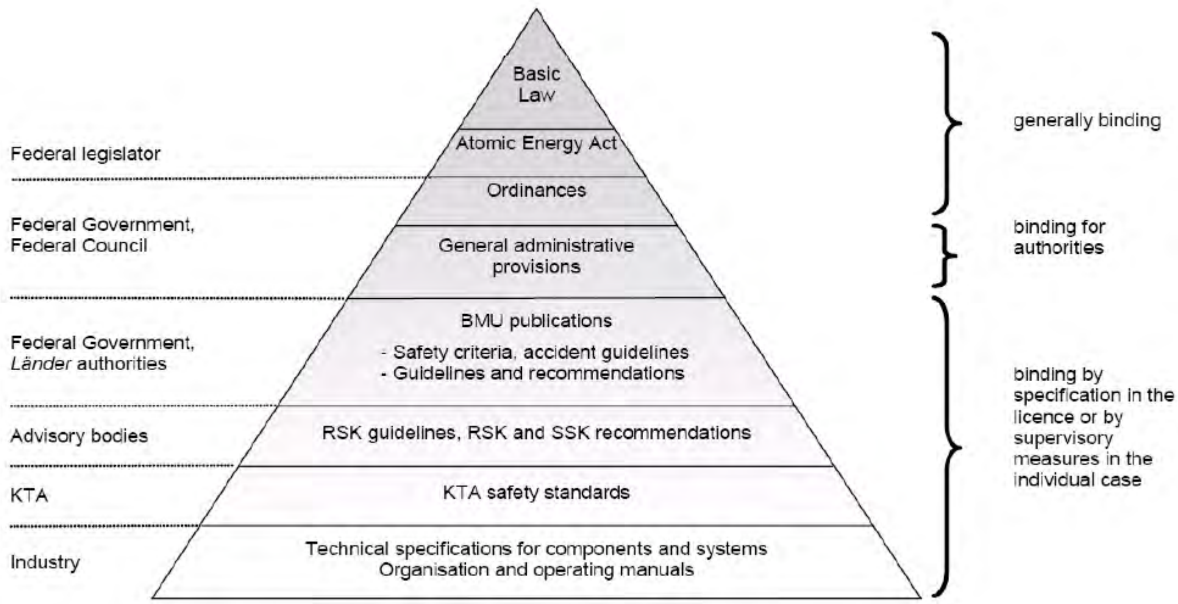

Figure 7-2
Hierarchy of German regulations (BMU, 2008). Used with permission of the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety.

Note: KTA = Safety Standards Commission.

The AtG also includes the general national regulations for protection and precautionary measures, protection and the management of radioactive waste, and forms the basis for supporting Ordinances. The amended Act of 2002 also stipulates the coordinated phase-out of commercial nuclear energy.

Originally, the AtG called for the reuse of fissile material in irradiated nuclear fuel via reprocessing and recycling. In 1994, amendment of the AtG gave nuclear power plant operators the option of reprocessing or direct disposal of irradiated nuclear fuel. A subsequent amendment in 2002 enacted a full prohibition on reprocessing of irradiated nuclear fuel effective July 2005, which restricted the long-term management of irradiated nuclear fuel to direct disposal thereafter. All shipments of irradiated nuclear fuel to France and UK for MOX production ceased in June 2005.

The Radiological Protection Ordinance regulates the protection of humans and the environment against radioactivity, including associated naturally occurring radioactive materials (NORM). The Ordinance establishes an annual dose limit for members of the public of 1 mSv/year from direct radiation and discharges from the operation of nuclear facilities.

Legally, all steps in the treatment of radioactive waste up to disposal are funded according to the polluter-pays principle, while disposal is the responsibility of the Federal Government. This means that waste producers are responsible for the safe management of all aspects of the treatment of radioactive waste, in particular the transportation and reprocessing of irradiated nuclear fuel prior to July 2005, as well as interim storage of irradiated nuclear fuel on nuclear power plant sites. Thereafter, according to the AtG, the federal government takes responsibility for establishing radioactive waste repositories.

The AtG includes two mechanisms for repository funding. The *Ordinance on Advanced Payments* stipulates how waste producers pay for site characterization and repository construction. In addition, the AtG allows for disposal fees that cover future repository operational costs to waste producers in the form of a fee charged at the time of waste delivery. The utilities make provisions for future disposal costs from present-day electricity revenues. These provisions are free of corporate taxation (Lempert and Biurrun, 1999).

The German philosophy for radioactive waste disposal is that all radioactive waste is to be disposed of in deep geological formations or mines. For example, Lower Saxony's regional mining, energy and geology authority recently approved (January 2008) the operational plan for the construction of the Konrad repository for non-heat-producing radioactive waste. In this context, the Federal Mining Act is important in the licensing procedure for geologic repositories (as discussed in Section 7.5.2).

Local and centralized interim storage facilities are available for non-heat-generating waste. Current licensing conditions also allow heat-generating waste to be stored in local and centralized interim storage facilities. The latter are used for HLW returned from reprocessing.

New safety requirements for the geologic disposal of heat-generating radioactive waste were issues in 2009, replacing the previous 1983 Safety Criteria (BMU, 2009). These requirements cover the planning, site exploration, construction, operation and closure of a final repository. Site selection and licensing are not addressed in this document.

The reference time period, for which evidence of safety (safety case) must be provided, is one million years. With regard to the optimization of safety, the Requirements state (BMU, 2009):

> *"A robust barrier system in which the safety functions of the final repository system and its barriers are insensitive to internal and external influences and disturbances, the behavior of the isolating rock zone is very predictable, and the results of the safety analysis are insensitive to deviations from the underlying assumptions, is pivotal to the reliability of safe, long-term containment."*

In the absence of criteria in the Radiological Protection Ordinance for assessing the safety of future generations, the 2009 Safety Requirements provide the necessary quantitative criteria in the form of risk:

- For all 'likely' future evolutions of the repository system (where 'likely' implies cases with a likelihood of occurrence exceeding 10%), the total risk should be $< 10^{-4}$ with regard to individuals with a 'lifetime' exposure (lifetime = current expected life expectancy).
- For all less-likely events, the additional risk to affected individuals shall be $<10^{-3}$, i.e. for such events a greater risk of radionuclide release and individual exposure is allowed because the events themselves are less likely. Where supporting quantitative data on probability exist, the likelihood attached to such events is >1% (but <10%).
- For improbable events for which the likelihood is <1%, no risk limit is specified, However, in the context of optimization, some effort should be made to see if the resultant risk can be reduced "with a reasonable input".

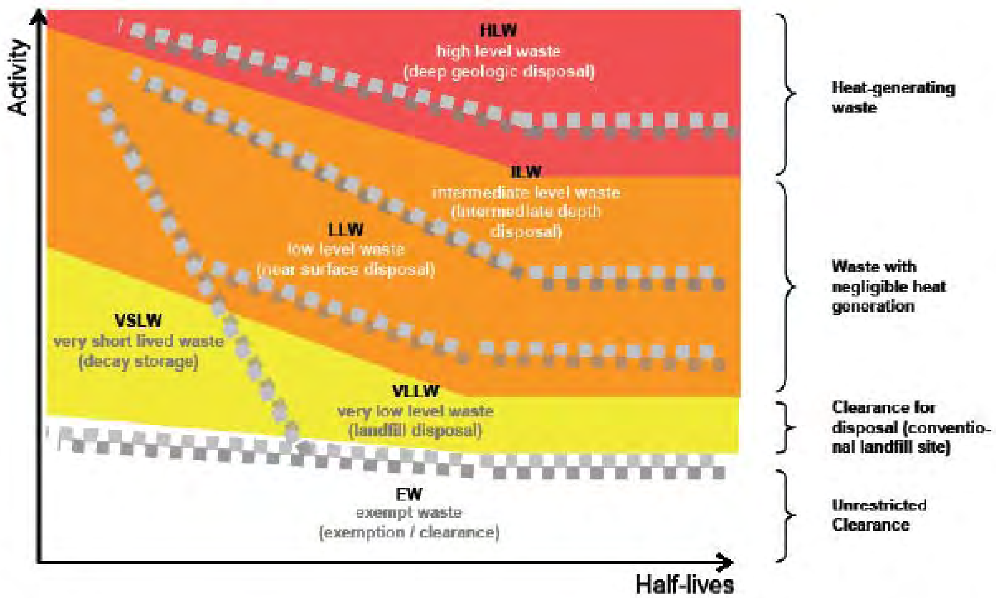

Figure 7-3
Comparison of German and IAEA waste classification systems (BMU, 2008). Used with permission of the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety.

7.2.3 Waste Classification

The definition of radioactive waste in Germany covers materials considered to be unsuitable for reuse and which must therefore by disposed of in a controlled way, according to the AtG and the Radiation Protection Ordinance.

The management policy for all types of radioactive waste has always been directed towards disposal in deep geologic formations. Consequently, the IAEA distinction between short-lived and long-lived radioactive waste that normally identifies the disposal route (near-surface or deep geological repository, respectively) is not applicable in Germany.

With regard to categories of waste, the authorities have moved away from previous classifications of: low-activity waste (LAW), medium-activity waste (MAW), and high-activity waste (HAW). Instead, a more basic subdivision has been adopted, i.e.:

- Heat-generating waste; and
- Waste with negligible heat generation.

The latter is quantified in the context of the Konrad repository planning so that “the increase in temperature at the wall of the disposal chamber caused by decay heat from the radionuclides in the waste package must not exceed 3 Kelvin on average.” BMU (2008) notes that such a temperature increase is roughly equivalent to the geothermal gradient associated with an increase in depth of ~100 m. To place the German waste categories in some context, Figure 7-3 compares the German classification system with that of IAEA.

7.2.4 Funding

The costs of conditioning, storage, and disposal of HLW and used fuel from nuclear power generation are the responsibility of the waste producers – primarily the electric utilities. Utilities set aside and manage the funds internally to cover HLW management costs according to established legislative guidelines.

7.3 Geological Studies for Deep Disposal

7.3.1 Early Studies

The Federal Ministry of (BMWi) has given a detailed account of the background to the selection of Gorleben for a repository for heat-emitting radioactive waste. A few highlights are presented below.

Between 1964 and 1976, the Federal Government funded a search for salt structures as suitable candidates for a repository. The sites identified in these selection processes were chosen on the basis of the experience and results of numerous scientific field studies, with pre-defined criteria for selection and exclusion, carried out from the mid-1960s to the early 1970s.

Recommendations and research carried out by the American National Academy of Sciences, the then German Soil Research Institute and Nuclear Energy Commission led to early support for the disposal of radioactive waste in deep salt deposits (Martini 1963). The original (1964) selection criteria, primarily geological attributes, implemented by the German Soil Research Institute were (BMWi, 2008):

- Salt formation 400-500 m thick, “reasonably” homogeneous and with sufficient lateral distribution;
- Surface of salt body at a depth of 300-800 m;
- Low permeability of overlying formation(s);
- Nearby formation(s) for brine disposal;
- Good transport infrastructure;
- Proximity to nuclear power plant sites.

At that time (early- to mid-1960’s), based on the above criteria, the Soil Research Institute identified seven salt domes as being potentially suitable and recommended these domes for detailed study. Interestingly, Gorleben was not included in this original screening, because it was not located in the vicinity of an existing power plant site and limited information was available at the time.

In 1973, the then Ministry for Research and Technology (BMFT) commissioned the nuclear fuel reprocessing company KEWA to identify sites for a centralized waste management center (WMC), where reprocessing, fuel fabrication, conditioning and final disposal activities would occur at one site. This search took place from 1973 to 1976 and focused on identifying a site that would provide for environmentally-sound operation of surface facilities for the reprocessing and conditioning of radioactive waste. The repository was treated largely as an auxiliary element at the time, although the site criteria that were developed were tailored to siting of a radioactive waste disposal facility.

Of 140 salt domes selected initially for evaluation, 23 were selected for further study. Further refinement in selection criteria reduced the number of sites to 13 and, eventually 4 sites, after applying criteria in three separate areas: (i) safety and the environment; (ii) infrastructure (transportation); (ii) structural policy.

As a result of site investigations involving these four candidate salt sites, Gorleben was selected by Lower Saxony as a suitable site in 1977.

The project initially enjoyed support at all political levels and was received positively with regard to job creation in what was an economically less-developed border area. However, disputes and protests directed primarily at the reprocessing plant eventually eroded support over time.

Public involvement in the licensing process began only after the site selection was completed, corresponding to the “DAD” approach (decide, announce, defend). A public symposium (“Gorleben Hearing”) was held in 1979, attended by a number of international participants. As a result of this meeting, the responsible Federal Minister recommended against building the reprocessing plant but did encourage all parties to remain committed to the Gorleben repository project.

7.3.2 Detailed Geological Studies

Detailed investigations of the Gorleben site began in 1979. The exploration program consisted primarily of stratigraphic investigations, hydrogeological investigations, geophysical measurements, seismological surveys, boreholes to the salt leaching surface, deep boreholes into the salt dome, and exploration boreholes for the shafts. After the reunification of east and west Germany, investigations were expanded to an area of the former east Germany north of the Elbe River. Surface investigations were completed in 1997.

The following surface investigations were carried out between 1979 and 1985 to investigate the geology and hydrogeology of the Gorleben site (Langer and Röthemeyer, 1996):

- 4 boreholes, each about 2,000 m deep to investigate the salt dome itself;
- 44 boreholes to investigate the cap rock and the underlying salt beds;
- 2 preliminary boreholes for the shafts Gorleben 1 and Gorleben 2;
- 156 km seismic profiles;
- 145 boreholes to investigate the Cenozoic cover;
- 326 boreholes for the installation of piezometers;
- 4 long-term pumping tests (~3 weeks per test);
- 1 borehole to investigate the Palaeogene (pre-Quaternary) formations in the syncline.

Results from these investigations led to the following conclusions (Langer and Röthemeyer, 1996):

- An isolation potential of repositories in rock salt of millions of years
- Analysis of fluid inclusions provide an indication of the depth to which glacially-influenced erosion processes affected the salt dome, and implied an isolation period of 250 million years at depths of 860 m to 1360 m
- Average rate of uplift of salt in the top part of the diapir of < ~0.02 mm per year for the past 25 million years, with a present-day uplift of ~0.01 mm per year.

Hydrogeological investigations were carried out over an area of ~300 km^2 around the salt dome in order to study the aquifer system in the sediments above the dome. On scales of tens of meters to kilometers, the site is characterized by a predominance of aquifers separated by aquitards. The salt dome is crossed by a sub-glacial erosion channel in which the lowermost aquifer is partly in contact with the cap rock or the salt itself. The density of the groundwater was found to vary considerably, generally increasing with depth up to the density of saturated brine at conservable depth. Changes in groundwater density influence groundwater movement significantly and, therefore, radionuclide transport (Lüdwig *et al.*, 2001).

The underground exploration, supervised by BfS and BGR, started in 1986 with the sinking of two shafts, followed by the construction of the underground infrastructural area.

In addition to the geological exploration, *in situ* experiments have been performed for the geotechnical characterization of the potential host rock. Results from the *in situ* and laboratory tests provided a database to identify potential areas for disposal. The overall results of the geological and geotechnical investigations indicated a large volume of potentially suitable host rock in a simple anticline structure of the salt dome. To date, there is no indication that the Gorleben Salt Dome is unsuitable for a repository (Bornemann *et al.*, 2008).

Again for political reasons, exploration of the Gorleben salt dome stopped in 2000 for at least three and at most ten years, in order to clarify "*conceptual and safety questions*". Thirteen issues were identified, including the build-up in pressure from gas generation from corrosion, and a comparison of salt to clay and granite. BfS were supposed to respond by the end of 2004 but after a year's delay, BfS summarized the results of both the research projects on the open issues and discussions with repository experts (BfS, 2005):

> *"The primary result of the investigation is an illustration of both the possibilities and the limitations of a generic comparison of host rocks, and answers to the twelve open issues. According to the results, no host rock exists that always guarantees maximum repository safety. Tailored and specially engineered repository concepts can be developed for all the relevant host rocks in Germany."*

and also:

> *"it is only possible to compare various options if specific sites and repository concepts are compared. This results in the need for a site comparison".*

Thus, BfS called for the focus on site-specific analyses in the comparison of different sites and the planning of a repository in an iterative, parallel way (BfS, 2005). BGR emphasized that "*the results indicated in the individual expert reports do not contradict the positive appraisal of the geological findings at the Gorleben site*" (BGR, 2007).

No specific URL was constructed in the process for identifying a specific site for a HLW repository, although the Asse Mine was used for extensive underground studies associated with L/ILW disposal. The underground explorations at Gorleben involving *in situ* tests and experiments served a similar purpose to a URL. More recently, German participation in URL programs in other countries has increased as a result of extending consideration of the types of rock for repository host formation beyond salt (see Section 7.5.2).

7.3.3 Gorleben

Figure 7-4 shows a cross-section of the Gorleben repository (Vogel and Schelkes, 1996).

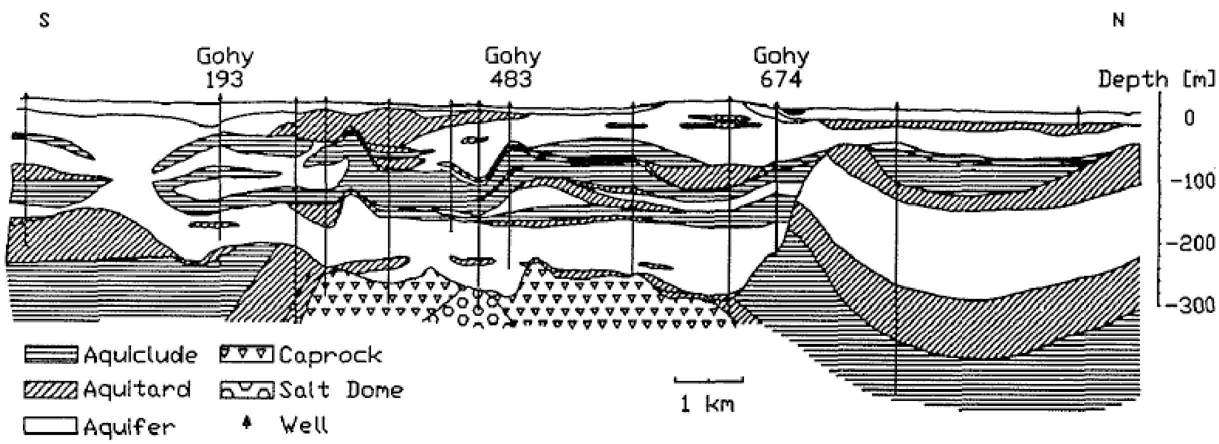

Figure 7-4
Cross-section along Gorleben erosion channel showing key hydrogeological features (Vogel and Schelkes, 1996). Copyright International Association of Hydrological Science, 1996. Used with permission.

The results of the detailed *in situ* geological and hydrogeological investigations led to a number of conclusions:

- The central section of the salt dome primarily consists of pure rock salt layers, which are largely solution-free and are particularly well suited for final disposal.
- In one of the exploration areas (EB1) within the Gorleben salt dome, the rock salt layers identified at the depth intended for the final disposal of radioactive waste extend further than originally predicted on the basis of surface exploration.
- An impermeable salt barrier, approximately 600 m thick, extends across the planned emplacement area to the overburden.
- The assumption that rock salt can act as a barrier has been affirmed.

- No hydraulic connections or flowpaths were detected between the rocks in the planned emplacement area and the aquifers in the overburden.
- The creep rates (convergence rates) measured in the salt dome mean that compact inclusion of the radioactive waste can be expected.

7.3.4 Alternative Geologies

BMU appointed a group of experts to develop a comprehensive procedure for the selection of sites that are both suitable for safe disposal *and* acceptable to the public. The procedure was based on well-founded criteria, with a clear and transparent structure. In contrast to previous site-selection procedures, public participation was considered an indispensable part from the beginning (AKEND, 2000).

The AkEnd committee charged with developing a selection procedure for disposal sites started work in 1999. This committee reached some general conclusions, including:

- The 'Concentrate and Contain' principle, rather than the 'Disperse and Dilute' principle, is the basis for an acceptable repository concept.
- Only disposal in deep geological formations, i.e., at least several hundred meters below ground, is to be considered.
- The disposal facility shall be constructed as a mine, in accordance with the state-of-the-art engineering techniques, with cost considerations to be taken into account.
- The isolation timeframe should be on the order of one million years.
- A robust multi-barrier system in a favorable geological setting determines the relative importance of geological, geotechnical, and technical barriers.

Subsequently, in 2003, the BMWi commissioned BGR to conduct a supplementary study on the distribution of argillaceous rock in Germany (Hoth *et al.* 2007) and summarize the results of existing studies in a survey map. Internationally-recognized criteria for exclusion and consideration, as well as the criteria for exclusion and minimum requirements drawn up by AkEnd (2002) formed the basis for these investigations. Due to its unfavorable properties in comparison with rock salt and argillaceous rock, information on crystalline rock was not recorded. BGR's report summarized the results of research on rock salt, crystalline rock and argillaceous rock as potential host rocks in Germany, identifying favorable and unfavorable properties (Figure 7-5; BGR 2007, BMWi, 2008). The key characteristics of repository concepts in different host formations were also presented (Figure 7-6; BGR 2007, BMWi, 2008).

Property	Rock salt	Clay/ argillaceous rock	Crystalline rock (e. g. granite)
Thermal conductivity	High	Low	Medium
Permeability	Practically impermeable	Very low to low	Very low (unfractured) to permeable (fractured)
Strength	Medium	Low to medium	High
Deformation behavior	Visco-plastic (creep)	Plastic to brittle	Brittle
Stability of cavities	Self-supporting	Artificial reinforcement required	High (unfractured) to low (highly fractured)
In-situ stress	Isotropic	Anisotropic	Anisotropic
Dissolution behavior	High	Very low	Very low
Sorption behavior	Very low	Very high	Medium to high
Heat resistance	High	Low	High

Favorable property   Average   Unfavorable property

Figure 7-5
Properties of host rocks in Germany relevant to a geologic repository (BMWi, 2008). Used with permission of the Federal Ministry of Economics and Technology.

Components	Rock salt	Clay/argillaceous rock	Crystalline rock
Emplacement depth	Approx. 900 m	Approx. 500 m	500 – 1200 m
Storage technique*	Drifts and deep boreholes	Drifts and/or short boreholes	Boreholes or drifts
Storage temperature	Max. 200° C	Max. 100° C	Max. 100° C (bentonite backfill)
Backfill*	Crushed salt	Bentonite	Bentonite
Interim storage period (fuel rods and HLW canisters)	Min. 15 years	Min. 30 – 40 years	Min. 30 – 40 years
Drift reinforcement	Not necessary	Necessary and potentially very complex	Necessary in severely fractured zones
Container concept	Established	New development required for Germany	New development required for Germany
Mining experience	Very extensive (salt-mining)	Hardly any	Extensive (iron ore mining)

Favorable property   Average   Unfavorable property

Figure 7-6
Comparison of different repository concepts (BMWi, 2008). Used with permission of the Federal Ministry of Economics and Technology.

*: Adapted to host rock in question.

7.4 Disposal Concept

7.4.1 General Philosophy

The German disposal concept is based on two principles:

- All radioactive waste generated should be finally disposed of in Germany in deep geological formations.
- The geological barrier plays the most important role in isolating the waste, i.e., the rock salt in the case of Gorleben.

In making recommendations for a repository in salt, Germany relied on over 100 years' experience in industrial salt-mining, and more than 40 years of in-depth research, which produced an extensive knowledge base in Germany. This knowledge and mining experience indicated that:

- Stable underground structures can be constructed in salt and, due to favorable mechanical properties of rock salt, cavities can be created and maintained for decades without any special support.
- Rock salt reacts to mechanical load with a slow, flowing movement known as ‘creep’. This particular property of rock salt causes self sealing of cavities and, consequently, an effective geological barrier function is guaranteed in a natural way over long periods of time once emplacement is complete.

Consideration of retrievability and reversibility plays no part in the German disposal concept for irradiated nuclear fuel, which calls for the backfilling of disposal caverns as well as sealing of shafts and tunnels.

The reference concept envisages that irradiated nuclear fuel assemblies are placed in thick-walled casks, then sealed and emplaced in geologic formations. To promote this concept, a pilot-scale conditioning plant was designed and constructed at Gorleben. However, as a result of the 2001 agreement with the federal government, this facility is licensed only for the repair of damaged or defective casks for irradiated nuclear fuel or for vitrified HLW. There were multiple appeals against the existing operating license but all of these appeals were dismissed in court.

7.4.2 Repository Layout

An early concept of the perceived layout for a repository in salt is shown in Figure 7-7. The design was considered generic for a repository in salt formation. The diagram shows a central area with the underground infrastructure underneath the vertical shafts. Two wings lead off this central area for disposal, one for heat-emitting waste and the other for non-heat-generating waste. The heat-emitting waste is disposed of in boreholes. A more recent diagram of the envisaged repository in salt is shown in Figure 7-8.

Crushed salt is the intended backfill in rock salt. Its sealing capability depends largely on the degree of compaction and the resulting pore volume. Different research organizations in Germany have carried out laboratory studies to evaluate the influence of temperature, moisture, and mineral additives on the performance of crushed salt. Mineral additives are intended to accelerate compaction. Pre-compacted, crushed salt (bricks) have also been tested as backfill material. Some of this research is being performed as part of an EC-funded Integrated Project NF-Pro that began in 2004.

Research concerning nuclear waste disposal in rock salt was cut back significantly after 2000. During the recent years, such activities focused on dismantling the Thermal Simulation of Drift Emplacement field test in the Asse mine and analyzing the results (BAMBUS II) for topics such as the compaction behavior and sealing characteristics of crushed salt as backfill, laboratory studies on evolution and long-term behavior of the EDZ, and demonstration experiments on geotechnical barriers.

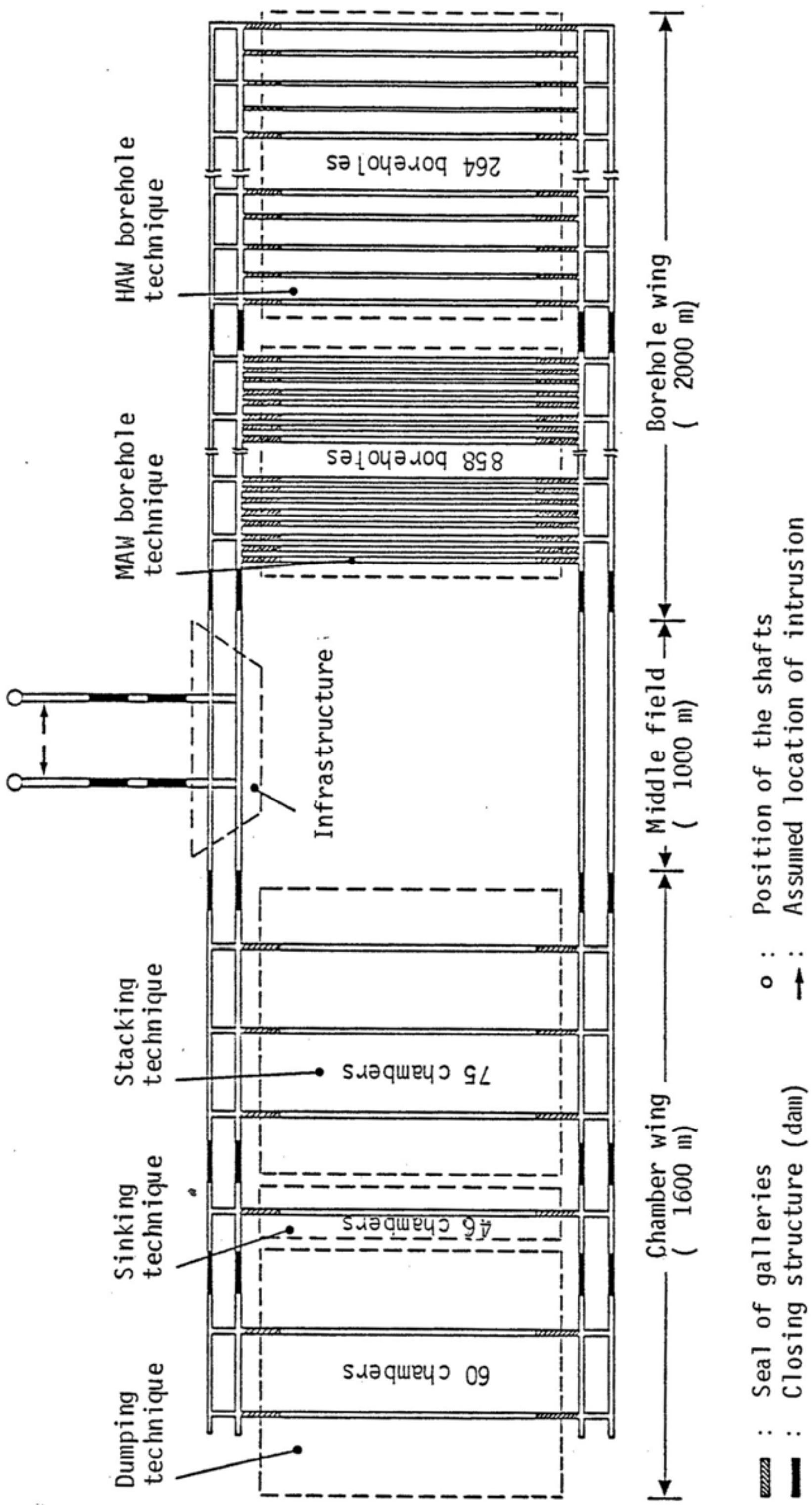

Figure 7-7
Schematic horizontal cross-section of early repository concept layout (PSE, 1985). Figure courtesy of Project Sicherheitsstudien Entsorgung.

NOTE: MAW = ILW; HAW = HLW; stacking, sinking and dumping represent three different disposal techniques.

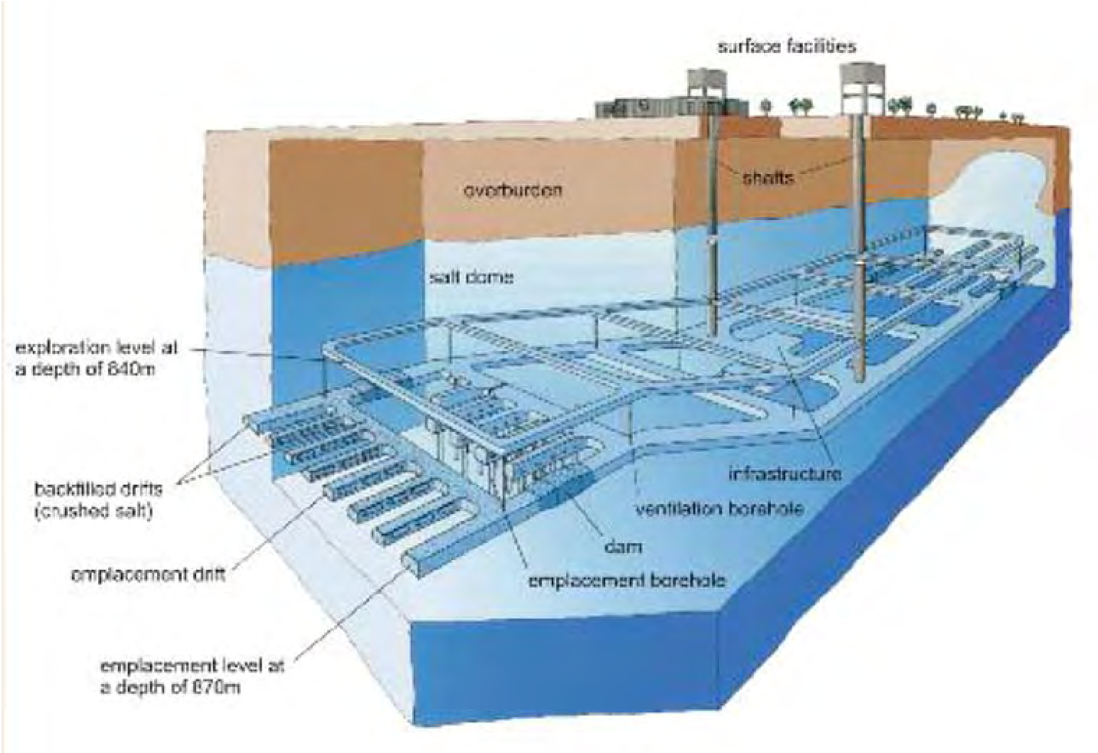

Figure 7-8
Conceptual diagram of potential repository layout in salt (BMWi, 2008). Used with permission of the Federal Ministry of Economics and Technology.

7.5 Transparency and Stakeholder Involvement

7.5.1 Public Involvement

Public involvement in large-scale projects, such as waste management and disposal facilities, are specified in the Nuclear Licensing Procedures Ordinance as well as the Environmental Impact Assessment Act, which are designed to ensure that the general public has access to all necessary information on a project. Such projects are publicly announced and associated documents are available to the public.

The Nuclear Licensing Procedures Ordinance specifies that the licensing authority must involve the general public, in particular those likely to be affected by the planned facility. Importantly, participation of the public is not limited to the *Land* in which the nuclear (disposal) facility is to be sited. Information on the proposed facility is published in the Official Bulletin as well as local newspapers once the relevant documents have been submitted. The Ordinance further states that the application, safety evaluation report, description of the proposed facility and the associated environmental impacts. Are to be made available to the general public. There is also a public hearing, where objections to the proposed facility can be raised for discussion. Those people raising objections are able to explain them at the public hearing and the licensing authority must take them into account and address them in its final decision.

7.5.2 International Involvement

Recent German R&D activities for nuclear waste disposal have been driven by the government's decision, in 2000, to pursue research activities in clay and granite. This decision required establishing new priorities for a number of current research projects as well as initiating new projects. Consequently, the importance of international cooperation in joint projects, and particularly the German participation in the underground research laboratory programs of fellow European countries, has increased significantly.

The long-term evolution of rock salt is also one of the objectives of the NF-Pro project. Laboratory studies have been carried out to analyze the failure and healing behavior of rock salt, dependent on the state of stress and temperature.

Shaft, gallery, and borehole seals, at full scale dimensions, are being tested in operating salt mines. These tests are being carried out as part of the research concept "Underground Disposal of Chemo-toxic Waste." In addition, emplacement techniques are being tested as part of an EC-funded Integrated Project "Engineering Studies and Demonstrations of Repository Designs (construction-operation- closure)" (ESDRED).

The current research on nuclear waste disposal in Germany, funded by the German government and the EC, focuses mainly on clay and granite as host rocks, and on clay as backfill and sealing material in salt, clay, and granite. Since no underground research laboratories in granite and clay exist in Germany, *in situ* experiments are being conducted as part of several international cooperation efforts, at underground laboratories in Sweden (Äspö), Switzerland (Grimsel, Mont Terri), France (Tournemire, Meuse/Haute Marne), and Belgium (Mol). (BMWi, 2008).

7.6 Safety Assessment and Licensing

7.6.1 Safety Assessment

Assessments carried out in support of geologic disposal in salt have been generic in nature.

In the early assessment work (PSE, 1985), one scenario that was considered consisted of a water or brine intrusion from the overlying rock into the residual cavities of a filled repository during the post-operational period. The timing of this scenario was assumed to be variable. Possible pathways included:

- A hypothetical permeable layer of rock between salt rock;
- Future drilling into repository;
- Backfill of shafts;
- Brine intrusion pathway between the mine workings and the brine inclusion of limited volume.

It was assumed conservatively that a permeable pathway was formed immediately on closure.

Based on a series of qualitative arguments, the scenario finally selected for assessment was intrusion of brine from the overlying rock into the middle area of the repository (middle field, see Figure DE-8) via the permeable pathway at the beginning of the post-operational period.

The quantity of radioactive waste considered in this particular assessment was based on an assumed energy output of 2,500 GWa, resulting in an estimated 1,825,000 waste packages of which 3% are deposited in HAW boreholes. Relative releases from the total repository were <1% "for most radionuclides" (PSE, 1985).

7.6.2 Licensing Process

The various agencies, and interactions between agencies, relevant to the licensing procedure for nuclear facilities are shown in Figure 7-9.

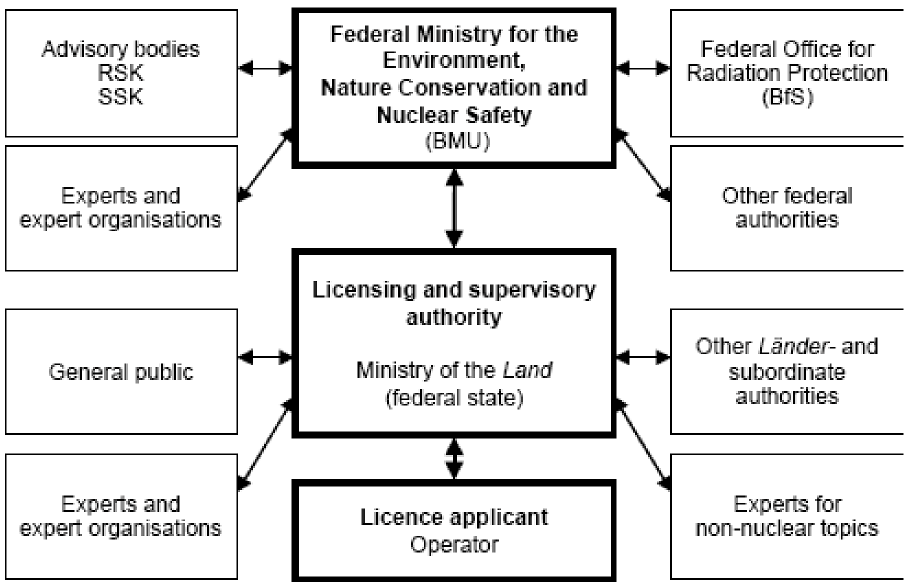

Figure 7-9
Parties involved in licensing of nuclear facilities (BMU, 2008). Used with permission of the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety.

The general licensing procedure is identified in AtG and StrSchV, which stipulate that construction and operation of nuclear facilities are subject to regulatory licensing. The AtG also states that licensing and supervision of such nuclear facilities are conducted by the *Land* where the particular facility is located. For the Gorleben facility, the relevant *Land* is Lower Saxony. The Federal Mining Act is also important for the licensing of a repository in a deep geological formation.

Construction of a disposal facility requires a special type of license – plan approval (Planfeststellung). In fact, approval is given if the requirements of the normal licensing application for a nuclear facility and submission documents are followed. One factor that is not the same as that for a normal nuclear facility is the liability, as the state itself has responsibility for this. Thus, the relevant section of the Atomic Energy Act states that the Federal Government and the Länder "are not obliged to make liability provisions."

The key difference in the plan approval procedure for disposal facilities is that all the legal areas are contained in one document, i.e., all the relevant licenses necessary are incorporated.

The licensing procedure stipulates that the "Safety Criteria for the Permanent Storage of Radioactive Waste in a Mine" must be taken into account, and this was the case in the licensing procedure for the Konrad repository. The criteria cover backfilling and closure of the repository and access shafts. For the post-operational phase of a repository, the safety criteria specify an annual dose limit of 0.3 mSv/year; this dose limit was superseded in the 2009 Safety Requirements by the specification of a risk limit.

7.6.2.1 Plan-Approval Procedure for Repositories

For the plan approval procedure, BfS is the applicant and is subject to the decisions taken by the licensing authority. Legal supervision regarding application of nuclear law by the respective *Land* is carried out by the federal ministry responsible for nuclear safety. The corresponding *Land* authority decides on plan approval.

The participants in the planning application and licensing of a waste repository are shown in Figure 7-10. As shown in this figure, BfS has a dual role. Supervision of compliance with both nuclear law and the requirements of the plan approval is also carried out by BfS in its self-supervisory department. BMU notes that separation of the departments responsible for repository development and regulatory overview avoids any issues concerning conflict of interest. Thus, the self-supervisory department is totally independent.

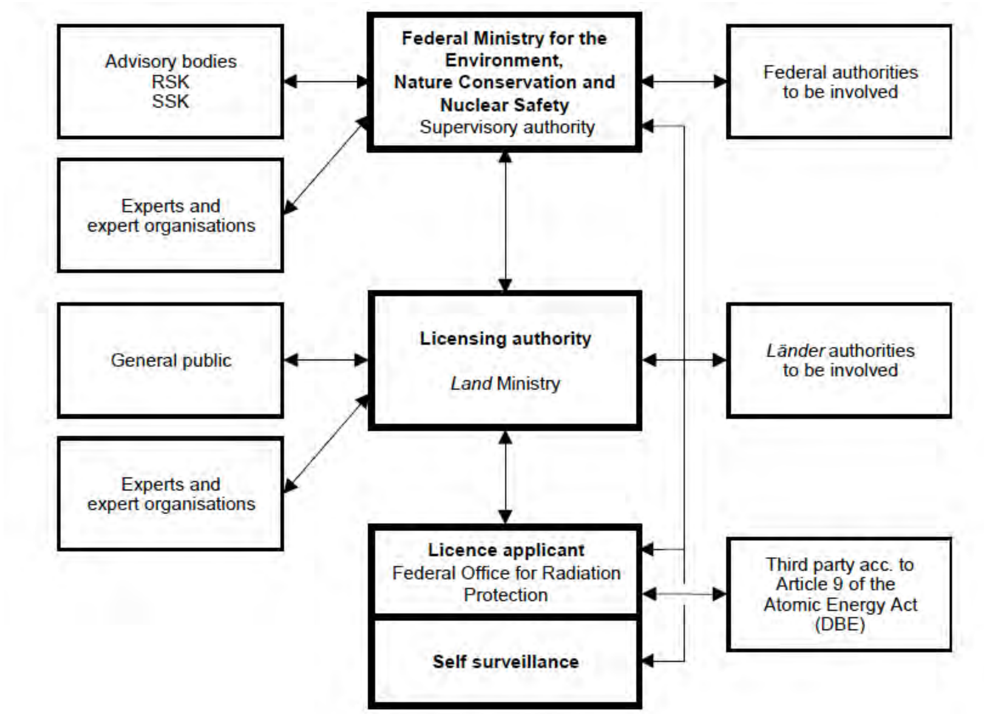

Figure 7-10
Participants in approval plan procedure for repository (BMU, 2008). Used with permission of the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety.

7.6.2.2 Experience with Konrad

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

7.6.2.3 Interim Storage Licensing Procedure

The specific application procedure, submission of supporting documents, and involvement of the general public are described in the *Nuclear Licensing Procedure Ordinance*. The Ordinance also addresses a staged licensing process as well as the environmental impact assessment, both of which are relevant to the licensing of a disposal facility.

One of the key documents is a safety evaluation report, which describes the facility, its operation and related impacts. Other topics that must be addressed in this report include:

- Concept description and explanation;
- Safety-relevant design principles;
- Description of precautionary measures to avoid / minimize damage to the facility and its surroundings.

Licenses for interim storage facilities are generally valid for a period of 40 years starting from the emplacement of the first storage cask.

7.6.2.4 Safety Requirements

Safety requirements for the disposal of radioactive wastes in a mine setting were published in 1983[22] and have since been updated to reflect international consensus (recommendations and standards) on deep geologic disposal, specifically for heat-generating wastes. Key components of the update in safety requirements included:

- Stepwise approach to repository development and assessment;
- Safety optimization;
- General protection of the environment.

BMU notes that there are no quantitative requirements for the disposal of heat-generating wastes. The current criteria state that "*the thermal output and the surface temperature of the packages for the disposal of heat-generating radioactive wastes should be determined in such a way that the specified properties of the packages are maintained and integrity of the geological formations is not endangered*" (BMU, 2008).

Criteria exist for leak tightness of storage casks as well as contamination on the outside of the cask, the latter applicable to all types of waste package.

7.7 Current Status

Wallner *et al.* (2006) describe recent developments in radioactive waste disposal in Germany as "*characterized by stagnation*". This opinion is based mainly on the Federal Government's 2001 agreement with the utilities to limit the future use of nuclear power, since when there have been limited developments in the field of disposal of all types of radioactive waste disposal. Exploration of the Gorleben salt dome is on hold and has been since October 2000. Interim storage facilities have been licensed at 12 reactor sites to avoid transportation of irradiated nuclear fuel to existing centralized storage facilities.

[22] *Safety Criteria for the Permanent Storage of Radioactive Wastes in a Mine.*

On the other hand, the amendment to the AtG in 2002 specified the intention to construct a repository for heat-generating radioactive waste around 2030. In announcing the new waste disposal policy, the Ministry for the Environment stated the following political objectives:

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

In addition, a solution was eventually reached concerning the final disposal of non-heat-generating L/ILW. In March 2007, the plan approval for the former Konrad iron ore mine near Salzgitter was finally endorsed by the Federal Administrative Court and cannot be appealed. The mine is now being converted into a repository, with plans to put it into operation in 2013.

Two sites are to be selected by 2010 for underground exploration for a repository for heat-generating waste, with salt not necessarily the only option.

The AkEnd committee (AkEnd, 2002) identified three phases for the site-selection procedure, from development to implementation, all of which include public involvement:

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

The Federal Government is currently striving for an agreement on financing site exploration. The nuclear industry, however, has already declared that they consider further site selection activity unnecessary and therefore are not willing to pay for it (Nies, 2004).

There still appears to be substantial support in the local councils of Gorleben as well as the neighboring Gartow community, for the continuation of the exploration work. At this stage, the community representatives demand an end to the political gridlock and want clarity on the future of the site. However, there continues to be a strong protest movement against Gorleben, not only supported by some local citizens but also driven by opponents of nuclear power from all over Germany.

BMU (2008) notes that the EBS disposal concept for irradiated nuclear fuel assemblies has reached "technical maturity", including a prototype of a fully-shielded POLLUX cask and the alternative concept of the unshielded rod canister.

In 2010, BMU commissioned GRS (the Association for Plant and Reactor Safety), supported by a number of organizations, to conduct a preliminary safety analysis for Gorleben and summarize all available information on the salt dome and the results of exploration activities to date. The main objective of this safety analysis is to develop a clear conclusion, on the basis of available information, on whether the Gorleben site can comply with the new safety requirements for the final disposal of heat-generating, radioactive waste. BMU notes that interim results and reports will be made available to the interested public on its website[23]. BMU also announced that the preliminary safety analysis, which is expected to be submitted by the end of 2012, will be reviewed by an international peer review group.[3]

7.8 Summary and Key Observations

- *Policy on Geologic Disposal*: In the 1970s, the German utilities planned to construct an integrated nuclear disposal center combining waste management facilities (reprocessing, fuel fabrication, waste conditioning) as well as a repository, all located at Gorleben. While exploration of the Gorleben salt dome has been on hold since October 2000, the amended AtG (2002) specified the intention to construct a repository for heat-generating radioactive waste around 2030.
- *Institutional Arrangements*: BfS, under the authority of BMU, is implementer, supported by the DBE, which has the responsibility for planning, designing, constructing, and operating the repository. In addition to its supervisory role of BfS, BMU is responsible for nuclear safety and radiation protection and has the oversight to enforce regulations issued by the regulatory authorities in the States. With federalism reform in 2006, the federal government was given exclusive powers in the regulation of nuclear power, so that the Länder only have legislative authority if explicitly authorized in the (federal) Basic Law. Waste management costs from nuclear power generation are the responsibility of the waste producers – primarily the electric utilities. Utilities set aside and manage the funds internally to cover HLW management costs, including repository development, according to established legislative guidelines.
- *Key Laws and Regulations*: A hierarchy of laws, ordinances, and regulations govern radioactive disposal in Germany. New safety requirements for the geologic disposal of heat-generating radioactive waste were issued in 2009 covering the planning, site exploration, construction, operation and closure of a final repository. Site selection and licensing are not addressed in this document.
- *Site Screening and Selection*: Salt was recognized in the early 1960's as a potential geological medium for a HLW repository, and a large-scale study between 1964 and 1976 was devoted to the identification of potential salt structures in Germany. Criteria for suitable salt formations were formulated in 1964. Of 140 salt domes selected initially for evaluation, 23 were selected for further study. Further refinement in selection criteria reduced the number of sites to 13 and, eventually 4 sites, after applying criteria in terms of (i) safety and the environment; (ii) infrastructure (transportation); (ii) structural policy. After detailed site investigations involving the four candidate salt sites, Gorleben was eventually selected by Lower Saxony as a suitable site. This salt dome consists primarily of pure rock salt layers, which are largely solution-free. An impermeable salt barrier, approximately 600 m thick, extends across the planned emplacement area to the overburden. Further study was halted for political reasons, although the consensus of experts who have reviewed various documents on Gorleben is that the results "do not contradict the positive appraisal of the geological findings at the Gorleben site". More recently, Germany is considering the possibility of geologic disposal in formations other than salt, e.g. low-permeability argillaceous rocks as an alternative.

[23] BMU press release No. 118/10, Berlin, 05.08.2010.

- *Repository Design Concepts*: In the German disposal concept, the geological barrier plays the most important role in isolating the waste, i.e., the rock salt in the case of Gorleben. As well as providing a mechanically stable rock formation, the self-sealing property of salt allows total isolation. For this reason, retrievability plays no part in the German disposal concept. The effort in Germany in terms of design concept has been devoted to the development of thick-walled casks, which reached the demonstration stage only. Crushed salt is the intended backfill. Germany is currently the only country where salt is the principal host geology being considered for heat-generating HLW.

- *Performance Metrics and Assessments*: The 2009 Safety Requirements provide the necessary quantitative criteria in the form of risk:

  – For all 'likely' future evolutions of the repository system (likely = cases where supporting quantitative data indicate a likelihood >10%), the total risk should be < 10^{-4} with regard to individuals with a 'lifetime' exposure (lifetime = current expected life expectancy).

  – For all less-likely events, the additional risk to affected individuals shall be <10^{-3}, i.e. for such events a greater risk of radionuclide release and individual exposure is allowed because the events themselves are less likely. Where supporting quantitative data on probability exist, the likelihood attached to such events is >1% (but <10%).

  The reference time period, for which evidence of safety (safety case) must be provided, is one million years.

  Assessments carried out in support of geologic disposal in salt have been generic in nature, as well as being performed prior to the 2009 Safety Requirements. In the early assessment work, a scenario under consideration consisted intrusion of a water or brine from the overlying rock into the residual cavities of a filled repository during the post-operational period. The timing of this scenario was assumed to be variable. It was assumed conservatively that a permeable pathway was formed immediately on closure. Based on a series of qualitative arguments, the scenario finally selected for assessment was intrusion of brine from the overlying rock into the middle area of the repository via the permeable pathway at the beginning of the post-operational period. Relative releases from the total repository were <1% "for most radionuclides".

- *Independent Peer-Review and Advisory Bodies*: Once selection of the Gorleben site had been announced, a public symposium ("Gorleben Hearing") was held in 1979, to which by a number of international participants were invited. There has not yet been an opportunity for international peer review of any assessment work. A key advisory body to the government is AkEnd, which was commissioned to recommend the selection of possible sites for heat-generating waste as well as developing procedures and criteria for selecting sites. Independent advisory bodies to the regulatory division of BMU include the Reactor Safety Commission (RSK) and the Commission on Radiological Protection (SSK), both of which make recommendations that feed directly into the licensing process.

- *Stakeholder and Public Involvement*: The public had minimum involvement in the decision leading to the selection of the Gorleben salt dome as a potential host formation for a geologic repository, until the decision was announced at which time there was fierce opposition to the program. Current legislation now ensures that public involvement in the process is significant, via the EIA process, with those raising objections being given the opportunity to explain their opposition at public hearings.
- *Program Maturity*: After a period of site selection activities leading to the selection the Gorleben salt dome as potential site for heat-generating waste, the repository program in Germany has essentially been on hold for almost 10 years due largely to political reasons; the moratorium on characterization of the Gorleben site ended in 2010.

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8
JAPAN

8.1 Introduction

8.1.1 General Nuclear Profile

Japan is the third largest producer of nuclear electricity, with 263 TWh generated in 2009 representing 29% of the country's total electricty production (WNA, 2010a). Its commercial nuclear power fleet comprises 54 operating LWRs (a mixture of BWRs and PWRs spanning Generation II and III reactor designs) and the Monju prototype sodium-cooled fast reactor, which was recently restarted in 2010 after an extended maintenance period following serious problems with sodium leakage during initial startup testing in 1995. Three of the operating LWRs are advanced BWR designs and therefore represent the latest in deployed reactor technology worldwide; construction of newer designs are also planned or underway, which explicitly allow for operation on 100% MOX cores. This list includes an advanced BWR at Ohma under construction and planning for two advanced PWRs at Tsuruga (WNA, 2010b).

Japan is aggressively pursuing further growth of nuclear capacity to reduce dependence on fossil fuel imports. Accordingly, additional plants are under construction or planned (METI, 2008). Japan as a nation has also adopted an energy policy that seeks to maximize energy recovery from imported uranium resources through extensive Pu-recycling as MOX in LWRs under its so-called "Pluthermal" program and eventual closure of the fuel with deployment of fast spectrum reactors. In 2009, began loading partial MOX cores after years of delay. As of 2010, three Japanese reactors were running on partial MOX cores (WNA, 2010).

Reprocessing of used fuel is an integral part of its energy strategy and back-end fuel cycle. In advance of establishing its own commercial-scale reprocessing infrastructure, Japan shipped 2900 and 4200 MTHM of irradiated nuclear fuel to AREVA in France and BNFL in the United Kingdom, respectively, for reprocessing services. The estimated HLW to be returned to Japan from foreign reprocessing corresponds to a total of 2,200 canisters. Within Japan, over 1000 MTHM of irradiated fuel was reprocessed at the pilot-scale Tokai-mura plant over a three-decade period. Meanwhile, Japan is awaiting startup of the industrial-scale (800 MTHM/yr) Rokkasho reprocessing facility. Construction of the facility has been completed after lengthy delays, and commercial operation has been further delayed (until late 2012 at the earliest) due to problems with the associated vitrification facility.

Japan's strategy for implementing geologic disposal of HLW has centered on a volunteer approach to site selection, whereby communities volunteer preliminary investigation areas (PIAs) as potential candidate sites. This strategy has not been successful to date. Geologic disposal is referred to as *Category 1* waste disposal (METI, 2008; see also Section 8.2.2). Wet storage is the principal method of storage of used nuclear fuel at commercial reactor sites, although dry storage has been implemented to a limited extent at two reactor sites. In addition, Tokyo Electric Power Company and Japan Atomic Power Company jointly formed the Recycle Fuel storage Company in order to develop the first commercial interim fuel storage facilities in Japan (NISA/METI, 2008).

8.1.2 Used Fuel / HLW Inventory and Projected Disposal Requirements

The used fuel inventory as of the end of March 2008 was 12,190 MTHM of UOX in wet storage, and 2,576 MTHM used UOX and MOX fuel elements at reprocessing plants (NISA/METI, 2008). A total of 1,664 HLW canisters of HLW glass have been generated so far, with the equivalent of an additional 22,200 HLW canisters waiting to be reprocessed. By 2020, the estimated total inventory of HLW for disposal will be on the order of 44,000 canisters (NUMO, 2009).

8.2 Institutional Arrangements

8.2.1 Institutional Framework

The HLW geologic disposal program in Japan was effectively launched in 1976 with the Japanese Atomic Energy Commission's (AEC's) first formal statement on the strategy for radioactive waste management in Japan (Masuda, 2002). The AEC (2000) stated that HLW from the reprocessing of irradiated nuclear fuel was to be vitrified, stored for a period of 30 to 50 years for cooling, and finally disposed of in a stable geological environment deep underground. Figure 8-1 shows a schematic of the relationship between the key waste management organizations in Japan. Principal roles are described below.

POLICY and OVERSIGHT - Under the 2000 Act, the *Ministry of Economy, Trade and Industry* (METI) has overall responsibility for establishing Basic Policy (MITI, 2000a)[24] as well as a Final Disposal 10-year Plan (MITI, 2000b), the latter to be updated every 5 years.

IMPLEMENTER - The *Nuclear Waste Management Organization of Japan* (NUMO), established in 2000, is the implementing organization for HLW disposal, responsible for the R&D necessary for implementing final disposal, with the focus on safety, feasibility, and cost efficiency. NUMO's activities include repository site selection, preparation of relevant license applications, as well as the construction, operation, and closure of the repository. As the first step in the siting process, NUMO announced an overall procedure for selecting potential candidate sites, followed by the identification of siting factors and a repository-concept catalogue to be provided to all municipalities in Japan. NUMO reports to METI, as shown in Figure 8-1.

[24] METI succeeded MITI (Ministry of International Trade and Industry) in 2001.

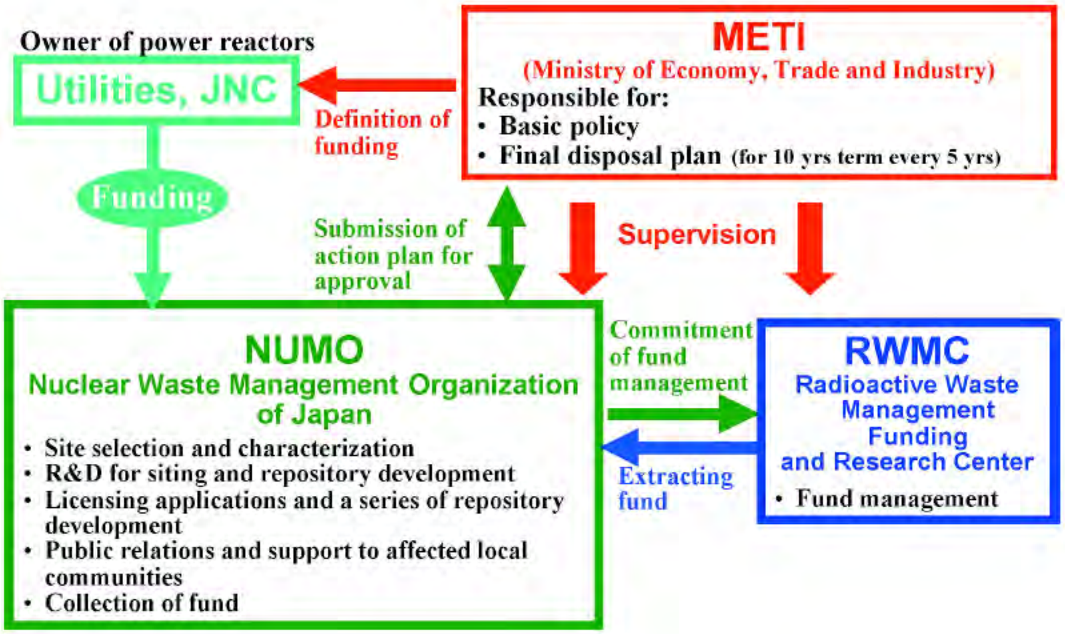

Figure 8-1
Institutional framework for HLW management and disposal program in Japan (Masuda and Kawata, 2001). Used with permission.

NOTE: As noted in the text, JNC has integrated with JAERI to form JAEA.

REGULATOR - Three agencies hold some portion of regulatory responsibilities for waste produced in commercial nuclear power plants. These are, in order of importance:

- *Nuclear Safety Commission* (NSC): provides safety overview, supervision and audit of regulatory bodies, acting on behalf of METI;
- *Nuclear and Industrial Safety Agency* (NISA): issues licenses for commercial nuclear facilities, formulates decrees, approves safety programs and carries out periodic inspections of facilities;
- *Japan Nuclear Energy Safety Organization* (JNES): provides technical support to NISA and NSC,

METI has overall responsibility for JNES and NISA.

ADVISORY and SUPPORT - *The Japanese Atomic Energy Agency* (JAEA) conducts research on behalf of the utilities and also supports NUMO as implementer. One key responsibility of JAEA is to further ensure the reliability of disposal technology and safety assessment (SA) methodologies, based particularly on studies at the Mizunami and Horonobe underground research laboratories (URLs; see Section 8.3.3), as well as the ENTRY and QUALITY facilities at JAEA's nuclear site at Tokai. JAEA was formed in October 2005 by the merger of the Japanese Nuclear Cycle Development Institute (JNC)[25] and the Japanese Atomic Energy

[25] Formerly the Power Reactor and Nuclear Fuel Development Corporation (PNC).

Research Institute (JAERI). Previously, JNC and its predecessor, PNC, had been the lead organization responsible for implementing research activities during the generic research and development (R&D) phase to provide a scientific and technical basis for the geologic disposal of HLW in Japan as well as promoting an understanding of the HLW safety concept with stakeholders. Thus, PNC submitted its first R&D progress report, referred to as H3, for the geologic disposal of HLW in Japan in September 1992, summarizing key results and identifying priority issues for further study (PNC, 1992). Since the H3 report and its successor report H12 (JNC, 2000), the R&D framework for geologic disposal was restructured according to government plans as defined in the Basic Policy Plan (MITI, 2000a), the Long-Term Program (AEC, 2000), and the Report of the Nuclear Sub-Committee (METI, 2001).

8.2.2 Legal and Regulatory Framework

No specific regulation for HLW disposal exists in Japan, only guidelines. Prior to the H12 assessment (JNC, 2000), AEC Guidelines made a number of recommendations, in particular that:

- National standards and international recommendations should be taken into account;
- Annual dose should be adopted as the basic indicator of safety.

No timeframe was specified for the safety assessment in AEC's guidelines, although the Commission was interested in seeing when the peak dose occurred and for how long.

Besides the general Atomic Energy Basic Law, the key piece of legislation governing HLW disposal is the Specified Radioactive Waste[26] Final Disposal Act, which came into force in June 2000. The Act specifies the procedures for site selection, the implementation body and funding in relation to the disposal of vitrified waste originating from the reprocessing of irradiated nuclear fuel. With regard to the disposal of other types of radioactive waste, in particular, TRU waste originating from reprocessing and MOX fuel fabrication facilities, relevant laws were amended in 2007 to allow for the possible co-disposal with HLW.

The 2000 Act calls for the establishment of safety regulations, and discussions related to the development of these safety regulations have been ongoing between the NSC and the NISA. The NSC is responsible for providing guidelines for safety regulations and has published three key documents:

- "*First Report on the Basis for Safety Standards for HLW Disposal*" (NSC, 2000) concerning safety standards for the geological disposal system, which took into account the results from the 2000 assessment (H12; JNC, 2000) as well as public response; and
- "*Requirements on the Geological Environment for Selecting Preliminary Investigation Areas for HLW Disposal*" (NSC, 2002). The requirements contained in this document are expected to be reflected in the initial selection of sites.
- "*Licensing Procedure Relating to the Safety Regulation of Specified Radioactive Waste Disposal and the Role of the Nuclear Safety Commission*" (NSC, 2007).

[26] Japan's terminology "Specific radioactive waste" signifies HLW from the reprocessing of irradiated nuclear fuel.

The NISA identified the key research and development issues related to developing safety regulations in a 2001 report (NISA, 2001).

Since the H12 assessment, the NSC has issued a report (NSC, 2007) on the application of risk assessment to address uncertainties concerning the long-term safety evaluation of the post-closure period of a waste repository. Basing its recommendations on those of ICRP, the NSC identified three types of post-closure scenario: Basic Scenario, Changeable Scenario, and Human/Very Rare Event Scenario" for which the corresponding dose-based criteria for radiation protection are 10 μSv/year, 300 μSv/year and 10-100 mSv/year, respectively. Those values were derived from a dose/probability analysis for each scenario. Currently, the NSC is revising its "Basic Guide for Safety Review for Radioactive Waste Burial Facilities". Again, no timeframe is specified.

With regard to radiation protection, NISA/METI (2008) note that the recommendations of ICRP have been incorporated into national legislation and regulation, citing an effective dose limit of 1 mSv/year, but stating that operators of waste disposal facilities must define a target value lower than 1 mSv/year based on the ALARA principle.

8.2.3 Waste Classification

As stated above, amendments were made to the relevant laws in 2007 (June) to establish a safety regulatory system for the final disposal of HLW. Two main classifications of waste disposal were identified:

- *Category 1* waste disposal – disposal of HLW;
- *Category 2* waste disposal – disposal of LLW; further subdivided into:
  - long-lived low-heat-generating radioactive waste (TRU),
  - uranium wastes,
  - power-plant waste, and
  - research-facility waste.

The amendments then set in place the development of new ordinances prescribing detailed procedures for Category 1 and Category 2 waste disposal. These new ordinances came into force in April 2008.

8.2.4 Funding

The electric power utilities as producers of used fuel and, ultimately, the HLW collectively bear the costs of final disposal. To cover these costs, the utilities make contributions to a national disposal fund according to the amount of electricity generated. Under the 2000 Act, the Radioactive Waste Management Funding and Research Center (RWMC), formerly the organization aimed at promoting research associated with radioactive waste management, was designated as the funding management organization. While the RWMC manages the national disposal fund, METI is responsible for allocating NUMO's budget. A recent estimate of the disposal program is ~3 trillion yen, corresponding to 0.13 yen/kW-hour for a repository that can accommodate 40,000 HLW canisters (NEA, 2010). An additional fee of 0.07 yen/kW-hour is charged to utilities to cover waste management activities prior to the national fund being established (NEA, 2005). Apart from its funding management role, RWMC's other responsibilities include supporting national policy-making on geological disposal by pursuing R&D activities on sociological issues and advanced technological options, as well as gathering, analyzing, and producing information as appropriate to disseminate to a variety of stakeholders. As with NUMO, the RWMC reports to METI (Figure 8-1).

8.3 Site Screening, Selection, and Characterization

8.3.1 Early Studies

Much of the early work on geological studies was coordinated by PNC/JNC in the absence of an implementing body. The work involved reviews of Japan's geology and geological structures, taking advantage, where possible, of information and measurement/data from underground resources, e.g. mines. At that time, seismic activity, uplift, erosion, igneous activity, climate change and sea level change were listed as major natural events that could potentially affect the long-term stability of the geological environment.

The major types of rock in Japan were classified into igneous, sedimentary, and metamorphic rocks, pyroclastic material, and pyroclastic rock, and the mineralogical and geochemical characteristics of each type as well as its occurrence were reviewed. With regard to groundwater in Japan, the principal characteristics reviewed were hydraulic conductivity, hydrology, and geochemistry.

With regard to seismic stability, records of seismic activities and fault activities were reviewed. In addition, records on igneous activity (volcanic eruptions, magmatic intrusion and hydrothermal processes) were examined. Japan has also undergone uplift and erosion during the Quaternary period, with uplift rates in the range 0.15-1 mm/year. Some major coastal plains are areas of submergence (PNC, 1992).

The occurrence of natural events and their impacts on the geological environment investigated over time scales of several hundred thousand years or more, based on field studies in regions where the history of these events could be observed. These studies indicated that the location of sudden localized events, such as volcanic activity and major fault movement, can be specified sufficiently well that their effects can be avoided by selecting an appropriate disposal site.

As an example, tracing back the history of volcanic activity in the Quaternary shows that the locations where such activity occurred are restricted to distinct regions and that there is little change in these locations. In addition, the direct effects of volcanic activity that could significantly influence the repository performance were expected to be located no more than a few tens of kilometers from the activity centers (Figure 8-2).

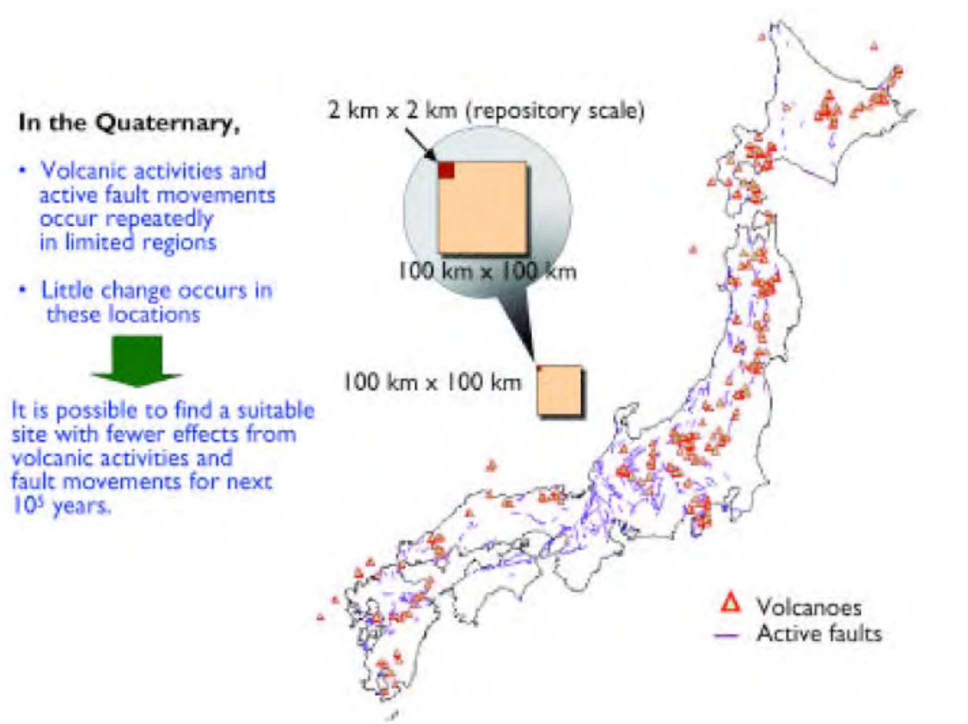

Figure 8-2
Stability of the geological environment in Japan – distribution of volcanoes and active faults (Masuda and Kawata, 2001). Used with permission.

On the other hand, gradual phenomena such as uplift and erosion, or climate and sea-level changes, are more widespread. However, JNC argued that it was possible to estimate their future trends and potential effects by extrapolating data obtained from field studies. Thus, the conclusion from these geological stability studies was that it is possible to select a sufficiently stable environment for geological disposal (JNC, 2000).

8.3.2 Site Selection Activities

PNC/JNC acknowledged the significant challenge in identifying potential sites in Japan given the tectonic history of the country. PNC/JNC also noted the difficulty in avoiding the specification of preferred locations or host rocks as a basis for developing specifications for the Engineered Barrier System (EBS), the challenge being a compromise between trying to be as generic as possible and identifying a specific host rock in order to develop a detailed specification for the EBS. The resulting general disposal concept is shown in Figure 8-3.

- From JNC's perspective, the geological barrier serves to provide two key functions:
- Long-term stability of the repository site; and
- An appropriate barrier function in terms of suitable hydrogeological, geochemical, and geomechanical conditions.

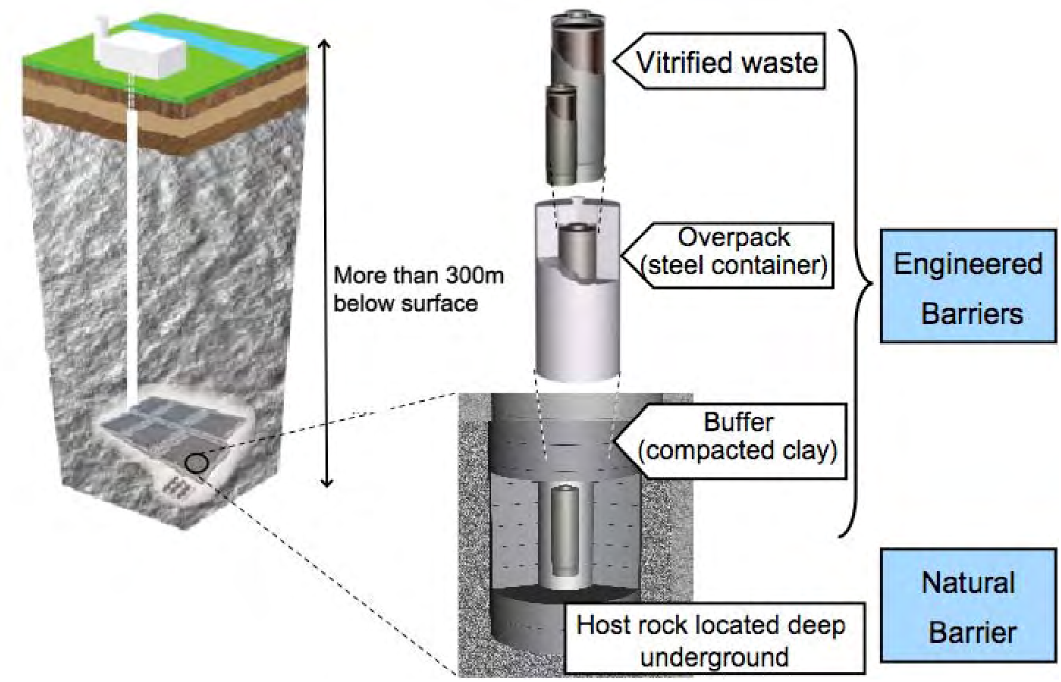

Figure 8-3
General disposal concept (NUMO, 2009). Used with permission of NUMO.

At the time of the H3 Progress Report (PNC, 1992), the Japanese AEC had hoped for a wide range of sites to cover the geological features in Japan, but this hope did not materialize. Subsequently, the AEC issued a set of Guidelines in 1997 concerning research and development activities for HLW disposal, including the technical issues that had to be addressed in the subsequent progress report. One of these issues was the technical basis for selecting repository sites. As a result, the subsequent H12 Report (JNC, 2000) addressed these technical issues but in a generic way (see Section 8.6.1).

More recently, NUMO has been promoting R&D activities aimed at providing the scientific background of siting factors for the selection of Preliminary Investigation Areas (PIAs), as well as approaches and methods for developing repository concepts tailored to volunteer sites. These two major issues are summarized in two technical documents published and made available to stakeholders (NUMO, 2004a; NUMO, 2004b).

The 2000 Act specifies that the siting process consist of three stages, as shown in Figure 8-4:

- *Stage 1*: PIAs for potential sites are nominated based on regional literature surveys, focusing on the long-term stability of the geological environment.
- *Stage 2*: Detailed investigation areas (DIAs) for candidate sites are then selected from the PIAs via surface-based investigations, including boreholes drilled to evaluate the characteristics of the geological environment.
- *Stage 3*: Detailed site characterization, including underground experimental facilities, leading to the selection of the repository construction site.

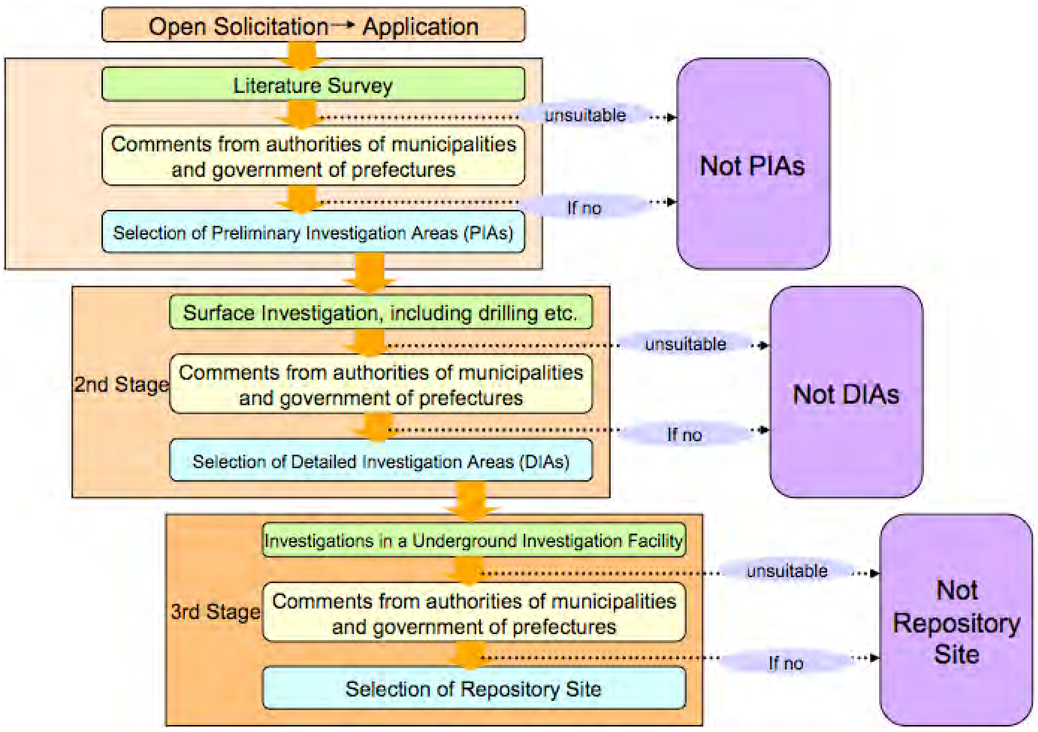

Figure 8-4
Site selection procedure developed by NUMO (2009). Used with permission of NUMO.

8.3.3 Detailed Geological Studies / URLs

JNC has been particularly active in establishing and demonstrating site characterization methodologies via two generic URL projects, one at Mizunami (MIU) in crystalline rock and the other at Horonobe in sedimentary rock. Detailed geological and hydrogeological studies had previously been carried out in the Tono mine, Mizunami, in a tunnel ~100 m underground (Yusa et al., 1992). Also, since 1988, underground investigations have taken place at the Kamaishi iron and copper mine, in a gallery 300 m below the ground surface with a second gallery 700 m below the ground surface (Takeda and Osawa, 1993). Both Tono and Kamaishi are located in granite host rock.

With regard to the new URLs:

- *Mizunami URL*: Surface-based investigations (Phase 1) began in March 2002 at the MIU construction site, on land owned by Mizunami City (JNC, 2002). Phase 2, construction of the URL began soon after. The design of the Mizunami URL consists of two 1,000 m deep shafts. Shaft excavation began in July 2003 and construction of the Mizunami URL is expected to be completed by the end of 2010.
- *Horonobe URL*: Surface-based investigations (Phase 1) started in 2001 (Goto and Hama, 2003). Based on regional investigations, an area for detailed site investigations (URL area) was identified. The land for construction of the URL site was then acquired within the URL area, and site preparation began in July 2003. The design of the Horonobe URL consists of two 500 m deep access shafts. Shaft excavation (Phase 2) started in 2005 and construction of the URL is expected to be completed by the end of 2010.

Figure 8-5 shows the locations of these URLs. JNC emphasized the research nature of these URLs, distinct from site-specific underground test facilities planned to be constructed at potential waste disposal sites by NUMO.

The surface-based investigations carried out in Phase 1 of the Mizunami and Horonobe projects has been integrated into a "geosynthesis" as a basis for further development during Phases 2 and 3. These URL projects also offer the opportunity to develop and test the tools and methodologies required for site characterization.

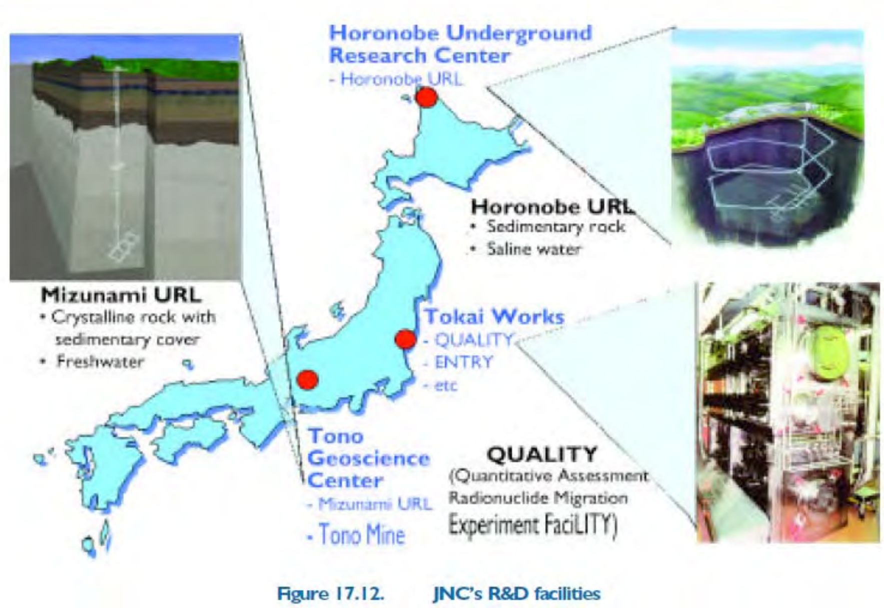

Figure 8-5
URL's in Japan: Mizunami and Horonobe (Masuda and Kawata, 2001). Used with permission.

8.3.4 R&D Studies

For JNC's R&D program since the H3 Report (PNC, 1992), three major areas of research were established following AEC Guidelines:

- Evaluation of the geological environments for hosting a repository;
- Engineering technology for a repository and EBS design, and
- Performance assessment of the disposal system.

JNC developed the Geological Disposal Technical Information Integration System (JGIS), a structured database, to promote program integration and sharing of technical information between the site investigation, repository design, and safety assessment teams (Uchida et al., 2003). The JGIS system consists of a relational database of technical information stored in the form of a structured flowchart. In addition to the development of such a structured knowledge base, JNC published a state-of-the-art "H17" report (JNC, 2005) that documents R&D results based on Phase 1 activities in the two URLs as well as studies in ENTRY and QUALITY.

8.4 Disposal Concept

8.4.1 Background

PNC carried out and published an assessment in a report referred to as the H3 Progress Report (PNC, 1992), taking into account a preliminary concept for the EBS. H3 summarized the results of R&D activities up to March 1992 and identified priority issues for further study. PNC submitted this report to the authorities in September 1992. JNC completed the second Progress Report, referred to as H12 (JNC, 2000) and submitted it to the AEC in November 1999. H12 was intended to demonstrate the technical feasibility and reliability of the specified disposal concept more rigorously and transparently, and provided input for future siting and regulatory processes.

In the absence of a specific site/location for the H12 assessment, JNC's feasibility study used data that were based on case studies and field measurements. Feasibility included the practical feasibility of designing and emplacing the EBS for a wide range of physical rock properties.

8.4.2 Repository Concept

JNC's multi-barrier disposal concept is shown in Figure 8-6 and is similar to that of Switzerland's disposal concept for granite host rock. As noted by JNC, greater emphasis is placed on the near-field barriers, given the complexity of the geological environment (Umeki, 1994).

JNC (2000) identified the characteristics of the geological environment considered important in terms of EBS barrier function, in particular:

- Groundwater flow rates,
- Rock permeability,
- Geochemical features of groundwater,
- Thermal and mechanical properties of rock formations, and
- Solute transport properties.

As a result of laboratory and engineering studies carried out after the H3 report, the H12 assessment was able to take advantage of results from the H3 assessment. In particular, the thickness of the overpack and buffer were reduced by ~30% resulting in an overall volume reduction for the EBS of ~50%. The composition of the buffer was also changed from pure bentonite to a mixture of bentonite and quartz sand.

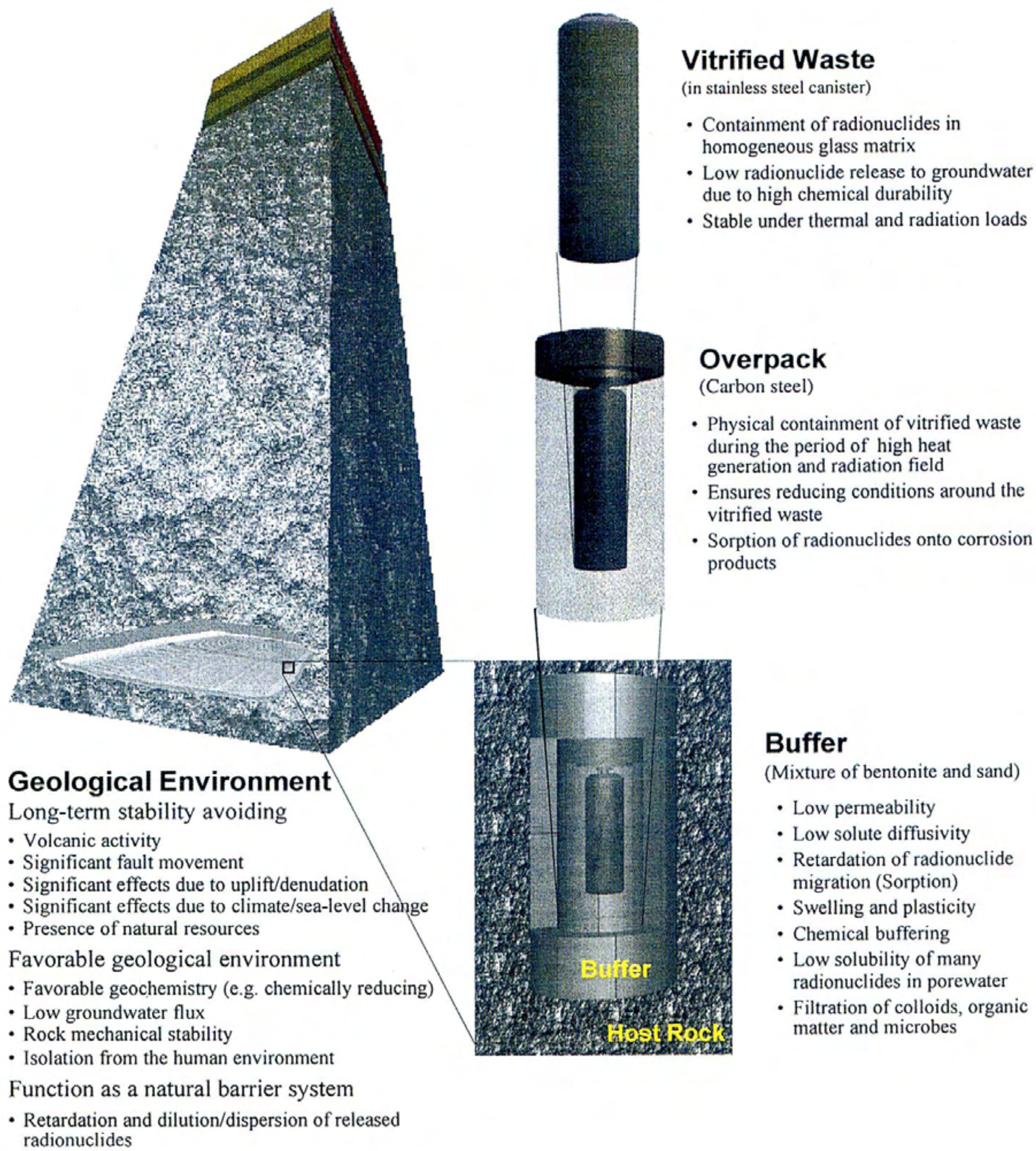

Figure 8-6
Multi-barrier concept for Japanese disposal concept (JNC, 2000). Used with permission of JAEA.

8.5 Transparency and Stakeholder Involvement

8.5.1 Public Involvement

In its overview of the H12 Project, JNC acknowledges the political sensitivity associated with the selection of potential sites for a HLW repository. Since the 2000 Act, the Basic Policy requires that at every stage of the site selection process, NUMO must solicit the opinions of local residents, and METI must solicit the opinions of governors and mayors, and that these opinions be documented and respected. NUMO's report on site investigations will be made available to local residents, and will be open for comments by them.

For the siting process, NUMO started open solicitation in December 2002, distributing an information package to all (3,239) municipalities in Japan. This package contained four documents:

- *Application Instructions*: General information including the approximate extent of the volunteered area.
- *Repository Concepts*: Information on what the planned repository might look like and how it will be developed for siting environments at potential candidate sites selected, taking into account the siting factors. An overview of the performance of different repository concepts is also provided in the document. One of the repository concepts, for coastal / sedimentary rock, is shown in Figure 8-7.
- *Siting Factors for the Selection of Preliminary Investigation Areas*: Guidance on area-specific surveys for PIAs, including factors specified in the 2000 Act (NSC, 2000), and the NSC environmental requirements (NSC, 2002), e.g., no record of significant tectonic movement, no evidence of unconsolidated sediments, and no mineral resources. Factors for exclusion included (Kitayama et al., 2006):
  - Clearly identified active faults;
  - Regions within a 15 km radius of Quaternary volcanoes;
  - Uplift of >300 m during th last 100,000 years;
  - Unconsolidated Quaternary deposits;
  - Economically valuable mineral deposits.
  - *Outreach Scheme*: Description of the benefits to volunteer municipalities, not only from a financial perspective, but also with respect to other positive social aspects.

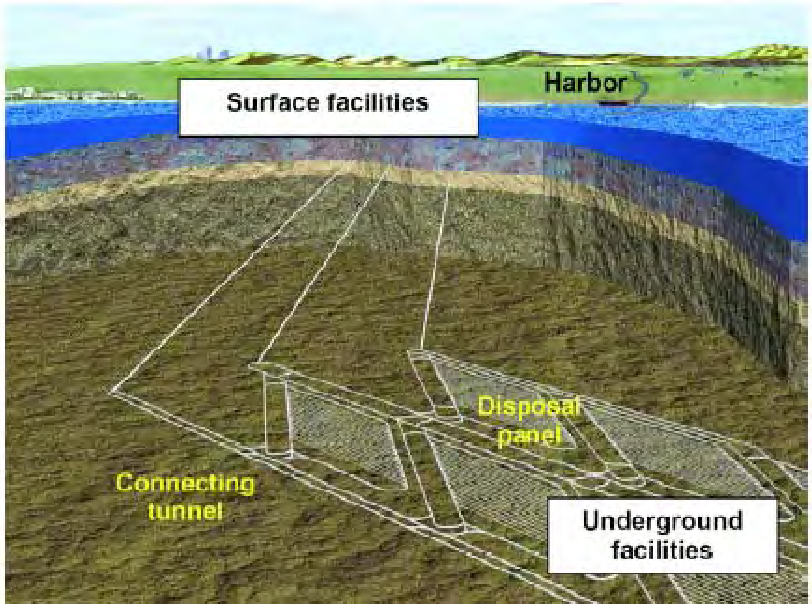

Figure 8-7
Example of one of the repository concepts for coastal/sedimentary rock (Kitayama et al., 2006). Used with permission of NUMO.

In support of the information package, NUMO also published two technical documents (NUMO 2004a, 2004b), intended for stakeholders with more specialist knowledge and interest. The objectives of these technical documents are to:

- Provide the scientific and technical basis to support the messages in the information package;
- Provide convincing arguments to experts;
- Describe all the details related to siting factors;
- Clarify the future direction for developing repository concepts tailored to potential disposal sites.

For public outreach, NUMO has been involved in a number of initiatives:

- *Public meetings* in over 30 locations, with local media hosting the meetings.
- *Round-table talks* with local politicians at these locations, with reporting by local newspapers.
- *Information campaigns* in leading newspapers, on TV, and in magazines, etc. A poster campaign was also conducted at major train stations.
- *Interactive website* for public dialogue, containing the information package, technical documents, booklets, videos and pamphlets, some of which have been produced in English, with the opportunity to submit comments.

JNC also developed a special demonstration tool named *Geofuture21* to help the general public understand the H12 assessment (see Figure 8-8). This presentation has been in operation at the JNC Tokai PA center since December 1999. In the Geofuture21, people enter a virtual repository 1,000 m underground, via a combination of scientific simulation, 3-D visualization, and a motion system. Participants are able to experience an earthquake of magnitude 7.5 on the Richter scale deep underground and compare the experience with that on the ground surface. They are also able to observe the behavior of the EBS (such as the swelling of bentonite and degradation of overpack) and to follow nuclide movement through the bentonite buffer. JAEA comments that about 90% of over 12,000 visitors have responded via a questionnaire that they were able to understand geological disposal reasonably well.

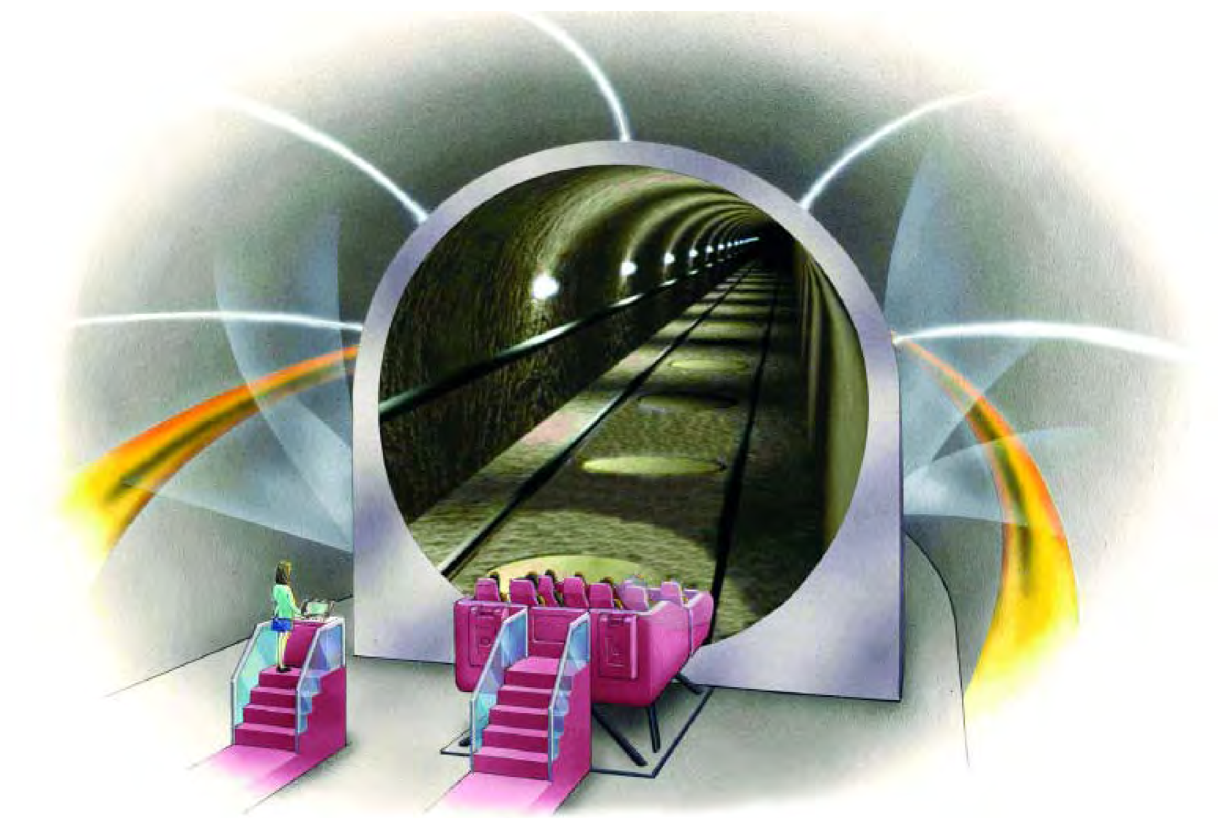

Figure 8-8
Geofuture21 virtual repository (Kitayama et al., 2006). Used with permission of NUMO.

8.5.2 International Involvement

International collaboration has played a key role in the Japanese HLW disposal program. NUMO, JNC and RWMC have international bilateral collaboration agreements with several organizations such as Nagra, SKB, and ANDRA. NUMO also joined EDRAM (International Association for Environmentally Safe Disposal of Radioactive Materials) and established an International Advisory Committee, a peer-review group of experts with specialist knowledge and expertise in subject areas relevant to NUMO's project activities. NUMO recognizes the importance of such collaboration in terms of its contributions both to the preparation of JNC's

Progress Reports such as H12 and to RWMC's R&D activities. NUMO is keen for this international collaboration to continue within the Japanese program, in order to further share experience and apply that experience (NUMO, 2009).

JAEA scientists are also gaining valuable experience by participating in foreign URL projects, and applying this experience to Japanese conditions and requirements. Such activities outside Japan provide an important knowledge base, not only for the implementer, but also for the regulator who needs to assess how the key characteristics of a site are derived and what uncertainties are associated with this process.

8.6 Safety Assessment and Licensing

8.6.1 Safety Assessment

8.6.1.1 Scenario Development

For the H3 report, PNC compiled a list of natural phenomena that could potentially affect the geological environment (PNC, 1992). These phenomena included earthquakes and faulting, volcanic activity, uplift and erosion, climate fluctuation, and sea-level change. Subsequent work involved literature surveys of these phenomena published in Japan and elsewhere.

JNC developed a systematic methodology for use in the H12 assessment. A comprehensive list of features, events, and processes (FEPs) was first developed by collating FEP lists developed in other projects, e.g., the OECD/NEA International FEP Database (1999) and Nagra (1994). The only scenarios modeled in detail in H12 are "groundwater scenarios", in which flowing groundwater provides the pathways for the transfer of radionuclides from the repository to the surface environment. These scenarios included:

- A *Base Scenario*, in which external events and processes such as natural geological and climatic phenomena, initial defects, and future human activities were excluded.
- A set of *Perturbation Scenarios*, in which the potential impacts of external events and processes were evaluated.

A Reference Case was defined for the Base Scenario, incorporating a particular set of geological characteristics, design features, model assumptions, and parameter values. The nuclide transport pathways considered in the Reference Case are illustrated in Figure 8-9. Alternative calculation cases were also defined for the Base Scenario, with alternative geological settings, design features, model assumptions, or parameter values, as well as for perturbation scenarios.

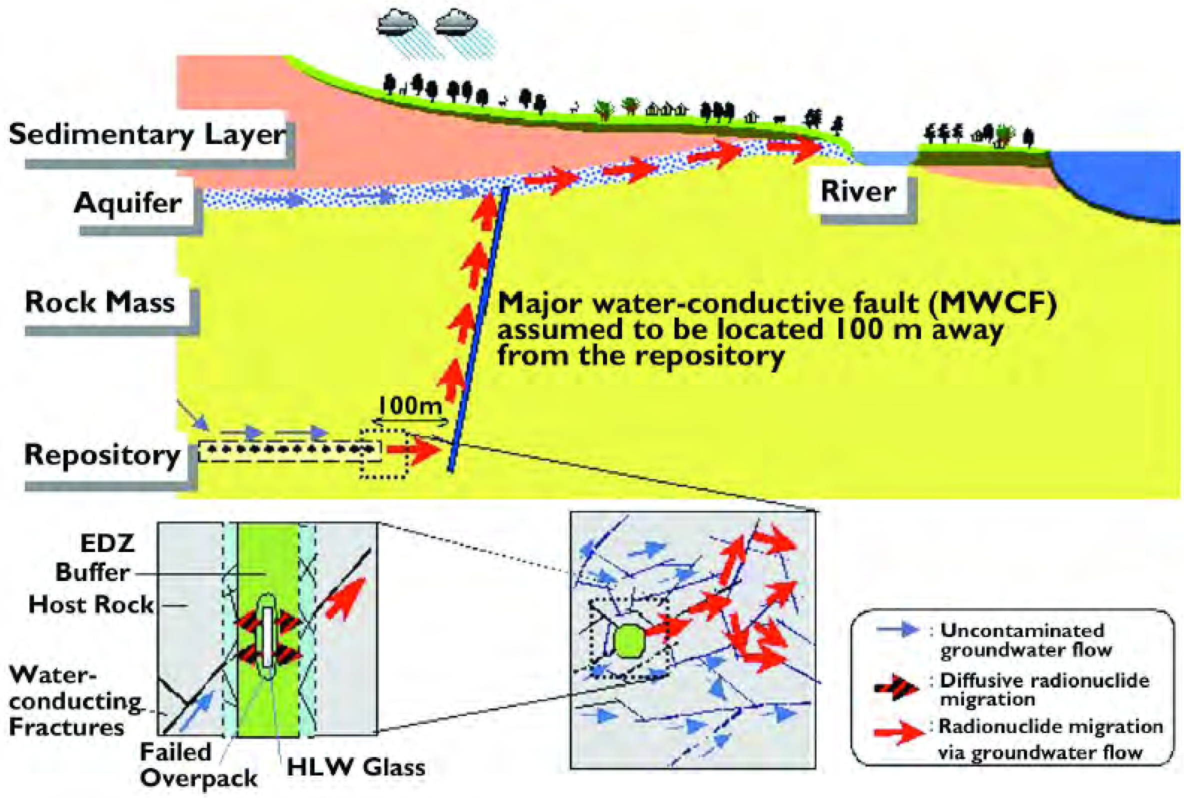

Figure 8-9
Reference Case scenario and conceptual models for groundwater pathway (Masuda and Kawata, 2001). Used with permission.

8.6.1.2 Assessment Results

The models representing different components of the repository system were integrated into a safety-assessment-model chain for the H12 assessment, which allowed analysis of the performance of the entire disposal system. Using this model chain, the Reference Case was analyzed for a repository containing 40,000 vitrified HLW packages.

A key assumption of the analysis of the Reference Case was failure of all waste packages at 1,000 years post-closure. The results for the Reference Case indicate that sufficient containment of radionuclides can be achieved by the EBS and the near-field host rock, provided that the groundwater flow rate is reasonably low (Figure 8-10). The peak annual dose shown in Figure 8-10 is ~10^{-5} mSv/year, roughly 4 orders of magnitude lower than the typical range of regulatory annual dose limits found in national regulations outside Japan (i.e., 0.1 – 0.3 mSv/year).[27] This peak dose is seen to occur almost 1 million years after closure of the repository.

[27] Currently, specific Japanese regulations for repository performance do not exist.

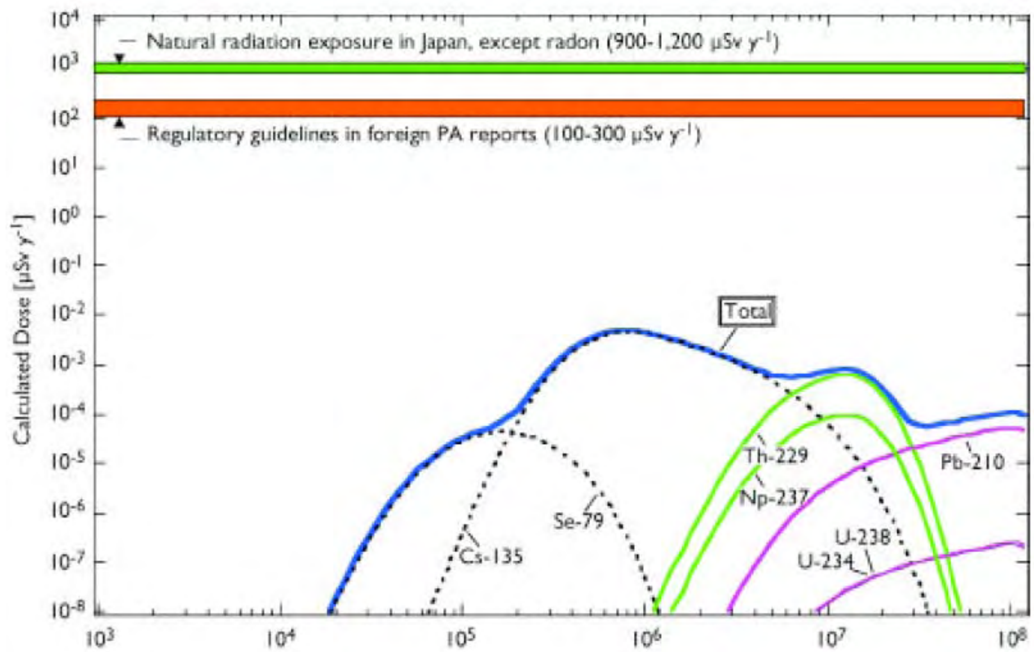

Figure 8-10
Dose results for the Reference Case: 40,000 HLW packages (Kitayama et al., 2006). Used with permission of NUMO.

Model and data uncertainty cases were analyzed for the Base Scenario by considering a wide range of alternative models and parameter values. JNC concluded that, despite remaining uncertainties at the generic stage of the R&D program, the assessment demonstrated that a geological repository would lead to negligible doses calculated to be sufficiently lower than the safety guidelines established in other countries and by international organizations.

8.6.1.3 Peer Reviews

The H12 assessment was reviewed during its preparation by external advisors, the public, and an independent international team.

The assessment documents, consisting of a Project Overview Report together with three supporting technical reports covering the three major fields relevant to the disposal system, *Geological Environment in Japan*, *Repository Design and Engineering Technology*, and *Safety Assessment of the Geological Disposal System, were* reviewed by an Advisory Committee and opened to the public in April, 1999, to solicit comments from Japanese experts in different disciplines. Also, according to AEC Guidelines, the draft H12 was reviewed independently by a group of international experts coordinated by OECD/NEA, before submission of final reports to the Japanese Government. Based on the review comments made by these reviews, JNC finalized the H12 and submitted it to the AEC in November 1999.

For the host rock and geosphere, a one-dimensional multiple-pathway model modeled transport along a set of representative channels (flowtubes), taking into account the heterogeneity of real fractures and channels by discretizing transmissivities.

A different approach was taken to biosphere modeling for the H12 assessment compared with the previous assessment. No attempt was made to model the evolution of the surface environment and the lifestyles of future generations, due to uncertainties that are inherently irreducible. Rather, certain sets of assumptions were made about these aspects of biosphere modeling, generating stylized representations of the biosphere for dose calculations (e.g., "Reference Biospheres", BIOMASS, 1999).

8.6.2 Licensing Process

The existing licensing system was established together with the safety regulatory system in 2007 with amendments to the Reactor Regulation Law. The licensing system requires the approval of the Minister of METI of the design and construction methods for the disposal facility prior to any construction work being carried out. Pre-operational inspection of the construction and properties of the facility is also required, including the inspection of welding methods. Thereafter, periodic inspections of the facility are also required during its operational phase. Current NSC guidelines also state that periodic assessments supporting the facility be carried out periodically with no longer than 20 years between assessments (NISA/METI, 2008).

8.7 Current Status

The Final Disposal Plan includes a schedule for the disposal program, which currently specifies that repository operation could start in the mid- to late-2030s (see Figure 8-11).

R&D on repository concepts and performance assessment is focusing on key issues and uncertainties identified in H12, in order to improve technical arguments and present a convincing safety case. Basic studies and experiments in two surface-based laboratories, ENTRY for mainly engineering-scale experiments and the QUALITY facility for experiments using radionuclides, continue to improve understanding of the long term behavior of the geological disposal system, as well as complement results obtained from the two URLs.

Given the importance in Japan of natural perturbation phenomena (volcanic activity, fault movement, uplift/erosion, climate/sea-level change) and their assessment, the simplified and stylized approach in H12 to assessing such scenarios has been updated by developing more realistic models that simulate the safety-relevant impact of these perturbations. This approach is intended to help guide and focus site investigation programs (NUMO, 2009).

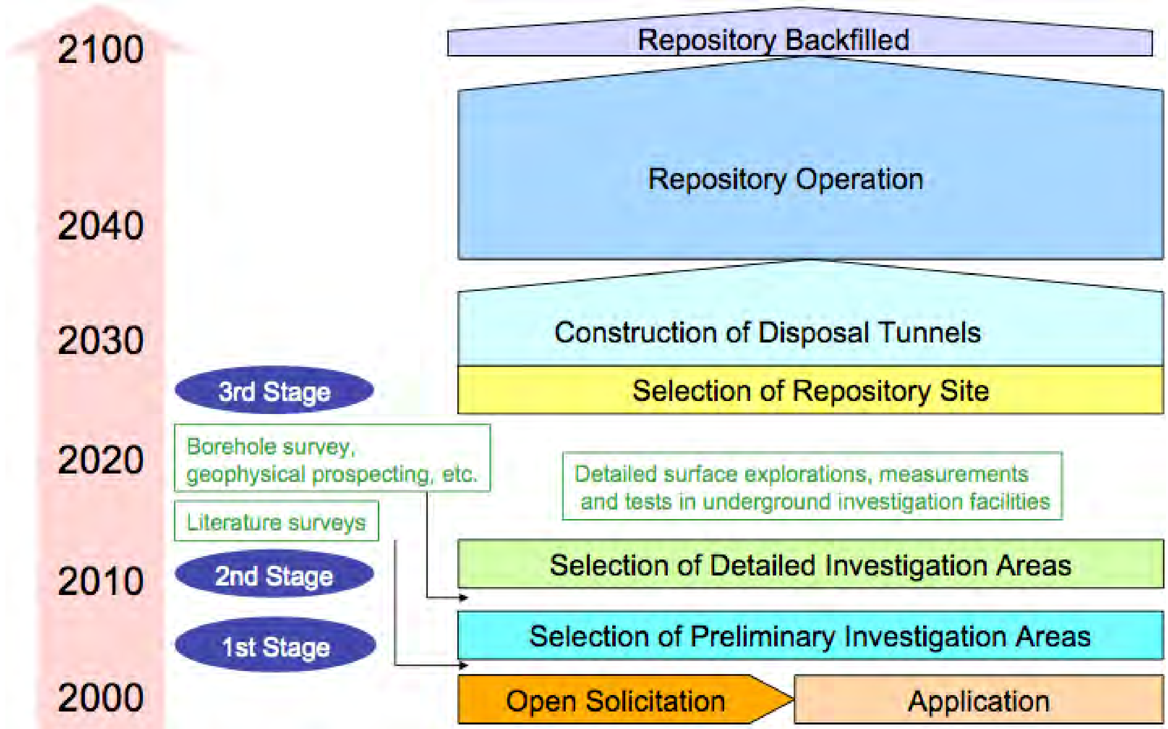

Figure 8-11
Current plans for HLW disposal program (NUMO, 2009). Used with permission of NUMO.

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

- Long-term glass dissolution;
- Diffusivities and porosities of gouge material within fractures;
- Colloid-facilitated radionuclide transport in fractured rock;
- Complexation of key radionuclides with natural organic substances.
- Development and update of key databases, in particular thermodynamic and sorption data for safety-relevant elements under relevant conditions, including hyper-alkaline and saline systems.

With regard to URL programs, the underground research projects offer opportunities for collaboration with universities and other research institutes. The URL programs consist of three phases:

- *Phase 1*: Development of techniques for characterizing the geological environment from the surface to deep underground, based on surface investigations (Phase 1). This will take into account requirements relating to the design of the disposal system and safety assessment.
- *Phase 2*: Data obtained from investigations during the previous excavation phase will serve to verify the results from the surface-based investigation (Phase 1) and characterize the evolution of the geological environment during drift excavation. Models will also be refined based on the geological data.

- *Phase 3*: Detailed investigations and experiments in the URL will help to refine geological investigation techniques that take into account the requirements of disposal system design and safety assessment. Data will also be compiled on the geological conditions to verify the reliability of models.

With regard to site selection and the volunteer approach, there have been several ‘false starts’ whereby communities volunteered initially but subsequently pulled out. In the most recent example, meetings had been held since September 2006 under the auspices of Toyo, Kochi Prefecture and other organizations, and the mayor of Toyo submitted an offer to NUMO in early 2007 to start preliminary site investigations. NUMO subsequently filed an application in accordance with the Specific Radioactive Waste Final Disposal Act, but Toyo cancelled the offer in April 2007 so that the preliminary investigations were not carried out (NISA/METI, 2008).

NUMO continues to pursue a volunteer approach to site selection in the belief that the support of local communities is essential to the success of this highly public, long-term project extending over more than a century (Figure 8-4). As discussed in Section 8.5.1, NUMO is using a variety of techniques to improve the dialogue with the public.

8.8 Summary and Key Observations

- *Policy on Geologic Disposal*: Japan’s waste management policy has included geologic disposal for vitrified HLW since 1976. Japan is currently pursuing a volunteer approach to identifying candidate sites that will be thoroughly investigated in several stages leading to planned selection of a disposal site in the 2020’s.
- *Institutional Arrangements*: NUMO is the implementer responsible for HLW disposal, reporting to METI. NUMO is supported by a number of research organizations, in particular RWMC and JAEA. JAEA is responsible for constructing and developing two URLs. The key regulatory bodies with regard to geologic disposal are JNES and NSC. NSC is responsible for issuing regulations concerning geologic disposal, and JNES provides technical support to NSC. With regard to the funding, the utilities are responsible for sharing the costs of final disposal and make contributions to a disposal fund according to the amount of electricity generated. METI is responsible for allocating NUMO’s budget and RWMC manages the national fund. A recent estimate of the disposal program is ~3 trillion yen. An additional fee of 0.07 yen/kW-hour is charged to utilities to cover waste management activities prior to the national fund being established
- *Key Laws and Regulations*: No specific regulations for HLW disposal exist in Japan, only guidelines. NSC is conducting Workshops to solicit public opinions, and is responsible for developing safety requirements for geologic disposal. In the absence of regulations, existing AEC Guidelines identify annual dose as the basic indicator of safety as well as stressing the importance of taking into account international (ICRP) standards.
- *Site Screening and Selection*: Site selection is a greater challenge in Japan than in most other countries, because of its active tectonic setting at the juncture of multiple convergent plate margins. At the end of a general geologic review period leading to the submission of a Progress Report in 1992, AEC had hoped for a wide range of sites to be identified covering most of the geological features in Japan, but this did not happen. More recently, in keeping with the 2000 Act, which specifies a three-stage siting process, Japan’s current strategy is a

volunteer approach, whereby communities volunteer PIAs as potential candidate sites. Three stages are envisioned, an initial literature survey stage, followed by a preliminary investigation area (PIA) stage, and a final detailed investigation area (DIA) stage leading towards the selection of a site in the 2020's. To aid the first step in the siting process, NUMO announced an overall procedure for selecting potential candidate sites, followed by the identification of siting factors to be provided to all municipalities in Japan. To date, no community has volunteered. Meanwhile, research continues in two URLs, one in crystalline rock (Mizunami) and the other in sedimentary rock (Horonobe). Despite the siting challenges posed by the volcanic and tectonic setting of Japan, NUMO is confident of identifying one or more sites suitable for geologic disposal.

- *Repository Design Concepts*: Japan's repository design concept favors the multi-barrier approach. In the absence of a specific site, the disposal concept is similar to that of Switzerland's disposal concept for granite host rock: carbon steel overpack for vitrified HLW, surrounded by bentonite-sand buffer.

- *Performance Metrics and Assessments*: JNC (the responsible organization prior to JAEA) carried out a preliminary assessment (H12 assessment) of the feasibility of geologic disposal in Japan. In the absence of a specific site, a granitic environment was assumed. Due to the lack of regulations / standards governing performance of geologic repository, JNC took note of regulatory annual dose limits found in national regulations outside Japan. The assessment methodology followed a scenario approach and the main assumption for the analysis of the Reference Case was failure of all waste packages (40,000) at 1,000 years post-closure. The results for the Reference Case indicate that radionuclides can be contained adequately by the EBS and near-field host rock, provided that the groundwater flow rate is reasonably low. The peak annual dose obtained was ~10^{-5} mSv/year, about 4 orders of magnitude lower than annual dose limits outside Japan. This peak dose occurred almost 1 million years after closure of the repository.

- *Independent Peer-Review and Advisory Bodies*: According to AEC Guidelines, the H12 study was reviewed independently by a group of international experts coordinated by OECD/NEA, before submission of final reports to the Japanese Government. The H12 assessment was also reviewed during its preparation by external advisors. JNC took into account review comments before finalizing the H12 report and submitting it to the authority in November 1999.

- *Stakeholder and Public Involvement*: Since the 2000 Act, the Basic Policy requires that at every stage of the site selection process, NUMO must solicit the opinions of local residents, and METI must solicit the opinions of governors and mayors, and that these opinions be documented and addressed. To aid the volunteer siting process, NUMO distributed an information package to all (3,239) municipalities in Japan, which contained among other things a description of the benefits to volunteer municipalities, not only from a financial perspective, but also with respect to other positive social aspects. NUMO has followed up this initial contact with an active public communications program including public meetings, round-table talks with local politicians and an interactive website.

- *Program Maturity*: Japan has been actively seeking a site for geologic disposal since 1976. While a wealth of information has been collected on different regions throughout Japan, no sites have been selected, although considerable research and development is continuing on repository concepts, knowledge management, and site characterization methods at URLs.

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9
SPAIN

9.1 Introduction

9.1.1 General Nuclear Profile

Spain's eight operating commercial nuclear reactors generated 51 TWh in 2009, representing 18% of the electricity supply (WNA, 2010a). The Spanish nuclear fleet comprises six PWRs and two BWRs located on six different plant sites.The plants are owned and operated by the companies Endesa and Iberdrola, with ENUSA, a state-owned company, in charge of front-end activities (WNA, 2010b).

Spain's national policy since 1983 has been to maintain open fuel cycle with no reprocessing; previously, fuel from Vandellós I, a gas-cooled reactor undergoing decommissioning, was sent to France for reprocessing. A final decision on disposal has not been made (NEA, 2005). Spain does not believe that any characteristics different from those of other nuclear facilities are necessary for locating an interim storage site apart from a suitable geographic location and suitable geological characteristics, so that several sites are possible (MITYC, 2008).

9.1.2 Used Fuel / HLW Inventory and Projected Disposal Requirements

Estimates of used fuel and HLW inventories eventually requiring disposal total to 12,800 m^3, including approximately 6750 MTHM of used fuel and the 12 m^3 of vitrified waste from the reprocessing of Vandellós I used fuel (NEA, 2005; WNA, 2010b). Irradiated nuclear fuel is currently managed primarily in wet storage at the reactor sites. At the Trillo plant, interim dry storage of irradiated nuclear fuel has been implemented due to spent fuel pool storage limitations. Radioactive waste management plans assume 40-year reactor lifetimes. In 2006, Parliament approved plans for a centralized dry storage (the Almacen Temporal Centralizado) facility for 6700 MTHM of used nuclear fuel and the vitrified HLW from reprocessing of Vandellós I fuel. Plans are to be developed by 2010, with facility siting to be done on a voluntary basis.

9.2 Institutional Arrangements

Submission by the implementer of a General Radioactive Waste Plan for approval by the government, provides the framework for radioactive waste management policy for the next 10 years approximately, at which a new Plan is submitted for approval. The most recent Waste Plan identifies geologic disposal as the ultimate goal for the management of used nuclear fuel, although site-specific studies are not currently being pursued.

9.2.1 Institutional Framework

Figure 9-1 shows the main organizations involved in radioactive waste management in Spain. The key roles are described below.

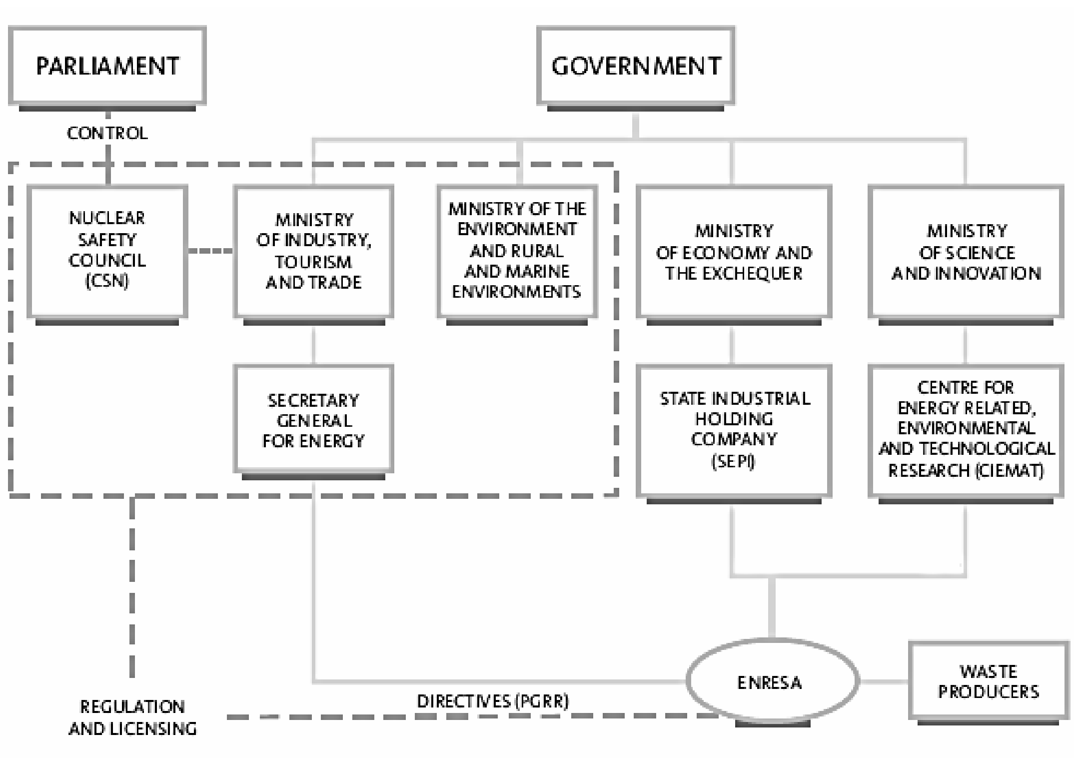

Figure 9-1
Institutional framework for radioactive waste management and licensing of nuclear facilities in Spain (adapted from MITYC, 2008).

POLICY and OVERSIGHT - The *Ministry of Industry, Tourism and Trade (MITYC)* issues authorizations for the construction and operation of nuclear facilities. As shown in Figure 9-1, MITYC oversees the General Secretary for Energy as well as the Director General for Energy Policy and Mines (not shown). MITYC is also responsible for approving EIS's.

IMPLEMENTER - *Empresa Nacional de Residuos Radiactivos* (*ENRESA*) was established in 1984 as the implementer of radioactive waste management programs in Spain. ENRESA is a state-owned company whose shareholders are CIEMAT (80%) and SEPI (20%), both government agencies. ENRESA operates as a management company, its main role being to develop a reference framework for national spent fuel and radioactive waste management strategies. The agency achieves this role by periodically submitting for approval by the Spanish Government a proposal for a General Radioactive Waste Plan (in Spanish, *Plan General de Residuos Radiactivos*, *PGRR*).

ENRESA's 5th PGRR (ENRESA, 2001), which established HLW research and development plans for most of the 2000's, laid out the waste management options for irradiated nuclear fuel and HLW that were under consideration in Spain (Figure 9-2), *viz.*

- Extended storage via a Centralized Temporary Storage (CTS) facility;
- Separation and transmutation; and
- Geological disposal.

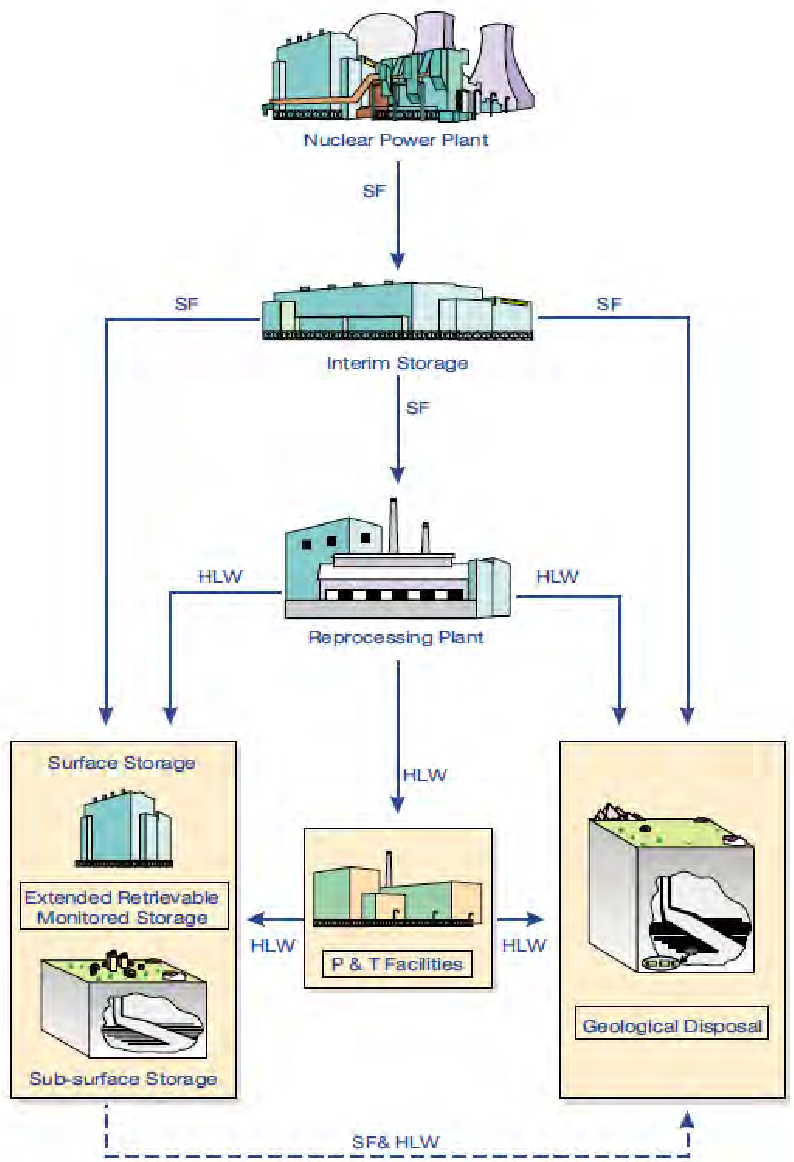

Figure 9-2
Waste management options for irradiated nuclear fuel and HLW under consideration in Spain (ENRESA, 2001). Used with permission of ENRESA.

Table 9-1 contains a summary of the key issues identified by ENRESA for these three waste management options. As shown in this table, ENRESA acknowledges that extended storage is the easiest option to promote politically, although it does not provide a final solution. ENRESA also noted that flexibility regarding geological disposal is a key issue, and that some degree of reversibility should be incorporated in the planning and development of a repository.

In addition to the issues identified in Table 9-1, ENRESA also recommended a stepwise approach to repository development and, in this context, viewed a URL as a key component. This 5th PGRR stated that no decision would be made on the final disposition of irradiated nuclear fuel and HLW before 2010, but that ENRESA's work program would include:

- Integrated studies on non site-specific deep geological repositories, one each in granite, clay and salt formations;
- Compilation of R&D results and geological information from the former Site Selection Plan.

REGULATOR – The Ministry of Economy (MINECO) is responsible for enforcing nuclear legislation and for granting licenses. *The Nuclear Safety Council (CSN)* is the regulatory authority for nuclear safety and radiation protection. The Council is independent of the Government and reports on its activities directly to Parliament. CSN issues licenses for nuclear facilities.

ADVISORY and SUPPORT – The key technical organization supporting ENRESA, as well as being a shareholder, is the Center for Energy-Related, Environmental and Technological Research (CIEMAT), which itself reports to the Ministry of Science and Innovation.

Table 9-1
Summary table of options for irradiated nuclear fuel and HLW management in Spain (adapted from ENRESA, 2001).

Issue	Extended Storage	Geological Disposal	Partitioning & Transmutation
Ethics	Options for dealing with SF and HLW are not foreclosed, but it passes responsibility for real action to future generations, which are burdened with societal, technological and financial risks.	The generation that benefited from the activities that produced the SF and HLW bears the financial and political costs and provides future generations with a solution.	A further step in the back-end of the nuclear fuel cycle to be developed and implemented by future generations.
Technology	Technology is available. No major showstoppers are envisaged.	Technology is being developed. Extensive international research programs and cooperation. No major showstoppers are envisaged.	The technology is at an early stage of development. It faces great technological challenges.
Safety	Safety requires active control and human care of the storage facilities. Long-term safety is vitally dependent on human actions.	Low doses predicted very far into the future. The uncertainties in projecting long-term performance and the prediction of future events are two major issues.	It would entail a further reduction of a relatively small, perhaps only hypothetical, risk to workers far into the future. The exposure risk in the present or in the near future would increase appreciably due to the complexity of the fuel cycle.
Safeguards	Safeguards are not an issue in the short or medium-term.	Disposal prevents the diversion of nuclear materials for harmful purposes because it is arduous, costly and easily discernible.	Increased risk of spreading technology that could be use for production of nuclear weapons material.
Society	The easiest decision to be taken from the political and sociological points of view.	Loss of political will and momentum. Increase of public scrutiny resulting in delays in repository development.	A political commitment with the continued use of energy is demanded. Public acceptance is more than questionable by the required additional nuclear facilities.
Economy	Favored by short-term financial pressures. It does not avoid the cost of repository development and creates financial uncertainties.	Monetary and material resources to implement geologic disposal are available today.	A long-term venture, which faces institutional challenges and requires long lead times and large investments.
Timing	It is not per se a long-term definitive solution and it does not assure any future solution to SF and HLW management	A convergence is observed on: • a stepwise process interspersed with decision points for technical, regulatory, policy and public review; and • a flexible approach to facilitate the reversibility in repository planning and implementation.	Current unanimity in considering that: • it is not a short-term alternative but it could be incorporated into future cycles in a scenario with continued use of nuclear energy; • it might reduce the volume and the radiotoxic inventory of the HLW to be managed; and • it would not avoid geologic disposal.

9.2.2 Legal and Regulatory framework

The main general nuclear law in Spain is the Nuclear Energy Act (Spanish acronym LEN), complemented by supporting laws, regulations and Ministerial Orders on specific aspects within the LEN.

The licensing process for nuclear and radioactive waste management and disposal facilities is governed by the *Regulation on Nuclear and Radioactive Facilities* (Spanish acronym RINR).

Other relevant legislation includes:

- *Law 33/2007* creating CSN and guaranteeing the organization's independence.
- *Royal Decree 35/2008*, concerning the operational requirements for nuclear facilities.
- *Instruction IS-08* (2005), on criteria concerning radiological protection.

CSN has also issued some relevant Instructions relevant to waste management:

- *Instruction IS-16* (2008) concerning the time periods over which documents and records of nuclear facilities must remain filed.

Spain's Regulation on Protection against Ionizing Radiations (Spanish acronym RPSRI) is the Spanish version of EU Directive 96/29 EURATOM concerning the general principles (justification, optimization, limitation) and standards associated with the protection of workers and the members of the public against the risks deriving from ionizing radiations. RPSRI established an annual effective dose limit for members of the public of *1 mSv per year*. A higher annual dose may be authorized as long as the five-year annual average does not exceed 1 mSv/year.

More relevant, with regard to radioactive waste disposal facilities, CSN established the following criterion (CSN, 2008):

> *The level of individual risk used shall be lower than 10^{-6}/year, or the risk associated with an annual equivalent dose to individuals in the critical group lower than 0.1 mSv.*

No timeframe associated with this criterion is specified.

9.2.3 Waste Classification

Spain's classification of radioactive waste follows closely the IAEA Classification scheme (IAEA, 2006), with LL-LILW and HLW the relevant categories, both unsuitable for surface disposal.

9.2.4 Funding

With regard to funding, a contractual relationship exists between ENRESA and Spain's radioactive waste producers, which must be approved by MITYC. This relationship was formalized in a 1984 Decree that established ENRESA as waste management company. Financing is based on payments being made into an interest-bearing fund, these payments being in the form of a levy (on the order of 1% of cost) on all sales of electricity, i.e. during the operational life of commercial reactors (WNA, 2010b). The size of the levy is calculated annually, taking into account revised projected costs for the back-end of the nuclear cycle and how much funds remain at the time of review. For smaller radioactive waste generators, there is a tariff for services rendered by ENRESA set by the Government (MITYC, 2008).

9.3 Geological Studies for Deep Disposal

9.3.1 Early Studies

Studies started in the mid-1980's towards identifying the availability of sites (formations) for the deep disposal of irradiated nuclear fuel and HLW. Detailed analysis of potential repository sites in Spain was carried out between 1986 and 1998 under a Site Selection Plan (SSP). Sequential analysis was performed at different scales in the SSP (del Olmo, 1996), *viz.*

- Identification of potentially suitable formations in the whole of Spain, documented in the *Favorable Formations Inventory* (IFA) at a scale of 1:500,000, and also included in the European Catalogue of Favorable Formations.
- From the IFA information, high-level regional screening studies were carried out, in which 150,000 km^2 areas were evaluated at a scale of 1:200,000, resulting in the *Inventory of Regional Areas* (ERA).
- Even more detailed studies were undertaken covering 22,000 km^2 areas at the 1:50,000 scale, identifying more than one thousand municipalities with the potential for deep repository construction.

The geological and socio-economic characteristics of the region were evaluated, applying (at each scale) several feasibility and suitability criteria and taking into account all existing underground and surface geological information, as well as new information obtained specifically for the SSP.

As a result of these studies, the SPP identified several candidate formations for disposal, the major types of formation being granite and clay, with some, more limited, salt formations also being identified. Beyond the identification of potential formations, several locations were identified as being potentially suitable. Although ENRESA's 3rd and 4th PGRRs identified several sites for detailed characterization in the period from 2000 to 2010, no further geological studies towards geologic disposal were authorized after the 5th PGRR was issued. The government required that existing geological data be used as input for PA calculations of a deep geological repository.

The characterization of clays in terms of favorable characteristics generally followed those already accepted by the international community: mineralogical purity, retention properties, plasticity, low permeability, high swelling pressure and thermal conductivity (Villar et al., 2006). Initial studies resulted in the selection of deposits of clay (smectites) from the Cabo de Gata region (Almería) and the Tertiary Basin of Madrid (Toledo). Detailed geological characterization of the formations was carried out via examination of borehole material by several laboratories. The Cortijo de Archidona deposit (Almería) was finally selected for detailed evaluations and the bentonite taken there has been the object of various research projects that resulted in this clay being one of the most well characterized in terms of mineralogical, thermal, hydraulic, mechanical, geochemical and alterability properties. The behavior of the bentonite was also studied under repository-type conditions at laboratory and natural scales, and the long-term evolution of the barrier is being analyzed by natural analogues studies in the Cabo de Gata area (Villar et al., 2006).

9.3.2 URL Programs / Experience

Within Spain, Spanish researchers carried out substantial work at the El Berrocal uranium mine (uranium-bearing quartz vein in a Paleozoic two-mica granite) as part of ENRESA's R&D program. The deposit had been mined from an open pit until the late 1960s. After the mine was abandoned, an underground gallery was excavated to serve as a research laboratory (Reyes et al., 1998). The mine is located about 90 km SW of Madrid.

The granite/U-bearing quartz vein system at El Berrocal has been studied as a natural analogue of a HLW repository, the main objectives being to understand and model the migration processes which have controlled the distribution of naturally occurring radionuclides in a fractured granitic environment. For example, carbonation processes were studied from a mineralogical and isotopic ($^{87}Sr/^{86}Sr$ and $\delta^{13}C$) viewpoint, given that carbonate is a strong complexing agent for U(VI) under both low-temperature hydrothermal and environmental conditions (Reyes et al., 1998). The El Berrocal Project was an international study, with several countries represented.

Elsewhere, substantial R&D work has been done sharing facilities in other countries, in particular in Canada, France, and Switzerland, e.g., the FEBEX (Full-Scale Engineered Barriers Experiment) experiment carried out at Grimsel, Switzerland and subsequently a similar emplacement experiment in clay (ENRESA, 2005). Spanish researchers have been involved in both granite and clay URLs. Astudillo (2001) summarizes Spain's involvement internationally with URLs.

For example, the FEBEX *in situ* test at Grimsel consisted of a full-scale simulation of a HLW disposal facility, emulating ENRESA's reference concept for granite (Astudillo, 2006; see Figure 9-3). The test comprised two electrical heaters, of dimensions and weight equivalent to those of HLW canisters, in a 2.28 m diameter drift specifically selected and excavated in granite. The entire space surrounding the heaters was filled with blocks of compacted bentonite to complete the 17.4 m thick barrier for the test section. The test zone was closed with a concrete plug.

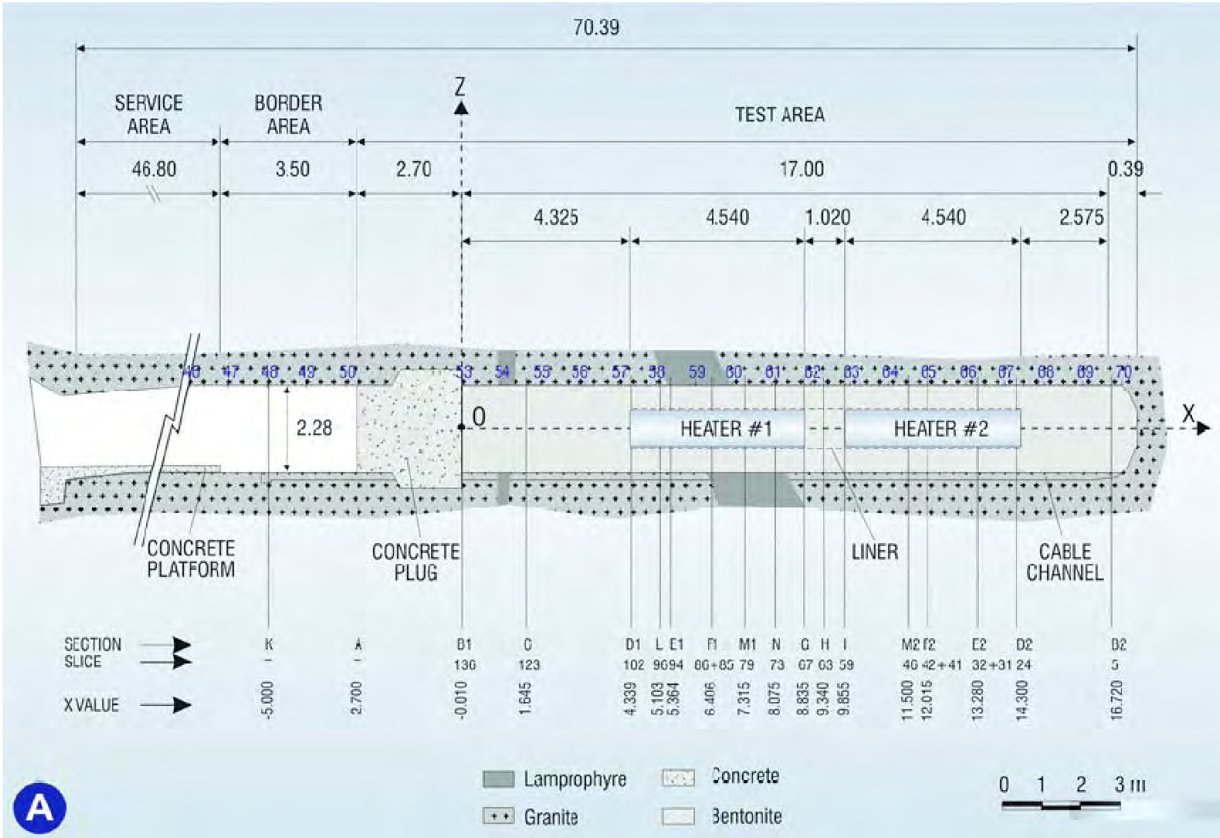

Figure 9-3
Layout of FEBEX experiment at the Grimsel granite URL, Switzerland (Astudillo, 2006). Used with permission of ENRESA.

Hydration of the buffer progressed as expected although the rate of hydration became slower than initially predicted by THM models after approximately 3 years. The EC-funded international project NF-PRO was able to extend the *in-situ* test for an additional three years, which helped to confirm previous observations as well as provide a comparison with a computational model over a longer period of time.

Dismantling started in 2002 (FEBEX-II) with the extraction of the first heater, together with a systematic and exhaustive sampling and *in situ* analysis of the surrounding bentonite. The FEBEX test was able to verify most of the hypothesis on the THM processes in the transient phase of the clay barrier, especially in the presence of water vapor (Astudillo, 2006).

9.4 Disposal Concept

9.4.1 Background

Studies on backfilling and sealing materials started as early as 1987 with an EC-funded project (Mingarro et al., 1991). The reference concept developed by ENRESA, conforms to the multi-barrier principle but exists only at a fundamental level. A generic-type reference conceptual design was first developed for a repository for irradiated nuclear fuel in a granite formation (ENRESA, 1994), in which solute concentrations in groundwater are generally low. The original concept comprised:

- Disposal at a depth of about 500 m of irradiated nuclear fuel in canisters having a corrosion allowance material (carbon steel) with a service life of at least 1,000 years;
- Buffer consisting of blocks of compacted bentonite surrounding the steel canisters;
- Horizontal galleries excavated by mechanical methods, i.e. tunnel boring machine.

The conceptual design has evolved since 1994 aiming for compatibility with repository concepts in clay and salt. Convergence with regard to design was achieved in the repository conceptual design published in 1996. The main features in the context of granite formations are:

9.4.2 Repository Layout in Granite

Details of ENRESA's repository lay out in granite (Figure 9-4):

- Depth 500 m, where ambient temperature is 30.5 ºC.
- Disposal drift length: maximum of 500 m;
- Drift diameter: 2.4 m;
- Carbon steel canister diameter: 0.90 m;
- Carbon steel canister wall thickness: 0.10 m;
- Canister waste contents: 4 PWR elements or 12 BWR elements;
- Heat output per canister (at deposition time / 50 years cooling): 1,200 W;
- Buffer characteristics: Ca-Na bentonite of Spanish origin (Serrata de Nijar);
- Thickness: 0.725 m;
- Dry density: 1.6 tm/m^3 (equivalent to 2.00 tm/m^3 bulk density);
- Initial saturation degree: 55%;
- Swelling pressure (at full saturation): 5 MPa.

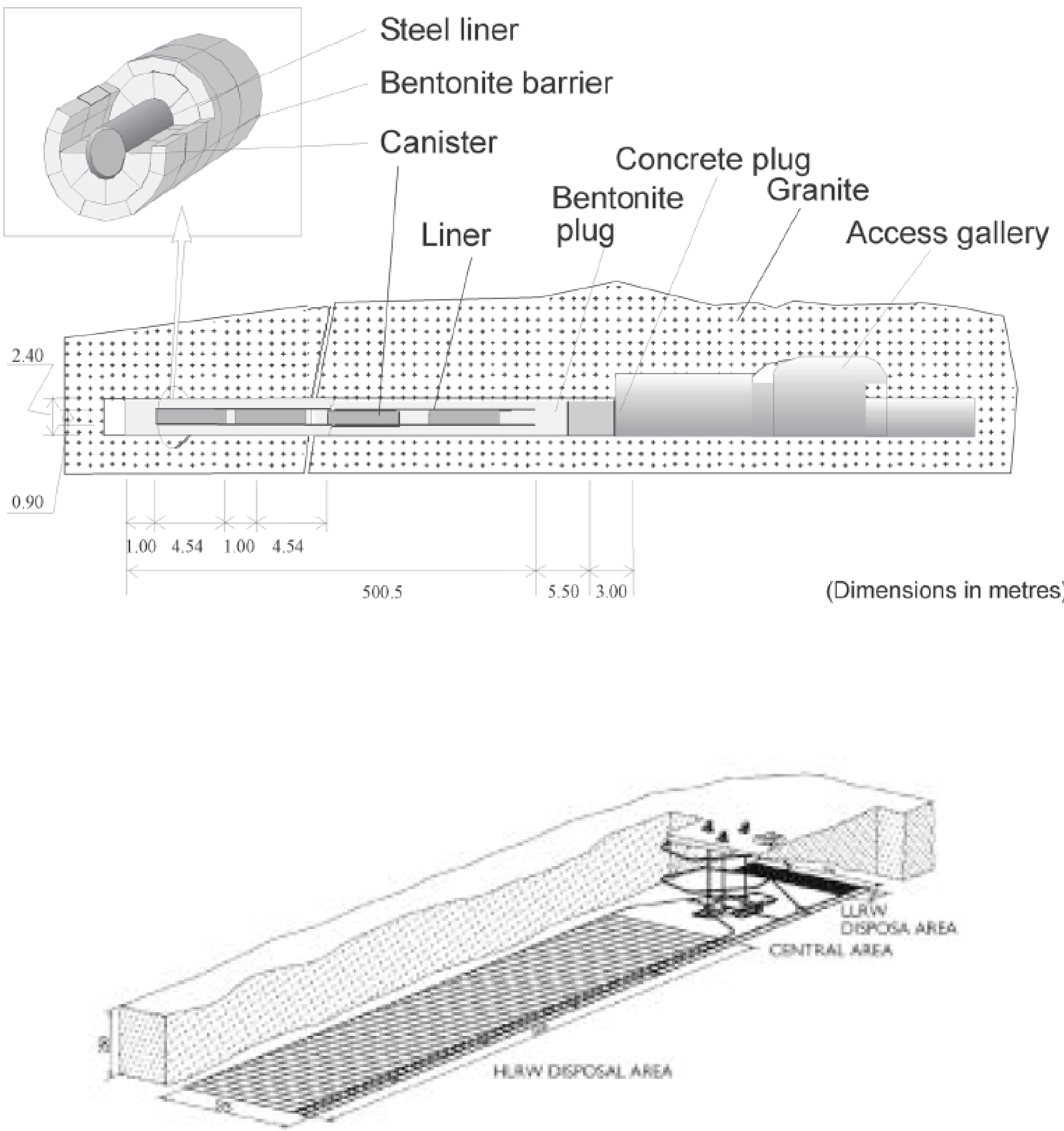

Figure 9-4
Repository layouts for granite (upper) and clay (lower) (ENRESA, 1995; Astudillo, 2001). Used with permission of ENRESA.

Based on thermal calculations, the distance between deposition drifts was determined at 35 m and the distance between canisters was 1 m (5.54 m between canisters). ENRESA's PA exercise used an updated thermal output of 1,220 W per canister, increased from 1,200 W after reassessment of the characteristics of Spanish irradiated fuel. This reference design formed the basis of the FEBEX mock-up experiment.

The deposition method (granite) involves automatic insertion of individual canisters into a carbon steel liner placed in the inner part of the buffer. The canister separation distance is based on a thermal constraint of 100 ºC at any point in the buffer.

Disposal drifts are sealed with a bentonite plug 5.5 m long with the same characteristics as the buffer, followed by a concrete plug, 3 m long. The backfill comprises 90% sand and 10% bentonite.

In 2004, ENRESA conducted a review of its EBS conceptual design, reaching a number of design changes (see Figure 9-5):

- Deposition drift diameter reduced to 2.0 m;
- Buffer thickness reduced from 0.725 m to 0.55 m (PA sensitivity analysis indicated that a reduction to 0.41 m would provide similar performance to 0.725 m);
- Steel sleeve aimed at facilitating canister deposition no longer required (as a result of the FEBEX experiments).

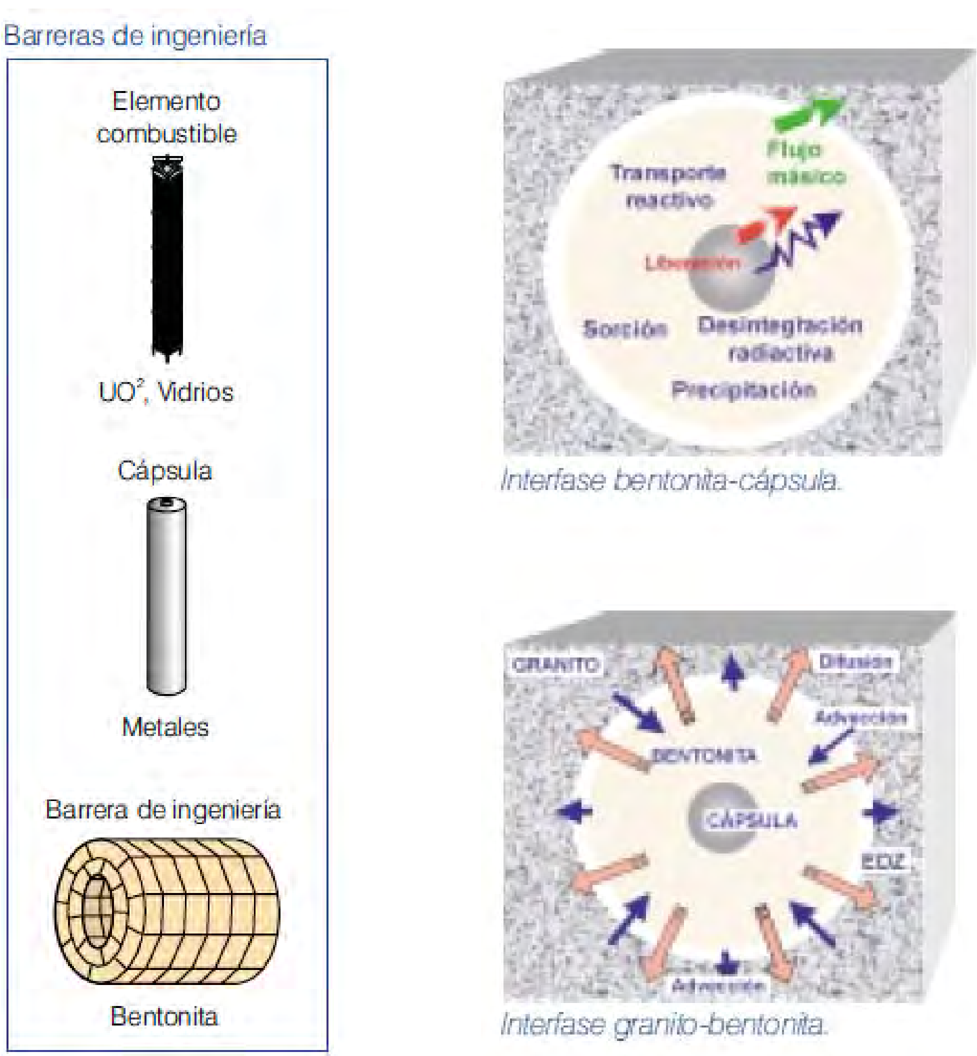

Figure 9-5
Conceptual EBS design for repository in granite (ENRESA, 2004). Used with permission of ENRESA.

Additional buffer modifications were (ENRESA, 2004):

- Increase in size of bentonite blocks to facilitate disposal cell construction, aiming for 3 or 4 blocks per section, cf. reference design of 16 blocks and 36 blocks per section used in FEBEX;
- Increase in initial humidity of bentonite saturation (to ~80%) to improve thermal performance of the buffer;
- Reduction in dry density of buffer, from 1.6 tm/m^3 to 1.5 tm/m^3 (total density of ~1.9 tm/m^3 under saturated conditions), to limit the impact of near-field stresses caused by pressure build-up from gas generation and increase in volume of steel corrosion products;
- New specification for thermal constraint – 100 ºC to the external 0.41 m of buffer thickness (reduced conservatism);
- Minimum separation between canisters of 0.5 m to allow easy access during construction of disposal cells.

9.5 Transparency and Stakeholder Involvement

9.5.1 Public Involvement

Because Spain is focusing more on interim storage in the short-term rather than geological disposal, there is little evidence of public involvement in any siting process associated with geological disposal. More generally, with regard to nuclear issues, once ENRESA has submitted its periodic General Radioactive Waste Plan to MITYC, stakeholders, including the public, are requested to comment.

Both the RINR (Regulation on Nuclear and Radioactive Facilities) and a 2006 Law (9/2006) on the Assessment of the Environmental Effects of Certain Plans and Programs require public involvement in the licensing process, which involves access to information relevant to the preliminary authorization of the facility. The 2006 Law specifically promotes transparency and public participation.

A more general Law (38/1995) recognizes the right of any person to access information on the environment in the hands of the public Administrations, as well as the obligation of the latter to make available such information.

The Spanish government initiated a Panel of Dialogue on Nuclear Energy in 2006, organized by MITYC and involving a wide range of stakeholders, including NGOs. The meetings held under the Panel of Dialogue framework emphasized that the mechanism for selecting a site for the CTS facility involves a democratic process, requiring the voluntary participation of the local communities as well as their involvement in decisions affecting them significantly. Once a candidate municipality has been identified, the general licensing process begins, involving EIA considerations and associated public hearings.

9.5.2 International Involvement

In addition to ENRESA's El Berrocal mine URL and its participation in other URL programs, ENRESA participates in an International Association for Environmentally Safe Disposal of Radioactive Materials (EDRAM). The other nine countries and organizations involved are Belgium (ONDRAF/ NIRAS), Canada (Ontario Power Generation), Finland (POSIVA OY), France (ANDRA), Germany (DBE) (BFS), Sweden (SKB), Switzerland (Nagra), United Kingdom (NIREX), and the United States of America (DOE-OCRWM).

9.6 Safety Assessment and Licensing

9.6.1 Safety Assessment

Preliminary non-site specific evaluation exercises were carried out in the 1990's using generic data compiled during the Site Selection Plan process (Astudillo and Alonso, 2004):

- ENRESA 97 (granite); and
- ENRESA 98 (clay).

A more detailed but still non-site specific performance assessment of disposal in granite was completed at the end of 2001 (ENRESA, 2001) and a similar exercise was completed for clay in 2004. A first revision of an integrated study for deep geological disposal in granite was completed at the end of 2003 (AGP Basico Granito, 2003). These documents are not available in the public domain.

As part of the assessment process and EC-funded projects, ENRESA developed a structured top-down FEP database, initially for the bentonite buffer, but later expanded to cover the entire near-field.

The main cause of canister failure was assumed to be either penetration of the canister wall due to localized corrosion, or mechanical collapse after the canister wall thickness has been reduced by general corrosion. The latter is assumed to occur conservatively once the thickness of steel reaches 4.25 cm. Average failure time was assessed to occur at least 70,000 years after closure, with a less-conservative estimate of >100,000 years (as a result of localized corrosion) or 500,000 years (as a result of general corrosion).

For radionuclide migration modeling, radionuclide concentrations in natural groundwater were assumed to be zero and releases from individual canisters were considered independent.

9.6.2 Licensing Process

As stated previously (Section 9.2.1; see also Figure 9-2), MITYC is responsible for granting authorizations and does so once CSN has approved the submissions of the licensee and submitted its mandatory report. In common with the licensing of all nuclear facilities, a Preliminary Safety

Analysis report is one of the documents that must be submitted for review by CSN and it should include specific items:

- Description of facility with relevant safety-related criteria addressed in the facility's design;
- Analysis of potential accidents and the resultant consequences;
- Radiological analysis combined with potential environmental impacts;
- Design of monitoring program to address potential impacts;
- Evidence of construction QA program.

9.7 Current Status

The 6th PGRR (ENRESA, 2006), approved by the Government's Cabinet of Ministers in June 2006, highlighted (a) the current strategy of 'temporary' management of irradiated nuclear fuel and HLW and, in this context, (b) the need to identify a site for the construction of a vault-type CTS facility. Such a facility, intended for the storage of used nuclear fuel, HLW and long-lived L/ILW, is expected to be operational by 2012.

The CTS-based strategy was proposed to the Government by unanimous resolution of the Congressional Commission for Industry, consisting of representatives of all Spain's parliamentary groups, in December 2004. An Inter-ministerial Commission subsequently drew up the technical and legal requirements that are necessary to proceed with identifying a site for the CTS facility. In this context, the allocation of funds envisaged for volunteer municipalities surrounding such facility has been established by Ministerial Order of July 13th, 1998[28]. According to this, towns surrounding a CTS will receive an annual allocation made up of both fixed and variable portions (MITYC, 2008). Additionally, MITYC envisages that a CTS will create highly-qualified workforce and stimulate the local economy.

With regard to geologic disposal, the 6th PGRR envisages (ENRESA, 2006):

- *2025-2040*: Characterization of a disposal site;
- *2041-2050*: Construction of disposal facility.
- *2050*: Operational phase of disposal facility.

The starting date for each of the above stages will act as decision-making points. ENRESA emphasizes, however, that the above schedule is primarily for economic and planning purposes. The above dates represent a delay of 15 years relative to the corresponding dates provided in the 5th PGRR.

Thus, the focus of Spain's current efforts is on the finalization and construction of a CTS facility for irradiated nuclear fuel. In terms of preventing undue burden on future generations, MITYC has not addressed this issue in the context of geological disposal, but rather existing facilities and extended storage therein.

[28] Official State Gazette 17-7-98.

Non site-specific conceptual repository designs have been developed for a geologic repository in granite, clay and salt, which ENRESA will now update / modify to introduce the capability of retrievability of emplaced waste.

ENRESA notes that future safety assessments of geologic repositories will combine geological information, repository design and the results of R&D. These assessments will provide quantitative information about the evolution of the repository for guiding R&D activities as well as optimizing facility design. These studies will also consider the possible effects of the new fuel cycle technologies associated with separation and transmutation of long-lived radionuclides.

In future PGRR's, MITYC expects ENRESA to report on updated management options, the feasibility of any new technologies, as well as generic projects summarizing the knowledge acquired in relation to geologic disposal.

9.8 Summary and Key Observations

- *Policy on Geologic Disposal*: Geologic disposal is seen in Spain as the final waste management option for used nuclear fuel and HLW, although the most recent policy plan stated that no decision would be made on the final disposition of irradiated nuclear fuel and HLW before 2010. The bulk of the current effort is being devoted to developing a centralized facility for the storage of irradiated fuel.
- *Institutional Arrangements*: ENRESA, a state-owned company, is the implementer in Spain, reporting to the Minister of Energy. ENRESA's main role is to develop a reference framework for national spent fuel and radioactive waste management strategies. CSN responsible for nuclear safety and radiation protection. CSN is independent of the Government and reports on its activities directly to Parliament. With regard to funding, ENRESA's financing is based on payments made to an interest-bearing fund, these payments being in the form of a levy on all sales of electricity, i.e. during the operational life of commercial reactors. The size of the levy is calculated annually, taking into account revised projected costs for the back-end of the nuclear cycle and how much funds remain at the time of review.
- *Key Laws and Regulations*: The licensing process for nuclear and radioactive waste management and disposal facilities is governed by the Regulation on Nuclear and Radioactive Facilities. Other relevant laws include Royal Decree 35/2008, concerning the operational requirements for nuclear facilities, and Instruction IS-08 issued by CSN on criteria concerning radiological protection.
- *Site Screening and Selection*: Detailed analysis of potential repository sites in Spain was carried out between 1986 and 1998 under a Site Selection Plan that identified the sequential analysis to be performed at different scales, from the identification of suitable geologic formations, to screening studies at the regional level, down to detailed studies at a more local scale. Although ENRESA's 3rd and 4th research plans identified several sites for detailed characterization in the period from 2000 to 2010, no further geological studies towards geologic disposal were authorized after the 5th plan was issued in 2001. These initial studies identified a number of deposits of clay (smectites) in different regions of Spain as being potentially suitable. Additional research activities were carried out in a granite/U-bearing quartz vein system at El Berrocal, which has been studied as a natural analogue of a HLW repository.

- *Repository Design Concepts*: The initial reference concept developed by ENRESA conforms to the multi-barrier principle but exists only at a fundamental level. A generic-type reference conceptual design was first developed for a repository for irradiated nuclear fuel in a granite formation. The repository design comprises horizontal galleries at a depth of about 500 m, waste packages consisting of irradiated nuclear fuel in carbon steel canisters with a service life of at least 1,000 years. The buffer that surrounds the steel canisters consists of blocks of compacted bentonite. The conceptual design has evolved since 1994 aiming for compatibility with repository concepts in clay and salt.
- *Performance Metrics and Assessments*: With regard to radioactive waste disposal facilities, CSN established a criterion that stated that the level of individual risk that applies must be lower than 10^{-6}/year, or the risk associated with an annual equivalent dose to individuals in the critical group lower than 0.1 mSv/year. Preliminary non-site specific evaluation exercises were carried out in the late 1990's using generic data compiled during the Site Selection Plan process, one for a granite site and one for a clay site. A more detailed but still non-site specific performance assessment of disposal in granite was completed at the end of 2001, and revised in 2003, and a similar exercise was completed for clay in 2004. These documents are not available in the public domain. The main cause of canister failure and subsequent radionuclide release was assumed to be either penetration of the canister wall due to localized corrosion, or mechanical collapse after the canister wall thickness has been reduced (to 4.25 cm) by general corrosion.
- *Stakeholder and Public Involvement*: Both the Regulation on Nuclear and Radioactive Facilities and a 2006 Law on the Assessment of the Environmental Effects of Certain Plans and Programs require public involvement in the licensing process, which involves access to information relevant to the preliminary authorization of the facility. The 2006 Law specifically promotes transparency and public participation. Current dialogue with the public is continuing in the context of identifying a site for the centralized storage facility.
- *Program Maturity*: While geological investigations have been carried out in order to identify potential sites for geologic disposal, the current effort is focused on finding a site for the planned centralized storage facility for irradiated fuel. ENRESA's most recent waste management plan envisages site characterization activities for a geologic repository occurring between 2025 and 2040, construction of the repository between 2040 and 2050, and the operational phase beginning around 2050.

9.9 References

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10
SWEDEN

10.1 Introduction

10.1.1 General Nuclear Profile

Sweden currently has a fleet of 10 operating nuclear reactors at three sites representing an installed capacity of 9399 MWe, as well as 2 shutdown reactors at Barsebäck. Nuclear power represents over 40% of Sweden's total electricity production, generating 50 TWh in 2009, with hydroelectric power providing an equivalent electrical capacity and fossil sources rounding out the remainder capacity at just under 10% (WNA, 2010a,b). A moratorium was imposed by a national referendum on the construction of any new nuclear power plants in Sweden (SKBF, 1983); this decision was reversed with legislation from the Riksdag effective 1 January 2011, but the outcome of pending national elections could lead to a reinstatement of the moratorium.

All nuclear power plants are located along the coast and have direct access to harbor facilities, facilitating water borne transport of nuclear fuel using a specially designed Roll on-Roll off (Ro-Ro) ship. Currently, the used fuel is cooled onsite for one year in wet storage following discharge from the reactor and is then transported to a centralized wet storage facility, the Central Storage for Spent Fuel (CLAB), which is co-located with the Oskarshamn nuclear power plant (Figure 10-1). The CLAB has been in operation since July 1985.

Swedish law explicitly requires that all nuclear waste produced in Sweden be subject to permanent disposal in Sweden. The provisions of the Nuclear Activities Act (SFS 1984:3) stipulate that the owners of the reactors, i.e., the producers, are responsible for all measures needed to execute and fund management of the nuclear sector's lifecycle, including disposition of nuclear waste and decommissioning of facilities. As a parliamentary democracy, the permanent Governmental agencies and the elected Parliament ("Riksdag") bear overall responsibility, and the various regulatory authorities ensure that the nuclear power licensees meet the requirements according to law via an oversight and licensing role.

The current national policy (SKB, 1998) is direct disposal of the used fuel in a deep geological repository (once-through fuel cycle) at a depth of ~500 m (see Section 10.4). In addition to the deep repository, SKB's disposal plan also requires licensing of an encapsulation plant where the used fuel is placed in canisters and the canisters sealed. This encapsulation plant is be co-located with the CLAB facility to minimize transportation; the licensing for this facility is proceeding in parallel with the license application for the deep geologic repository.

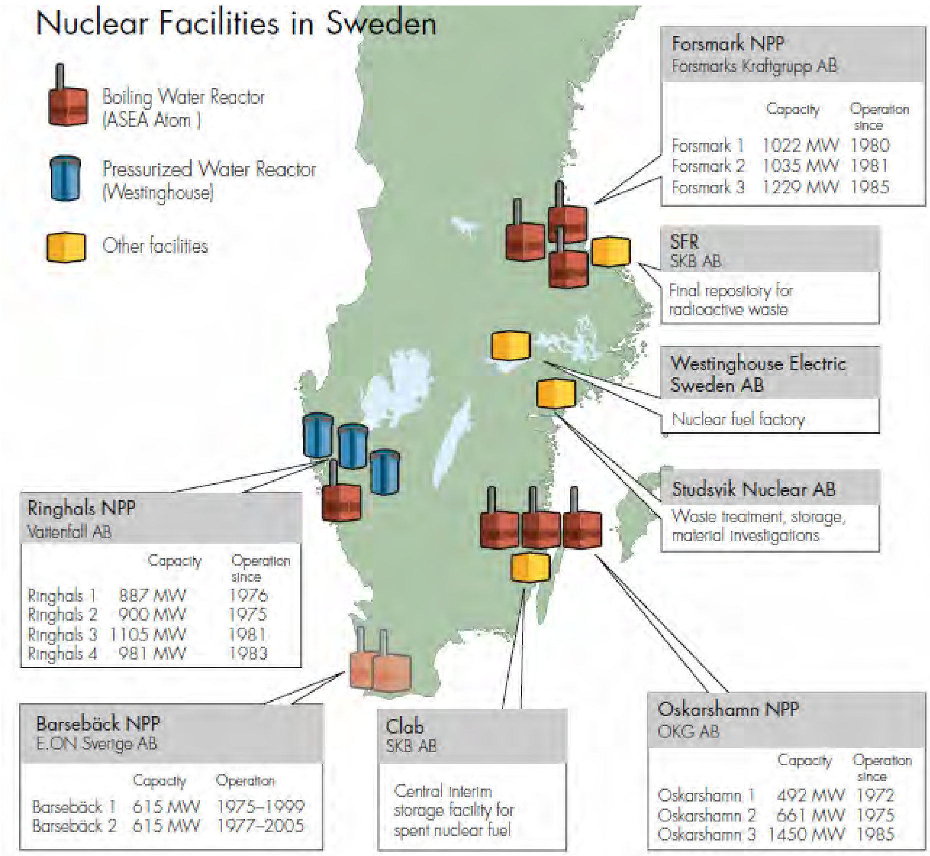

Figure 10-1
Locations of nuclear facilities in Sweden (Source: MoE, 2010). Used with permission of the Swedish Ministry of the Environment.

10.1.2 Used Fuel / HLW Inventory and Projected Disposal Requirements

By the end of 2007, the reactors had produced an estimated 5332 MTHM of used nuclear fuel, the majority of which is consolidated in wet storage at the CLAB facility, which has been expanded to a capacity of 8,000 MTHM (MoE, 2008). For the projected reactor lifetimes, a total of 12,000 MTHM or 19,000 m^3 of used nuclear fuel is expected along with 60,000 m^3 of LLW/ILW reactor waste (NEA, 2010; MoE, 2008).

10.2 Institutional Arrangements

10.2.1 Institutional Framework

Figure 10-2 shows schematically the regulatory responsibilities for the general control of nuclear safety and radiation protection in Sweden. Regulatory authorities involved in siting a nuclear waste repository have both decision-making and advisory roles. In addition, non-regulatory agencies in Swedish society have had a strong influence on decision-making.

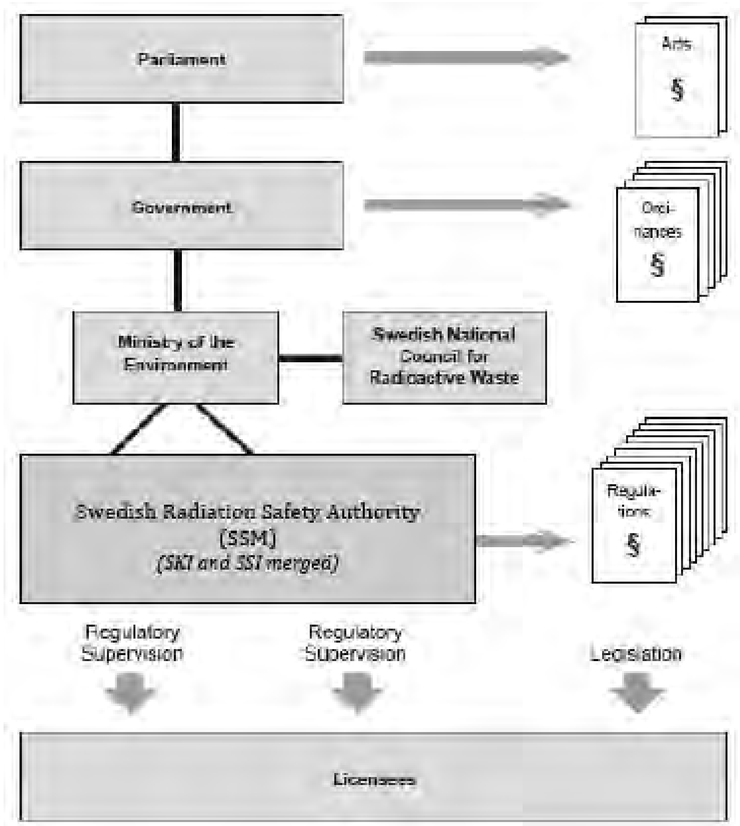

Figure 10-2
Organizational structure for regulatory control of nuclear safety and radiation protection in Sweden (MoE, 2008). Used with permission of the Swedish Ministry of the Environment.

Nuclear safety and radiation protection matters have previously been handled by the regulatory authorities the Swedish Nuclear Power Inspectorate (SKI) and the Swedish Radiation Protection Institute (SSI), respectively. These organizations have now been merged to form the *Swedish Nuclear Power Authority* (SSM, discussed further below). The government also has access to an advisory body, the Swedish National Council for Nuclear Waste.

With respect to planning and environmental protection, other central governmental agencies are involved in the siting process primarily as advisors. Sweden is divided into 288 local municipalities. The *Municipality Council* of the candidate sites has a significant influence on the site selection both formally and informally as a siting decision generally has to be accepted by the Council. Finally, the *general public* and non-governmental organizations (NGOs) have an influence on siting. Indirectly the public also has a strong influence, since they elect both the Government and the Municipality Council were they live.

Key roles in management of used fuel in Sweden are described below:

POLICY and OVERSIGHT – Ministry of Environment (MoE). According to the Environmental Code and the Act on Nuclear Activities (see section 10.2.2.2), most major decisions regarding the nuclear waste management program, including deciding on the permissibility of a license application to construct a nuclear waste repository, are taken by the

Swedish Government. Most authorities involved in the licensing process report to the Ministry of the Environment (MoE). However, in a decision-making role in which the Government acts as an entity, all ministries are part of the decision under the leadership of the Prime Minister. The Government can only rule according to the laws, which are decided by Parliament.

IMPLEMENTER – Swedish Nuclear Fuel and Waste Management Company (SKB). The reactor owners formed a jointly owned company, the Swedish Nuclear Fuel and Waste Management Co. (SKB), to act on their behalf as implementing agency for managing the used fuel and other nuclear waste produced in Sweden. Thus, SKB has the primary responsibility for developing the technical method, finding a suitable site, and preparing the necessary license applications for a final geological repository. Activities are funded by resources in the waste fund.

SKB operates several existing facilities for the back-end of the fuel cycle. In addition to CLAB, there is a licensed and operating final repository, SFR, for disposal of the short-lived L/ILW (Figure 10-1). SFR has an on-shore entrance, but the disposal facility itself is located under floor of the Baltic Sea. SKB also owns the transportation vessel m/s Sigyn. SKB's extensive research and development (RD&D) program[29] is aimed at establishing final disposal for the spent nuclear fuel and other long-lived wastes not suitable for disposal in the SFR repository.

REGULATOR - Swedish Radiation Safety Authority (SSM). Until 2006, two separate regulatory agencies, SKI and the SSI, held oversight authority for the planned geological disposal of spent fuel in Sweden. SKI had a mandate similar to the Nuclear Regulatory Commission in the US, which was to make sure that all nuclear activities in Sweden were carried out in a way to assure public health and safety and to meet safety standards. SSI's role was to establish such safety standards, a role similar to that of the Environmental Protection Agency for the US spent fuel disposal program. Each agency developed its own documentation and reports, which have now been fully incorporated into the combined SSM organization. Because of the different perspectives and histories of these two components of SSM, the roles and responsibilities of SKI and SSI are summarized below, noting that such roles and responsibilities have now been unified into SSM.

SKI (now integrated into SSM): The main responsibility of SKI had been to ensure that all nuclear activities were conducted in a safe manner and in accordance with the Act of Nuclear Activities. SKI would evaluate license applications according to the Act and presents its views and suggest stipulations before licensing matters were decided by Government. (SKI had also been able to make decisions on minor issues themselves).

One of SKI's tasks had been to evaluate SKB's RD&D program, prepared and submitted every three years, including asking for review comments from other authorities, universities, involved municipalities and NGO's. SKI would then submit its own review, with additional comments and its own conclusions to the Government. The Government presents its conclusions in the form of a special Government Decision and can at this point also issue stipulations for the future conduct of SKB's program. The Government usually also makes other general statements, for

[29] Required by law and, since 1992, called the Research Development and Demonstration Program, RD&D, by Industry.

example concerning repository siting. Historically, SKI's recommendations and the associated Government decisions have had a profound impact on the SKB work, as noted in later sections.

SKI was also authorized to issue regulations regarding safety, including regulations for the safety of final geological repositories. The main components of SKI's regulations governing safety in the final disposal of nuclear waste are described as follows (SKI, 2000):

"*Safety, in both the short and long term, shall be based on a system of passive natural and engineered barriers and that an undetected deficiency or unexpected failure in safety function that might arise in one of the barriers should not impair the safety of the final repository.*"

"*The barriers should provide radioactive protection in accordance with the regulations formulated by the Swedish Radiation Protection Institute:*

- *The repository site should be selected such that the rock provides sufficiently favorable conditions such that the barriers will function as intended over sufficiently long time. These conditions concern temperature, flow of water, rock mechanics and geochemistry.*
- *The barrier system should be constructed with regard to the best possible technology.*
- *Actions taken for post-closure monitoring or retrieval should be reported.*
- *Features, events and processes that are of importance for the post-closure safety of a final repository should be analyzed in a safety assessment before the repository is built, before it is commissioned and before it is closed.*
- *A safety assessment shall comprise a time period as long as necessary for evaluating the safety function, but the time period should be at least 10,000 years and only needs to comprise 1,000,000 years.*
- *The validity of models, parameter values and other conditions used in the safety assessment should be proven.*"

The question of whether the fundamental requirements are met for a deep repository on a specific site was to be considered in conjunction with the regulatory review of the safety assessments and environmental impact statements, which SKB was obligated to submit.

It should be noted that SKI's regulations did not directly stipulate requirements on the performance of different parts of the deep disposal system, but discussed in more general terms requirements on the repository system as a whole. SKI's view was that laws and regulations couldn't be used directly to stipulate requirements or preferences on the properties of the rock, or even the engineered barrier system. Such requirements or preferences can only be derived indirectly, based on the impact they may have on the safety of the repository.

<u>SSI (now integrated into SSM):</u> SSI had been the Swedish Government authority with the task of protecting people and the environment from the harmful effects of radiation. SSI's mission was to ensure that the risks and benefits inherent to radiation and its use are compared and evaluated. SSI also developed competence on radiation to minimize the risk involved for the individual and decided the dose limits for the general public and for workers exposed to radiation.

SSI issued safety standard regulations concerning final disposal of spent nuclear fuel (SSI, 1998). The more important paragraphs imply that:

- The final disposal of spent nuclear fuel shall be radiologically optimized and based on the best possible technology.
- A final repository for spent nuclear fuel or nuclear waste shall be designed so that the annual risk of injury after closure is no more than 10^{-6} for a representative individual in a group that is exposed to the greatest risk.
- Final disposal shall be carried out in such a manner that biological diversity is preserved and a sustainable utilization of biological resources is protected against the harmful effects of radiation.

The timeframe for safety assessment is to be one million years after repository closure, although the specific risk limit is applicable only for the first 100,000 years following repository closure. Thereafter, the risk limit is to be used as a general basis for discussing the protective capability of the repository system. "Harmful effects' refer to cancer and hereditary effects. The risk limit corresponds to an effective dose limit of about $1.4 \cdot 10^{-5}$ Sieverts/ year. This is about 1% of the effective dose due to natural background radiation in Sweden.

"Optimization" means limitation of radiation doses to man as far as is reasonably possible, taking into account both economic and societal factors. By "best possible technology (BAT)" it is meant tried and tested technology keeping with accepted scientific principles and taking account of both the benefit and the cost of the measures. This demand for best technology is in accordance with the general requirements of the Environmental Code.

SSI's general acceptance criterion was expressed in terms of risk to individuals, where risk is defined as the product of the probability of receiving a radiation dose and the harmful effects of the dose. It should be noted that this 1998 SSI safety standard meant that Sweden would convert from a previously used 'dose criterion' to a risk criterion. The regulations did not specify the size of the group to be considered being exposed to the greatest risk, but in its general advice SSI suggested that this group should be rather large. As an alternative, a smaller group may be considered. In response to this 1998 SSI criterion, SKB conducted its safety assessment SR 97 (SKB, 1999) and suggested that individuals in a smaller group than that conceived in the SSI regulation may be exposed to a higher annual risk (10^{-5}).

By "biological diversity" SSI meant diversity in species and genetic varieties. In the absence of established methodology for estimating risks to biota, SSI suggested that the "precautionary principle" should apply, i.e. the very suspicion of harmful effects on the environment should be sufficient to intervene or refrain from a given activity.

Although SSI stated that harmful effects in the future should not be regarded as less important than the harmful effects to which man or the environment are exposed today, the first 1000 years after repository closure are the most important to evaluate. The regulations also require an account of a case based on the assumption that the biosphere conditions prevailing at the time of the license application do not change. The period after 1000 years shall also be investigated, and different kinds of uncertainties for different time periods should be considered.

ADVISORY and SUPPORT - Swedish National Council for Radioactive Waste (formerly KASAM). The Swedish National Council for Radioactive Waste (formerly KASAM) was established in 1985 as an independent committee attached to the Ministry of the Environment. The Council's mandate is to study issues relating to nuclear waste and the decommissioning of nuclear installations and to advise the Government and certain authorities on these issues. It has broad similarities to the Nuclear Waste Technical Review Board (NWTRB) in the US.

The Government authorized the Minister of Environment to appoint a Chairman and up to ten other Members. The members are independent Swedish experts within different areas of importance for the final disposal of radioactive waste, not only within technology and science, but also within areas such as ethics, psychology, law and social sciences.

According to its instructions (Dir. 1992:72), KASAM shall:

- Present reports on the state of knowledge in the nuclear waste area every third year. Such reports have been presented - in Swedish only - in 1986, 1987, 1989, 1992 and 1995.
- Present an independent review of the research and development program for the final disposal of spent nuclear fuel that the nuclear power utilities prepare once every three years.
- Act as an advisory committee – upon request – to the authorities within the nuclear field (the Swedish Nuclear Power Inspectorate, SKI, and the Swedish Radiation Protection Institute, SSI) on matters connected with nuclear waste and the decommissioning of nuclear power plants.

Besides some technical seminars, the Council also arranges seminars with the aim of opening up a dialogue between different interest groups that are seriously interested in nuclear waste-related issues.

10.2.2 Legal and Regulatory Framework

The general requirements on siting, licensing, construction, and operation of deep geologic repository for used nuclear fuel from commercial nuclear power generation emanate from several acts of law passed by the Swedish Parliament. The most important laws in this regard are the Environmental Code, the Nuclear Activities Act and the Radiation Protection Act, described below. Separate permits granted under the auspices of the Environmental Code and the Nuclear Activities Act, are required for repository construction.

10.2.2.1 The Environmental Code

The Environmental Code (SFS 1998:808) contains a requirement for the issuance of permits for siting of the deep repository and the preparation of an Environmental Impact Statement (EIS) as part of the license application (discussed further in Section 10.6.2). The rules on EIS are adopted from European Union Directives (85/337/EG and 97/11/EG). The Code also imposes limits on the environmental impacts of the deep repository.

The Code specifies that a license application for a nuclear waste repository should be handed over to the Environmental Court. However, before the Court can try the case the Government first has to consider and affirm the permissibility of the activity (i.e. to construct a nuclear waste repository). If the Government finds the activity to be permissible the Environmental Court has to grant a license, but the Court can (and shall) issue stipulations for the activity as part of this license.

As part of the siting process followed in Sweden under the Environmental Code, the Government can only permit the activity if the municipal council of the affected municipality accepts it, i.e., the council can veto an application. However, for nuclear waste repositories and some other activities judged to be of major national interest, there were some formal possibilities for the Government to overrule the veto. According to the Environmental Code, a veto could not be overruled if another municipality accepts the facility or if the Government judged that another site is more suitable.

10.2.2.2 Nuclear Activities Act and Radiation Protection Act

Requirements on safety and radiation protection are set forth in the Nuclear Activities Act (SFS 1984:3) and the Radiation Protection Act (SFS 1988:220). The Nuclear Activities Act prescribes in general that nuclear activities shall be conducted in a safe manner and in accordance with Sweden's international undertakings concerning the non-proliferation of nuclear weapons. The Radiation Protection Act prescribes in general that with regard to conducting activities with radiation, measures and precautions are needed to prevent harm to humans, animals and the environment. Ordinances issued by the Government pursuant to the Nuclear Activities Act and the Radiation Protection Act contain some more detailed provisions and regulate the activities of the nuclear regulatory authority SSM. These additional ordinances are still couched in general terms regarding requirements on safety and radiation protection in the deep repository.

The Nuclear Activities Act and the Radiation Protection Act put far-reaching responsibility on licensees to own or operate a nuclear reactor. To ensure that they meet these responsibilities, licensees are also required to conduct a comprehensive research, development and demonstration (RD&D) program.

10.2.3 Waste Classification

There is no legally defined waste classification system in Sweden. Sweden's waste classification system is based on waste management practices, in particular disposal route (NEA, 2009):

- VLLW: operational and decommissioning waste that can be cleared for disposal in municipal facilities or designated for shallow burial at nuclear sites;
- LILW – short-lived: operational and decommissioning radioactive waste that is disposed of in the existing near-surface repository for short-lived waste at Forsmark (SFR);
- LILW – long-lived: internal reactor parts and high-alpha content (TRU) waste that is designated for disposal in the planned repository for long-lived waste (SFL);

- Spent fuel: irradiated nuclear fuel that is designated for disposal in the planned geologic repository for used fuel (SFK).

10.2.4 Funding

Figure 10-3 provides an overview of the funding mechanisms for radioactive waste management in Sweden. The Nuclear Activities Act puts the full responsibility to finance all waste management programs on the reactor owners. The details are provided in the "Act on Financing Future Expenses for Spent Nuclear Fuel, etc'' (SFS 1992:1537). According to this Act, the reactor owners pay a fee to the Nuclear Waste Fund. The authority is overseen by a Board of Governors appointed by the Government. The average charge between 1982 and 1996 was 0.019 SEK per kWh, but has since dropped to an average 0.005 SEK per kWh (Ministry of the Environment, 2008). The latter assumes an operational period of 40 years and a minimum remaining operating time of 6 years. The fund can also be used to cover certain other expenses, in particular costs for informing the general public.

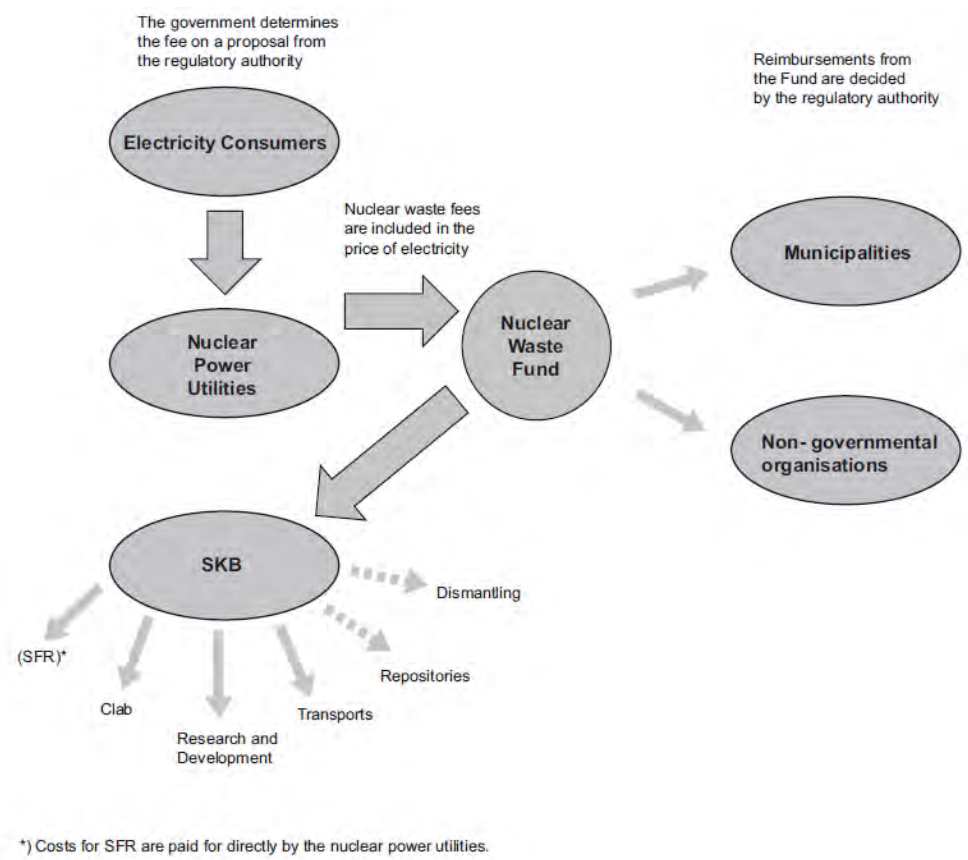

Figure 10-3
Funding of radioactive waste management activities associated with nuclear power in Sweden. (MoE, 2008). Figure used with permission of the Swedish Ministry of the Environment.

The regulatory authority SSM also has duties regarding the financing of the waste management, by reporting to the Government on how much funding is needed for future management and disposal of the nuclear waste and for decommissioning the nuclear installations. Based on these estimates the Swedish Government decides on the fees on the electricity generated in nuclear power plants.

10.3 Site Screening, Selection and Characterization

10.3.1 Deep Disposal – Geological Studies and Site Selection - Overview

The SKB plan for site selection and characterization was divided into three consecutive stages (Figure 10-4), *viz.*

- General siting and feasibility studies,
- Site investigations, and
- Detailed investigations.

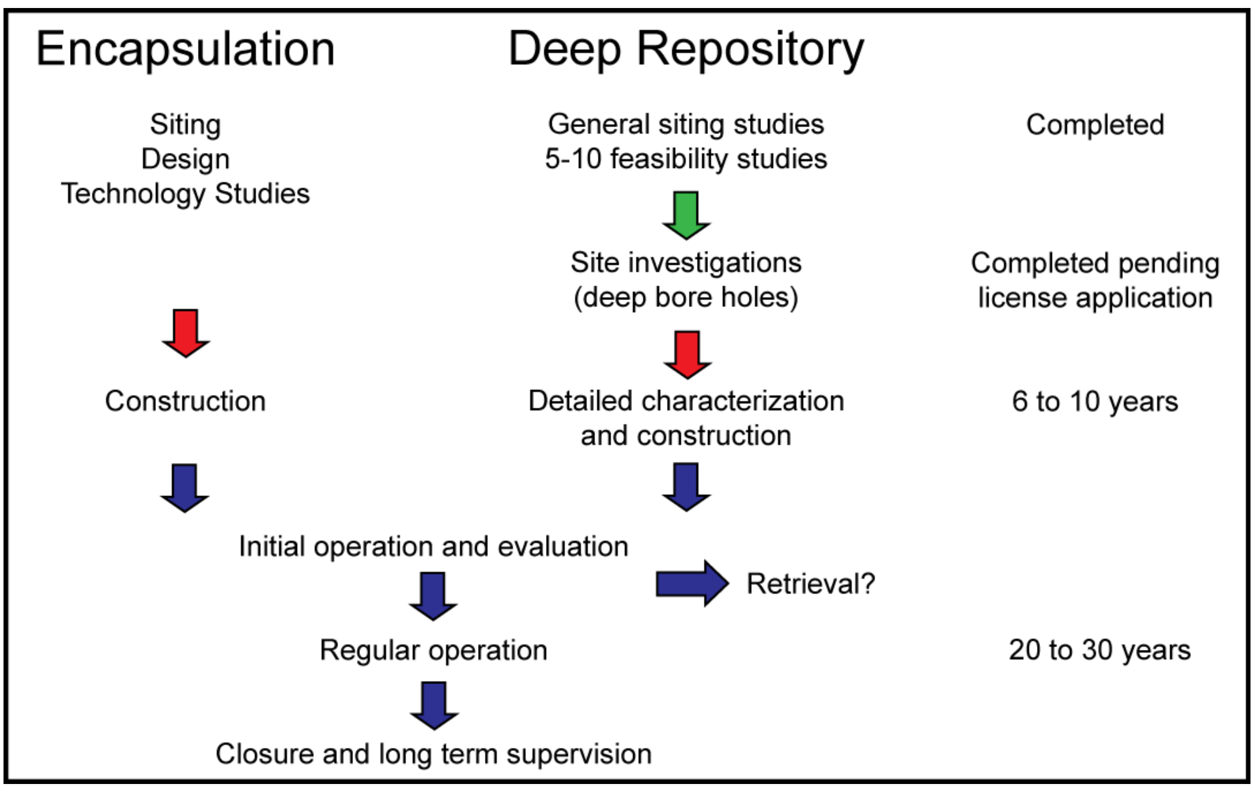

Figure 10-4
The different siting steps together with the estimated duration of these steps. Note: Red arrows signify that a formal license is required, blue arrows signify that an operational permit or a license is needed, and green arrows imply that no formal license is needed. (SKB, 2009). Used with permission of SKB.

The timing of the stages changed over time, in part because SKB decided to seek the consent of the local municipalities for each stage. The first two steps of "General siting studies" and "Site investigations" started in the early 1980's and were completed in 2009.

10.3.1.1 Legal Requirements on Siting

The different Acts and Laws in Sweden do not directly specify criteria for selecting a site for a nuclear waste repository. Instead the legislation generally requires that an EIS is produced comparing alternative sites and ensures that there is a competent review of any license application. The regulations also stipulate that the ultimate decision be made by the Government and the local (the Municipality Council) decision makers. Swedish legislation also authorizes the regulator to issue regulations on the safety of a nuclear waste repository.

The former regulations and general advice issued by SKI on siting (see Section 10.2.1.3) did not offer much practical guidance to SKB as implementer. Rather the regulations specified that the siting decision must be properly defended, regardless if the site selection was based on a staged approach or not.

10.3.2 Results from SKB Research and Development Program

The siting process and associated site characterization in Sweden has evolved through several stages as conducted by SKB and reviewed by governmental agencies, as summarized below.

10.3.2.1 Exploration for suitable study areas (1980-1992)

In the 1970's and early 1980's SKB (and its predecessors) carried out several "study sites" site investigations, including boreholes drilled to ~600 m and standard hydrogeological and geochemical investigations. None of the studied sites was ever officially called a potential repository site - the exploration was done for research, although this philosophy eventually changed.

SKB's 1983 Safety Assessment (SKBF, 1983) demonstrated that the explored sites were potentially suitable. SKB's plan was to continue to explore these, and possibly a number of other sites and then as a result of the investigations suggest a site for the final repository (SKB, 1989). Interestingly, during the entire 1980's, SKB's principal focus with respect to siting criteria was on finding sites with favorable hydrological properties.

During most of this period, SKI generally agreed with the SKB's approach, while pointing out that extensive explorations, possibly including underground constructions, would be necessary before a site could be declared suitable (SKI, 1987). Subsequently in 1989, at SKB's presentation of its RD&D program (SKB, 1989) SKI asked for a more systematic site selection process (SKI, 1990). The regulator suggested that sites should be selected through a process of "systematic elimination", while acknowledging that it is impossible to design a process leading to the "best site". SKI also questioned the adequacy of SKB's site investigation program, expressing concern that the investigation methods might not be suitable to discover deep, sub-horizontal fracture zones.

One of the advisory bodies, the Swedish National Board for Spent Nuclear Fuel (SKN), which at that time was responsible for reviewing SKB's RD&D Program 1989, suggested that the siting program should proceed in stages. It also made other useful suggestions (see Section 10.3.4).

SKB abandoned the "study area" approach around 1990. The main reason was probably the difficulties with local public opinion at the sites, and possibly the criticisms by review bodies including the regulator that a systematic approach was lacking.

10.3.2.2 SKB report on siting factors important to safety (1992-1994)

With the publication of R&D 92 (SKB, 1992a) SKB launched a concrete program for site selection and geological investigation. In its proposed program, referring to the safety assessment SKB-91 (SKB, 1992), SKB claimed that there are good possibilities to find suitable sites from a technical and geological standpoint at many locations in Sweden. Importantly, to support its revised approach to siting, SKB stated (SKB, 1992):

> *"The safety of such a repository is only slightly dependent on the ability of the surrounding rock to retard and sorb leaking radioactive material. The primary function of the rock is to provide stable mechanical and chemical conditions over a long period of time so that the long-term performance of the engineered barrier system is not jeopardized."*

By this statement, SKB was acknowledging that hydrological aspects such as flow and sorption by rock minerals (which could vary strongly from one site to another) were far less important to long-term safety than the isolation capabilities and performance of the engineered barrier system (EBS). In this way, SKB argued that many sites in Sweden could be considered suitable if they were geomechanically and geochemically 'stable'.

SKB's plan called for site selection to start with overview studies of the whole country, followed by 'feasibility studies' in municipalities that accepted SKB's presence. These studies would explore a variety of factors including legal, technical/geological, societal, and political. Based on the results of these studies, SKB then planned to select two sites for "site investigations", followed by underground explorations at one site ("detailed investigations"). Operation would then start with a minor amount of waste emplacement in a first stage, followed by a phase of evaluation with possibilities for waste retrieval. This first stage was called a "demonstration phase". Effectively, though with modifications, this is the program that SKB has followed since 1992.

10.3.2.3 SKB's development of siting factors and feasibility studies (1994-2000)

SKB's RD&D 92 report discussed general siting guidelines and factors to be considered in its Feasibility Studies. SKI (1993b), however, criticized the lack of well-defined siting factors, stressing the need for evaluating safety with an integrated safety assessment. SKI accepted the idea of selecting volunteering communities, while noting that the degree of local acceptance would depend on the possibility to clearly show that the selected sites are indeed safe. Several other review bodies, including KASAM and several municipalities, also expressed a need for the presentation of siting factors prior to final siting decisions. In response, the Government stipulated that SKB had to amend its RD&D program with "… *the criteria and methods which can form the basis for the selection of suitable sites for a repository… and a program for the identification of design specifications for an encapsulation plant and repository*" (Government, 1993).

SKB amended its approach (SKB, 1994) accordingly and described "siting factors" under four different headings;

- Safety,
- Engineering,
- Environment, and
- Society.

SKB differentiated between basic safety requirements and factors/conditions that at different stages during the siting work appeared to be favorable, unfavorable or directly disqualifying for further siting work in a particular area. The siting factors for a certain stage were determined based on the knowledge available at that stage. Areas that were not expected to fulfill requirements were eliminated from further study, whereas more detailed siting studies were carried out for favorable areas.

The factors and criteria were presented in a quite general form. Primarily, favorable sites were those with stable and suitable geochemical and mechanical conditions, enhancing low radionuclide releases and slow radionuclide transport.

The approach and the factors presented were accepted by the Government and the regulatory authorities *"...as a suitable point of departure for the continued work*". However, according to SKI (1995), the factors must be more precisely defined and updated before the selection of sites for site investigations. According to SKI "… *an in-depth analysis of the connection between measurable site characteristics and safety requirements should among other things be carried out in connection with the development of a program for site investigations*". The view that siting factors should be established prior to site investigation was also strongly pushed by several municipalities. The Government also decided that 5 to10 feasibility studies should be completed before SKB could move to the next stage: *site investigations* (implying investigations from deep bore holes).

In the General Siting Study 95 (SKB, 1995b), SKB described conditions on a national scale as a general background to the fundamental prospects for siting a deep repository. Based on review comments from SKI and most other reviewers, the Government's decision (Government, 1996) in response to SKB's RD&D Program 95 (SKB, 1995a) stated that SKB's General Siting Study should be supplemented. The government suggested that, prior to selection of a site for "site evaluation" (i.e. sinking boreholes), SKB needed to provide:

- the assembled collection of SKB's general overview studies and feasibility studies,
- other background or comparison material,

- criteria for assessment of the sites, and
- factors that exclude further studies of a site.

In the RD&D Program 98 (SKB, 1998) SKB formulated the interim goal of being able to choose at least two sites for site investigations in 2001. In order to meet this goal SKB initiated several projects in order to prepare the various pieces of background information needed according to government decisions and regulatory advice, *viz.*

- An extensive project starting in 1996 to identify all the parameters that can be determined in a geoscientific site investigation (Andersson *et al.*, 1998).
- Following on from Andersson et al. (1998), a project aimed at quantifying requirements and preferences for a suitable repository site for the purpose of deriving siting factors and siting criteria for site selection before and after site investigations (Andersson *et al.*, 2000).

SKB's systematic approach to siting factors and criteria (Andersson *et al.*, 2000) are much more recent than those developed and applied in the US in the early 1980's. Because Andersson *et al.*, 2000 reflects more recent thinking about siting factors, Section 10.3.3 provides an overview of the Andersson *et al.* (2000) report and a discussion on how the derived siting factors fulfilled the needs expressed by Swedish Government and authorities.

10.3.2.4 SKB's initial site investigations and site selection (2000-2009)

By 2000, SKB had completed Feasibility Studies in six municipalities, all of which have nuclear installations or are direct neighbors to municipalities with nuclear installations. These Feasibility Studies were confined to information that already existed. SKB also conducted Feasibility Studies in two municipalities in the northern inland part of the country but, following negative votes in local referenda on being a possible repository site, SKB canceled further consideration of these sites.

In 2000, SKB selected three sites for initial site investigation during the fall of 2000, at Simpevarp, Laxemar and Forsmark. The Simpevarp and Laxemar sites are subareas around Oskarshamn, and the Simpevarp subarea was subsequently set aside because it presented less flexibility than Laxemar with respect to available space for waste deposition.

According to the adopted nomenclature, "site investigations" implied selection of a detailed study area large enough to host the repository (5 – 10 km^2), with careful exploration of the selected area and its surroundings by means of drilling deep boreholes. SKB published a general site investigation and evaluation program (SKB, 2000), which was slightly adapted to the selected sites during the investigations. The site investigations were planned to be conducted in two phases

- Initial site investigations, and
- Complete site investigation.

The initial site investigation stage was formally completed with the publication of the SR-Can report (SKB, 2006), in which SKB formally compared the performance of a standard "KBS-3 type" repository design using site-specific data from Laxemar and Forsmark. Both sites were shown to comply with the SSI's 100,000-year risk criterion, and were accordingly judged to be suitable for selection as the repository site.

Thus, after a multi-year set of site characterization programs, SKB formally announced in the summer of 2009, its selection of the Forsmark site as the site for the deep geologic disposal of used fuel in Sweden. SKB (2009) concludes:

> *"The systematic examination of conditions on the sites shows that Forsmark offers the best prospects for achieving long-term safety in practice. The most important reasons are the great differences in frequency and permeability of conductive fractures. The rock conditions in Forsmark also permit a more robust and efficient execution than in Laxemar.*
>
> *It is conceivable that further research will show that we have treated the risks of bentonite erosion much too pessimistically today. If this turns out to be the case, the radiological risk would be lower at both sites and the relative advantage for Forsmark would thereby decrease. But it is difficult to imagine that the safety ranking between the sites would be changed by additional knowledge.*
>
> *The industrial prospects for establishing and operating the final repository in a good way are judged to be good at both sites. The differences that do exist cannot be accorded any decisive importance for site selection. The same applies to the environmental impact which the project will cause. In view of this, SKB concludes that Forsmark is the most suitable site for the final repository."*

This decision effectively closed SKB's initial site investigation phase and initiated activities for preparing and submitting a license application (see Figure 10-4).

10.3.3 SKB's Siting Guidelines and Siting Criteria

10.3.3.1 Background to SKB siting guidelines and criteria

Andersson *et al.* (2000) gives an account of what general *requirements* SKB would apply on sites, what *preferences* of a site would be considered advantageous, and how the fulfillment of requirements and preferences (siting criteria) should be judged prior to the selection of sites. The conclusions and results of Andersson *et al.* (2000) were based on the knowledge and experience acquired by SKB over many years of research and development. The knowledge gained during SKB's SR-97 safety assessment (SKB, 1999) was a primary source of such insights on key siting factors.

The stipulated criteria in Andersson *et al.* (2000), governed by Swedish laws and regulations, apply to a repository for used fuel to be emplaced in a "KBS-3 type" repository, i.e. a repository where the used fuel is contained in copper canisters embedded in bentonite clay at a depth of 400–700 m in crystalline (i.e., 'granitic/ gneissic') basement in Sweden. If the repository concept were to be changed, or if new technical/scientific advances are made, certain requirements, preferences or criteria may need to be adjusted. Thus, it should be emphasized that the Andersson *et al.* (2000) study can only used as a basis for siting of other types of repositories or in other geological settings by careful consideration of caveats and limitations cited in the report.

Andersson *et al.* (2000) analyzed how the rock's different geological conditions, mechanical properties, thermal properties, hydrogeological properties, chemical properties and transport properties influence the functions of the deep repository, and whether it was possible to determine requirements and preferences regarding the influence of these properties. Where possible, these requirements or preferences were then translated into requirements or preferences regarding the individual properties (parameters). Parameters that could be used to determine whether *requirements* or *preferences* were satisfied were called '*geoscientific suitability indicators'*. In order to be able to determine at different stages during a site investigation whether requirements and preferences for a given parameter are satisfied, criteria were formulated that were based on the quantities that could be measured or estimated at the relevant stage of the investigation.

It was noted in Andersson *et al.* (2000) that terms such as '*siting factor'* and '*siting criteria'* were often used without any precise definition. In order to increase precision in terminology they used the definitions shown in Table 10-1.

Table 10-1
Definition of terms used in Andersson *et al.* (2000) on siting factors.

Term	Definition
Function	Purpose which the deep repository is intended to serve, for example to have an isolating and retarding function.
Parameter	Physical or chemical quantity (property, condition in the rock).
Requirement	Condition that must be satisfied, refers to actual conditions regardless of siting stage. All requirements must be satisfied.
Preference	Condition that ought to be satisfied regardless of siting stage. All preferences do not have to be satisfied, however.
Geoscientific suitability indicators	Measurable or estimable site-specific parameters that can be used in a given siting stage to assess whether requirements and preferences are satisfied.
Criteria for site evaluation	Values for suitability indicators in a given siting stage that are decisive for the assessment of whether a site satisfies stipulated requirements and preferences.

10.3.3.2 Basis for formulating requirements and preferences

Andersson *et al.* (2000) contains assessments of the required or desired function of the host rock at a candidate site, both from a long-term safety perspective and from an engineering perspective. The development of more precise descriptions of *requirements* and *preferences* were derived by a stepwise procedure:

- Identification of key barrier functions using safety assessment results.
- Formulating engineering type requirements and preferences.
- Exploring rock property impact on key barriers and engineering conditions.
- Assessment of rock properties in order to formulate requirements and preferences.
- Assessment to what extent the important rock properties can be assessed at various stages of the site selection process.

The analyses of Andersson *et al.* (2000) were based on many years of experience, analyses and development around the KBS-3 concept. The assessment of what factors are essential from the viewpoint of long-term safety were based on analyses performed within the framework of SR 97 (SKB, 1999), complemented with previous knowledge and experience.

For each site-related parameter needed to describe the safety assessment's initial state, the question was asked whether this parameter should be a '*geoscientific suitability indicator*'.

Safety assessment was to be used to seek an answer to the question of whether there are value ranges in a site property where the deep repository's isolation can be threatened. As a precaution, such value ranges comprised a basis for formulating requirements, even though it was not always clear that the deep repository would definitely be unsafe if the requirements were not met. The requirements could only be reconsidered in the light of new knowledge or if the design of the repository was significantly altered.

Safety assessment was also to be used to determine the basis for preferences regarding value ranges that contribute to enhanced containment or high isolation. Such value ranges result in *desired function*, but do not necessarily define the borderline to *unacceptable function*. Such a borderline is, in many cases, influenced by other parameters. Furthermore, the borderline can be relative to assumptions on other factors, such as the influenced of repository layout.

The *requirements* defined conditions that may not occur. The *preferences* defined conditions that lead to enhanced containment or high isolation, but a deep repository site could be safe even if many *preferences* were not satisfied. The *requirements* and *preferences* were formulated to provide guidance in the siting work and to be able to prioritize investigation activities in site investigations. They do not take the place of integrated and complete safety assessments.

Andersson *et al.*, (2000) noted that *requirements* and *preferences* that were framed from the rock engineering perspective were of a somewhat different character than those *requirements* and *preferences* for direct safety requirements. The repository layout was designed primarily to achieve as good performance and safety as possible: canister and tunnel spacing were to be determined by requirements on temperature in and around the repository, major discontinuities are avoided, etc. Furthermore, pure rock excavation aspects, such as water seepage and rock stability in tunnels, would be taken into account. The general rule was that conditions that were favorable from a safety viewpoint should also entailed good constructability and a safe working environment. Good constructability and a stable rock facility were further advantageous for safety during the operation of the facility. There was, therefore, according to Andersson *et al.* (2000) seldom any conflict between the requirements and preferences that can be formulated from different viewpoints.

Andersson *et al.*, (2000) presented the following general *requirements* from a civil engineering perspective:

- The working environment shall be safe,
- The environmental impact of investigations, of construction and of operation shall be limited and kept within acceptable levels,
- Construction shall only have a limited and transient impact on the safety functions of the deep repository,
- Excavation of deposition areas shall be able to take place at the same time as deposition in other areas.

Beyond this there are *preferences* that:

- The rock work can be done with as few interruptions and as little use of extraordinary reinforcement and sealing measures as possible (good constructability),
- The deposition area does not have to be split up into a very large number of subareas, and that it is possible to position deposition tunnels in a flexible manner in the selected deposition areas.
- During and after site investigations, a construction analysis must be carried out for the chosen repository layout, in which constructability, time and material consumption, environmental impact, working environment, etc. for the rock construction were analyzed. If the safety assessment or construction analysis indicated unreasonable consequences or costs for the chosen layout, the construction analysis needs to be changed. In other words, the construction analysis does not impose any absolute requirements, since adjustments can generally be made to suit prevailing conditions. There are, on the other hand, a number of factors that influence constructability and costs, which were enumerated in Andersson *et al.* (2000).

10.3.3.3 Resulting requirements and preferences

Numerous conditions need to be determined in a site investigation in order to develop a fundamental understanding of the site. Only certain conditions, however, are of direct importance regarding whether the investigated site is suitable for a repository or can accommodate the layout of the repository. The following *requirements* were identified in Andersson *et al.* (2000) on the rock or the placement of the deep repository in the rock, again stressing that these requirements are specific to the Swedish host rock and repository design being considered in the Andersson *et al.* (2000) report:

- The rock in the repository's deposition zone may not have any ore potential, i.e. may not contain such valuable minerals that it might justify mining at hundreds of meters' depth. The main reason for this requirement is mitigate the risk of human intrusion (for more detail, see Andersson *et al.*, 2000 section 4.4).
- Regional plastic shear zones shall be avoided if it cannot be demonstrated that the properties of the zone do not deviate from those of the rest of the rock. The main reason for this requirement is that regional plastic shear zones in the crystalline rock often are very heterogeneous and contain several water-bearing fracture zones with reduced mechanical strength (for more detail, see Andersson *et al.* 2000, section 4.5).
- It must be possible to position the repository with respect to the fracture zones on the site. Deposition tunnels and deposition holes for canisters may not pass through or be positioned too close to major regional and major local fracture zones. Deposition holes may not intersect identified local minor fracture zones. The main reason for this requirement is that fracture zones usually are more permeable and have reduced mechanical strength (for more detail, see Andersson *et al.* 2000, section 4.6).
- The rock's strength, fracture geometry and initial stresses may not be such that large stability problems may arise around tunnels or deposition holes within the deposition area. The main reason for this requirement is to ensure reasonable feasibility in constructing the repository and for safeguarding workers health (for more detail see Andersson *et al.* 2000, section 5.2).
- The groundwater at repository level may not contain dissolved oxygen. Absence of oxygen is indicated by a negative Eh, occurrence of Fe(II), or occurrence of sulphide. The main reason for this requirement is that if there were dissolved oxygen in the groundwater, this could lead to corrosion of the copper canister (for more detail, see Andersson *et al.* 2000, section 8.2).
- The total salinity (TDS = Total Dissolved Solids) in the groundwater must be less than 100 g/l at repository level. The main reason for this requirement is that the swelling capacity of the bentonite buffer may be substantially reduced at higher salinities (for more detail, see Andersson *et al.* 2000, section 8.4).

In addition to the above *requirements*, there are a large number of *preferences*, i.e. conditions that were judged in Andersson *et al.* (2000) to be desirable and should be taken into account when positioning the repository in the rock:

- Since it can be difficult to predict how different rocks and minerals will be used in the future, it is preferable to site the deep repository in commonly occurring rock types. With this preference, risk of future intrusion would be further mitigated.
- Moderate density (fracture surface-area per rock volume) of local minor fracture zones is preferable, along with moderate density of fractures. A low fracture density will make it easier to find suitable volumes for deposition holes in the rock as fractures are associated with higher permeability and reduced rock mass strength.
- It is generally a preference if the initial rock stresses at the planned repository depth do not deviate from what is normal in Swedish crystalline bedrock. With rock stresses within the desired interval there is good experience from underground engineering work.

- It is preferable that the strength and deformation properties of the intact rock be normal for Swedish bedrock, since experience has shown it is possible to carry out rock works with good results in such bedrock.
- It is preferable that the coefficient of thermal expansion have normal values for Swedish bedrock (i.e. within the range 10^{-6} to 10^{-5} K^{-1}) and that it not differ markedly between the rock types in the repository area. Within the desired range there is good understanding of the effect of thermal expansion of the rock resulting from the residual heat of the waste.
- The rock should have a higher thermal conductivity than 2.5 W/(m•K). Areas with a high potential for geothermal energy extraction should be avoided. The undisturbed temperature at repository depth should be less than 25°C. If these preferences are fulfilled there is no need to adjust repository design in order to keep the temperature on the canister surface below the performance objective of 100 °C.
- It is an advantage if a large part of the rock mass in the deposition zone has a hydraulic conductivity (K) that is less than 10^{-8} m/s. For a rock with lower hydraulic it is easier to demonstrate its retention capacity, i.e. if the preference is fulfilled the safety case is made easier.
- Fracture zones that need to be passed during construction should have such low permeability that they can be passed without problems, which means the zones should have a transmissivity (T) that is lower than 10^{-5} m^2/s and are furthermore not problematical from a construction-related viewpoint.
- It is an advantage if the local hydraulic gradient is lower than 1% at repository level, but lower values do not provide any additional advantage. The hydraulic gradient drives the groundwater flow, but very low measured gradient may in fact be an indication of regions of high permeability.
- Undisturbed groundwater at repository level should have a pH in the range 6–10, a low concentration of organic compounds ([DOC]<20mg/l), low colloid concentration (lower than 0.5 mg/l), low ammonium concentrations, some content of calcium and magnesium ($[Ca^{2+}]+[Mg^{2+}]>4$ mg/l) and low concentrations of radon and radium. With a groundwater composition within the preferred ranges the geochemical databases for sorption are well known.
- It is preferable that it be possible to find canister positions in a large fraction of the rock that has a Darcy velocity lower than 0.01 m/y on a canister hole scale, since lower fluxes increase the retardation of important radionuclides.
- It is preferable that a substantial retardation of important radionuclides take place in the geosphere. A quantitative preference can be expressed in the form of the transport resistance (a so-called ***F*** parameter), where Darcy velocity, flow distribution and the flow-wetted surface area per volume of rock (or equivalent parameter) are such that a large fraction of all flow paths have F greater than 10^4 y/m.
- It is desirable that matrix diffusivity and matrix porosity not be much lower (by a factor of 100 or more) than the value ranges analyzed in the safety assessment SR 97. The accessible diffusion depth should at least exceed a centimeter or so. A fulfilled preference means that it is comparatively easy to demonstrate the retention capacity of the rock.

- Areas where biological diversity or species worth protecting may be threatened and areas which are or may be important water sources, soil sources or farmland should be avoided for the deep repository's surface facilities. Areas protected by law are avoided.

As a general rule, Andersson *et al.* (2000) state that satisfied preferences lead to greater safety margins, lower costs, simpler investigations or simpler construction of the repository. The authors also stress that not all preferences have to be satisfied for a site to be approved for a deep repository - "poorer" values for certain parameters may well be compensated by "better" values for others. An integrated safety assessment and a construction analysis are, therefore, always needed to assess safety and performance.

In addition to the above preferences that have directly to do with the properties of the rock, there *are preferences that facilitate the characterization of the site*. In particular:

- That there be a high proportion of exposed rock and otherwise moderate soil depth (preferably less than about 10 m), since this facilitates determination of the lithological and geological-structural conditions in the underlying bedrock from the ground surface.
- That the bedrock be homogeneous with few rock types and regular fracturing, although a small-scale variation in mineral composition, e.g., a gneiss, is no disadvantage.

Even though the requirements and preferences were formulated on the basis of different safety and construction viewpoints, there was scarcely any example of a conflict between different requirements or preferences. As a rule, conditions that lead to good long-term safety are also advantageous from the construction viewpoint.

10.3.3.4 Site selection criteria

By assessing to what extent *preferences* and *requirements* could be estimated at different stages of a site investigation (and site selection), Andersson *et al.*, (2000) formulated criteria to be used at two distinct stages:

- *Before* selection of areas for site investigations,
- *After* a site investigation with deep boreholes.

Andersson *et al.* (2000) also discussed how to select areas for site investigations. It was stated that requirements and preferences regarding the rock should be used as far as possible to formulate criteria for selection of sites for site investigations. Good knowledge of the conditions on the ground surface usually exists after completion of a Feasibility Study, while knowledge of conditions in the deep rock might still be very limited. Criteria could, therefore, normally only be formulated for the following *geoscientific suitability indicators*:

- After completion of a feasibility study, continued studies and investigations are only conducted in areas that are deemed to be homogeneous and to consist of commonly occurring rock types, and that are not deemed to have a potential for occurrence of ore or valuable industrial minerals.

- During the feasibility study, the study site is selected and adapted so that a deep repository can be positioned with good margin in relation to regional plastic shear zones and the regional fracture zones interpreted in the feasibility study.
- Areas protected by law are avoided, and areas for further investigations are chosen so that they have few conflicting interests (e.g., a water source) and so that the surface portion can be adapted with little impact on the near-surface ecosystem.
- Areas with an unsuitably high topographical gradient on a regional scale (greater than 1%) are rejected.

Even if the Feasibility Studies identified areas with a good potential to have suitable conditions, further site investigations (from boreholes) would be necessary to confirm this. However, Andersson *et al.* (2000) also made a survey of the generic knowledge of the Swedish crystalline bedrock. It was shown that good prospects should exist to find many sites in Sweden that satisfy all *requirements* and most of the essential *preferences*.

10.3.3.5 Disqualifying criteria for abandoning a site during site investigations

According to Andersson *et al.* (2000), the overall safety assessment combined with the overall construction analysis make up the essential background material in an integrated assessment of whether a site is suitable. The site is only acceptable (but not necessarily selected) if it is possible to show by means of an integrated safety assessment that a safe deep repository can be implemented. During a site investigation, when measurement data have been obtained from repository depth, but before the overall assessment has been carried out, criteria must be used to check whether the *requirements* and *preferences* are (or are likely to be) satisfied. The criteria provide guidance on the outcome of the assessments and can, therefore, also be used to review a safety assessment.

The following criteria, *specific to the KBS-3 concept and specific to the general conditions of the geological setting in Sweden,* were judged so important that the site investigation should be discontinued and another site chosen if they cannot be met:

- If large deposits of ore-bearing minerals or valuable industrial minerals are encountered within the repository area, the site should be abandoned.
- During the site investigation, the repository is adapted more precisely to the then-identified fracture zones. Suitable respect distances to major identified regional and local major fracture zones can only be determined site-specifically, but it is assumed that a distance of at least several tens of meters to major local zones and at least 100 m to regional zones is appropriate. If the repository cannot be positioned in a reasonable manner (necessitating being split into a large number of parts) in relation to regional plastic shear zones, regional fracture zones or local major fracture zones, the site is not suitable for a deep repository.
- If the repository cannot be reasonably configured to avoid extensive and general stability problems, the site is unsuitable and should be abandoned. Extensive problems with "core-disking" of drill cores should arouse suspicions that such problems may arise.
- At least one of the indicators for the absence of dissolved oxygen; negative *Eh* values, the occurrence of dissolved Fe^{2+}, or sulfide, must be demonstrated by measurements of

groundwater composition at repository depth. If none of the indicators can clearly indicate the absence of dissolved oxygen, a more thorough chemical assessment is required. If additional studies cannot indicate oxygen-free conditions, the site must be abandoned.

- Measured TDS values at repository level must be lower than 100 g/l. Occasional higher values can be accepted if it can be shown that the water is located in areas that can be avoided and that the water will not be able to flow to the repository area.

Besides these direct disqualifying criteria, which conform to most of the requirements discussed in Section 10.3.3.3, the suitability of the site can be questioned if a large fraction of the rock mass between fracture zones has a hydraulic conductivity > 10^{-8} m/s. High permeability of the rock requires local precision adaptation of the repository if the safety margins are to be met.

10.3.4 Underground Research Laboratory – Äspö

In its review of SKB's RD&D program 1989, SKN suggested that a "demonstration facility should take the place of the final repository in SKB's current planning", SKN (1990). The Government decision (Government, 1990) endorsed SKN's views and, in addition to a staged siting program, the Government suggested that SKB should explore the possibilities of "…including a demonstration-scale final repository as a stage in the process…". Thus, as part of its R&D work, SKB has constructed and operates the Äspö underground Hard Rock Laboratory (HRL), located near the Oskarshamn nuclear power plant.

The *Construction Phase* for the HRL took place was carried out in two Stages from 1990-1995; Stage 1 down to 330 m and Stage 2 to 460 m, with total tunnel length ~3600 m. During tunnel excavation, groundwater samples were collected from probe holes drilled ahead of the tunnel face. Much of the geochemical work carried out involved monitoring the changes that occurred to rock and groundwater as a result of the excavation / construction work. Planning for the *Operational Phase* included the development of advanced tools and support laboratory facilities for radionuclide migration experiments. Of particular note was the CHEMLAB probe, a "downhole laboratory", which allowed radionuclide migration experiments to be performed on rock samples within the probe using formation water.

During the *Operational Phase*, SKB has conducted a wide range of geoscientific research projects in the fields of geology, hydrogeology, geochemistry, and rock mechanics. SKB outlined the main goals of its HRL program (SKB 1989b). These were intended to:

- Test the quality and appropriateness of different methods for characterizing the rock mass with regard to conditions of importance for a geologic repository;
- Refine and demonstrate methods for adapting a deep repository to the local properties of the rock mass with regard to design, planning, and construction; and
- Collect material and data of importance for the safety of the geologic repository and for confidence in the quality of the safety assessments.

While additional goals were established as the schedule proceeded, the key objective for SKB was to understand the Äspö HRL rock mass properties as well as to demonstrate the technology for different components of the repository system.

Specific projects that have been carried out include (SKB, 2007b):

- Geological Mapping and Modeling;
- Development of New Technique for Underground Surveying
- Seismic Influence on the Groundwater System;
- Inflow Predictions;
- Hydro-Monitoring Program;
- Monitoring of Groundwater Chemistry;
- Rock Mechanics; and
- Äspö Pillar Stability Experiment.

In addition, the Backfill and Plug Test was used to investigate the hydraulic behavior of bentonite mixtures in parts of the EBS.

The Äspö HRL has served and continues to serve as a resource not only to SKB but also to numerous international organizations including all the main countries' national waste management implementing agencies.

10.4 Disposal Concept

SKB's fundamental conceptual design for disposal, known as "KBS-3" (SKBF, 1983; SKB, 1998; SKB, 2006), calls for the waste package to be encapsulated in copper/cast iron canisters, which will be disposed in vertical deposition holes and surrounded by a bentonite-clay buffer (Figure 10-5) at a depth of 400-700 m in the Swedish basement crystalline rock (i.e., 'granitic/gneissic').

The buffer barrier performance is particularly critical because it:

- Assures diffusion-limited transport of reactants to the copper surface assuring containment times >100,000 years,
- Assures diffusion-limited release of dissolved radionuclides into the geosphere,
- Prevents release of any radionuclide-bearing colloids to the geosphere, and
- Decouples the performance of the EBS from any future changes in the hydrological conditions of the geosphere.

In reviewing previous RD&D programs, SKI and the government had accepted SKB's suggested design as the main research option. The KBS-3 methodology was also accepted in 1984 by the government as a basis for licensing the two latest reactors, where it was necessary to show that there exists a feasible solution for final disposal of the waste. On the other hand, the government asked SKB to present alternatives to KBS-3, including the alternative of prolonged storage at the CLAB, partitioning and transmutation, and disposal in very deep boreholes (e.g., SKB, 2007; SKB, 2010).

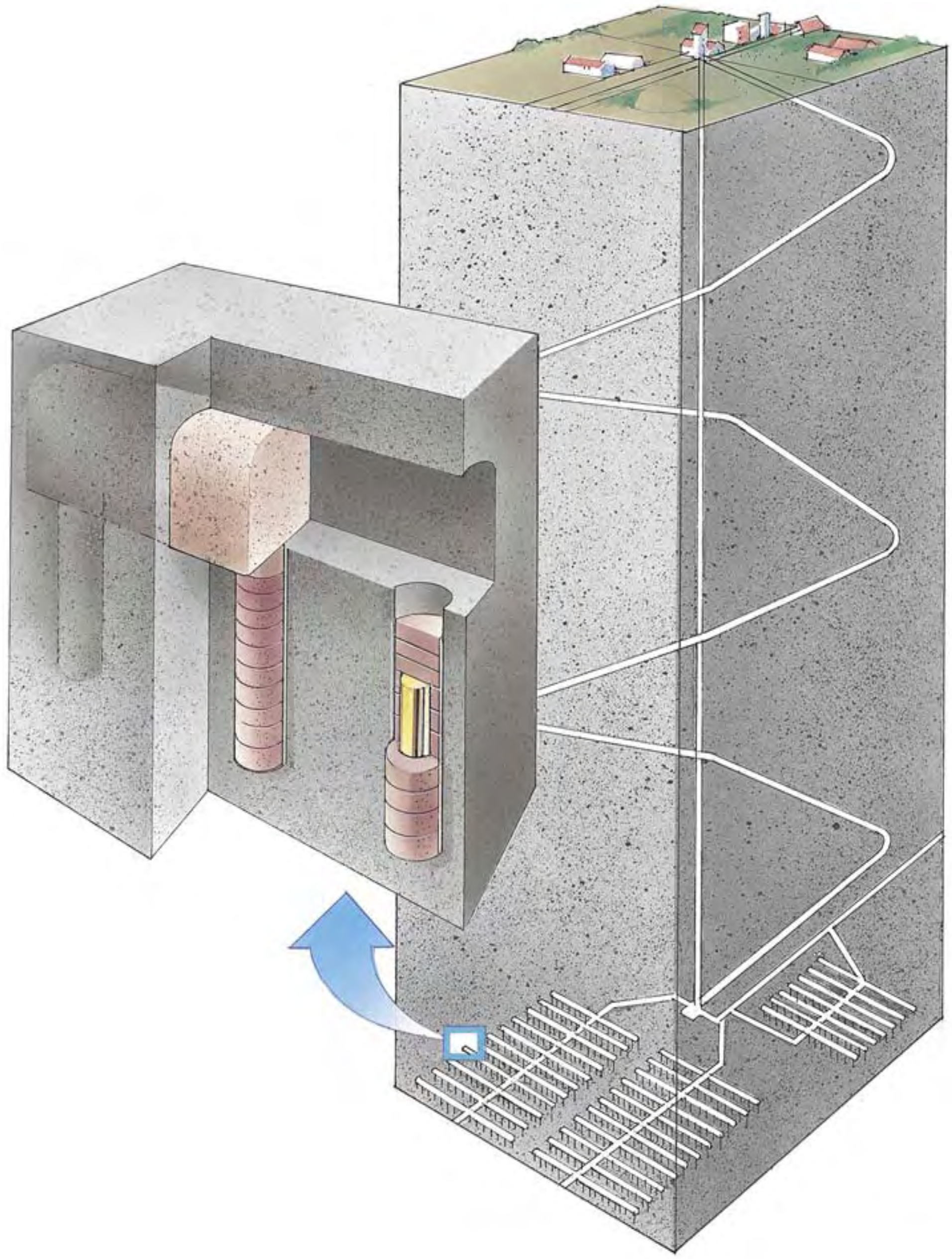

Figure 10-5
The KBS-3 concept accomplishes direct disposal of spent fuel in copper/iron canisters surrounded by bentonite clay and placed about 500 m deep in the Swedish basement rock (from SKB SR97, SKB, 1999). Used with permission of SKB.

10.5 Transparency and Stakeholder Involvement

10.5.1 Public Involvement

10.5.1.1 Feasibility Studies and the DIALOGUE Project

Before the present pre-licensing phase in Sweden, there had been a long history of close interactions among implementer, regulatory authority, local and national government representatives, and the general public. Such interactions have aided SKB in its development of the EIS that will be included in its license application.

An early initiative was taken by SKI in 1990, with the DIALOGUE project (SKI, 1993a). The project collected representatives from some authorities, some municipalities and some NGO's. SKB did not participate at that stage. The DIALOGUE project discussed, among other things, how to define procedures for site selection and licensing in which many different groups could have confidence. It appears that the conclusions of the DIALOGUE project had a profound impact on future actions taken by authorities, the Government and the municipalities.

When the Feasibility Studies started in 1994, there were extensive interactions between the stakeholders involved. SKB organized an extensive series of public information exchange meetings, some attended by the authorities. All of the municipalities were given the chance to review the feasibility study reports (e.g., SKB, 1997), the expenses for these reviews being covered by the NWF. Reviews were based on comments from experts invited by the municipality, and by comments from a multitude of organizations, political groups and members of the public in the communities. Examples of issues raised in the municipalities are (SOU, 1999):

- How to combine the necessary cooperation with the implementer with the need to critically review its studies?
- Will the community have access to sufficient competence for following and reviewing the future studies?
- How can one ensure that the public has full knowledge on what is going on?
- How does the municipality contribute to the ability of the public to evaluate a future suggestion to site a repository in the Municipality?

Also the Special Advisor in the nuclear waste area has taken some initiatives. Since 1997 he has organized an informal national EIS-forum on the nuclear waste area. The participants included representatives from the municipalities and the counties with feasibility studies, SKI, SSI, the Swedish Board of Housing, Building and Planning, the Environmental Protection Agency and SKB. Other stakeholders or information sources were invited when judged necessary. All records from the meetings were publicly available. The general public and NGO's were not part of the forum *per se*, but had access to all meeting minutes and any other materials produced. The purpose of the forum was to provide a platform for discussing issues of a more general character than the discussion in an individual municipality. A few meetings were held each year. The meetings have in particular discussed alternatives to the KBS-3 methodology and different ethical issues.

For preparing the EIS for a nuclear waste repository the proponent (e.g. SKB) needs to consult with the county administrative board, other authorities, the local municipality and with the public. Also local non-governmental organizations (NGO's) are considered part of the impacted public. The consultations need to start at such an early stage that it can practically lead to changes of the plan, including changes of the proposed siting. The EIS rules are also applied in other acts, including the Nuclear Activities Act and the Radiation Protection Act (see Section 10.2.2).

10.5.1.2 Consultations During Initial Site Investigations

Chapter 6 of the Environmental Code requires consultation by SKB with concerned national authorities, the participating municipalities, the public and organizations that can be expected to be impacted by siting decisions for both an encapsulation plant and a site for a geological repository. The formal consultation period started in 2000 when SKB announced the sites for "site investigations" (Government, 2000), and continue to the present day.

The consultations are aimed at dealing with siting and design activities, as well as the form and content of the eventual supporting EIS. SKB's goal with consultations is that everyone, from private citizens to concerned organizations to impacts local and national authorities, who wants to get involved in these aims should be given the opportunity to participate. The consultation process will continue until just before submittal of the license applications. SKB publishes an annual compilation and summary of its consultations.

10.5.2 International Involvement

As mentioned in Section 10.3.4, the Äspö HRL has been a valuable resource for URL experiments and testing for many national waste management agencies. In addition, SKB has bilateral agreements with most national implementing agencies and is active in other waste management programs via SKB International AB, a subsidiary of SKB. SKB International operates on a commercial basis and works towards assisting other countries increase safety and reduce environmental risks associated with the handling and storage of radioactive operational waste and used nuclear fuel.

SKB publishes all its technical documents, making them available in the public domain most of them electronically. SKB's SR 97 safety assessment was reviewed by an international group of OECD/NEA experts as well as by the Swedish regulator. In addition, SKB's SR-Can was reviewed by an international safety assessment methodology review team commissioned by the Swedish regulator (Sagar et al., 2008).

The Swedish regulator also cooperates extensively with international organizations, primarily within the IAEA (International Atomic Energy Agency), OECD/NEA (Nuclear Energy Agency) and the European Union (EU). SSM also maintains bilateral agreements with nuclear regulatory agencies in other countries.

10.6 Safety Assessment and Licensing

10.6.1 Safety Assessment

SKB has carried out periodic safety assessments periodically, approximately every ten years, the most recent assessment being SR-Can (SKB, 2006).

In this report, SKB formally compared the performance of a standard “KBS-3 type” repository design using site-specific data from Laxemar and Forsmark. A wide variety of expected and unlikely scenarios were evaluated for both sites, principally associated with effects from future continental-style glaciation advance and retreat across the site. In particular, impacts were calculated for (a) a buffer-erosion scenario driven by deep circulation of dilute glacial melt waters, and (b) a shearing scenario assuming a limited number of waste packages mechanically disrupted from re-activation of faults due to glacial unloading of the repository rock-block. The derived total risk assessments for both sites from SR-Can are shown in Figure 10-6. Both sites were shown to comply with the SSI’s 100,000-year risk criterion, and were accordingly judged to be suitable for selection as the repository site.

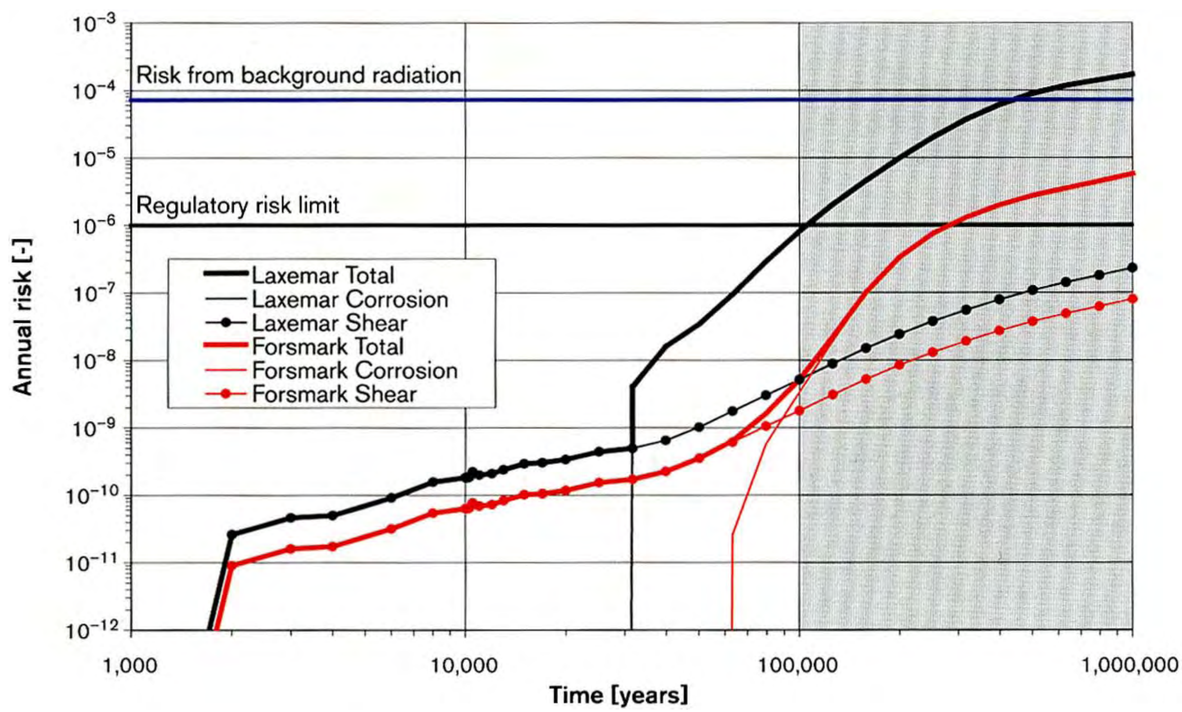

Figure 10-6
Summary risk calculations for the Laxemar and Forsmark sites from SR-Can (SKB, 2006). Used with permission of SKB.

Since SR-Can, an additional safety analysis commissioned jointly be SKB and its Finnish equivalent Posiva, has been carried out on the KBS-3H disposal concept, in which canisters containing used nuclear fuel are emplaced in a horizontal configuration rather than a vertical one (Smith et al., 2008), at a depth of about 400 m. For this assessment, the possibility of canister failure was addressed via a range of assessment cases, with the following canister failure modes:

- Initial penetrating defect;
- Canister failure due to corrosion;
- Canister failure due to rock shear.

Although the site selected for the safety analysis was Finnish (Olkiluoto), the assessment relied heavily on data from SR-Can (SKB, 2006). In all the calculation cases, the results complied with Swedish regulatory criteria (Smith et al., 2008).

10.6.2 Licensing Process

10.6.2.1 General Requirements Including an EIS

The general requirements and principles for management and disposal of Swedish used fuel are set forth in the *environmental requirements* of the Environmental Code, the *safety requirements* in the Nuclear Activities Act, and the *radiological protection requirements* in the Radiation Protection Act, as described in Section 10.2.2.

The construction of a nuclear waste repository (including shaft sinking) needs a license according to the Nuclear Activities Act as well as the Environmental Code. The license application is tried by the Government, but is first examined by the regulatory authority (now SSM – see Section 10.6.2.3), which also recommends a course of action. When issuing a license, the Government can also issue the stipulations thought necessary to ensure safety and radiation protection. The latter are to be based on the Radiation Protection Act, but a special license is not necessary if a license has been issued according to the Nuclear Activities Act.

The license application needs to include an EIS. Both the Nuclear Activities Act and the Radiation Protection Act contain rules on EIS, based on the rules in the Environmental Code. In common with all large projects, the EIS must include all the information necessary to make an overall assessment of the environmental impact and the effect of a project on natural resources. The EIS should also explore alternative sites and alternative designs, including the consequences if the project is not undertaken (“zero alternative”). The chosen alternative needs to be justified.

10.6.2.2 Preparation for Joint License Application for Constructing Encapsulation Plant and Repository

The first licensing step is for a permit to construct the encapsulation plant and repository underground facilities prior to operations and receiving radioactive material. In November 2006, SKB submitted an application under the Nuclear Activities Act for a permit to build and eventually operate an encapsulation plant at Oskarshamn. This application also contained an EIS, based in part on previous consultations as discussed previously.

Now that the Forsmark site has been selected (SKB, 2009), SKB has started planning for a detailed investigation for the Forsmark site, including plans to carry out detailed investigations from tunnels and shafts. According to a decision by the Swedish government in May 1995 (Government, 1995), these detailed investigations are seen as ”construction of a repository” and, thus, require a full license and nuclear safety case with, among other things, consent from the Municipality Council.

Since the CLAB interim storage facility, the encapsulation plant and the final repository are part of an integrated management system, SKB will apply for license permits for all of these facilities under the Environmental Code including a joint EIS that updates the previous EIS for the encapsulation plant. SKB’s approach involves separate but parallel licensing of the encapsulation plant (to be operated in coordination with the CLAB interim storage facility for used fuel) and a geological repository at the selected Forsmark site. The previous permits for CLAB will also be reviewed as part of the integrated licensing process.

10.6.2.3 Evaluation of the License Application for Construction

As stated in Section 10.6.2.1, SSM evaluates SKB’s license application, and, in order to do this:

- Sends out the material for review to a wide range of authorities and organizations,
- Starts an evaluation project,
- Issues requests for supplementary information, and
- Prepares an evaluation statement summarizing all views and concluding with recommendations to the Government for further actions.

Key groups to receive the application for review are all parties involved in the EIS process. When answers are received from reviewers, SSM historically has had the obligation to compile and comment the views put forward. If the SSM views deviate from the reviewers, SSM needs to further justify its position.

10.7 Current Status

SKB is currently preparing a series of reports centered on the SR-Site Report and developing its SR-Site documentation that will form the basis for the license application. The intended submittal date for this license application and SR-Site Report is early 2011.

SSM's main effort will be an evaluation project of SKB's license applications for CLAB, the interim storage facility, the encapsulation plant and the geological repository.

The expectation is that the implementer will need to produce a safety case that establishes its confidence that the safety regulations are met. SSM may choose to focus its review to identify and independently evaluate specific topics and issues that can be shown to be of high risk-importance. In addition, evaluations will likely be undertaken by SSM, ranging from simple quality checks of some calculations contained in the license application to exploration of additional scenarios or alternative conceptual models. If the results of these calculations by SSM lead to a deviatory view on key, risk-important factors, these results may be used as a basis for asking supplementary questions to SKB. No independent data will be collected by SSM, however - if additional data are needed, SKB will be asked to provide such data. SSM staff and management will then prepare the evaluation statement and conclusions, based on the review comments and the results of their own evaluation project, and send these to the Government.

A divergence in views between SSM and SKB may emerge regarding how the 'best available technology" clause of the safety standard should be applied in evaluating SKB's repository license application. In Section 13.3.4 of the SR-Can report (SKB, 2006), for example, SKB discusses its views on the potential conflict between BAT and optimization of the repository system. In this context, the key point may be whether primary emphasis should be placed on "best" or "available" aspects with respect to alternative technologies.

10.8 Summary and Key Observations

- *Policy on Geologic Disposal*: The current national policy actively being pursued is to dispose of used fuel directly in a deep geological repository. Sweden has selected a site and is proceeding toward submittal of a license application for construction planned for March 2011.
- *Institutional Arrangements*: SKB is the implementer responsible for radioactive waste management and is currently preparing the documentation necessary to support its license application for the construction of a geologic repository. The Swedish regulator is SSM, formed in 2009 from the merger of SKI and SSI. Regulations issued separately by these previous organizations have been reissued under SSM. With regard to funding, existing legislation requires the reactor owners to pay a fee to the Nuclear Waste Fund, which is overseen by a Board of Governors appointed by the Government. The average charge was 0.019 SEK per kWh between 1982 and 1996, but has since dropped to an average of 0.005 SEK per kWh.

- *Key Laws and Regulations*: The Environmental Code (SFS 1998:808) contains a requirement for the issuance of permits for siting of the deep repository and the preparation of an Environmental Impact Statement (EIS) as part of the license application. The rules on EIS are adopted from European Union Directives (85/337/EG and 97/11/EG). The Code also imposes limits on the environmental impacts of the deep repository. The Nuclear Activities Act stipulates that the owners of the reactors, as waste producers, are responsible for all measures needed to execute and fund management of the nuclear sector's lifecycle, including the disposition of nuclear waste and decommissioning of facilities. The Nuclear Activities Act also specifies the funding mechanism for radioactive waste management. The Radiation Protection Act prescribes in general that with regard to conducting activities with radiation, measures and precautions are needed to prevent harm to humans, animals and the environment. Ordinances issued by the Government pursuant to the Nuclear Activities Act and the Radiation Protection Act contain some more detailed provisions and regulate the activities of the nuclear regulatory authority SSM. SSM regulations also call on the implementer to consider optimization and best available technology.
- *Site Screening and Selection*: SKB's plan for identifying a site for a geologic repository identified three stages: general siting and feasibility studies, site investigations, and detailed investigations including subsurface. SKB were criticized at an early stage by the regulator and the government for not having site selection criteria to guide the site selection process. SKB updated its approach to include general siting factors based on engineering, safety, environmental and societal considerations. SKB was requested to make its factors more specific. In its 1998 research plan, SKB formulated the interim goal of being able to choose at least two sites for site investigations in 2001 and funded a project to quantify requirements and preferences for a suitable repository site for a more systematic approach to site selection. From six feasibility studies completed in 2000, SKB selected three sites (later reduced to two: Laxemar and Forsmark) for more detailed investigation in two stages. The initial site investigation stage was completed in 2006 with the publication of the SR-Can report, in which SKB formally compared the performance of a standard "KBS-3 type" repository design using site-specific data from Laxemar and Forsmark. Both sites were shown to comply with the regulator's 100,000-year risk criterion, and were accordingly judged to be suitable for selection as the repository site. Finally, after a multi-year set of site characterization programs, SKB formally announced in 2009 its selection of the Forsmark site as the site for the deep geologic disposal of used fuel in Sweden. As part of its research activities, SKB constructed the Äspö underground Hard Rock Laboratory between 1990 and 1995 and continues to carry out experiments and testing in this facility.
- *Repository Design Concepts*: SKB's basic conceptual design for disposal, known as "KBS-3", calls for the waste package to be encapsulated in copper/cast iron canisters, which will be emplaced in vertical deposition holes and surrounded by a bentonite-clay buffer at a depth of 400-700 m in the Swedish basement crystalline rock. The buffer barrier is an important component in terms of performance, providing a diffusive barrier both to the inward transport of reactants as well as the release of radionuclides once a canister has failed. More recently, SKB has also considered a KBS-3 design with horizontal rather than vertical emplacement of canisters.

- *Performance Metrics and Assessments*: The relevant regulator at the time SSI issued safety standard regulations in 1998 addressing the final disposal of spent nuclear fuel. These regulations stipulate an annual risk of injury after closure of no more than 10^{-6} for a representative individual in a group that is exposed to the greatest risk. This standard has a risk-based focus on the 1000 years after repository closure, a risk-compliance target out to 100,000 years, and qualitative examination of repository performance terminated at one million years. SKB has carried out periodic safety assessments periodically, approximately every ten years, the most recent assessment being SR-Can in 2006 in which SKB formally compared the performance of a standard “KBS-3 type” repository design using site-specific data from two sites, Laxemar and Forsmark. Impacts were calculated for a range of different scenarios, in particular (a) a buffer-erosion scenario driven by deep circulation of dilute glacial melt waters, and (b) a shearing scenario assuming a limited number of waste packages mechanically disrupted from re-activation of faults due to glacial unloading of the repository rock-block. Both sites were shown to comply with the SSI’s 100,000-year risk criterion, and were accordingly judged to be suitable for selection as the repository site.

- *Independent Peer-Review and Advisory Bodies*: SKB’s 1997 safety assessment was reviewed by an international group of OECD/NEA experts as well as by the Swedish regulator. In addition, SKB’s SR-Can assessment was reviewed by an international safety assessment methodology review team commissioned by the former Swedish regulator SKI, which has now been combined with SSI to form SSM. An advisory body to the government is the Swedish National Council for Radioactive Waste (formerly KASAM), consisting of Swedish experts in relevant technical areas. SSM is planning to have the NEA conduct an independent review of SKB’s 2011 license application.

- *Stakeholder and Public Involvement*: There has been a long history in Sweden of close interactions among implementer, regulatory authority, local and national government representatives, and the general public. When SKB’s studies started in 1994, there were extensive interactions between the stakeholders involved, including an extensive series of public information exchange meetings, some attended by the authorities. All of the municipalities were given the chance to review the feasibility study reports, with the expenses for these reviews being covered by the NWF. In order to prepare the EIS for a nuclear waste repository SKB needs to consult with the county administrative board, other authorities, the local municipality and the public. Local NGO’s are considered part of the impacted public.

- *Program Maturity*: The Swedish waste management program for used nuclear fuel is at an advanced stage, with the submission of a license application for the construction of an encapsulation plant as well as repository underground facilities expected in early 2011.

- *Licensing Process*: The construction of a nuclear waste repository (including shaft sinking) needs a license according to the Nuclear Activities Act as well as the Environmental Code. The license application is tried by the Government, but is first examined by the regulatory authority, which also recommends a course of action. The license application needs to include an EIS, which in turn requires a formal safety analysis. SKB is currently preparing documentation in support of a license application for the construction of both an encapsulation facility as well as repository underground facilities.

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11
SWITZERLAND

11.1 Introduction

11.1.1 General Nuclear Profile

Five commercial nuclear power reactors (3 PWR, 2 BWR) are in operation at four sites in Switzerland, generating 26 TWh in 2009 – 40% of the country's electricity supply. In 1990, while the public agreed to keep the existing reactors operating, there was a 10-year moratorium on granting a license to any new power reactor. Two new initiatives proposed in 1999 aiming to the ban of the construction of new NPPs until 2010 and mandate closure of all NPPs after a 30-year life-span were rejected by the public in 2003. Parliament then passed a new Nuclear Energy Law in 2005 allowing for the continued operation of the existing fleet and the construction of new reactors; the law and the corresponding Nuclear Energy Ordinance came into force in 2005. Subsequently, general license applications for three new nuclear units have been submitted and at least one has received strong support from the host canton (NEA, 2009; WNA, 2010a,b).

The Nuclear Energy Law leaves the decision to choose reprocess and recycle or directly disposal of used nuclear fuel in the hands of the utilities. The current strategy chosen by NPP operators includes both reprocessing and storage of irradiated nuclear fuel. Reprocessing of used fuel in France or UK is followed by MOX fuel fabrication for recycling in Swiss reactors and the generation of HLW for disposal. Single recycle of MOX also generates used MOX fuel for disposal. The 2005 Nuclear Energy Act brought initiated a ten-year moratorium shipments of used fuel abroad for reprocessing, starting in 2006.. The country is committed to development of a deep geologic repository capable of accommodating both used fuel and HLW, although participation in a future multinational program appears as a possible alternative.

11.1.2 Used Fuel / HLW Inventory and Projected Disposal Requirements

As of the end of 2007, the amount of irradiated fuel in storage in Switzerland was 856 MTHM; an additional 1,139 MTHM had been shipped abroad for reprocessing (UVEK, 2008). Contracts between the Swiss NPP operators and foreign reprocessing companies (AREVA NC in France and NDA in the UK) provide for the reprocessing of approximately 1,200 MTHM of used fuel. A total amount of 4,200 MTHM of irradiated nuclear fuel (regardless of reprocessing or direct disposal path) is projected to be discharged from the current fleet (UVEK, 2008); this assumes a 60-year operational life for each reactor. The reference inventory used by Nagra in its safety assessment Project Opalinus Clay (Nagra, 2002) was:

- Irradiated UOX and MOX fuel: 2065 canisters containing 3,217 MTHM;
- HLW: 730 canisters containing HLW from the reprocessing of 1,195 MTHM irradiated fuel;
- ILW: 1680-1880 canisters of long-lived ILW with total volume 1,400 m^3.

All used fuel and HLW is stored at reactor sites in wet and dry configurations and a centralized interim dry storage facility. The Central Storage Facility (ZZL) operated by the company ZWILAG in Würenlingen began operation in 2001 and provides capacity for irradiated fuel, HLW and other radioactive waste classes. At Beznau NPP, a dedicated storage facility (ZWIBEZ) includes a hall for dry storage of used fuel. Additional wet storage capacity was added at Gösgen NPP in the form of a spent fuel pool. By the end of 2007 about 50% of vitrified HLW had been returned from France to Switzerland for interim storage at ZZL (UVEK, 2008).

11.2 Institutional Arrangements

11.2.1 Implementation framework

Figure 11-1 shows the key players concerning radioactive waste management in Switzerland. The following key bodies make up the regulatory and licensing framework for radioactive waste management in Switzerland (Swiss acronyms used):

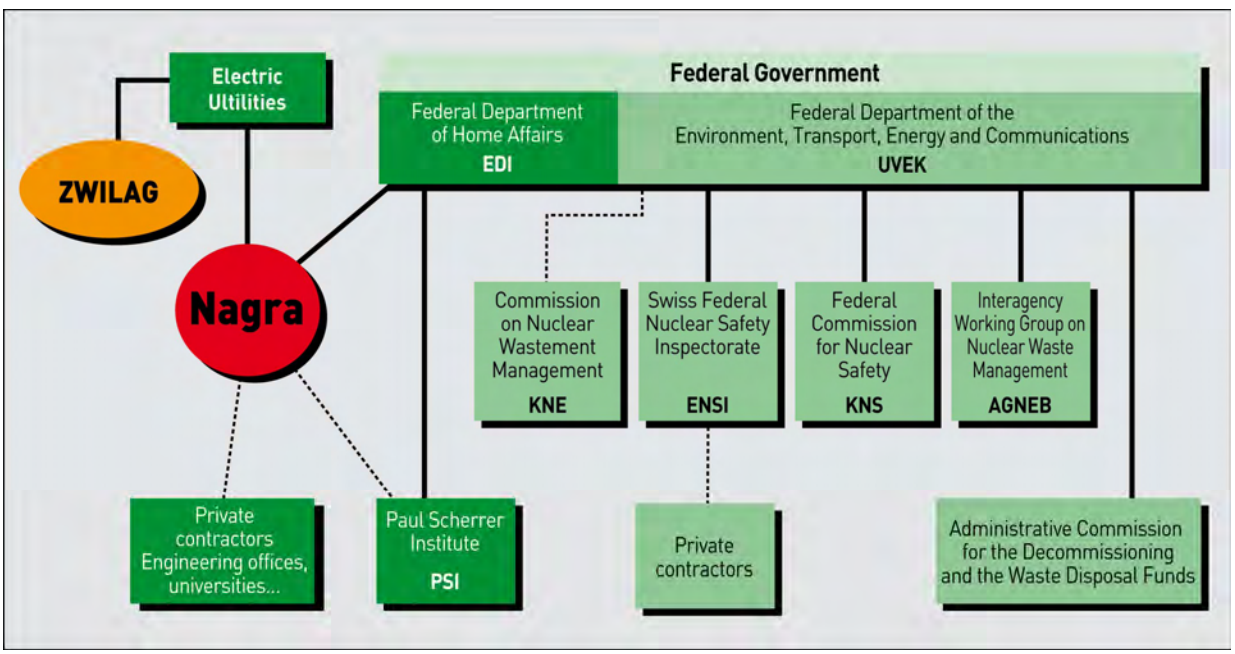


Figure 11-1
Institutional framework for nuclear waste disposal in Switzerland (UVEK, 2008). Image courtesy of Nagra. Copyright Swiss Federal Nuclear Safety Inspectorate (HSK), 2008. Used with permission.

POLICY and OVERSIGHT – The Federal Council (federal government) issues the general license that is initially needed for each nuclear facility. The general license has to be approved by Parliament and is subject to a facultative referendum at the national level. The Federal Council also issues the closure order for disposal facilities. Other key federal departments and agencies include:

- *Federal Department of Environment, Energy and Communication (UVEK)*: Issues construction and operational licenses, as well as decommissioning orders.
- *Federal Office of Energy (BFE)*: Issues other types of license (transport, transit, import and export).
- *Federal Department of Home Affairs (EDI)*: EDI represents the Swiss Confederation in the Nagra Cooperative.

IMPLEMENTER – The National Cooperative for the Disposal of Radioactive Waste (Nagra) was formed by the operators of nuclear power plants and the federal government as a public-private entity. Nagra is responsible for the disposal of all kinds of radioactive waste, including irradiated nuclear fuel if declared as waste, with a view to implementing its permanent and safe disposal.

REGULATOR – The Swiss Federal Nuclear Safety Inspectorate (ENSI, formerly HSK) serves as the Swiss government's independent regulatory authority (established by the ENSI Act in 2009) responsible for the safety of nuclear energy facilities and installations, including radioactive waste disposal.

ADVISORY and SUPPORT – A number of entities serve in advisory roles with respect to the management of radioactive waste in Switzerland. These include: the Federal Nuclear Safety Commission (KNS), Inter-Departmental Working Group on Radioactive Waste Management (AGNEB), Geological Commission on Nuclear Waste Management (KNE).

11.2.2 Legal and Regulatory Framework

The legislation governing irradiated nuclear fuel and radioactive waste management consists mainly of the following laws and ordinances (Swiss acronyms):

- Nuclear Energy Law (KEG);
- Radiological Protection Act (StG);
- Nuclear Safety Inspectorate Act (ENSIG);
- Nuclear Energy Ordinance (KEV);
- Radiological Protection Ordinance (StSV);
- Ordinance on the Decommissioning and Waste Management Funds;
- Ordinance on the Collection of Radioactive Waste.

The legal framework regarding nuclear energy was extensively revised in recent years, resulting in the new Nuclear Energy Law (KEG, 2003), which came into force in February 2005, together with the corresponding Ordinance (KEV, 2004). The new Energy Law keeps the nuclear option open and addresses a number of key issues related to radioactive waste, including a concept of monitored long-term geological disposal of radioactive waste that combines elements of final disposal and reversibility. A stronger commitment from the Swiss Federal Government for the implementation of repositories for all types of waste is generally required. Furthermore the legislation (KEG, 2003) introduced a 10-year-moratorium on the export of nuclear fuel for reprocessing from 2006 to 2016. It includes provisions for decommissioning and a system for funding the costs of decommissioning and of radioactive waste management. It also simplifies licensing procedures and introduces the general right of appeal.

StG and StSV form the legal basis for operational radiation protection in Switzerland. The Radiological Protection Ordinance sets the dose limit for individuals of the population to 1 mSv per year.

KEG (2003) requires the safe and permanent management and disposal of all radioactive waste, with long-term safety of disposal facilities detailed in ENSI's Regulatory Guideline ENSI-G03. The overall objective of radioactive waste disposal and the principles to be observed, which are stated in the guideline ENSI-G03, are derived from the internationally agreed IAEA requirements. The safety requirements are expressed in the form of protection objectives. In order to determine whether the protection objectives have been fulfilled, quantitative protection criteria are used:

- For each future evolution classified as likely, the release of radionuclides may not lead to an individual dose exceeding 0.1 mSv per year.
- Future evolutions classified as less likely that are not considered under [the previous] protection criterion may not, taken together, constitute an individual radiological risk of fatality exceeding one in a million per year. There is no time cutoff in the Swiss regulations.

Compliance with the protection criteria has to be shown in a safety demonstration.

All radioactive waste is to undergo final disposal in repositories situated in suitable geological formations - surface and near-surface disposal is not allowed. Thus, two geological repositories are foreseen, one for L/ILW and the other for LL-ILW and HLW (including irradiated nuclear fuel if not reprocessed). At present, the need for the operation of a HLW repository is not acute, due in part to the cooling time in storage required.

The Swiss nuclear energy legislation requires a feasibility demonstration of safe and permanent disposal of radioactive waste. For L/ILW, this demonstration was formally approved by the Federal Council in 1988. For LL-ILW and HLW, the feasibility project was completed in 2002 and the demonstration approved by the Federal Council in 2006. According to the nuclear energy legislation, radioactive waste generated in Switzerland must in principle be disposed of domestically. However, disposal of waste within the framework of a bilateral or multilateral (i.e., multinational) project is maintained as an option, but not actively pursued.

For both L/ILW and HLW repositories, the Nuclear Energy Ordinance (KEV, 2004) states that the site selection procedure must be defined in a so-called "Sectoral Plan" within the framework of the existing legislation on spatial planning. In the past both 'voluntary' and 'nominative' approaches have been followed for site selection for geological disposal. In contrast to the recent trend toward the 'voluntary' approach in several countries, Switzerland has adopted a 'nominative' site selection approach that was successfully initiated in 2008. The Swiss experience with this siting approach is discussed in detail in Appendix A, including lessons learned from the its application and key characteristics that differentiate the Swiss aproach from a 'traditional' nominative approach.

11.2.3 Waste Classification

StG and KEG both define radioactive waste as "*… radioactive material or radioactively contaminated material that is not further used*." Thus, irradiated nuclear fuel for which disposal without reprocessing is foreseen, is radioactive waste by definition and must be disposed of accordingly. To date in Switzerland, no irradiated nuclear fuel has definitively been declared as waste by its owner.

KEV defines the following classification of radioactive waste:

- HLW (HAA): Irradiated fuel elements that are no longer used as well as vitrified fission product solutions resulting from the reprocessing of used fuel elements.
- Alpha-toxic waste (ATA): Waste with a concentration of alpha-emitters exceeding 20,000 Bq/g of conditioned waste.
- L/ILW (SMA): All other radioactive waste.

HAA corresponds to the IAEA class HLW. ATA and SMA roughly correspond to the IAEA classes LL-LILW and SL-LILW, respectively.

11.2.4 Funding

Based on the Ordinance on the Decommissioning and Waste Management Funds for Nuclear Facilities, two funds were established to ensure sufficient resources for the decommissioning of nuclear facilities and the final management of irradiated nuclear fuel and radioactive waste. One of these, the Waste Management Fund, covers the management costs arising after shutdown of the nuclear reactors. Current expenditures related to irradiated nuclear fuel reprocessing and storage of irradiated nuclear fuel and radioactive waste, as well as to research and development, planning, geological investigations and, eventually, construction and operation of disposal facilities, are continuously paid for by the reactor operators as part of their annual budget.

The Waste Management Fund covers waste disposal costs up to, and including, the closure of the repository. Under the Nuclear Energy Act, funding is achieved via a levy of about 1 Swiss cent/kWh of nuclear power production.

11.3 Site Screening, Selection and Characterization

11.3.1 Geological Studies - Crystalline

Zuidema et al. (2006) note that site selection for HLW disposal is constrained by the small size of Switzerland and by its geological setting. Current geological consensus concludes that uplift of the Swiss Alps, of ~1-2 mm/year, is likely to continue. Thus, excluding alpine areas and other complex geological structures associated with the Jura Mountains and the Rhine Graben leaves only limited areas in Central and Northern Switzerland that would be potentially suitable. Within this area, three host-rock variants have been considered – either the crystalline basement or one of the two overlying, low-permeability sediment layers (first priority Opalinus Clay, Lower Freshwater Molasse as a reserve).

The regional investigations of the crystalline basement were completed and documented in 1984 (Nagra, 1984a; 1984b). Geological studies showed that the extent of accessible crystalline basement was much less than originally thought because of the presence of a previously unknown, extensive Permo-Carboniferous trough that dissects the region. Thus, only two relatively restricted areas remain for the selection of a possible site, each covering an area of about 50 km^2. Despite this limitation, Nagra still believed that it would be feasible to find a suitable repository for the required low volume of waste (Nagra, 1994a).

11.3.2 Geological Studies - Sedimentary

The Federal Government also required that sediments be investigated as alternative potential host rocks for the disposal of HLW. Thus, in parallel with the crystalline basement studies, investigations of the sedimentary options proceeded from desk studies to select potential host formations to identification of potential siting areas. The two sedimentary host rocks investigated in detail were Opalinus Clay, which exists in a laterally extensive but rather thin layer in Northern Switzerland, and Lower Freshwater Molasse, where the formations are large but somewhat heterogeneous (Nagra, 1994b).

Nagra identified Opalinus Clay as the higher priority option and conducted a field program in the "Zürcher Weinland" potential siting area. Based on existing seismic data supported by information obtained from the previous drilling of deep boreholes, this host rock was expected to form a fairly simple, almost flat layer in the Zürcher Weinland exploration area. Because it showed good velocity contrast with neighboring formations, the site characterization focused on close-mesh 3-D reflection seismics (Nagra, 2001a). Given the densely populated country like Switzerland, the seismic work presented a considerable logistical challenge. However, detailed processing of the resulting database allowed direct visualization of displacements within the Opalinus Clay, of a magnitude 3-10 m. Additional, state-of-the-art, image processing allowed continuous structures with even smaller displacements to be clearly identified.

Interpretation of the seismic profiles was calibrated by an exploration borehole drilled in the middle of the study area (Nagra, 2001b). This not only confirmed the 3D seismic results, but also provided basic information on the rock-mechanical, hydrogeological, lithological, and geochemical properties of the Opalinus Clay and surrounding formations. These characteristics were further supported and complemented by studies off site at the Mont Terri underground test site (see Section 11.3.3 below) and observations in other deep boreholes (e.g., hydrocarbon exploration) and tunnels intersecting these sediments.

11.3.3 URL Programs

Despite having a relatively small program, Switzerland is in the fortunate situation of having two major underground test sites—Grimsel in crystalline rock and Mont Terri in Opalinus Clay.

The Grimsel Test Site (GTS) is situated below ~ 500 m of overburden in granite/ granodiorite of the Swiss Alps. Over a period of almost 20 years, this site has become an international center for multi-phase *in situ* research supporting nuclear waste management. Earlier phases of work concentrated more on development and testing of methodology to characterize the subterranean environment. More recently, emphasis has shifted towards large-scale demonstration of technology for waste emplacement, testing models of evolution for the engineered barrier system and the immediately surrounding rock, development and testing of monitoring technology, and long-term studies of processes influencing radionuclide migration in natural or perturbed (high pH plume, colloids) fracture flow systems

Nagra and 17 other organizations from nine countries involved in work at the GTS are now investigating options for a further phase of work when the present "Phase V" ended in 2002/2003. GTS now serves as an international center of excellence for specific *in situ* studies where considerable experience has been gained (Zuidema et al., 2006).

The second test site, at Mont Terri in the Jura Mountains, also provides horizontal access to the Opalinus Clay under an overburden of about 300 m. The Mont Terri project (managed by the Swiss Federal Office of Water and Geology) is now entering a seventh one-year phase involving Nagra and partner organizations from six countries. At this site, continuing projects characterize the thermal, hydraulic, mechanical and geochemical properties of this rock. (Owing to the rock's high content of swelling clay minerals, such properties tend to be inherently coupled.) Work has, however, also commenced on studying solute migration in this rock (constrained by negligible advective flow in undisturbed rock) and engineered barrier system emplacement/evolution.

11.4 Disposal Concept

11.4.1 General Philosophy

Two objectives for permanent disposal were stated in the original Swiss regulator's Guideline HSK-R-21:

- The repositories must ensure the safety of humans and the environment from the harmful effects of radiation, which translated to the safety requirement that individual annual doses resulting from "*realistically assumed processes*" must not exceed 10 mrem per year, i.e. 0.1 mSv/year. As stated previously (Section 11.2.2), there is no cutoff time in the Swiss regulations.

- The responsibility for radioactive waste disposal must not be passed on to future generations. This meant that a repository must be able to be closed/sealed within a few years of being filled with waste.

Nagra's conceptual repository design was developed taking into account the potential host rocks, the very low volumes of HLW expected, and the government requirement for an early, convincing demonstration of waste disposal safety as a condition of extending reactor operating licenses. These factors together led to designs that were considered robust (or even overdesigned), but were not regarded as optimized in an economic or operational sense.

The Swiss disposal concept (Zuidema et al., 2006) has the following features:

- Deep disposal, about 500 m to 1 km below surface, in a specially constructed facility;
- In-tunnel emplacement of HLW waste packages in a geologic medium (sediment or crystalline basement) that physically protects the EBS, has low water flows and favorable groundwater chemistry, and acts (especially for Opalinus Clay) as an efficient radionuclide transport barrier;
- Engineered barriers: In addition to the vitrified waste in its steel fabrication canister or irradiated UO_2/MOX fuel within its cladding, a thick steel overpack is envisaged, surrounded by compacted bentonite clay.
- Co-disposal of long-lived ILW, but in tunnels in a separate part of the repository.

The role of the geosphere with regard to repository safety includes:

- Protection against deliberate human intrusion;
- Protection against erosion;
- Protection from the effects of tectonic strain and earthquakes;
- Low groundwater supply and low flow rates in repository area;
- Long flow paths combined with low groundwater flow ensures long migration times for radionuclides once they are released from the near-field
- Favorable groundwater chemistry (reducing conditions) - results in low corrosion rates as well as low solubility limits and increased sorption (retardation).

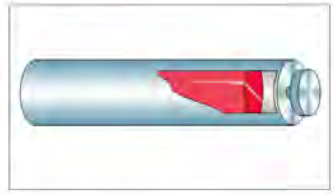
As shown in Figure 11-2, Nagra proposed a multi-barrier concept for delaying the release of radionuclides to the biosphere:

- Waste cylinders (carbon steel for HLW) surrounded by compacted layer of bentonite, so that diffusion is the only transport mechanism;
- Geological formation is selected for long radionuclide travel times, including natural rock retardation mechanisms.

Safety barrier system for HLW

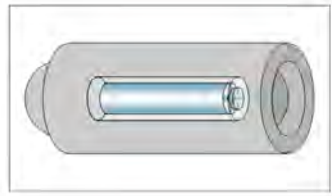
Waste matrix (glass)

- Fixation of radionuclides in the glass matrix
- Low glass corrosion rate



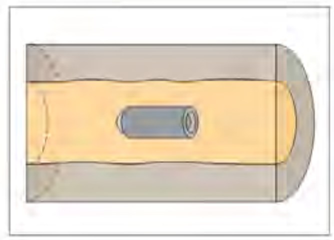
Disposal container

- Containment of the waste for several thousand years
- Thereafter: Restricted water inflow and retention of radionuclides (sorption on corrosion products)



Backfill (bentonite)

- Suitable interface between disposal containers and the host rock
- Favourable radionuclide retention properties
- Favourable conditions for long-term stability of the disposal containers



Host rock

- Low groundwater flux
- Favourable structure of the pore space
- Favourable geochemical conditions for radionuclide retention
- Favourable conditions for long-term stability of the engineered barriers

Geological situation

- Favourable host rock configuration for arrangement of the disposal tunnels
- Favourable conditions for long-term stability of the barrier system
- Absence of raw materials deposits that would be workable in the foreseeable future

Placement of disposal tunnels deep underground

- Isolation of the waste
- Prevention of unwanted access
- Protection from unintentional intrusion
- Protection from erosion

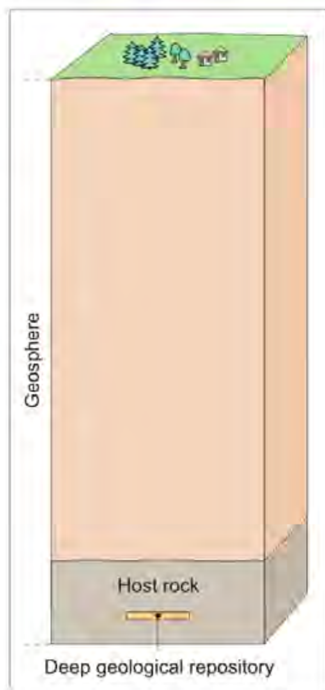


Figure 11-2
Basic disposal concept for HLW disposal in Switzerland (Nagra, 2009). Used with permission of Nagra.

In its main report on Project Gewähr (Nagra, 1985), Nagra noted, with regard to optimization, that the data in the safety assessment were not optimized but used only to demonstrate that the protection objectives of the repository "*are met with sufficiently safety reserve*". It was also emphasized in the report that site selection was not the focus of Project Gewähr. For Project Opalinus Clay (Section 11.6.1.3), Nagra envisaged that the emplacement tunnels for HLW and used fuel would be excavated as needed and backfilled and sealed concurrently with waste emplacement, so that a given tunnel would remain open for a maximum of two years.

11.4.2 Repository Layout

The proposed repository layout for a repository in Opalinus Clay is shown in Figure 11-3. Separate areas are provided for HLW or irradiated nuclear fuel, and LL-ILW, with silos for alpha-containing ILW (i.e., containing transuranic nuclides) immobilized in bitumen and cement.

11.5 Transparency and Stakeholder Involvement

11.5.1 Public Involvement

The licensing procedure includes a public consultation. Documentation on the project, including the safety analysis report, the regulatory review report and the views and opinions of the cantons, is made available, and any person (including those from neighboring countries) can give input or raise objections. KEG specifically requires that the site canton as well as neighboring cantons and countries, are involved in decision-making with regard to the general license. Furthermore, bilateral agreements have been established between Switzerland and neighboring countries (France, Germany, Austria and Italy) in order to exchange information on planned or operating nuclear facilities that are situated close to the common national borders. The involvement of the public and other stakeholders with regard to site selection is discussed in Section 11.7.

11.5.2 International Involvement

Apart from active participation in the IAEA and the NEA and the in-depth R&D activities ongoing at the two URLs in Switzerland, Nagra has formal agreements with European Community (EC), Belgium (CEN/SCK), the Czech Republic (RAWRA), Spain (ENRESA), France (CEA/ANDRA), Germany (GSF/ BGG), Japan (JNC, CRIEPI, NUMO, JNFL a n d Obayashi), Taiwan (AEC/FCMA and INER), Sweden (SKB), Finland (Posiva), USA (DOE and NRC) and the UK (NIREX). Additional informal collaborations add to the above the list.

Since 1997, Nagra has also expanded the provision of technical support services to organizations in other countries as well as applications outside the nuclear waste management field. Nagra believes that this approach has advantages for both Switzerland and those other countries – the experience accumulated at considerable cost (~800 M SFr over ~30 years) within the Swiss national program can be made available for other purposes, while Nagra staff have the possibility to further widen their experience, which provides a perspective that is essential for such a small program (Zuidema et al., 2006).

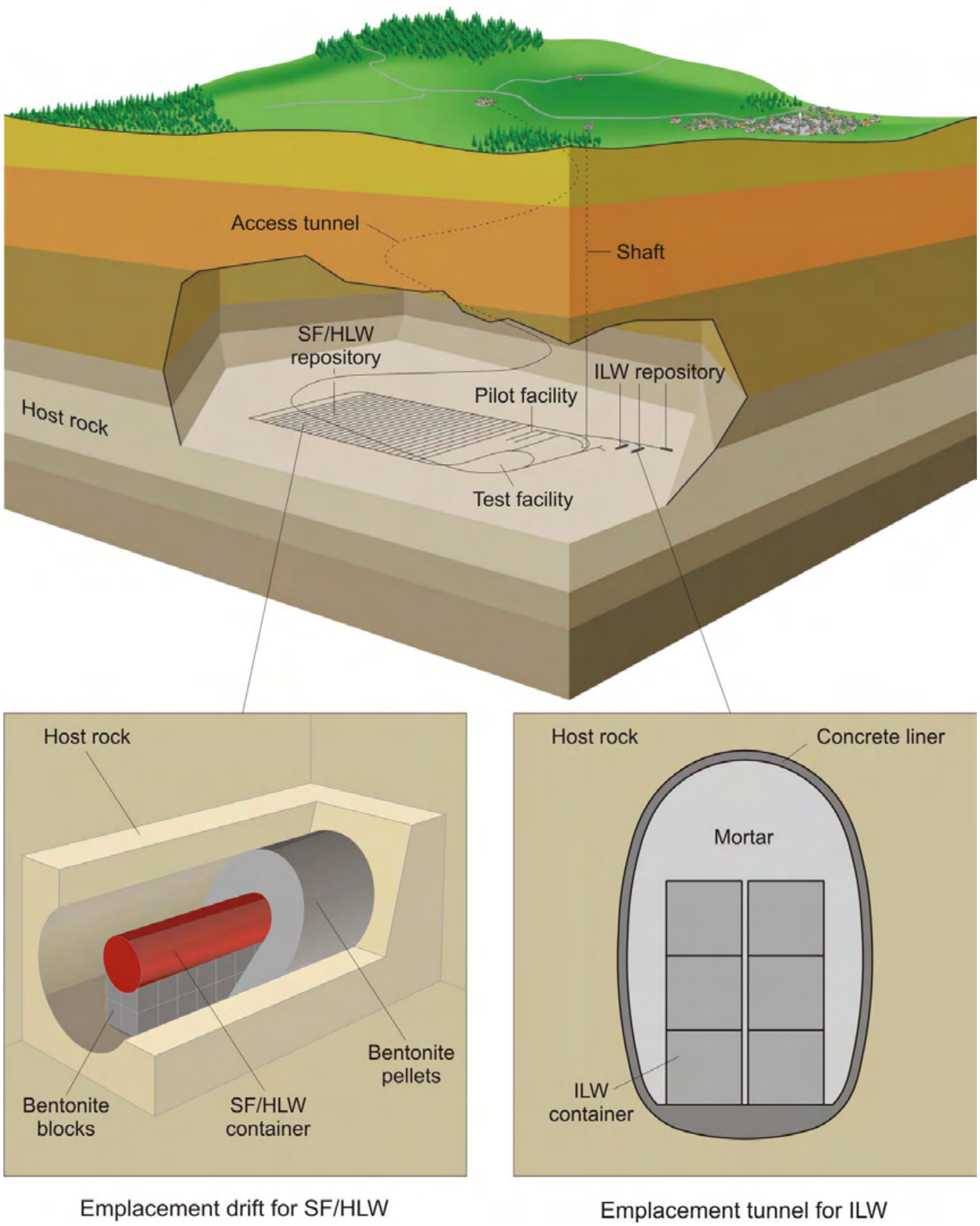


Figure 11-3
Envisaged layout for a repository in Opalinus Clay (Nagra, 2009). Used with permission of Nagra.

11.6 Safety Assessment and Licensing

11.6.1 Safety Assessment

As discussed in the next few sub-sections, a number of assessments have been carried out by Nagra: two addressing disposal in crystalline rock and one addressing disposal in sedimentary formations.

11.6.1.1 Project "Gewähr"

The Project Gewähr assessment (Nagra, 1985) covering disposal of HLW and LILW in crystalline rock was carried out according to the HSK Guideline, which specified that deterministic safety analyses should be carried out and that error limits and uncertainties were to be estimated.

Careful evaluation of the inventory of radioactive waste was based on the assumption of a total output (from all NPPs) of 6 GW per year, 8 NPPs, and an average lifetime for each NPP of 40 years, i.e., the “240 GWa scenario”. Reprocessing of irradiated nuclear fuel was also assumed, although the presence of irradiated nuclear fuel assemblies was given limited consideration, e.g., greater tunnel lengths to accommodate the longer waste packages.

Project "Gewähr" considered a repository for Type C radioactive waste (HLW and LL-ILW) as well as for Type B (short-lived L/ILW) radioactive waste, in the crystalline basement of northern Switzerland. Initially, 30 groups of radioactive waste were identified, but this number was reduced to three types for disposal in the Type C repository, *viz.*

- Vitrified HLW (1,120 m^3);
- Hulls and end caps from reprocessing (5,600 m^3);
- LL-ILW, bitumenized precipitates and concentrates (4,320 m^3).

Nagra (1985) defined three steps in its safety evaluation:

- Specification of the repository concept;
- Scenario analysis;
- Determination of consequences in terms of radiation doses.

The host rock selected was Böttstein granite within a Permo-Carboniferous trough (geologically/tectonically stable region) in the crystalline basement of northern Switzerland, at depths in the range 300-1,500 m. This choice was based largely on surface investigations (seismic) as well as borehole data. Ambient rock temperature was 55 °C based on the maximum temperature (100 °C) established by Nagra for the bentonite backfill.

As shown in Figures 11-2 and 11-3, the repository concept consisted of a system of mined tunnels and silos at a depth of ~1,200 m in crystalline basement of northern Switzerland. The repository was assumed to be located in a stable granite block, several kilometers wide, between two major faults. The diameter of the parallel tunnels was 3.7 m and no lining was envisaged.

The primary function of the canisters was to ensure containment for a specific length of time, in the case of Nagra, 1000 years. Nagra's reference canister (GS 40) was a self-supporting canister shell made of cast steel for HLW, maximum length 2.0 m and maximum outer diameter 940 mm., with cylinder wall thickness 250 mm, taking into account a corrosion allowance of 50 mm. After 1,000 years, corrosion was estimated to reduce the thickness of the steel by 29 mm. Thus, the realistic lifetime of a canister was expected to be much greater than 1000 years.

The bentonite buffer was intended to provide a mechanical and hydraulic protection zone. A sand-bentonite backfill was designed for good structural properties and low water permeability.

In terms of scenario analysis and for a repository in granite below the water table, the Base Scenario involved release of nuclides into groundwater and subsequent transport to the biosphere. Nagra's approach to the existence of low-probability scenarios is shown in Table 11-1.

Table 11-1
Nagra's treatment of release scenarios in Project Gewähr (Nagra, 1985)

Consequences	High Probability	Low Probability
Significant	Avoided by repository concept and design	Quantitative discussion
Insignificant	Quantitative consequence analysis	Qualitative discussion

Geological scenarios that were deemed relevant to the long-term safety, i.e., over periods of 10^6 to 10^7 years, comprised two generalized scenarios in the context of recent uplift of ~1.0-1.5 mm/year:

- Residual isostatic readjustment following the end of Alpine orogeny[30];
- Continuing Alpine orogeny with horizontal movements of ~2-3 mm/year causing the observed uplift.

A climate scenario (glaciation) was also taken into account.

The Base Case assumed that groundwater enters the waste-filled storage caverns and eventually reaches the surface of the waste canisters. The canisters corrode and are assumed to fail simultaneously after 1,000 years. After failure, the waste matrix begins to corrode and radionuclides are released. These radionuclides diffuse through the bentonite buffer and reach the host rock.

[30] Severe structural deformation of the Earth's crust.

Retention is achieved through limited solubilities for the first barrier (waste matrix), and is further enhanced by:

- Incorporation of radionuclides in leach- and corrosion-resistant glass matrix;
- Diffusion resistance from compacted bentonite;
- Steel canister providing reducing conditions even after canister failure;
- Low groundwater recharge and favorable water chemistry (reducing conditions).

Radionuclides are transported from the near field through the geosphere to overlying aquifers and eventually to surface waters. Doses result primarily from radionuclide intake via drinking water from a well and food.

The resultant maximum possible / peak dose from the 'realistic' case was 6 x 10^{-8} mrem/year (6 x 10^{-10} mSv/year) due mainly to Pd-107 and Cs-135. Conservative parameters were selected in combination to yield a 'conservative' case in which the peak dose was ~10^{-6} mrem/year (10^{-8} mSv/year). Again, the key (dose-contributing) nuclide was found to be Cs-135 due mainly to a low value for the nuclide's sorption coefficient. Nagra noted that a peak dose close to the regulatory limit of 10 mrem/year (0.1 mSv/year) could be reached only via a combination of many conservative assumptions, which did not correspond to "*realistically conceivable*" that is included in the Swiss Guideline R-21.

11.6.1.2 Kristallin-I

The Kristallin-I safety assessment provided a re-evaluation of HLW disposal in the crystalline basement of northern Switzerland as a host rock (Nagra, 1994a). This more recent assessment took advantage of additional geological, hydrogeological and hydrochemical data that had been collected since the earlier safety report, as well as improved assessment tools. Kristallin-I is considered an update of the part of Project Gewähr that dealt with HLW and LL-ILW, but also reflected an updated approach (see Figure 11-4) to demonstrating safety via a robust *safety case* (McCombie et al., 1991).

The Protection Objectives (PO) for Kristallin-I were similar to those for Project Gewähr except that for PO-2, the concept of risk was introduced: "*the individual radiological risk of fatality from a sealed repository subsequent upon unlikely processes and events not taken into consideration in Protection Objective 1 shall, at no time, exceed one in a million per year*."

Again, no time cutoff was specified for post-closure assessments. Nagra addressed and discussed uncertainty in the context of:

- Future evolution of the repository,
- Appropriate models for the relevant features, events and processes, and
- The data and parameter values selected.

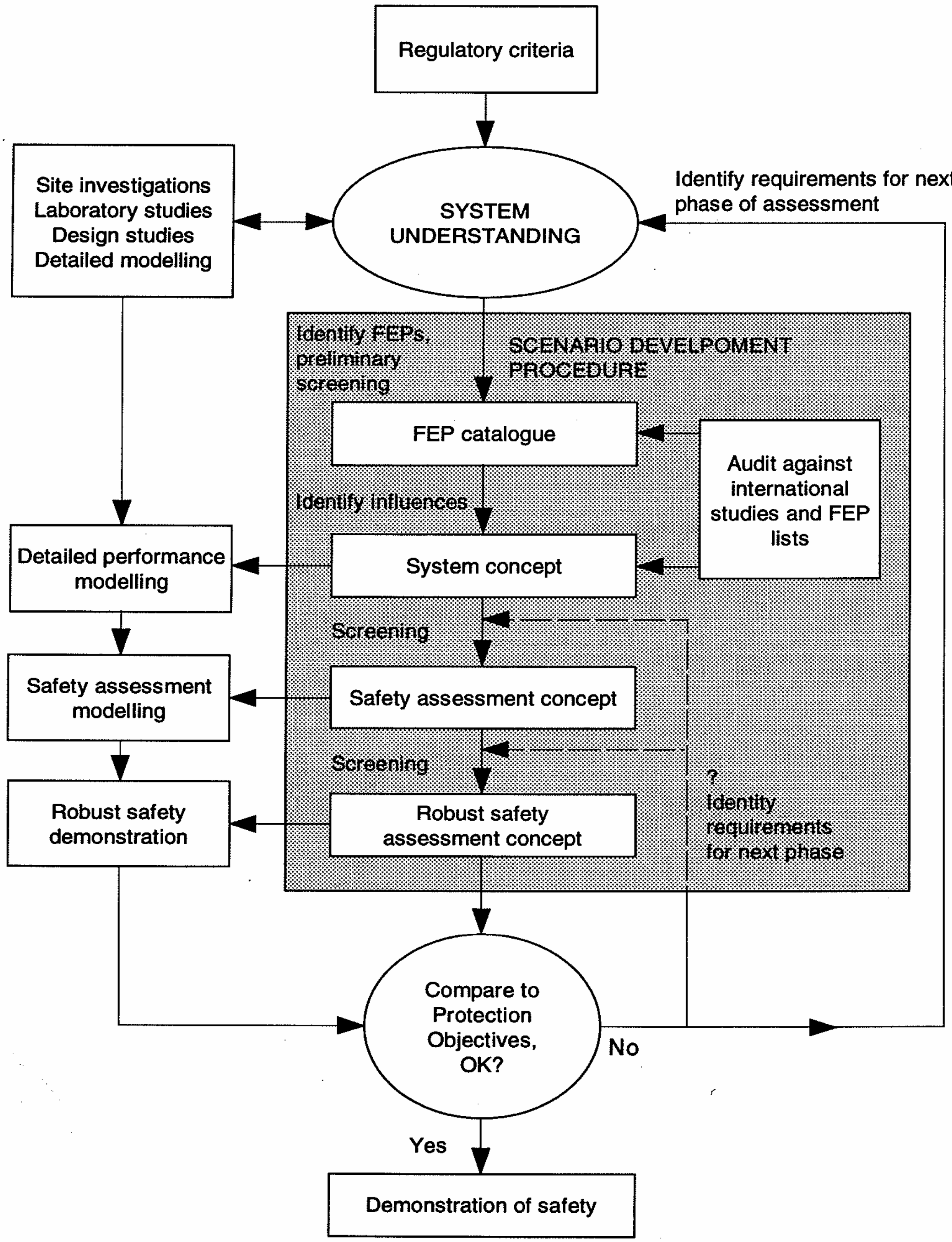


Figure 11-4
Nagra's approach to demonstrating safety for Kristallin-I (Nagra, 1994a). Used with permission of Nagra.

The nature of the updated assessment was again deterministic, and Nagra noted that conservatisms were introduced throughout the assessment assumptions and calculations such that the results should not be considered as best or central estimates of performance. McCombie et al. (1991) note that in a robust disposal system, most phenomena (features, events, processes) that could be harmful to safety are excluded or forced to very low probability or consequence by the repository design and siting concept.

Figure 11-5 shows a schematic cross-section through the crystalline basement of northern Switzerland. The system of safety barriers is the same as that shown in Figure 11-2. The repository concept has essentially remained the same, but supported by additional data. Temperature evolution within the EBS was re-evaluated and, using conservative values for key properties of the bentonite, the maximum temperature of the bentonite in contact with HLW canisters was 150 ºC, but decreasing to 100 ºC for the outer half of the bentonite buffer.

Key elements of the Reference Scenario are:

- Resaturation of the bentonite soon after repository closure;
- Corrosion and simultaneous failure of canisters at 1000 years;
- Corrosion of the HLW glass matrix and congruent release of radionuclides, with concentrations of radionuclides being controlled by solubility limits;
- Diffusion of radionuclides through bentonite buffer, with sorption;
- Advection of radionuclides from the bentonite-host rock boundary and through the principal water-conducting features, including matrix diffusion;
- Relatively rapid transport of radionuclides to the surface aquifers;
- Release of radionuclides in the biosphere and resultant exposures to humans.

The key processes are shown in Figure 11-6. A chain of assessment models covered release and transported of radionuclides in the near-field, migration through the geosphere and within the biosphere.

The resultant peak dose (annual individual dose from all pathways) for the Reference Case was $<10^{-3}$ mSv/year (2.10^{-4} mSv/year), cf. regulatory guideline of 0.1 mSv/year. Cs-135 was found to be the key dose-contributing radionuclide (see Figure 11-7). As shown in Figure 11-7, assessment calculations were carried out until 10^{7} years, and in this context, Nagra acknowledges the increasing uncertainty in terms of future evolution of the repository system.

Calculations were performed in Kristallin-I for Reference Case, Alternative Scenarios (deep groundwater well, tunnel / shaft seal failure, alternative climate-related scenarios), and also a robust scenario involving immediate transport from near field to biosphere).

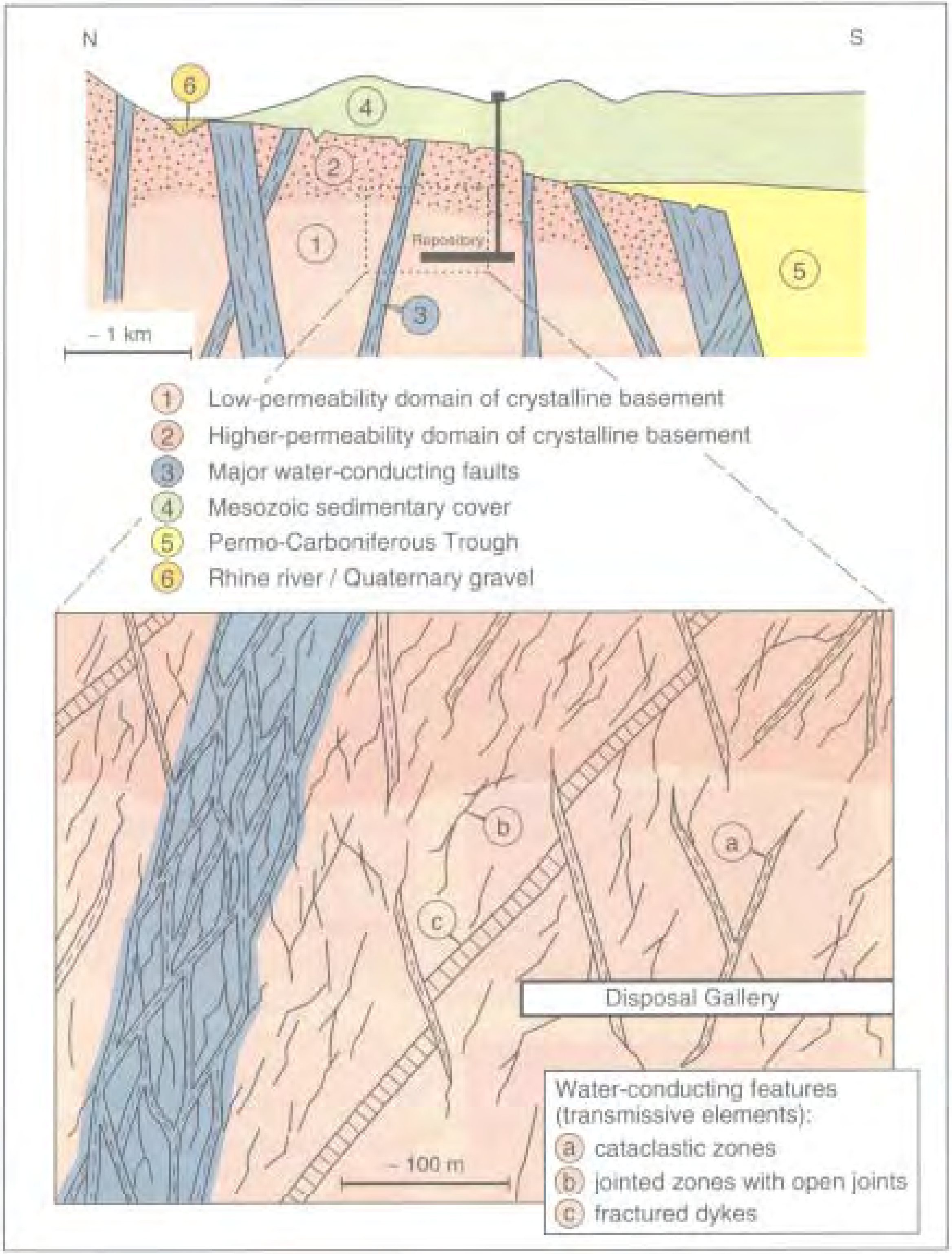


Figure 11-5
Schematic cross-section through the crystalline basement of Northern Switzerland (Area West) showing the main structural components affecting repository siting and construction (Nagra, 1994a). Used with permission of Nagra.

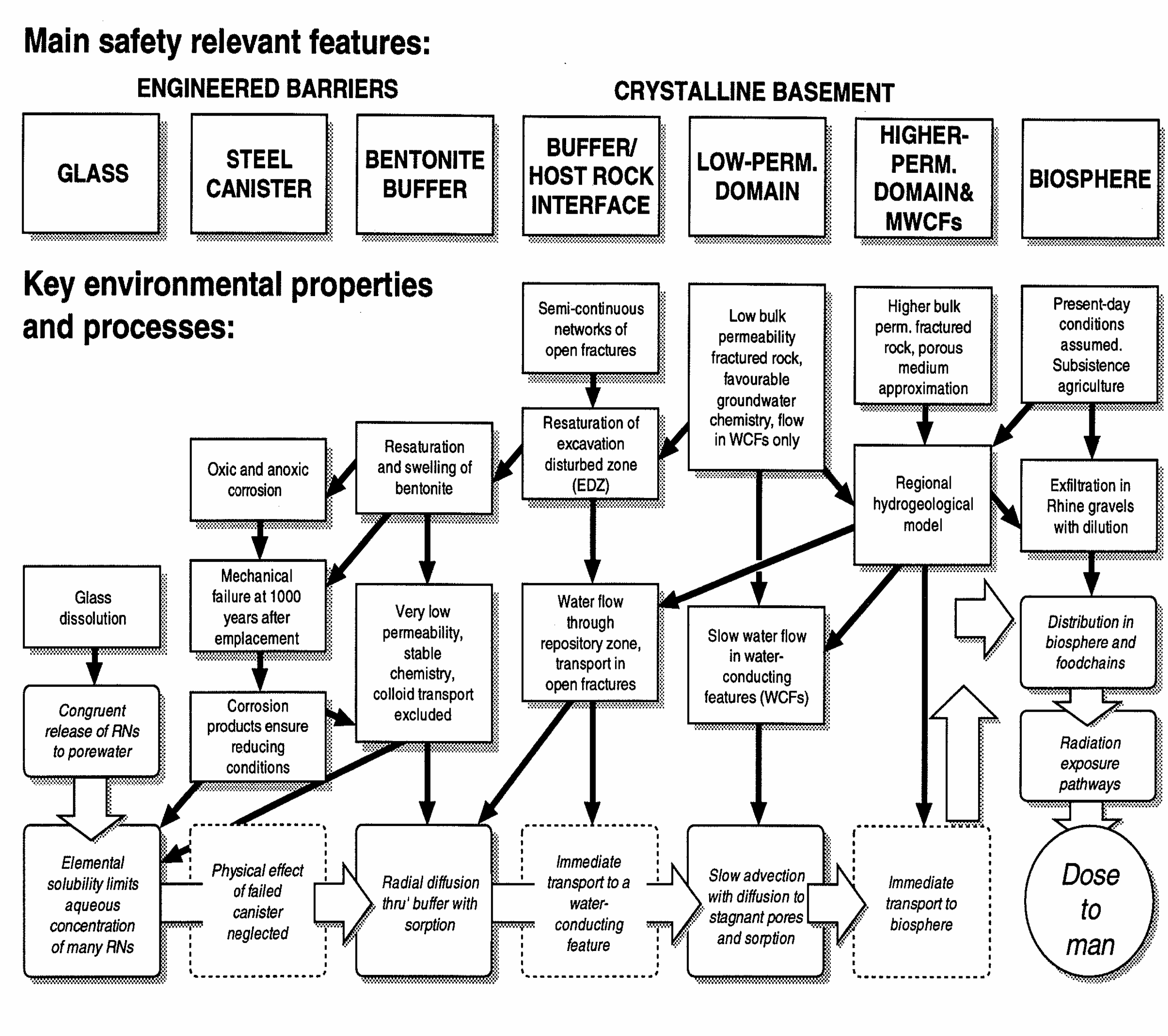


Figure 11-6
Main safety-relevant features incorporated in Reference Case for Kristallin-I (Nagra, 1994a). Used with permission of Nagra.

For the robust scenario, i.e., bypassing the geosphere, the resultant peak dose was ~10^{-2} mSv/year, dominated by Cs-135 at early times and nuclides in the 4n+2 decay chain at late times (> 10,000 years). A suite of calculations was performed to illustrate the effects of uncertainty in the properties of the EBS as well as the geosphere; not simply changes in parameter values but alternative scenarios and different modeling assumptions. In particular, uncertainty in the geological information necessitated a highly conservative representation of radionuclide transport in the geosphere, demonstrating via a robust scenario, that the EBS properties are sufficient to provide an adequate level of safety. Thus, the main function of the geosphere is to provide a degree of protection for the repository as well as favorable geochemical characteristics and low groundwater flow rates.

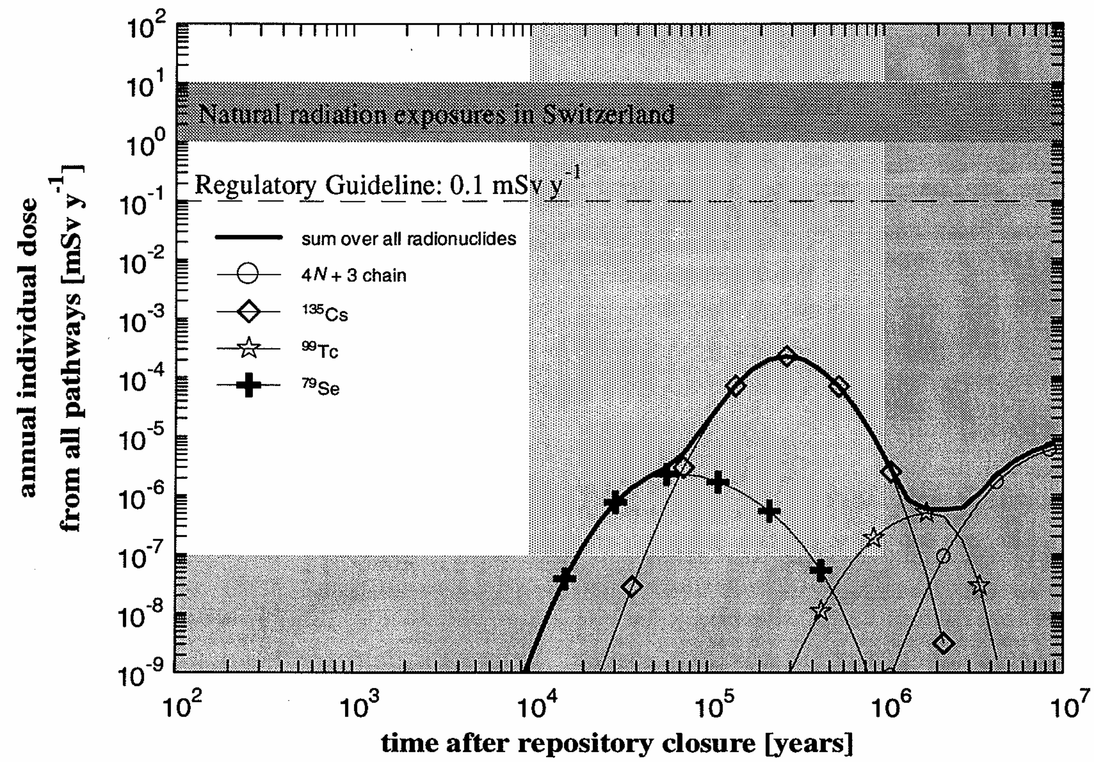


Figure 11-7
Results (annual individual dose) for Reference Scenario for Kristallin-1 (Nagra, 1994a). Used with permission of Nagra.

11.6.1.3 Project Opalinus Clay

In December 2002, Nagra submitted to the Swiss government, the documentation of Project *Entsorgungsnachweis* (demonstration of disposal feasibility) for the disposal of used fuel, HLW and LL- ILW in the Opalinus Clay of the Zürcher Weinland in northern Switzerland. Feasibility was discussed in terms of a suitable geological environment, feasible construction and operation of the repository, and long-term safety. The project was also designed to provide a platform for discussion as well as a foundation for decision-making on how to proceed with the Swiss HLW program.

Figure 11-3 shows the envisaged layout of a repository in Opalinus Clay, and Figure 11-8 shows a corresponding plan view (Nagra, 1994a).

The Reference Scenario involved release of radionuclides via the groundwater pathway to the surface but within this scenario there were 22 calculation cases involving alternative conceptualizations and parameter variations. Corrosion by water is the primary process leading to failure of used fuel or HLW canisters. Nagra also identified alternative scenarios, some representing alternative pathways (release of volatile gases and human activities) while others related to design options and biosphere uncertainty.

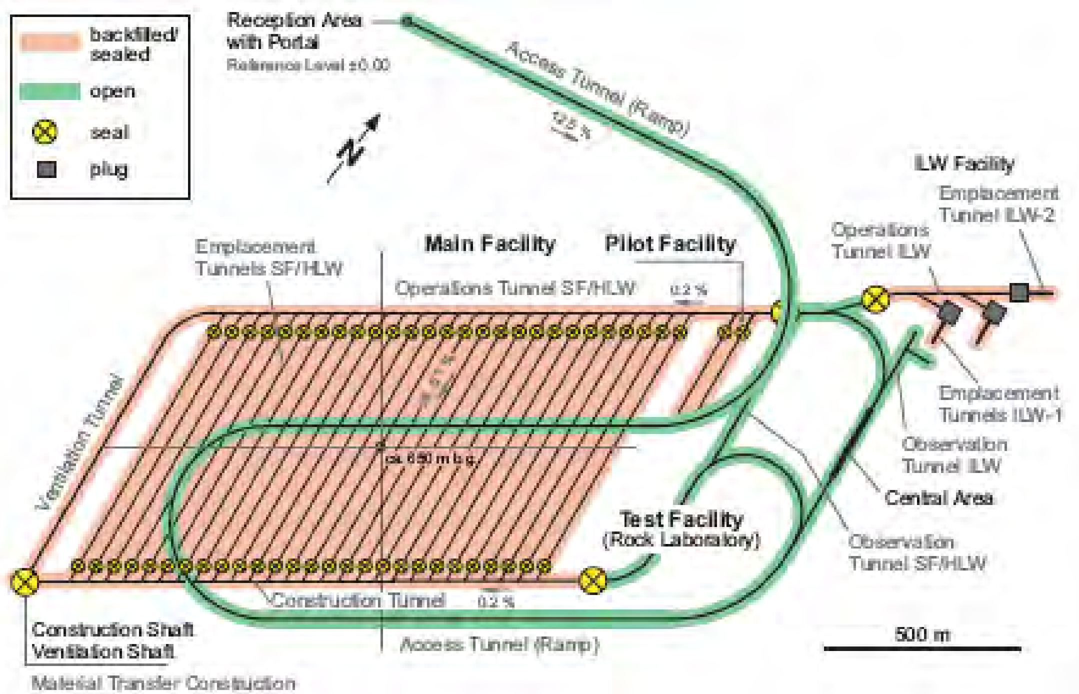


Figure 11-8
Plan view of repository for used fuel, HLW and LL-ILW in Opalinus Clay (Nagra, 1994a). Used with permission of Nagra.

NOTE: The status is shown after waste emplacement but before final sealing and closure of repository.

For the Reference Case, Nagra assumed that there are no initially defective canisters and that all canisters are breached after 10 000 years. Nagra considered the assumption of simultaneous breaching rather than a more likely distribution over time to be conservative. Carbon steel was selected for the used fuel and HLW canisters, the steel being 150 mm thick for the used fuel canisters and 250 mm thick for the HLW canisters. The design lifetime set by Nagra for both types of canister is 1,000 years but the expected lifetime is 10,000 years (Johnson and King, 2003). No credit was taken for the stainless steel flask containing the HLW glass or the Zircaloy cladding for used fuel.

Deterministic and probabilistic calculations were performed and Nagra carried out modeling simulations over a time span of up to 100 million years, but acknowledging that, for times beyond 1 million years, it becomes increasingly difficult to exclude significant changes in the geological environment.

Figure 11-9 shows a summary of the deterministic results for the Reference Case in terms of annual dose (mSv/year) and separated into annual doses attributable to used fuel, HLW and LL-ILW. The annual dose attributable to used fuel dominates the overall dose, with maximum annual dose for the Reference Case being 4.8 x 10^{-5} mSv/year and dominated by I-129.

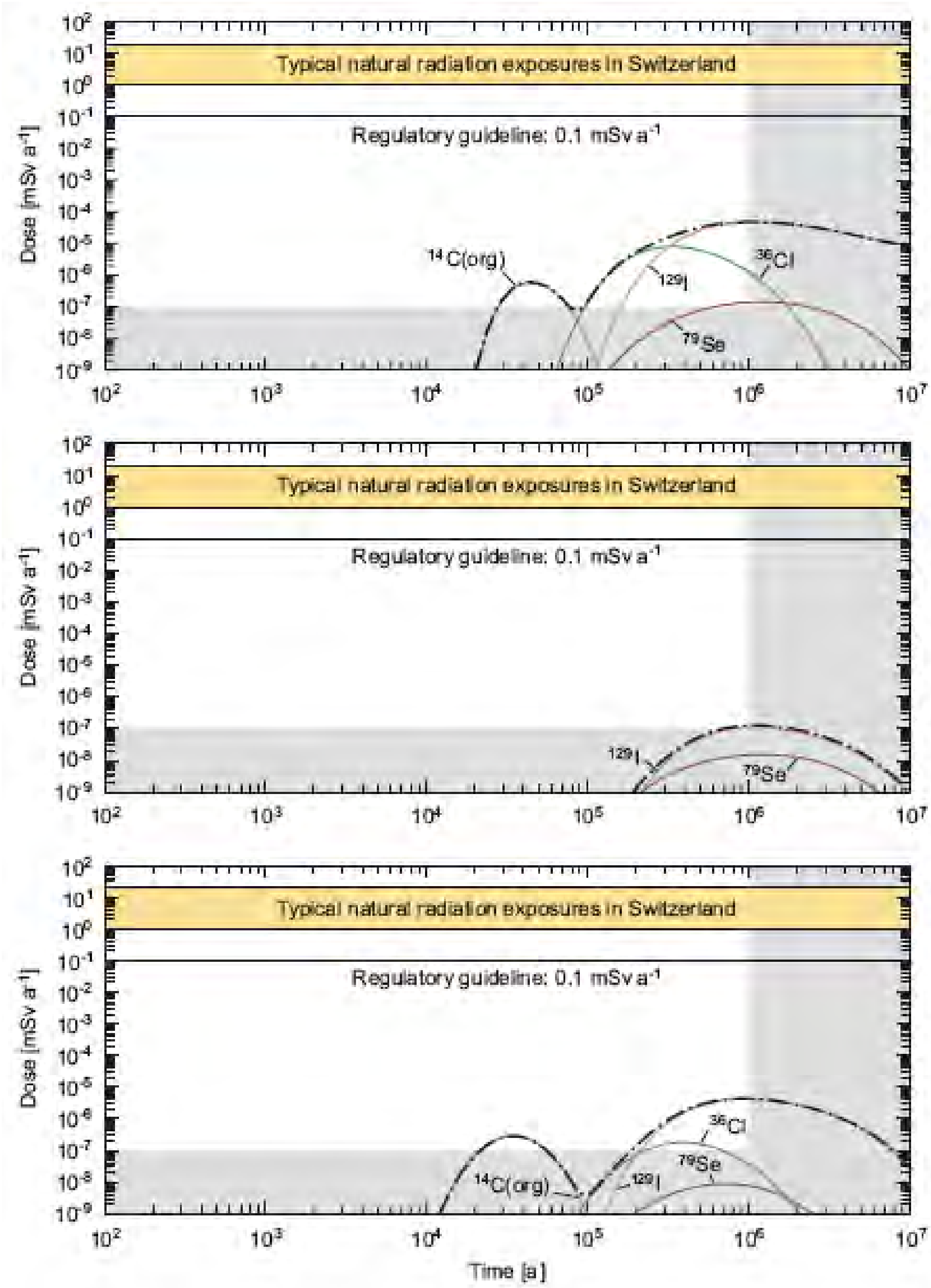


Figure 11-9
Predicted annual doses (mSv per year) for the Reference Case as a function of time: upper diagram = used fuel; middle diagram = HLW; lower diagram = ILW (Nagra, 2002). Used with permission of Nagra.

The other key (dose-contributing) nuclides are Cl-36, C-14, and Se-79. Due to the retentive properties of the Opalinus Clay, the predicted peak dose occurs about one million years in the future when uncertainties, including those concerning the biosphere, are significantly greater than the present day.

11.6.1.4 Safety Assessment Summary

Nagra's work in the 1980's concentrated on a demonstration of the feasibility of a repository in Switzerland for HLW and LL-ILW, culminating in the submission of Project "Gewähr" (Nagra, 1985) and, subsequently, Kristallin-I (Nagra, 1994a). These assessments were the result of a legal requirement to demonstrate the feasibility of safe disposal of all categories of radioactive waste, a precondition to the continued operation of NPPs in the country. While the engineering feasibility and the safety of such a repository were demonstrated, the authorities were not satisfied that blocks of crystalline rock of sufficient size and suitable properties were available. Thus, the Federal Council required that Nagra's research be continued and extended to sedimentary rocks.

As a result of a broad selection process, Nagra chose the Opalinus clay formation in the north of the canton of Zurich for further geological investigations. The results of these investigations formed the basis of another feasibility demonstration, which was submitted for review to the federal authorities in December 2002. The technical review by the competent Swiss authorities was concluded in August 2005 after a broad public consultation. The Federal Council approved the feasibility demonstration in 2006.

Project Opalinus Clay was reviewed by an International Review Team (IRT) that considered Nagra's Safety Case to be "*at the forefront of international practice*" (NEA, 2004). The IRT noted that Nagra's bentonite buffer concept was different from most other countries in terms of:

- The use of bentonite pellets; and
- The maximum design temperature in the inner portion of the bentonite greater than 100 °C.

Given that gas generation and transport through the clay formation is a key issue, the IRT concluded that Nagra's treatment in the assessment, incorporating direct two-phase flow through the clay, had been addressed satisfactorily and conservatively.

11.6.2 Licensing Process

The Nuclear Energy Act establishes the need for a series of licenses regarding nuclear materials, radioactive waste and nuclear facilities:

- *General license*: Mainly a political decision prior to the realization of a nuclear facility. The general license is granted by the federal government and must be approved by Parliament. The approval is subject to a referendum.
- *Construction license:* The licensing authority is UVEK.
- *Operation license*

- *Closure Order*: Applicable only to disposal facilities. The Order is issued by the Federal Council upon expiry of the monitoring period after termination of emplacement of waste packages. After closure, the Federal Council may order further surface monitoring for a limited period of time, after which it will declare that the disposal facility is no longer subject to the nuclear energy legislation.

The licensing process is conducted by BFE and generally consists of the following main steps:

- Submission of the application - description of project and safety analysis report;
- Review of the safety aspects of the project by ENSI. Comments are also provided by KNS and possibly by other organizations or experts;
- Consultation with federal offices and cantonal governments;
- Deposition of the license application documentation for public consultation. Individuals, communities and organizations can raise objections against the project;
- Compilation by BFE of all comments collected, and proposal to UVEK for a decision;
- Decision by UVEK, typically together with a list of license obligations. Appeals against this decision may be filed with a Board of Appeals.

As shown in the first bullet, a safety assessment report is part of the documentation required by the Swiss nuclear energy legislation for each licensing step of a nuclear facility, including irradiated nuclear fuel management facilities. An EIA is required at the general license and construction license stages based on the Environmental Protection Act. HSK carries out comprehensive reviews of the safety assessments, and BAFU reviews the EIA.

Safety assessments have to be updated by the applicant and reviewed by HSK at each step of the licensing procedure (general, construction, and operation licenses).

11.7 Current Status

For both L/ILW and HLW repositories, the Nuclear Energy Ordinance (KEV) states that the site selection procedure has to be defined in a so-called “sectoral plan” within the framework of the existing legislation on land-use planning, also taking into account socio-economic aspects. The conceptual part of the “sectoral plan for deep geological repositories” was prepared by the federal authorities under the lead of the BFE and, following a widespread consultation process, was approved by the Federal Council in April 2008. Implementation of the sectoral plan started in October 2008 when Nagra proposed six geologically suitable siting regions based on safety criteria. The government decision on the proposals is pending.

The site selection procedure is based on a stepwise approach involving three stages.

- *Stage 1*: Suitable siting regions are identified for each repository type, based on safety criteria defined by the regulatory authority.
- *Stage 2*: Potential repository sites are defined in the previously proposed siting regions and compared on the basis of provisional safety assessments. Socio-economical factors are also taken into account at this stage.
- *Stage 3*: Detailed investigation of at least two sites for each repository type. Such investigation includes a full safety assessment for each selected site. Based on the results of this process, a repository site will be selected for each repository type.

Broad public consultation occurs at the end of each step, in Switzerland and also in the concerned neighboring countries. The focus of the selection procedure is primarily on safety, but stakeholder involvement is an important component. With a favorable outcome, the above three-step process ends with approval of selected sites by the Federal Council. The overall process is expected to take between eight and ten years, followed by the general licensing procedure according to nuclear energy legislation. The general license will be granted by the Federal Council and must be approved by Parliament. The approval is subject to a referendum. Thus, the operation of the repository for HLW is expected to be no earlier than 2040.

11.8 Summary and Key Observations

- *Policy on Geologic Disposal*: The Energy Law specifies the requirement for the safe and permanent management and disposal of all radioactive waste. Nuclear utilities are allowed to choose between reprocessing and direct disposal of the irradiated nuclear fuel, with a geologic repository able to accommodate either type of waste.
- *Institutional Arrangements*: The operators of nuclear power plants and the federal government formed Nagra which is responsible for the disposal of all kinds of radioactive waste, including irradiated nuclear fuel if declared as waste, with a view to implementing its permanent and safe disposal. Nagra is supported by PSI as well as independent contractors. ENSI (formerly HSK) is the regulatory authority responsible for the safety of nuclear energy facilities and installations, including radioactive waste disposal. Current expenditures related to waste management while plants are operational are continuously paid for by the NPP operators as part of their annual budget. By law a separate dedicated waste fund was established to cover post-reactor-shutdown expenses associated with disposal via a levy of about 1 Swiss cent/kWh of nuclear power production.
- *Key Laws and Regulations*: The new Nuclear Energy Law, KEG, which came into force in 2005, provides an extensive revision of Switzerland's legal framework, together with the corresponding Ordinance. As well as requiring the safe and permanent management and disposal of all radioactive waste, KEG keeps the nuclear option open and addresses a number of key issues related to radioactive waste, including a concept of monitored long-term geological disposal of radioactive waste that combines elements of final disposal and reversibility. StG and StSV form the legal basis for radiation protection in Switzerland, the latter specifying the annual dose limit for individuals of the population to 1 mSv per year. ENSI's Regulatory Guideline ENSI-G03 specifies the requirements for long-term safety of disposal facilities.

- *Site Screening and Selection*: Three host-rock variants have been considered —either the crystalline basement or one of the two overlying, low-permeability sediment layers. Geological studies showed that the extent of accessible crystalline basement was much less than originally thought. Thus, only two relatively restricted areas remain for the selection of a possible site, each covering an area of about 50 km^2. Despite this limitation, Nagra believes that it would be feasible to find a suitable repository for Switzerland's required low volume of waste. In parallel with the crystalline basement studies, investigations of the sedimentary options proceeded from desk studies to select potential host formations to identification of potential siting areas. With regard to sedimentary formations, Nagra identified Opalinus Clay as the top priority and conducted a field program in the potential siting area. Geological investigations have benefitted from two URL programs, at Grimsel in crystalline rock, and at Mont Terri for clay studies. Despite the small size of Switzerland and its geological setting, Nagra is confident of identifying a suitable site for a geologic repository and has been able to successfully submit a Safety Case in support of a repository in clay.

- *Repository Design Concepts*: The current Swiss multi-barrier disposal concept involves geologic disposal at a depth of about 500 m to 1 km below surface, with in-tunnel emplacement of HLW packages in crystalline or sedimentary rock that physically protects the EBS, has low water flow, favorable groundwater chemistry (reducing conditions) and, in the case of Opalinus Clay, provides an efficient radionuclide transport (diffusion) barrier. In addition to the vitrified waste in its steel fabrication canister or irradiated UO_2/MOX fuel within its cladding, a thick steel overpack is envisaged, surrounded by compacted bentonite clay. Co-disposal of long-lived ILW is planned, but in tunnels in a separate part of the repository.

- *Performance Metrics and Assessments*: The key performance metric is that the annual dose to individuals must be less than 0.1 mSv/year. There is no time cutoff in Swiss regulations. Nagra completed the Project Gewähr assessment in 1985 covering disposal of HLW and LILW in crystalline rock, with a disposal concept similar to the current version, and carried out according to HSK's Guideline, which specified that deterministic safety analyses should be carried out and that error limits and uncertainties were to be estimated. Nagra's Base Case assumed that groundwater enters the waste-filled storage caverns and eventually reaches the surface of the waste canisters. The canisters corrode and are assumed to fail simultaneously after 1,000 years. After failure, the waste matrix begins to corrode and radionuclides are released and diffuse through the bentonite buffer to the host rock. The resultant maximum peak dose from the 'realistic' case was $6 \bullet 10^{-10}$ mSv/year. Nagra noted that a peak dose close to the regulatory limit of 0.1 mSv/year could be reached only via a combination of many conservative assumptions, which did not correspond to "*realistically conceivable*" that was specified in the HSK Guideline. In the subsequent assessment, Kristallin I, similar criteria were specified, but with an additional risk criterion (1 in a million per year) to be applied to unlikely events. Again, no timeframe cutoff was stipulated. The resultant peak annual individual dose from all pathways for the Reference Case was $< 2.10^{-4}$ mSv/year. Nagra also submitted (2002) a safety assessment (Project Opalinus Clay) for the disposal of used fuel, HLW and LL-ILW in Opalinus Clay. For the Reference Case, Nagra assumed that there are no initially defective canisters and that all canisters are breached after 10 000 years. In this case, the annual dose attributable to used fuel dominated the overall dose, with maximum annual dose for the Reference Case being 4.8×10^{-5} mSv/year.

- *Independent Peer-Review and Advisory Bodies*: Project Opalinus Clay was reviewed by an International Review Team that considered Nagra's Safety Case to be "*at the forefront of international practice*". With regard to advisory bodies, a number of bodies provide advice to the government on nuclear-related safety matters, including KNS, AGNEB, and KNE.
- *Stakeholder and Public Involvement*: The licensing procedure for the construction of disposal facilities includes a public consultation. Documentation on the project, including the safety analysis report, the regulatory review report and the views and opinions of the cantons, is made available, and any person (including those from neighboring countries) can give input or raise objections. KEG specifically requires that the site canton as well as neighboring cantons and countries, are involved in decision-making with regard to the general license.
- *Program Maturity*: The Swiss geologic disposal program is well-developed and has been successful technically in terms of the development of a disposal concept for both crystalline rock and sedimentary (clay) rock, and submission of safety assessments indicating the feasibility of geologic disposal. Progress has been delayed to some extent by the social and political climate although the current site selection process (sectoral plan) is now proceeding in a stepwise approach involving three stages with broad public consultation at the end of each stage.
- *Licensing Process*: According to KEG, a series of licenses is required for a geological repository: General license (subject to a referendum), Construction License, Operation License. And a Closure Order. Submission of an application involves a project description and a safety assessment report, which is part of the documentation required for each licensing step. An EIA, which involves a public hearing, is part of the process for the General and Construction Licenses.

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12
TAIWAN

12.1 Background

12.1.1 General Nuclear Profile

Taiwan began generating electricity from nuclear power in 1978, and currently has six operating reactors (4 BWRs and 2 PWRs) at three separate sites, all operated by Taiwan Power Company (TPC), or Taipower. This nuclear fleet generated 40 TWh of electricity in 2009 representing approximately 20% of the total national electricity supply (WNA, 2010a,b). Two additional reactors, Advanced Boiling Water Reactors (ABWRs), are being constructed at a fourth location. All operating reactors, as well as the two ABWRs under construction, in Taiwan are imported from the United States. Most nuclear fuel is also of U.S. origin. Taiwan's policy for irradiated nuclear fuel is direct disposal, although reprocessing is under consideration, subject to terms of agreements with supplier nations.

12.1.2 HLW / Used Nuclear Fuel Inventory

The operation of eight reactors, each for 40 years, is expected to result in a total amount of 7,340 MTHM of irradiated nuclear fuel (AEC, 2006). While irradiated fuel is exclusively stored in pools at the nuclear power plant sites, there is a need for interim dry storage facilities for irradiated fuel until a repository is available.

12.2 Institutional Arrangements

12.2.1 Institutional Framework

An overview of organizations involved in radioactive waste management in Taiwan is shown in Figure 12-1 (Cheng and Wu, 2001).

POLICY and OVERSIGHT– The Atomic Energy Council (AEC) is the lead executive branch agency within the Taiwanese government with responsibility for radioactive waste management The AEC was created in 1968 by the Taiwanese government. Also created at this time was the Institute of Nuclear Energy Research (INER), responsible for nuclear-related studies.

IMPLEMENTER – Taipower (TPC), the owner/operator for all the nuclear power plants, is a state-owned utility under the Ministry of Economic Affairs. In common with most other countries in this review, Taiwan's policy is that the nuclear utility also has primary responsibility for the management of used nuclear fuel.

REGULATOR - Taiwan's nuclear regulator is the Fuel Cycle and Material Administration (FCMA), which is administratively located within the executive branch government agency policy body for nuclear energy and waste management, the AEC.

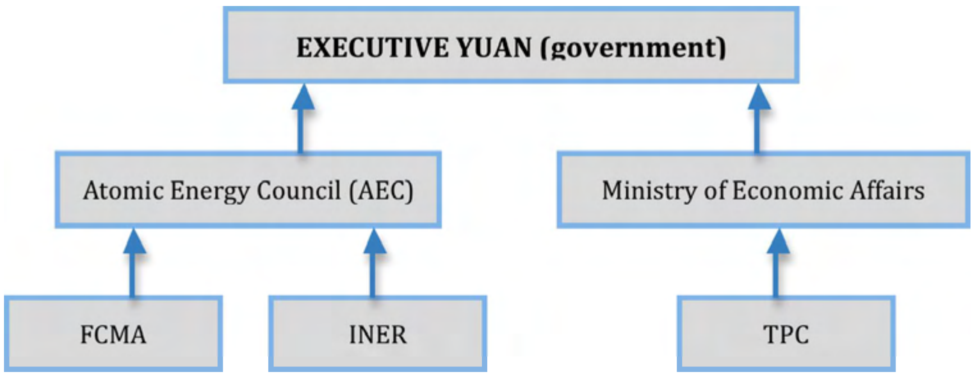

Figure 12-1
Top-level radioactive waste management organizations in Taiwan (Cheng and Wu, 2001). Used with permission of Lawrence Berkeley National Laboratory.

12.2.2 Legal and Regulatory Framework

The management of commercial used nuclear fuel in Taiwan is subject to bilateral agreements with the United States under the "The Republic of China and United States Civilian Atomic Energy Cooperation Agreement". According to this agreement, reprocessing of Taiwanese irradiated nuclear fuel either domestically or by shipping to a third country must be approved by the U.S. government. Taiwan currently does not have reprocessing facilities and does not plan to carry out reprocessing in the near future. However, the government has not ruled out the possibility of reprocessing irradiated nuclear fuel outside Taiwan.

Currently, the government strategy for the final disposal of used nuclear fuel includes interim storage lasting for decades while exploring the possibility of an international repository. The irradiated nuclear fuel will first be cooled in the cooling pools near the power plant and then will be stored in dry casks in surface storage facilities. Until a feasible proposal for a final disposal repository has been accepted, however, Taipower must comply with the Nuclear Materials and Radioactive Materials Management Act." The relevant Articles in this Act include:

- *Article 29*: The waste producer is responsible for the treatment, transportation, storage and final disposal of the waste either by the producer or by employing domestic and international qualified vendors. The producer will be fined if failing to comply with this Article.

- *Article 46*: The nuclear power company must apply 2% or more of the nuclear post-processing (backend operation) fund towards final disposal research and development activities. (Note that the post-processing fund has already been established.)
- *Article 49*: The producers of irradiated nuclear fuel should have a program to understand the feasibility for constructing domestic facilities for the final disposal of this waste.

Another law, the Enforcement Rules for Nuclear Materials and Radioactive Waste Management Act, requires that the producer of irradiated nuclear fuel must propose a final disposal plan for this type of waste. The producer must revise the plan every four years and every year, the producer must report the progress towards execution of the long-term plan. To comply with this law, Taipower must propose the final disposal program and carry out site selection, construction, operation, closure, and monitoring activities.

Furthermore, the government issued a report entitled *The Spent Nuclear Fuel Final Disposal Plan* (AEC, 2006), which acknowledges direct geologic disposal as government policy for irradiated nuclear fuel. While continuing to seek international cooperation on final disposal and irradiated nuclear fuel reprocessing, this program is a guideline of extended research and development activities to understand the feasibility for constructing domestic facilities for the final disposal of spent nuclear fuel within Taiwan.

12.3 Site Screening, Selection and Characterization

12.3.1 Early Geological Study

TPC initiated in 1986 the first stage of a program to identify suitable locations for the final disposal of used nuclear fuel. Geological survey information obtained during this early study indicated that potential host rocks including granite, thick shale and mudstone layers, exist at appropriate depths in Taiwan (Cheng and Wu, 2001). A long-term plan for detailed geological investigation studies was formulated in 1991 and has since undergone several revisions/ modifications leading to the most recent disposal plan discussed below.

12.3.2 Repository Program

"The Spent Nuclear Fuel Final Disposal Plan", submitted by TPC and approved by AEC (2006), proposed a division of the used nuclear fuel disposal program into five phases as follows. The outcome at the end of each phase is also described below:

1. Regional investigation (2005 – 2017): completion of the report on the final disposal technical feasibility evaluation and recommendation on site investigation areas.
2. Preliminary site investigation and selection (2017 – 2028): completion of the performance and safety assessment for the selected candidate sites and prioritization of the recommended sites for further study.
3. Detailed site investigation and testing (2028 – 2038): completion of the feasibility research report and environmental impact statements for the selected site.

4. Repository design and safety assessment (2038 – 2044): completion of the site safety analysis and obtain construction license.
5. Repository construction (2044 – 2055): completion of repository construction and operational license obtained.

Currently, the repository program is at the beginning of the first phase of the Final Disposal Plan described above. The objective is to complete the preliminary technical feasibility assessment and recommend sites for further investigation. The specific scope of work for Phase 1 includes (Taipower, 2006):

- Preliminary geological investigation and drilling exploratory boreholes in selected areas;
- Establish a methodology for repository performance and safety assessment;
- Complete a preliminary technical feasibility assessment report by 2009 (the report has not yet been released); and
- Complete technical feasibility assessment report by 2017.

The program investigated host rocks throughout Taiwan and other large islands under Taiwan's jurisdiction.

Taiwan's geology is of Mesozoic and Cenozoic origin. Candidate geologies for the hosting of a repository are listed below in Table 12-1.

Table 12-1
Summary of main geologies in Taiwan and large islands under Taiwan's jurisdiction (adapted from Taipower, 2009).

Region / Location	Principal Geology	Age (Ma)	Geological Stability and Siting Considerations
Eastern Taiwan	Granite	80 – 90	Geological stability to be confirmed by further study
Penghu, Kinmen, and Mazu islands	Granite	100 – 160	Geologically stable since the Yanshanian orogeny; very low tectonic and seismic activity; currently the favored region under consideration for hosting a repository (Kinmen Island)
Southwest and central mountain region of Taiwan	Mudstone	0.4 – 5.5	Thick rock layer (4 – 7 km); however dense population, extensive groundwater use leading to subsidence, high seismic activity including frequent, shallow earthquakes, and presence of methane, CO_2, and natural gas within sedimentary rock formations are considered unfavorable for siting of a repository

12.3.3 Site Investigations

The area currently under study is on Kinmen Island located west of the Taiwan Strait and close to mainland China. Deep (500 m) boreholes have been drilled for geophysical, geochemical, and hydrogeological investigations. Various geophysical methods have been used to study fractures, such as gamma-ray logging, spontaneous potential, short-normal (16N) and long-normal (64N) resistivity, lateral resistivity, caliper logging, full waveform sonic log, gamma spectroscopy, acoustic televiewer, dipmeter log, and a Borehole Televiewer. The resultant measurements are intended to understand fracture distribution within the granite. The results demonstrate that the granite near borehole No.6 is almost intact and possesses high resistivity and high acoustic speed characteristics. Most fractures are filled with minerals and exist in deeper parts of the borehole.

Geochemical data from the borehole were obtained in such a way as to avoid disturbance from sample extraction. Water samples were taken from water-bearing fractured zones. Hydrogeological data were obtained using the borehole micro-flow measurement and hydraulic packoff test. Data from this test can help identify layers with relatively high hydraulic conductivities.

Part of the site investigation program involves developing appropriate technologies for geological conceptual model development. Scientists utilized three deep boreholes in the Kinmen Island test area to conduct a small-scale field test to synthesize detailed information on the geology, tectonics, geophysics, geochemistry, hydrogeology, and rock mechanics, in order to better understand the fracture distribution and fracture characteristics.

12.4 Disposal Concept

Taiwan has adopted the general KBS-3 disposal concept of the Swedish program, although the specific disposal concept has yet to be decided, i.e. the program is evaluating both vertical and horizontal emplacement options (e.g., Figures 12-2 and 12-3 below). Bentonite is planned for use as buffer material. Laboratory studies have been carried out on MX-80 bentonite, whereby bentonite material was mixed with synthesized seawater, and new experiments are ongoing. Radionuclide sorption coefficients are the focus of these studies, with most experiments being conducted under anaerobic conditions. Aerobic conditions were also used and the results obtained were compared with those under anaerobic conditions (Taipower, 2009).

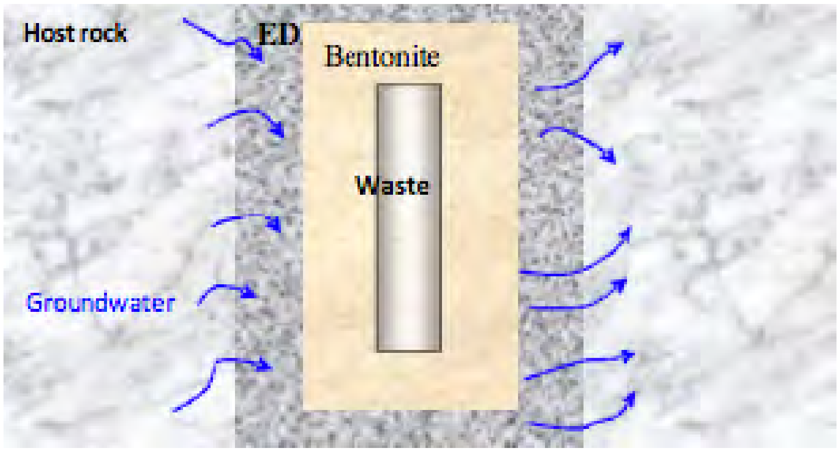

Figure 12-2
Near-field conceptual model of the repository for the *vertical disposal concept* for irradiated nuclear fuel (Zhou 2000). Used with permission of author.

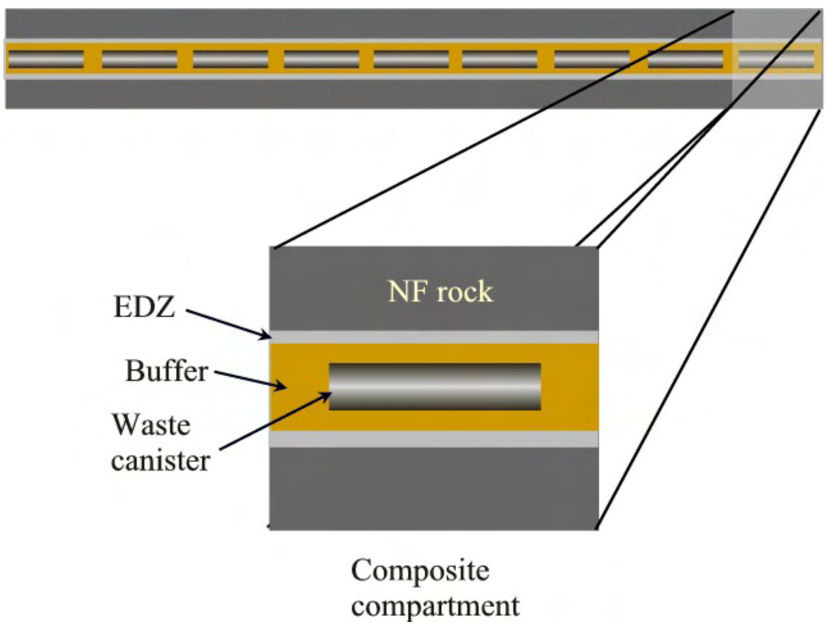

Figure 12-3
Near-field conceptual model for the *horizontal disposal concept* for irradiated nuclear fuel (Zhou and Apted 2005). Used with permission of authors.

12.5 Transparency and Stakeholder Involvement

12.5.1 Public Involvement

The Taiwanese public view concerning nuclear waste geological disposal is complicated by both social and political factors. In general, the general public has negative feelings towards nuclear power and nuclear waste disposal, which have been used in political campaigns in general elections. Such sentiments reached a peak when the former president Shui-bian Chen was elected. As the first democratically elected president from the long-time opposition party, he stopped construction of the commercial nuclear power plants at the new (fourth) location. Later, he allowed construction to continue based on more rational and practical considerations.

The Taiwanese population consists of a combination of immigrants from mainland China and aboriginal ethnic groups and there is a long history of conflict between these two groups over land use, which has made site selection especially difficult. For example, areas of low population density are all occupied by aboriginal groups, making the issue of site selection highly political.

Under such social and political pressures, the government is very cautious in dealing with the public. The government keeps all communication channels open and regards public involvement and acceptance as the most important factors for the success of the final disposal program. Documents and regulations are available online including program plans and progress reports. Considering uncertainty in public opinions, as well as the emergence of new technologies and strategies in international collaboration, the government set the program goal as a "moving target", i.e. adopting a flexible approach. Such an approach conforms to Article 37 of the Enforcement Rules for the Nuclear Materials and Radioactive Waste Management Act: "*The high level waste final disposal plan shall be reviewed and amended every four years. Upon amendment, reasons and correction measures shall be submitted to the competent authorities for approval prior to execution*."

In terms of openness, the Taiwanese program has strived to keep the public well informed as well as seek potential opportunities to dispose of waste outside Taiwan.

12.5.2 International Involvement

Taiwan has long appreciated that international collaboration enhances the technical accreditation of the country's waste management program and helps avoid unnecessary repetition of typical problems that have occurred in other countries during early phases of their programs. In particular, Taiwan has reached out to overseas consulting groups to help develop their initial PA modeling and simulation capabilities as well as train technical staff. Technical communications have also been maintained with other national programs including US DOE, Swedish, Japan, Switzerland, and Finland. To this end, bilateral agreements exist with a number of countries, in particular the US (via the national laboratories) and Switzerland (via Nagra).

Workshops and conferences have been held in Taiwan that sometimes include tours to the Orchid (Lan-Yu) Island facility[31]. Taiwanese scholars also attend international conferences to present their waste management program difficulties and appeal for collaboration. Every year, the program budget covers collaboration activities and includes a review of the latest developments in repository and reprocessing/recycling of irradiated nuclear fuel.

[31] LLW storage facility.

12.6 Safety Assessment and Licensing

12.6.1 Performance Assessment

The other important part of the long-term disposal program comprises the performance and safety analyses that cover safety requirements, disposal concept, repository facility planning and assessment, waste container material preliminary assessment, scenario development, system assessment, and integration technology.

Performance assessment (PA) capabilities have been developed within the Taiwanese program at the same pace as site investigations. In 2000, INER was selected to lead the PA work. Since then, INER has established the INPAG model/code system to carry out near-field and far-field radionuclide transport simulations (Zhou 2000, 2002, 2004; Zhou and Apted, 2005), later integrating INPAG with GoldSim in order to carry out probabilistic PA calculations. The program frequently reviews progress and integrates new information/data, such as geological and hydrogeological R&D data, into PA calculations. Results to date indicate repository performance exceeding regulatory requirements by sizeable margins, with projected annual effective doses that fall below the regulatory limit of 0.25 mSv/year[32] by more than 3 orders of magnitude (Taipower, 2009).

12.6.2 Licensing Process

No specific regulations or guidelines govern the licensing of disposal facilities for used nuclear fuel, but, based on existing regulations for a LLW disposal facility, it is likely that the site selection process will involve an Environmental Impact Assessment (EIA) that will involve the local population and local government prior to approval by the Executive Yuan.

Prior to construction of a disposal facility, a Safety Analysis Report (SAR) must be submitted by TPC to AEC. FCMA is the appropriate authority for approval of both the EIA and the SAR. Following construction of a disposal facility, an updated SAR must be submitted and approved by FCMA prior to obtaining an operational license.

12.7 Current Status

As discussed in Section 12.3.2, Taiwan is at the first stage of a five-phase program towards the development of a geologic repository for used nuclear fuel. This phase is not expected to be completed until 2017. The preliminary technical feasibility report has still to be published, but the final version of this report is not due until 2017.

The output from the technical feasibility report will identify several sites for more detailed geological and hydrogeological investigations. Thus, a repository is not expected to be constructed until at least 2052.

[32] As specified in the Enforcement Rules for the Nuclear Materials and Radioactive Waste Management Act (Articles 19 and 34).

12.8 Summary and Key Observations

- *Policy on Geologic Disposal*: The government issued The Spent Nuclear Fuel Final Disposal Plan in 2006, which acknowledges direct geologic disposal as government policy for irradiated nuclear fuel, although reprocessing is under consideration. Current plans are for the geologic repository to be operational by ~2032. The government's strategy includes interim storage lasting for decades while exploring the possibility of an international repository. A domestic repository must also be considered in case an international repository is not possible.
- *Institutional Arrangements*: Taipower, the owner/operator for all the nuclear power plants, is a state-owned utility under the Ministry of Economic Affairs, has primary responsibility for the management of used nuclear fuel including disposal. Taiwan's nuclear regulator is FCMA, which reports to the executive government branch, AEC. INER is the main organization responsible for nuclear-related studies on behalf of AEC. With regard to funding, the nuclear power company is expected to apply 2% or more of the nuclear post-processing (backend operation) fund towards final disposal research and development activities.
- *Key Laws and Regulations*: The Nuclear Materials and Radioactive Materials Management Act states that the waste producer is responsible for the treatment, transportation, storage and final disposal of the waste. This Act also states that producers of irradiated nuclear fuel should have a program to understand the feasibility for constructing domestic facilities for the final disposal of this waste
- *Site Screening and Selection*: Geological survey information obtained during an early study (late 1980's, early 1990's) indicated that potential host rocks including granite, thick shale and mudstone layers, exist at appropriate depths in Taiwan. The Spent Nuclear Fuel Final Disposal Plan approved by AEC in 2006 identified a five-stage process to select a suitable site, starting with regional investigations. Such geologic investigations are ongoing, including exploratory boreholes. This stage is expected to be completed in 2017 with the preparation and submission of a technical feasibility report. The area currently under study is in a granite formation on Kinmen Island, west of the Taiwan Strait and close to mainland China.
- *Repository Design Concepts*: Taiwan has adopted the general KBS-3 disposal concept of the Swedish program, although the specific disposal concept, whether vertical or horizontal emplacement, has yet to be decided. The buffer is bentonite.
- *Performance Metrics and Assessments*: The Enforcement Rules for the Nuclear Materials and Radioactive Waste Management Act specify an annual effective dose limit of 0.25 mSv/year. No assessment timeframe is specified although assessment calculations have been carried out to 10^7 years. PA capabilities have been developed within the Taiwanese program with INER responsible for leading this work. Biosphere results indicate a peak dose of ~10^{-8} Sv/year around

- *Stakeholder and Public Involvement*: In general, the general public has negative feelings towards nuclear power and nuclear waste disposal, which have been used in political campaigns in general elections. Thus, the government is cautious in dealing with the public, while regarding public involvement and acceptance as the most important factors for the success of the final disposal program. The Taiwanese program has strived to keep the public well informed by providing online access to regulations and documents including program plans and progress reports.
- *Program Maturity*: Taiwan is at an early stage in the development of a geologic repository for used nuclear fuel, with an operational facility not expected before 2052.

12.9 References

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13
UNITED KINGDOM

13.1 Introduction

13.1.1 General Nuclear Profile

The United Kingdom's (UK's) fleet of 19 operating reactors comprises an unusual portfolio of nuclear technology with 14 advanced gas-cooled reactors (AGR), 4 Magnox reactors, and one PWR. Nuclear generation contributes approximately 20% of the country's electrical energy supply, with 63 TWh of electricity generated in 2009 (WNA, 2010a,b). Originally all nuclear generation was publicly owned until 1996, when the 15 later generation reactors (AGRs + PWR) were privatized under British Energy. Following the 2009 acquisition of British Energy by Electricité de France (EdF), these reactors are now owned and operated by wholly-owned subsidiary EDF Energy. The remaining four operating Magnox-type reactors have remained under public ownership in anticipation of their imminent shutdown and decommissioning in the 2011-12 timeframe.[33] To date, 22 of the 26 Magnox power reactors have been shutdown. All of the current operating reactors except for the Sizewell B PWR are scheduled for shut down by 2023. In light of this pending loss of nuclear generating capacity, an aggressive replacement program totaling 19 GWe is being pursued, with the first new nuclear generation expected to be online by 2018 (WNA, 2010b).

The UK is one of a short list of countries having a commercial reprocessing capacity, although moving ahead, the future of reprocessing of used fuel is a commercial decision in the hands of the owners of the used fuel, subject to regulatory requirements. Under current UK Government policy, irradiated nuclear fuel is not categorized as waste while the option of reprocessing the fuel remains open and future use for the fuel is possible. However, a prevailing assumption for new nuclear reactors is that the associated irradiated LWR fuel would not be reprocessed and would be subject to direct disposal.

The HLW management picture in the UK is complex, reflecting the complex history associated with the development of nuclear power in the UK, the large scale deployment of Magnox reactors, and the large scale deployment of reprocessing infrastructures for both oxide fuels from LWRs and metal fuels from its domestic fleet of gas cooled reactors including Magnox fuel requiring reprocessing as opposed to long-term storage available for other more stable fuel types. Following public rejection of a proposed underground research laboratory (URL) for geologic

[33] Ownership is now under the Nuclear Decommissioning Authority (NDA) following transfer of civilian nuclear liabilities from British Nuclear Fuels Limited (BNFL) in 1998.

disposal of HLW at Sellafield, the Government remains committed to geologic disposal for HLW, although a firm direction has not been established and the decision for recycling or direct disposal is left to the commercial interests involved.

13.1.2 Used Fuel / HLW Inventory and Projected Disposal Requirements

Based on the 2007 UK Radioactive Waste Inventory (UK, 2008a), DEFRA (2008) states that there are 1,937 m^3 of HLW in the UK in storage of which 1,090 m^3 is in liquid form and 847 m^3 has been vitrified and packaged. The projected total volume of vitrified HLW over the lifetime of existing NPPs is 1,420 m^3 (DEFRA, 2008) The inventory of irradiated fuel from commercial reactors is summarized in Table 13-1. HLW resulting from the reprocessing of irradiated fuel from other countries will be returned to the country of origin[34]. The first shipments were expected to commence in financial year 2008/09 but eventually started in early 2010. The government's policy also allows for waste substitution, whereby the UK returns a greater amount of HLW to the customer, the difference being balanced by retaining a radiologically equivalent amount of L/ILW.

Table 13-1
UK Inventory of used nuclear fuel of UK origin as of March 31, 2008 (DEFRA, 2008)

Location	Fuel Type	Approximate Inventory (MTHM)
Magnox Power Stations	Magnox	180
Sellafield	Magnox AGR SGHWR*	1200 2800 120
British Energy	AGR and PWR	440
UKAEA	Various	13

NOTE: * SGHWR = Steam Generating Heavy Water Reactor, from UKAEA Winfrith.

13.2 Institutional Arrangements

13.2.1 Institutional Framework

The key responsibilities for radioactive waste management in the UK are:

POLICY and OVERSIGHT - In addition to its overview responsibility for the implementer, the *Government's Department of Energy and Climate Change (DECC), as of October 2008,* brings together policy responsibilities for both energy and climate change mitigation, which previously had been handled separately (with the civil nuclear industry formerly under the responsibility of the Department for Business, Enterprise and Regulatory Reform, or BERR).

[34] Reprocessing contracts entered into and signed since 1976 allowed for this option.

IMPLEMENTER - The *Nuclear Decommissioning Authority (NDA)* is a non-departmental public body, established in April 2005 under the Energy Act 2004 to provide a countrywide focus on the decommissioning and cleanup of nuclear sites. One of its first tasks was to prepare its decommissioning strategy, submitted in April 2006 for approval by the government. In 2007, the Government gave responsibility to the NDA for developing and maintaining a national strategy for the handling of LLW. The NDA is also responsible for the disposal and safe and secure interim storage of waste on civil nuclear sites. Within NDA, the Radioactive Waste Management Directorate (RWMD) has the responsibility for developing a geologic disposal facility (GDF) for higher-activity radioactive waste (HAW).

For more than twenty years, Nirex served as the UK nuclear industry's expert body on the long-term management of 'highly-active' radioactive waste (HAW), maintaining and developing the UK's knowledge base on long-term waste management options. In October 2006, in response to CoRWM's recommendations, the Government decided that responsibility for geological disposal of HAW should fall to NDA. The Government then transferred its shares in Nirex to NDA, with Nirex successfully integrating into NDA. NDA now performs the functions previously undertaken by Nirex.

NDA made a commitment in its 2006 Strategy to carry out an assessment of the life-cycle implications of irradiated nuclear fuel management, eventually publishing a report in October 2007. The study included an assessment of the risks (financial, socio-economic and environmental) and opportunities associated with three broad scenarios: disposal, storage or use. Thus, the three management strategy options for the next 300 years are:

- To treat the used fuel as waste, put it in a form suitable for geologic disposal and proceed with this as soon as possible.
- To store the used fuel for the long-term, on the assumption it may have a value at some point up to 300 years in the future.
- To reprocess the fuel now for recycle. This would see uranium stocks put back into enrichment and fuel fabrication, and plutonium used as an input to mixed oxide fuel (MOX).

REGULATOR - The role of the Environment Agencies in radioactive waste disposal is discussed in Section 13.2.2. In terms of regulatory and licensing authorities for nuclear facilities, the following organizations are relevant:

- The *Health and Safety Executive (HSE)* is the UK's independent licensing authority and regulates safety at all nuclear facilities in the UK.
- *Environment Agencies*: With regard to the authorization of the disposal of radioactive wastes from nuclear licensed sites, the *Environmental Agency (EA)* is the principal environmental regulator for England and Wales and the *Scottish Environmental Protection Agency (SEPA)* has broadly equivalent responsibilities in Scotland.
- *The Health Protection Agency (HPA)*, established in 2005, incorporates the National Radiological Protection Board (NRPB) and other agencies in a more holistic approach to protecting UK public health. The Radiation Protection Division of the HPA undertakes research and provides training courses on all aspects of ionizing radiation, as well as providing expert advice to government agencies.

ADVISORY and SUPPORT - The *Committee on Radioactive Waste Management (CoRWM)* is an independent, non-governmental advisory organization, established in 2003 by the government. The Committee provides recommendations towards government policy regarding radioactive waste management (see next section). In its updated role, CoRWM will review the long-term management option for the UK's HAW, in particular NDA's and the Government's proposals, plans and programs to develop a geological disposal facility, as well as interim storage.

At the request of the government CoRWM undertook an extensive program of public and stakeholder engagement as a basis for making recommendations towards the management of HAW in July 2006. The three key elements of CoRWM's recommendations were that:

- Waste should be managed by means of geological disposal.
- Implementation should be based on the principles of volunteerism and partnership between communities and implementers.
- Disposal should be preceded by safe and secure interim storage, mainly in storage pools but with some in dry storage.

The government accepted CoRWM's recommendations and in 2007, the Secretary of State published a White Paper on an energy strategy (UK, 2007) in which it emphasized the importance of nuclear power and the Government's conclusions in relation to the management of radioactive waste produced by new NPPs, *viz.*

> *"Having reviewed the arguments and evidence put forward, the Government believes that it is technically possible to dispose of new higher-activity radioactive waste in a geological disposal facility and that this would be a viable solution and the right approach for managing waste from any new nuclear power stations. The Government considers that it would be technically possible and desirable to dispose of both new and legacy waste in the same geological disposal facilities and that this should be explored through the Managing Radioactive Waste Safety Program. The Government considers that waste can and should be stored in safe and secure interim storage facilities until a geological facility becomes available."*

Following this White Paper, the Secretary of State issued another White Paper in 2008 specifically on nuclear power (UK, 2008b), confirming that nuclear power was a central component of its energy policy.

13.2.2 Legal and Regulatory framework

Figure 13-1 shows a schematic of the different responsibilities for environmental radiation protection in the UK.

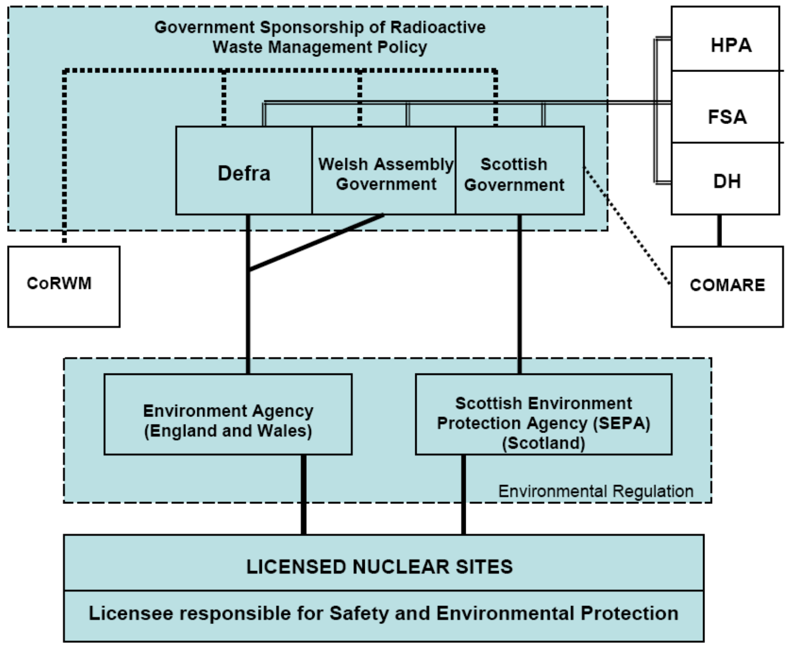

Figure 13-1
Entities in the UK pertinent to control of environmental effects from used fuel, reprocessing and radioactive waste management (DEFRA, 2008). Crown copyright 2008. Used with permission.

<u>*NOTE*</u>: COMARE = Committee on Medical Aspects of Radiation in the Environment; DH = Department of Health; FSA = Food Standards Agency.

The above organizations provide separate strands to the legislative and regulatory framework relevant to radioactive waste management in the UK. The first strand, via HSE, addresses nuclear safety and operational radiation protection aspects of irradiated nuclear fuel and radioactive waste management, and the second strand addresses, via the various UK Environment Agencies, environmental protection. Given this context, the relevant regulations are:

- The *Health and Safety at Work Act 1974 (HSWA74) and Nuclear Installations Act 1965 (NIA65):* The main legislation covering the safety of workers and the general public at nuclear installations, whereas NIA65 applies specific regulatory controls to nuclear plants. Under this Act, the HSE must grant a license before nuclear activities can proceed.
- The *Ionising Radiations Regulations 1999, (SI 1999/3232)*, issued under HSWA74, are designed to ensure that exposure to ionising radiation arising from work activities, whether man-made or natural, is kept as low as reasonably practicable and does not exceed specified dose limits for individuals. Enforced by the HSE.
- The *Radioactive Substances Act 1993 (RSA93)* is concerned with control over the security of radioactive materials and in ensuring the safe accumulation and disposal of radioactive waste. RSA93 is regulated in England and Wales by the Environment Agency (EA), in Scotland by the Scottish Environment Protection Agency (SEPA) and in Northern Ireland by the Industrial Pollution and Radiochemical Inspectorate (IPRI). RSA93 requires prior authorization to dispose of radioactive waste including that from nuclear installations.

- The *Energy Act 2004* established a new non-departmental public body, the Nuclear Decommissioning Authority (NDA), which took over the responsibility for decommissioning, and operation via civil contracts with operators pending decommissioning of designated civil nuclear legacy sites.
- The *Environmental Act 1995* established the basis for the regulatory framework with respect to environmental protection. It also established the EA and SEPA as regulators together with their funding arrangements.
- The *Town and Country Planning (Environmental Impact Assessment) (England and Wales) Regulations (TCPA)* and the *Planning (Environmental Impact Assessment)* Regulations (Northern Ireland) implement the national requirements relating to EIAs. In some instances, an application for planning permission may be requested by the relevant Minister for ministerial decision, which usually reflects the development being of national importance, e.g. the (previously) proposed L/ILW repository at Sellafield, Cumbria.
- The *Environmental Permitting (England and Wales) Regulations 2010* are intended to provide a simplified process for regulators to meet the objectives of 18 European Directives regulating emissions to air, water and soil, as well as waste management. These regulations provide a unified system for permitting waste operations, among other activities, including radioactive substance regulations.

With regard to government policy and radioactive waste management, the environment agencies (EA/SEPA/Department of Environment of Northern Ireland) issued "*Guidance on Requirements for Authorisation of Disposal Facilities on Land for Low and Intermediate level Radioactive Wastes*" (Environment Agencies, 1997). This Guidance established a series of principles and requirements intended primarily for the disposal of L/ILW primarily but also relevant to long-lived radioactive waste and, hence, HLW disposal. Of these principles and requirements, three requirements are particularly relevant:

- *Requirement R2 - Period after Control is Withdrawn (Risk Target):* After control is withdrawn, the assessed radiological risk from the facility to a representative member of the potentially exposed group at greatest risk should be consistent with a risk target of 10^{-6} per year (i.e., 1 in a million per year).
- *Requirement R3 - Use of Best Practicable Means:* The best practicable means shall be employed to ensure that any radioactivity coming from a facility will be such that doses to members of the public and risks to future populations are ALARA.
- *Requirement R5 – Multiple-Factor Safety Case*: The overall safety case for a specialised land disposal facility shall not depend unduly on any single component of the case.

The Environment Agencies have now updated this guidance document and issued separate, but similar, requirements for short-lived L/ILW and long-lived ILW/HLW. With regard to geologic disposal facilities, the wording for the new Requirements (having different numbers) corresponding to those above is:

- *Requirement R6 – Risk guidance level after the period of authorization:* After the period of authorization, the assessed radiological risk from a disposal facility to a person representative of those at greatest risk should be consistent with a risk guidance level[35] of 10^{-6} per year (i.e. 1 in a million per year).
- *Requirement R8 - Optimization:* The choice of waste acceptance criteria, how the selected site is used and the design, construction, operation, closure and post-closure management of the disposal facility should ensure that radiological risks to members of the public, both during the period of authorisation and afterwards, are as low as reasonably achievable (ALARA), taking into account economic and societal factors.
- *Requirement R3 – Environmental Safety Case*: An application under RSA93 relating to a proposed disposal of solid radioactive waste should be supported by an environmental safety case[36].

In addition, there is a Requirement, *R5*, on dose constraints during the period of authorization, which covers the operational phase and any period of active institutional control. These limits, described as the "upper bound of optimization", should not exceed a source-related dose (0.3 mSv/year) or a site-related dose (0.5 mSv/year). In this context, the Environment Agencies (2009) also note that the HPA has recommended a dose constraint of 0.15 mSv/year for exposure to the public from a new radioactive waste disposal facility.

A total of 14 Requirements are provided in the Guidance Document (Environment Agencies, 2009) and discussed in detail, as discussed in subsequent sections. No timeframe is specified in connection with the period after authorization, although it is acknowledged that the evolution of the disposal system "becomes increasingly uncertain with time".

13.2.3 Waste Classification

Radioactive waste is classified in the UK under a number of broad categories, essentially according to the heat-generating capacity and activity content of the waste (DEFRA, 2008). Thus the key categories are HLW, ILW, LLW, and VLLW. While LLW and ILW are not subdivided explicitly into short-lived and long-lived, the existing categories are generally in accordance with the definition of radioactive waste in the Joint Convention. A significant portion of the ILW component is equivalent to TRU waste in the US.

13.2.4 Funding

The prevailing principle with respect to HLW management is that waste generators bear the responsibility for assessing, planning for, and paying the costs associated with waste management, including disposal of HLW and used fuel (NEA, 2010). Given the complex history of nuclear power development, the execution of such a "polluter pays" philosophy is more

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[35] 'Risk guidance level' is intended to mean the same as the 'risk target' quoted in Environment Agencies (1997).

[36] Additional guidance refers to a multiple-function environmental safety approach, therefore similar to the earlier (1997) 'multiple' factor safety case'.

mixed. The NDA bears the responsibility for managing the sizeable portfolio of nuclear liabilities that remain in the public sector; resources to cover these activities come from direct government funding[37] as well as from revenue from commercial activities at NDA sites, such as electricity generation and ongoing fuel cycle services. Costs associated with management, conditioning, and disposal of wastes from the private sector are to be borne by the utilities that own the wastes, including used fuel and HLW (NEA, 2010).

In its review for the UK Government, CoRWM (2009) addressed the issue of funding and the legitimate concern of stakeholders regarding the conflict between the short-term nature of public spending and the long-term nature of geological disposal. Although not making specific recommendations for a long-term funding mechanism, CoRWM stated that "there is a need for Government and NDA to consider and explain more fully how they will ensure that appropriate funding will be available during the various phases of the implementation of geological disposal" (CoRWM, 2009). Funding for the regulation of radioactive waste management comes from waste producers, via cost-recovery schemes and NDA is involved in this process. Thus, CoRWM considers that ensuring NDA funding will also assure regulatory funding.

With regard to the funding of stakeholder activities, CoRWM recommended that the Government should fund engagement activities and the Government's White Paper appeared to accept this recommendation. Thus, CoRWM believes that the government is committed to resolution of this issue (CoRWM, 2009).

13.3 Site Screening, Selection, and Characterization

13.3.1 Overview

In the 1970's a standing Commission on Environmental Pollution examined a range of issues with implications on the quality of the environment, publishing reports based on its findings. One of these reports, published in 1976 (RCEP, 1976) and generally referred to as the Flowers Report, addressed issues associated with nuclear power, including the safety and siting of nuclear reactors, security and control of plutonium, energy strategy, and radioactive waste management (Chapter VII). With regard to radioactive waste management, the Commission reviewed a number of options, taking into account reprocessing of irradiated nuclear fuel. Thus, with regard to disposal, the Commission envisaged the disposal of vitrified HLW in stainless steel cylinders and recognized that associated timeframes in terms of safety would extend far beyond the hundreds of years necessary for fission products. The Commission stated (Paragraph 391):

> *"Neither the AEA nor BNFL in their submissions to us gave any indication that they regarded the search for a means of final disposal of highly active waste as at all pressing, and it appears that they have only recently taken firm steps towards seeking solutions. We think that quite inadequate attention has been given to this matter, and we find this the more surprising in view of the large nuclear programmes that both bodies envisage for the coming decades, which would give rise to much greater quantities of waste. In the last year, and possibly prompted by our own inquiry or by the availability of*

[37] Through the Department for Energy and Climate Change and the Treasury.

> *funds for the purpose from the European Commission in Brussels, the AEA have commissioned the Institute of Geological Sciences (IGS*[38]*) to conduct a desk study of the possibilities of disposal to a geological formation on land [Gray et al., 1976]. BNFL have asked the NRPB [National Radiological Protection Board] to examine the radiological consequences of disposing of blocks of vitrified high-level waste direct to the deep ocean."*

Soon after the publication of the Royal Commission's Report and the IGS report, geological investigations into geologic disposal began in different areas throughout England, Wales and Scotland, with a view to identifying a suitable site for geologic disposal. However, largely as a result of intensive public opposition, especially in Scotland, these geological investigations stopped in 1980.

Favorable characteristics identified by the Royal Commission included the following attributes:

- Lack of fissures or inter-granular pore spaces in the host rock;
- The ability of the host rock to withstand heat;
- The area in which the repository is located should be seismically quiet;
- The region in which the repository is located should be of low or moderate relief and remote from large bodies of surface water; and
- The repository should be at least 300 m below surface.

The Commission reviewed the advantages and disadvantages of the three principal types of rock considered internationally: rock salt, clay and hard crystalline rocks. For salt, the Commission identified salt being a valuable resource as the main disadvantage, which appeared to eliminate this type from further consideration.

Despite the lack of progress on identifying possible sites for geologic disposal, the report by Gray et al. (1976) did identify certain favorable properties that were relevant. Subsequently, Chapman et al. (1986) identified generic types of environment that would be suitable, including:

- Thick mudstones;
- Low relief fractured hard rock;
- Basement rocks under sedimentary cover (BUSC); and a
- Small island.

In June 2007, as the next stage of its new waste management program, the UK Government and corresponding administrations for Wales and Northern Ireland consulted on recommendations for how to select a site for the long-term disposal of HAW. The consultation document entitled "*Managing Radioactive Waste Safely: A Framework for Implementing Geological Disposal*" (UK, 2007) sought views on the technical aspects of developing a geological disposal facility and on the process and criteria to be used in deciding where the future facility should be located.

[38] IGS became the British Geological Survey (BGS) in 1984.

In the 1990's, geological studies focused on Sellafield in Cumbria, in the north-east of England, in connection with a potential repository for L/ILW. Twenty nine deep boreholes were drilled to investigate the geological characteristics around the potential site.

13.3.2 URL Program / International Involvement

In the early 1990's, as part of its investigations into a potential repository for L/ILW at Sellafield, Nirex identified the need for more detailed subsurface investigations via construction of an underground research laboratory (URL), referred to as a Rock Characterization Facility (RCF). The objective of this RCF was to investigate *in situ* the detailed properties of the potential host rock. A significant effort was devoted to obtaining the necessary planning permission for the RCF, with the application being submitted in 1994. Cumbria County Council rejected this application in December 1994 and, although Nirex appealed the decision, which resulted in a public inquiry, the rejection was upheld by the Secretary of State for the Environment in 1997, dealing a major blow to progress towards a GDF. Only now are proposals being sought for site investigations in connection with a geological facility (see Section 7).

Apart from this initiative, UK Nirex and its UK contractors (e.g. British Geological Survey) were involved in international URL programs. In addition, Nirex provided expertise to other national radioactive waste disposal programs involving geologic disposal.

13.4 Disposal Concept

Recent generic design options for the UK's HLW Geological Disposal Facility have been presented (Chapman et al., 2008). The NDA adopted three generalized geological host environments as a basis for exploring disposal concepts:

- 'Strong, hard rocks',
- 'Less strong, sedimentary rocks', and
- 'Evaporites'.

The resultant disposal concepts that were identified by Chapman et al. (2008) are discussed in detail in Appendix B to this volume. Table 13-2 (abstracted from Table B-2, Appendix B) summarizes these different concepts. As noted in this Appendix, internationally developed concepts for granite/ gneiss/ 'crystalline' rock (e.g. Finland, Sweden, China, Taiwan, Koreas, Switzerland) fall into the first group, clays/ marls/ argillaceous rocks (e.g. France, Belgium, Switzerland) into the second group, and dome and bedded salt formations (e.g. Germany, WIPP in the US) into the last group.

Table 13-2
Key features and variants leading to the used nuclear fuel and HLW disposal concepts (adapted from Appendix B, Table B-2 and Chapman et al., 2008).

Key Feature	Variants
In-tunnel (borehole)	Vertical borehole Horizontal borehole
In-tunnel (axial)	Short-lived canister Long-lived canister
In-tunnel (axial) with supercontainer	Small working annulus Small annulus + concrete buffer Large working annulus
Caverns with cooling, delayed backfilling	Steel MPC + bentonite backfill Steel or concrete/DUCRETE container + cement backfill
Mined deep borehole matrix	
Hydraulic cage	Around a cavern repository
Very deep boreholes	

13.5 Transparency and Stakeholder Involvement

13.5.1 Public Involvement

Generally in the UK, one of the statutory objectives of the environment agencies (EA and SEPA) is to encourage a close and responsive dialogue with all stakeholders – in particular the public, local authorities and other representatives of local communities and regulated organizations. To this end, HSE and the EA have corporate policies to promote transparency through the availability of information to the public.

More specifically, one of the Requirements of the Environment Agencies recent Guidance document for developing geologic disposal facilities (Environment Agencies, 2009), *R2 – Dialogue with potential host communities and others* - states that the "developer should engage in dialogue with the planning authority, potential host community, other interested parties and the general public on its developing environmental safety case for a geologic disposal facility".

Planning permission is obtained from the relevant local authority under the TCPA 1990 for England and Wales, or the TCPA 1997 for Scotland. The planning application process provides an opportunity to inform and obtain views from the public. For major developments such as a radioactive waste repository, this would take place through the public inquiry process.

Such a process was demonstrated when Nirex appealed the decision of the Cumbria Country Council to reject the application for a geological facility for L/ILW disposal. The Secretary of State noted (Parliamentary Office of Science and Technology, 1997) that the "poor design, layout, and arrangements for access of the proposed RCF, together with adverse impacts on visual amenities, a protected species (badgers), and the natural beauty of the English Lake District, were serious enough to warrant refusal of the planning application". A number of technical concerns related to scientific uncertainties were also specified.

Since this application refusal, there has been an extended period of consultation with all UK stakeholders, carried out by different bodies including the House of Lord's Select Committee on Science and Technology (SCCT), NDA, and the UK Center for Economic & Environmental Development (CEED).

In its enquiry into the management of nuclear waste, the SCCT invited evidence from a wide range of experts and stakeholders. Their report, *The Management of Nuclear Waste* (House of Lords, 1999), was published in March 1999. Although the comments are related primarily to the disposal of L/ILW, they are also relevant to HLW disposal. Thus, an extract of the major conclusions in the executive summary of the report is presented below:

"II. The bulk of nuclear waste that exists now and is certain to arise in the future originates from past military and civil nuclear programs. The problem exists and has to be solved. It could not be avoided by deciding today to discontinue nuclear power production or the reprocessing of spent fuel.

IV. The long time scales involved might be thought to be a reason for postponing decisions. The contrary is the case, since existing storage arrangements have a limited life and will require replacement and eventually the repackaging and transfer of stored waste. Reliance on supervision for very long periods increases the probability of human error.

V. We received a great deal of evidence on the technical issues and conclude that phased disposal in a deep repository is feasible and desirable….The phased approach which we recommend would allow decisions to be taken in a considered way as technical confidence and experience develop, and would avoid premature decisions which may be difficult to reverse.

VI. The future policy for nuclear waste management will require public acceptance…. Central to this is the need for widespread public consultation before a policy is settled by Government and presented to Parliament for endorsement.

VII. Present policy for nuclear waste management is fragmented. There are wastes for which no long-term management option has yet been decided and there are a number of significant materials, for which no use is foreseen, which are not categorized as waste at all. This leads to uncertainties in the planning of future facilities and to the continued storage of hazardous materials in an essentially temporary state.

VIII. These problems require changes in the present organizational structure for nuclear waste management."

CEED held a 'Consensus Conference' on radioactive waste management in London in May 1999. Pioneered in Denmark, such a medium is a relatively new way of involving the public in the assessment of key issues of science and technology. Consensus Conferences create a forum for a Citizens' Panel comprising members of the public, to take part in an informed debate with expert witnesses of their choice.

In this case, the Panel, made up of fifteen citizens recruited from throughout Britain, came together in London to debate the issue of radioactive waste management, following two weekends of intensive preparation. At the end of the conference, the Panel issued a report (CEED, 1999) on its views as to the key issues for circulation to the Government, media, and other interested parties. In this way, the report was able to initiate a debate in a topic area normally dominated by scientists and specialists. Most of the conclusions of the report are directly relevant to the possible development of a deep repository and are reproduced below.

- *"Radioactive waste must be removed from the surface and stored underground, but must be monitorable and retrievable. Cost cannot be an issue. We must leave options open for future solutions.*
- *We recommend the appointment of a neutral body by the Government to deal with waste management, including the selection of a national storage site. The criteria for site selection should be open and publicized.*
- *All institutions handling radioactive waste should conform to the same high standards, which should include random scrutiny.*
- *Research and development must be continued on a much larger scale and international cooperation should be encouraged.*
- *We see no problem with privatization within the nuclear industry if done properly, with adequate safeguards.*
- *At present, there is a lack of trust and understanding, and public awareness must be raised. The public needs to be fully informed of the problems and solutions available. Decision-making must be open and transparent. Radioactive waste issues should be made part of the Government's education strategy.*
- *We are not fundamentally opposed to nuclear power, but it should not be expanded until a way is found to deal adequately with the waste problem.*
- *A new and internationally accepted method of waste classification is needed that clearly and openly communicates information about nuclear waste to the public as well as industry.*
- *Existing international reprocessing contracts should be honored, but no new ones should be taken up.*
- *Finally, while the industry has in the past had a well-deserved reputation for secrecy, we have in the course of the conference noted a welcome shift in culture and a new feeling of openness in dealing with these difficult issues."*

Elsewhere, Nirex reviewed activities in connection with the RCF that led to refusal of its planning application. Nirex's findings (Hooper, 2001; Future Foundation, 2000) are summarized in terms of three key issues: process, structure, and behavior.

With regard to *process*, in the years leading up to the RCF Public Inquiry, decisions taken by Nirex were not transparent, and there was a lack of stakeholder involvement. With regard to *structure*, there was a lack of clarity and visibility to regulators and other stakeholders in both short-term and long-term decisions, in particular short-tem decisions that had long-term implications. With regard to the issue of *behavior*, Nirex called for a new openness, transparency, and accountability.

13.6 Safety Assessment and Licensing

13.6.1 Safety Assessment

As part of its submission to its application for an RCF at Sellafield (see Section 13.3.2), Nirex carried out a Preliminary Safety Assessment in 1995. Nirex described this assessment as a preliminary analysis rather than a comprehensive safety assessment. The objective of the assessment was to explore the impact on the risk of identified uncertainties in the hydrogeological conceptual model, in order to develop understanding of the groundwater flow and repository system. Since this preliminary analysis takes into account barriers (the EBS) that would not be typical of a HLW repository, it is not discussed further.

The Environment Agencies (2009) have given clear guidance on the requirements for authorization of geologic disposal facilities. In particular, they discuss in detail the elements of the environmental safety case, defining the latter as “a set of claims concerning the environmental safety of disposals of solid radioactive waste, substantiated by a structured collection of arguments and evidence. It should demonstrate that the health of members of the public and the integrity of the environment are adequately protected.” The Agencies view the development of the environmental safety case as an iterative process, with the level of detail required increasing as the project proceeds (see Figure 13-2). The Agencies specify the periods of the project where the dose constraint (operational period and any period of active institutional control) and the risk guidance level (post-closure period) apply. There is no quantification of the timeframe over which the post-closure assessment must be performed, however.

As stated in Section 13.2.2.1, one of the Requirements, *R6*, specifies that after the period of authorization, the assessed radiological risk from a disposal facility to a person representative of those at greatest risk should be consistent with a risk guidance level of 10^{-6} per year (i.e. 1 in a million per year).

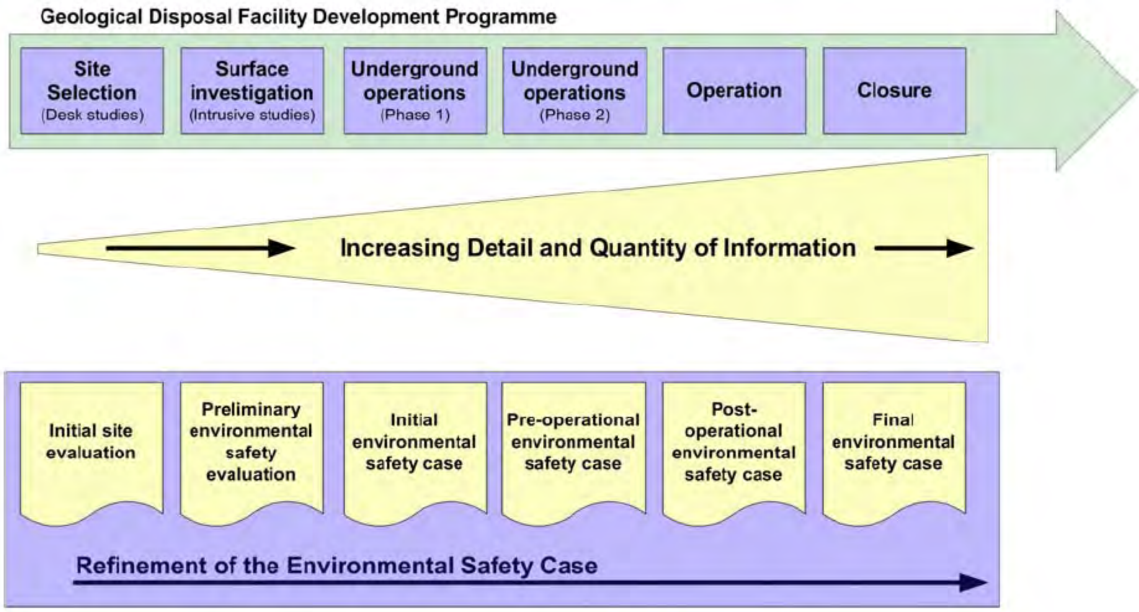

Figure 13-2
Refinement of the environmental safety case during the development program Copyright © Environment Agency 2009 all rights reserved. Used with permission.

13.6.2 Licensing Process

The form and structure of the license is the same for all nuclear facilities. The relevant considerations for facilities for irradiated fuel or HLW are shown in Table 13-3.

In the situation where the Secretary of State decides to review an application for planning permission under the TCPA, a local Public Inquiry must set up. In England and Wales the independent Planning Inspectorate arranges for one of its inspectors to hear and receive evidence for or against the proposal. The Inspector then makes a report and a recommendation to the Secretary of State for Communities and Local Government or to the Welsh Assembly Government.

Similarly, EA and SEPA will consult on a developer's application for the authorization of disposal of radioactive waste in a repository.

In determining applications for radioactive waste disposals on, or from, sites licensed under NIA65, the Environment Agencies consult other statutory bodies such as local and health authorities, fisheries and agriculture committees, in addition to the FAS and HSE. The agencies also undertake wide public consultation. After considering all the views expressed, they publish a "decision document" setting out their decision and the reasons behind it, including their response to issues raised during consultation. In Scotland, SEPA also consults with the Scottish Government for applications made.

Table 13-3
Licensing requirements for radioactive waste management facilities in the UK (adapted from UK, 2008a).

Activity	Legislation	Enforcing Authority	Type of License
Construction, commissioning, operation and decommissioning of any spent fuel or radioactive waste management facility required as a result of nuclear industry activities, including accumulation, and prescribed under the NIA65 cannot take place without a nuclear site license. [The license provides the powers to shut down any operations in the interests of safety.]	NIA65	HSE	Nuclear Site License
Keeping / using radioactive material (other than on licensed nuclear sites)	RSA93	EA (E&W) SEPA (S EHS (NI)	Registration
Accumulation of radioactive waste (other than on licensed nuclear sites)	RSA93	EA (E&W) SEPA (S EHS (NI)	Authorization
Disposal of radioactive waste	RSA93	EA (E&W) SEPA (S EHS (NI)	Authorization
Installations for: • Processing of used nuclear fuel or HLW • Final disposal of used nuclear fuel or radioactive waste •Storage of used fuel or radioactive waste in a different site from the production site	TCPA [EIA (England &Wales) Regulations]; EIA (Scotland) Regulations; Planning (EIA) Regulations (NI)	Local Planning Authority	Planning Consent (including EIA)

13.7 Current Status

Following the failure of a previous geologic disposal program, and after a period of extended consultation with UK stakeholders, the Government[39] published a White Paper describing the way forward for radioactive waste disposal (UK, 2008c), in particular the higher-activity component (HAW). The document describes the framework for implementing geologic disposal in the future in the UK. It is interesting to note, however, that with the partial devolution of government powers to Scotland and Wales, obtaining consensus across all countries of the United Kingdom is now more difficult.

In the preparation and planning for geologic disposal, the government identifies a "robust programme of interim storage" as an integral part of the implementation framework. With regard to a GDF, retrievability is discussed and, while no specific decision is made concerning this issue, the White Paper notes that in the planning and construction of such a facility, retrievability is not excluded.

A notable component of the future implementation framework is the site selection or site assessment process (see Figure 13-3) whereby the Government is committed to an approach based on "voluntarism and partnership" (UK, 2008c). The Government is also keen to ensure that the process is as flexible as possible. Thus, the White Paper identifies three types of community:

- *Host Community*: the community in the area where the GDF will be constructed;
- *Decision-Making Body*: the local government decision-making authority;
- *Wider Local Interests*: Communities other than the Host Community, which may have a vested interest in the construction of a GDF.

The Government acknowledges that close liaison among all three types of community will be essential to progress of a facility. In this context, two key early decision points are recognized (see Figure 13-3):

- Expression of Interest – the stage at which a community registers interest in being a host community, but with no obligation at this stage.
- Decision to Participate – the point at which a formal commitment to participate in the GDF siting process is made by the Decision-Making Body or Bodies.

[39] The Government in power, which features in this section, has actually been replaced and the country is under a different political regime although the commitment towards geological disposal appears not to have changed.

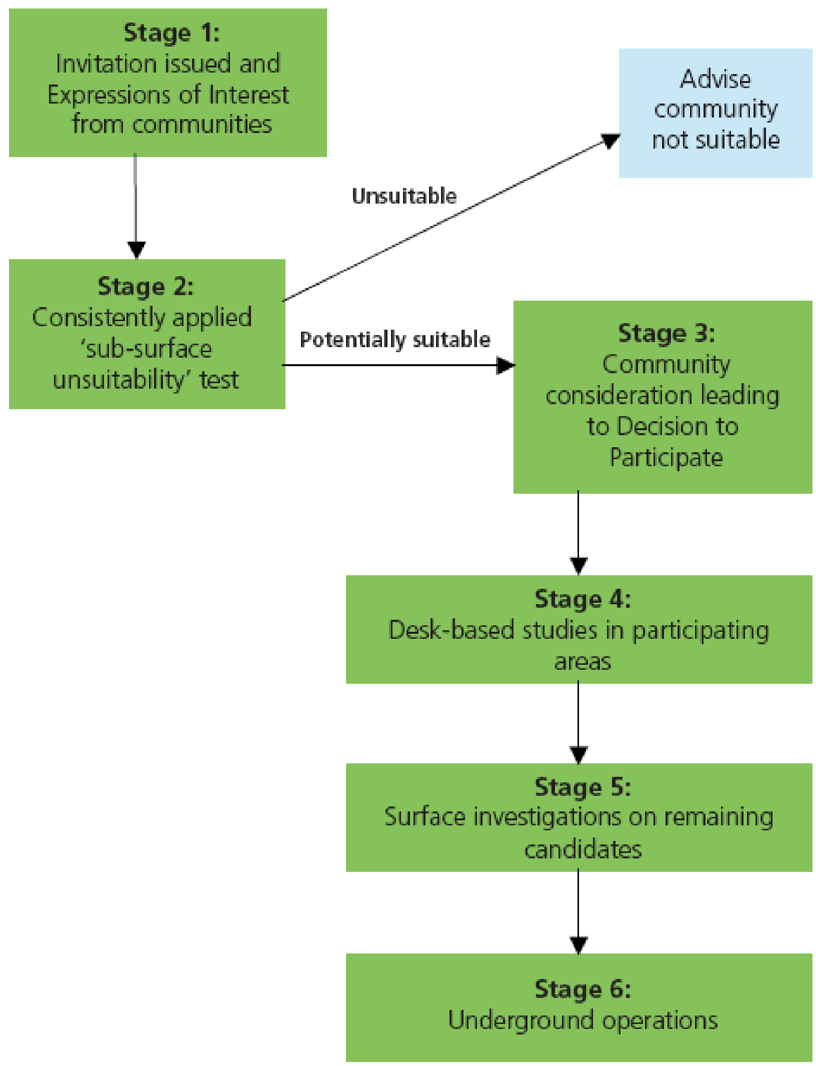

Figure 13-3
Schematic representation of the UK site selection process (UK, 2008c). Crown copyright 2008. Used with permission.

The partnership approach reflects the fact that more than host community will be impacted by the construct ion of a GDF. The voluntarism approach also includes the Right of Withdrawal whereby the host community can opt out up until underground construction begins.

Several EoIs have been made including one by Cumbria County Council (CoRWM, 2009), which interestingly was the local authority associated with the failed RCF application.

NDA's current strategic aim for HAW is to achieve passive safety as soon as reasonably practicable for interim storage and eventual disposal in a GDF. The division of NDA that is responsible for the development of a GDF, the RWMD has recently filled the key post of Research Manager, who is responsible for delivering a GDF and for coordinating the associated R&D program. In this context, the RWMD is in the process of establishing contracts to carry out a number of R&D activities including characterization work to support existing EoIs.

13.8 Summary and Key Observations

- *Policy on Geologic Disposal*: The UK is committed to geologic disposal, but the Government leaves the decision whether to reprocess irradiated nuclear fuel (and if so, when) or to seek alternative management options, to the commercial judgment of the owners of the irradiated nuclear fuel, subject to meeting regulatory requirements. The Government also adopts the position that irradiated nuclear fuel should not be categorized as waste while the option of reprocessing the fuel remains open and future use for the fuel is possible.

- *Institutional Arrangements*: NDA, a non-departmental public body, is responsible for the disposal and safe and secure interim storage of waste on civil nuclear sites. RWMD, a Division within NDA, has the responsibility for developing a geologic repository for higher-activity radioactive waste. The EA in England and Wales, and SEPA in Scotland, are the principal environmental regulators overseeing the disposal of radioactive wastes from nuclear licensed sites. Costs associated with management, conditioning, and disposal of wastes from the private sector are to be borne by the utilities that own the wastes, including used fuel and HLW. The NDA is funded by a combination of direct government funding and income from commercial activities.

- *Key Laws and Regulations*: The Energy Act 2004 established NDA as a new non-departmental public body. The Environmental Act 1995 established the basis for the regulatory framework for environmental protection and established EA and SEPA as the relevant regulators. More generally, The Radioactive Substances Act 1993 is concerned with ensuring the safe accumulation and disposal of radioactive waste. The Town and Country Planning (Environmental Impact Assessment) (England and Wales) Regulations (TCPA) and the Planning (Environmental Impact Assessment*)* Regulations (Northern Ireland) implement the national requirements relating to EIAs, which are part of the licensing process.

- *Site Screening and Selection*: In the 1970's, geological investigations related to geologic disposal began in different areas throughout England, Wales and Scotland, with a view to identifying a suitable site for geologic disposal. Largely as a result of intensive public opposition, these geological investigations stopped in 1980. Generic types of environment in the UK that were considered at an early stage as potentially suitable for locating a geologic repository are areas of thick mudstones, low-relief fractured hard rock, or basement rocks under sedimentary cover, and small islands. In a bid to pursue subsurface investigations concerning a subsurface repository for L/ILW, the implementer at the time submitted an application to construct a URL, but this application was rejected after a public hearing. Since then, NDA has adopted three generalized geological host environments as a basis for exploring disposal concepts: strong, hard rocks, less strong, sedimentary rocks, and evaporates. With regard to the future implementation framework for site selection or site assessment, the Government is committed to an approach based on "voluntarism and partnership".

- *Repository Design Concepts*: In a recent review conducted for NDA, twelve generic disposal concepts (Appendix B) have been recognized for the three different types of geological environments identified in the previous bullet point. With regard to retrievability, while no specific decision has been made concerning this issue, the Government in a White Paper notes that in the planning and construction of such a facility, retrievability is not excluded.

- *Performance Metrics and Assessments*: With regard to geologic disposal facilities, the Environment Agencies updated the previous guidance document, which provided a risk guidance level (the same as a risk target) after the period of authorization", *viz.* the assessed radiological risk from a disposal facility to a person representative of those at greatest risk should be consistent with a risk guidance level of 10^{-6} per year (i.e. 1 in a million per year). In addition, there is a requirement on dose constraints for the period of authorization, which covers the operational phase and any period of active institutional control. These limits must not exceed a source-related dose of 0.3 mSv/year or a site-related dose of 0.5 mSv/year. To date, no assessments have been carried out related to geologic disposal of HLW in the UK.
- *Independent Peer-Review and Advisory Bodies*: With regard to advisory bodies, CoRWM, an independent, non-governmental advisory organization, provides recommendations towards government policy regarding radioactive waste management. In its updated role, CoRWM will review the long-term management option for HAW, in particular NDA's and the Government's proposals, plans and programs to develop a geological disposal facility.
- *Stakeholder and Public Involvement*: One of the stated requirements of the Environment Agencies recent Guidance Document for developing geologic disposal facilities states that the "developer should engage in dialogue with the planning authority, potential host community, other interested parties and the general public on its developing environmental safety case for a geologic disposal facility". Since rejection of the implementer's application to construct a URL at Sellafield, the UK underwent a major period of public consultation aimed at obtaining consensus on the way forward.
- *Program Maturity*: The UK geologic disposal program is at an early stage, based largely on the program halting following major public opposition to geological investigation in the 1970's, with subsequent reluctance by successive governments to address the need for geologic disposal. Only recently, the NDA (RWMD) is more actively pursuing site selection in connection with a geologic disposal facility. In the context of future site selection / assessment, the Government is fully committed to an approach based on "voluntarism and partnership".

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14
SUMMARY

14.1 Introduction

All countries seriously pursuing programs for management of used fuel and/or HLW have chosen or are considering deep geologic disposal as a feasible means to provide safe, secure isolation from the biosphere. The specific approaches countries are taking with their disposal programs are as varied as the countries themselves. However, in spite of this variation, there are also notable similarities, convergences, and common experiences. Together this collective experience offers a wealth of information for informing planning and execution of geologic disposal programs, including in the US where the long-standing Yucca Mountain program has been effectively terminated by Executive Branch policy changes and actions.

As stated in the Introduction, twelve countries were selected as a representative sample in an attempt to capture a wide range of characteristics such as country size, population, and geologic diversity. Accordingly, the list should not be construed as complete but rather as illustrative. Table 14-1 provides a general overview of the countries in terms of nuclear generation, fuel cycle infrastructure and policy, and current management of used fuel.[40]

This Chapter summarizes key observations derived from the individual country reviews provided in Chapters 2 – 13, grouped by topic, to facilitate comparison of geologic disposal programs among the diverse collection of countries selected for this review. These are:

- Policy on geologic disposal;
- Institutional arrangements;
- Key laws and regulations;
- Site screening and selection;
- Repository design concepts;
- Performance metrics and assessments;
- Independent peer-review and advisory bodies;
- Stakeholder and public involvement; and
- Program maturity.

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[40] The information contained in Table 14-1 is derived from a number of sources and represents a snap shot in time subject to change. Given the complexity associated with the various topics addressed, the table itself reports status on a simplified basis with extensive footnotes providing more nuanced explanations where appropriate.

In addition to the detailed summaries below, a number of more succinct tables are provided at the end of this Chapter (Tables 14-2 to 14-4). The institutions that implement and provide funding for execution of disposal programs for HLW and used fuel are highlighted in Table 14-2. General site selection and repository design considerations are summarized in Table 14-3. Key attributes of national repository programs with respect to criteria for assessing repository performance and regulatory compliance are summarized in Table 14-4.

14.2 Policy on Geologic Disposal

Belgium - Although the decision to dispose of HLW has not yet been made, a Belgian research program on geologic disposal is well established. Belgian policy is reviewed every 10 years. A government decision on waste management of used nuclear fuel, including the option of reprocessing and the decision on HLW disposal, is expected in 2011.

Canada – Canada is committed to the geologic disposal of used nuclear fuel, although its first attempt to progress towards finding a suitable site ended in the mid-1990's when the government decided there was insufficient public support for the existing program. Since then, the implementer has placed emphasis on gaining a broad consensus among stakeholders for the geologic disposal program before proceeding further with site selection and characterization.

China - The Chinese government's official plan calls for the development of a geologic repository but at a pace that is compatible with the development of nuclear power and the construction of nuclear reactors.

Finland - Finland is actively pursuing a geologic repository for used nuclear fuel based on a once-through fuel cycle and has reached the licensing stage for the construction of such a repository. A license application is anticipated at the end of 2012.

France - France is firmly committed to the geologic disposal of HLW and submitted a feasibility report to the safety authorities in 2005. Following a favorable review, the French parliament set in motion the next phase of repository development, and research is continuing into the design and siting of the disposal facility.

Germany - In the 1970s, the German utilities planned to construct an integrated nuclear disposal center combining waste management facilities (reprocessing, fuel fabrication, waste conditioning) as well as a repository, all located at Gorleben. While exploration of the Gorleben salt dome has been on hold since October 2000, the country remains committed to repository construction for heat-generating radioactive waste, and the Gorleben site remains the principal candidate.

Japan - Japan's waste management policy has included geologic disposal for vitrified HLW since 1976. Japan is currently pursuing a volunteer approach to identifying candidate sites that will be thoroughly investigated in several stages leading to planned selection of a disposal site in the 2020's.

Spain - Geologic disposal is seen in Spain as the final waste management option for used nuclear fuel and HLW, although the most recent policy plan stated that no decision would be made on the final disposition of irradiated nuclear fuel and HLW before 2010. The bulk of the current effort is being devoted to developing a centralized facility for the storage of irradiated fuel.

Sweden - The current national policy actively being pursued is to dispose of used fuel directly in a deep geologic repository. Sweden has selected a site and is proceeding toward submittal of a license application for construction planned for March 2011.

Switzerland – Swiss law specifies the requirement for the safe and permanent management and disposal of all radioactive waste. Nuclear utilities are allowed to choose between reprocessing and direct disposal of the irradiated nuclear fuel. A geologic repository is to be constructed capable of accommodating both used fuel and HLW.

Taiwan – The Taiwan government plans formally acknowledge direct geologic disposal as the country's policy for irradiated nuclear fuel, and a domestic repository is being investigated. Reprocessing remains under consideration, as does the possibility of an international repository. The government's near-term waste management strategy focuses on interim storage.

United Kingdom - The UK is committed to geologic disposal as the final disposition path for used fuel and HLW, but the decision to reprocess and recycle used fuel or other alternative management options is left to the owners of the used nuclear fuel - subject to regulatory requirements. Used nuclear fuel is not categorized as waste while reprocessing and recycling remains a viable option.

14.3 Institutional Arrangements

Belgium - The Belgian National Agency for Radioactive Waste and Fissile Materials (ONDRAF/NIRAS) is the implementer responsible for radioactive waste management including repository development and is supervised by the Ministry of Energy. The Federal Agency for Nuclear Control (FANC) is the relevant nuclear regulatory authority in Belgium for disposal facility licensing and is under the Ministry of the Interior. Utilities pay into a dedicated fund for waste management managed by a commercial entity (Synatom) albeit with some government oversight.

Canada – The Nuclear Waste Management Organization (NWMO) is the implementer responsible for the long-term management of Canada's irradiated nuclear fuel and is overseen by the federal Ministry of Natural Resources (NRCan). The Canadian Nuclear Safety Commission (CNSC) regulates the emplacement of radioactive waste in a deep geologic repository. The CNSC stands as an independent federal agency but reports to the Canadian Parliament through NRCan. Financial resources for ensuring the long-term management of used nuclear fuel in Canada are mandated by law. The utilities and Atomic Energy of Canada Limited (AECL) make annual contributions to dedicated trust funds administered exclusively by NWMO.

China - The implementing body will be the Chinese National Nuclear Corporation (CNNC), the country's major (State-owned) nuclear utility. The regulatory authority is the Division of Radioactive Waste Management within the Department of Nuclear and Ionizing Radiation Safety Management under the supervision of the Ministry of Environmental Protection. Government plans call for the establishment of a dedicated irradiated nuclear fuel post-processing fund.

Finland - The two utility companies in Finland, Teollisuuden Voima Oy (TVO) and Fortum, joined together in 1995 to form Posiva Oy as an independent private company to plan, construct, and operate a geologic repository for final disposal of used fuel. The Radiation and Nuclear Safety Authority (STUK) is the independent Finnish nuclear regulatory authority. Licensees of nuclear facilities pay into a dedicated waste fund that is independent of the State budget but is controlled and administered by the Ministry of Employment and the Economy. The waste fund charges are set annually by the government according to the assessed liabilities for each company.

France - The implementer in France is the National Radioactive Waste Management Agency (ANDRA), an independent public body separated from the Atomic Energy Commission (CEA) in 1991, funded by government and industry, and supervised by the Ministries for Industry, Research and the Environment. The responsible regulatory authority, the Nuclear Safety Authority (ASN), authorizes the different stages of construction and operation of the URL as well as the eventual repository. ASN reports to the Ministries of Industry, Environment and Health. Funding for geologic disposal programs is provided principally by the reactor operators. Each principal nuclear operator in France (i.e., Electricite du France (EdF), AREVA, and CEA) administers its own separate, internal fund - subject to conditions set forth in law to ensure adequacy of the assets set aside.

Germany – The Office for Radiation Protection (BfS), under the authority of the Federal Ministry of the Environment, Nature Conservation and Nuclear Safety (BMU), is the implementer along with the Company for the Construction and Operation of Waste Repositories (DBE), which has the responsibility for the actual execution of repository development - planning, design, construction, and operations. In addition to its supervisory role of BfS, BMU is responsible for nuclear safety and radiation protection and has the oversight to enforce regulations issued by the regulatory authorities in the States. Recent government reforms in 2006 gave the federal government exclusive powers in the regulation of nuclear power such that the Länder only have legislative authority if explicitly authorized in federal law. Waste management costs from nuclear power generation are the responsibility of the waste producers – primarily the electric utilities. Utilities set aside and manage the funds internally to cover HLW management costs, including repository development, according to established legislative guidelines.

Japan - The Nuclear Waste Management Organization of Japan (NUMO) is the implementer responsible for HLW disposal, reporting to the Ministry of Economy, Trade and Industry (METI). NUMO is supported by a number of research organizations, in particular the Radioactive Waste Management Funding and Research Center (RWMC) and the Japanese Atomic Energy Agency (JAEA). The JAEA is responsible for constructing and developing two URLs. The key regulatory bodies with regard to geologic disposal are the Japanese Nuclear Energy Safety Organization (JNES) and the Nuclear Safety Commission (NSC). NSC is responsible for issuing regulations concerning geologic disposal, and JNES provides technical support to NSC. The utilities are responsible for sharing the costs of final disposal and make contributions to a disposal fund based on the amount of electricity generated. METI is responsible for allocating NUMO's budget, while RWMC manages the national fund.

Spain – The Empresa Nacional de Residuos Radiactivos (ENRESA), a state-owned company, is the implementer in Spain and reports to the Ministry of Energy. ENRESA's main role is to develop a reference framework for national spent fuel and radioactive waste management strategies. The Nuclear Safety Council (CSN) is the independent nuclear regulatory authority responsible for nuclear safety and radiation protection in Spain and reports directly to Parliament. Funding for the repository program is provided by utility payments into an interest-bearing fund in the form of a levy on electricity sales throughout the operational life of commercial reactors. The size of the levy is revised annually in light of projected costs for the back-end of the nuclear cycle and the fund balance.

Sweden – The Swedish Nuclear Fuel and Waste Management Company (SKB) is the implementer responsible for all radioactive waste management activities including repository development and is currently preparing for a license application for repository construction. The Swedish Radiation Safety Authority (SSM), formed by the recent merger of two separate agencies (SKI and SSI), serves as the independent nuclear regulatory authority. By law, the utilities pay into a dedicated waste fund on a per kWh basis subject to review by SSM and adjustment by the Government. The waste fund is administered by a government appointed Board of Governors.

Switzerland - The operators of nuclear power plants and the federal government formed the National Cooperative for the Disposal of Radioactive Waste (Nagra) as a private/public consortium to implement radioactive waste disposal in Switzerland, including HLW and used nuclear fuel if declared as waste. The Swiss Federal Nuclear Safety Inspectorate (ENSI, formerly HSK) represents the independent nuclear regulatory authority responsible for the safety of nuclear energy facilities and installations, including radioactive waste disposal. Current expenditures related to waste management while plants are operational are continuously paid for by the NPP operators as part of their annual budget. By law a separate dedicated waste fund was established to cover post-reactor-shutdown expenses associated with disposal via a levy of about 1 Swiss cent/kWh of nuclear power production.

Taiwan – Taipower, the owner/operator for all the nuclear power plants, is a state-owned utility under the Ministry of Economic Affairs, and has primary responsibility for the management of used nuclear fuel including disposal. Taiwan's nuclear regulator is the Fuel Cycle and Material Administration (FCMA) and is administered by the executive branch agency the Atomic Energy Commission (AEC). In terms of funding for geologic disposal, Taipower is expected to apply 2% or more of the nuclear post-processing (backend operation) fund towards final disposal research and development activities.

United Kingdom – The Nuclear Decommissioning Authority (NDA) is a non-departmental public body responsible for the disposal and safe and secure interim storage of waste on civilian nuclear sites. The Environment Agencies (EAs) in England, Wales, and Scotland are the principal environmental regulators overseeing the disposal of radioactive wastes from nuclear licensed sites. Costs associated with management, conditioning, and disposal of wastes from the private sector are to be borne by the utilities that own the wastes, including used fuel and HLW.

14.4 Key Laws and Regulations

Belgium - The key regulation governing radiation protection is GRR-2001, the Royal Decree of July 20, 2001. This umbrella regulation specifies an annual dose limit of 1 mSv/year to members of the public from all controllable practices and sources. No timeframe is specified in GRR-2001 in the context of the period over which radiological assessments should be performed. No regulations have been prepared specifically addressing geologic disposal.

Canada – The Nuclear Safety and Control Act of 2000 (NSCA) established clear responsibilities on the part of owners of irradiated nuclear fuel concerning long-term waste management approaches and led to the formation of NWMO. With regard to regulations addressing geologic disposal, the regulatory document R-104 was issued by the CNSC's predecessor and provided the basis for AECL's 1994 assessment. The CNSC's equivalent document to R-104 issued in 2006 is Regulatory Guide G-320.

China - The Radioactive Pollution Prevention and Treatment Law, passed in 2003, requires HLW disposal from all sources in a central repository located in a deep geologic formation. While China does not have specific regulations and technical standards addressing the geologic disposal of HLW, three government organizations were involved in drafting a recent document covering key technical and safety issues associated with geologic disposal.

Finland - STUK developed its regulations largely based on a document prepared originally by the combined Nordic Radiation Protection and Nuclear Safety Authorities (the Nordic Flag Book) and published in final version in 1993. The Finnish Government provided STUK with additional guidance in its Decision 1999/478, which defined the regulatory compliance period as "*an assessment period that is adequately predictable with respect to assessments of human exposure but that shall be extended to at least several thousands of years.* […] *Beyond the assessment period referred to above, the average quantities of radioactive substances over long time periods, released from the disposed waste and migrated to the environment, shall remain below the nuclide specific constraints defined by the Radiation and Nuclear Safety Authority.*" Current regulatory guidelines by STUK (YVL 8.4 and YVL E.5) state that calculations for the quantitative safety assessment be extended to at least "several thousand years" after the closure of the repository (the so-called 'environmentally foreseeable future") to ensure that: (1) the annual effective dose to the most exposed members of the public remains below 0.1 mSv, and (2) the average annual effective doses to other members of the public remains insignificant. The acceptability of these doses depends on the number of exposed people, but they must not be more than 1/100 to 1/10 of the constraint for the most exposed individuals, i.e. no more than 0.001 to 0.01 mSv/year.

France - A 1991 Law changed the direction of radioactive waste management in France and put into motion a 15-year research period into the feasibility of deep geologic disposal, with a stepwise approach to nuclear waste management. Following ANDRA's successful completion of this 15-year research phase, Parliament passed the 2006 Planning Act authorizing ANDRA to continue its research using its URL and to pursue a license application for repository construction - subject to the regulatory findings and recommendations.

<u>Germany</u> – A hierarchy of laws, ordinances, and regulations govern radioactive disposal in Germany. New safety requirements for the geologic disposal of heat-generating radioactive waste were issued in 2009 covering the planning, site exploration, construction, operation and closure of a final repository. Site selection and licensing aspects are not addressed.

<u>Japan</u> - No specific regulations for HLW disposal exist in Japan, only guidelines. The NSC is conducting workshops to solicit public input and is responsible for developing safety requirements for geologic disposal. In the absence of regulations, existing AEC Guidelines identify annual dose as the basic indicator of safety as well as stressing the importance of taking into account international (International Commission on Radiological Protection, ICRP) standards.

<u>Spain</u> - The licensing process for nuclear and radioactive waste management and disposal facilities is governed by the Regulation on Nuclear and Radioactive Facilities. Other relevant laws include Royal Decree 35/2008, concerning the operational requirements for nuclear facilities, and Instruction IS-08 issued by CSN on criteria concerning radiological protection.

<u>Sweden</u> - The Environmental Code (SFS 1998:808) contains a requirement for the issuance of permits for siting of the deep repository and the preparation of an Environmental Impact Statement (EIS) as part of the license application. The rules on EIS are adopted from European Union Directives (85/337/EG and 97/11/EG). The Code also imposes limits on the environmental impacts of the deep repository. The Nuclear Activities Act stipulates that the owners of the reactors, as waste producers, are responsible for all measures needed to execute and fund management of the nuclear sector's lifecycle, including the disposition of nuclear waste and decommissioning of facilities. The Nuclear Activities Act also specifies the funding mechanism for radioactive waste management. The Radiation Protection Act prescribes in general that with regard to conducting activities with radiation, measures and precautions are needed to prevent harm to humans, animals and the environment. Ordinances issued by the Government pursuant to the Nuclear Activities Act and the Radiation Protection Act contain some more detailed provisions and regulate the activities of the nuclear regulatory authority SSM. SSM regulations also call on the implementer to consider optimization and best available technology (BAT).

<u>Switzerland</u> - The new Nuclear Energy Law of 2005 (KEG) provides an extensive revision of Switzerland's legal framework. KEG keeps the nuclear option open and addresses a number of key issues related to radioactive waste, including a concept of monitored long-term geologic disposal of radioactive waste that combines elements of final disposal and reversibility. The Radiological Protection Act (StG) and the Radiological Protection Ordinance (StSV) form the legal basis for radiation protection in Switzerland, the latter specifying the annual dose limit for individuals of the population to 1 mSv per year. ENSI's Regulatory Guideline ENSI-G03 specifies the requirements for long-term safety of disposal facilities.

<u>Taiwan</u> – The Nuclear Materials and Radioactive Materials Management Act states that the waste producer is responsible for the treatment, transportation, storage and final disposal of the waste. This Act also states that producers of irradiated nuclear fuel should have a program to understand the feasibility for constructing domestic facilities for the final disposal of this waste.

United Kingdom – The Energy Act of 2004 established NDA as a new non-departmental public body. The Environmental Act 1995 established the basis for the regulatory framework for environmental protection and established EA and SEPA as the relevant regulators. More generally, The Radioactive Substances Act 1993 is concerned with ensuring the safe accumulation and disposal of radioactive waste. The Town and Country Planning Regulations (TCPA) providing for Environmental Impact Assessments in England and Wales and the Planning Regulations, providing for Environmental Impact Assessments in Northern Ireland, implement the national requirements relating to EIAs, which are a key part of the licensing process..

14.5 Site Screening and Selection

Belgium - Studies within a late 1970's EC program as well as an independent Belgian study identified only argillaceous rocks as being potentially suitable in Belgium, from which two main groups were recognized: hard rocks (shales) and poorly consolidated, plastic clay. Preliminary results from studies of the latter identified Boom Clay as being suitable, its key features being low hydraulic conductivity, reducing conditions (Eh = -0.25V), and high retention properties for most radionuclides. A decision was made to construct an underground R&D facility (HADES) on and under the premises of SCK•CEN at Mol-Dessel. The initial construction phase started in 1980 and resulted in the completion of the URL in 1983.

Canada – A 1970's Commission recommended emplacement of used fuel in a deep underground repository within the Canadian Shield (i.e., crystalline rock). AECL then moved towards constructing an underground facility for detailed *in situ* geological studies, also incorporating in its URL program experiments and testing of emplacement techniques. Reconnaissance studies and surveys were used to identify an appropriate site for the URL in a granite intrusion. Recent site characterization of the low-permeability sedimentary formations at the Bruce reactor site in Ontario for the disposal of L/ILW operational wastes may lead to an expansion in the rock types considered for future disposal of Canadian used fuel.

China - In the absence of specific regulations, China's siting program follows, in principle, the IAEA guidelines. Six locations, distributed throughout the country, were selected initially as the basis for initiating site selection. Currently, the R&D work is focused only on a long-term feasibility study of a site located in northwest China, Beishan, in the Gobi desert. The area has no groundwater resources, no important mineral resources, a relatively low population density, and no economic prosperity. The host rock is primarily granite with low permeability and porosity with low water outflow in the area. The deep groundwater is almost stagnant with a circulation time of thousands of years. Chemical analysis indicates high salinity, slight-alkaline, and reducing conditions. While site investigation studies are continuing at Beishan in the Gobi desert, existing national policy requires comparative site selection before the final choice is made.

Finland - TVO (superseded by Posiva in 1996) has been conducting a step-wise program for screening and selecting candidate sites for a final geologic repository since 1983. As a first step in screening of sites, TVO evaluated the bedrock of all of Finland during 1983-1985 in order to locate potentially suitable candidate sites for further characterization. Following the development of a range of screening and geoscientific criteria, TVO provided summary reports on 102 sites to

STUK and Government authorities in late 1985. In 1987, following review and recommendations by STUK and the government, TVO selected five areas for preliminary site characterization, which included many of the principal candidate rock types, all within the category of crystalline rock. The actual choice of these sites was influenced by land ownership and discussions with local municipalities with regard to their acceptance for further characterization as a potential repository site. From 1993-2000, detailed characterization was carried out on four sites, including drilling deep boreholes, from which one site was identified for further detailed evaluation, on Olkiluoto, a small island to the southwest of Finland, and already hosting two nuclear power reactors.

France - Early studies identified two types of generic rock, granite and clay, as being suitable for geologic disposal. Subsequent geological investigations towards identifying a site for geologic disposal due to intense socio-political opposition. The 1991 Law required ANDRA to identify at least two sites that would be suitable for geologic disposal, one in clay and one in granite. From ANDRA's initial investigations, four sites were originally proposed for more detailed investigations. Of these, one was rejected on hydrogeological grounds and two clay sites in the East Paris basin were combined, leading to detailed geological investigations on two sites, one in clay (Bure, east Paris Basin) and one in granite (Vienne, western France). Eventually, based on a National Commission's review of ANDRA's submissions for each site, the government approved continued research at the clay site but rejected the granite site based on technical concerns of the review commission. The Bure site consists of a series of almost horizontal layers, with relatively simple geology, low tectonic activity, and low permeability. Homogeneity of the clay appears to be relatively high with few discontinuities. Geochemical conditions are reducing, with pH buffered by carbonate minerals.

Germany – Salt was recognized in the early 1960's as a potential geological medium for a HLW repository, and a large-scale study between 1964 and 1976 was devoted to the identification of potential salt structures in Germany. Criteria for suitable salt formations were formulated in 1964. Of 140 salt domes selected initially for evaluation, 23 were selected for further study. Further refinement in selection criteria reduced the number of sites to 13 and, eventually 4 sites, after applying criteria in terms of (i) safety and the environment; (ii) infrastructure (transportation); (ii) structural policy. After detailed site investigations involving the four candidate salt sites, Gorleben was eventually selected by Lower Saxony as a suitable site. This salt dome consists primarily of pure rock salt layers, which are largely solution-free. An impermeable salt barrier, approximately 600 m thick, extends across the planned emplacement area to the overburden. Further study was halted for political reasons, although the consensus of experts who have reviewed various documents on Gorleben is that the results "do not contradict the positive appraisal of the geological findings at the Gorleben site". More recently, Germany is considering the possibility of geologic disposal in formations other than salt, e.g. low-permeability argillaceous rocks as an alternative.

Japan - Site selection is a greater challenge in Japan than in most other countries, because of its active tectonic setting at the juncture of multiple convergent plate margins. At the end of a general geologic review period leading to the submission of a Progress Report in 1992, AEC had hoped for a wide range of sites to be identified covering most of the geological features in Japan, but this did not happen. More recently, in keeping with the 2000 Act, Japan's current strategy is a volunteer approach, whereby communities volunteer as potential candidate sites. Three stages are envisioned, an initial literature survey stage, followed by a preliminary investigation area

(PIA) stage, and a final detailed investigation area (DIA) stage leading towards the selection of a site in the 2020's. To aid the first step in the siting process, NUMO announced an overall procedure for selecting potential candidate sites, followed by the identification of siting factors to be provided to all municipalities in Japan. To date, no community has volunteered. Meanwhile, research continues in two URLs, one in crystalline rock (Mizunami) and the other in sedimentary rock (Horonobe).

Spain - Detailed analysis of potential repository sites in Spain was carried out between 1986 and 1998 under a Site Selection Plan that identified the sequential analysis to be performed at different scales, from the identification of suitable geologic formations, to screening studies at the regional level, down to detailed studies at a more local scale. Although ENRESA's 3rd and 4th research plans identified several sites for detailed characterization in the period from 2000 to 2010, no further geological studies towards geologic disposal were authorized after the 5th plan was issued in 2001. These initial studies identified a number of deposits of clay (smectites) in different regions of Spain as being potentially suitable. Additional research activities were carried out in a granite/U-bearing quartz vein system at El Berrocal, which has been studied as a natural analogue of a HLW repository.

Sweden - SKB's plan for identifying a site for a geologic repository identified three stages: general siting and feasibility studies, site investigations, and detailed investigations including subsurface. SKB were criticized at an early stage by the regulator and the government for not having site selection criteria to guide the site selection process. SKB updated its approach to include general siting factors based on engineering, safety, environmental and societal considerations. SKB was requested to make its factors more specific. In its 1998 research plan, SKB formulated the interim goal of being able to choose at least two sites for site investigations in 2001 and funded a project to quantify requirements and preferences for a suitable repository site for a more systematic approach to site selection. From six feasibility studies completed in 2000, SKB selected three sites (later reduced to two sites, Laxemar and Forsmark) for more detailed investigation in two stages. The initial site investigation stage was completed in 2006 with the publication of the SR-Can report, in which SKB formally compared the performance of a standard "KBS-3 type" repository design using site-specific data from Laxemar and Forsmark. Both sites were shown to comply with the regulator's 100,000-year risk criterion, and were accordingly judged to be suitable for selection as the repository site. Finally, after a multi-year set of site characterization programs, SKB formally announced in 2009 its selection of the Forsmark site as the site for the deep geologic disposal of used fuel in Sweden. As part of its research activities, SKB constructed the Äspö underground Hard Rock Laboratory between 1990 and 1995 and continues to carry out experiments and testing in this facility.

Switzerland - Three host-rock variants have been considered —either the crystalline basement or one of the two overlying, low-permeability sediment layers. Geological studies showed that the extent of accessible crystalline basement was much less than originally thought. Thus, only two relatively restricted areas remain for the selection of a possible site, each covering an area of about 50 km^2. Despite this limitation, Nagra believes that it would be feasible to find a suitable repository for Switzerland's required low volume of waste. In parallel with the crystalline basement studies, investigations of the sedimentary options proceeded from desk studies to select potential host formations to identification of potential siting areas. With regard to sedimentary formations, Nagra identified Opalinus Clay as the top priority and conducted a field program in the potential siting area. Geological investigations have benefitted from two URL programs, at Grimsel in crystalline rock, and at Mont Terri for clay studies.

Taiwan – Geological survey information obtained during an early study (late 1980's, early 1990's) indicated that potential host rocks including granite, thick shale and mudstone layers, exist at appropriate depths in Taiwan. The Spent Nuclear Fuel Final Disposal Plan approved by AEC in 2006 identified a five-stage process to select a suitable site, starting with regional investigations. Such geologic investigations are ongoing, including exploratory boreholes. This stage is expected to be completed in 2017 with the preparation and submission of a technical feasibility report. The area currently under study is in a granite formation on Kinmen Island, west of the Taiwan Strait and close to mainland China.

United Kingdom – In the 1970's, geological investigations related to geologic disposal began in different areas throughout England, Wales and Scotland, with a view to identifying a suitable site for geologic disposal. Largely as a result of intensive public opposition, these geological investigations stopped in 1980. Generic types of environment in the UK that were considered at an early stage as potentially suitable for locating a geologic repository are areas of thick mudstones, low-relief fractured hard rock, or basement rocks under sedimentary cover, and small islands. In a bid to pursue subsurface investigations concerning a subsurface repository for L/ILW, the implementer at the time submitted an application to construct a URL, but this application was rejected after a public hearing. Since then, NDA has adopted three generalized geological host environments as a basis for exploring disposal concepts: strong, hard rocks, less strong, sedimentary rocks, and evaporates. With regard to the future implementation framework for site selection or site assessment, the Government is committed to an approach based on "voluntarism and partnership".

14.6 Repository Design Concepts

Belgium - Two reference designs exist, one for HLW disposal and one for used nuclear fuel, with a multi-barrier disposal concept for each – stainless steel waste container, clay-sand-graphite buffer, and concrete lining surrounded by the host clay formation. The repository layouts are on a single level. While emphasizing that disposal implies no intention to retrieve waste, ONDRAF/NIRAS has discussed and identified certain aspects of the repository design that allow retrievability over a limited period of time.

Canada – The federal government and the provincial government of Ontario both stated (1980) that selection of a site for HLW disposal would not proceed until the concept for disposal had been reviewed and assessed. Thus, a generic, rather than site-specific, disposal concept was developed and AECL was given the responsibility to develop a disposal concept compatible with disposal in the Canadian Shield. This multi-barrier disposal concept consisted of a titanium overpack with inner steel container as waste package, and bentonite buffer. The most recent repository conceptual design involves horizontal emplacement in a repository at a generic location in the Canadian Shield. The current reference waste package design has an inner carbon steel lining (80-100 mm thick) and an outer oxygen-free phosphorus-doped (OFP) copper shell (25 mm thick), the interior of the steel vessel being filled with an inert gas such as helium. The design life for such a reference container is 100,000 years, comparable with the design lifetimes of Swedish and Finnish canisters. The stepwise management approach now being adopted by

NWMO allows for retrievability of the irradiated nuclear fuel over an extended period, until future society makes the determination on final closure.

China - The waste form is HLW. According to current plans, the final repository will be located in a deep granite formation. The intended engineered barrier system (EBS) is similar to the Swedish KBS-3 type concept, with similar tunnels and shafts, and bentonite as the buffer material, 0.8m thick, but with vitrified HLW as waste form. The vitrified waste form will be contained in a 0.07 m thick (inner) canister. The possibility of retrievability has not been discussed.

Finland - At the same time as TVO submitted its list of 102 potential sites to the government, it also established the Swedish KBS-3 repository concept as its reference in order to begin relating future siting activities with future research, development and design activities on the repository. The reference design is for vertical emplacement in a one-level underground facility with disposal tunnels at a depth of approximately 420 m. With regard to the EBS, whole used fuel assemblies will be sealed inside a canister structure consisting of a massive cast iron insert covered by a 50-mm thick copper overpack. Canisters will be placed individually inside deposition holes spaced at intervals along the floor of long, horizontal deposition tunnels, the void spaces between the canister and rock in the deposition hole being filled with rings and blocks of compacted bentonite. Each deposition tunnel will be backfilled with a clay-rich material after all deposition holes have had waste packages and buffer emplaced. Retrievability does not play a part in Finland's disposal concept.

France - A Safety Rule issued to complement the 1991 Law specifies a multi-barrier concept. France also places a high priority on reversibility. The disposal concept calls for the entire repository to be laid out on a single level in the middle of the clay formation (Callovo-Oxfordien). The disposal concept also calls for the co-disposal of HLW, a small quantity of irradiated nuclear fuel, and long-lived ILW, with separate *zones* for different types of waste, each zone separated by a substantial thickness (250 m) of clay. Tunnels will have a low-alloy steel lining during waste emplacement, for stability and to aid retrievability. The overpack for HLW consists of 55-mm thick P235 steel, with a lid of similar material electron beam welded to the casing. The overpacks are fitted with ceramic runner pads to facilitate emplacement. ANDRA's disposal concept for HLW has no engineered clay barrier but relies on the hydraulic properties of the natural clay. Retrievability and reversibility are important considerations in France's repository concept and design.

Germany – In the German disposal concept, the geological barrier plays the most important role in isolating the waste, i.e., the rock salt in the case of Gorleben. As well as providing a mechanically stable rock formation, the self-sealing property of salt allows total isolation. For this reason, retrievability plays no part in the German disposal concept. The effort in Germany in terms of design concept has been devoted to the development of thick-walled casks, which reached the demonstration stage only. Crushed salt is the intended backfill.

Japan - Japan's repository design concept favors the multi-barrier approach. In the absence of a specific site, the disposal concept is similar to that of Switzerland's disposal concept for granite host rock: carbon steel overpack for vitrified HLW, surrounded by bentonite-sand buffer.

Spain - The initial reference concept developed by ENRESA conforms to the multi-barrier principle but exists only at a fundamental level. A generic-type reference conceptual design was first developed for a repository for irradiated nuclear fuel in a granite formation. The repository design comprises horizontal galleries at a depth of about 500 m, waste packages consisting of irradiated nuclear fuel in carbon steel canisters with a service life of at least 1,000 years. The buffer that surrounds the steel canisters consists of blocks of compacted bentonite. The conceptual design has evolved since 1994 aiming for compatibility with repository concepts in clay and salt.

Sweden - SKB's basic conceptual design for disposal, known as "KBS-3", calls for the waste package to be encapsulated in copper/cast iron canisters, which will be emplaced in vertical deposition holes and surrounded by a bentonite-clay buffer at a depth of 400-700 m in the Swedish basement crystalline rock. The buffer barrier is an important component in terms of performance, providing a diffusive barrier both to the inward transport of reactants as well as the release of radionuclides once a canister has failed. More recently, SKB has also considered a KBS-3 design with horizontal rather than vertical emplacement of canisters.

Switzerland - The current Swiss multi-barrier disposal concept involves geologic disposal at a depth of about 500 m to 1 km below surface, with in-tunnel emplacement of HLW packages in crystalline or sedimentary rock that physically protects the EBS, has low water flow, favorable groundwater chemistry (reducing conditions) and, in the case of Opalinus Clay, provides an efficient radionuclide transport (diffusion) barrier. In addition to the vitrified waste in its steel fabrication canister or irradiated UO_2/MOX fuel within its cladding, a thick steel overpack is envisaged, surrounded by compacted bentonite clay. Co-disposal of long-lived ILW is planned, but in tunnels in a separate part of the repository.

Taiwan – Taiwan has largely adopted the general KBS-3 disposal concept of the Swedish program, with a bentonite buffer, although the specific disposal orientation, whether vertical or horizontal emplacement, has yet to be decided.

United Kingdom – In a recent review carried out for NDA, twelve generic disposal concepts (Appendix B) have been recognized as suitable for the appropriate geologic environment. While no specific decision has been made concerning retrievability, the UK Government has indicated that retrievability is not excluded from potential attributes of an eventual repository design.

14.7 Performance Metrics and Assessments

Belgium - In the absence of specific regulations to guide the long-term safety evaluation, ONDRAF/NIRAS prepared and submitted two 'feasibility' assessments (SAFIR and SAFIR2) for a deep geologic repository in clay. ONDRAF/NIRAS considered timeframes out to a million years and presented results in terms of annual dose (Sv/year). Peak dose was on the order of 10^{-5} Sv/year for both HLW disposal and used fuel disposal.

Canada – The regulation used as performance metric for AECL's performance assessments was the former Atomic Energy Control Board's Regulatory Document R-104 in which the regulator required that the risk be summed over "*all significant scenarios*" and that the estimated annual risk be less than 10^{-6} per year, specified as being equivalent to an annual dose of 0.05 mSv per year. With regard to timescale of concern, R-104 stated: "*The period for demonstrating compliance with the individual risk requirements using predictive mathematical models need not exceed 10,000 years. Where predicted risks do not peak before 10,000 years, there must be reasoned arguments that beyond 10,000 years the rate of radionuclide release to the environment will not suddenly and dramatically increase, and acute radiological risks will not be encountered by individuals.*" While no time limit is specified in the current regulatory document, Regulatory Guide G-320, similar to R-104, the assessment timeframe is expected to include the time of maximum impact. In addition, the assessment itself is expected to provide a rationale for timeframe. AECL's assessment results, submitted as part of an EIS, presented as the estimated mean dose rate and risk to an individual of the critical group out to 100,000, yielded a maximum risk more than an order of magnitude less than the regulatory criterion at the time. The most recent assessment calculated the predicted annual dose for the most exposed member of the critical group, a self-sufficient farmer residing near the surface discharge location. The resultant peak dose of ~10^{-7} Sv/a occurred after nearly 500,000 years.

China - While no known performance assessments have yet been carried out, Chinese scientists are acquiring knowledge and experience in assessment tools. A formal performance assessment is anticipated in the near future.

Finland - Specific performance metrics are provided in STUK's regulatory documents. For quantitative safety assessment, the STUK requirements lead to the identification of three time periods:

- *0 to ~10,000 years*: the "environmentally foreseeable future" and the period that STUK's defined dose rate constraints apply. Biosphere transport and dose assessments need to be performed for those radionuclides that might be released to the biosphere during this period.
- *10,000 to several 100,000's of years*: the period of "large-scale climate changes" when episodes of permafrost and glaciations are expected, and radiation protection criteria are based on STUK's geo-bio flux constraints. No biosphere analyses are needed and dilution plays no role in the fulfillment of the regulatory constraints. Doses can still be used as safety indicators to gain additional insight to repository performance during this period.
- *Several 100,000's to 1,000,000 years*: A "very far future" period for which no rigorous quantitative safety assessment is required but the judgment of safety can be based on more qualitative considerations.

Assessments were carried out in support of the site selection program and, more recently, to support the EIA as part of the license application submission. In the EIA safety analysis, no one specific site was considered, but the four potential host sites. Detailed descriptions of the natural and repository systems were given for five time periods: first hundred years; 100-10,000 years; 10,000-100,000 years; 100,000-1,000,000 years; and beyond a million years. While no releases were obtained for the Reference Scenario (intact canisters), the possibility of one or more canisters failing over a one million year time period could not be discounted. Thus, variants of the Reference Scenario included a leaking canister, either as a result of a small pinhole (cross-sectional area ~5 mm^2), or from a larger hole (~1 cm^2) as well as a scenario involving the total failure of a canister. The peak dose from the latter scenario, the most constraining, yielded an annual dose < 10^{-7} Sv per year. The most recent assessment, of a KBS-3H (horizontal emplacement of canisters) yielded results that complied with the regulatory criteria.

France - The reference situation for geologic disposal is provided in the 1991 Safety Rule that complemented the 1991 Law. The Safety Rule provides an annual dose limit to an individual of 0.25 mSv/year over a post-closure period of 10,000 years. Beyond 10,000 years, 0.25 mSv/year is treated as a reference rather than a strict limit. The Rule also provides guidance on the assessment process itself. ANDRA's assessment (Argile 2005) of the Reference Scenario yielded a peak annual dose of 0.022 mSv/y, due primarily to radionuclide releases from irradiated nuclear fuel. ANDRA also tested barrier functions with specific scenarios and also considered a relatively extreme, severely degraded scenario that combined a higher permeability of the clay host rock with defects in seals, waste packages and overpacks. This scenario resulted in a peak annual dose of 0.12 mSv/year.

Germany – The 2009 Safety Requirements provide the necessary quantitative criteria in the form of risk. For all 'likely' future evolutions of the repository system (likely = cases where supporting quantitative data indicate a likelihood >10%), the total risk should be < 10^{-4} with regard to individuals with a 'lifetime' exposure (lifetime = current expected life expectancy). For all less-likely events, the additional risk to affected individuals shall be <10^{-3}, i.e. for such events a greater risk of radionuclide release and individual exposure is allowed because the events themselves are less likely. Where supporting quantitative data on probability exist, the likelihood attached to such events is >1% (but <10%).

The reference time period, for which evidence of safety (safety case) must be provided, is one million years. Assessments carried out in support of geologic disposal in salt have been generic in nature, as well as being performed prior to the 2009 Safety Requirements. In the early assessment work, a scenario under consideration consisted intrusion of a water or brine from the overlying rock into the residual cavities of a filled repository during the post-operational period. The timing of this scenario was assumed to be variable. It was assumed conservatively that a permeable pathway was formed immediately on closure. Based on a series of qualitative arguments, the scenario finally selected for assessment was intrusion of brine from the overlying rock into the middle area of the repository via the permeable pathway at the beginning of the post-operational period. Relative releases from the total repository were <1% "for most radionuclides".

Japan – The Japanese Nuclear Cycle Development Institute (JNC, later re-formed as JAEA) carried out a preliminary assessment (H12 assessment) of the feasibility of geologic disposal in Japan. In the absence of a specific site, a granitic environment was assumed. Due to the lack of regulations / standards governing performance of geologic repository, JNC took note of regulatory annual dose limits found in national regulations outside Japan. The assessment methodology followed a scenario approach and the main assumption for the analysis of the Reference Case was failure of all waste packages (40,000) at 1,000 years post-closure. The results for the Reference Case indicate that radionuclides can be contained adequately by the EBS and near-field host rock, provided that the groundwater flow rate is reasonably low. The peak annual dose obtained was ~10^{-5} mSv/year, about 4 orders of magnitude lower than annual dose limits outside Japan. This peak dose occurred almost 1 million years after closure of the repository.

Spain - With regard to radioactive waste disposal facilities, CSN established a criterion that stated that the level of individual risk that applies must be lower than 10^{-6} per year, or the risk associated with an annual equivalent dose to individuals in the critical group lower than 0.1 mSv/year. Preliminary non-site specific evaluation exercises were carried out in the late 1990's using generic data compiled during the Site Selection Plan process, one for a granite site and one for a clay site. A more detailed but still non-site specific performance assessment of disposal in granite was completed at the end of 2001, and revised in 2003, and a similar exercise was completed for clay in 2004. These documents are not available in the public domain. The main cause of canister failure and subsequent radionuclide release was assumed to be either penetration of the canister wall due to localized corrosion, or mechanical collapse after the canister wall thickness has been reduced (to 4.25 cm) by general corrosion.

Sweden - The relevant regulator at the time, SSI, issued safety standard regulations in 1998 addressing the final disposal of spent nuclear fuel. These regulations stipulate an annual risk of injury after closure of no more than 10^{-6} for a representative individual in a group that is exposed to the greatest risk. This standard has a risk-based focus on the 1000 years after repository closure, a risk-compliance target out to 100,000 years, and qualitative examination of repository performance terminated at one million years. SKB has carried out periodic safety assessments periodically, approximately every ten years, the most recent assessment being SR-Can in 2006 in which SKB formally compared the performance of a standard "KBS-3 type" repository design using site-specific data from two sites, Laxemar and Forsmark. Impacts were calculated for a range of different scenarios, in particular (a) a buffer-erosion scenario driven by deep circulation of dilute glacial melt waters, and (b) a shearing scenario assuming a limited number of waste packages mechanically disrupted from re-activation of faults due to glacial unloading of the repository rock-block. Both sites were shown to comply with the SSI's 100,000-year risk criterion, and were accordingly judged to be suitable for selection as the repository site.

Switzerland - The key performance metric is that the annual dose to individuals must be less than 0.1 mSv/year. There is no time cutoff in Swiss regulations. Nagra completed the Project Gewähr assessment in 1985 covering disposal of HLW and LILW in crystalline rock, with a disposal concept similar to the current version, and carried out according to HSK's Guideline, which specified that deterministic safety analyses should be carried out and that error limits and uncertainties were to be estimated. Nagra's Base Case assumed that groundwater enters the waste-filled storage caverns and eventually reaches the surface of the waste canisters. The canisters corrode and are assumed to fail simultaneously after 1,000 years. After failure, the waste matrix begins to corrode and radionuclides are released and diffuse through the bentonite buffer to the host rock. The resultant maximum peak dose from the 'realistic' case was 6×10^{-10} mSv/year. Nagra noted that a peak dose close to the regulatory limit of 0.1 mSv/year could be reached only via a combination of many conservative assumptions, which did not correspond to "*realistically conceivable*" that was specified in the HSK Guideline. In the subsequent assessment, Kristallin I, similar criteria were specified, but with an additional risk criterion (1 in a million per year) to be applied to unlikely events. Again, no timeframe cutoff was stipulated. The resultant peak annual individual dose from all pathways for the Reference Case was $< 2 \times 10^{-4}$ mSv/year. Nagra also submitted (2002) a safety assessment (Project Opalinus Clay) for the disposal of used fuel, HLW and LL-ILW in Opalinus Clay. For the Reference Case, Nagra assumed that there are no initially defective canisters and that all canisters are breached after 10,000 years. In this case, the annual dose attributable to used fuel dominated the overall dose, with a maximum annual dose for the Reference Case of 4.8×10^{-5} mSv/year reported.

Taiwan – The Enforcement Rules for the Nuclear Materials and Radioactive Waste Management Act specify an annual effective dose limit of 0.25 mSv/year. No assessment timeframe is specified, although assessment calculations have been carried out to 10^7 years. PA capabilities have been developed within the Taiwanese program. Biosphere results indicate peak doses falling orders of magnitude below the 0.25 mSv/year limit.

United Kingdom – With regard to geologic disposal facilities, the Environment Agencies updated the previous guidance document, which provided a risk guidance level (the same as a risk target) after the period of authorization", *viz.* the assessed radiological risk from a disposal facility to a person representative of those at greatest risk should be consistent with a risk guidance level of 10^{-6} per year (i.e. 1 in a million per year). In addition, there is a requirement on dose constraints for the period of authorization, which covers the operational phase and any period of active institutional control. These limits must not exceed a source-related dose of 0.3 mSv/year or a site-related dose of 0.5 mSv/year. To date, no assessments have been carried out related to geologic disposal of HLW in the UK.

14.8 Independent Peer-Review and Advisory Bodies

Belgium - After submission of the first safety evaluation (SAFIR), the Secretary of Energy established a Safety Commission comprising Belgian and foreign experts to evaluate this report, as well as review ONDRAF/NIRAS' 5-year R&D program plan. An international peer group under the oversight of NEA reviewed the second safety evaluation SAFIR2, in addition to the internal Evaluation Commission. The conclusion was that the selection of the Boom Clay in the Mol-Dessel area is a good one. These assessments are not considered as safety assessments in support of licensing but more a basis for discussions between implementer and regulator.

Canada – AECL's 1994 assessment was reviewed by a special Review Panel of the EIA, which concluded that the disposal concept was technically safe and met current regulatory requirements.

China - Numerous government research institutes and universities participate in R&D activities associated with the long-term program towards an HLW repository. These organizations regularly hold meetings, workshops, and seminars and have close ties with their international counterparts.

Finland - An international peer group reviewed the latest safety assessment for the final disposal of used nuclear fuel according to the KBS3-V disposal concept on behalf of STUK. STUK also had an independent audit of its role in the Finnish repository program conducted through its membership in the European Union. While the government has no specific Advisory Group, it is open to comments received from expert groups within Finland.

France - France has a National Assessment Commission (CNE) with responsibilities for reviewing ANDRA's work and making recommendations to the government. A draft version ("2001 Argile") of the 2005 assessment was provided to this Commission as well as to a group of NEA international experts for peer review. The regulator, ASN, is supported by ISRN (Institute of Radiation Protection and Nuclear Safety) as well as by an advisory group of experts.

Germany – Once selection of the Gorleben site had been announced, a public symposium ("Gorleben Hearing") was held in 1979, to which by a number of international participants were invited. There has not yet been an opportunity for international peer review of any assessment work. A key advisory body to the government is AkEnd, which was commissioned to recommend the selection of possible sites for heat-generating waste as well as developing procedures and criteria for selecting sites. Independent advisory bodies to the regulatory division of BMU include the Reactor Safety Commission (RSK) and the Commission on Radiological Protection (SSK), both of which make recommendations that feed directly into the licensing process.

Japan - According to AEC Guidelines, the H12 study was reviewed independently by a group of international experts coordinated by OECD/NEA, before submission of final reports to the Japanese Government. The H12 assessment was also reviewed during its preparation by external advisors. JNC took into account review comments before finalizing the H12 report and submitting it to the authority in November 1999.

Spain - No information available.

Sweden - SKB's 1997 safety assessment was reviewed by an international group of OECD/NEA experts as well as by the Swedish regulator. In addition, SKB's SR-Can assessment was reviewed by an international safety assessment methodology review team commissioned by the former Swedish regulator SKI, which has now been combined with SSI to form SSM. An advisory body to the government is the Swedish National Council for Radioactive Waste (formerly KASAM), consisting of Swedish experts in relevant technical areas. SSM is planning to have the NEA conduct an independent review of SKB's 2011 license application.

Switzerland - Project Opalinus Clay was reviewed by an International Review Team that considered Nagra's Safety Case to be "*at the forefront of international practice*". With regard to advisory bodies, a number of bodies provide advice to the government on nuclear-related safety matters, including KNS, AGNEB, and KNE.

Taiwan – No information available.

United Kingdom – With regard to advisory bodies, CoRWM, an independent, non-governmental advisory organization, provides recommendations towards government policy regarding radioactive waste management. In its updated role, CoRWM will review the long-term management option for HAW, in particular NDA's and the Government's proposals, plans and programs to develop a geologic disposal facility.

14.9 Stakeholder and Public Involvement

Belgium - Public involvement, in terms of construction of the HADES R&D facility, has been relatively small. According to the current program, public consultations on the long-term management plan for LL-ILW and HLW started in January 2009, allowing opportunities for consultations with the general public. Any license application for the construction of disposal facilities involves the preparation of an EIS, which calls for public input via public hearings.

Canada – After almost two decades of research towards the geologic disposal of used nuclear fuel, the government acknowledged in 1998 that there was insufficient public support to move forward and implement a repository-siting program. Since that time NWMO, the implementing organization, has engaged in widespread dialogue with Canadian stakeholders, taking part in almost three years of discussions with aboriginal peoples, the public, and specialists. The result of these interactions indicated that Canadian public generally agreed that the national waste management focus should be on (i) geologic disposal and interim storage, both (ii) on site and (iii) at a dedicated centralized storage facility. NWMO has since settled on a fourth option, the Adaptive Phased Management approach, which is a combination of the three options originally proposed.

China - The HLW disposal program has not attracted significant public attention given its early stage of development. A 2007 decree "Environmental Information Disclosure Methods" provides a legal basis for public access to documents and involvement in the geologic disposal of HLW.

Finland - The Finnish public are generally well informed about nuclear power and the need for waste management, such that Posiva's proposal to construct a repository on Olkiluoto garnered strong local community support, with the Eurajoki Council, which had the right to veto the decision, voting 20 to 7 in favor of constructing the repository. In addition, the EIA process, a key component licensing process, involves the general public during the public hearings part of the process, giving them the opportunity to comment. In 2000-2001, the Finnish government reviewed STUK's favorable preliminary safety appraisal of the Posiva repository program and the accompanying municipal acceptance, and issued a formal Decision in Principle (DiP) that represented public acceptance and approval to proceed to the planned construction license application in 2012.

France - Public involvement in site investigation and selection activities in France was limited prior to the 1991 Law; thereafter, legislation provided for greater public involvement by requiring the formation of Local Information and Follow-up Committees to promote dialogue among local stakeholders at each candidate site concerning a deep repository. In addition, France's Environmental Code requires a public inquiry associated with an Environmental Impact Assessment.

Germany – The public had minimum involvement in the decision leading to the selection of the Gorleben salt dome as a potential host formation for a geologic repository, until the decision was announced at which time there was fierce opposition to the program. Current legislation now ensures that public involvement in the process is significant, via the EIA process, with those raising objections being given the opportunity to explain their opposition at public hearings.

Japan - Since the 2000 Act, the Basic Policy requires that at every stage of the site selection process, NUMO must solicit the opinions of local residents, and METI must solicit the opinions of governors and mayors, and that these opinions be documented and addressed. To aid the volunteer siting process, NUMO distributed an information package to all (3,239) municipalities in Japan, which contained among other things a description of the benefits to volunteer municipalities, not only from a financial perspective, but also with respect to other positive social aspects. NUMO has followed up this initial contact with an active public communications program including public meetings, round-table talks with local politicians and an interactive website.

Spain - Both the Regulation on Nuclear and Radioactive Facilities and a 2006 Law on the Assessment of the Environmental Effects of Certain Plans and Programs require public involvement in the licensing process, which involves access to information relevant to the preliminary authorization of the facility. The 2006 Law specifically promotes transparency and public participation. Current dialogue with the public is continuing in the context of identifying a site for the centralized storage facility.

Sweden - There has been a long history in Sweden of close interactions among implementer, regulatory authority, local and national government representatives, and the general public. When SKB's studies started in 1994, there were extensive interactions between the stakeholders involved, including an extensive series of public information exchange meetings, some attended by the authorities. All of the municipalities were given the chance to review the feasibility study reports, with the expenses for these reviews being covered by the National Waste Fund (NWF). In order to prepare the EIS for a nuclear waste repository SKB needs to consult with the county administrative board, other authorities, the local municipality and the public, including local non-governmental organizations.

Switzerland - The licensing procedure for the construction of disposal facilities includes public consultation. Documentation on the project, including the safety analysis report, the regulatory review report and the views and opinions of the cantons, is made available, and any person (including those from neighboring countries) can give input or raise objections. KEG specifically requires that the site canton as well as neighboring cantons and countries, are involved in decision-making with regard to the general license.

Taiwan – In general, the general public has negative feelings towards nuclear power and nuclear waste disposal, which have been used in political campaigns in general elections. Thus, the government is cautious in dealing with the public, while regarding public involvement and acceptance as the most important factors for the success of the final disposal program. The Taiwanese program has strived to keep the public well informed by providing online access to regulations and documents including program plans and progress reports.

United Kingdom – One of the stated requirements of the Environment Agencies recent Guidance Document for developing geologic disposal facilities states that the "*developer should engage in dialogue with the planning authority, potential host community, other interested parties and the general public on its developing environmental safety case for a geologic disposal facility*". Since rejection of the implementer's application to construct a URL at Sellafield, the UK underwent a major period of public consultation aimed at obtaining consensus on the way forward.

14.10 Program Maturity

Belgium - Although the country has not committed to this option, the Belgian R&D program on the feasibility of geologic disposal has been active for over 30 years, working with a reference design and repository layout for both HLW and used nuclear fuel. Ongoing work in this area has focused on constructing, and carrying out experiments and tests in the underground facility HADES in clay underneath one of the main nuclear sites in Belgium.

Canada – AECL had worked on assessing the concept for HLW emplacement in a deep repository excavated in the plutonic rock of the Canadian Shield, culminating in the submission of an EIS in 1994. Following public hearings, the government found the concept to be technically safe and in compliance with current regulatory requirements, but it also concluded that there was insufficient public support to move forward and implement a repository siting program. Since then, the new implementer NWMO has carried out an extensive dialogue with stakeholders on a fair process to identify a community willing to host facilities for the management of Canada's used nuclear fuel for the long term. Guided by the public input, NWMO developed a Proposed Process for Selecting a Site for review and comment in 2008.

China - The relatively recent radioactive waste management program is commensurate with the country’s developing commercial nuclear power program. Rapid growth in both the construction and commissioning of commercial nuclear reactors and continued pursuit of an operational geologic repository are expected.

Finland - The Finnish program to identify a site for a geologic repository has developed in a stepwise manner since the first study of potential sites in the early 1980’s, culminating in the identification of a specific site at Olkiluoto. Posiva is now preparing a license application for the repository construction phase at this site.

France - Following abandonment of a disposal program in the late 1980’s, the renewed French pursuit of a geologic repository has proceeded in a stage-wise manner, with careful review by a National Commission consisting of respected scientists and academics. Subject to the favorable outcome of a public debate, ANDRA is close to final site selection, with subsequent submittal of license application for the construction anticipated thereafter.

Germany – After a period of site selection activities leading to the selection the Gorleben salt dome as potential site for heat-generating waste, the repository program in Germany has essentially been on hold for almost 10 years due largely to political reasons.

Japan - Japan has been actively seeking a site for geologic disposal since 1976. While a wealth of information has been collected on different regions throughout Japan, no specific candidate sites have been selected, although considerable research and development is continuing on repository concepts, knowledge management, and site characterization methods at URLs.

Spain - While geological investigations have been carried out in order to identify potential sites for geologic disposal, the current effort is focused on finding a site for the planned centralized storage facility for irradiated fuel. ENRESA's most recent waste management plan envisages site characterization activities for a geologic repository occurring between 2025 and 2040, construction of the repository between 2040 and 2050, and the operational phase beginning around 2050.

Sweden - The Swedish waste management program for used nuclear fuel is at a high level of maturity, with the submission of a license application for the construction of an encapsulation plant as well as repository underground facilities expected in early 2011.

Switzerland - The Swiss geologic disposal program is well-developed and has been successful technically in terms of the development of a disposal concept for both crystalline rock and sedimentary (clay) rock, and submission of safety assessments indicating the feasibility of geologic disposal. Progress has been delayed to some extent by the social and political climate although the current site selection process (sectoral plan) is now proceeding in a stepwise approach involving three stages with broad public consultation at the end of each stage.

Taiwan – Taiwan is at an early stage in the development of a geologic repository for used nuclear fuel, with an operational facility not expected before 2052.

United Kingdom – The UK geologic disposal program is at an early stage, due largely to an extended period of inactivity following major public opposition to geologic investigations in the 1970's and subsequent reluctance by successive governments to address the need for geologic disposal. Only recently, the NDA is more actively pursuing site selection in connection with a geologic disposal facility. In the context of future site selection / assessment, the Government is fully committed to an approach based on "voluntarism and partnership".

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Table 14-1
General Nuclear Fuel Cycle Context for National Geologic Disposal Programs[a]

	Operating Nuclear Fleet for Electric Power Generation[b]			Commercial Reprocessing Infra-structure	Used Fuel Storage[c]				Pu-Recycle in LWRs as MOX	Fuel Cycle Policy[d,e]
					At Reactor		Away from Reactor			
	GWe Net	No.	%		wet	dry	wet	dry		
Belgium	5.9	7	52	no[f]	yes	yes	no	no	yes	no final decision
Canada	13	18	15	no	yes	yes	no	no	no	OTC
China	10	13	2	yes pilot scale with planned expansion to commercial	yes	yes	yes	no[g]	planned	MOFC with plans for fuel cycle closure
Finland	2.7	4	33	no	yes	no	no	no	no	OTC
France	63	58	75	yes commercial scale for domestic and export	yes	no	yes	no	yes	MOFC with plans for fuel cycle closure
Germany	20	17	26	no[h]	yes	yes	no	yes	yes	MOFC shifting to OTC
Japan	47	55	29	yes pilot scale with commercial scale imminent	yes	yes	yes	no[g]	yes	MOFC with plans for fuel cycle closure
Spain	7.4	8	17	no	yes	yes	no	no[g]	no	OTC
Sweden	9.4	10	35	no	yes	no	yes	no	no	OTC
Switzer-land	3.2	5	39	no	yes	yes	no	yes	yes	MOFC with option for direct disposal
Taiwan	4.9	6	21	no	yes	no[g]	no	no	no	OTC
United Kingdom	11	19	18	yes[i] commercial scale for domestic and export	yes	no	yes	no	no[j]	no final decision, OTC for LWRs likely

a. Data on nuclear fleet and capacity are from World Nuclear Association. *World Nuclear Power Reactors & Uranium Requirements*. Updated 1 November 2010. <http://www.world-nuclear.org/info/reactors.html> Accessed 8 November 2010. Information on nuclear fuel cycle infrastructure, approach, and policy are from multiple sources referenced Chapters 2 – 13, including:

-- World Nuclear Association country profile reports available at: *WNA Public Information Service Full List of Pages*. <http://www.world-nuclear.org/infomap.aspx> Accessed 23 November 2010.

-- Organization for Economic Cooperation and Develop, Nuclear Energy Agency (OECD/NEA) member country reports and profiles available at: *Radioactive Waste Management Programmes in OECD/NEA Member Countries*. <http://www.nea.fr/rwm/profiles/> Accessed 23 November 2010.

-- International Atomic Energy Agency (IAEA) member country reports and profiles available at: Net-Enabled Radioactive Waste Management Database (NEWMDB). <http://newmdb.iaea.org> Accessed 23 November 2010.

b. Categories include: installed capacity (GWe Net), number of operating reactors (No.), and percentage contribution to total electric generation (%).

c. Used fuel storage includes storage (1) for the purpose of managing the fuel on an interim or indefinite basis until permanent disposal occurs or a decision is made regarding other fuel cycle options, and (2) for temporary storage until reprocessing occurs. Considers operational facilities only and is subject to change. This category does not include storage of vitrified HLW.

d. Fuel cycle policy reflects interpretation of current national policy and is subject to change. Those nations having direct disposal as an option will also have irradiated uranium fuel for disposal – predominately UOX from LWRs. Nations with planned, current, or historical Pu-recycling programs will have commercial HLW from reprocessing as a dominate waste form for disposal in a repository and will also have an inventory of used MOX fuel for disposal if not recycled further in LWRs or fast reactors.

e. Fuel cycle acronyms are defined as follows: OTC = once through (open) fuel cycle; MOFC = modified-open fuel cycle, which includes single Pu-recycle in LWRs as MOX and the options of multi Pu-recycle and use of reprocessed uranium (RepU). The OTC implies direct disposal of used UOX fuel; the MOFC would involve disposal of HLW from reprocessing as well as the disposal of used MOX fuel if further reprocessing and/or fuel cycle closure does not occur.

f. Belgium pioneered reprocessing and MOX fuel fabrication technology and operated facilities at an industrial scale; these facilities have been shut down for decommissioning.

g. Planned or under construction.

h. Germany operated pilot scale reprocessing facility for two decades prior to shut down in 1991.

i. The UK context is particularly complex. The UK originally developed an industrial scale reprocessing infrastructure to support the management of irradiated metallic fuel from its Magnox reactor fleet beginning in 1964. The UK's reprocessing capacity was extended to domestic and foreign oxide fuel with the startup of the Thermal Oxide Reprocessing Plant (Thorp) in 1994; however, chronic operational and maintenance problems have severely restricted production far below design capacity.

j. The UK to date has chosen not to recycle Pu domestically as MOX fuel for use in its power reactor fleet.

Table 14-2
Key attributes of national programs with respect to implementer and funding.

Country	Nature of Implementer	Funding for Program Implementation
Belgium	government entity (ONDRAF/NIRAS) under ministry (Energy)	utilities pay into dedicated waste fund administered by private entity (Synatom)
Canada	government entity (NWMO) under ministry (NRCan)	dedicated national waste fund administered by implementer (NWMO) and funded by waste producers
China	State-owned utility	government plan calls for establishment of waste fund
Finland	private utility consortium (Posiva)	licensees of nuclear facilities pay into independent waste management fund administered by government entity (MEE); annual fees established by government
France	independent government entity (ANDRA) supervised by ministries (Industry, Research and the Environment)	principal waste generators pay into a fund internal to each entity
Germany	commercial entity (DBE) operating under contract with government entities (BfS under BMU)	utilities as principal waste producers set aside and manage internal funds according legislative guidelines
Japan	government entity (NUMO) supervised by ministry (METI)	dedicated national waste fund managed by government entity (RWMC); utilities pay into fund based on power generated
Spain	government entity (ENRESA) reporting to ministry (Energy)	utilities pay into interest-bearing fund via levy on electricity sales during operational life of reactors
Sweden	private utility consortium (SKB)	utilities pay fee (per kWh) into dedicated national waste fund overseen by government appointed Board of Governors
Switzerland	government and private utility consortium (Nagra)	dedicated national waste fund paid for by levy per kWh
Taiwan	Taipower, state-owned utility reporting to ministry (Economic Affairs)	2% of fuel cycle backend fund allocated for waste disposal activities
United Kingdom	non-departmental government entity (NDA) reports to department (Energy and Climate Change)	Funding for NDA and disposal activities paid for by government and commercial activities; waste producers bear costs of waste management and disposal

Table 14-3
General site selection and repository design considerations.

Country	Geographical Size	Geological Diversity	Primary Candidate Geologies	Hydrology and Geochemistry	Primary Repository Design Concept
Belgium	small (10^4 km^2)	Limited	Clay	Saturated, reducing	Used fuel and HLW waste forms considered; stainless steel overpack and sleeve; clay+sand+graphite backfill; concrete lining; clay formation
Canada	continental (10^7 km^2)	High	Crystalline, sedimentary	Saturated, reducing	Generic concept – single layer, in-room emplacement (variants considered); used fuel waste form; carbon steel inner liner; copper overpack (previously titanium); compacted bentonite blocks; cement or clay backfill
China	continental (10^7 km^2)	High	Crystalline	Saturated, reducing	Vitrified waste form mainly; small amount of used fuel; bentonite buffer; no information on canister
Finland	regional (10^5 km^2)	Limited	Crystalline	Saturated, reducing	Used fuel waste form; KBS-3 concept (see Sweden), vertical emplacement
France	regional (10^6 km^2)	Moderate	Clay (crystalline site rejected)	Saturated, reducing	Co-disposal of HLW and used fuel with ILW-LLW in separate sections of repository; primarily vitrified waste form with some used fuel; steel sleeve for tunnels; P235 steel overpack; no buffer (unless fracture zone indicated); individual disposal cells sealed with clay
Germany	regional (10^5 km^2)	Limited	Salt, sedimentary	Saturated, reducing	Generic design; reliance on host rock for performance; thick-walled overpacks for HLW / used fuel; crushed salt for backfill
Japan	regional/island (10^5 km^2)	Limited	Crystalline	Saturated, reducing	Generic concept similar to Switzerland
Spain	regional (10^5 km^2)	Moderate	Crystalline,	Saturated, reducing	Generic reference design; used fuel in carbon steel overpacks; compacted bentonite buffer blocks; concrete and bentonite sealing plugs
Sweden	regional (10^5 km^2)	Limited	Crystalline	Saturated, reducing	KBS-3 (used fuel waste form; copper / steel canisters; vertical emplacement; compacted bentonite buffer; sand-bentonite backfill
Switzer-land	small (10^4 km^2)	Limited	Crystalline, clay	Saturated, reducing	Mainly HLW, but used fuel also considered; thick-walled carbon steel overpack; compacted bentonite buffer
Taiwan	small/island (10^4 km^2)	Limited	Crystalline, mudstone	Saturated, reducing	General KBS-3 disposal concept (see Sweden); vertical or horizontal emplacement configuration yet to be decided
United Kingdom	regional/island (10^5 km^2)	Moderate	Crystalline, sedimentary, mudstone	Saturated, reducing	Generic disposal concepts compiled for review (see Appendix B)

Table 14-4
Key aspects of repository program for assessing performance and regulatory compliance.

Country	Timeframe	Approach and Metrics	Demonstration[a]
Belgium	None specified; published results of assessments extend beyond 10^6 years	Annual dose limit: 1 mSv/yr	Non-formal assessments used as basis for discussions with regulators
Canada	Previously (R-104) 10,000 years; reasoned arguments required where peak risks occur >10,000 years Currently expected to include period of maximum impact	Previous - annual risk: 10^{-6} /yr corresponding to ~ 0.05 mSv/yr) Current - acceptance criterion less than regulatory dose limit: dose constraint of 0.3 mSv/yr	PA / EIS
China	No information available	No information available	No formal PA carried out
Finland	0 – 10,000 yrs 10,000 – 100,000’s yrs 100,000’s – 10^6 yrs	Dose constraint: 0.1 mSv/yr Radionuclide release activities Qualitative considerations	PA No biosphere analyses Bounding analyses and natural analogs
France	0 - 10,000 yrs 10,000 - 10^6 yrs	Annual dose limit: 0.25 mSv/yr Reference dose: 0.25 mSv/yr	PA
Germany	0 – 10^6 yrs	Likely (prob. >0.1) scenario total risk: $<10^{-4}$ Less-likely (0.01 < prob. <0.1) scenario total risk: $<10^{-3}$	Generic PA
Japan	No timeframe specified; Published assessment results extend beyond 10^6 years	Expected to track annual dose limits adopted in other countries, i.e., ~ 0.1 -0.3 mSv/yr	PA
Spain	No timeframe specified	Individual risk: $<10^{-6}$ per year (or <0.1 mSv/yr risk equivalent in critical group)	Preliminary non-specific assessments performed for granite and clay sites
Sweden	100,000 years for risk criterion; Published assessments carried out to 10^6 yrs	Annual risk to representative individual of most exposed group: $<10^{-6}$ /yr	Periodic PA (conducted ~ every 10 years)
Switzerland	No timeframe specified	Annual dose limit for likely events: 0.1 mSv/yr Annual risk limit for less likely events: 10^{-6} /yr	PA (1985), PA / Safety case (2002)
Taiwan	No timeframe specified; published assessment results beyond 10^6 yrs	Annual dose limit: 0.25 mSv/yr	PA
United Kingdom	No timeframe specified; increase in uncertainty with time acknowledged by regulator	Annual risk guidance level or risk target: 10^{-6} /yr	PA / Safety case

a. Acronyms defined as follows: PA = performance assessment; EIS = environmental impact statement or assessment.

A

NOMINATIVE SITING APPROACH: A CASE STUDY FROM THE SWISS NATIONAL REPOSITORY PROGRAM (NAGRA)

A.1 Introduction: General Context of Swiss National Program for Geological Disposal of Nuclear Wastes

According to the Swiss Nuclear Energy Act (KEG, 2003) and the corresponding Nuclear Energy Ordinance (KEV, 2004), which came into force in February 2005, "All radioactive waste generated in Switzerland is to undergo final disposal in repositories situated in suitable geological formations - surface and near-surface disposal is not permitted". Two geological repositories are foreseen, one for L/ILW[41] and the other for HLW[42] (including irradiated nuclear fuel if not reprocessed) and LL-ILW[43].

The nuclear power plant operators and the Swiss federal government founded the National Co-operative for the Disposal of Radioactive Waste (Nagra), which is responsible for the disposal of all kinds of radioactive waste, including irradiated nuclear fuel if declared as waste, with a view to implementing its permanent and safe disposal. It is important to stress that Nagra, the Swiss implementer, is not part of the Swiss Federal Government and its legal form is a "Cooperative". This is fundamentally different the past US approach in which the implementer for geological disposal was the US Department of Energy (USDOE), a formal part of the US government with all of the implications and restrictions arising from being a government agency.

The Swiss Nuclear Energy Act establishes the need for a series of licenses regarding nuclear materials, radioactive waste and nuclear facilities (general, construction, operation, closure). The last three are granted by DETEC (see Section A.6 for terminology); they are based on scientific and technical criteria, which are evaluated by ENSI. The general license is the decision prior to the realization of a nuclear facility, granted by the Swiss federal government. The general license is approved by the Parliament. The Nuclear Energy Act (KEG, 2003) specifies that this approval is subject to a facultative referendum.

[41] L/ILW: low and intermediate level waste

[42] HLW: high-level waste (vitrified)

[43] LL-ILW: long-lived intermediate level waste

Decisions by the federal government, cantonal government (canton is equivalent to a 'State' in the US), and a community government are normally subjected to a facultative referendum, unless otherwise explicitly specified in the law. The facultative referendum at each level is initiated by a specified minimum number of signatures and the result is binding (i.e., not consultative as in other countries). Federal law has precedent over cantonal law, and cantonal law over community law. According to the Nuclear Energy Act (KEG, 2003) the general license granted by the government and approved by the Parliament (equivalent to Senate and Congress in the USA) can only be subjected to a national referendum.

For both L/ILW and HLW repositories, the Nuclear Energy Ordinance (KEV, 2004) states that the site selection procedure must be defined in a so-called "Sectoral Plan" within the framework of the existing legislation on spatial planning. In the past both 'voluntary' and 'nominative' approaches have been followed for site selection for geological disposal. In contrast to the recent trend toward the 'voluntary' approach in several countries, Switzerland has adopted a 'nominative' site selection approach that was successfully initiated in 2008. Lessons learned from the application of this approach and the particular characteristics that differentiate it from a 'traditional' nominative approach are highlighted.

A.2 The Swiss Sectoral Plan for Deep Geological Repositories

A.2.1 What is a Sectoral Plan?

The sectoral plan is a tool used by the Swiss Federal Government in the area of spatial planning. No fewer than 20 Swiss federal agencies have remits which touch upon spatial planning in the widest possible variety of areas: these include agriculture, transportation, regional policy, energy, security strategy and public buildings and installations. The Federal Office for Spatial Development (FOSD) has, since 1980, regularly produced a publication entitled "Overview of federal activities affecting spatial planning". This describes the tasks affecting spatial planning that the Federal Government performs independently or in conjunction with the cantons and presents the relevant legal bases, concepts and sectoral plans. It also outlines specific projects of the Federal Government, as well as a range of studies on the subject of spatial planning. The publication provides all the information necessary for coordinating tasks between the Federal Government and the cantons and also contains a number of themed maps illustrating different spatial planning-related activities.

A sectoral plan generally consists of two parts: in the strategy (conceptual) part, the basic principles, boundary conditions and objectives are set out, while the topical (implementation) part contains boundary conditions, procedures and specifications for the individual projects. The Federal Government prepares such sectoral plans in order to plan and coordinate its activities in so far as they significantly impact living space and the environment. Sectoral plans coordinate tasks with spatial character and integrate them into the overall context of territorial development. Already effective examples are sectoral plans for roads, rail/public transport, aviation, overhead transmission lines, etc.

A.2.2 Sectoral Plan for Deep Geological Repositories in Switzerland

The strategy (conceptual) part of the sectoral plan defines the siting procedure based on 3 stages (Figure A-1). Highest priority is given to safety-technical criteria: the long-term protection of man and the environment. The results of the individual stages are documented in a results report and so-called “object sheets”. These include maps and text and show the extent of the geological siting area, the planning perimeter and, in stages 2 and 3, the sites. They also contain the results of the assessment of safety and feasibility and the evaluation of spatial and environmental aspects. They provide guidelines for implementation in the subsequent stage and for the approval of the general licence. After each stage, the object sheets and the findings in the results reports are approved by the Federal Council and thus become part of the sectoral plan.

In Stage 1, on behalf of the waste producers Nagra should propose potential sites based on the safety and engineering feasibility criteria stated in the sectoral plan. The SFOE informs the affected Cantons, communities and neighbouring states and sets up the Cantonal Commission (see also Section A.6 for description of the various entities). The general feasibility of these sites is evaluated under the auspices of the Federal Office for Spatial Development (FOSD) in terms of spatial planning perspectives. The safety and engineering feasibility aspects are evaluated by safety authorities. Given a positive result, the SFOE then prepares the so-called “object-sheets” for the potential siting regions and submits these to the Federal Council for approval.

Stage 2 is dedicated to the assessment of spatial planning and socio-economic aspects and the performance of provisional safety analysis for sites within the approved siting regions. The affected regions have the opportunity to participate in the process of narrowing down and identifying the potential sites and in locating the surface installations. Regional participation groups are set up for this purpose. On the basis of a comprehensive assessment of the safety and engineering aspects, considering the results of the provisional analysis and including the potential locations for the surface facilities, the waste producers must propose at least two potential sites for each repository, i.e. one repository for L/ILW and one for HLW. Economic analyses are conducted and the economic impact of the repository on regional development is studied. Following review by the authorities, the “object-sheets” are updated by the SFOE and, with the approval of DETEC, the identified sites are incorporated into the sectoral plan (interim result).

In Stage 3, the waste producers propose one site each for L/ILW and for HLW (or a single site for all waste categories). With the involvement of the affected regions, the projects are further developed, detailed investigations of the socio-economic effects are made and support measures defined. The geological information for each potential site is brought to a level where sites can be compared from a safety perspective. Stage 3 ends with the specification of a site in the sectoral plan and the granting of the general license by the Federal Council, which then must be approved by Parliament and is subject to an optional binding national referendum.

The Sectoral Plan also specifies a schedule as shown in Figure A-2.

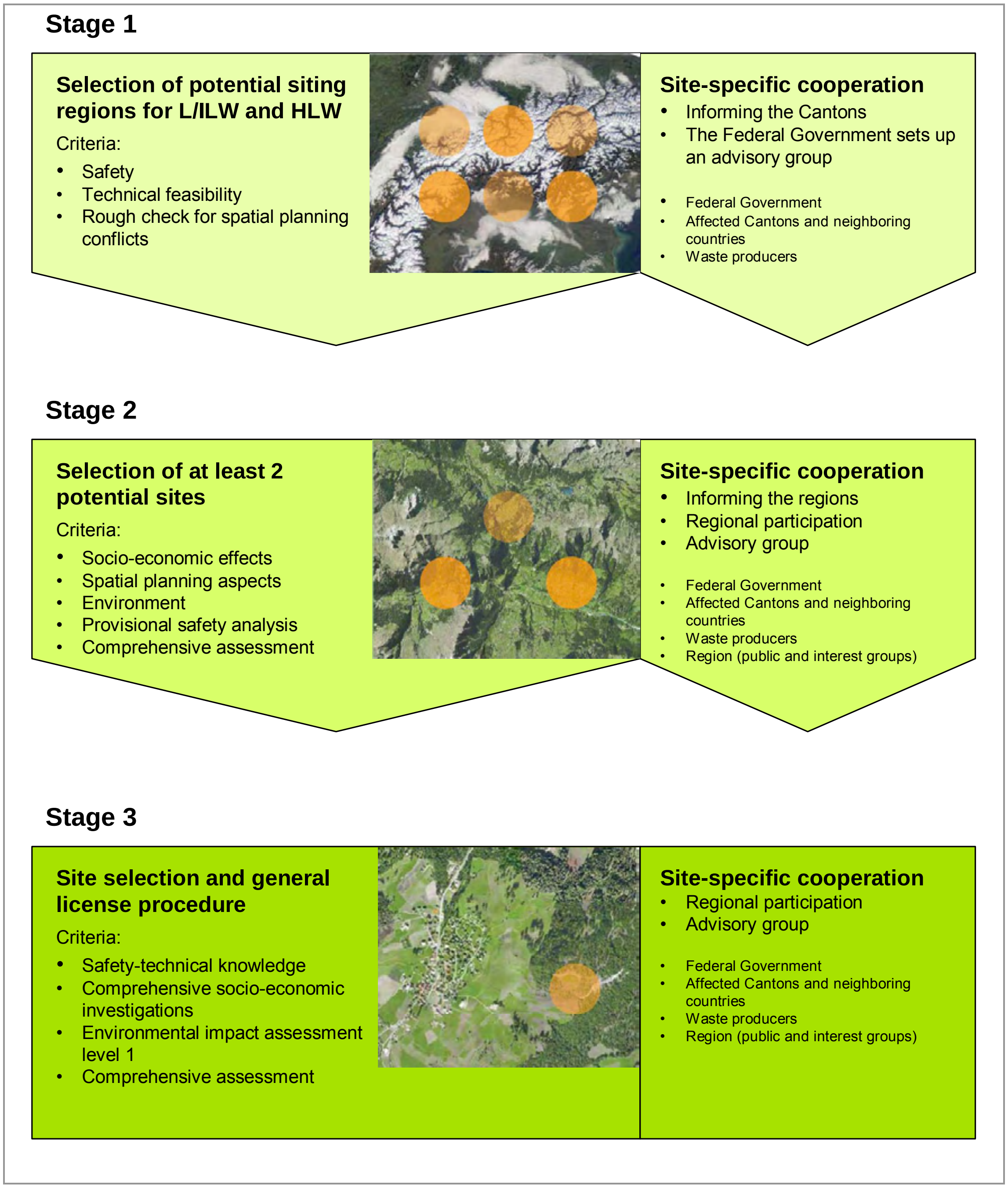

Figure A-1
Overview of main tasks in each of the three stages of the Sectoral Plan (SFOE, 2008). Copyright Swiss Federal Office of Energy. Used with permission.

Timeframe

		Date	
Sectoral plan for deep geological repositories Preparation of conceptual part	Approval by Federal Council	2008	
Sectoral plan for deep geological repositories Implementation	**Procedure according to Spatial Planning Act and Ordinance**		**Procedure according to Nuclear Energy Act**
Stage 1: Selection of geological siting areas (2.5 years)	• Cooperation • Hearings and participation • Settlement • Decision on object sheets		
Stage 2: Selection of at least two sites (2.5 years)	• Cooperation • Hearings and participation • Settlement • Decision on object sheets		
Stage 3: Site selection and general licence procedure (2.5-4.5 years)	• Cooperation • Hearings and participation • Settlement		• Preparation and submission of general licence application • Review and approval procedure
Decision of Federal Council (1.5 years)	Approval of object sheets	By 2016/18[1]	Granting of general licence
		By 2017/19	Approval of general licence by Parliament (1 year) • Possible national referendum
		By 2019/23	Approval of geological investigations, construction licence for rock lab at site (2-4 years) • Licence can be opposed before Federal Administrative Court and Federal Supreme Court
		L/ILW by 2025/31 HLW by 2035/41	Supplementary investigations, construction of access tunnel including underground exploration, construction and operation of rock lab at site and construction licence procedure for repository (L/ILW 6-8 years, HLW 16-18 years) • Construction licence can be opposed before Federal Administrative Court and Federal Supreme Court
		L/ILW by 2030/38 HLW by 2040/48	Construction of disposal tunnels/caverns and operating licence (5-7 years) • The operating licence is prepared and granted during the construction phase; licence can be opposed before Federal Administrative Court and Federal Supreme Court
		L/ILW from 2030 HLW from 2040	Earliest start of operation • Additional criteria are taken into account for the start of emplacement, particularly the temperature of the fuel elements to be emplaced in the HLW repository

Figure A-2
Schedule [Timeplan 2008 – 2035/45] specified in the Sectoral Plan (Status: 2 April 2008). [1]Deadline depends mostly on whether, e.g., additional test boreholes are necessary. [2]Also applies for the operating license. (SFOE, 2008) Copyright Swiss Federal Office of Energy. Used with permission.

A.2.3 Advisory and Coordinating Bodies Established under the Sectoral Plan

The Swiss site selection process involves a large number of actors. In order to achieve optimum coordination, the Federal Government has cooperated with the Cantons in which siting is being considered to set up several bodies.

National Steering Committee

First, it is recognized that the Federal Government leads the site selection process. Overall management lies with the Steering Committee, consisting of the general secretary of the responsible Department (DETEC), the director of the SFOE, the director of the FOSD and a representative of ENSI. The Steering Committee monitors and guides the Sectoral Plan process in terms of top-level coordination between the Federal Government and the Cantons and in terms of timetable. It supports the SFOE in cases of conflict and in planning communication.

Operational responsibility lies with the SFOE, which is supported by the Project Management, consisting of representatives of the SFOE, FOSD and ENSI. They plan and coordinate the procedural steps and ensure collaboration among the involved federal authorities. Other responsibilities relate to quality control, reporting and risk management.

Cooperation with Nagra takes place as part of the project management meetings between the Federal Government and Nagra and in the various bodies in which Nagra is represented.

Waste Management Advisory Council

The Waste Management Advisory Council (Beirat Entsorgung) was established by DETEC. Thanks to its independent nature and function as a national advisory board, it is able to offer an outside, objective perspective. It promotes dialogue among all involved players and helps to identify process risks and barriers. It advises DETEC on the implementation of the site selection procedure for deep geological repositories.

Cantonal Commission

DETEC and the SFOE have established a Cantonal Commission (Ausschuss der Kantone) for consultation with the involved Cantons. The Commission comprises representatives from the siting Cantons, neighbouring Cantons and neighbouring countries. It is responsible for ensuring the required cooperation between official representatives from the siting Cantons, neighbouring Cantons and neighbouring countries, supports the Federal Government in the implementation of the selection procedure and makes recommendations for the attention of the federal authorities.

Working Group on Information and Communication

The information and communication activities to be carried out during the three stages of site selection and the subsequent decisions and justifications have to be comprehensible and transparent. Providing the general public with clearly understandable information via a variety of channels (special events with question-and-answer sessions, lectures, brochures, internet, etc.) requires coordination among the involved authorities at federal and cantonal level and Nagra. It is the responsibility of the Working Group on Information and Communication (Arbeitsgruppe Information und Kommunikation) to coordinate both the activities and the information content and presentation.

Working Group on Spatial Planning

In Stage 2 of the siting procedure, a spatial planning assessment has to be carried out for the siting regions proposed in Stage 1. The spatial planning assessment and the comparison of the sites should be underway by 2011; this is done together with the siting Cantons. The Working Group on Spatial Planning (Arbeitsgruppe Raumplanung) was established to deal with these and related issues and comprises representatives from the Federal Government and the responsible authorities of the involved Cantons.

Technical Forum on Safety

The Technical Forum on Safety (Technisches Forum Sicherheit) discusses and answers technical and scientific questions posed by the public, communities, siting regions, organisations, Cantons and authorities in neighbouring countries.

Technical Coordination Group of Siting Cantons

Five of the nine members of the Technical Coordination Group of Siting Cantons (Fachkoordination Standortkantone) are also members of other bodies introduced in this section.

Cantonal Working Group on Safety

The Cantonal Working Group on Safety (Arbeitsgruppe Sicherheit Kantone) is made up of delegates from the cantons of Zürich, Aargau, Basel-Land, Nidwalden, Obwalden, Schaffhausen, Solothurn and Thurgau.

Cantonal Expert Group on Safety

The Cantonal Expert Group on Safety (Kantonale Expertengruppe Sicherheit) consists of three geologists from the private sector and academia.

Committee of Government Representatives

Like the Working Group on Information and Communication, the Committee of Government Representatives (Ausschuss der Regierungsvertretenden) was originally set up in 2004 for the Opalinus Clay Project. The members are government representatives from Cantons covering the siting regions for a repository for high-level radioactive waste (Zürich, Aargau, Thurgau, Schaffhausen) and representatives from the German State of Baden-Württemberg and the

German administrative district of Waldshut. Additional committee members are representatives of the SFOE and ENSI.

A.2.4 Procedure and Criteria for Switzerland's Sectoral Plan for Deep Geological Repositories

Different requirements apply to the engineered and natural barriers depending on the category of waste for disposal. The current waste management concept foresees two repositories, one for HLW and one for L/ILW. The alpha-toxic waste (ATW) could be assigned to one or the other of the repositories or divided between them. Some of the L/ILW could also be allocated to the HLW repository. If a site fulfils the requirements for both a HLW and a L/ILW repository, the selection procedure could culminate in one site for all types of waste (shared surface facilities while the repositories are located at different depths). When selecting potential siting regions, however, it has to be clear from the beginning, in general, which waste categories and types of waste are foreseen for disposal at a particular site.

Various aspects have to be considered when identifying potential siting regions and repository sites. Firstly, the large-scale criteria that are essential for ensuring long-term safety have to be applied. Criteria that are relevant on a smaller scale then have to be taken into consideration. The selection procedure must clarify the following:

- What requirements apply to site-specific geological conditions, taking into account the allocated waste inventory and the associated safety and barrier concepts?
- Where are there suitable large-scale geological-tectonic areas that fulfil the safety requirements?
- Which rock formations in these areas are potentially suitable as host rocks and as effective containment zones?
- Where are there potential host rocks in suitable configurations (composition, depth, thickness, accessibility)?

The safety and technical (engineering feasibility) criteria are shown in Figure A-3. Site selection is made based on the properties of the underground environment and the overall geological situation as understood on the basis of expert geological knowledge combined with general information and investigation results. The procedure should take into account the fact that a suitable site will not be identified as such on the basis of one single feature. The characteristics that are required to be evaluated are generally interdependent in terms of their safety-related effects. They are also dependent on the waste inventory and the design of the engineered barrier system. An appropriate narrowing-down procedure is developed (see also Figure A-4) to prevent the situation where a suitable site is ruled out as a result of applying overly strict requirements with respect to a single site characteristic (for example, when applying individual quantitative criteria).

Criteria group	Criteria
1. Properties of the host rock and the effective containment zone	1.1 Spatial extent 1.2 Hydraulic barrier effect 1.3 Geochemical conditions 1.4 Release pathways
2. Long-term stability	2.1 Stability of the site and rock properties 2.2 Erosion 2.3 Repository-induced influences 2.4 Conflicts of use
3. Reliability of geological findings	3.1 Ease of characterisation of the rock 3.2 Explorability of spatial conditions 3.3 Predictability of long-term changes
4. Engineering suitability	4.1 Rock mechanical properties and conditions 4.2 Underground access and drainage

Figure A-3
Criteria groups and criteria for the site evaluation form the viewpoint of safety and technical feasibility (SFOE, 2008). Copyright Swiss Federal Office of Energy.

Stage 1: Selection of Siting Regions

The waste producers must describe and evaluate the siting regions proposed in terms of safety and technical feasibility, taking into account the expected waste inventory and the provisional design of the engineered barriers. The five steps to be followed are shown in Figure A-4.

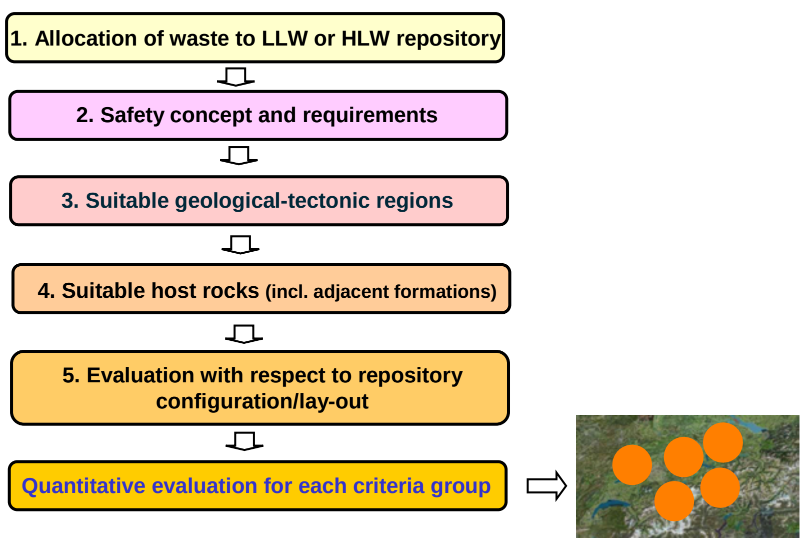

Figure A-4
Methodology for deriving the siting regions (Nagra). Used with permission.

Reviewing safety and engineering feasibility

When evaluating the proposed siting regions, the authorities have to consider the following questions:

- Are the requirements applying to the host rock and the site, as derived by the waste producers, transparent and sufficient?
- Have the producers taken into account all relevant geological information and is this sufficient for the purposes of preliminary orientation?
- Have the producers adequately taken into account the pre-defined criteria for preparing proposals for potential sites?
- Is the procedure followed by the producers in preparing proposals for potential sites transparent and reproducible?
- Can the authorities approve the proposals from the viewpoint of safety and feasibility?

Stage 2: Selection of at least two sites

In Stage 2, from the siting regions proposed and approved in Stage 1, at least two are proposed for integration into the sectoral plan as an interim result. This proposal is made in two steps:

- ***Step 1: Identification of sites in selected regions.*** The waste producers first identify potential sites within the regions. In cooperation with the affected Cantons and regions, proposals are prepared for the arrangement and layout of surface facilities and for the underground disposal areas. This leads to at least one site per region. For all sites identified in this way, safety evaluations are then carried out that comprise a qualitative evaluation using safety criteria and a quantitative analysis of the safety functions. The latter provides a first quantitative estimate of containment of the waste and potential release of radionuclides.
- ***Step 2: Selection of at least two sites.*** The results of the safety evaluations – together with the results of the assessment of other aspects according to the strategic part of the sectoral plan – lead to selection of at least two sites. When preparing the proposals, the waste producers have to observe the following rules:
  - No site can be proposed as an interim result if it has been assessed as being clearly less favorable than the others in terms of safety criteria and the quantitative evaluation.
  - Socio-economic aspects can only be decisive for the selection in the case of sites that are comparable from the viewpoint of safety.

The waste producers then carry out a provisional safety analysis for the proposed sites (see G03/e).

Stage 3: Site selection and general license procedure

In this stage, the waste producers, with the involvement of the affected regions, select the site at which the repository will be constructed and prepare a general license application. This is done in further two steps:

- ***Step 1: Selection of the site.*** From the sites that have been integrated into the sectoral plan as an interim result, the waste producers select the site for repository construction, based on a safety-oriented comparison. To justify this selection, the level of knowledge of the different sites has to be comparable with verified site-specific information; it is thus foreseen that additional geological investigations will have to be carried out, as necessary. The results – together with the evaluation of further aspects in accordance with the strategic part of the sectoral plan – lead to an overall evaluation of the site selection by the waste producers.
- ***Step 2: Preparing and submitting a general license application.*** The waste producers have to prepare the necessary data and reports for the general license application. The suitability of the site must be confirmed by geological investigations. For this purpose, additional investigations may be necessary, in so far as they have not already been carried out when selecting the site. Besides a safety and security report, the documentation submitted has to include an environmental impact report and a report on the situation with respect to spatial planning.

A.3 Switzerland's Sectoral Plan – Status of Site Selection May 2010

A.3.1 Nagra's Proposals for Siting Regions

On 6 November 2008, at a media conference organised by the Swiss Federal Office of Energy, Nagra's siting proposals for the L/ILW and HLW repositories were announced.

In accordance with the requirements set out in the sectoral plan, Nagra's selection is based exclusively on criteria relating to safety and technical feasibility. The 13 criteria shown in Figure A-3 were further expressed in a total of 49 indicators.

Criteria → Indicators → Requirements

Nagra proposed to apply the criteria through the definition of corresponding indicators in order to increase the transparency of the selection process. The indicators were chosen so that clear requirements could also be defined for each of them. For most of the indicators these requirements were quantitative. Two examples to illustrate this approach are shown below.

Example - Criterion 1.1: Spatial extent for the HLW-repository

Spatial extent belongs to the group of criteria associated with the host rock (Figure A-3). The following four indicators and respective requirements were proposed by Nagra for this criterion applied in the context of siting for a HLW-repository:

- Thickness of the host rock; requirement: larger than 100 m (> 100 m)
- Lateral extent of the host rock; requirement: larger than 4 sq. km (> 4 km2)
- Depth-max (for construction); requirement: 900 m below surface (<900 mbs)
- Depth-min (erosion and decompaction); requirement: 400 m below surface (<400 mbs)

Example - Criterion 3.2: Spatial explorability

Spatial explorability is an example of indicators that are evaluated qualitatively. The following four indicators were defined:

- Regional structural model and spatial conditions for the repository
- Continuity of formations of interest
- Explorability from underground
- Explorability from the surface

The requirements for indicator 1 above were the potential to locate and explore with a high degree of reliability zones in which emplacement caverns could be located. For a given geologic-tectonic unit the following evaluation scale was then defined:

- very favourable: many zones, including also large ones are expected
- favourable: if several zones are expected
- unfavourable to less favourable: if only a few zones are expected

Similar requirements and scales were defined for the other indicators.

Nagra’s report NTB 08-03 documents, evaluates and justifies Nagra's proposals in detail; it follows the procedure suggested in Figure A-4. The siting regions are shown in Figure A-5 and they are as follows:

For low- and intermediate-level waste (6 siting regions):

- Südranden (Canton Schaffhausen)
- Zürcher Weinland (Cantons Zürich and Thurgau)
- North of Lägeren (Cantons Zürich and Aargau)
- Bözberg (Canton Aargau)
- Jura-Südfuss (Cantons Solothurn and Aargau)
- Wellenberg (Cantons Nidwalden and Obwalden)

For high-level waste (three siting regions):

- Zürcher Weinland (Cantons Zürich and Thurgau)
- North of Lägeren (Cantons Zürich and Aargau)
- Bözberg (Canton Aargau)

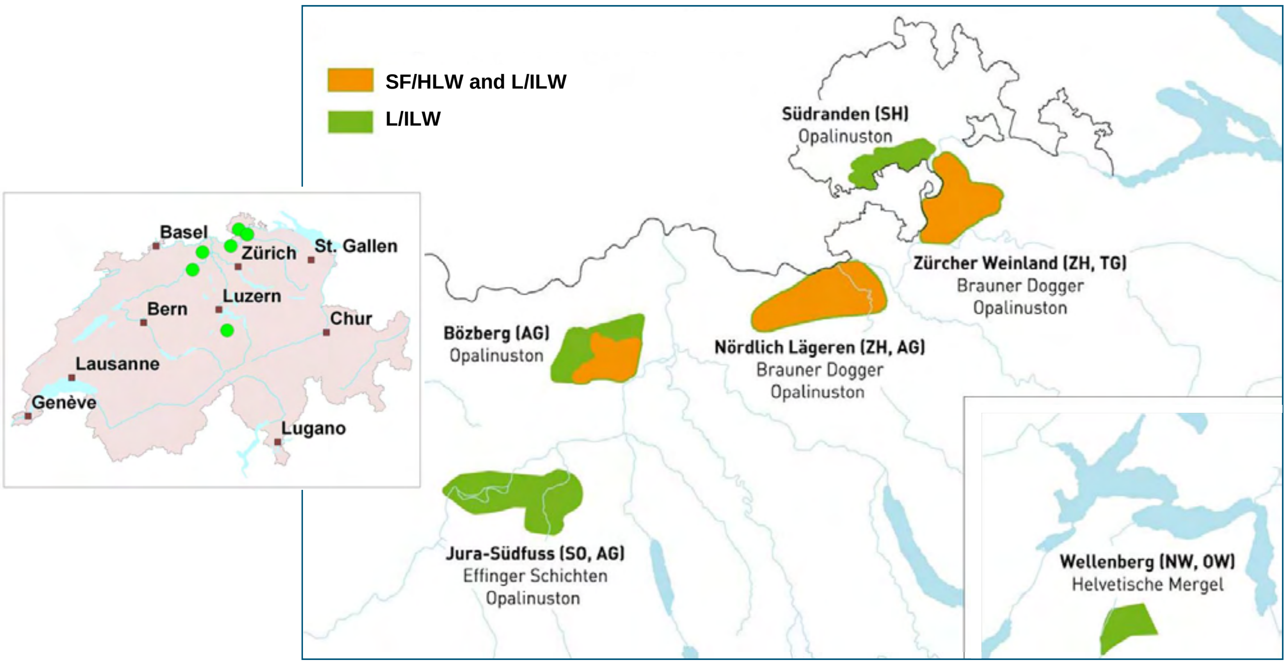

Figure A-5
Proposed siting regions and the proposed host rocks (Nagra, 2008). Used with permission.

The three siting regions Zürcher Weinland, North of Lägeren and Bözberg could also be considered for a combined repository (for all waste categories).

Nagra's selection does not represent a decision on one or more sites, but forms the basis for further reviews and investigations, on which the cantons, local communities, neighbouring countries and the federal authorities can comment.

A.3.2 Review by the Nuclear Safety Inspectorate (ENSI)

The review results from the authorities were presented by the Swiss Federal Office of Energy (SFOE) and ENSI at a press conference on 26 February, 2010. The main review was performed by ENSI (ENSI 2010) but it also included input from:

- KNE (Commission for Nuclear Disposal);
- Swisstopo (Swiss Federal Office of Topography, National Geological Survey); and
- Experts in several specialised areas.

In parallel the expert opinion of the German “Expert Group on Swiss Geological Repositories (ESchT)” has been published. The expert opinion of KNS on the ENSI review was published on 6 May 2010.

The evaluation of the siting criteria from Nagra, ENSI and KNS is shown in Figure A-6, for the HLW/SF type repository. For the overwhelming majority of the criteria, the evaluation by ENSI and KNE is exactly the same. Where there are deviations in evaluation, these are always within adjacent pair of criteria (very favourable → favourable; favourable → less favourable).

The reservations expressed in the evaluation of ENSI (shown below) will be addressed by Nagra explicitly in Stage 2:

- Technical feasibility: requirements for tunnel construction at depths > 600 m in clay;
- Degree of characterisation of host rocks other than Opalinus Clay; and
- Siting in alpine regions.

Summarising their evaluation, ENSI answered the questions defined in the Sectoral Plan as follows:

- **Allocation of the waste**
  The allocation of the waste to both repository types (L/ILW and HLW) was considered to be appropriate.
- **Requirements on geology**
  The requirements relating to geotectonic conditions, host rock and sites were confirmed as being plausible.
- **Geological information**
  The relevant information for Stage 1 of the site selection process was comprehensively taken into account.
- **All relevant criteria taken into account**
  All safety-relevant criteria listed in the Sectoral Plan were taken into account and correctly applied.
- **Transparency and traceability**
  The preparation of the proposals for geological siting regions as specified in the Sectoral Plan was found to be transparent and traceable.

In conclusion, ENSI confirms Nagra's analysis of the geological information to be technically justified, comprehensive and transparent and approves all six proposed siting regions. In May 2010, the Swiss Nuclear Safety Commission (KNS) also published its position on ENSI’s review. As informed in the press release of May 6, 2010, KNS concluded that ENSI has followed Nagra’s process for the siting region selection in detail and has evaluated comprehensively the proposed siting regions. KNS agreed with ENSI’s approval of the proposed siting regions.

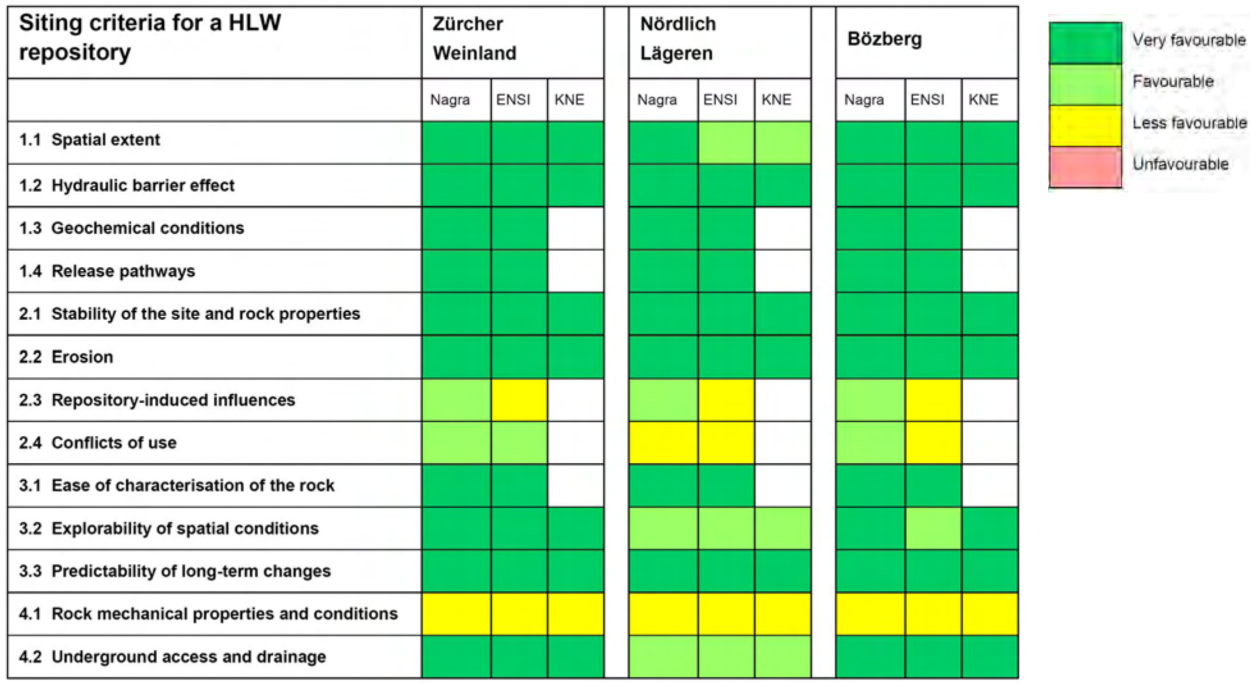

Figure A-6
Nagra, ENSI and KNE evaluation of the siting regions proposed by Nagra for the HLW/SF type repository. The evaluation is shown against the safety and technical criteria in the Sectoral Plan (ENSI, 2010). Copyright Swiss Federal Nuclear Safety Inspectorate.

A.4 The Swiss Nominative Site Selection Approach: Lessons Learned

The following key characteristics define the Swiss Site Selection approach and are considered to be the basis for the successful implementation.

- Nomination

The process itself is a nominating process with respect to the host rock and the siting regions, on the basis of safety and engineering criteria defined explicitly and in advance. Nagra will propose suitable host rocks and regions and Nagra has the burden of proof, as the implementer on behalf of the waste producers. For the surface facilities and their location, the site selection process is characterized by a strong participatory component by local communities, which will respect the concerns of the hosting regions including relevant socio-economic aspects.

- Clear roles and responsibilities

The Sectoral Plan defines very clearly the roles and responsibilities of the various political, administrative and technical bodies.

- Strong leadership by the Swiss Federal Office of Energy

The lead for the site selection process is taken by the Federal Office of Energy and participation of the concerned regions (communities, cantons, neighboring countries) is explicitly defined. However, the proposals for siting regions, potential sites and finally, the two geologic repositories in Switzerland, for low- and intermediate-level waste and for high-level / spent fuel waste are made by Nagra and are evaluated by ENSI; both bodies are outside the ministerial/governmental bodies.

- Regional participation

The process of regional participation (both local communities and Cantons) started early in the process. It has been conducted in a professional way, with predefined tasks and without false expectations.

- Step-wise decision making and approval

This is the major difference to previously adopted nominative approaches, in particular with respect to the decisions taken at each step and approved by the authorities and the Federal Council. The end of each stage defined in the Sectoral Plan is reached when the proposals made by the implementer (siting regions, at least two sites, site for each type of repository) are approved by the Federal Council.

- Technical competency of implementer and regulator

The Swiss siting process has been aided by the recognized and accepted technical and scientific know-how by both the implementer and the regulator.

- Transparency and traceability

All of the siting decisions made were traceable and transparent, as was recognized by the regulatory and national political authorities in their review.

- Commentary on the different types of site selection approaches

International site selection approaches followed to date can be broadly divided to nominative or voluntary. The nominative approach, also referred to as decide-announce-defend, is the approach used traditionally for major construction projects; it was also the one introduced at the earlier stages of geologic repository development. Over the last decade a shift has been made towards the voluntary approach, whereby the implementer announces exclusion criteria and compensation schemes for the interested communities and issues a call for volunteers. The approach has had success for surface engineered-based repositories, but not yet for deep geologic repositories.

The Swiss site selection approach is a further evolution of the earlier nominative approaches, with the main characteristics as described in the previous sections. Table A-1, summarizes the three approaches for ease of comparison. Notice that in all approaches the last three steps are exactly the same. The difference is what ‘drives’ or ‘initiates’ the process.

Table A-1
Different types of site selection approaches.

<table>
<tr><th>Nomination</th><th>Voluntary Site Selection Approach</th><th>Swiss Stepwise Nominative Site Selection Approach</th></tr>
<tr><td rowspan="4">Implementor investigates and evaluates various sites</td><td>Implementor announces criteria and call for volunteers</td><td>Plan, criteria, roles, responsibilities are defined and agreed upon</td></tr>
<tr><td>One or more community volunteer</td><td rowspan="2">Implementor proposes siting regions
Approval</td></tr>
<tr><td rowspan="2">Implementor investigates and evaluates sites</td></tr>
<tr><td>Implementor proposes (at least) two sites
Approval</td></tr>
<tr><td>Implementor proposes site</td><td>Implementor proposes site</td><td>Implementor proposes site</td></tr>
<tr><td>Authorities evaluate</td><td>Authorities evaluate</td><td>Authorities evaluate</td></tr>
<tr><td>Approval</td><td>Approval</td><td>Approval</td></tr>
</table>

In Switzerland, for example, having assigned the highest priority to the safety and engineering aspects of the geologic repository site, it is also logical to assign the first step to an organization that can evaluate the safety and engineering aspects of possible sites and have these aspects as the ‘driving’ force of the process. A voluntary approach could be perceived as driven from other aspects, not safety and engineering related, which could be detrimental to the credibility of the whole process. Because ‘perceptions’ however, are also influenced by cultural differences, in different countries this perception for the voluntary approach may vary.

In addition, the new aspect introduced in the Swiss approach is the stepwise decision-making and narrowing-down of the options to the specific site.

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A.6. Glossary of Key Documents and Terms for Switzerland

The following key bodies make up the regulatory and licensing framework for radioactive waste management in Switzerland:

- *Federal Council (federal government)*: issues the general license which is initially needed for each nuclear facility; the general license has to be approved by Parliament and is subject to a facultative referendum at the national level; the Federal Council also issues the closure order for disposal facilities.
- *Federal Department of Environment, Transport, Energy and Communication (DETEC)*: issues construction and operational licenses, as well as decommissioning orders.
- *Federal Office of Energy (SFOE)*: issues other types of license (transport, transit, import and export).
- *Swiss Federal Nuclear Safety Inspectorate (ENSI)*: Government's supervisory authority, (established as an independent body by the ENSI Act in 2009, formerly HSK), responsible for the safety of nuclear energy facilities and installations, including radioactive waste disposal.
- *Federal Office of Public Health*: responsible for radioactive material that does not fall under Nuclear Energy Act.
- *Advisory Bodies*: Federal Nuclear Safety Commission (KNS), Inter-Departmental Working Group on Radioactive Waste Management (AGNEB), Geological Commission on Nuclear Waste Management (KNE).

With regard to *implementer*, the producers of nuclear waste are responsible for management of all waste categories. The federal government is responsible for the management of radioactive waste generated by the use of radioisotopes in medicine, industry and research. The operators of nuclear power plants and the federal government formed the National Co-operative for the Disposal of Radioactive Waste (Nagra), which is responsible for the disposal of all kinds of radioactive waste, including irradiated nuclear fuel if declared as waste, with a view to implementing its permanent and safe disposal. Figure CH-1 shows the key players concerning radioactive waste management in Switzerland.

B
CURRENT INTERNATIONAL REPOSITORY CONCEPTS FOR THE GEOLOGICAL DISPOSAL OF USED NUCLEAR FUEL AND HLW

B.1 Introduction

If the US abandons its current license application for a geological repository at Yucca Mountain for the permanent disposal of commercial used nuclear fuel (UNF) and defense high-level waste (HLW), it will be left with future decisions about where and how to dispose of such wastes (or new HLW arising from the reprocessing of commercial used fuel). Similar situations now exist in several other countries that have deferred implementation of geological disposal for their nation's used fuel and HLW[44].

This uncertain US situation is compounded by the fact that it is unclear which of two approaches the US would pursue if future new siting were to be needed:

- A 'nominative' approach, in which a technical implementer organization for geological disposal would either evaluate and screen potential, diverse geological sites on the basis of technical criteria, as was done in the US from 1982 to 1987 under the Nuclear waste Policy Act (see Report II of this series of EPRI reports), or screen candidate sites having a specific type of host rock (see Swiss example, Appendix A of this report); or
- A 'volunteer' approach that begins with potentially interested local and regional communities coming forward and expressing an interest in the possibility of hosting the repository.

Clearly, either approach could lead towards almost any geological environment for geological disposal, although there would need to be broad geological 'exclusion criteria' to remove obviously unsuitable locations. It is not clear, at present, which geological environment(s) might emerge from a re-started US repository-siting program. What is clear, however, is that internationally there have been numerous repository concepts developed for a wide variety of host rocks. Consequently there is a considerable knowledge base and design flexibility that could aid in the rapid and cost-effective implementation of a safe repository concept rather than necessitating new design, testing and construction development programs.

[44] For example, the United Kingdom has recently commenced a major program to develop geological disposal concepts for its accumulated long-lived radioactive wastes and possible future arisings of wastes from a new fleet of nuclear power reactors expected to begin around 2020. This program is managed, on behalf of the government by the Radioactive Waste Management Directorate (RWMD) of the UK Nuclear Decommissioning Authority (NDA). With the permission of the NDA, this Appendix draws upon a previous NDA study of international repository concepts

From the wide range of concepts that have been considered worldwide over the last 30 or more years, it has been possible to identify twelve basic groups of design concepts that capture the full range of possible options that it might be necessary to consider when diverse geological sites begin to be considered. This Appendix summarizes these twelve design concepts, and is based upon detailed reports that are available from the NDA and references published in the last two years [1-5]. Three generalized geological host environments have been adopted by the NDA in the UK:

- ‘strong, hard rocks’;
- ‘less strong, sedimentary rocks’; and
- ‘evaporites’.

It can be seen that internationally developed concepts for granite/ gneiss/ ‘crystalline’ rock (e.g. Finland, Sweden, China, Taiwan, Koreas, Switzerland) fall into the first group, clays/ marls/ argillaceous rocks (e.g. France, Belgium, Switzerland) into the second group, and dome and bedded salt formations (e.g. Germany, WIPP in the US) into the latter.

The remainder of this Appendix examines the twelve basic repository design concepts for disposal of used fuel and/or HLW.[45] It should be noted that this review does not consider hydrogeologically unsaturated groundwater environments such as that of Yucca Mountain, and focuses instead on conceptual design s for HLW disposal in saturated sites.

B.2 Generic Geological Disposal Concepts

The objective of a geological disposal system is to contain and isolate wastes deep underground until such time as radioactive decay has reduced their hazard to levels similar to naturally occurring radioactive materials, whilst simultaneously ensuring that any releases of radioactivity, at any time in the future, do not cause unacceptable hazards to people or the environment. This is achieved by a combination of natural and engineered barriers that work in concert with each other at different times in the future evolution of the disposal system. After repository closure, the repository functions *passively* to achieve these objectives and is illustrated schematically in Figure B-1. Different weights are placed on the performance and interaction among the barriers for expected environmental conditions for diverse host rocks and sites, as well as credible perturbations to such conditions for different host rock and sites.

[45] NDA has also published a parallel analysis of existing repository design concepts for the disposal of ILW, either separately or together with HLW (6).

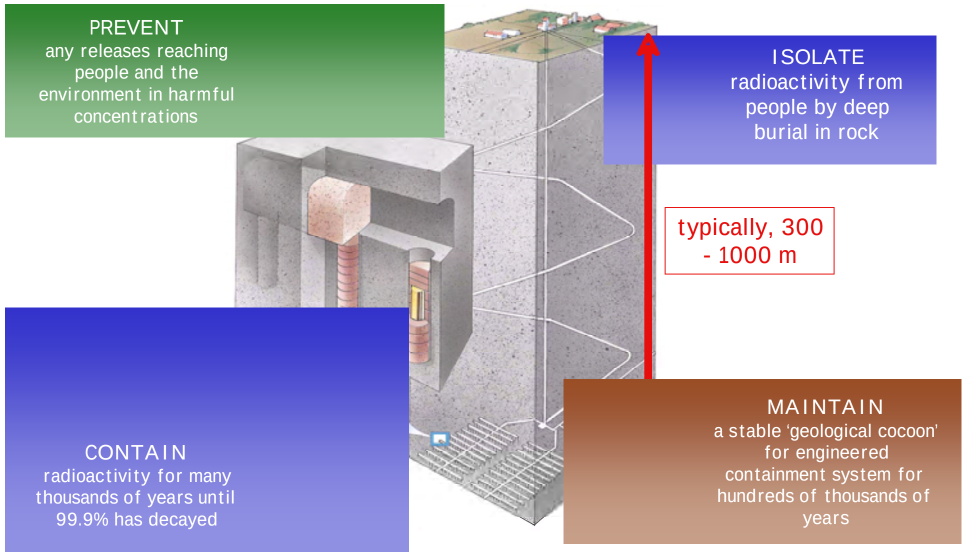

Figure B-1
Schematic illustration of the objectives of geological disposal – to provide a passively safe system to isolate and contain wastes, allow them to decay to low levels of radioactivity and prevent any releases that do occur from reaching people and the environment in harmful concentrations.

Geological repository concepts for disposal are driven by two basic safety strategies (Apted and Ahn, 2010):

- containment ('delay and decay') barriers and processes, and
- concentration attenuation (dispersion and dilution) barriers and processes.

Design implementations of these strategies are essentially combinations of materials and design features with specific safety functions, matched to the particular geological conditions of a site, i.e., features commonly referred to as the THMC (thermal, hydrological, mechanical, and chemical) properties of the host environment, with geochemical and hydrological dominating. Table B-1 illustrates the relative performance of typical geological host environments with respect to common requirements of a repository. The actual performance of a repository system will be strongly formation-, environment- and location-specific and the intention in presenting Table B-1 is to give an illustrative, stylized picture of relative strengths and weaknesses to illustrate how any option represents a balance of properties that must themselves be balanced by the properties of a repository design to achieve an optimal solution. It is important to note that no environment is ideal in all respects, hence the need to match EBS and NBS properties to attain adequate system performance.

Table B-1
Highly generalized indication of the relative performance of typical geological environments with respect to various possible requirements of a repository.

	Hard Rocks (such as granite)	Sedimentary Sequences	Evaporites (salt formations)	Unsaturated Tuffs
Heat conduction	Good	Variable	Extremely good	Good
Host rock hydraulic conductivity	Good	Very good	Extremely good	Good
Stable near-field hydrochemistry	Variable	Very good	Extremely good	Variable
Low groundwater flux geological environment	Variable	Extremely good	Extremely good	Good
Intrusion potential	Good	Variable	Possible	Very good
Construction flexibility	Extremely good	Variable	Good	Very good

The principal elements from which a repository concept can be developed are:

- the waste container and any overpack material: for UNF and HLW these are generally metals, with various degrees of corrosion resistance, depending on how their performance is intended to fit into the overall safety concept;
- buffer (called “backfill” only in the US) material that is emplaced around the waste packages to help isolate them physically and chemically from the host rock and groundwater system;
- engineered systems designed to assist emplacement of packages and buffer that may be left in situ in some concepts (e.g. borehole liners);
- the geometrical configuration of the elements above: the amount, spacing, layout, thickness of materials and items.

A complete system design also includes other elements such as backfill materials for void spaces, seals and plugs for tunnels, shafts and boreholes, support systems for openings that may be left in situ when the repository is closed (e.g. tunnel liners) etc.

The information to develop the Generic Disposal Concepts was gathered from numerous international studies that have been carried out over many years, all of which have considered feasible geological disposal solutions for national waste management programs. Some of these studies have progressed only so far as outline concepts, while others are underpinned by decades of research and development and form the basis of mature national programs. Some of the international studies are only for HLW or only for UNF, whilst others consider disposal of both in the same repository.

It is important to appreciate that it is inevitable that the basic, Generic Concepts described here, when deployed in any repository program, would need to go through several stages of adaptation and tailoring to site-specific factors, engineering requirements and stakeholder preferences. A repository development program, from siting through to implementation, is expected to last decades. The design that would eventually emerge for licensing and implementation would thus certainly look different in detail to the Generic Concepts illustrated here.

The set of twelve deep geological Disposal Concepts was derived from a list of seven key features that were considered to differentiate between Disposal Concepts. For each of these key features, such as emplacement in short boreholes from a disposal tunnel, variants were identified that give rise to specific repository Concepts; for example, vertical and horizontal boreholes are considered as separate repository Concepts. The list of key features and their variants is given in Table B-2. The Concepts considered are suitable for either HLW or UNF, or (generally) for both.

Table B-2
Key features and variants leading to the UNF and HLW disposal Concepts.

Key Feature	Variants	Concept No.
In-tunnel (borehole)	Vertical borehole	1
	Horizontal borehole	2
In-tunnel (axial)	Short-lived canister	3
	Long-lived canister	4
In-tunnel (axial) with supercontainer	Small working annulus	5
	Small annulus + concrete buffer	6
	Large working annulus	7
Caverns with cooling, delayed backfilling	Steel MPC + bentonite backfill	8
	Steel or concrete/DUCRETE container + cement backfill	9
Mined deep borehole matrix		10
Hydraulic cage	Around a cavern repository	11
Very deep boreholes		12

Some variants could be common to several Concepts. For example, the use of a long-lived or short-lived overpack/canister is examined as a variant of an ‘in-tunnel (axial)’ Disposal Concept. However, use of long-lived or short-lived overpack/canisters could also be a variant of several other Disposal Concepts such as ‘in-tunnel (borehole)’ and ‘in-tunnel (axial) with supercontainer’ and their variants. Evaluation of closely related variants would duplicate a good deal of material, with no obvious benefit.

Each of the twelve generic Geological Disposal Concepts is described in a tabular format, showing how and for what environment or host rock the Concept was developed, what its advantages and shortcomings are with respect to implementation and safety and how much work has been carried out on it worldwide. The tables are drawn and adapted from the source study (Baldwin et al., 2008).[46]

It was assumed that all of the Concepts could be developed so as to provide levels of safety that are acceptable to regulatory authorities. Consequently, their safety performance from international studies over many years and other precedents is used simply to comment on the degree of confidence in safety for each Concept, as well as to show what uncertainties exist and where they lie (expressed in terms of how straightforward it may be to make a safety case).

The diagrams of Disposal Concepts presented in this Appendix were developed specifically for the NDA evaluation, to show the main features of each Generic Concept in the form of a schematic illustration. The diagrams illustrate the overall geometry of the repository system and include a blow-up section showing more detail of the engineered barrier system. Note that the diagrams, in particular the blow-up sections, are not drawn to scale.

The twelve basic repository concepts are illustrated in the following schematic graphics (Figures B-2 to B-12) and are adapted from a report prepared for the UK Nuclear Decommissioning Authority and used with permission from the authors and NDA.[47]

[46] Adapted and incorporated with permission of the authors and the UK NDA.

[47] Baldwin, T., Chapman, N. A. and Neall, F. 2008. Geological Disposal Options for High-Level Waste and Spent Fuel. Contractor Report Prepared for the UK Nuclear Decommissioning Authority, January, 2008.

Concept 1	In-tunnel (vertical borehole) with long-lived or short-lived canister
Main characteristics of the Concept	
Waste is emplaced in short (typically 6-8 m), medium to large diameter (e.g. 0.6 to 1.5 m) boreholes drilled in the floor of disposal tunnels. The waste is emplaced in a metal canister (or "overpack" in the case of HLW containers). Where long-lived containment within the canister/overpack is required, a corrosion-resistant copper canister with an iron insert is commonly used. Alternatively, a short-lived (some hundreds to a few thousands of years) steel canister can be used in environments where the safety case places less emphasis on long-term canister integrity. The annulus around the waste package is usually filled with a buffer material to isolate and protect the canister, typically highly compacted bentonite, but other materials could be used, for example, crushed salt in a salt host rock, as described for Concept 4.	
Main drivers for the Concept	
The Concept was originally developed for disposal of spent nuclear fuel in stable crystalline basement rocks in Sweden in the KBS study in the late 1970s and early 1980s. The vertical deposition boreholes were developed in response to uncertainty about the extent and properties of the EDZ (excavation damaged and disturbed zone) around the deposition tunnel. It was unknown whether the EDZ could provide a long, interconnected, high porosity and permeability zone in which water flux would be high and which could provide fast advective pathways to water-conducting fracture zones, thus circumventing part of the geosphere barrier. The boreholes were intended to isolate each waste package and place them in less disturbed rock beyond the tunnel EDZ so that radionuclide releases would be slowed by transport through undisturbed rock before reaching the tunnel EDZ or other geosphere pathways (note however, that this places significant requirements on the quality of the rock at each borehole site). The use of a copper (or titanium) canister to provide a very long period of containment was driven by the realization that the instant release fraction (IRF) of radionuclides not held within the UO_2 matrix (or the zircaloy cladding) of the UNF could be released as a pulse. This may not be greatly attenuated by the geosphere, due to the chemical properties of these radionuclides (e.g. very high solubility and low sorption) and the hydrogeological properties of the geological environment. The IRF could thus give rise to unacceptably high releases to the biosphere unless the waste were contained for long enough to allow radioactive decay of radionuclides such as ^{14}C. Moreover, the longevity of the canisters means that failures are likely to be spread over a long period of time and to be distributed throughout the repository area, thus being dispersed and diluted in time and space. In geological environments where a high degree of containment is provided by the host rock and geosphere, short-lived steel canisters are also considered suitable.	
Important Aspects of the Concept	
The Concept is flexible with respect to disposal of both UNF and HLW and to implementation in a wide range of host rocks. The Concept is mature for implementation in crystalline rocks and has a substantial knowledge base – amongst the best of any Concept worldwide – from 30 years of focused research, development and demonstration by many national programs, most notably those undertaken by SKB (Sweden) and Posiva (Finland). Where corrosion-resistant materials are used, the engineered barrier provides long containment and significant decay of radionuclides. This reduces the emphasis on the host rock and geosphere properties, as long as the natural barrier can be relied upon to isolate and protect the EBS in the long-term. The Concept results in a relatively large excavated volume per waste package, and a proportionately larger repository than some axial emplacement, because of the size of the disposal tunnels required to handle the waste packages (especially long PWR UNF elements). The emplacement of compacted bentonite buffer to the density specifications and geometrical tolerances required is difficult in wet host rocks and additional development and demonstration of procedures for emplacing the waste and the EBS is required. The acceptable conditions for a disposal borehole (e.g. fracturing, inflow rate) are critical for long-term safety performance but operational criteria have not yet been fully established and will be site-specific.	

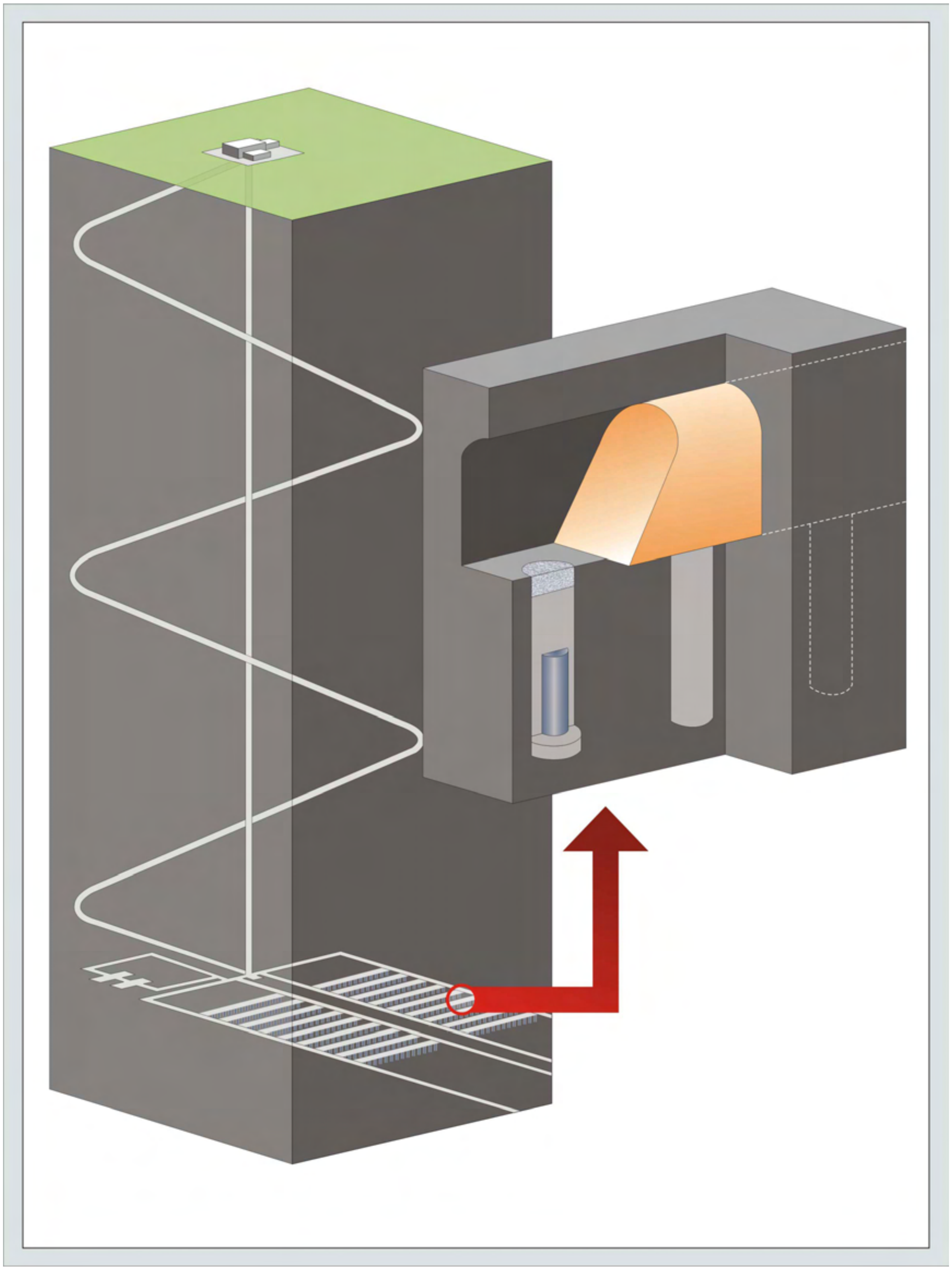

Figure B-2
Schematic illustration of Concept 1, in-tunnel (vertical borehole) with long- or short-lived canister. In this illustration, 'panels' of disposal tunnels are shown incomplete and the extent of the repository and additional ventilation shaft(s) is not indicated. In all the Concepts, construction (in some panels) and waste emplacement (in others) could be carried out in parallel. This is an operational and strategic decision. From Baldwin et al. (2008); used with permission.

Concept 2	**In-tunnel (horizontal borehole) with long-lived or short-lived canister**
Main Characteristics of the Concept	
One or more waste packages are emplaced in short, horizontal or near-horizontal, large diameter (~0.7-1.5 m) boreholes drilled in the walls, usually on both sides, of the disposal tunnels. The waste is emplaced in a metal canister (or “overpack” in the case of HLW containers). Where long-lived containment within the canister/overpack is required, a corrosion-resistant material is used, for example a copper canister with an iron insert. Alternatively, a short-lived (some hundreds to a few thousands of years) steel canister can be used in environments where the safety case places less emphasis on long-term canister integrity. A liner may be used to support the boreholes and a buffer (e.g. bentonite) may be included to fill the space around and between the waste packages.	
Main drivers for the Concept	
The original Concept was developed by SCK·CEN (Belgium) and ANDRA (France) for disposal of mainly vitrified HLW in clay host rocks. A similar Concept has been developed in the Netherlands for disposal in salt. For use in crystalline, fractured rocks, the Concept would be closely related to Concept 1. The Concept has usually only been considered for deposition holes which contain one to a small number of waste packages, that is, holes up to a few tens of meters in length, because of rock mechanical considerations in weaker rocks which limit the borehole length unless a very substantial liner is used. Also, with longer holes, small changes to borehole diameter are more likely (both from excavation and later deformation) and could disrupt emplacement operations of waste packages that are only slightly smaller than the hole and are simply pushed into place. ANDRA note that the use of horizontal boreholes, up to a few tens of meters in length, makes better use of the laterally (rather than vertically) extensive argillaceous formations present in their proposed repository siting area, which outweighs the extra difficulties of handling packages in horizontal rather than vertical deposition holes. ANDRA also prefers to use metal-lined deposition holes with no buffer material as this provides adequate containment in the host rocks identified whilst reducing hole size, use of concrete and the volume of clay disturbed by the excavation. The use of a substantial borehole liner also allows retrieval of waste emplacement for a significant period after emplacement, which is a requirement of the ANDRA disposal program.	
Important Aspects of the Concept	
Implemented as envisaged by ANDRA, that is, in argillaceous rocks, which provide a good natural barrier, and without the use of backfill, the Concept is relatively simple, practical and efficient in terms of potential emplacement rate, excavated volume per waste package and the repository footprint. The Concept is flexible in principle with respect to disposal of both HLW and UNF, although outstanding uncertainties regarding the thermal impact of UNF on the argillaceous host rock and steel liner (especially if reversibility is required) need to be addressed. Should waste retrieval be required, this Concept may offer advantages over axial tunnel emplacement Concepts, which require completion of the whole disposal tunnel up to sealing, and also vertical emplacement options. The Concept is not very mature either for crystalline or sedimentary host rocks, despite its relatively long history, with demonstration testing particularly lacking. However, for crystalline host rocks, much of the extensive knowledge base of the KBS-3 Concept would be relevant and, for argillaceous rocks, studies from other programs (e.g. France, Belgium, Switzerland) would provide significant information. Although the Concept is, in principle, flexible with respect to different host rocks, significant development work would be required, for example, for stronger, fractured rocks where buffer emplacement is required. In this case, problems of emplacing bentonite in wet host rocks, similar to those for Concept 1, may be expected. For fractured rocks particularly, the acceptable conditions in the emplacement boreholes (e.g. inflow rate, fracture size and density) could be important for long-term safety performance and will require very detailed site characterization information.	

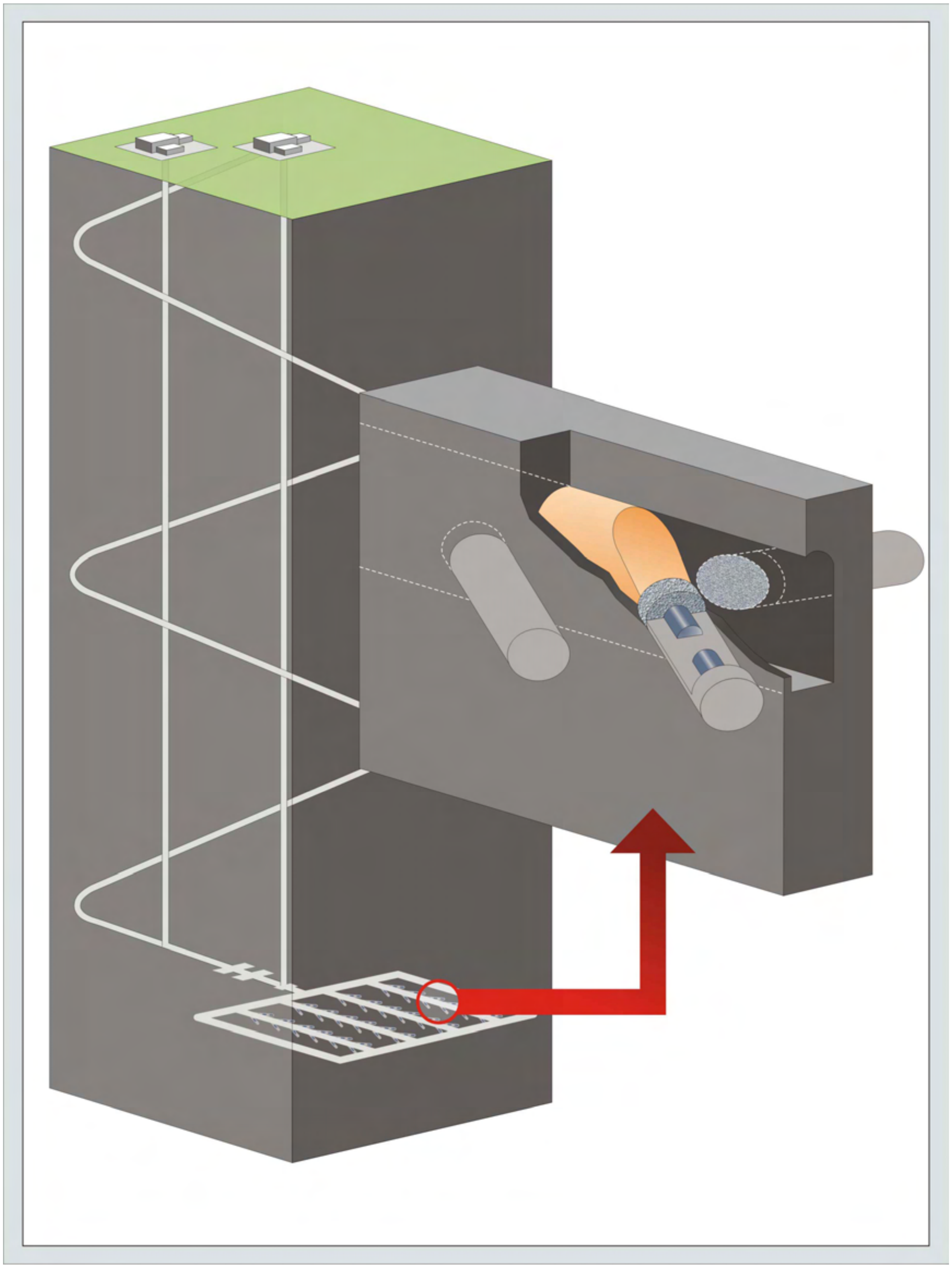

Figure B-3
Schematic illustration of Concept 2, in-tunnel (horizontal borehole) with long- or short-lived canister. From Baldwin et al. (2008); used with permission.

Concept 3	In-tunnel (axial) with short-lived canister and buffer
Main Characteristics of the Concept	
Waste, encapsulated in steel overpacks, which provide complete containment for some hundreds or thousands of years, is emplaced axially along disposal tunnels surrounded by a thick buffer layer of bentonite that completely fills the tunnel with no further backfill. Waste packages are also separated by sections of buffer. The disposal tunnels are closed immediately after completion of waste emplacement with very substantial seals to resist the bentonite swelling pressure.	
Main drivers for the Concept	
The Concept was developed by Nagra for HLW disposal for Project Gewähr 1985, the legally required demonstration of HLW disposal feasibility in Switzerland, for crystalline basement rocks in Northern Switzerland, and also by Ondraf/Niras in SAFIR 2 for disposal in the plastic Boom clay in Belgium. In Switzerland, based on extensive experience with tunnel excavations, it was expected that the EDZ would be minor if tunnel boring machine (TBM) technology could be used to excavate the circular cross-section disposal tunnels in crystalline rock. Waste packages could be well isolated along the axis of the tunnel with the use of a thick bentonite buffer, avoiding the influence of a tunnel EDZ in this way, rather than by using vertical boreholes (Concept 1). Axial deposition in circular tunnels was also seen as an efficient disposal method as the excavation volume per waste package is minimized due to the relatively small tunnel diameter and no individual disposal boreholes are required. Similarly, in the Boom clay, the properties of the host rock suggested that an EDZ developing around the disposal tunnels during excavation would close due to creep after tunnel sealing and would not remain a significant issue for long-term safety. The Concept has also been adopted by Nagra for UNF/HLW co-disposal in a clay host rock (the Opalinus clay in Northern Switzerland). The relatively large tunnel diameter (3.7 m in Swiss crystalline rocks; 2.5 m in Opalinus clay; 2.0 m in Boom clay) allows for a thick bentonite buffer around the waste packages (up to 1 m in diameter). In the Concept as envisaged for crystalline host rock, the buffer has an initial role to protect the waste canister/overpack but, since this was assumed to provide containment for only about 1,000 years, the main long-term role was as a barrier to transport of radionuclides into the geosphere. The SAFIR 2 Concept in the plastic Boom clay used a steel tube in the center of the buffer to facilitate emplacement of the waste packages. This Concept has been abandoned by the Belgian program, however, because there were concerns relating to the corrosion of the canister and also with respect to the practical implementation, particularly regarding thermal expansion of the central steel tube. This Concept is also under consideration by DBE, Germany, for disposal of UNF in a salt dome formation at Gorleben; in this case crushed salt is envisaged as the backfill around the canisters.	
Important Aspects of the Concept	
The Concept is flexible with respect to disposal of both UNF and HLW and to implementation in a wide range of host rocks, although relatively dry host rocks are favored. The Concept results in a small excavated volume per waste package, and a proportionately small repository, because the disposal tunnels are used for emplacement of the waste axially and there is no need to invert/rotate the waste packages into boreholes, nor for additional excavation of boreholes. The Concept is mature for implementation in both crystalline and sedimentary rocks (including evaporites) and has a very substantial knowledge base from 30 years of focused research, development and demonstration by several national programs, including those in Switzerland (Nagra), Japan (NUMO), Spain (Enresa), Belgium (Ondraf/Niras) and Germany (DBE Technology). The short containment period provided by the steel canister means that all waste packages are likely (and must be assumed) to fail over a relatively limited period of time. This places emphasis on performance of the geosphere, especially for UNF and the IRF, as the natural barrier must be relied upon to attenuate the radionuclide releases. The emplacement of compacted bentonite buffer to the density specifications required is difficult in a wet host rock. Bentonite granules may provide a suitable medium in relatively dry environments but may not produce a sufficiently dense material in wet, fractured rocks. Acceptable conditions for a disposal position (e.g. fracturing, inflow rate) are likely to be very important for long-term safety performance; acceptance criteria are probably site-specific and will require a highly detailed site characterization program, especially in fractured host rocks.	

Concept 4	In-tunnel (axial) with long-lived canister and buffer
Main Characteristics of the Concept	
Waste, encapsulated in a copper (or titanium) corrosion-resistant canister, which provides a long period of complete containment, is emplaced axially along disposal tunnels surrounded by a thick buffer layer of bentonite that completely fills the tunnel, with no further backfill. Waste packages are also separated by sections of buffer. The disposal tunnels are closed immediately after completion of waste emplacement with substantial seals to resist the bentonite swelling pressure.	
Main drivers for the Concept	
The Concept similar in essentials to Concept 3, but uses a long-lived copper or titanium canister, with an iron insert for mechanical strength, in place of the short-lived steel overpack. The Concept has so far only been developed by Ontario Power Generation (OPG) in Canada, for disposal of UNF in the crystalline rocks of the Canadian Shield. The use of the long-lived canisters confers advantages over steel canisters with respect to the instant release fraction (IRF) of the UNF, as the longer containment time allows significant decay of some safety-relevant IRF nuclides (e.g. ^{14}C), as well as spreading canister failures, and thus releases, over a very long time period. The OPG design also differs from Concept 3 in having two UNF waste packages side-by-side at each disposal position to accommodate the very strong stress anisotropy found in parts of the Canadian Shield that requires the excavation of oval cross-section tunnels for stability of the openings. This design supersedes an earlier Canadian (AECL) design using short vertical boreholes in the floor of disposal room as the stress anisotropy gave problems with borehole stability. In the OPG design, the buffer has several component parts, using bentonite and bentonite/sand mixtures with different initial densities. The high-density bentonite around the waste package is emplaced as a single unit, surrounding both waste packages, on to a bentonite-sand base. This simplifies and speeds the operation by avoiding the use of many complex shaped blocks, as well as providing additional shielding during the emplacement operation.	
Important Aspects of the Concept	
The Concept is flexible with respect to disposal of both UNF and HLW and to implementation in a wide range of host rocks, especially in the circular cross-section tunnel version similar to Concept 1. The Concept can result in a smaller excavated volume per waste package than Concept 1, and a proportionately smaller repository, because the disposal tunnels are used for emplacement of the waste axially and there is no need to invert/rotate the waste packages into boreholes, nor for additional excavation of boreholes. The long containment times provided by the engineered barrier, with the use of copper or titanium canisters, allows significant decay of radionuclides, especially important for the UNF IRF, and spreads the releases over a very long period of time. This reduces the emphasis on the host rock and geosphere, as long as the natural barrier can be relied upon to isolate and protect the EBS in the long-term. Axial emplacement of waste using a long-lived canister is not a Concept preferred in any major nuclear country, although it combines elements of two of the most highly developed disposal Concepts (Concept 1 and Concept 3), and the substantial knowledge base developed for these Concepts is relevant and could be applied during implementation of Concept 4. The emplacement of compacted bentonite buffer to the density specifications required is difficult in a wet host rock. The OPG procedure as specified is likely to be practical as specified only in dry host rocks. Use of bentonite granules may also provide a suitable method in relatively dry environments but may not produce a sufficiently dense material in wet, fractured rocks. Acceptable conditions for a disposal position (e.g. fracturing, inflow rate) are likely to be important for long-term safety performance requiring a highly detailed site characterization program, especially in fractured host rocks. This may be less important for the thick, compound buffer of the OPG Concept than for a circular tunnel with the more usual 0.3 – 0.7 m thick buffer, similar to Concept 1.	

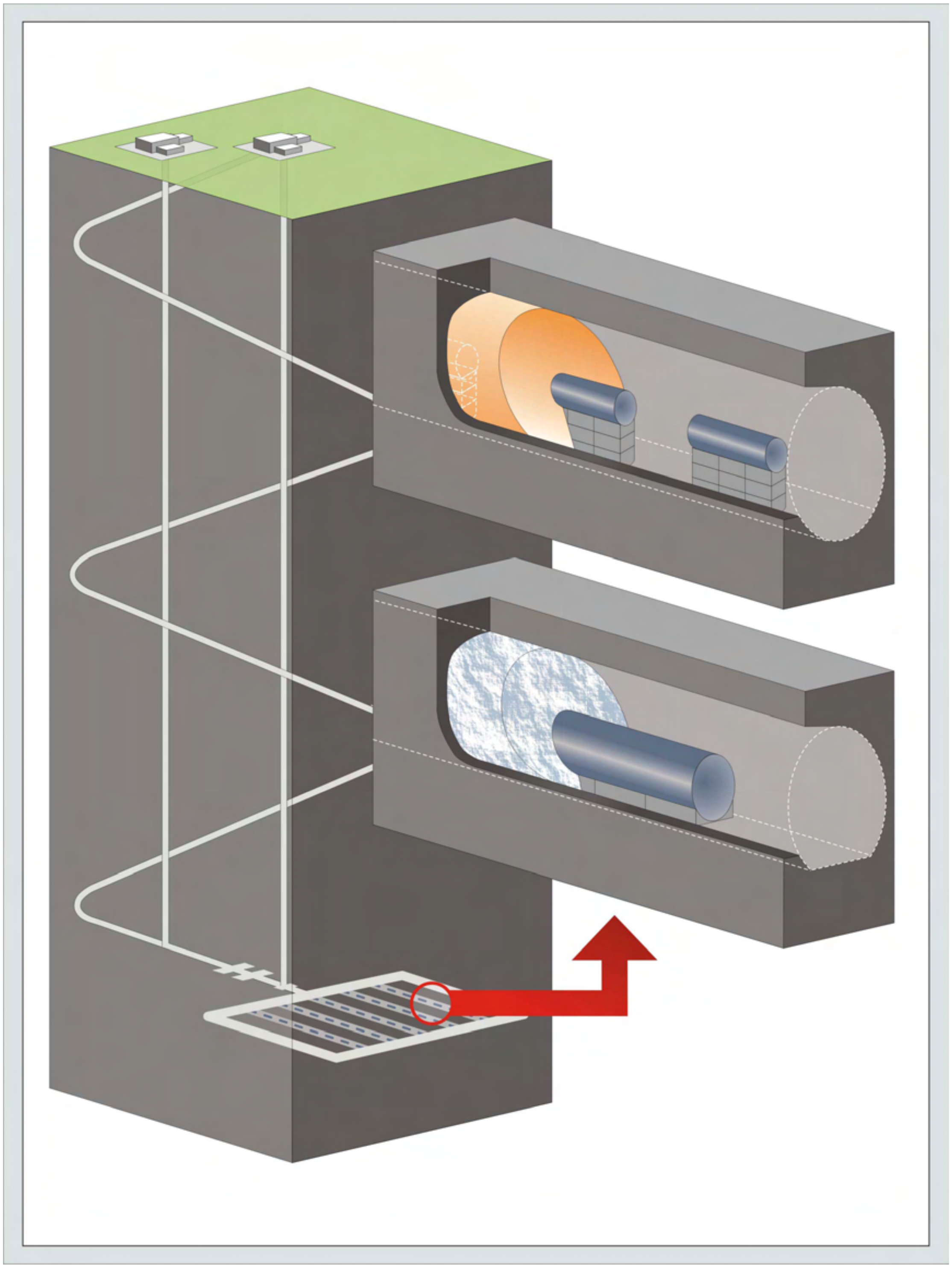

Figure B-4
Schematic illustration of Concepts 3 and 4, in-tunnel (axial) with long- and short-lived canister and buffer (the lower figure illustrates disposal in salt, with a salt backfill). From Baldwin et al. (2008); used with permission.

Concept 5	In-tunnel (axial) with supercontainer (small annulus)

Main Characteristics of the Concept

Waste is emplaced axially in circular tunnels in the form of a supercontainer in which the waste, overpack or canister and buffer are pre-assembled at a surface facility into an enclosed, perforated steel handling shell. The tunnel is as small as possible to minimize the void space around the supercontainer, which must be filled by swelling of the buffer as no additional backfill or buffer material is used. Bentonite buffer sections are used to isolate one or more supercontainers.

Main drivers for the Concept

SKB (Sweden) and Posiva (Finland) decided to assess a horizontal version of the KBS-3 Concept, known as KBS-3H, to avoid potential problems with spalling around vertical deposition boreholes.

In addition, axial emplacement of the waste packages made it possible to take advantage of a reduced EDZ around the disposal tunnel by using a small diameter, TBM-excavated tunnel. Placing of the bentonite buffer in the form of pre-formed bentonite blocks in high humidity conditions was, however, problematic due to premature swelling of the bentonite causing cracking and disintegration. This led to the development of the supercontainer in which the buffer and waste package are emplaced as a single unit inside a perforated handling shell.

To avoid the necessity for additional backfill, the tunnels are as small as is feasible; the supercontainers are emplaced using water cushion equipment that requires only a very small (few centimeters) effective clearance to lift and transport the supercontainer into position.

Bentonite in the supercontainer will swell and extrude out of the handling shell once saturation begins. However, as the amount of bentonite is only just sufficient to fill the 5 cm annulus to the required density, it is important that no bentonite is displaced. Thus each supercontainer is isolated in the tunnel by a bentonite-only section (distance block). The distance blocks also serve to space out the waste packages to ensure the thermal limits on the bentonite are not exceeded.

Important Aspects of the Concept

The Concept is flexible with respect to disposal of both UNF and HLW. It is also flexible, in principle, with respect to implementation in a range of host rocks, although any tendency to friability in the host rock that could introduce loose material into the annulus would preclude weaker rocks (see below).

The Concept can result in a small excavated volume per waste package – and a proportionately small repository – because the disposal tunnels are used for emplacement of the waste axially and there is no need for the space required to invert/rotate the waste packages into boreholes, nor for additional excavation of boreholes.

The Concept is relatively mature for implementation in crystalline rocks (e.g. Sweden and Finland) as it builds to a large extent on the very substantial knowledge base developed for Concept 1, including site-specific information of SKB and Posiva.

Acceptable conditions for a disposal position (particularly water inflow rate) are critical for avoidance of bentonite erosion from the supercontainer and the immediate annulus. Besides the requirement for detailed characterization of disposal tunnels, this may also limit suitability of this Concept in wet, fractured host rocks if significant lengths of disposal tunnels are unusable.

Uncertainty about the long-term effect of interaction between bentonite and the corrosion products of the handling shell means that the effective thickness of the buffer may be reduced from the nominal emplaced thickness. Thus the necessity to avoid the detrimental effects of interaction with a concrete tunnel liner could preclude implementation in many weaker rocks.

The very small annulus around the supercontainers means that recovery/reversal of emplacement of a supercontainer using the water cushion equipment (which must slide under the supercontainer) may be awkward, especially after the bentonite has begun to extrude in wetter sections of the tunnel. Thus the difficulties of reversal or retrieval may be greater than for in-hole Concepts.

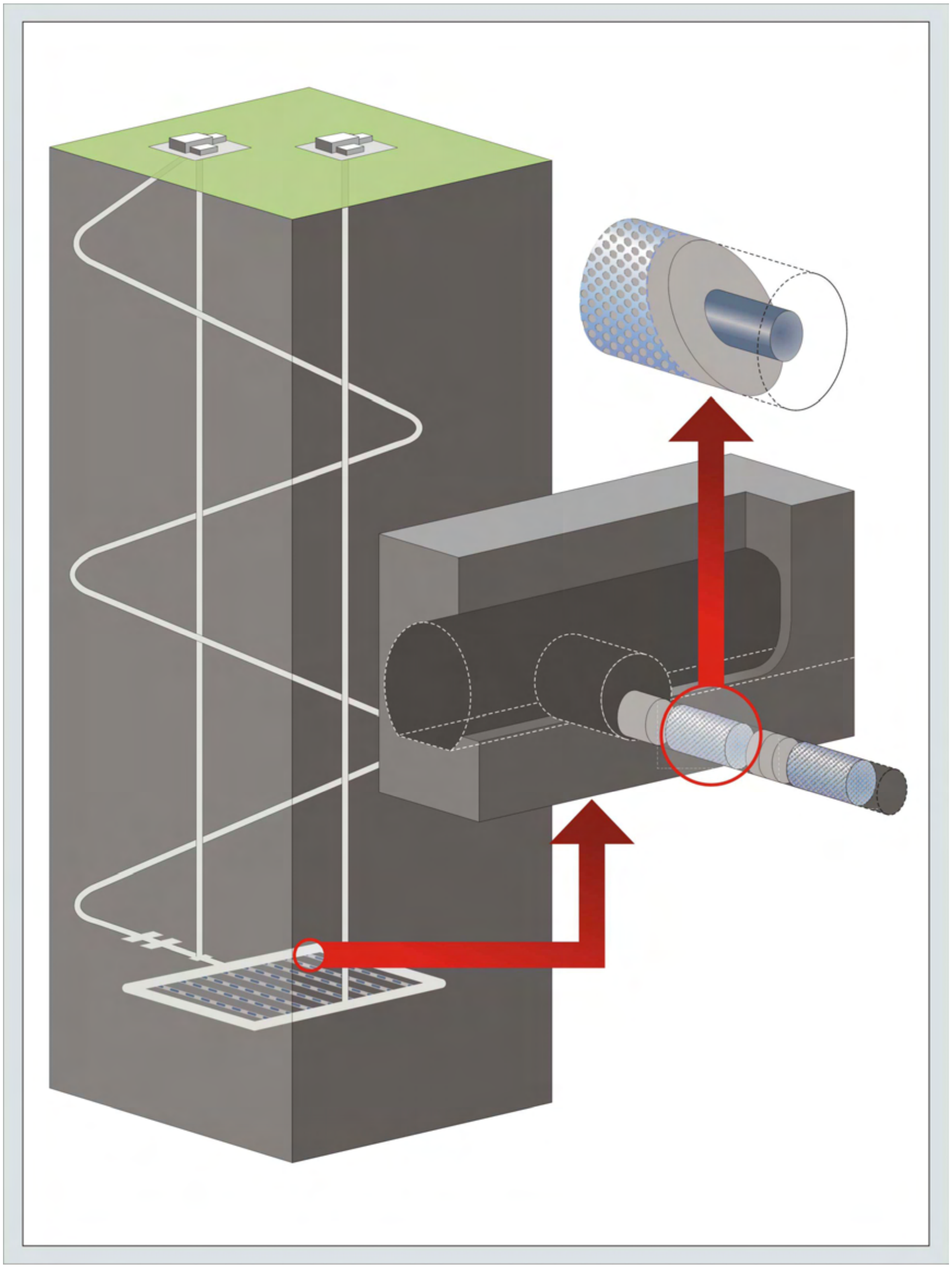

Figure B-5
Schematic illustration of Concept 5, in-tunnel (axial) with supercontainer (small annulus). From Baldwin et al. (2008); used with permission.

Concept 6	In-tunnel (axial) with supercontainer (concrete buffer)
Main Characteristics of the Concept	
Waste is emplaced axially in circular tunnels, probably lined for support, in the form of a supercontainer in which the waste, canister or overpack and buffer are pre-assembled at a surface facility into an enclosed steel handling shell. The buffer material is ordinary Portland cement (OPC)-based concrete. The tunnel diameter may be significantly larger than the supercontainer, which sits on a small pedestal at its disposal position. Where a large annulus is used, additional cement-based backfill is used to fill around and between the supercontainers. The tunnels may be some hundreds of meters long to take advantage of a laterally extensive formation.	
Main drivers for the Concept	
The Concept was developed by Ondraf/Niras for the plastic Boom clay (Belgium) to address potential problems which were identified with the earlier SAFIR 2 Concept in which HLW flasks in thin steel overpacks were slid into position along a steel tube positioned axially in a bentonite-filled disposal tunnel (Concept 3). The SAFIR 2 design was replaced primarily because of uncertainty regarding EBS performance. In particular, it was considered possible that certain types of corrosion, such as localized corrosion or stress corrosion cracking, might threaten the integrity of the overpack during the thermal phase. There were also questions with respect to the practical implementation, particularly regarding thermal expansion of the central steel tube. Expansion, coupled with the swelling pressure exerted by the clay buffer, would cause the steel to be highly stressed, possibly leading to plastic deformation. Scoping calculations showed that the maximum permissible tube length would be less than 20 m. Other uncertainties related to the difficulty of transport and emplacement of an unshielded overpack, and quality assurance of the engineered barriers. By using a supercontainer the waste is shielded throughout the emplacement procedures, which are simplified (although the supercontainer is larger and heavier to handle), and there are no detrimental interactions between the buffer, backfill and the concrete tunnel liners. The OPC-based concrete buffer (and backfill used) is designed to provide alkaline, passivating conditions around the steel overpack, enhancing longevity by reducing corrosion. After failure of the overpack, it will continue to provide a favorable chemical environment as well as a diffusion/transport barrier.	
Important Aspects of the Concept	
The Concept is flexible with respect to disposal of both UNF and HLW. The Concept can result in a small excavated volume per waste package, and a proportionately small repository, because the disposal tunnels are used for emplacement of the waste axially and there is no need for the space required to invert/rotate the waste packages into boreholes, nor for additional excavation of boreholes. The use of the supercontainer increases safety during the emplacement operations by making the procedures simpler and more robust (e.g. fewer components to emplace) and also by use of a self-shielded waste package. The Concept is based on approximately five years of research and development and it builds to a large extent on the substantial knowledge base developed for the SAFIR 2 Concept and investigations in the Boom Clay. Detailed development is still ongoing by Ondraf/Niras and practical demonstration of many aspects, for example, backfilling with cement-based mortar, has not yet been undertaken. The long-term performance of the OPC-based buffer material in terms of a transport/diffusion barrier is uncertain, especially in an environment where cracking could occur due to rock stresses. The Concept is also most appropriate for host rocks where the interaction between the highly alkaline pore fluids from the concrete and the surrounding rock will be limited to the vicinity of the near field and not significantly affect the geosphere. The large, heavy supercontainers place particular demands on the transport and handling systems, as well as on access tunnels.	

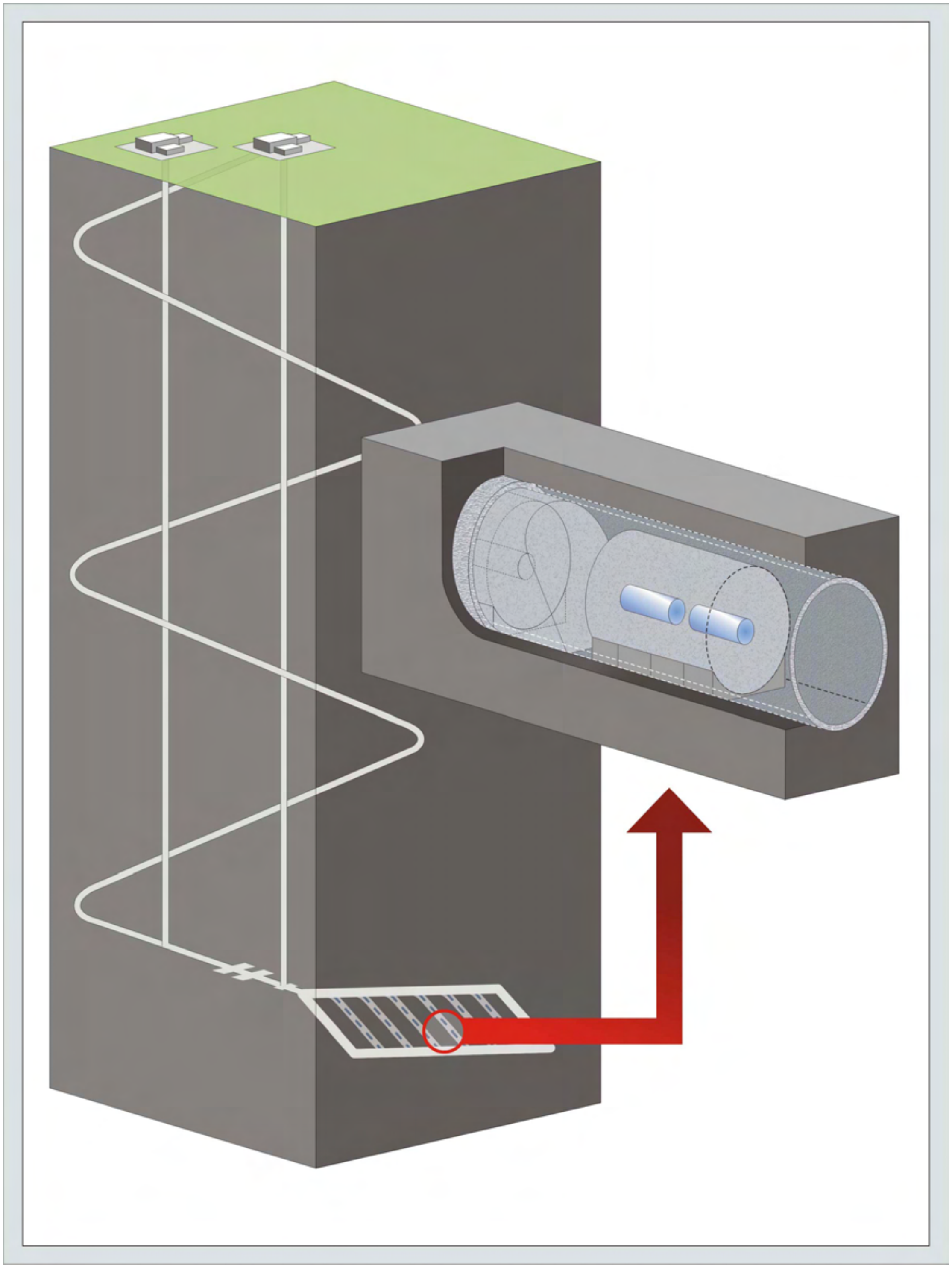

Figure B-6
Schematic illustration of Concept 6, in-tunnel (axial) with supercontainer (concrete buffer). From Baldwin et al. (2008); used with permission.

Concept 7	**In-tunnel (axial) with supercontainer (large annulus)**
Main Characteristics of the Concept	
Supercontainers, comprising waste, overpack and compacted bentonite buffer assembled in a robust steel handling shell, are emplaced horizontally along the disposal tunnels on short pedestals which facilitate handling but do not necessarily place the supercontainer in the centre of the tunnel. The tunnels are larger than the supercontainer diameter by ~1 m or more to allow easier emplacement, and recovery if required. Depending on the heat output of the waste, the supercontainers can be placed end-to-end or spaced out along the tunnels, which may be some hundreds of meters long. The annulus around the supercontainer and any spaces between them are filled with a non-compacting backfill such as a mixture of crushed rock and bentonite.	
Main drivers for the Concept	
Difficulties were encountered with use of bentonite during the FEBEX experiment at the Grimsel Test Site (GTS) in Switzerland, which involved the construction of a full-size EBS section of compacted bentonite with an axially placed waste package (substituted by a heater, in the experiment) in a specially excavated tunnel (essentially Concept 3). Construction of the bentonite buffer in the form of pre-formed compacted bentonite blocks, as described for the Project Gewähr / Kristallin-I studies, in high humidity conditions was problematic due to premature swelling of the bentonite causing cracking and disintegration of pre-formed blocks. As a result, a modified Concept including a prefabricated EBS module (also called a supercontainer) was considered as part of a study of alternative disposal Concepts for HLW by NUMO (Japan) for generic sites in recognition that candidate repository sites may be considerably less dry than the GTS. However, unlike the layout envisaged in Concept 5, it was seen to be fundamentally more robust from an operational safety perspective to have a larger working annulus around the supercontainer so that, in the event of an operational incident, recovery or reversal of the emplacement process would be relatively straightforward. The properties of the backfill are undefined and there have been varied suggestions, for example, that it should be 100% bentonite, although less dense than the buffer, but extending the diffusion barrier, or higher permeability material to act as a hydraulic cage around the supercontainer and reduce the potential effects of localized water inflow (this may then also require intermediate tunnel seal sections).	
Important Aspects of the Concept	
The use of the totally sealed supercontainer means that Concept is flexible with respect to implementation in a range of host rocks, as a liner could be used as a temporary (operations phase) hydraulic barrier in wet, fractured rock or for support in weaker rocks. When using a liner, the backfill would be expected to buffer any chemical interactions (e.g. from concrete) to prevent alteration of the bentonite once the supercontainer handling shell corrodes. As the supercontainer is fabricated under controlled conditions (in a hot cell), the density of the bentonite could be greater than normally considered (e.g. greater than 2.0 tonnes/m^3), providing high swelling pressure on saturation to reduce any small unfilled voids in the backfill. The Concept can result in a small excavated volume per waste package, and a proportionately small repository, because the disposal tunnels are used for emplacement of the waste axially and there is no need for the space required to invert/rotate the waste packages into boreholes, nor for additional excavation of boreholes. The Concept is only at the desk study stage, thus there is significant development required in all areas including technology, long-term safety functions of components and safety assessment. The Concept is nominally similar to Concept 5 with the addition of backfill. However, the functions of the backfill in the concept are as yet undefined and there are options to tailor the backfill properties to fulfill functions beyond that of a simple hydraulic barrier (e.g. chemical buffering between the bentonite and a concrete tunnel liner). Uncertainty about the long-term effect of interaction between bentonite and the corrosion products of the handling shell means that the effective thickness of the buffer may be reduced from the nominal emplaced thickness.	

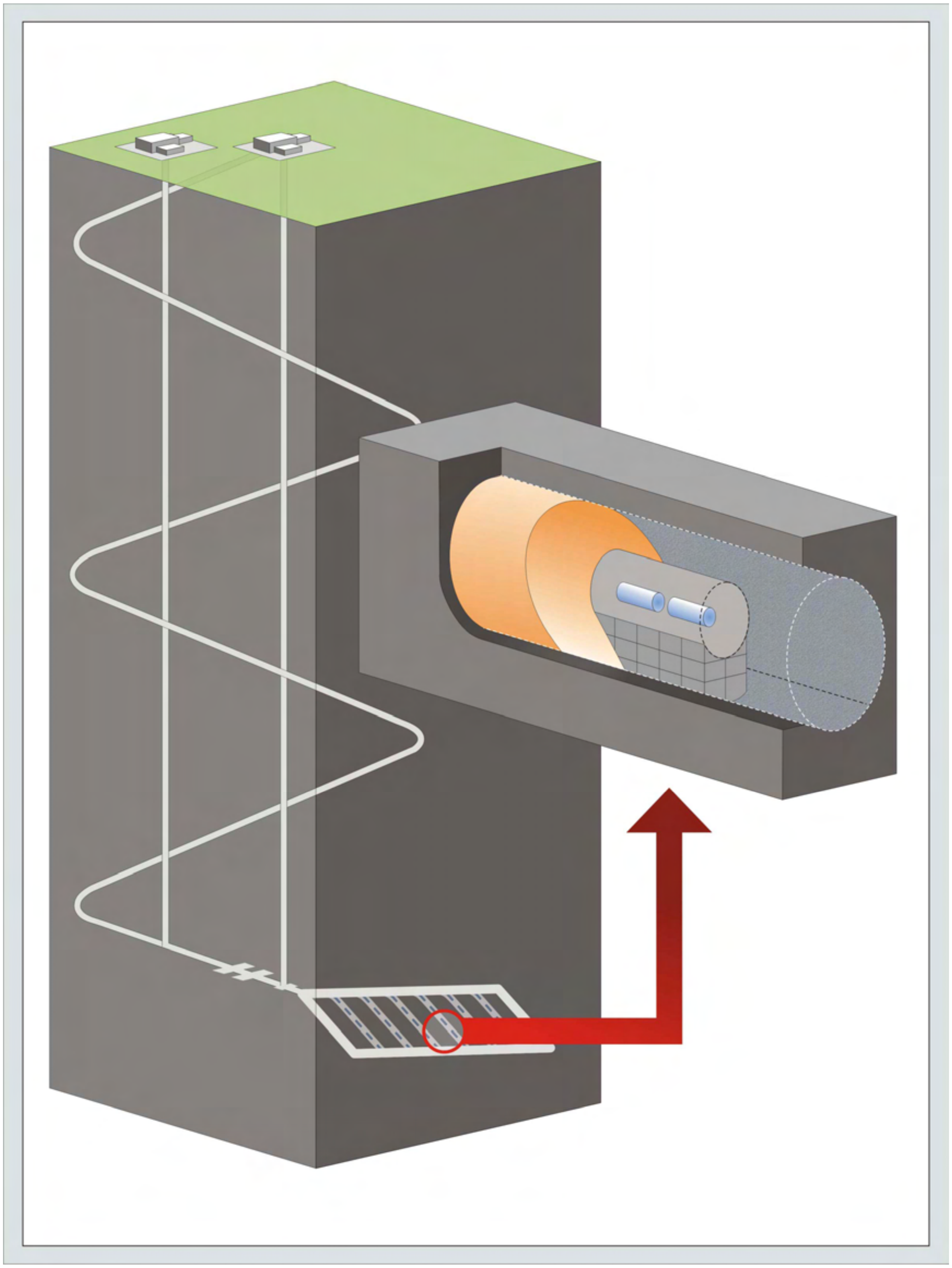

Figure B-7
Schematic illustration of Concept 7, in-tunnel (axial) with supercontainer (large annulus). From Baldwin et al. (2008); used with permission.

Concept 8	Caverns with steel MPC (bentonite backfill)
Main Characteristics of the Concept	
Large, steel multi-purpose transport/storage/disposal containers (MPCs), which can hold up to around 20 HLW flasks or multiple fuel assemblies, are emplaced upright in large ventilated caverns for a period up to 300 years to allow cooling and inspection. The period before backfilling (i.e. until the MPCs are cool enough to backfill without detriment to the bentonite) will depend on the type, amount and age of the waste in the MPCs, but could vary from less than 100 years for old HLW (heat generation of 400W per HLW flask or less) to more than 200 years for UNF or MPCs containing 20 young HLW flasks (>1000 W per HLW flask). After the open period, bentonite backfill materials are emplaced around the containers, cavern seals are emplaced and access tunnels are backfilled for final closure of the repository.	
Main drivers for the Concept	
The general Concept of cavern disposal has been considered feasible for many years but not developed in any detail until recently. The most current development work is in Japan (where the Concept is referred to as the Cavern Retrievable Concept – CARE) driven by (a) concerns that conventional repository Concepts were not appropriate if waste retrieval was required for a significant period and (b) the requirement for the implementing organization to purchase all the land above the repository footprint, which favors smaller footprint Concepts. The cooling of the wastes for a period of up to 300 years means that one of the factors influencing repository layout – namely the thermal output of the waste – is much diminished and the Concept provides a very compact repository footprint even for the large Japanese inventory (the reference inventory for the first repository is 40,000 HLW flasks). The Concept is included in the NUMO alternatives for HLW disposal in Japan.	
Important Aspects of the Concept	
The Concept is flexible with respect to disposal of both UNF and HLW. It is also flexible, in principle, with respect to implementation in a range of host rocks since cavern liners would be used for operational safety grounds in most, if not all, host rocks. The Concept can result in a very small repository footprint because of the packaging of waste into a relatively small number of large containers and the delay to backfilling, which allows the wastes to cool. There is, however, a trade-off between size of footprint and the required storage period as dense packing of the waste using minimum numbers of MPCs means that longer delay to backfilling will be required. Conversely, spreading the waste in more MPCs and over more caverns reduces the maximum temperature in the backfill allowing earlier closure. The Concept allows straightforward retrieval of waste during the open period, as well as inspectability, but with associated security issues. The Concept has only been the subject of limited desk studies to date. Although much of the technology already exists (e.g. transport casks, surface interim stores), many aspects of the long-term safety of the Concept are still uncertain, including the requirements for the backfill and the technology developments for its deployment. The Concept is not flexible with respect to early closure if this is earlier than the originally planned cooling period. If earlier closure were required, alternative backfill materials would need to be investigated, with the concomitant reconsideration of the safety case. The Concept is vulnerable to incomplete closure or loss of institutional control during the long open period.	

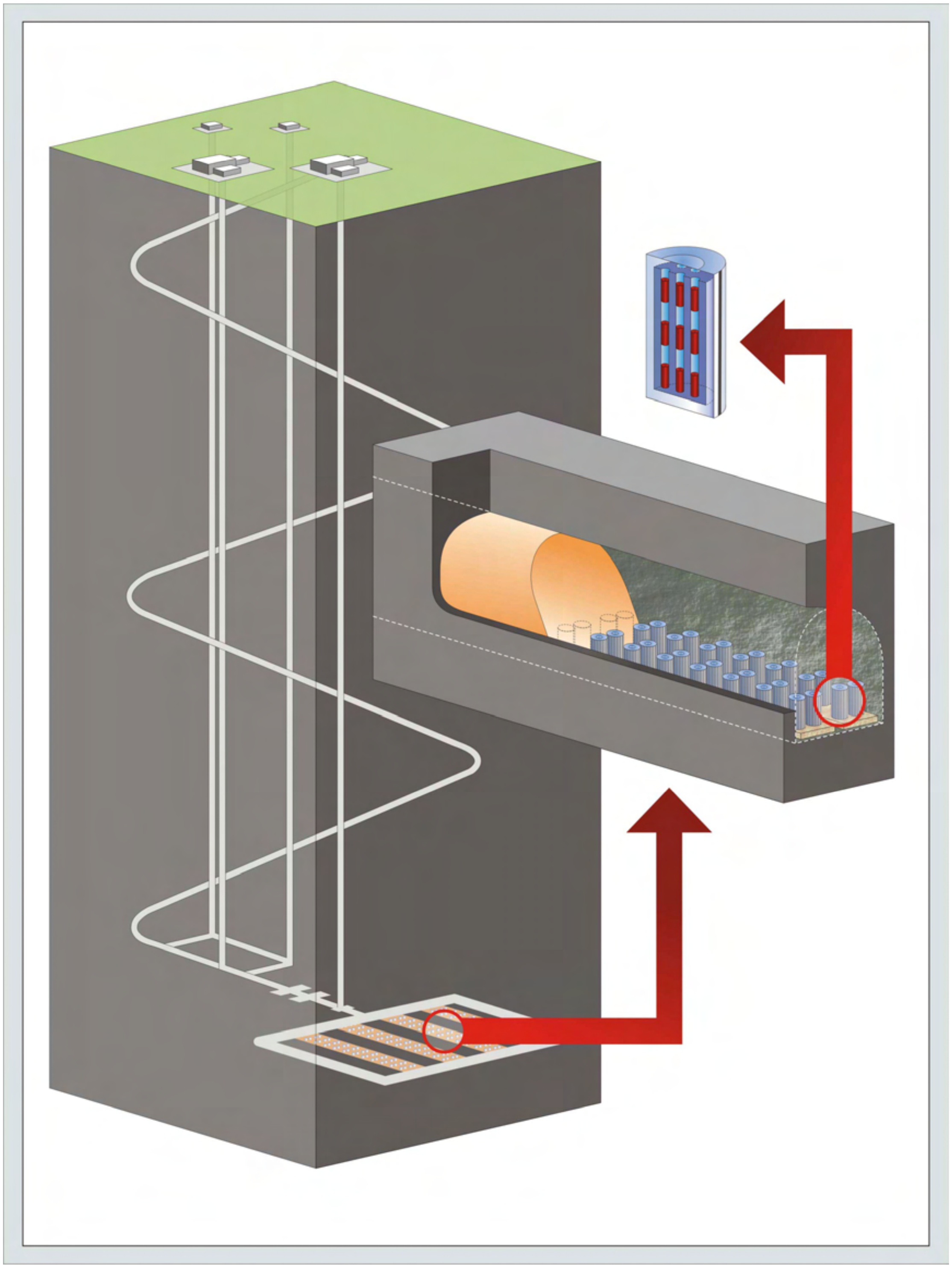

Figure B-8
Schematic illustration of Concept 8, caverns with steel MPC (bentonite backfill). Note that there is scale change on the blow-up diagram: the caverns are of the order of 10-20 m wide and high, compared to the previous tunnel Concepts, where the tunnels were only a few meters in diameter. From Baldwin et al. (2008); used with permission.

Concept 9	Caverns with steel MPC or concrete/DUCRETE CDC (cement backfill)
Main Characteristics of the Concept	
Large, steel multi-purpose transport/storage/disposal containers (MPCs) or concrete disposal casks (CDCs), which can hold up to approximately 20 HLW flasks or multiple fuel assemblies, are emplaced upright in large ventilated caverns for a period up to 300 years to allow cooling and inspection. The period before backfilling (i.e. when the MPCs/CDCs are cool enough to backfill without detriment to the bentonite) will depend on the type, amount and age of the waste in the MPCs, but could vary from less than 100 years for old HLW (heat generation of 400W per HLW flask or less) to more than 200 years for UNF or MPCs containing 20 young HLW flasks (>1000 W per HLW flask). After the open period, cement-based backfill is emplaced around the containers, cavern seals are emplaced and access tunnels are backfilled for final closure of the repository.	
Main drivers for the Concept	
The general Concept of cavern disposal has been considered feasible for many years but not developed in any detail until recently. The most current development work is in Japan (see Concept 7) driven by a) concerns that conventional repository Concepts were not appropriate if waste retrieval was required for a significant period; b) the requirement for NUMO to purchase all the land above the repository footprint, which favors smaller footprint Concepts. Cooling of the wastes for a period of up to 300 years means that one of the factors influencing repository layout – namely the thermal output of the waste – is much diminished and the Concept provides a very compact repository footprint. The original Concept assumed that the backfill material would be based on bentonite. However, it was also recognized that emplacement of bentonite-based backfill would be time consuming as the expected long-term safety performance is dependent on the barrier properties and, thus, on homogeneity and density of the emplaced material. Consequently, for use when rapid closure was desirable, cement-based alternatives have been considered as possibilities, although no studies have been carried out to examine the implications for long-term safety. A variant on this Concept is the use of concrete disposal casks (CDCs), possibly incorporating depleted uranium (if declared a waste). DU-containing concrete ("DUCRETE") fulfils the double role of providing additional radiation shielding compared to conventional concrete (or a smaller, thinner walled CDC) and disposing of unwanted DU and may introduce an additional benefit into the EBS by influencing UNF solubility (although this depends on the form of the DU in the concrete).	
Important Aspects of the Concept	
The Concept is flexible in principle with respect to implementation in a range of host rocks since cavern liners will be used for operational safety grounds in most, if not all, host rocks. The Concept can result in a small repository footprint because of the packaging of waste into few large containers and the delay to backfilling allowing wastes to cool. There is however a trade-off between the size of footprint and the required open period: dense packing of the waste means that backfilling would have to be delayed for longer. Conversely, spreading waste over more caverns reduces the maximum temperature in the backfill allowing earlier closure. The Concept allows straightforward retrieval of waste during the open period, as well as inspectability, but with associated security issues. The Concept has only been the subject of limited desk studies to date. Although much of the technology already exists (e.g. transport casks, surface interim stores), many aspects of the long-term safety of the Concept are still uncertain including the implications of a cement-based backfill. The Concept is not flexible with respect to early closure, although deployment of cement-based backfill could allow rapid closure. The Concept is vulnerable to incomplete closure or loss of institutional control during the long open period.	

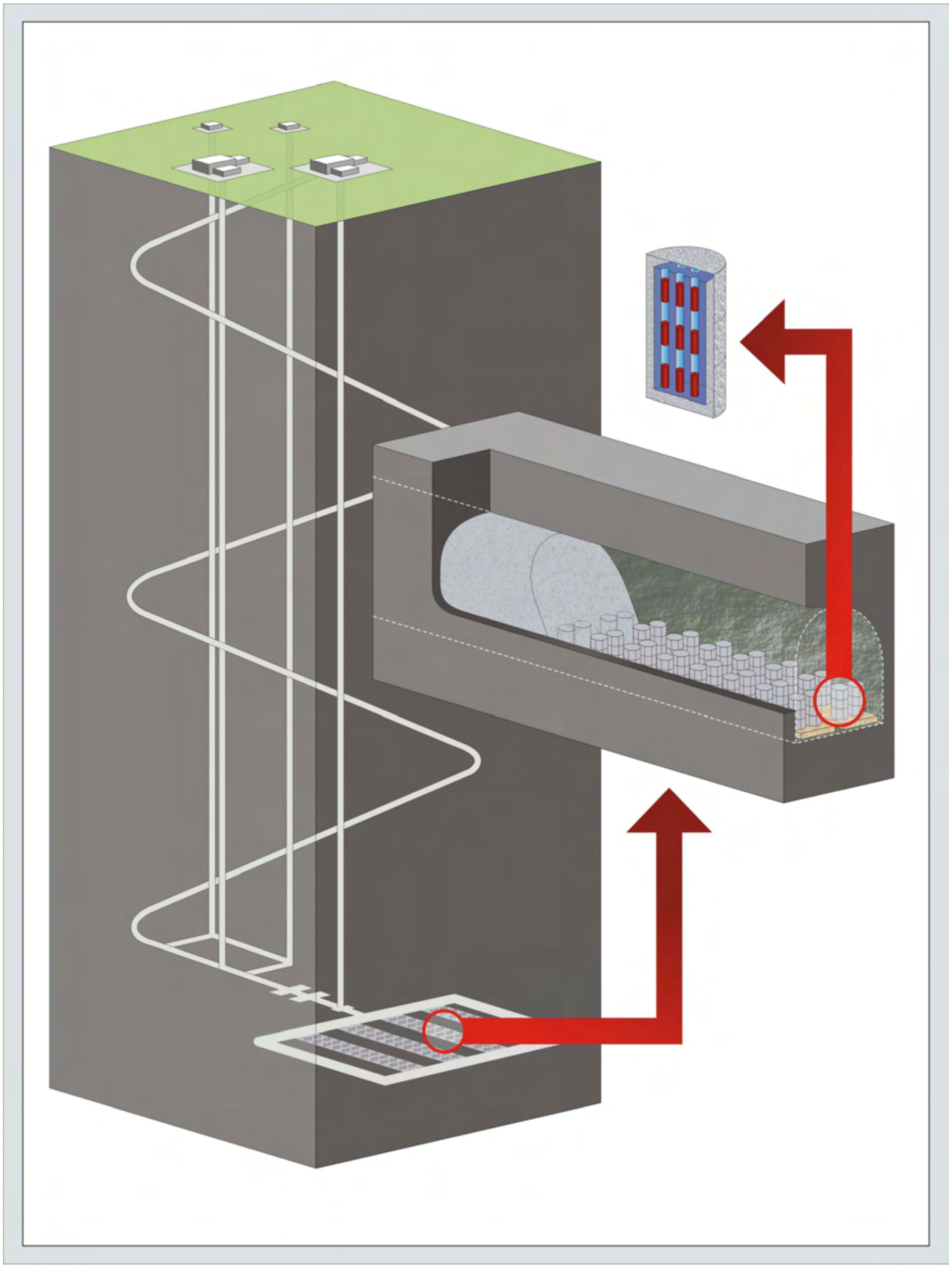

Figure B-9
Schematic illustration of Concept 9, caverns with steel MPC or concrete/DUCRETE CDC (cement backfill). As with the previous illustration, note that there is scale change on the blow-up diagram: the caverns are of the order of 10-20 m wide and high, compared to the previous tunnel Concepts, where the tunnels were only a few meters in diameter. From Baldwin et al. (2008); used with permission.

Concept 10	Mined deep borehole matrix
Main Characteristics of the Concept	
Waste packages are emplaced in stacks in long (~200 m or more) vertical boreholes which are bored from deep underground either directly from a disposal tunnel or between an upper operational cavern and a lower cavern, which is used for excavation of the boreholes only and backfilled and sealed before emplacement begins. In the latter case, raised boring allows excavation of large diameter holes (1.5 - 2 m) over several hundreds of meters. The waste packages may be prefabricated EBS modules (supercontainers) containing waste, overpack and buffer in a handling shell or waste in an overpack/canister around which buffer material may be emplaced.	
Main drivers for the Concept	
The Concept arose in studies of alternative repository designs for both hard rock and salt. AECL (Canada) and NUMO (Japan) both considered disposal of waste packages in moderately deep bore holes without conventional surface excavation operations as a potential option for hard rock. The German reference Concept for disposal in salt domes considered a matrix of 300 m deep boreholes and this was subsequently developed in detail up to 1999 as one of the two design options for the Gorleben repository. The attraction of the Concept is the use of the vertical extent of a host rock and (from the Japanese perspective) the small repository footprint while reducing the surface operations, compared to conventional deep borehole Concepts (see Concept 12) – although, even in the AECL study, the boreholes were only some hundreds of meters long, not kilometers. Also, compared to deep boreholes, the technology for construction and operation of the mined borehole Concept is already standard. The original AECL study concluded that although the Concept was feasible in principle, there were large uncertainties in respect of emplacing the buffer around the waste packages. As a result, the NUMO studies considered the use of supercontainers to simplify emplacement of the EBS. The DBE Technology Concept for salt (Gorleben) remains a favored option and differs significantly from the hard rock Concepts as it involves neither buffer nor overpack.	
Important Aspects of the Concept	
The Concept is flexible with respect to disposal of both UNF and HLW. It is also flexible with respect to implementation in a range of host rocks although, in weaker rocks, limitations on size of openings at depth may limit the practical depth for the lower cavern and thus borehole length. This may be less of a constraint if boreholes are drilled directly from the upper cavern, although this is likely to require a larger cavern. The Concept can result in a small repository footprint, due to the vertical extent of the repository, and also a relatively small excavated volume per waste package, which reduces if longer boreholes can be used and fewer disposal caverns are required. The Concept is immature and there are considerable uncertainties over the long-term safety performance, especially in fractured hard rocks, given the practical difficulty of emplacing backfill/buffer to the sort of quality/properties normally considered acceptable. The relatively close packing of the waste in 3D suggests that thermal convection within the borehole matrix during a possibly extended thermal period could be an issue. The implications on local groundwater flow and geochemistry have not yet been investigated. A major technology requirement is likely to be the development of weight-bearing interim seals which prevent the weight of upper waste packages crushing the lower ones and potentially contaminating the borehole fluids while the borehole is still open (it is assumed that neither waste packages nor supercontainers would be emplaced into a dry borehole and that dense bentonite mud would be used partly for borehole support and partly to control the descent of the waste packages, especially in the case of equipment failure).	

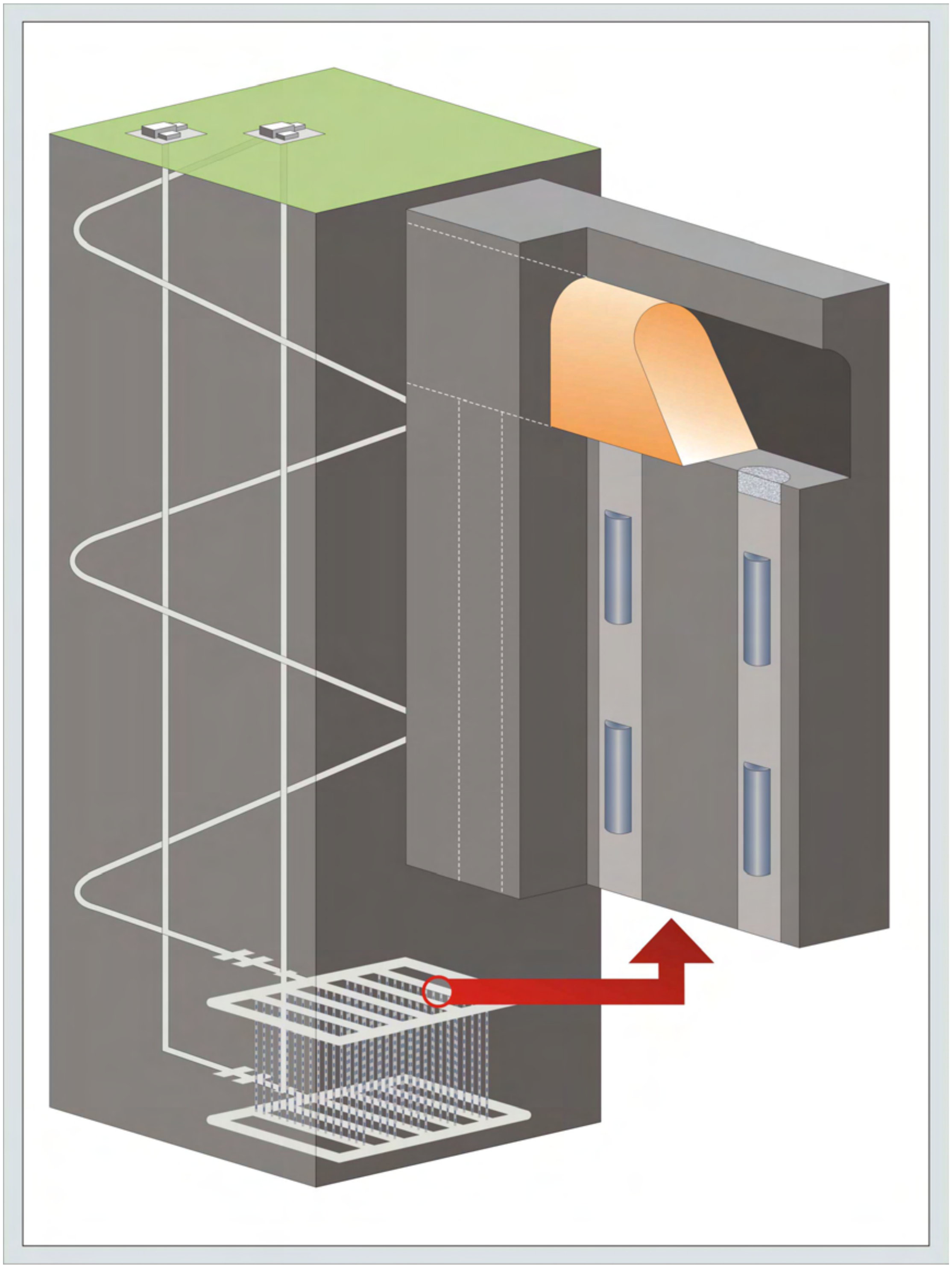

Figure B-10
Schematic illustration of Concept 10, mined deep borehole matrix. From Baldwin et al. (2008); used with permission.

Concept 11	Hydraulic cage (around a cavern repository)
Main Characteristics of the Concept	
The concept of a hydraulic cage around a disposal volume (vault, cavern or whole repository) commonly refers to the use of a zone of material that has a high permeability compared to the average host rock and to the repository volume or the EBS. This zone acts to channel the water away from the disposal volume. The zone may be a simple trench excavated around a cavern and backfilled with high porosity material such as coarse gravel and cobbles or coarsely crushed and graded rock. A more complex option for a larger repository volume is the excavation of a screen of outlying boreholes that intersect the flow field and divert water away from the repository.	
Main drivers for the Concept	
The use of a hydraulic cage in a repository Concept for UNF or HLW was first formalized in the WP Cave Concept by SKB (Sweden), which employed a low permeability bentonite zone, but the general Concept of engineering a high permeability zone which intercepts and diverts water, creating reduced advective flow behind it has mainly been considered for ILW repositories (e.g. in Sweden and Japan). The original WP Cave brought the hydraulic cage Concept together with an unusual repository EBS, but hydraulic cages have been more widely considered as a way of modifying the geological barrier properties around any reasonably compact repository. For sites with higher groundwater fluxes, the reduction of flux through the repository volume by using a hydraulic cage could make a more robust safety case. Although the usual objective is to modify water flux for long-term safety, it could also be possible to use a hydraulic cage to improve conditions during the operational phase. Thus, use of a hydraulic cage in conjunction with the CARE Concept (Concept 8) using ventilation tunnels adjacent to the caverns to create the hydraulic barrier has been suggested in Japan. Here, we describe the use of a hydraulic cage around the CARE concept (Concept 8) caverns. The Concept has not been specifically proposed by any waste management organization for HLW or UNF (although SKB has proposed a hydraulic cage around ILW vaults in the SFL3-5 concept) and is presented as an illustration of how hydraulic cages could be integrated with other Concepts.	
Important Aspects of the Concept	
A hydraulic cage around a compact repository Concept could improve the ease of making a safety case at sites with higher water flows, although this clearly depends on the type of site, host rock and the repository Concept envisaged. The use of a hydraulic cage might be restricted to Concepts with a small repository footprint as engineering a hydraulic cage around an extensive repository could be impractical. A hydraulic cage around a cavern repository (Concept 8) could mitigate some of the consequences of maintaining drained, ventilated openings deep underground on the local groundwater flow by diverting groundwater away from the caverns. The theory of the hydraulic cage is well established but large-scale, practical implementation that would function over the very long timescales of interest is uncertain and currently no programs are actively pursuing this option for HLW or UNF. When engineered around caverns, the excavated openings must be some meters larger than the caverns require, to allow for construction of the cage. Thus, for 15 m-wide CARE-type caverns, the excavated opening would need to be around 20 m. This is feasible in good, hard rock at depths up to 700 m or more. However, in weaker rocks requiring massive tunnel liners for support, the depth of deployment will be restricted and it may be necessary to restrict cavern size, making less efficient use of cavern space.	

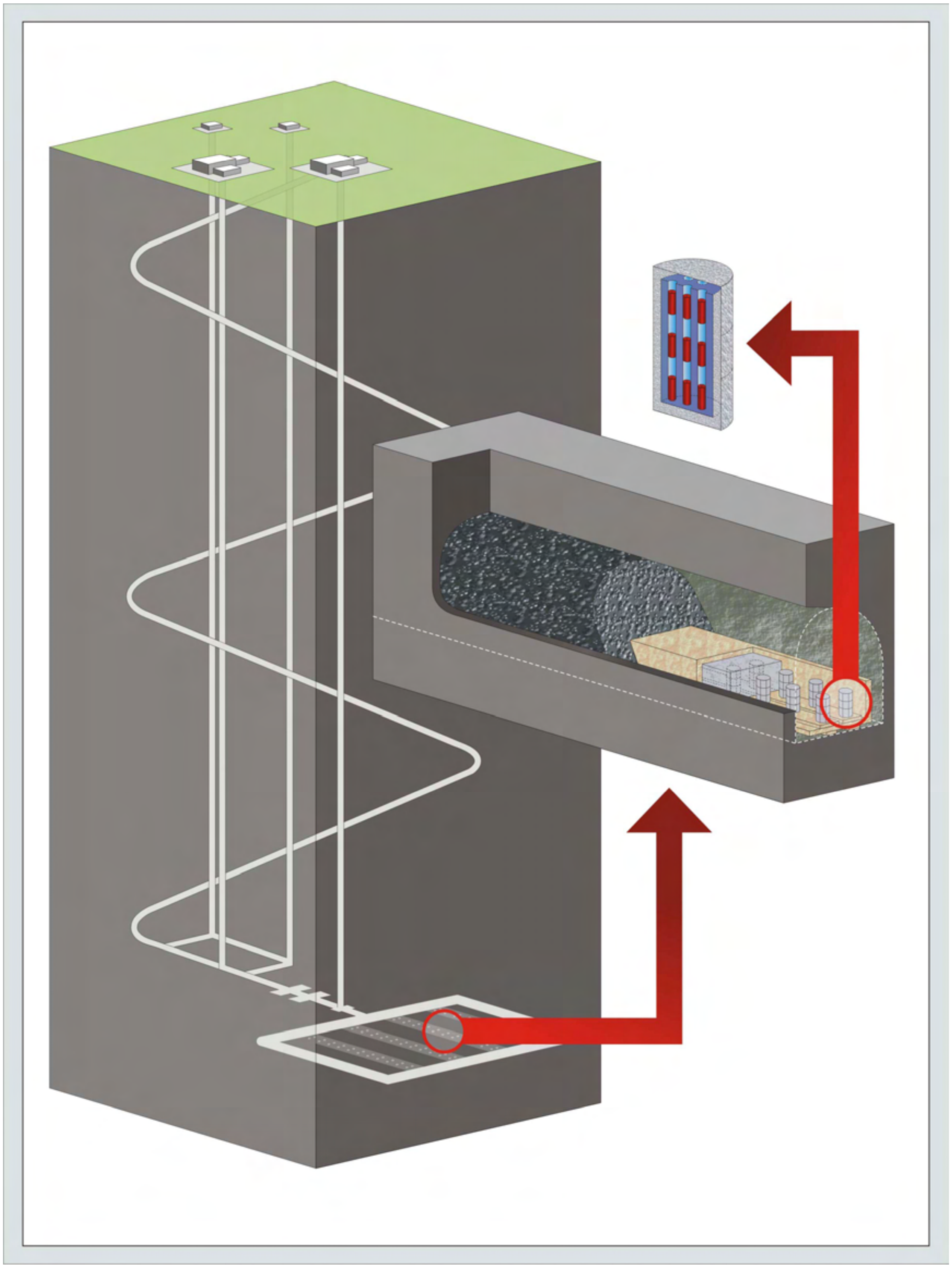

Figure B-11
Schematic illustration of Concept 11, hydraulic cage (around a cavern repository). As with the other cavern concepts, note that there is scale change on the blow-up diagram: the caverns are of the order of 10-20 m wide and high, compared to the previous tunnel Concepts, where the tunnels were only a few meters in diameter. Note that this cavern concept includes a concrete inner vault (brown) that is backfilled with cement (grey). From Baldwin et al. (2008); used with permission.

Concept 12	Very Deep Boreholes
Main Characteristics of the Concept	
Simple metallic waste packages with no overpack are emplaced in the lower region (the bottom 1000 – 2000 m) of a borehole drilled from the surface to a depth of about 3 to 5 km. The borehole is fully lined with metal casing from the surface and should be of sufficient diameter to leave a generous annulus to ensure ease of emplacement. Various options are considered for backfilling the disposal zone of the borehole. The long, upper section of the borehole is sealed. If required, the uppermost part of each borehole could be destroyed, to ensure that wastes are practically irrecoverable. Each disposal borehole is drilled either singly and vertically, from its own drilling pad, or as part of a group from a central location of limited area (probably a few hectares, depending on the amount of waste), using directional drilling technology to deviate the lower sections by a few degrees to ensure the disposal zones in each hole are some tens of meters or more apart.	
Main drivers for the Concept	
The very deep hole Concept dates from the earliest days of geological disposal studies. Many variants have been considered, for depths down to 10 km, including Concepts that involve using the heat from UNF or HLW to melt the rock at great depth and seal-in the waste. Most attention has been on disposal of cooled waste, however. Evaluations have been carried out in Sweden, and Nirex reviewed the history of the Concept in detail. The origin lies in the idea of using great depth to provide very high levels of isolation in what is considered to be a geologically remote and stable environment. Other attractions of very deep boreholes are the relatively limited requirements of site investigations compared to, for example, the detailed fracture characterization required for some conventional repository Concepts, the practical recoverability of the wastes once emplaced, the small surface facilities and structures required and the potential for disposal campaigns carried out as needed, without the necessity to keep open an underground facility. Although there are several very deep scientific boreholes (deepest over 12 km) around the world that show that deep borehole drilling capability per se exists, there have been no practical tests to link large-diameter hole drilling technology with waste disposal technology, no detailed studies of waste handling and operational and post-closure safety, and no integrated disposal Concept has been developed. The Concept has been proposed for UNF, HLW and fissile materials, such as separated Pu. There may be operational issues with UNF disposal (due to the instant release fraction of the UNF) that make this Concept poorly suited to this material. The Concept appears most suited to relatively small volumes of HLW and especially appropriate for fissile materials.	
Important Aspects of the Concept	
The key issue with this Concept is the lack to date of any detailed design or performance assessment study. The Concept is flexible with respect to implementation in a range of host rocks since the key issue is the hydrogeological environment at depth, in particular the lack of active groundwater movement, rather than the properties of the specific rocks at that depth. The Concept can be implemented in disposal ‘campaigns’ with no activities required in interim periods and the surface area required for the excavation (and emplacement) operations may be very small. The Concept provides very secure disposal of waste with effectively little chance of recovering waste without major technological investment. There are uncertainties about the operational procedures for this Concept as most evaluations have focused on the feasibility of borehole excavation and less on the operational safety and practicality: for example, the need to support or isolate waste packages so that the lower ones are not crushed, potentially contaminating the borehole fluid during emplacement operations. The safety case centers on the isolation provided by the deep geosphere but no detailed, comprehensive safety assessment has yet been performed. The size of the waste package for practical implementation in the near future (that is, without major development of ~1 m diameter boreholes) means that, although suitable for HLW, very little UNF, for example, only one PWR fuel assembly, could be contained in one waste package. Thus, the Concept could be inefficient for UNF disposal, as well as potentially poorly suited for the operational safety reasons mentioned above.	

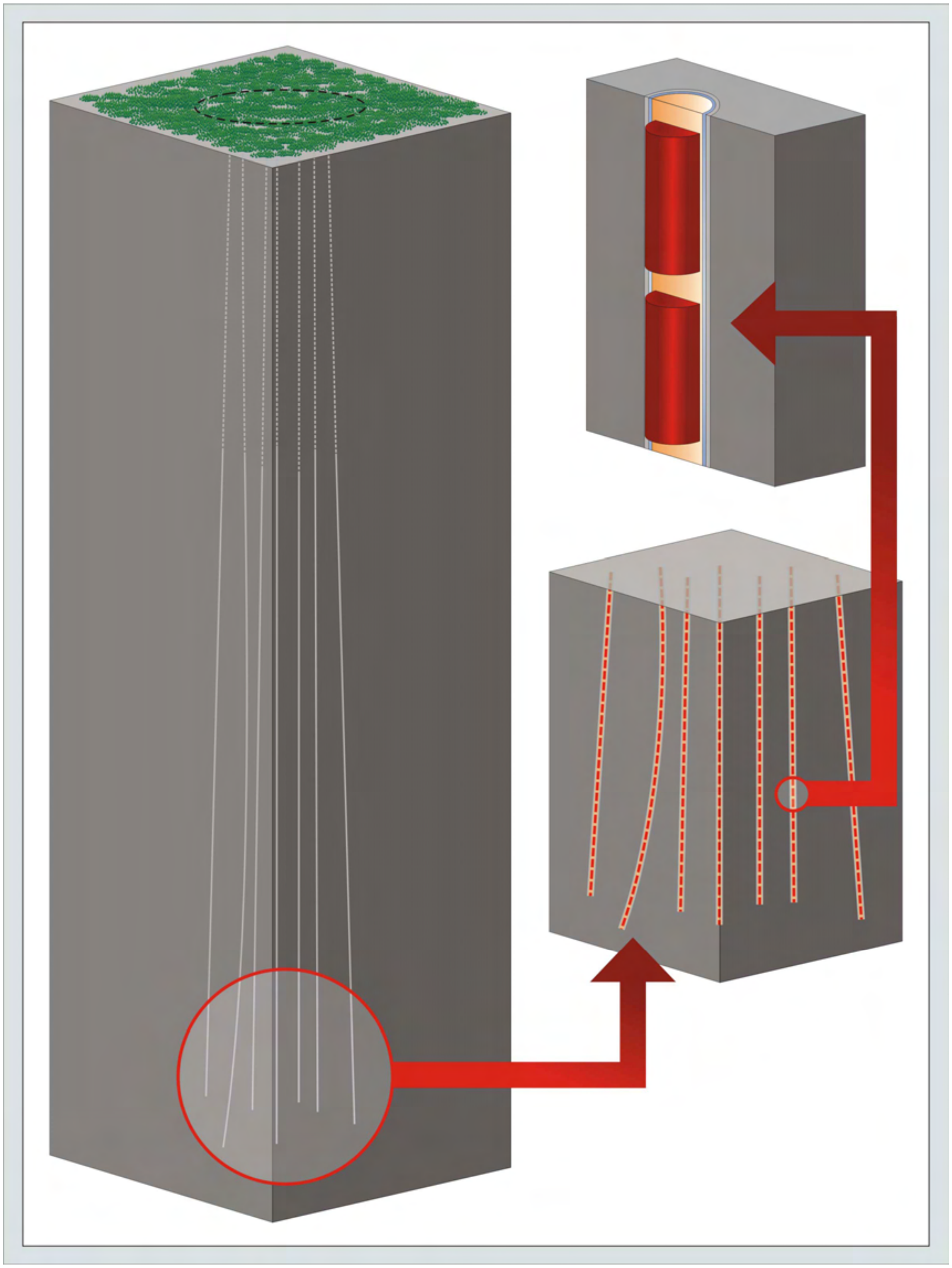

Figure B-12
Schematic illustration of Concept 12, very deep boreholes. The scale on the main image is vastly different to the previous example concept illustrations; with depth in the range of 3 to 5 km, rather than around 500 m (the green surface represents a forest). The black circle indicates the area over which borehole operations will be carried out. From Baldwin et al. (2008); used with permission.

B.3 Lessons Learned from Existing Repository Concepts

This Appendix, bringing together international experience to show the range of approaches that could be adopted in what may be a future 'new start' in the US, can prove valuable in several ways:

- Solutions for geological disposal of highly radioactive materials exist, often in well-developed form, for many different geological environments.
- Using the common mode of presenting disposal Concepts and options in this Appendix provides a useful tool for initiating discussions with decision makers and national review groups such as the Blue Ribbon Commission on America's Nuclear Future.
- The schematic diagrams in this Appendix also offer a ready method of showing potentially interested host communities what a repository might look like for the type of geological environment(s) that are available at the possible host area.
- It confirms the fact that the US, in concert with these other international programs developing repository concepts, will require a repository for UNF, HLW (or both), along with disposal options for other sources long-lived wastes arising from nuclear power generation and waste processing.
- Depending on the direction and timing that a potential re-start of a siting program in the US might take, well-tested, analyzed and documented repository concepts exist that can aid in rapid and cost-effective repository development and deployment.
- Certain design concepts, such as the very deep boreholes (Concept 12) may present both special advantages and special challenges that will require further investigation before they can be considered as proven and ready for deployment.

Perhaps the two most important 'lessons learned' can be drawn from assembling the various repository design concepts in this Appendix.

- ***Flexibility in design.*** Given siting uncertainties, it is important to maintain as much flexibility as possible in all aspects of a geological repository program and not to commit to specific designs until it becomes necessary. Indeed, it is highly likely that the initial license approved repository design will be modified, improved and optimized during the multi-decade emplacement and operational phase of any repository implementation.
- ***Wide number of options.*** There already exists a wide range of design options available (including many minor variants) that can cater for almost any 'generally suitable' geological environment. 'Generally suitable' means environments that, from the first principles of siting repositories (see EPRI Report II in this series for the siting principles applied in the US site selection program of the 1980's), clearly offer some potential as hosts.

In these two ways, designs can be found that will match to most generally adequate geological conditions and the degree to which the designs will have to be adapted and optimized will be highly site-specific.

The latter conclusion is critically important for a volunteer siting approach, where a location might be advanced that displays geological conditions that deviate somewhat from what might be regarded by some reviews as ‘standard’. It is based upon the positions that (a) it is never feasible or necessary to identify the ‘safest’ (e.g. geologically ‘best’) location in any country[48] and (b) matching an engineered barrier system (EBS) to the geological conditions of the natural barrier system (NBS) has such a degree of adaptability and flexibility built in that, for generally adequate geological conditions, a total system can be achieved that will satisfy the requirements of a safety case. While a more technically led siting program, which endeavors to identify sites on their technical merits first and then overcome the societal issues, might provide a location with more robust geological properties than a volunteer process, it is a highly uncertain process and the technical outcome may be no more attractive than one provided by careful EBS-NBS matching, to meet regulatory safety requirements.

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[48] Although it is clearly possible to identify the best site from among a group of contenders by considering all their attributes, technical and non-technical, and weighting these to provide the best match to the requirements of the main stakeholders.

C
ALTERNATIVE REPOSITORY CONCEPTS OFFERING FLEXIBLE, SAFE AND SECURE MANAGEMENT OF USED NUCLEAR FUEL AND HLW

C.1 Introduction

The Blue Ribbon Commission (BRC) on America's Nuclear Future is charged with reassessing management options for America's nuclear waste, dominated by existing and future commercial used nuclear fuel, existing defense HLW, and any wastes resulting from possible future reprocessing of commercial used fuel. Consideration is likely to focus on conventional storage and disposal options, coupled with evaluation of the advantages and disadvantages of reprocessing. The latter linkage to reprocessing is particularly tricky as it involves highly uncertain extrapolations of nuclear power generation, reactor and reprocessing technology, price of uranium and other fissile materials, trends in public acceptance, etc. Indeed, these factors are not only strongly coupled, they are also back-coupled to success in expansion of nuclear power generation and implementing acceptable storage/disposal options. Increased acceptance of the need for new nuclear power plants is often tied to increased concern about the management of resulting wastes.

When expressed in terms of fundamental requirements on conventional storage and disposal concepts, major conflicts arise between:

- Desire to show progress towards implementation of disposal of waste and desire to allow flexibility for possible future reprocessing of used fuel
- Higher safety / security of geological disposal (especially with regard to possible terrorist attacks) and long timescales of implementation
- Generally better public acceptance of long-term storage and the acknowledged fact that storage is not a final solution of the waste problem
- Perceived reduction in the volume of waste for storage and disposal following reprocessing and the fact that this considers only high-level waste (HLW) and not the much more problematic intermediate level waste (ILW) waste streams, for which there are currently no specific recognition in US regulations.

Such conflicts mean that there is no clear "best" management option: pros and cons of different variants need to be carefully assessed to optimize trade-offs between their strengths and weaknesses. The need to consider a wide spectrum of options is well recognized, particularly in countries with a history of commitment to reprocessing of commercial spent fuel (e.g. France, UK, Japan, Switzerland) and has led to projects to develop and assess novel management concepts.

Two innovative concepts affecting the flexibility, safety and feasibility of geological disposal are discussed in this Appendix. It is recognized that there are other new developments, but these two concepts may be of particular interest for current BRC review. These innovative repository concepts enable consideration and fulfillment of diverse, multiple national priorities, including complying with nuclear safeguard requirements, an adaptive staging of waste management activities, optimizing thermal management of nuclear waste to be disposed, addressing any key technical or operational topics through extended confirmatory testing or design adaption, minimizing or obviating the need for multiple disposal sites, and allowing maximum accommodation for public-policy debate and decisions on national waste management strategies.

C.2 Cavern Retrievable (CARE) Concept

The CARE concept (McKinley et al., 2008) was explicitly developed to resolve conflicts arising from conventional management strategies for spent fuel, recognizing that these include both technical and socio-political aspects of project implementation. In effect, it consists of an underground cavern storage option (in the parlance of the US, a "monitored retrievable storage", or MRS, option) that is dimensioned for operation over a timescale of centuries (300 years is often taken as a reference), but allows for the option of the facility being readily transformed into a final repository whenever the decision is made to do so. Figure C-1 shows three key stages of phases for a CARE facility of an initial Emplacement Phase, followed by an extended Storage Phase, and concluding with a final Disposal Phase (Steps A, B and C in Figure C-1).

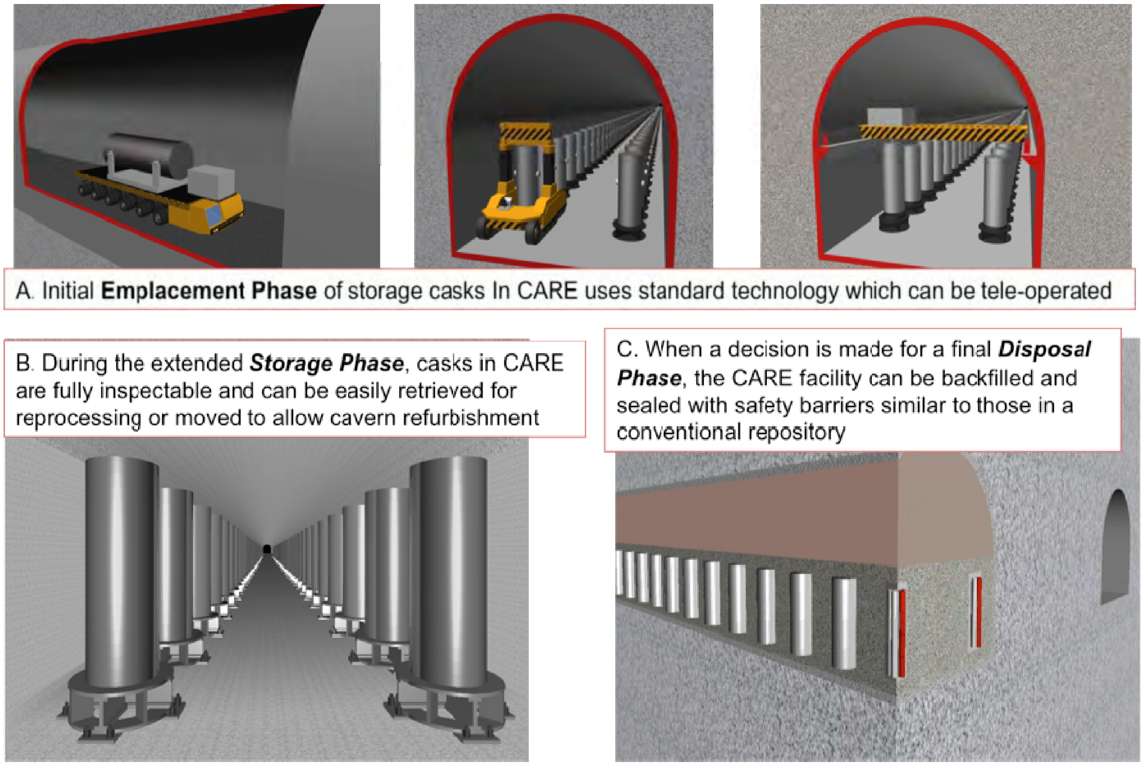

Figure C-1
The Monitored Retrieval Storage CARE concept, showing three distinct phases that allow for enhanced flexibility and linkage among storage, reprocessing, safeguards, and final disposal activities.

C.2.1 Advantages of the CARE Concept

Some features of note for the CARE concept include:

Initial dry storage in large caverns constructed at depths of several hundred meters below surface in a suitably stable geological formation. The technology for assessing site suitability and for construction of such facilities is well established and, given political support at national and local levels, could be implemented relatively rapidly. Indeed, the existing site characterization performed and documented in the Yucca Mountain site license application would clearly address this need.

Although variants are possible, analysis to date has considered storage of used fuel and/or solidified HLW from reprocessing in large multi-purpose transportation / storage / disposal casks. These provide high levels of operator protection during transport, storage and any other handling operations. During the storage phase, waste is fully inspectable and some or all can be easily retrieved at any time. Furthermore, casks already licensed and used in the US for dry cask storage could be adapted to CARE.

A single facility has many safety and security advantages compared to distributed storage of used fuel at reactor sites, while storage deep underground provides effectively complete protection from surface perturbations, including potential terrorist attacks.

Even if the facility is implemented predominantly for used fuel, if a future decision to reprocess is made, used fuel can be easily removed and the resulting HLW and "2nd generation" used fuel (e.g. MOX) returned to the facility in reconditioned or modified casks.

Minimum operation times of the facility are expected to lie within the range of 50 - 100 years. During this period, considerable advances are to be expected in understanding of the site, the science and technology of engineered barriers and the capabilities of performance assessment methodology, which will be reflected in an improved safety concept for transfer to the disposal phase.

An extended storage period of several hundreds of years under ventilated conditions would lead to significant removal of radiogenic heat (dominated by the decay heat of isotopes with half lives of 30 years or less), enabling a much closer emplacement of waste packages at the time of final disposal, minimizing and possibly obviating the need for additional repository sites.

It must be noted that this implied timescale of several hundreds of years is not unprecedented: many near-surface disposal concepts include institutional control periods of 300 – 500 years and large contaminated sites may require active site management "for the indefinite future". With respect to the ground support system to maintain the physical integrity of the tunnels walls, the Yucca Mountain License Application, for example, already envisions the ability to refurbish ground support as needed during the multi-decade period of waste emplacement prior to repository closure.

When desired, potentially after a period of several 100's years, a CARE facility can be readily and at low cost transformed from a storage site into a final repository. This would require backfilling of the open void space, using materials and emplacement techniques already well developed for more conventional single-step repository concepts (see Appendix B).

A key point is that a CARE facility already entails a large proportion of the costs necessary for constructing a repository. By constructing a CARE facility with nuclear waste funds now (from the generation that benefited from the nuclear power), there will be far less costs imposed on a future generation when it makes a decision on closure of a CARE facility as a final repository.

While international development and analyses of the CARE concept have focused on implementation in saturated host rock sites, the CARE concept could also be readily adapted to the Yucca Mountain site, for either the vertical-emplacement configuration shown in Figure C-1 or in a horizontal, axial-emplacement configuration closely matching the Yucca Mountain License Application reference design. For adaptation to the unsaturated Yucca Mountain site, backfilling of the eventual nuclear waste for final disposal would entail emplacement of a high-isolation (so-called "Richards barrier') capillary-breaking backfill around the disposal packages.

Preliminary generic analysis suggests that disposal within massive casks surrounded by a suitable buffer / backfill (e.g. based on compacted bentonite) can form the basis of a very robust safety case, even with existing levels of science and technology.

If, in the unlikely event that a safety case as a geological repository cannot be made for a CARE facility after extended storage, the facility could be run further in a storage mode until waste can be transferred to a disposal site elsewhere or another final management option is implemented.

Public acceptance may likely be a much greater challenge for disposal than for storage. In the CARE option, the storage period includes continuous effort to build acceptance through:

- Building familiarity with the underground facility and demonstration of its safety, beneficial spin-offs for the local community and low environmental impact;
- Establishing long-term (10's to 100's of years) demonstration projects to show directly the performance of proposed disposal options;
- Establishing local research infrastructure that maintains an overview of the applicability of developing science and technology to revised and updated disposal concepts; and
- Active involvement of the local community in decision-making, continually building acceptance of the project.

Compared to current single-step repository disposal concepts, CARE is presently vague in terms of implementation details for the disposal phase, as these are expected to evolve with time and develop in consensus with key stakeholders. Indeed, the CARE concept envisions stakeholders and decision makers many generations in the future, so it is possibly counter-productive to unduly restrict options for a far-future generation.

The CARE concept is compatible with modern, holistic approaches to management of radioactive and other toxic wastes, which are of growing interest in several countries. Indeed, given the slow progress in development of fusion power, it may be prudent to leave open the option of long-term, large-scale fission power, possibly with associated introduction of breeder reactors, which may be facilitated by integrating used fuel management with that of depleted uranium.

C.2.2 Siting of a CARE Facility

This CARE disposal option arose from assessment of alternative designs for deep geological disposal of HLW in Japan, which involves a volunteer approach to siting in a country with active tectonics and complex geology. Although there are published exclusion criteria to assure that completely unsuitable sites are not considered, the geological barrier is not assumed to be as powerful as in many other national disposal concepts. Nevertheless, extensive civil engineering experience in Japan shows that construction and operation can be carried out safely and cost-effectively in a very wide range of settings. In terms of post-closure safety, the geological barrier is initially assumed to physically isolate and protect the engineered barrier system (EBS), but not provide a high level of assured resistance to radionuclide release and transport. Key to the safety case, therefore, is performance of a "robust" EBS, which is assured by the use of large quantities of well-understood materials that provide geochemical buffering in a diffusion-dominated solute transport system.

The limited constraints on the host rock – effectively avoiding active faults, volcanoes, natural resources and regions of rapid uplift – would allow such a facility to be sited in almost any region of the US and site suitability to be confirmed relatively rapidly using existing characterization technology. Given the large number of potential sites, practical constraints such as public acceptance and ease of access / waste transportation can be given high weighting in the siting process. Given the cost and political sensitivity of waste transportation and the potential for implementing such a concept quickly and cheaply, the option of parallel siting of 2 or 3 facilities might be considered (analogous to the compacts for disposal of LLW).

The fixed features of the CARE design are associated with essential infrastructure for maintaining caverns open and accessible for periods of up to several hundred years. The layout of the emplacement tunnels will be determined to a considerable extent by the properties of the host rock. The cavern emplacement option offers considerable benefits in terms of reducing the repository footprint (or increasing the inventory that can be emplaced in a specific site) in cases when this would be of value, with the consequence of increased thermal loading as delayed backfilling considerably reduces the significance of thermal impacts.

C.2.3 Cask Design for CARE

The multi-use transport/storage/disposal cask envisioned for CARE is currently based on a CASTOR design. There are already decades of experience in fabricating, handling, transporting and storing such casks. These are extremely robust and are routinely tested with drop, crash, fire and water immersion tests which go beyond anything which would be expected in CARE operational scenarios during storage or in preparation for conversion to a final repository for waste disposal.

Corrosion during dry storage is expected to be negligible compared to the wall thickness and is likely to be concentrated in areas where contact with water or other materials occurs – predominantly on the base, which can easily accept the maximum possible degradation (<< 1cm) as this is trivial in comparison to the total base thickness. Regular inspection and refurbishment would ensure that quality of the cask meets specifications for disposal. The main concerns involve how a cask based on this basic design would perform under post-closure conditions and, in particular, if such performance could be degraded by processes occurring during the long period of storage.

C.2.4 Operations in a CARE Facility

After construction, the emplacement of waste is, in the reference case, expected to extend over a period of around 50-100 years (see Figure C-1). This could involve remote-operated procedures using rail transportation although, from a radiation protection viewpoint, there would be no difficulty in assuring shielding to allow manually controlled handling. To minimize thermal loading, waste emplacement can be staggered both within and between the caverns. Over the following 100-300 years, say, waste packages can be regularly removed for inspection and to allow the emplacement caverns to be refurbished – requiring an extra buffer storage cavern for the expected case that refurbishment will be carried out only on completely empty caverns. With a 50-year inspection / refurbishment cycle for the entire waste inventory, work activity levels will be similar to those during emplacement, allowing relatively constant manpower levels to be maintained. This provides both a valuable commitment to the local community and also assures that the experienced workforce needed for final repository closure would be available. Further, as the facility would be envisaged to be the focus for R&D in disposal technology, the materials and processes used will be continually improved to correspond to the evolving state-of-the-art.

Ventilation is critical during the operational period and could be assured by a combination of active and passive systems (the efficiency of the latter being very dependent on the actual setting but could be significant in the case of disposal in hilly or mountainous topography).

C.2.5 Timing and Impact of Buffer and Backfill Emplacement

The potential need for adding backfilling a CARE facility for final disposal has been considered for two particular cases; the reference case of sealing after the thermal output of the waste has decreased significantly (after around 300 years) and an early closure case where the decision is made to close the repository, despite the high residual thermal loading. For both these cases, various sealing materials and approaches have been considered from the viewpoints of easing quality assurance (QA), assuring the robustness of the post-closure safety case and minimizing costs.

Based on studies that have previously been carried out for prefabricated EBS modules (see Section C.3), advantages have been shown for multiple layers of buffer / backfill which have complementary roles. There are also cost aspects involved. In the following, the term buffer will be reserved for components of the system that play a critical role in defining the performance of the enclosed waste, while backfill will be used for other material used to fill residual space.

A potential buffer design includes a layer of silica sand around the cask, which is, in turn, enclosed in a thick layer of a compacted mixture of sand and bentonite. The buffer is emplaced on a “floor” of sand or gravel and void spaces above these primary engineered barriers are backfilled with finely crushed rock. Such a concept effectively provides a diffusion-dominated system within more conductive zones, which may function as a hydraulic cage and which should be inherently robust to local perturbations. The hydraulic cage effect can be enhanced by backfilling the overlying ventilation tunnels with course gravel, after plugging the connecting holes to the emplacement cavern. Nevertheless, to avoid risks of short-circuits, high-quality sealing of the ventilation shafts and any drainage from the ventilation tunnel would be needed. Typically, temperatures would remain below 100°C throughout the EBS for planned closure after 300 years.

In the early sealing case, a major concern is the influence of high temperatures within the EBS and surrounding host rock. The buffer is assumed to be composed of the same materials but the thickness of the bentonite / sand layer is increased. As long as the storage period is in the order of 100 years, maximum temperatures should remain below about 150°C, which should not cause problems of stability for a bentonite based backfill. For closure after shorter times – in the most severe case immediately after emplacement caverns are filled (in the order of 50 years after operations commenced), peak temperatures would lie above the presently accepted stability limits of bentonite and hence alternative buffer materials would be used. Provisionally, designs using vermiculites have being considered, but more work is needed to select the optimal material for this purpose.

C.2.6 Post-Closure Performance of a CARE Repository

After sealing of the emplacement caverns, the lining materials will gradually degrade and the buffer and backfill materials will saturate with inflowing water. Although very dependent on the rate of supply of water from the host rock, this process is likely to initially be rather localized and total saturation may well take many decades to centuries. Indeed, the saturation process may be hindered by the build up of an internal pressure of hydrogen, resulting from the anoxic corrosion of the steel casks.

When evaluating buffer and backfill designs, it is important to consider the impact on long-term safety of the use of high quality cavern linings constructed from conventional concrete materials, which are expected to be required both for operational safety and minimization of local hydrogeological or geochemical perturbations over the extended open period, even when cavern excavation is in good rock. Potentially detrimental interactions between the concrete and any clay-based buffer and backfill materials such as bentonite will be mitigated, to an extent, by carbonation of the concrete during the open period, especially as the ventilation required for temperature control within the caverns will provide a continuous supply of atmospheric carbon dioxide.

As yet, the models used to assess post-closure performance are simplistic but, if anything, probably underestimate the power of such an EBS to limit releases. The massive local inventory of steel corresponds to a major “redox trap” which, on the basis of natural analogue studies, could well be expected to reduce releases of most (or all) key radionuclides to insignificant levels for as long as reducing agents remain in the system – probably hundreds of thousands of years for expected repository settings.

C.2.7 Assessing Advantages Compared to Other Options

As noted in the Introduction, different back-end management options have particular strengths and weaknesses and these must be weighted and balanced against each other in order to find an optimal solution for specific program boundary conditions. This type of evaluation – usually involving some form of multi-attribute analysis – has been carried out in many national programs. Additionally, various approaches have been used to illustrate and highlight pros and cons of management options.

As a simple example, Figure C-2 indicates a qualitative, subjective view of the strengths of CARE (higher scores lie closer to the outer circle) compared to the original YMP concept, a deep borehole disposal option, and a policy of continued, long-term surface storage at reactor sites.

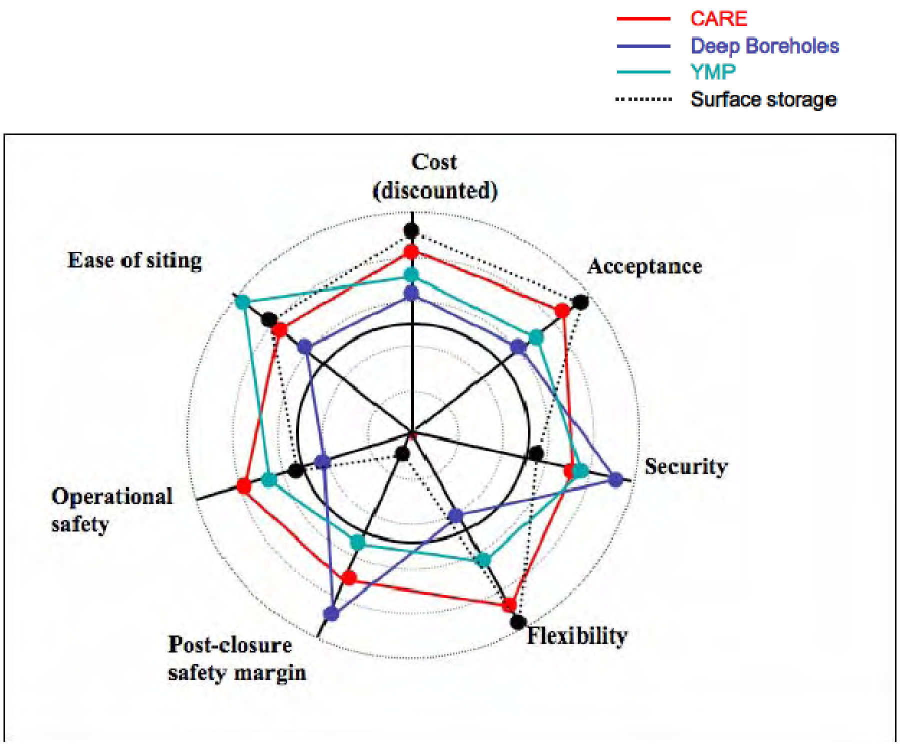

Figure C-2
Qualitative comparison of key attributes of CARE concept, deep boreholes, and the US Yucca Mountain program (YMP) disposal concept.

C.2.8 Lessons Learned about the CARE Concept

The CARE concept offers several advantages for HLW / used fuel management relative to other approaches, particularly if flexibility to respond to uncertainties in the evolution of the future nuclear power management options is to be emphasized. The basic CARE concept has been examined in some detail for the specific boundary conditions in other national programs for saturated host-rock sites. Some additional effort would be required to tailor variants of CARE to US boundary conditions, either as an adaptation for the Yucca Mountain site in unsaturated tuff or as deployed in some other site and type of host rock. Nevertheless, the investment involved would be minor compared to the total program cost or, indeed, the potential management flexibility and life-cycle cost savings that could be realized for the CARE concept.

C.3 Prefabrication Emplacement Modules (PEMs)

Another important innovative concept affecting disposal of radioactive wastes is the prefabricate emplacement module, or PEM (Apted, 1998; JNC, 2000). The basic concept, as illustrated in Figure C-3, is to create a unified system of engineered barriers and nuclear waste into a single package that can be fabricated, sealed, quality assured and even temporarily stored in surface or sub-surface (i.e., CARE) storage facilities until the time of final emplacement and closure of a geological repository. The PEM concept was initially referred to as an “inside-out” waste package, because the buffer barrier in conventional repository designs (see Appendix B) is emplaced outside of the canister, whereas for the PEM the buffer is emplaced inside of a thin-walled outer container.

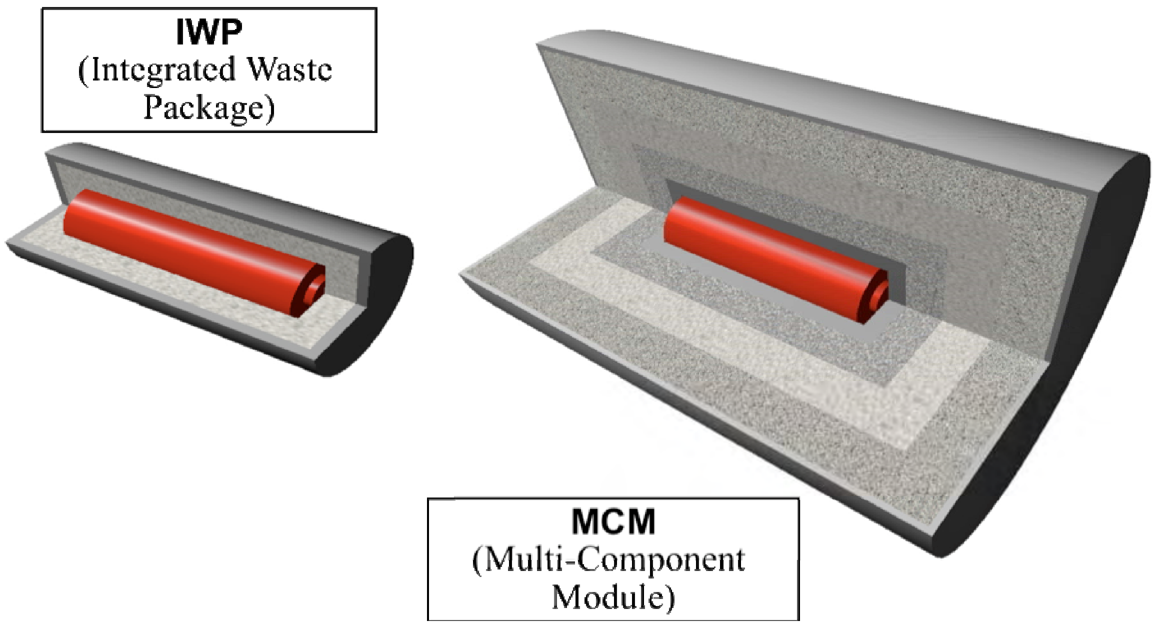

Figure C-3
The PEM concept, illustrated by two basic forms; a simple integrated waste package with a minimum of container and buffer barriers, and a multi-component module with additional engineered barriers, to possibly address additional levels of assurance for safe handling (i.e., self-shielding) and performance.

C.3.1 Advantages of the PEM Concept

There are a number of potential advantages to the PEM concept for disposal of HLW and used fuel. These include:

- Placement of the mild steel container outside the buffer, rather than inside, precludes any potential disruption of the buffer from hydrogen gas generated during anaerobic corrosion of the steel.
- The formation of iron corrosion products at the interface with the host rock will act to prevent any possible piping or chemical erosion of the buffer into factures within the host rock.
- Placing the dense metal container outside the buffer assures there is no possibility of long-term sinking of the canister, which has been postulated for conventional repository design (see Appendix B).
- Locating the mild steel container at the surface of the tunnel will result in extremely rapid return to favorably reducing conditions for a saturated repository.

- The relatively rapid (several thousands of years) general corrosion of the mild steel container will promote a uniform re-saturation and swelling of the buffer once the container is breached.
- There are several styles of PEMs (Figure C-3), ranging from a simple IWP container with an inside buffer, to the more complex, multiple barrier MCM. The application of specific PEMs depends on the type of waste for disposal, the type of host rock, and the margins for safety required to meet national safety standards.
- The entire PEM is loaded in a surface facility with the nuclear waste form (e.g., a HLW stainless steel pour vessel), dry buffer material is hot-isostatically pressed around the waste form, and the container is welded. In the same surface facility, the entire waste package, including the weldment and the buffer, can be fully inspected and quality assured before emplacement. Such full quality assurance cannot be as readily established for conventional repository concepts (see Appendix B), especially with respect to the handling of compacted bentonite buffer and backfill materials that swell readily and rapidly in humid mine environments of a repository.
- Because of the prefabricated nature of the PEM, the rate of emplacement of PEMs into underground deposition sites can occur much more rapidly than is currently documented for traditional repository concepts. For example, emplacement rates of waste packages for a KBS-3 type repository range from 1 package every 2 days to 1 package a week; preliminary analyses for the PEM indicate emplacement rates easily as high as 10-20 packages a day.
- Safety assessment analyses show that PEMs are expected to achieve as good or slightly better performance per emplaced waste package compared to conventional repository concepts, although there is no requirement for any additional R&D because the PEM uses the same materials as envisioned for the conventional repository concepts.
- Taken together, PEMs are projected to be a much more cost-effective approach to final disposal, given savings in materials, fabrication, quality assurance, amount of rock to be excavated, and labor for transportation and emplacement.

C.3.2 Lessons Learned about the PEM Concept

The PEM concept was originally conceived for the Japanese repository program, which by national law is required to implement a repository concept capable of averaging the emplacement of 10 HLW waste packages per day of operation. Interest in the PEM expanded when additional benefits on issues such as feasibility of emplacement, quality assurance, hydrogen gas evolution, disruption of buffer safety functions and cost were subsequently evaluated. Further analyses on the PEM concept are being conducted in Europe and Asia, and there are potential advantages for a fuller consideration of PEMs if there is a restart to the US repository program.

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D
SAFETY CASE CONCEPT AS DEVELOPED IN THE UNITED KINGDOM

D.1 Introduction

The International Atomic Energy Agency (IAEA), the Nuclear Energy Agency (NEA), the NRC, the EPA, and international repository and regulatory programs endorse performance or safety assessments of repository systems as an integral part of licensing. Outside of the US, the "Safety Case" is widely applied. The US National Academy of Sciences (NAS, 2003) has asserted that the current US regulations and the international Safety Case are essentially equivalent. Two key aspects of the Safety Case include the extensive use of supporting arguments (such as industrial and natural analogue data) and alternative safety indicators (e.g., comparison of repository releases to the flux of natural radionuclides from a site). The latter aspect is especially useful as application of dose-models based on present-day human habits and activities becomes increasingly uncertain at time scales beyond several hundreds of years. An illustrative example of how the Safety Case concept and approach have been implemented in the UK is provided here, focusing on:

- History / background of the term "Safety Case" and its use;
- Outline of the main components of a Safety Case; and
- Appropriate examples of Safety Case, both nuclear and non-nuclear (e.g., railways).

D.2 Safety Case - History / Background

The Safety Case concept was developed initially for the nuclear power industry in the UK in the 1960s. The Nuclear Installations Act of 1965 required that all nuclear plants obtain a license from the UK Health and Safety Executive (HSE) prior to operation. One of the standard license conditions required the production and maintenance of an adequate safety case, the primary objective being to reduce the probability of a major incident occurring. In the late 1970s, the Nuclear Installation Inspectorate (NII) division of HSE developed Safety Assessment Principles for use by its own inspectors when making judgments on a licensee's safety cases and these principles were published so that all stakeholders were aware of the standards to be met.

Following the loss of the North Sea oil platform Piper Alpha which claimed 167 lives in 1988, the Safety Case was introduced into this industry by the Cullen Enquiry to bring safety requirements for North Sea (offshore) operations to a higher standard. Unfortunately, prior to 1988, the offshore industry had been insulated from the development of safety cases as a tool for managing major accident hazards. Since then, however, the approach has also been adopted by the offshore petroleum industry in Europe, Australasia and elsewhere (Sutton, 2004).

The major environmental disaster at Seveso, Italy in 1976 led to a European Directive - the so-called Seveso Directive, which was significantly influenced by developments in the UK on this topic.

There is nothing specific within the safety case methodology that limits its application to any particular industry or location. Thus, Safety Cases have been prepared for a wide range of industries in the UK:

- Chemical plants;
- British railway system; and
- British naval operations.

Although we have not found a formal definition of safety case issued by UK HSE, Bishop and Bloomfield (1995) provide a general definition of the term in the UK as:

> "A documented body of evidence that provides a convincing and valid argument that a system is adequately safe for a given application in a given environment".

Compare this definition with that of IAEA regarding safety case in the context of the post-closure phase of a repository (IAEA/NEA, 2004):

> "An integration of arguments and evidence that describe, quantify and substantiate the safety, and the level of confidence in the safety, of a geological disposal facility ".

The intended benefits of safety cases in general include:

- An improved understanding of the hazards and risks of a system;
- An enhanced knowledge of the technical an managerial controls required to manage the hazards/risks; and
- Better oversight by the regulator.

Together, these benefits lead to a reduction in the number and consequences of major accidents.

D.3 Components of a Safety Case

D.3.1 Basic Philosophy Concerning Safety in the UK

In the UK, the responsibility for safety is placed on the body which controls the hazard and hence the risk - the operator, in an organizational rather than personal, sense. The role of the regulator is to ensure that this responsibility is carried out adequately and that risks are being properly controlled. For nuclear facilities, regulation is the responsibility of the Health and Safety Executive's Nuclear Installations Inspectorate (HSE/NII). The requirement to produce a safety case is embedded in the standard conditions that are attached to all licenses, through powers in the Nuclear Installations Act.

D.3.2 Principal Features of Safety Case Systems

The main requirement of all safety case regimes is that via their safety case, operators must demonstrate that they have identified and assessed all the relevant risks and that they have taken all the steps necessary to minimize these risks. The key general features of all UK safety case regimes are (Wilkinson, 2002):

- Safety cases must be prepared by the operator of an installation.
- The safety case must identify the safety critical aspects of the installation, both technical and managerial.
- Appropriate "performance standards" must be defined for the operation of the safety critical aspects. A "performance standard" is a statement which can be expressed in qualitative or quantitative terms, of the performance that is required of a system, which is used as the basis for managing the 'hazard' throughout the life cycle of the installation. Thus, performance standards are key to a safety case system.
- Generally, the workforce must be involved.
- The safety case is reviewed by a competent and independent regulator who may prohibit the activity if there are serious shortcomings in the safety case submitted.

The principle here is that those who create the risk must manage it, and it is the operator's job to assess relevant processes, identify and evaluate the risks, and implement appropriate controls.

D.3.3 Safety Case and Nuclear Installations

In the case of nuclear installations, a potential operator, as part of the licensing process, must produce a safety case that includes the identification of operating limits and conditions, safety mechanisms and maintenance requirements. HSE requires licensees to undertake strategic planning for radioactive waste management, including the development of programs for the disposal of waste accumulated at nuclear sites within a suitable timeframe. The strategy, together with safety case(s) for associated facilities, is intended to demonstrate that the facilities are adequately safe.

Clearly, the above aspect of a safety case is appropriate for the operational stage of a facility and not for the post-closure phase of a disposal facility. Although HSE is responsible for all aspects of the regulation of waste management on nuclear-licensed sites, the regulation of disposal of radioactive waste under the Radioactive Substances Act of 1993 (RSA 1993) is the responsibility of the Environment Agencies - the Environment Agency in England and Wales, and the Scottish Environmental Protection Agency in Scotland.

The post-closure safety case is widely (internationally) considered to be the primary focus of regulatory review for assessing the long-term safety of the disposal system and establishing limits on its operation, e.g., in terms of waste acceptability and disposal limits. In the UK, the regulatory authorities may place equal or greater emphasis on the qualitative aspects of the safety case, including a demonstration of the ALARA principle.

Typically, a safety case is refined at each stage in the decision-making process (from site characterization through site closure). In the case of a radioactive waste disposal facility, this refinement approach provides a strong link to site characterization and selection, operations, and overall waste management.

D.4 Examples of Safety Case Application in the UK

Two examples are provided in this section, although there are a number of others to draw upon. The examples described below are:

- Drigg Post-Closure Safety Case
- Railways Safety Case Regulations.

D.4.1 Post-Closure Safety Case for BNFL's Drigg LLW Facility

As discussed in Section D.3.2, the regulation of disposal of radioactive waste under the Radioactive Substances Act of 1993 (RSA 1993) is the responsibility of the Environment Agency in England and Wales and the Scottish Environmental Protection Agency in Scotland. Close cooperation between HSE and the Environment Agencies is ensured via statutory consultation under the terms of the Environment Act of 1995.

The Certificate of Authorization for the Drigg site (DOE/MAFF, 1988) requires BNFL to use best practicable means to limit the migration of radionuclides from the waste. As part of its periodic review process of authorizations for nuclear licensed sites, the Environment Agency (hereafter referred to as the Agency) developed a methodology for the planned review of BNFL's Post-Closure Safety Case (PCSC) for the Drigg site (Streatfield et al., 2002).

According to this approach, approval of the Safety Case relies on an ongoing evaluation lasting several years. For example, the Agency's review program has progressed in 3 phases:

- Phase I (12/96 to 3/99): Review and assessment of interim working documents from BNFL's PCSC development Program.
- Phase II (1/99 to 9/02): Continuation of the review and assessment of interim working documents, as well as assessment of BNFL's responses to issues raised in Phase I, and review of BNFL's Status Report (BNFL, 2000).
- Phase III (9/02 onwards): Review of the Drigg PCSC and review of authorization for disposal of low-level radioactive waste at Drigg.

Therefore, the PCSC is not simply a submission of safety assessment results, but a continuous interaction process involving document submission and review. During Phase I, for example, 21 documents from BNFL's PCSC Development Program, and an additional 7 supporting documents, were reviewed.

The Agency's extended review process had several objectives (Streatfield et al., 2002):

- To determine whether the developing PCSC is likely to be consistent with current legislation and guidance, in order to provide a suitable basis for regulatory decision making.
- To determine whether the developing PCSC is supported by BNFL's existing program of site investigation and characterization, research, safety analyses and monitoring.
- To identify regulatory and technical issues that may affect the post-closure safety assessment.
- To identify the need for additional work by the Agency, such as audits.
- To identify the need for any additional work by BNFL for the preparation of the PCSC to meet regulatory expectations.

The Agency emphasizes that regulatory decisions concerning Drigg (and in general) will not be made on a quantitative risk calculation alone, i.e., the PCSC is expected to include multiple and complementary lines of reasoning against the principles and requirements set out in the 1997 GRA guidance document (Environment Agency, 1997). Note that those GRA criteria were not prescriptive. Thus, it was the applicant's responsibility to justify the information provided as being appropriate and adequate.

The GRA require, for example, demonstration of the use of good engineering practice in design construction and operation of a radioactive waste disposal facility. A requirement also existed for the application of good science in:

- investigating the suitability of a site,
- supporting R&D work,
- interpreting the resulting data; and in
- developing safety assessment methodologies.

The key issues associated with the review of documents from BNFL's PCSC Development Program, from the Agency's perspective and in terms of the PCSC, were:

- Sufficiency: sufficiency of data to support the safety case.
- Gaps in R&D: whether gaps exist in BNFL's research and assessment program; and
- Overall Approach: appropriateness of approach for assessing post-closure safety.

Following reviews conducted during Phases I and II, the Agency commented to BNFL (Streatfield et al., 2002, 2003) on specific areas of the program, as a means of improving its Safety Case:

- Regulatory interpretation
- Comprehensiveness and integration of BNFL's approach
- Optimization
- Models and flow data
- Treatment of uncertainty; and
- Traceability of assumptions and data.

D.4.2 The Railways (Safety Case) Regulations

The UK's railway system, one of the most dense traffic systems in the world, was privatized in 1994. One of the aims of privatization was to comply with EU Directives to separate the infrastructure management from train operations. Thus, the infrastructure became the responsibility of Railtrack (now Network Rail) and its numerous private contractors, while train operations were divided among 26 passenger and 3 freight Train Operating Companies (TOCs).

Safety Case regulations evolved alongside the privatization process. A study by the UK HSE recognized the significance of the critical technical and operating interfaces that would exist between such a large number of different organizations operating on the network. Accordingly, HSE recommended adopting a Safety Case approach for identifying and controlling those risks. This resulted in the creation of the Railways Safety Case Regulations (RSC) in 1994. These Regulations were based on interconnected safety links, from HSE through Railtrack to the TOCs. There was a clearly-defined process for acceptance of Railway Safety Cases, with the HSE accepting the Railtrack Safety Case, and Railtrack in turn accepting safety cases from the TOCs. Railtrack was also required to ensure TOC compliance with their safety cases by conducting an annual audit of each TOC, and monitoring their safety performance.

Because of the interconnected safety links, Railtrack had a quasi-regulatory role, which acted as a barrier to effective cooperation with the TOCs, who were Railtrack's major customers. Railtrack was required by its license to separate the safety issues from commercial issues by creating an independent directorate, the Safety and Standards Directorate, which oversaw the production of standards and also audited TOCs. However, it was always difficult to persuade external stakeholders that the Safety and Standards Directorate was completely independent of the rest of Railtrack.

Following a number of Accident Inquiries, the RSC Regulations 2000 were introduced. This resulted in the separation of the Directorate from Railtrack, becoming an independent company called Railway Safety. The new Regulations also effectively removed the interconnected safety links by requiring the HSE to accept all Railway Safety Cases, with Railtrack and Railway Safety independently assessing them and making recommendations to the HSE.

The new regulations also required a fundamental change in the content of safety cases. Up to that point, safety cases had been required to describe the risks, and measures to reduce them to as low as reasonably practicable. The regulations also required a development plan, committing each company to further improvements in safety management. Each operator was also required to review its safety cases to reflect these regulations.

The new Railtrack Railway Safety Case describes the risks and their controls, and considers the effectiveness of these control measures. It also commits to taking action to further improve them. This is linked closely with Railtracks annual Safety & Environment Plan, and with accident recommendations and other inputs, to provide a process of continual improvement. Thus, the safety case is central to the whole process of continuous review and improvement of railway safety.

D.5 Lessons Learned about the Safety Case Approach

While the Safety Case approach as applied initially in the nuclear industry may have differed from its current usage in the UK, the term as used at the present time in the context of a post-closure safety case for geologic disposal in the UK, appears essentially the same as that adopted internationally and by IAEA.

A Safety Case should be developed as a top-level document, and include the articulation of a strategy to achieve safety as distinct from the strategy for demonstrating compliance. The emphasis should be placed on obtaining and communicating understanding and facilitating dialogue with the relevant stakeholders. A Safety Case is the integration of relevant arguments in support of the long-term safety of the repository. In particular, a statement of confidence should be included, to describe the approach adopted to achieve sufficient confidence, and to acknowledge the remaining issues, together with a suggested strategy for resolving those issues.

D.6 References

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