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## R&D orientations for Technical Safety Organizations


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# 1 Content

# Abbreviations

| | |
|---|---|
| ALARA | as low as reasonably achievable |
| CARP | Coordinated Assessment and Research Plan |
| CNCS | Canadian Nuclear Safety Commission |
| DGD | deep geological disposal |
| EBS | engineered barrier system |
| ELI | Ministerie Van economische Zaken, Landbouw en Innovatie, Netherlands |
| ENPC | Ecole Nationale des Ponts et Chaussées (ENPC |
| FANC | Federal Agency for Nuclear Control, FANC, Belgium |
| FEP | Features, Events, Processes |
| HLW | high level waste |
| GD | geological disposal |
| GRS | Gesellschaft für Anlagen-und-Reaktorsicherheit, GRS, Germany |
| IAEA | International Atomic Energy Agency |
| ICRP | International Commission on Radiological Protection |
| IFSTTAR | French institute of science and technology for transport, development and networks |
| IRSN | Institut de Radioprotection et de Sûreté Nucléaire, France |
| IRSN | Institut de Radioprotection et de Sûreté Nucléaire, IRSN, France |
| LEI | Lietuvos Energetikos Institutas, Lithuania |
| MESA | Methods for Safety Assessment for Geological Disposal Facilities for Radioactive Waste |
| NEA | Nuclear Energy Agency |
| NWMO | nuclear waste management organisations |
| OECD | Organisation for Economic Co-operation and Development |
| OPG | Ontario Power Generation |
| PA | performance assessment |
| PAMINA | project Performance assessment methodologies in application to guide the development of the safety case |
| RAWRA | Radioactive Waste Repository Authority |
| R&D | research and development |
| SCK.CEN | Belgium Nuclear Research Center |
| SC | safety case |
| SF | spent fuel |
| SSG | Specific Safety Guide |
| SSR | Specific Safety Requirements |
| SUJB | Czech regulatory body |
| TES | Tournemire Experimental Station |
| THMC | thermo-mechanical-chemical |
| TSO | technical support organization |
| URL | underground research laboratory |
| WIPP | Waste Isolation Pilot Plant |

# 2 Foreword

The objective of the FP7 program SITEX project is to set up a network capable of harmonizing European approaches to safety review of geological disposals for radioactive waste. Lasting 24 months, SITEX brings together 15 organisations representing technical safety organisations (TSOs) and safety authorities, as well as civil society outreach specialists. In particular, SITEX plans to help establishing the conditions required for developing a sustainable network of technical safety experts who have their own scientific skills and analytical tools, independently of the operators, and who are capable of conducting their own research programs.

# 3 Summary

This report describes the common vision of technical safety organizations on the scientific and technical knowledge needed by experts for supporting the regulatory review of the safety case and assessing properly the key safety questions that will arise from the development of Deep Geological Disposal (DGD) project. In the framework of the Work Package 3 (WP3) of the SITEX project, each participating organization has determined:

- the key safety issues which characterize their national radioactive waste DGD project;
- according to these key safety questions, the associated scientific and technical knowledge which is considered as of major importance and priority in order to support the necessary skills needed for properly reviewing the DGD safety case.

Based on these national contributions, WP3 participants have first compiled the needs for knowledge for reviewing long term safety developed along three main issues:

- the quality of the data on which rest the safety demonstration provided by the waste management organisation;
- the understanding of the complex processes which may potentially influence the safety of the DGD;
- the assessment of the future evolution(s) (in spatial extent and intensity) of these processes, as well as the assessment of their impact on the DGD safety.

A specific attention to challenges in operational safety review has been paid in order to highlight needs for better characterisation of nuclear and conventional hazards possibly occurring in an underground nuclear facility.

# 4 Introduction

Nowadays, the long-term safety of DGD is an international key topic of research, addressed by different actors developing research activities, and notablyexisting and future DGD operators, TSOs and regulators. This broad topic of research encompasses the

characterization of site properties, the understanding of complex processes which may have an influence on the future evolution of DGD (e.g. Thermal, Hydrological, Mechanical and Chemical processes…) and the assessment of the influence of these processes on the evolution of the performances of the different components, on which the overall safety of the DGD relies.
Waste management organisations are generally mandated to develop a comprehensive research program aimed at supporting the development of the concept and of the safety demonstration. In order to be able to assess the technical and scientific arguments provided by WMOs in the safety case, reviewers have to develop and maintain high level of expertise in the scientific areas on which WMOs base the safety demonstration. It is the reason why a number of TSOs and regulatory bodies develop their own research program independently from the programs developed by WMOs.
This is notably required by the IAEA [1] who consider as one of governmental responsibilities to ensure that necessary scientific and technical expertise remain available both for operator and for the support of independent regulatory reviews and other national review functions. The independency of experts and/or independent technical safety organizations (TSOs) reviewing safety cases is therefore very important for ensuring the credibility of safety cases for all stakeholders, particularly from candidate sites. The benefit of such program is twofold. On the one hand it is an efficient way to develop and maintain high level skills up to date independently from the operators. The independent experts must have a deep understanding of complex and sometimes very specific processes occurring in a repository. For a good understanding it is not sufficient only to review the results of R&D activities of operators without own experience in a given subject. This can be only acquired by close involving in research activities in organizations that have good scientific capabilities to conduct such research activities. On the other hand, it allows addressing key safety questions and especially where gaps in knowledge may be identified, and how some phenomena and their related uncertainties can play a role on the confidence in the results provided by WMOs.

The main purpose of WP3 is to propose a framework for the development and maintenance of scientific capabilities of TSOs or duly appointed bodies to perform independent technical assessments and to encourage joined research and scientific and technical exchanges between TSOs and regulatory authorities.
The main aim of this deliverable (D3.1) is to identify areas where scientific and technical knowledge is needed for reviewing the safety case. This report is divided into three main chapters (5, 6 and 7). Chapter 5 shortly summarises DGD concepts and the state of development of DGD programmes of participating countries. Chapter 6 presents a list of the R&D actions that are or should be undertaken in order to support the technical review of safety cases and the key safety questions that are attached to these actions. Finally, chapter 7 presents a first broad vision of the participants of the possible mapping between the R&D actions identified in chapter 6 and the Strategic Research Agenda (SRA) developed by the Implementing Geological Disposal – Technical Platform (IGD-TP, operator network).

The following organizations participate in the WP3 of the SITEX project:

- Bel V, Belgium
- Canadian Nuclear Safety Commission, CNSC, Canada
- DECOM SA, Slovakia
- Federal Agency for Nuclear Control, FANC, Belgium
- Gesellschaft für Anlagen-und-Reaktorsicherheit, GRS, Germany
- Institut de Radioprotection et de Sûreté Nucléaire, IRSN, France
- Lietuvos Energetikos Institutas, LEI, Lithuania
- Ministerie Van economische Zaken, Landbouw en Innovatie, ELI, Netherlands
- UJV Rez, a.s. Czech Republic

These countries develop disposal programmes with very different state of maturity (from conceptual studies to selection of site and technological demonstration tests). This discrepancy impacts the nature of the work performed (generic work, calculations, experimentations…) and the scientific areas that are investigated by the SITEX partners in order to review or to prepare the future review of geological disposal safety caseMore detailed information about country programmes are reported in Appendix 1 of this deliverable.

# 5 Identification of common views on key safety issues and related scientific knowledge needed

In the framework of the SITEX/WP3, each participating organization has determined:

- the key safety issues which characterize their national radioactive waste Geological Disposal (GD) project;
- the associated R&D actions that should be undertaken in order to ensure a high level technical review of the safety case developed by the operator.

Based on these national contributions, SITEX participants have developed a common view on the key safety issues and the related needs for knowledge in order to perform high quality review. This analysis focuses on long term safety. Questions arising from operational safety itself are addressed as well, in particular where needs for better knowledge is identified. Depending of organisations, these needs for knowledge are already derived into R&D actions (see D3.2 deliverable of WP3) or used as basis to identify further R&D actions, possibly undertaken through joined programming activities. The scientific and technical knowledge needed can be categorized into the following issues :

- (A) the quality of the data on which rest the safety demonstration;
- (B) the understanding of the complex processes which may potentially influence the long term safety of the DGD;
- (C) the assessment of the future evolution (in spatial extent and intensity) of these potential processes, as well as the assessment of their impact on the DGD safety;

- (D) the identification and characterisation of the potential hazards to occur during the construction and operation of the DGD.

Such categorization of scientific knowledge needed is derived from the approach that is followed (see the work developed by WP4 on the review methodology) when reviewing the safety case. As a matter of fact, experts review in particular these aspects of the safety case demonstration : the capability of the implementer to properly justify the methods used to obtain data and the confidence in the data; the capability of the implementer to explain the processes that govern the performance of the components and their ability to achieve the safety functions; the capability of the implementer to perform the long term evolution of the disposal taking into account the influence of the uncertainties on the different potential evolutions as well as perturbations and unexpected events; the due account of potential hazards that could impair safe operation of the waste emplacement, considering the influence of accidents during operational phase on the long term safety.

Each category is described in the following 4 subsections. For each subsection, a set of scientific and technical knowledge needed (e.g. considered as necessary for performing the review) is identified. In the follow-up of SITEX project, these issues will be case by case analysed and, where appropriate, derived into research activities for the expertise function.

## 5.1 QUALITY OF INPUT DATA

The quality of the long-term and operational safety demonstrations that will be provided by the operator for the DGD facility notably rests on the quality of the data which characterize the GD system. The data evaluated for the safety demonstration include among others:

- **Data on the radioactive source term**

  E.g.: the identification of radionuclides and their individual activity, quantification of IRF (instant released fraction of radionuclides)…

- **Data on waste form properties**

  E.g.: the identification of leaching rates for individual radionuclides…

- **Data on canister corrosion development**

  E.g.: the identification of corrosion rates and potential defect occurrence…

- **Data on the thermal development and properties of the near-field and the host-rock**

  E.g.: heat conduction capacity of the host-rock, heat stability of the buffer and sealing material…

- **Data on the mechanical properties of the near-field and the host-rock**

  E.g.: the mechanical properties and stability of buffer and sealing material

- **Data on the chemical and migration properties of the near-field and the host-rock**

  E.g.: the sorption capacity of the buffer and host-rock, their diffusion properties…

- **Data on the hydrogeological properties of the near-field and the host-rock**

  E.g.: the host-rock permeability, seal performance, the presence of faults/joints in the host-rock…

As discussed in the following subsections, the quality of the data evaluated by the operator for the operational and long-term safety demonstrations principally depends on their accuracy and relevance, as well as on their representativeness of the in situ properties of the GD system.

### 5.1.1 Methodology adequacy and relevance

The accuracy and the relevance of the data used for the safety demonstration of the GD facility principally depend on the methodology followed by the operator to evaluate these data. Where necessary, TSOs and safety authorities undertake R&D actions to develop a better knowledge in the capacity and accuracy of these methods to provide accurate and relevant input data for the safety demonstration. On the basis of such R&D, assessors are able to identify the domain of validity of the investigated methods with respect to the type of data measured and the conditions of the experiment. Accordingly, appraisal of the implementer's work is done and potential questions may be asked to the implementer to clarify or justify its data acquisition strategy and its confidence in the data produced.

**Needs for knowledge:**

**A1** The adequacy and relevance of methods available for the evaluation of data necessary for long-term and operational safety demonstrations.

**Particular points of attention (not exhaustive list):**

- Method to characterize the source term;
- Method to characterize DGD thermal development;
- Method to characterize container corrosion rates;
- Method to characterize the geo-mechanical properties of the host-rock (advanced coupled (T)HM behaviour);
- Method to characterize diffusion properties of the near-field materials and the host-rock;
- Method to identify transport properties of the host rock (faults and joints);
- Method for description of geological properties of the site;
- Method to evaluate the sorption capacity of the near-field materials and the host-rock;
- Method to evaluate the permeability of near-field materials and the host-rock.

### 5.1.2 Representativeness of the evaluated data

DGD is a large complex multibarrier system, being constructed in heterogeneous host rock massives. Therefore not all the information can be easily generalized. Data can suffer from both simplification of complex system, processes and time scales and from size reduction, namely due to laboratory research. Therefore the data may not be fully representative of the whole DGD system and its future evolution. Consequently, TSOs and safety authorities should develop confidence in:

- the upscaling approaches for the data evaluated at small scale in order they can be representative for the whole DGD system;
- methods for data extrapolation in time (e.g. for the long-term safety demonstration);
- methods dealing with system heterogeneity.

Confidence in the transferability and in the time extrapolation in terms of data for the safety demonstration may notably be built by understanding the complex processes and phenomena that will alter the evaluated data under repository conditions (see section 6.2). The coupling between the Thermal, Hydrological, Mechanical and Chemical (THMC) properties of the DGD system is of particular concern for the transferability and the extrapolation in time of the evaluated data.

**Needs for knowledge:**

**A2** Representativeness of the data evaluated at small scale (in time and space) with respect to in situ repository conditions and future evolution.

**Particular points of attention (not exhaustive list):**

- Representativeness of rock properties (mineralogy and petrology, porosity, pore connectivity, fracturing, mechanical properties);
- Representativeness of hydraulic performance of seals and concrete liners;
- Representativeness of the geological structures in 3D dimensions (e.g. seismic methods);
- Representativeness of barriers transport properties (sorption, diffusion data, permeability).

## 5.2 UNDERSTANDING OF COMPLEX PROCESSES

In order to design the GD facility and to demonstrate its long-term and operational safety, operators have to develop understanding of the key processes (i.e. Thermal, Hydrological, Mechanical and Chemical processes and their related couplings) which govern the evolution of the GD system. TSOs and safety authorities have to build confidence in the understanding developed by the operator. This may be achieved notably by undertaking independent R&D actions devoted to the understanding of:

- the processes on which rest the performances of the four main components of the disposal system (waste forms, canister and overpacks, Engineered Barrier System (EBS) and geosphere);
- the processes resulting from potential internal and external perturbations of the disposal system (i.e. its long term evolution).

For these two topics, needs for knowledge are described in the two following subsections. The understanding of the (individual and coupled) processes might require complex experiments, large computing resources and time. Consequently, a preferred option might be in the development of small scale experiments and models. Action A2 (see section 6.1.2) should give insights in the upscaling of the acquired data and models to conditions representative of the disposal system.

## 5.2.1 Processes on which rest performances of individual components

This subsection defines R&D actions aiming at characterizing the processes on which rest the performances of the four main components of the disposal system (waste forms, canister and overpacks, remaining engineered barriers and geosphere).

#### 5.2.1.1 Waste forms

**Needs for knowledge:**

**B1** Understanding in the processes on which rest the performances of the waste form.

**Particular points of attention (not exhaustive list):**

- SNF disposal: characterization of the process responsible for the Instant Released Fraction (IRF) due to $^{129}I$ and $^{36}Cl$;
- Graphite bearing wastes (e.g. RBMK, UNGG reactor type): characterization of processes responsible for the $^{14}C$ and $^{36}Cl$ release from waste forms;
- Vitrified waste: influence of the initial fracturing state of glass on its dissolution rate (and thus the radionuclide release rate);
- The influence of the pH evolution on dissolution rate of the vitrified waste packages (e.g. high pH from the concrete environment);
- Influence of dissolutions rates for radionuclide release from SNF;
- The influence of glass leaching on the radionuclide release rate;
- Gas issue (in case of bituminised waste presence).

#### 5.2.1.2 Container & overpack

**Needs for knowledge:**

**B2** Understanding in the processes on which rest the performances of the waste container and its overpack.

**Particular points of attention (not exhaustive list):**

- The influence of the thermal dissipated power on the container and overpack properties;
- The assessment of corrosion mechanisms and rates (e.g. generalised or pitting corrosion) in reference conditions;
- The dimensioning of the container and overpack with respect to the different loads experienced under repository conditions (due to the host-rock behaviour and to the thermo-mechanical effects);
- The influence of corrosion layer on the extent of corrosion;
- Influence of microbial activity on corrosion rates;
- Influence of radiolysis process on corrosion kinetics.

#### 5.2.1.3 Surrounding engineered barriers (buffers and seals, construction materials…)

**Needs for knowledge:**

**B3** Understanding in the processes on which rest the performances of the engineered barriers surrounding the waste packages.

**Particular points of attention (not exhaustive list):**

- Processes affecting the geo-mechanical properties of the EBS (swelling capacity of bentonite materials, change of the chemical composition of bentonite materials…);
- Process affecting the hydraulic properties of the EBS (EBS permeability…);
- Processes of buffer erosion (causing loss of the performance);
- Processes on which rest the retention/sorption capacity of the EBS;
- Long term evolution of buffer and sealing materials.

#### 5.2.1.4 Geosphere

**Needs for knowledge:**

**B4** Understanding in the processes on which rest the performances of the Geosphere.

**Particular points of attention (not exhaustive list):**

- Long term stability of the geosphere (including seismic, orogenic properties);
- Processes affecting the mechanical properties of the geosphere and its healing (for clay rock and rock salt);
- Mechanisms of creation and propagation of natural heterogeneities in multilayer sedimentary system, leading to a “differential fracturing” between limestone and clayey layers at the whole system scale
- Influence of the mineralogical composition of the host rock on its sorption capacity;
- The migration of radionuclides in the host-rock at ambient temperature and considering temperature gradient representative of those that may occur in the near-field of the foreseen waste disposal system;
- Identification of fracture wetting surface in fractured rocks in order to determine the extent of radionuclide migration;
- Evaluation of indicators on the confinement capacity (i.e. diffusion dominated) at long-term, and its consistency with the DGD concept:
    - o Clay type DGD: verification that the concentration of solutes (such as Cl) shows a classical diffusion profile, which proves that diffusion prevailed up to present day (i.e. before DGD implementation); identification of effective porosity, i.e. the porosity available for radionuclide migration
    - o Granitic type DGD: identification of pore connectivity in the crystalline rock massives; identification of effective porosity, i.e. the porosity available for radionuclide migration

### 5.2.2 Processes resulting from internal and external perturbations

This subsection focuses on the understanding of processes resulting from internal and external perturbation of the GD system (e.g. Internal: potential interactions between the four main repository components, construction and operation of the repository… External: climate change, permafrost…). These perturbations may be critical for the long-term safety of the repository. As an example, the long term heat generation from the emplaced radioactive wastes (internal perturbation) can induce a perturbation of the geosphere properties and therefore jeopardize the long-term safety of the repository (e.g. permeability increase and change of the geochemical regime in the vicinity of the repository near-field).

Consequently, TSOs and safety authorities have to develop confidence in the potential internal and external perturbations identified by the operator in the safety case. This may be achieved notably by building an independent understanding of the potential internal and external perturbations of the disposal system and on their impact on the long-term safety of the GD facility. The following perturbations will be considered in the next subsections:

- Internal perturbations:
    - o Waste/Host-rock and Waste/surrounding EB interactions;
    - o EBS/Host-rock interactions;
    - o Perturbations during the operational phase;
    - o Perturbations due to constructions
- External perturbations.

#### 5.2.2.1 Internal perturbations: waste /Host-Rock and Waste/Surrounding EB interactions

**Needs for knowledge:**

**B5** Understanding in the internal perturbations of the disposal system resulting from the waste/host-rock and waste/EBS interactions.

**Particular points of attention (not exhaustive list):**

- Effects from waste forms:
    - o The introduction of a large quantity of new chemical component will perturb the in situ conditions and therefore the final geochemical conditions may not be the same as before the perturbation, even after the system returns to equilibrium.
    - o The compatibility between bitumen waste and the surrounding concrete and host-rock (with respect to their potential swelling).
    - o Generation of gases ($H_2$, $O_2$) upon radiolysis pore water, organic-based waste (e.g. bitumen).
    - o Radiation effects on materials.
- Effects of gas release from waste packages and from the corrosion or radiolysis of engineered barriers:
    - o The study of the pressure, the formation of either microcracks or macrocracks and the integrity of the geosphere
    - o The mechanisms of gas generation and migration and their effects on the mechanical (M) and hydrological (H) stability of the geosphere.
- Effects of temperature on cementitious materials:
    - o Chemical evolution (solid and pore solution)

    - Microstructure evolution
    - Composite effective diffusion coefficient evolutions

#### 5.2.2.2 Internal perturbations: EBS/Host-Rock interactions

**Needs for knowledge:**

**B6** Understanding in the internal perturbations of the disposal system resulting from the EBS/host-rock interactions.

**Particular points of attention (not exhaustive list):**

- Effects due to the corrosion phenomenon:
    - The corrosion behaviour of steel-based material in an anaerobic environment of deep repository.
    - The clay materials evolution due to iron-clay interactions (characterisation and modelling).
    - Effect of interaction of bentonite/steel on corrosion layer development and influence on the corrosion rate.
    - Anaerobic corrosion of iron metals is in addition connected with gas generation that can be detrimental to other components of a repository.
- Effects due to alkaline perturbation:
    - The clay materials evolution due to cement-clay interactions by characterisation and modelling.
    - Mineral transformations of bentonite in the alkaline front environment (it relates mainly to composition of penetrating alkaline waters, pH, single minerals transformation trend).

#### 5.2.2.3 Internal perturbations due to operation and construction

**Needs for knowledge:**

**B7** Understanding in the internal perturbations of the disposal system resulting from operational transients.

**Particular points of attention (not exhaustive list):**

- Effect of the oxidizing and desaturation transient process;
- Effects resulting from the presence of micro-organisms;
- Influence of defects, caused by handling, on canister corrosion;
- EDZ formation and extent.

#### 5.2.2.4 External perturbations

**Needs for knowledge:**

**B8** Understanding in the potential external perturbations of the disposal system.

**Particular points of attention (not exhaustive list):**

- General geological condition change (potential marine transgression, future permafrost/glaciations, site erosion, site seismicity, site uplift/subsidence);
- The changes of the properties of the natural and engineered barriers (through the changes in groundwater flow regime, chemical conditions) are strongly related to the condition changes which are expected in the future;
- Potential future human intrusion and activities (gas storage and extraction, geothermal energy, resources exploration etc.): the point is to possibly better know the actual resources but also to link this point with the scenario development strategy for assessing perturbations due to intrusions inside or close to the disposal facility.

## 5.3 ASSESSMENT OF EXTENT, INTENSITY AND IMPACT OF PROCESSES

In order to demonstrate the long-term and operational safety of the repository, operators have to assess the spatial extent and the intensity of the processes resulting from the internal and external perturbation of the GD system (see section 5.2) and the potential radiological impact that would results from these evolutions. This assessment is carried out by modelling activities.

The review of these assessments made by WMOs relies first on the assessment of the hypothesis selected and in particular the set of data used to feed the models (5.1). The reliability of the physical and numerical models developed by the operators, the comprehensiveness of scenarios modelled and the methodology followed to manage the uncertainties are also key parts of the review.

### 5.3.1 Reliability of FEPs and scenarios

**Needs for knowledge:**

**C1** The methodologies followed by the operators for the assessment calculations.

**Particular points of attention (not exhaustive list):**

- Features, events and processes that are potentially important for the safety of the disposal system should also be identified thus there is a priority suggested on the development of updated FEP databases. E.g. (not comprehensive list):
  - o Scenario caused by earthquake leading to the immediate failure of various numbers of the canisters in the time of mean lifetime of canisters.
  - o Scenario caused by denudation or erosion leading to the substantial shortening of radionuclide pathways to the environment.
  - o Scenario caused by the formation of preferential pathways in part of buffer in boreholes with immediate failed canisters due to earthquake.
  - o Intrusion scenario leading to the failure of one canister and buffer immediately after the end of institutional control of the repository (300 years).
  - o Climate change scenario during post-closure period, taking into account evolution of the possible biosphere behaviour.
- Safety functions of all system components should be developed for scenario development.

- FEP screening and scenario development should be supported by experimental demonstration and natural analogue observations.

### 5.3.2 Reliability of models

**Needs for knowledge:**

**C2** The extent and intensity of processes resulting from internal and external perturbations of the repository on human and the environment (including dose).

**Particular points of attention (not exhaustive list):**

- Relevance of underlying conceptual (phenomenological) models used for the performance assessment:
  - Source term model for the radionuclide release from waste packages and spent fuel. This modelling is of particular concern as it has a deep impact on the long term safety of radioactive waste disposal. As an illustration, the modelling of the spent fuel Instant Release Fraction (IRF) is an important point of attention (see "Data quality" section).
  - The modelling RN migration and sorption, account for the range of relevant geochemical processes.
- Investigate the influence of the level of abstraction and simplification (of mechanical, hydrological, thermal and chemical processes) on the results of the assessment calculations; so as the robustness of the data transfer from one system component to the other.
- Verification and validation of the models, both conceptual and mathematical ones. This point is particularly important as the assessment of the extent and the intensity of processes, as well as their influence on the long-term safety, is principally based on modelling activities. Moreover, the justification of models during the transient phase (several thousand years) and on long time scales (such as those considered for DGD) is a key aspect of confidence.

### 5.3.3 Management of uncertainties

**Needs for knowledge:**

**C3** Management of uncertainties

**Particular points of attention (not exhaustive list):**

- The challenges linked to the management of uncertainties concern several issues:
  - The identification of lack of knowledge (5.2), and link with upscaling process and long-term extrapolation in time
  - The influence of the device and measurements used to obtain input data (see section 5.1).
  - The technics for taking into account the uncertainties into the processes and the models
  - the bias linked to of abstraction/simplification of the reality leading to conceptual models.

- o Influence of Mathematical representation of processes included.
- o Applied software and its verification and validation; confirmation of its suitability.

### 5.3.4 Monitoring

**Needs for knowledge:**

**C4** Definition of the reference state of the system (normal/expected evolution) and monitoring methods in order to detect deviations

**Particular points of attention (not exhaustive list):**

- o During construction phase: monitoring like for civil engineering/mining objects.
- o During operational phase: to confirm short term evolution and update the safety case; To detect deviations from the expected domain of behaviour: what process to measure? Which components? Which parameter? What is the specified (safe) domain of functioning (the reference state) ?...
- o During post-closure phase: until the end of institutional control.

## 5.4 OPERATIONAL SAFETY

In parallel to the work developed by reviewers to increase knowledge and implement review methods of long term safety, the assessment of operational phase safety is an increasing challenge as far as disposal programmes progress. As example, the IAEA GEOSAF project on international harmonization of approaches in the evaluation of the safety case for a geological disposal launched in 2010 a pilot study on fire hazard and the GEOSAF 2 project was subsequently initiated in 2012 on the integration of operational phase and post-closure phase into the safety case.

In terms of methodology of regulatory review, existing methods already used for various nuclear facilities remains valid and serve as basis for reviewing safety of the operation of geological disposal. But some specific risks or situations need to be addressed without any substantial feedback experience from the operation of existing nuclear facilities (management of concomitant activities, management of fire…). Parameters associated to the characterization of the considered risks (fire, flood…) needs to take into account the peculiarities of such a facility. Finally, the identification of Limits, Controls and Conditions for the operational phase remains a challenge, since it has to integrate the dimension of long term safety: the numerous links between pre- and post-closure arguments of the safety case call for a methodology to verify continuously that the operator is always on the right track to achieving its target, namely the conditions of the repository at the time of closure which form the basis of the demonstration that the facility is sure in the long term.

These four findings lead to the unveiling of five consequences for the preparation of the regulatory review :

1. The analysis of the design, the maintenance, the coherence between the provisions adopted and the considered risks (especially those that are specific to a geological disposal) should be deepened. This underlines the questions related to the technologies used in the facility, the architecture, the components' robustness and easy maintenance, as well as a deep understanding of the mechanisms associated to the ageing of the abovementioned components.
2. The analysis of the scenarios used by the licensee, especially those that are envelope, should be carried out with a good understanding of the peculiar characteristics of a geological disposal.
3. Risks associated to activities running in parallel over extensive periods of time should be considered as essential.
4. The analysis of the adequacy of a monitoring and surveillance programme during the operational phase, which would consider several objectives, should be performed as well.
5. As stated above, a deeper knowledge of the various situations and parameters that influence the "initial state" of the closed repository (namely the characterization of the set of parameters that control the post-closure safety assessment) should be sought as well.

**Needs for knowledge:**

**D1** A methodology to review the hazards possibly occurring in an underground nuclear facility and scientific basis associated to development of conventional/nuclear hazardeous processes in underground

- fire hazard
  - characterization of fires in underground spaces
  - thermal response of ILW emplacement cells on temperature rise aggressions
  - quantification of effects of fire on specific target in confined environment
  - integration of different confined environment in IRSN's simulation tools
- handling hazard
  - characterization of situations of stopping the transfer of canisters and emplacement
  - consequences of these situation on the components relevant for safety and on the general level of risks in the facility
- hazard due to activities performed in parallel (co-activity)
  - methods (including in other industries) for organizing safely activities performed in parallel
  - definition of situations (such as evacuation in the case of a fire in the underground area) that should be taken into account in the analysis of these risks

# 6 Conclusions

This report displays the main issues to be addressed when reviewing the scientific elements to be developed by the implementer for assessing the LT safety of a deep geological disposal. At this step of the study, the set of major scientific issues to be well understood by the reviewer is not formally turned into a strategic research agenda. The current R&D national programmes yielded by TSOs and regulatory bodies are of various level of maturity depending of the progress of the national disposal programmes. For this reason, it appears quite difficult to draw full common lessons and perspectives in terms of R&D joint programming regarding the various key questions identified. Maybe this objective could be envisaged by addressing two main categories of scientific issues: on the one hand those related to processes where the scientific community already made progress and where additional efforts concern specific ongoing development designs in selected sites (for more advanced programmes), on the other hand those related to generic scientific topics that concern any kind of programmes (more related to the assessment methodology). However, collaborative programmes do remain of interest on some higher level scientific topics related to components and materials (behaviour of concrete, performance of seals…) or cross-cutting issues and integrated modelling (role of interfaces, coupling of processes for example, transient phase…).

At this step, a common view of SITEX WP3 participants has been reached on the needs for knowledge that are necessary in order to support the independent expertise function in reviewing the Safety Cases for geological disposal repositories. These needs are classified based on three main axes:

- the quality of the data on which rest the safety demonstration;
- the understanding of the complex processes which may potentially influence the long term safety of the DGD;
- the assessment of the future evolution (in spatial extent and intensity) of these potential processes, as well as the assessment of their impact on the DGD safety.

A specific attention to challenges in operational safety review has been paid in order to highlight needs for better characterisation of nuclear and conventional hazards possibly occurring in an underground nuclear facility. Critical issues appear at this step to be related to fire hazards, waste package handling and disposal activities in areas close to construction areas.

Following these issues, a number of R&D actions are already yielded by TSOs and regulatory bodies, as emphasised in the D3.2 deliverable entitled “Availability and needs for technical and scientific tools for TSO's”. This second deliverable of the SITEX WP3 will present the experimental and modelling capabilities of participants. On the basis of this analysis and available skills and tools identified, possible collaborations between TSOs will be addressed in the deliverable D3.3 “the strategy for implementing TSO’s R&D programmes”.

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# 8 Appendix 1: advancement state of national projects

In WP3 the following countries participate:

BELV, Belgium

Canadian Nuclear Safety Commission CNSC, Canada

DECOM SA, Slovakia

Federal Agency for Nuclear Control, FANC, Belgium

Gesellschaft für Anlagen-und-Reaktorsicherheit, GRS, Germany

Institut de Radioprotection et de Sûreté Nucléaire, IRSN, France

Lietuvos Energetikos Institutas, LEI, Lithuania

Ministerie Van economische Zaken, Landbouw en Innovatie, ELI, Netherlands

UJV Rez, a.s., Czech Republic

France has a developed programme, reaching the point of licence submission. In Belgium, after long term research in the Boom clay formation, the waste management organization submitted in 2011 to the government a waste plan recommending the geological disposal of SNF and long lived ILW and HLW. In April 2013, a decision in principle from the government is still pending. Germany after planning the disposal in the Gorleben mine stopped the decision process in 2000 and awaken it in 2010 with broader look to other potential host rock formation. Czech Republic has a small DGD development programme, heading to repository opening in 2065. Even though Netherlands considers SNF disposal, it played also an active role in project aiming in international repository (Arius Sapierr). Slovakia and Lithuania displayed by very small programmes, being at the beginning of the process. Lithuania is the only country considering also SNF shipment to other countries.

The concept of geological disposal of radioactive waste is notably based on the multi-barriers/multi-functions principle, in which the long-term safety is ensured by the natural geological barrier and the Engineered Barrier System (EBS) that act in tandem. Geological disposals are designed to be passively safe. In the following chapter, the EBS will be sub-divided into the following barriers:

- the waste form

It consists of the waste in its physical and chemical form after treatment and/or conditioning (resulting in a solid product) prior to packaging (e.g. waste immobilized in a concrete, glass or bitumen). The waste form is the very first engineered barrier against radionuclide release to its surrounding environment.

- The waste container (the vessel in which the waste is placed for disposal) and its eventual overpack.

For example, vitrified HLW waste and SNF are generally placed inside a specifically designed waste container (the canister). Then, the waste container may be placed inside an overpack. For instance, in Belgium and Czech Republic, the so called supercontainer option is currently considered: i.e. the waste container is enclosed in an overpack consisting of a carbon steel layer, a buffer of cement material and a stainless steel envelope (and bentonite in the Czech design). The waste form, the waste container and the overpack are the components of the waste package.

- the surrounding engineered barriers (e.g. the backfilling material);
- the host-rock which is the natural geological barrier ensuring radionuclide confinement.

These natural and engineered barriers will contain the migration of RN to the biosphere (e.g. by relying on the favourable geochemical conditions of the host-rock (i.e. reducing environment) and on the properties of the engineered materials steel, concrete, bentonite). This slow migration rate will allow a sufficient radioactive decay for a major part of the radionuclides before they reach the biosphere, and therefore contribute to the long-term safety of the repository.

The countries consider either disposal of SNF itself (Czech Republic, Belgium, Slovakia and Lithuania) or disposal of vitrified waste (France, Netherlands, Germany).

The long-term safety of a deep geological repository for RW will be strongly dependent on the performance of the geosphere (including the host-rock). The geosphere potentially isolates the RW from possible future intrusions by humans; provides a stable physical and chemical environment for the engineered barriers within the repository, insulating against external perturbations such as earthquakes and climate change; and prevents, delays and attenuates radionuclide transport by virtue of its hydraulic and sorption properties [5]. Different types of host rock are considered even within the countries, involved in the presented study. France, Belgium and Netherlands consider clay rock as the reference option for disposal. Clay rock is also considered by Slovakia, Lithuania and Canada, altogether with other options. Granite is under consideration in Czech Republic, Slovakia and Canada. Rock salt may be also the choice for Germany, Netherlands and Lithuania. Canada also evaluates sedimentary rock formation (marl) as a potential host-rock for radioactive waste disposal.

Short summarisation of various safety concepts in participating countries is gathered in . The **Safety Case** is an integral part of DGD programme development. It represents the collection of scientific, technical, administrative and managerial arguments and evidence in support of the safety demonstration of a disposal facility, covering the suitability of the site and the design, construction and operation of the facility, the assessment of radiation risks and assurance of the adequacy and quality of all of the safety related work associated with the disposal facility.

The Safety case was adopted by most countries, participating on the study. Hereby are listed several examples of different levels of maturity of safety case:

- Belgium: Preliminary safety analyses have been documented by the waste management organization in SAFIR reports I and II (in 1989 and 2003, respectively)

- Canada: not mention fully in Canada contribution, but illustrative case study of the current design and postclosure safety planed

- Czech Republic: preliminary safety study for hypothetic site in crystalline (2012)
- France: Dossier Argile 2005 (feasibility of DGD in Callovo-oxfordian formation, 2005)
- Germany: preliminary safety analyses on-going (2010-2012)
- Lithuania: generic safety assessment planned
- Netherlands: generic safety analyses PROSA (1990), within OPERA planned for clay and salt
- Slovakia: none

## 8.1 BELGIUM

The Belgian research and development programme on the geological disposal in clay of long-lived and/or high level radioactive waste was launched in 1974 by the Belgian Nuclear Research Centre (SCK•CEN). The Research Development and Demonstration (RD&D) programme rapidly focused on a poorly indurated geological formation located in the underground of the North-East of Belgium: the Boom Clay. As these preliminary studies had given confidence in the potential ability of the Boom Clay to contain radionuclides, SCK•CEN undertook in 1980 the construction of an experimental underground laboratory (HADES, High Activity Disposal Experimental Site) in the Boom Clay formation, in the vicinity of Mol.

In 1983, the responsibility for the RD&D programme was progressively entrusted to the Belgian Agency for Radioactive Waste and Enriched Fissile Materials (ONDRAF/NIRAS). In 1989, ONDRAF/NIRAS has provided its Supervising Authorities with the SAFIR report (Safety Assessment and Feasibility Interim Report), summarizing the RD&D achievements since 1974. The review of the SAFIR report highlighted notably that besides the option of Boom Clay, alternative host-rocks should be considered.

Consequently, the RD&D activities performed between 1990 and 2000 addressed a preliminary characterisation of an alternative Clay formation: the Ypresian Clay. However, the geological disposal in Boom Clay stayed the reference option of ONDRAF/NIRAS for the long lived and/or high level waste management. During this period, studies on Spent Fuel disposal were intensified as a moratorium was decided by the Belgian government on the recycling of spent nuclear fuel. The outcomes of the RD&D activities performed by ONDRAF/NIRAS between 1990 and 2000 were summarized in the SAFIR 2 report [1], which was subjected to an international Peer Review organized by the Nuclear Energy Agency (NEA) of the Organization for the Economic Cooperation and Development (OECD). The outcomes of this review were published in January 2003 [2]. Whereas the NEA Peer Review identified specific issues that should be further investigated, it concluded that the Belgian disposal programme is mature enough – from a scientific and technical point of view – to move progressively towards implementation.

In 2011, ONDRAF/NIRAS has developed a Waste Plan [3,4] recommending the geological disposal of high-level and/or long-lived radioactive waste. This Waste Plan has been submitted to the federal government for a decision-in-principle, in order to set a clear policy for the long-term management of high-level and/or long-lived waste in Belgium. In July 2012, the decision of the government is still pending.

In its advice on the Waste Plan, FANC supports a decision-in-principle "go-ahead for geological disposal" but considers that the decision to already select a specific type of host geological formation (i.e. poorly indurated clay) is premature. FANC advices to proceed to a systematic screening of the potential host formations present on the Belgian territory considering criteria reflecting both their confinement and their isolation performance before such a decision is taken.

If the government approves the principle of a geological disposal, the next milestone of the high-level and/or long-lived waste disposal programme, proposed by ONDRAF/NIRAS, is the development of a first "Safety and Feasibility Case" (SFC 1), planned in 2015. The SFC 1 would consider the construction of a reference disposal system in a definite zone of the Boom Clay formation. The Ypresian clay formation would be considered as an alternative host formation, with fewer details. According to ONDRAF/NIRAS, the main objective of the SFC 1 would be to substantiate that the proposed disposal system ensures a high level of operational and long-term safety. Based on the outcomes of the SFC 1, it would be considered whether to launch the siting phase.

A second Safety and Feasibility Case (SFC 2) would be expected to be finalised around 2020. The SFC 2 would be site-specific and would provide evidence that no major safety or feasibility issues prevent the implementation of the geological disposal facility. Based on the SFC 2, it would be considered whether to prepare the licensing application.

### 8.1.1 Description of DGD reference concept

The reference concept briefly described in this section is based on ONDRAF preliminary reports developed within the framework of the SFC-1 [5-7]. It is important to mention that several aspects are still under development by ONDRAF and that the concept has not yet been formally reviewed by the regulatory body (FANC and Bel V).

As discussed in the previous section, the current disposal solution foreseen for high level and long-lived radioactive waste is the Geological Disposal in poorly indurated clay. The foreseen geological repository (see **Erreur ! Source du renvoi introuvable.**, left) consists of horizontal disposal galleries lined with concrete wedge blocks. These disposal galleries will be connected to an access gallery, linking to the repository shafts. The disposal facility will be divided in two main sections: the HLW and the LILW-LL sections.

In the HLW section, the foreseen Engineered Barrier System (EBS) is mainly based on the concept of supercontainers. As depicted in **Erreur ! Source du renvoi introuvable.**, right, a supercontainer essentially consists of the primary container of vitrified waste or spent fuel, a watertight overpack of carbon steel, a thick protection layer of cement material and a stainless steel envelope. The supercontainer is intended to ensure the complete containment of radionuclides and chemical contaminants, at least during the thermal phase. In the Long-Lived waste section (non-heat-generating), waste will be immobilized in concrete containers, either in the form of conditioned or unconditioned waste packages, to form monoliths. Supercontainers and monoliths provide permanent radiological shielding for the workers and therefore facilitate underground operations.

Once the wastes will be deposited, the empty spaces in the repository galleries will be backfilled with a cement-based material. Finally, all the access galleries and all the shafts will be backfilled and sealed.

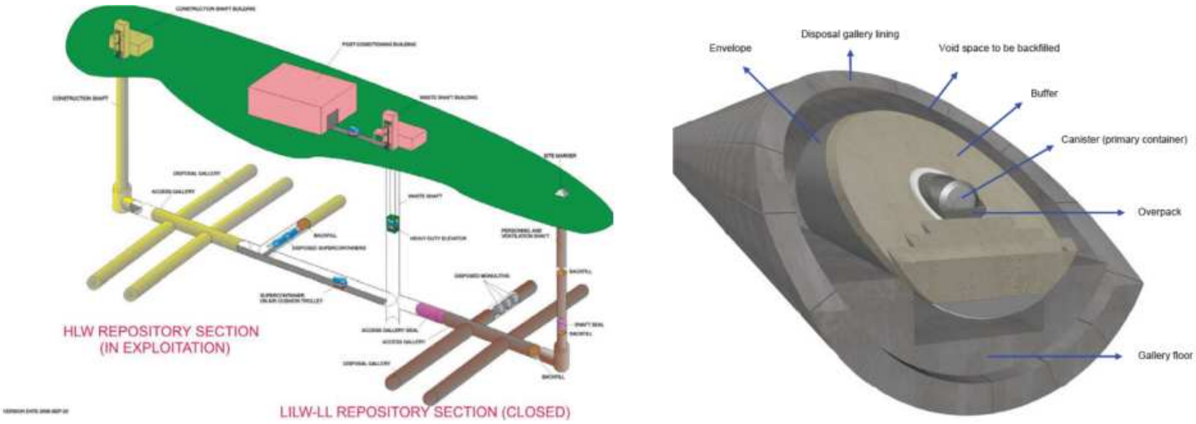



**Fig. 1** (left)**: Indicative scheme of the foreseen geological disposal facility [6]**, (right)**: Supercontainer for vitrified HLW in a disposal gallery [6]**

## 8.2 CANADA

### 8.2.1 Adaptive Phased Management for used nuclear fuel

The Nuclear Waste Management Organization (NWMO) was established in 2002 by Ontario Power Generation Inc., Hydro-Québec and New Brunswick Power Corporation in accordance with the Nuclear Fuel Waste Act (NFWA) to assume responsibility for the long-term management of Canada's used nuclear fuel. NWMO in its report to the government in November 2005 recommended the consideration of both granitic rocks and sedimentary rocks for its phased approach for the eventual geological disposal of used nuclear fuel. On June 14, 2007, the report from the NWMO was accepted by the Government of Canada for its Adaptive Phased Management (APM) for used nuclear fuel. APM is both a technical method and a management system, with an emphasis on adaptability. Technically, it is centralized containment and isolation of used nuclear fuel in a deep geological repository. The management system involves realistic, manageable phases – each marked by explicit decision points with continuing participation by interested Canadians.

APM moves towards a goal that Canadians themselves identified safe and secure long-term containment and isolation of used nuclear fuel produced in Canada, with flexibility for future generations to act in their own best interests. The NWMO is now developing a process for site selection to consider not only technical merits, but also social factors. The proposed site selection process is designed to use a partnership-based approach to:

- help ensure that any community that is selected to host this facility is both informed about the project and willing to host it;
- help ensure that any site that is selected to host this facility will safely contain and isolate used nuclear fuel for a very long period of time, in an appropriate geological formation, and that there is an acceptable way of transporting used fuel to the site;

- assist the potentially interested host community to consider carefully and thoroughly the project's potential benefits and risks when deciding whether to express interest, and ultimately, willingness to host the project;
- involve surrounding communities, regions and other jurisdictional levels potentially affected by the project and the transportation of used fuel in the identification and assessment of public health, environmental, social, economic and cultural effects of the project as part of a broader regional assessment;
- involve First Nations, Métis and Inuit who are potentially affected by the implementation of this project; and
- help foster an ongoing public conversation on questions to be answered and issues to be addressed throughout the site selection process.

As potentially suitable sites are identified with interested communities, detailed field studies and site characterization activities will be conducted to explore the potential to meet rigorous long-term safety requirements. Currently, 15 communities have expressed their interests in hosting a deep geological repository (DGD) for the management of Canada's used nuclear fuel for the long term.

In the meantime, the NWMO is maintaining a very active R&D program to keep up to date with the rapid progress in geosciences related to geological disposal. The NWMO is currently conducting conceptual design of a DGD for used nuclear fuel at a hypothetical but realistic site in both crystalline rock and sedimentary rock to provide an illustrative case study of the current multi-barrier design and postclosure safety of a deep geological repository. This hypothetical geosphere was derived in part from historic experience gained in the Canadian Nuclear Fuel Waste Management Program. It was developed for the purpose of this illustrative case study while the NWMO proceeds with the APM sitting process and selection of a preferred site in an informed and willing host community.

### 8.2.2 Ontario Power Generation's (OPG) DGD for low to intermediate level waste

Ontario Power Generation (OPG) is undergoing a multi-year planning and regulatory approvals process for a deep geological repository (DGD) for the long-term management of low and intermediate level waste (L&ILW). Although OPG's current storage practices (i.e. stored centrally at OPG's above-ground Western Waste Management Facility (WWMF)) are safe and could be continued safely for many decades, OPG's long-term plan is to manage these wastes in a long-term management facility – the DGD project.

In 2001, the Municipality of Kincardine, Ontario, where the WWMF is located, requested that OPG considers options for the long-term management of the WWMF's L&ILW. This led, in 2002, to a Memorandum of Understanding (MOU) between the parties. The MOU set out the terms for a plan to study the long-term management options. A DGD Hosting Agreement was signed in October 2004 between OPG and the Municipality of Kincardine to allow for the construction and operation of a deep geological repository for the long-term management of L&ILW waste from OPG's power plants. A Project Description for the DGD Project was prepared and filed with the Canadian Nuclear Safety Commission (CNSC) in December 2005, which initiated the regulatory approvals phase for the project. The CNSC presided over a public hearing in October 2006 in Bruce County for the purpose of determining the type of EA (Environment Assessment) process required for this undertaking. In December 2006, the Commission published its report with a recommendation to the federal Minister of Environment that the DGD Project should be referred to review panel. In June 2007, the federal Minister of Environment referred the project to a joint review panel. Final instructions and guidelines for the preparation of the project EIS (Environmental Impact Statement) were jointly released by the CNSC and the Canadian Environmental Assessment Agency in January 2009.

Beginning in 2006, a comprehensive program of field and technical studies and investigations has been undertaken including the disciplines of geoscience, safety assessment, environmental assessment, public communications and development of the engineering design. Expert review panels in the areas of geoscience, engineering and safety assessment were established to guide and review the study findings. Early involvement of the regulator in key issues or areas provided the guidance to the applicant on the preparation of submission. All work was completed in 2010 leading to the regulatory submission in April 2010. Currently, all submissions for environmental assessment and license to prepare site and construct are under public review. A license for site preparation and construction will be issued if the EA is past.

## 8.3 CZECH REPUBLIC

The systematic accumulation and assessment of scientific and technical data for developing deep geological repository, selecting EBS system and conducting safety assessment of DGD started in the Czech Republic already in 1993 in the project funded by Ministry of Trade and Industry and coordinated by Nuclear Research Institute Rez (NRI). The results from this project were summarized in so-called “Reference design report” of the deep geological repository in 1999 This report included preliminary report with requirements on a safety report needed to get permission for siting DGD from State Office for Nuclear Safety. The proposed reference, generic description of a repository was similar to Swedish KBS 3V concept of disposal of spent fuel assemblies in granite host rock in a vertical position, but with steel based canisters instead of copper canisters. The reasons for selecting this type of canister were partly economic, because canisters made of steel can be easily manufactured in Czech industry, partly site specific, because composition of granite water in most of granite sites of the Czech Republic contain much less chlorides than granite water in Swedish of Finnish sites and therefore it could be expected that pitting corrosion of stainless steel will be limited. The repository layout in granite host rock consists in spent fuel waste packages surrounded by bentonite bricks and located vertically in boreholes in the tunnels at least 500 m under the surface. The selection of granite host rock was made on basis of a comprehensive review of “pre-existing” geological information, where it was concluded that the most suitable host rock in the Czech Republic for a deep geological repository is granite. Clay or salt rocks are not available in dimensions sufficient for siting a repository.

In 1997, the Act on the Peaceful Utilization of Nuclear Energy and Ionizing Radiation (the so-called Atomic Act) established the Radioactive Waste Repository Authority (RAWRA) as a state organization which is responsible for preparation of Deep Geological repository, including the coordination of R&D activities and preparation of safety reports.

The Czech DGD development program was divided into the following parts:

a) Selection of suitable sites for DGD.

b) Proposal of repository design, including selection and characterization of suitable engineered barrier system.

c) Evaluation of safety of the disposal system and safety report preparation.

d) Related technical research and development.

Safety assessment activities focused primarily on studying processes occurring both in near field and far field of a repository in a hypothetical site in granite host rock and survey of candidate site for a repository, but the sitting process was stopped in 2004 for 5 years due to strong opposition of public in selected sites. It supposed that after agreement with municipalities it should start this year by more detailed geological survey, including drilling works.

After 2015 (or 2018 depending on Government decision and agreement with candidate site communities) to 2025 geological work should be performed on "main" site. On the basis of the geological work design of the repository would be updated. Properties of engineered barriers will be studied taking into account the results from the site.

In the period 2025-2030 underground laboratory should be prepared on site, if the site is found acceptable. The construction and operation of the laboratory can be, however, postponed on the basis of results from abroad laboratories.

In the period 2030-2065, underground laboratory and first modules of the repository can be excavated. After 2050 also the construction of the surface premises should start, including possibly an encapsulation plant.

It is expected that the deep geological repository will be put into operation after 2065.

Currently, the update of reference design of the project is before completion. The updated project was also based on KBS3 H Swedish design, but steel based canisters with spent fuel assemblies should be emplaced in horizontal boreholes in "super-containers" together with bentonite bricks (No definite decision, however, was made concerning the final DGD concept). The main effort is devoted to finding suitable sites and to get approval of local communities. The part of this project was safety case study which can be considered as the first safety case for Czech concept of DGD in a hypothetical site with steel based canisters. There were four sources of data for this study: 1) Inventory for spent fuel assemblies and other radioactive waste from Czech Nuclear Power Plants including new nuclear reactors under consideration, 2) hydrogeological and geochemical data from Czech sites similar to considered candidate granite host rock sites, 3) available data for candidate engineered barriers (Czech bentonites, data for lifetime of carbon steel overpack from NRI laboratories), and 4) literature data from abroad DGD programmes (mainly Swedish and Finish reports).

This preliminary safety report is burdened by a large number of uncertainties coming not only for shortage of data, but also for shortage of knowledge.

Radioactive Waste Repository Authority (RAWRA) that was established in 1997 according to Czech Atomic Law is responsible for the preparation of safety reports of radioactive waste repositories. RAWRA submit to State Office for Nuclear Safety (SÚJB) safety reports to get approval for siting, construction, operation and closure of radioactive waste repositories. Currently the own capabilities of RAWRA to implement the safety reports are, however, limited so that it utilizes contractors both for acquiring data, modelling, and preparing preliminary safety cases.

## 8.4 FRANCE

### 8.4.1 The DGD concept for HL and IL-LL wastes

Three main safety functions are assigned to the DGD:

1. Preventing water circulation along repository, which is achieved with levels of dead end drifts and with the emplacement of bentonite based seals at key points;
2. Limiting RN release and immobilize them in the repository, relying on the favourable geochemical conditions of the host-rock (i.e. reducing environment) and on the different wastes packages (steel, concrete);
3. Delaying and attenuating the RN migration, thanks to the very low permeability of the clay host-rock formation (typically below $10^{-12}$ m/s) which thus makes the solutes transfer dominated by diffusion.

The DGD concept is based on a multi-barriers concept, the last barrier being a clay-rich host-rock itself. Its architecture results from a declination of the main safety functions adapted the specific target clay-host rock, namely a sedimentary argillaceous rock layer (Callovo-Oxfordian argillites) located in the eastern part of the Paris sedimentary basin. It is based on the main following features (see Fig. 2):

- Surface installations that provides facilities for receiving and finalizing the wastes packaging : primary wastes drums are packed in secondary packages (carbon steel-based for HL, cement-based for IL-LL wastes);
- Both an access ramp and shafts for providing necessary links between surface installations and the repository level for operational purposes (ramp to transport wastes to the repository, ventilation shaft, access shafts);
- A repository level, located below 500m depth, close to the middle part of the 130 m thick clay host-rock, from which all access drifts and horizontal disposal cells are progressively built;
- Secondary wastes packages (HL and IL-LL), progressively disposed in their corresponding disposal cells;
- Disposal cells, grouped together according to the type of wastes (HL vs. IL-LL, and subtypes according to their more detailed chemical composition), in order to better control their possible interactions;

- A closure of disposal cells at the end of their operation, followed by a progressive closure of the several levels of repository seals.

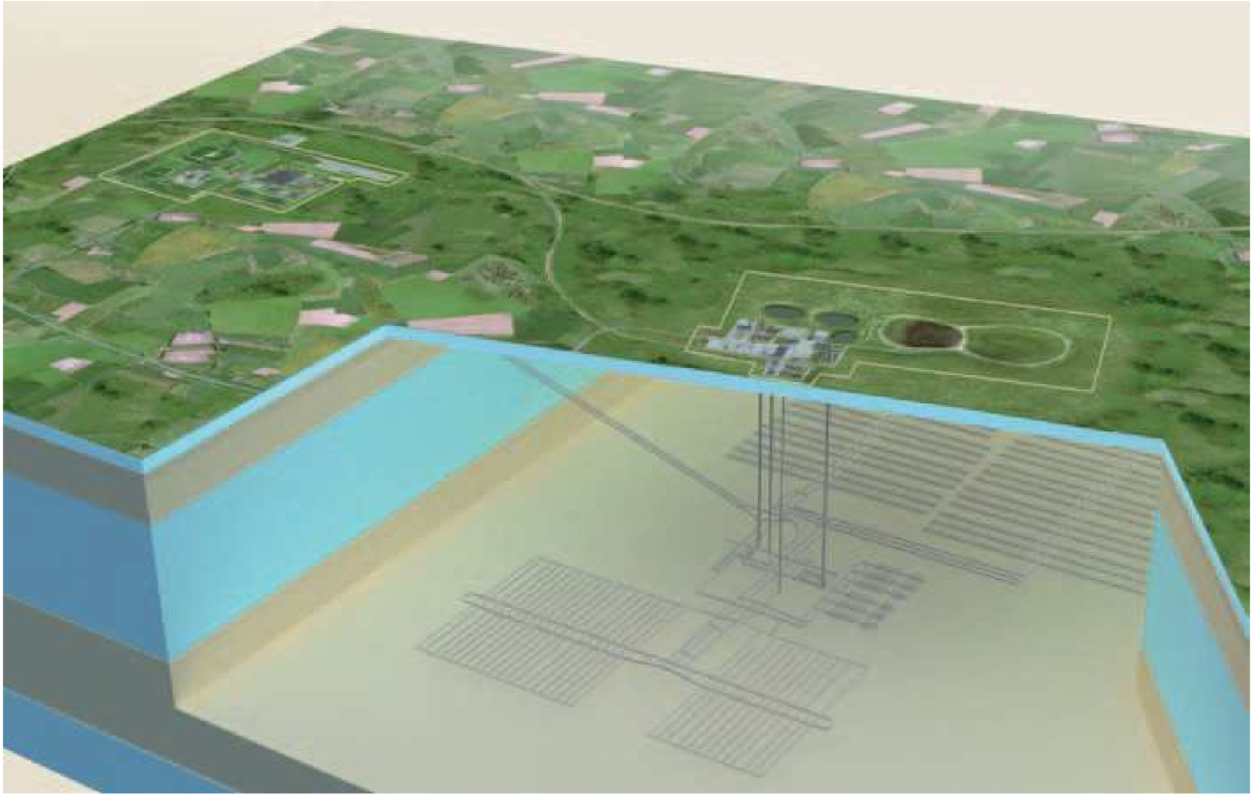


**Fig. 2. Provisional view of the French DGD for HL and IL-LL wastes (after Andra)**

### 8.4.2 Current state of the DGD development for HL and IL-LL wastes

Following the *feasibility Dossier* provided by Andra to the Government in 2005, the *2006 Planning Act* provides the general planning for the DGD development (see next Fig. 3).

- End 2009, Andra presented the *basic technical options* of the project related to the operational and the long-term safety, to the reversibility, to the proposal of various storage options to be studied and to the proposal of a reduced area (~30 km$^2$ within the transposition zone of ~250 km$^2$) where investigations concentrated after 2009 with a view to site the repository. These basic technical options were reviewed by IRSN in 2010.
- The current step involves preparing a *public debate*, based on a detailed description of the project objectives, its main characteristics, a proposal for an implementation site, its socio-economic stakes, its estimated cost and its significant impacts on the environment and on the regional development. This public debate aims to help enlightening future decisions, particularly regarding the draft law on reversibility conditions, which will be prepared by the Government once the licence application concerning the implementation of the repository will be reviewed. The public debate will take place in 2013 to leave enough time until 2015 for Andra and the Government to take into account the observations made by the public.

- The following step should lead to file the repository licence application for the implementation of the DGD at the end of 2014. This application will include notably a site plan, an environmental impact assessment, a preliminary safety report, a description of the planned management, closure and monitoring modalities for the facility and a scientific and technical supporting file. On the basis of the licensing process prescribed by the Planning Act, its review during 2015 could lead to promulgate a new law detailing reversibility modalities. The Planning Act of 28 June 2006 provides for a broad consultation, calling for the collection of the opinions of local communities concerning the application and a public inquiry that may take place during the second half of 2016. The implementation of the new disposal facility might very well be authorised by a decree at the end of 2016.
- Without prejudging the provisions to be adopted in 2016, it is estimated that a 8-year period will be required to build the repository and to conduct pre-commissioning tests. Such a rather short timescale in comparison with the scope of the project has already led to the drafting of a likely construction schedule in order to launch work as soon as possible.
- This should lead to the commissioning of the repository in 2025, subject to the grant of the required licence through a decree validated by the State Council and after public inquiry.
- The expected operational phase duration is about 100 years, thus the final closure could be expected around 2025.

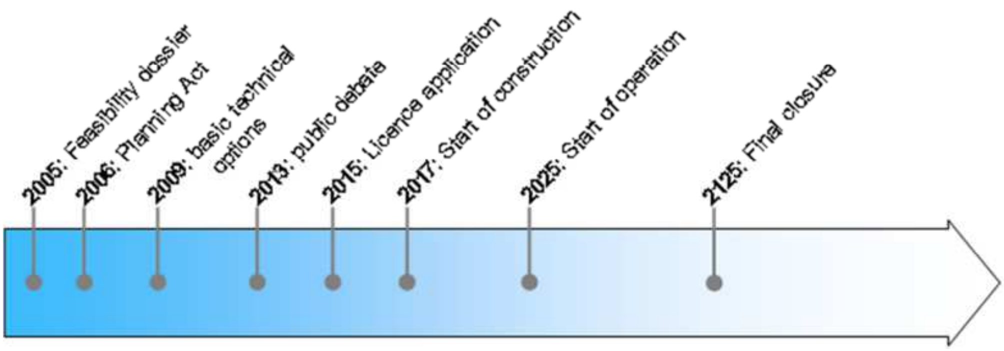



**Fig. 3. Development plan of the French DGD for HL and IL-LL wastes**

## 8.5 GERMANY

Since the early sixties, i.e. from its very beginning, radioactive waste disposal in Germany is based on the Federal Government decision that all types of radioactive waste (short-lived and long-lived) are to be disposed of in deep geological formations within the country. This applies to waste from reprocessing of spent fuel assemblies from German nuclear power plants as well as to waste from the operation and decommissioning and / or dismantling of commercially operated nuclear facilities, together with waste originating from the use of radioisotopes in research, trade, industry and medicine. Only solid or solidified waste is accepted for disposal; liquid and gaseous wastes are excluded from the emplacement in a geological repository.

Due to intention to dispose of all types of radioactive waste in deep geological formations there is no necessity to differentiate between waste containing radionuclides with comparatively short half-lives and waste containing radionuclides with comparatively long half-lives. As such, there are no measures or precautions required in Germany in order to separate radioactive waste in this way.

### 8.5.1 Site selection for a geologic disposal for heat-generating waste

In November 2011 the federal government has started negotiations with the governments of the federal states about a site-selection-process to determine a site for a geologic disposal, due to the still ongoing controversial political debate about the Gorleben site. A federal working group will, according to a road map of the Federal Ministry for the Environment, Nature Conservation and Reactor Safety (BMU), negotiate features of the site selection process and elaborate the legal framework for selecting and locating a repository site up to midyear 2012.

### 8.5.2 Exploration of the Gorleben salt dome

Since 1979 the Gorleben salt dome is subject of an ongoing site exploration programme in order to examine its suitability for hosting a repository for all kinds of radioactive waste, in particular heat-generating waste. The Exploration was interrupted for ten years.

On 1 October 2010 Federal Office for Radiation Protection resumed the exploration activities. The main objectives of the resumed site exploration are

- the investigation of local occurrences of hydrocarbon within salt rock, its distribution, its composition and its origin as well as the potential impact on heat-generating radioactive waste,
- and the position and blocking of main anhydrite within the Gorleben salt dome.
- Furthermore subsurface geoscientific exploration in other exploration areas is necessary in order to proof the occurrences of suitable potential emplacement areas sufficiently distant to brine inclusions and to the main anhydrite as well as to provide a complete set of data for a safety case.

Since July 2010 the Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) performs a preliminary safety analysis for summarising all available information on the Gorleben salt dome, including the results of exploration activities until now. The results of the preliminary safety analysis are expected by the end of 2012.

### 8.5.3 Status of existing facilities

Konrad:

In May 2002 the Konrad mine, a former iron ore mine, has been licensed for disposal of alpha-bearing low and intermediate level waste with negligible heat generation. This plan approval became final in April 2007 by court decision.

At present the construction of the Konrad repository for low and intermediate level waste proceeds. First new buildings have been erected and existing constructions have been altered for continued use. Underground drivage has been continued to prepare the first emplacement drifts. Revision of the existing detailed planning to ensure compliance with rules and standards is ongoing. Start of operation is planned 2019 earliest.

Morsleben:

The Morsleben repository (ERAM) was in operation until September 1998. An amount of approximately 37.000 cubic meters of low and intermediate level waste was emplaced in the Morsleben disposal. Now the plan-approval procedure for the decommissioning of the Morsleben disposal is in the final stage. In October 2011 the public hearing relating to the objections took place. Decommissioning can only start once the plan-approval procedure for the decommissioning of the ERAM has been completed. It is expected, that the plan-approval procedure will finish in the year 2014.

Asse:

Until 1978 the Asse salt mine was used as a disposal facility for low and intermediate level waste and as an underground research laboratory. An amount of 47.000 cubic meters waste has been emplaced. Since 1988 the mine is jeopardized by a groundwater inflow of about 12 cubic meters per day actual. The mine is under ongoing deformation, roof and pillars are weakened.

In 2010 the decision for full retrieval of all emplaced radioactive waste was taken by comparing the advantages and disadvantages of different options (retrieval, relocation, or complete backfilling). A compilation of facts for the retrieval has started. Reopening of two chambers is planned.

Multiple measures for operational safety and radiation protection needed to be in place before compilation of facts can start. For the full compilation of facts including drilling, opening the two chambers and testing the handling of the barrels with radioactive waste three years were planned originally.

The recent experiences of the Federal office for Radiation Protection show that the collection of facts takes much more time than previously expected.

### 8.5.4 Requirements

So far, the Federal Government has not made a decision on the further procedure for the construction of a repository for heat-generating waste. The “Safety Criteria for Disposal of Radioactive Waste in a Mine” as of 1983 do not comply with the actual recommendations of the IAEA, OECD/NEA or ICRP. Therefore, the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU) published in September 2010 revised “Safety Requirements Governing the Final Disposal of Heat-Generating Radioactive Waste” taking the latest state of the art into account [BMU, 2010]. In particular, these Safety Requirements regulate the following points:

- The protection objectives pursued by the final disposal of radioactive waste
- The safety principles to be observed
- A step-by-step approach and optimization with regards to radiation prevention, operational safety and reliability of the safe, long-term containment of waste, with due regard for feasibility
- Protection from damage caused by ionizing radiation
- Requirements governing safety analyses and the evaluation thereof for operation and long-term safety
- Design requirements governing the safety concept for the final repository during the operating and post-operational phases
- Safety management for construction and operation of the final repository
- Documentation of the final repository

## 8.6 LITHUANIA

There is only one nuclear power plant in the Republic of Lithuania - Ignalina Nuclear Power Plant (Ignalina NPP). It contains two RBMK-1500 water-cooled graphite-moderated channel-type power reactors. The first and the second reactors were shut down by the end of 2004 and by the end of 2009, respectively. During operation, the power plant has accumulated large quantities of radioactive waste, including spent nuclear fuel (SNF) [Poskas, et al., 2012].

The amount of Spent Nuclear Fuel (SNF) accumulated at the INPP consists of approximately 22,600 fuel assemblies. Unloaded from the reactor's core, fuel assemblies are cooled in water pools for at least one year. Then, fuel bundles can be separated from fuel assemblies (one fuel assembly contains two fuel bundles) for further cooling (prior dry storage SNF shall be cooled in water pools for at least 5 years), management and disposal. The current SNF management concept in Lithuania foresees dry storage of SNF bundles for about 50-100 years. During this time the final SNF management concept shall be developed.

### 8.6.1 Spent nuclear fuel management and disposal

The original management option for SNF to transfer it to Russia was considered. All competence related to SNF management was with central Soviet organisations and very little competence existed in Lithuania. With the independence Lithuania had to find alternative options for spent fuel management since at that time transfer of the SNF to Russia was no longer an option.

Lithuania has established an appropriate legislative and regulatory framework in order to govern safety of radioactive waste management. All the legal acts concerning radioactive waste management are prepared according to the best national and international practices, including International Atomic Energy Agency (IAEA) recommendations. The revised Strategy for Management of Radioactive Waste (approved by Lithuanian Government in 2008) indicates three potential options for management of SNF after interim storage:

- Disposal in the national repository.
- Disposal in the EU established regional repository.
- Transfer to other country which possesses suitable SNF management technologies and can take full responsibility for management of the waste.

If the international policy regarding to SNF transfer to other countries will not change and new SNF management technologies will not be developed, at earliest in 2030 the Lithuania will start siting process for the national repository. Also the alternative to prolong interim storage of SNF will be considered (depending on status of containers and storage facilities).

The national research program (2003-2007) managed by RATA (Radioactive Waste Management Agency) for assessing the possibilities of spent nuclear fuel and long-lived radioactive waste disposal had the following principal objectives:

- To describe inventory and characteristics of waste to be disposed in a deep geological repository;
- To identify and characterize geological formations in Lithuania suitable for waste disposal;
- To select one (or two) of these formations and to develop a generic design of the repository;
- To develop first preliminary generic safety assessment of the repository;
- To identify few potential sites in Lithuania suitable for deep repository;
- To prepare an information programme aiming at public understanding and acceptance of the long-term plan.

Several potential geological formations are available in Lithuania: crystalline basement, clay and anhydrite as well as rock salt (Permian evaporate) deposits. The research conducted with the assistance from Swedish experts (2000-2005) showed both, the Triassic clay formations, and the crystalline basement rocks to be promising. The two geological media merit further studies [Investigations of possibilities, 2005]. Implementing research on SNF disposal in Lithuania, the LEI with the assistance of Swedish experts proposed the concepts of deep geological repository in clay and in crystalline rock formations. Proposed concepts foreseen the emplacement of SNF in steel or copper canisters surrounded by the bentonite and located at certain depth in geological environment. There is no final decision on the long-lived ILW disposal in Lithuania also. The ILW (and spent graphite) are planned to be stored in the steel containers at interim storage facility until the final decision will be made. According to proposed generic repository concept of RBMK-1500 spent nuclear fuel (SNF) disposal, the long-lived intermediate level waste (ILW) could be disposed at the same repository at certain distance from SNF emplacement tunnels. While analysing the possibilities of SNF disposal in Lithuania, the costs assessment of geological repository installation was carried out and generic repository safety assessment was initiated.

The main objective of initial studies was to demonstrate that in principle it is possible to implement a disposal of SNF in Lithuania in a safe way. The objective does not imply that disposal of SNF will take place in Lithuania. Which option shall be used for disposal is to a large extent a political decision, and these investigations will be an important input to such decision once required. The following main conclusions were made during the studies [Poskas, et al., 2012]:

- Employing present technologies it would be possible to dispose SNF and other long-lived radioactive wastes in the repository built in the crystalline basement of Lithuania. Modelling of safety relevant radionuclide migration shows that doses to humans will not exceed the existing dose restrictions. Clays having very good confining properties are an alternative media to the crystalline basement.
- The internationally agreed safety standards that ensure protection of human health and the environment have been applied. Despite a scientific evidence of achievable safety, the implementation of a geological disposal encounters difficulties because of lack of confidence from the politicians and the public.

The results and conclusions of the work done in 2003-2007 were taken into consideration during the preparation of the national research program for 2008-2012. The following aspects foreseen to be carried out also [RATA, 2007]:

- Analysis of the possibilities of managing radioactive waste through joint efforts of several countries;
- Feasibility study of borehole disposal of spent sealed sources;
- Analysis and synthesis of information about spent nuclear fuel and other long-lived radioactive wastes, and assessment of properties important for disposal of these types of waste;
- Updating a concept of disposal of spent nuclear fuel and long-lived radioactive waste;
- Development of detailed geological investigation programme, development of database of geological investigation results;
- Performance of studies of geological formations in the Ignalina NPP region (in situ, laboratory tests), analysis of geological survey, identification of prospective geological formation;
- Safety assessment of geological repository for SNF and intermediate level waste;
- Updating the estimation of costs related to implementation/construction of repository.

Unfortunately not all topics will be covered during the planned period. The program of necessary geological studies was elaborated in “Geological Investigations Program for Possible Underground Disposal of Long-lived Radioactive Waste” in 2011. This program contains a regional overview of geological conditions in Lithuania and several potential regions suitable for geological disposal of spent fuel and other long-lived waste are identified. Also, it contains an outline of long-term site investigations.

The investigations were/are continued by LEI in the scientific research project “The use of numerical models in support of site characterization and performance assessment studies of geologic repositories”, “Treatment Requirements for Irradiated RBMK-1500 Graphite to meet Disposal Requirements in Lithuania” coordinated by IAEA and are currently being carried out under the projects: “Treatment and disposal of irradiated graphite and other carbonaceous waste (CARBOWASTE)”, “Fate of repository gases (FORGE)” financed by the Seventh Framework Programme (FP7) of the EU.

In the future the rest listed aspects of the national research program are supposed to be analyzed.

## 8.7 NETHERLANDS

### 8.7.1 Radioactive waste management principles in the Netherlands

The overarching policy on radioactive waste management in the Netherlands is that all hazardous and radioactive waste must be isolated, controlled and monitored. The Dutch policy is based on a document that was presented to the parliament by the Government in 1984 [15]. This policy document covered two areas. The first concerned the long-term interim storage of all radioactive wastes generated in the Netherlands, and the second concerned the Government research strategy for eventual disposal of these wastes. Thereby it is required that the radioactive waste must be disposed of in a long-term retrievable manner.

In the policy document of 1984 the government decided to confer the responsibility for the safe management of radioactive waste to a centralized organization, COVRA, the Central Organisation for Radioactive Waste. COVRA operates a facility at the industrial area Vlissingen-Oost in the south-west of the country. The main considerations underlying this decision were:

- the relatively small amounts of radioactive waste being produced in the Netherlands
- waste management requires specialist care, and this is easier to achieve in one central facility.
- financial reasons, since one central facility can be operated at lower cost.

Since 1984 COVRA collects, treats (if required) and stores all low and intermediate level wastes in purpose-built interim storage facilities in Borssele. In 2000 a storage building for the storage of very low level radioactive waste from ore processing industries was commissioned (TENORM waste)[1]. A dedicated facility for the long-term storage for (vitrified) high level waste, the HABOG (Highly-radioactive Waste Treatment and Storage Building), went into operation in 2003. The vitrified high level waste originates from the reprocessing, in France, of the spent nuclear fuel from the only nuclear power reactor in the Netherlands, the Borssele NPP. In addition, the construction of a storage facility for depleted uranium started in 2003 and the facility became operational in 2004. As of 15 April 2002 the State is 100% owner of COVRA.

The cumulative waste volume in the Netherlands is at present only a few thousand $m^3$. For such a small volume it is not economical to construct a deep geologic disposal facility, as the costs are mainly determined by the construction costs of such a facility. However, the waste volume collected in a period of 100 years of nuclear waste production is judged as large enough to make a disposal facility viable. Moreover, the recent decision to postpone the closure of the Borssele NPP to 2033 implies an additional 30 years of production of high level waste, as well as an additional 30 years of cost contribution to the disposal fund.

[1] Only applicable for VLLW that is required to have a license

An interim storage period of 100 years allow the heat-generating waste to cool down to a situation where cooling is no longer required. In addition, the 100-years period of accumulation of a capital fund is supposed to be sufficient for the then desired level to construct a deep geological repository. This brings the financial burden for today's waste to an acceptable level.

### 8.7.2 Stage of the DGD development

The Dutch concept for a Deep Geological Repository (DGD) for disposal of radioactive waste is still being developed. In the Netherlands attention mainly has been focused on suitable salt domes in the northern part of the country and Boom clay layers in the south and central part. Since the Dutch radioactive waste will be stored for a period of at least 100 years the determination of a suitable concept is at present not a critical issue. However there have been several options investigated in the Netherlands, mainly within the former PROSA and CORA programmes (focus mainly on rock salt), and currently running OPERA program (Boom clay). The different options include the concept of retrievability.

### 8.7.3 Description of the reference concept – rock salt

The research on radioactive waste disposal in the Netherlands started in the early seventies. It was pointed out that rock salt formations in the Netherlands could serve as host rock for a disposal facility. In the late 1980's the VEOS study (Safety evaluation of disposal concepts in rock salt) has been performed in the Netherlands [16]. The aims of this study were the evaluation of the post-closure safety of some possible disposal concepts and the determination of relevant characteristics. VEOS used a scenario approach followed by a deterministic consequence analysis and several deterministic sensitivity studies.

In the early 1990's a generic probabilistic safety analysis, PROSA, [17] of the Dutch generic reference disposal concept has been performed. The PROSA study determined the radiological effects on humans and derived safety relevant characteristics of a disposal concept for radioactive waste in rock salt. These characteristics have been derived from sensitivity analyses of the radiological consequences of some disposal concepts in rock salt formations. The PROSA study was restricted to the safety in the post-closure period.

The PROSA study was carried forward and extended in the CORA program [18], in which the options for retrievable storage and disposal of radioactive waste in the Netherlands were investigated, both for a salt-based and clay based repository. The concept adopted in the CORA program is shown in Fig. 4.

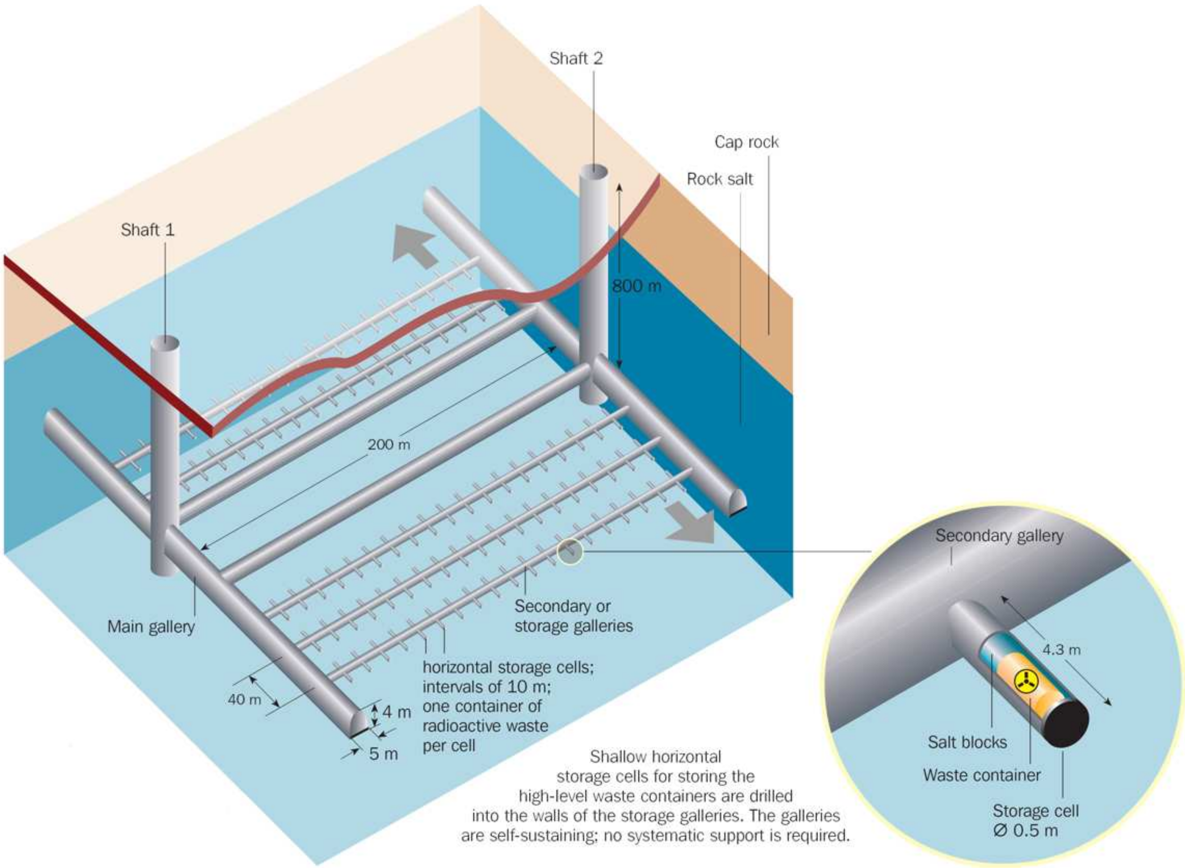



**Fig. 4. Rock salt based option for retrievable disposal of radioactive waste**

The underground facility is similar to a conventional mine and consists of two vertical shafts and a network of galleries and disposal cells in a rock-salt formation at a depth of approximately 800 metres.

The disposal concept contains short horizontal disposal cells drilled into the side walls of a gallery, each accommodating a single vitrified HLW container. The annular space around the container would then be filled with crushed salt and the cell is sealed off with a salt blocks, i.e. the sealing plug.

Salt shows creep under high pressure of the overburden and will gradually compress any voids thereby ensuring that the waste will eventually be sealed off regardless of human intervention. In other words, the long term disposal in salt offers the advantage of a fail-safe situation. However, creep is a slow process and experience shows that, with adequate maintenance, mines in rock salt can be kept open for more than 100 years.

In separate studies for the EU FP6 projects PAMINA and THERESA, NRG and the University of Utrecht performed specific studies on the behaviour of the envisaged rock salt plugs for sealing of the disposal cells. Both experiments and modelling efforts revealed that the salt creep under high pressure will compress the sealing plugs and surrounding voids, resulting in a state of the plugs comparable to that of natural rock salt. However, as a function of the moisture content and heat input from the radioactive waste, this process may take up to several hundred years. Up to that time, the so-called “time to plug closure”, the porosity and permeability of the plugs are enhanced compared to that of rock salt.

The studies performed in the CORA program did not reveal any factors prohibiting the technical feasibility of the three options for retrievable disposal examined: long-term surface storage and deep underground disposal in either salt or clay deposits. For the underground options retrievability requires however extra facilities. In addition a back-up surface storage may be needed to accommodate any waste retrieved from the underground disposal facility. Based on to-day’s knowledge retrievability can only be guaranteed for a few hundred years.

The already mentioned the “time to plug closure” may take up to several hundred years. During that time frame the sealing plug may be susceptible to enhanced brine inflow and outflow, if present. Other topics that were investigated in the PROSA and CORA programs were extensive analyses of FEPs and alternate scenarios (flooding, human intrusion), probabilistic safety assessment methodologies, and societal aspects of geological disposal. Future R&D efforts in the Netherlands will focus on a better fundamental understanding of the salt creep phenomenon of the sealing plugs and surrounding voids, and the consequences for the long term safety of a salt-based repository.

### 8.7.4 Description of the reference concept – Boom clay

Since July 2011 the new Dutch national programme OPERA has become effective [19]. OPERA (“Onderzoeksprogramma Eindberging Radioactief Afval” – Research Programme for the Disposal of Radioactive Waste) is a five-year R&D programme financed by the Dutch government and utilities, and focuses on the feasibility of hosting a geological repository in Boom clay layers in the Netherlands. The OPERA program builds on previous experiences gained in the CORA program, and separate studies performed for the Dutch government [e.g. 20] and within various projects of the EU Framework programme, e.g. BENIPA and PAMINA. In addition, the OPERA program benefits to a large extent from the decades of experiences that have been gained in Belgium with respect to R&D on geological disposal in Boom clay.

The aim of the OPERA program is the development of a Safety Case for a Dutch national geological repository for radioactive waste in the available host rocks, i.e. Boom clay, and, to a lesser extent, rock salt. The objective of the research is to collect and develop evidence and arguments to evaluate the long-term safety. The research therefore involves both technical as well as societal aspects. The Safety Case will be established and elaborated for the present conceptual stage of the Dutch DGD program. This implies that the R&D will be performed for a generic design of the foreseen facilities including the option of waste retrieval, and for host rock characteristics at a generic depth of at least 500 m. Siting, operation and closure will be elaborated in subsequent programs.

An outline of the present disposal concept is shown in Fig. 5, whereas Fig. 6 depicts the disposal cells for vitrified high level waste and for low and intermediate level waste.

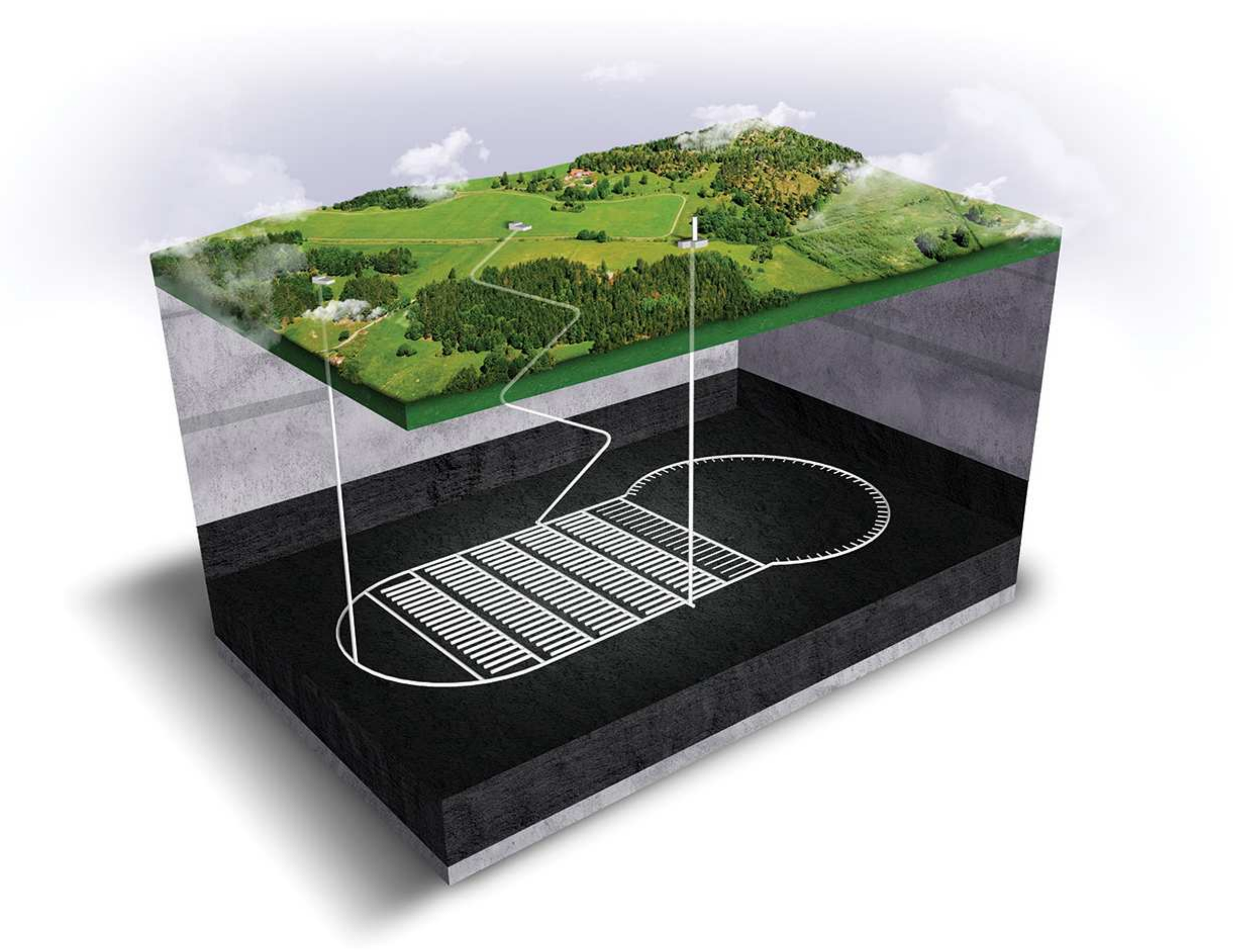


**Fig. 5. Artist impression of a geological repository for the disposal of radioactive waste in Boom Clay**

**Fig. 6. Artist impression of the disposal cells for HLW (left) and LILW (right)**

The OPERA program is organized in seven Work Packages which comprise most if not all elements that are considered important in modern Safety Cases for geological disposal of radioactive waste. The areas where R&D actions will be conducted in OPERA are treated in.

## 8.8 SLOVAKIA

### 8.8.1 SNF management and Disposal of radioactive waste

Slovakia operates 4 units of Nuclear Power Plants of VVER 440. Two units are under constructions (intended time for putting into operation is 2013-14). Two more units (NPP V1 in Jaslovske Bohunice) are decommissioned as well as the demonstration NPP A1 in Jaslovske Bohunice (150 $MW_e$, HWGCR, shut down in 1978).

The basic concept of spent fuel management in the current strategy can be characterized by the following points:

1. An open fuel cycle is applied in operation of nuclear reactors in Slovakia . Currently, in the Slovak Republic is not possible to implement a closed fuel cycle, because the VVER-440 reactors in Slovakia are not licensed to use MOX fuel (or fuel with regenerated 235U after reprocessing).
2. In the spent fuel management, there is not considered its shipping for reprocessing abroad, followed by the return of high level waste for its consecutive management.
3. Short-term storage of spent fuel for 3-7 years after its removal from the reactor is implemented in pools at reactors which are located on each unit.
4. Long-term storage of spent fuel (tens of years) away-from-reactor before its intended geological disposal. Currently, the spent fuel from both Slovak NPPs in operation (V2 in Jaslovske Bohunice and EMO 1,2 in Mochovce) is stored in the spent fuel storage facility (wet type) in Jaslovske Bohunice. There is an intention to erect the spent fuel storage facility also in the Mochovce site.
5. Slovakia considers the dual-track solution for the final step of spent fuel (and radioactive waste not disposable in existing near surface repository) management:
    a) Disposal in the deep geological repository developed, sited and constructed on the Slovak territory,
    *b)* Participating on activities potentially leading to implementation of shared geological repository (SAPIERR projects, activities of the Working Group studying and proposing various aspects of establishing the European Repository Development Organization).

### 8.8.2 Research Project of Deep Geological Disposal in Slovak Republic

Studies on the possibility of a DGD in the former Czechoslovakia had begun in the 1980s. After the separation of the Czech and Slovak Republics in 1993, research and development for a DGD in the Slovak Republic began in 1996 by first desk studies, along the lines of former federal activities.

In 1996, the program started with a critical review of information (no field investigations) and included a survey of published and archival data on: regional geology, hydrogeology, engineering and geophysics. The results identified 15 areas potentially suitable for a DGD, which were spread among granitic (7), clayey (4), metamorphic (3), and flyschoid (1) formations. The next years focused on screening of the candidate sites characteristics using the archive data, very limited field studies and measures (geophysical profiles and a few shallow boreholes).

Based on this analysis, three areas (five localities) were determined as prospective for the DGD. Three localities are situated in granitic rocks and two in argillaceous Neogene complexes.

The preliminary conceptual design for a DGD at a hypothetical site with two alternative geological host environments has been developed. Conceptual ideas for the DGD operational phase were focused on various aspects of surface and underground activities, approached from the perspective of technology, economy and feasibility.

Physical and chemical properties of source term and radionuclide vector of the SNF assemblies has been described. Activity of selected isotopes, total SNF activity, thermal power and the contribution of selected isotopes were calculated over various time periods after refueling. Potential leaching mechanism from SNF and vitrified or cemented matrixes of HLW were reviewed.

As the first step in the near field study, the critical review of available information and current research has been done. As a result, the carbon-steel container coated with a nickel layer has been proposed for the outer wall, with an inner wall made of stainless steel.

Far field research as well as three-dimensional modeling of the geological barrier was performed by the Slovak Geological Survey of Dionyz Stur. Interaction between host environment (granites or clays) and engineering barriers and possible alterations was analyzed.

Safety analysis were based on studying of internationally accepted safety assessment methodology, i.e. methodology of development of scenarios, including features, events and processes (FEP) review, followed by selection of appropriate conceptual and mathematical models for safety assessment.

In few studies the public involvement and confidence building were addressed, key aspects of public involvement had been emphasized: information, communication, participation, acceptance and compensation.

The DGD development program was stopped in 2001. In subsequent years few research activities, concerning very limited geological survey of potential sites was carried on. Limited field investigations (geophysical measurements electric, gravimetric, magnetic, seismic and shallow drilling down to 250 m, including hydrogeological and geophysical logging) were performed. Further reduction in the number of sites and narrowing of their surface is expected for the future, but two alternative host environments remain being considered.

Unfortunately, despite of national nuclear program development and despite of the fact that the strategy of back-end of nuclear energy (as approved by the Government in May 2008 or as it is updating at the present time) recons with the project resurection, there are practically no activities leading to the geological disposal at the present time (June 2012). In contrast to other countries with similar structure of peaceful use of nuclear energy, Slovakia has not yet established the subject (agency) responsible for development, implementation, construction and operation of radioactive waste/spent fuel disposal facilities. National near surface repository, together with many other activities and facilities (operation and consecutive decommissioning of NPP V1 in Jaslovske Bohunice, decommissioning of NPP A1 in Jaslovske Bohunice, operation of the spent fuel storage in Jaslovske Bohunice, operation of radioactive waste treatment and conditioning facilities in Jaslovske Bohunice and Mochovce, implementation of a new NPP) is operated by the JAVYS, a.s. company. This company was entrusted by the Ministry of economy (fall of 2011), according to amended provision of the atomic act, to be responsible for disposal in the Slovak Republic.

# 9 Appendix 2: National safety issues and R&D actions

## 9.1 BELGIUM

The present section identifies the main areas where R&D actions may be undertaken to support the technical review of the safety case for the future Belgian Geological Disposal Facility. This identification is based on the three following steps. The first puts the R&D activities of the Belgian regulatory body (FANC and Bel V) into context and describes the followed R&D methodology. Secondly, the key technical issues identified at the current stage of the repository development are highlighted. Finally the main areas for which R&D actions are needed in priority are defined.

### 9.1.1 R&D context and methodology

FANC and Bel V, constituting together the Belgian Regulatory Body, have both an important effort in R&D. By means of periodic meetings, FANC and Bel V keep each other mutually informed about on-going and planned R&D activities and participations.

By Law of 14 April 1994 on the Protection of the Public and the Environment against Radiation, FANC is responsible for maintaining scientific and technical documentation in the area of nuclear safety and radiological protection. It is also responsible for fostering and coordinating R&D and establishing relationships with national and international research organisations. Within this framework, FANC performs various R&D activities ranging from the participation in international working groups (e.g. IGSC, Clay-Club,…) and projects (e.g. FORGE) to the development and follow-up of in-situ experiments at the Mont Terri rock laboratory (CH) through a collaboration with the Belgian Nuclear Research Centre (SCK•CEN).

The R&D activities of Bel V are primarily related to the development and the maintenance of expertise in nuclear safety and to a lesser extent in radiation protection (the latter being covered extensively by FANC). Bel V’s R&D activities are fully integrated in its Quality Management System. Within that framework, Bel V issues about every 5 years a R&D Strategy, a yearly R&D Program (defining Task Leaders and budgets) and a yearly R&D Report. The overall R&D effort foreseen by Bel V has been recently significantly increased to about 10% of the total available time for the technical staff. Within the framework of radioactive waste disposal safety, Bel V participates in international working groups (like

IAEA PRISM and more recently IAEA GEOSAF) and previously participated in international projects like 6th FP EC MICADO, 6th FP PAMINA, …

Within the framework of the safety of radioactive waste disposal, a two-fold R&D methodology is followed by the Belgian regulatory body (FANC and Bel V). The Key technical issues and R&D priorities are first identified principally based on the following elements:

- the milestones of the Belgian waste management programme (see section 5.1) and their associated objectives and decisions;
- the safety requirements established in royal decrees and guidance applying to waste disposal;
- the safety issues associated to the safety concept developed by the operator.

R&D actions are then identified for each key issue. The three main types of R&D actions that may be undertaken can be summarized as follows:

A. Literature survey, participation to conferences or international working groups (IAEA, OECD).

Such R&D actions are undertaken to cover fundamental issues that are already addressed by the international community.

B. Sub-contract to other organisations (universities and research centres).

These R&D actions are undertaken to cover specific key issues for which the regulatory body (FANC and Bel V) does not have the necessary resources.

C. R&D within regulatory body (FANC and Bel V) or in collaboration with other organisations (Framework Programmes of EC…).

The choice of the action type is dictated by the importance and priority of the issue and by the availability of resources and competences within the regulatory body (FANC and Bel V).

### 9.1.2 Key technical issues

This section summarizes the technical issues considered as important by the regulatory body (FANC and Bel V) at the current stage of the repository development.

As mentioned in section 5.1.1, up to now no Safety Case has been submitted for the foreseen geological repository. The identification of the key technical issues is therefore principally based on the safety functions currently defined by the operator: the physical containment of waste, the delaying and the spreading of the releases, the limitation of water flow, the ensuring of stable conditions and the limitation of access. On the basis of these safety functions, technical issues that require further investigations may be identified by the regulatory body. These issues are mainly associated to the processes on which rest the safety functions and the potential perturbations that may compromise the long-term safety of the foreseen repository. In addition, technical issues are associated to the acquisition of key data on the host-rock and on the Engineered Barrier System (EBS), and to the modelling of the long-term radiological impact of the foreseen waste disposal facility.

Another starting point to identify key technical issues and develop R&D actions is the latest "Safety Assessment and Feasibility Interim Report" (SAFIR 2, see [7]), as well as its review by the Nuclear Energy Agency and the regulatory body [8]. Whereas the repository design and safety assessment have evolved since the SAFIR 2 report, several key technical issues highlighted by the NEA are still important attention points for the Belgian nuclear authorities.

#### 9.1.2.1 DATA ACQUISITION

Technical issues consist in acquiring key data on the reference and alternative host rocks and on the foreseen Engineered Barrier System (EBS). The geochemical compositions of the host rock and the EBS are key parameters for deriving input data (sorption, solubility) of the safety assessment. In addition, data should be acquired on the hydrogeology and biosphere, as their knowledge is essential for determining the input parameters of the safety assessment.

The NEA review of the SAFIR2 report has highlighted that the existence of alternative formations and how much flexibility is available for siting should be further investigated. Moreover, the NEA has stressed that an important point of attention is the transferability of data collected locally to the whole site of geological disposal. Two principal examples are:

- The Clay permeability: measurements at the metre scale have revealed a clay permeability of $10^{-12}$ m/s, whereas the calibration of the regional hydrodynamic model resulted in a value of $10^{-10}$ m/s. Clarification of the clay permeability is important since with the lower value diffusion dominates and with the higher value advection dominates (the transition being around $10^{-11}$ m/s).
- Radionuclide migration experiments have been carried out under specific conditions. Data extrapolation to different reference conditions and longer timescales is therefore surrounded by uncertainty.

#### 9.1.2.2 PROCESSES ON WHICH REST SAFETY FUNCTIONS

Technical issues are equally associated to the understanding of the processes on which rest the long-term safety functions of the foreseen waste disposal system. The safety functions developed by ONDRAF are: the physical containment of waste, the delaying and the spreading of the releases, the dilution and the dispersion of the releases and the limitation of access. The following technical issues have been highlighted:

- the migration of radionuclides in the host-rock and in the Engineered Barrier System (EBS) at ambient temperature and considering temperature gradient representative of those that may occur in the near-field of the foreseen waste disposal system;
- the isolation capacity of the host-rock and the EBS;
- the resilience capacity of the host-rock and the EBS;
- the containment capacity of the EBS.

Concerning the radionuclide migration, the NEA review of the SAFIR2 report has highlighted the following attention points:

- the partial pressure of $CO_2$ in the Boom Clay is a key parameter for deriving input data (sorption, solubility) for performance assessment. This parameter is important to constrain the carbonate system, the pH and the speciation of some important radionuclides;

- the mineralogical description of the host rock is of particular importance, as trace minerals can play an important role in the control of some aqueous species.

#### 9.1.2.3 LONG-TERM SAFETY STABILITY

Technical issues are associated to the potential perturbations that may affect the defined safety functions and therefore compromise the long-term stability of the foreseen geological waste repository. These potential perturbations may be either internal or external to the foreseen waste repository. The technical issues associated to the potential internal perturbations are:

- the influence of the construction works and the exploitation of the waste disposal facility on its long-term safety (conducting concurrent activities, potential clay oxidation/reduction, permeability increase, influence of microbial activity,…)
- the influence of interactions between the waste and the host-rock on the long-term safety of the waste disposal system (the gas generation and migration, the temperature increase, the nitrate plume, the clay radiolysis,…)
- the influence of interactions between the EBS and the host-rock on the long-term safety of the waste disposal system (alkaline plume, container degradation, …)

Moreover, the NEA review of the SAFIR2 report has stressed that:

- the dissolution rate of vitrified waste packages by high pH pore water from the surrounding concrete environment has to be better characterized;
- the compatibility between bitumen waste and the surrounding concrete and clay should be confirmed to ensure that the potential swelling of these waste packages does not jeopardize the long-term safety;
- in the current design, waste will be conditioned in supercontainers which consist notably of two steel envelopes. The corrosion of these steel envelopes is of particular concern for the long-term safety;
- the gas release from waste packages and from the corrosion or radiolysis of engineered barriers is an important point of attention;
- the initial thermal phase may modify the geochemical and mechanical properties of the surrounding clay. The impact of this potential modification on the radionuclide migration should be investigated;
- the introduction of a large quantity of new chemical component will perturb the in situ conditions and therefore the final geochemical conditions may not be the same as before the perturbation, even after the system returns to equilibrium.

Technical issues associated to external perturbations mainly consist of a potential marine transgression, future permafrost/glaciations, future human activities (gas storage and extraction, geothermal energy, …), site erosion, site seismicity…

The NEA review of the SAFIR2 report has highlighted that the proximity of the site to the Roer Valley Graben is an important attention point. Earthquakes that may have most impact in the foreseen repository area are related to the fault activity of the Roer Valley Graben. Another attention point stressed by the NEA is the proximity of the current chosen site to major aquifers. It is therefore necessary to clarify, at the level of policy and regulation, the required level of protection to ensure the acceptability of the repository.

#### 9.1.2.4 Feasibility

The feasibility of the foreseen waste disposal system should be investigated. Key technical issues that should be analysed are the construction strategy foreseen for the disposal facility, the manufacturing of envisioned materials and the strategy for the emplacement of waste and EBS into the geological repository.

#### 9.1.2.5 Assessment

Finally, technical issues are associated to the understanding of the main scenarios, hypotheses and models developed by the operator to assess the long-term safety of the foreseen geological waste disposal facility. Key issues that should be investigated are the assessment methodologies (scenario development, …), the model development (modelling of radionuclide migration, site hydrogeology and biosphere) and the treatment of uncertainty.

Specific attention points are:

- the management of parameter uncertainty: definition of PDFs. Within the framework of the 6FP EC *PAMINA* project Bel V has notably highlighted that the sensitivity of Safety Assessment to the Probability Density Functions associated with input parameters is high. The study stressed that a conservative assumption for PDFs is not possible in general without a dedicated study and detailed justifications;
- the modelling of the radionuclide release from waste packages and spent fuel. This modelling is of particular concern as it has a deep impact on the long term safety of radioactive waste disposal. As highlighted by the 6FP EC *MICADO* project in which Bel V participated, the modelling of the spent fuel Instant Release Fraction (IRF) is an important point of attention;
- the modelling RN migration: consideration of the Boom clay geochemistry. The conceptual model representing the migration of radionuclides inside the Boom Clay should account for the range of geochemical processes. A sorption model specific to the Boom Clay should be developed.

The NEA review of the SAFIR2 report recommends using a more up to date and systematic approach for the development of scenarios and to further develop a systematic management of uncertainties.

### 9.1.3 Identification and prioritization of R&D Actions

In order to support the technical review of the safety case for the future geological repository in Belgium, R&D actions (A, B or C; as defined in section 6.1.1) are envisaged by the regulatory body (FANC and Bel V) for each key issue identified in section 6.1.2. An overall and preliminary description of these R&D actions is proposed in Tab. 1.

**Tab. 1. Description of the R&D actions to be undertaken**

| Key issues | | R&D needs | Priority for regulatory body (H,M,L) | Action types (A, B, C) |
|---|---|---|---|---|
| Data acquisition (characteristics & parameters important for safety, for EBS, reference site and other potential host formations and sites) | | Host rock characterisation<br>Biosphere<br>Hydrogeology | H | A, B |
| | | EBS characterisation | L | A, B |
| Processes on which rest safety functions | | Host formation performance:<br>- Migration of RN (at ambient T° and considering T° gradient)<br>- Low advection<br>- Retention / retardation<br>- Isolation capacities<br>- Resilience capacities | H | A, B (, C)[2] |
| | | EBS performance<br>- Containment capacities<br>- Isolation capacities<br>- Migration of RN<br>- Resilience capacities | L | A, B (, C)[1] |
| Long-term stability | Internal perturbations | Construction works + exploitation<br>- Clay oxidation/reduction<br>- Permeability increase<br>- Microbial activity,… | H | A, B, C[3] |
| | | Waste / Host-rock interactions<br>- Gas generation and migration<br>- Temperature increase<br>- Nitrate plume<br>- Clay radiolysis,… | H | A, B, C[2] |
| | | EBS/Host rock interactions<br>- Alkaline plume<br>- Container degradation,… | M | A, B, C[2] |
| | External perturbation | - Marine transgression<br>- Permafrost/Glaciation<br>- Human activities (Gas storage & extraction, geothermal energy,…)<br>- Erosion,… | H | A, B |
| Feasibility | | Construction | H | A, B |
| | | Materials manufacturing | L | A, B |
| | | Emplacement of waste and EBS | L | A, B |
| Assessment | | Assessment methodologies | M | A, B |

[2] Some specific issues may be investigated through type C actions, whereas a systematic R&D by the Regulatory Body (FANC and Bel V) is not foreseen for this key issue.
[3] For instance, FANC currently collaborates with SCK•CEN for the R&D conducted in the Mon Terri URL.

| | | | |
|---|---|---|---|
| | (scenario development, …) | | |
| | Model development :<br>- RN Migration<br>- Hydrogeology<br>- Biosphere | M | A, B, C[4,5] |
| | Treatment of uncertainties | H | A, B, C[4] |

## 9.2 CANADA

### 9.2.1 Introduction

In Canada, unlike with many other nuclear regulators, the technical and scientific support functions for the Canadian Nuclear Safety Commission (CNSC) are provided by in-house technical staff; there is no separate technical support organization (TSO). The NWMO is responsible for maintaining an active R&D program to keep up to date with the rapid progress in geosciences and safety assessments related to geological disposal. The Canadian Nuclear Safety Commission also maintains a Research and Support Program to address key knowledge gaps for safety analysis of repositories for nuclear wastes.

Canadian law requires regulatory agencies to respond in a short period of time once a proposal for the nuclear waste management is filed. A fair and objective assessment is not possible, without the knowledge and background associated with the years of research and development work that contributes to the final proposal. Therefore, CNSC staff has started a Coordinated Assessment and Research Plan (CARP) in order to be ready for the assessment of upcoming proposals from OPG and the NWMO. The CARP in essence would consist of review of on-going publications from OPG and NWMO at critical milestones, combined with an independent research program through the CNSC research and support funding.

### 9.2.2 CNSC coordinated assessment and research plan (CARP)

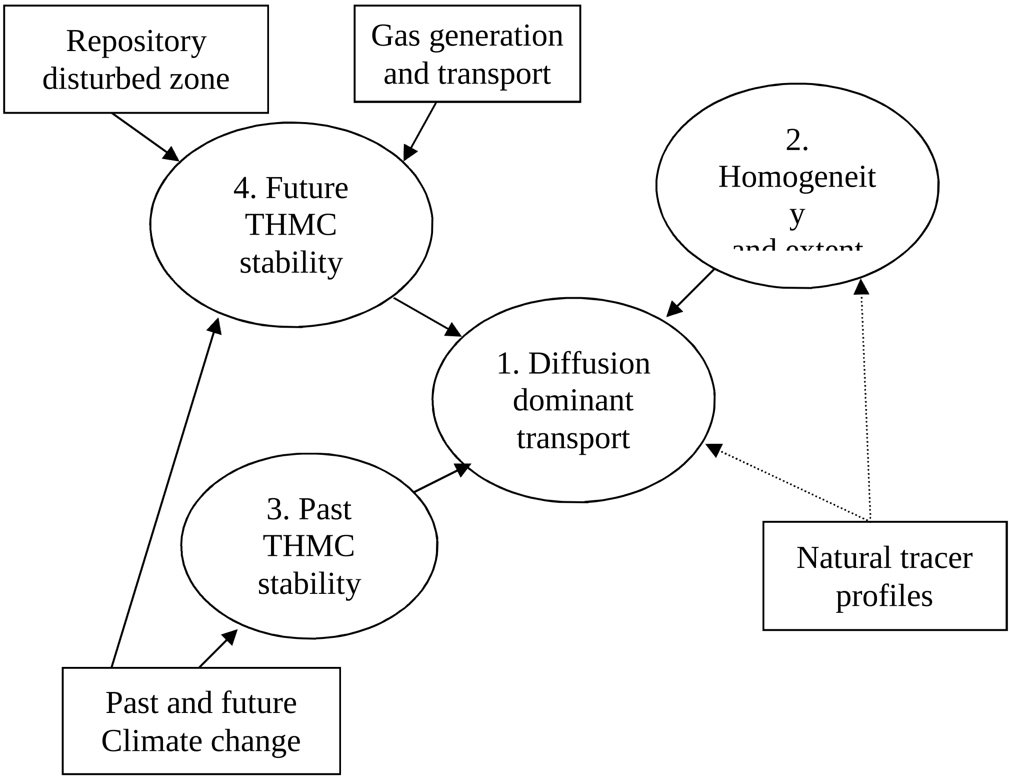
CARP is a multi-year research project with the objective to provide (or reject) confidence that sedimentary rock formations, especially the one in Southern Ontario, possess the attributes that would provide confidence on its long term efficiency and robustness. The attributes are:

- transport of solutes is dominated by diffusion processes, ensuring in some cases of total containment of contaminants from the wastes for hundreds of thousands years. Low permeability and low porosity are necessary properties for a host rock where diffusion is dominant.

[4] Whereas the development of a detailed model for assessing the radiological impact of the geological repository is not foreseen by the regulatory body (FANC and Bel V), the development of basic models for assessing specific key issues is envisioned.
[5] For instance, Bel V has participated in the 6 FP EC *PAMINA* project.

- homogeneity and sufficient lateral and vertical extent of the geosphere. Homogeneity makes it easier to characterize and predict the future behaviour of the geosphere. In addition, homogeneity is usually associated with a low degree of fracturing and/or the self-sealing nature of fractures, which in turn support the attribute of diffusion dominant transport. Sufficient vertical and horizontal extent of the formation ensures longer pathways before the contaminants could reach the biosphere.
- stability of the geosphere for the past millions of years. During those years, the host rock has been subject to tectonic forces, glaciation, erosion, etc. Proving that the Thermal-Hydro-Mechanical-Chemical regimes at hundreds of metres depths have been relatively unaffected by these processes is a strong argument of the robustness of the geosphere, and will also support the attribute of diffusion dominant transport.
- future stability of the geosphere due to the combined effect of the repository, including effects associated with the behaviour of the waste, and external geological forces (e.g. glaciation, seismicity, etc.).

The research subjects and strategy to build arguments for the safety case are shown in the figure below.




*(The items inside the oval boxes are the desirable attributes of the geosphere. The items in the rectangular boxes are research subjects that contribute or counter particular attributes)*

The synthesis of the research results will provide CNSC staff independent knowledge in order to provide the Commission and EA panel staff's opinion on the acceptability of the OPG and NWMO's applications. The research project includes four parallel but complementary components that will contribute to or refute safety arguments for geological disposal in sedimentary rocks (specifically to Southern Ontario). These four components of the project are almost equally important, to give an integrated assessment of the safety case. These four components are:

1. **The study of natural tracers:** Sedimentary rocks were deposited in a marine environment. At most sites, at depths, the saline content of the groundwater is very high in the low permeability layers. The shallower rocks connected to these deeper layers were open to fresh water supply at some time in the past. The concentration of solutes (such as Cl) in many cases shows a classical diffusion profile that proves that diffusion was the only plausible transport process that prevailed for the last millions of years.

2. **The study of past and future impacts of climate change**: Field evidence (for example oxygen-isotope ratio in marine sediments; high water pressure in rock formations at depths of more than a few hundred metres, etc.) demonstrates that the Northern part of the Northern Hemisphere has been subject to repeated glacial cycles during the last millions years, with dominant periods of the order of 100000 years, and with brief intervening interglacial period. It is likely that 120000 years before present, sub-zero average annual temperature develops in Canada, leading to permafrost conditions and the development of an ice sheet with a transient thickness that can attain 3km at its peak. The weight of the ice sheet produced a depression of the earth surface (up to a few hundred metres) and major eustatic variations of the sea level. A rapid glacial retreat (melting of the ice sheet) occurred, 8000 to 10000 years before present. Currently we are still in an interglacial period, however most experts agree on the possibility of a new glacial period within 10 000 to 50 000 years. Glaciation-deglaciation cycles profoundly affect the geochemical, hydraulic, mechanical and thermal conditions in the geosphere. However, past numerical simulations for the Canadian Shield show that these effects are restricted to the upper 200-300 m. In this proposal, we will gather geological data and perform numerical simulations to assess the impact of past and future glaciation-deglaciation cycles on sedimentary rock formations of Ontario. The results of this research will provide confidence (or refute) the thermal-hydro-mechanical-chemical (THMC) past and future stability of such formations at the depths (more than 500m) being considered for radioactive waste disposal.

3. **The study of the Thermal-Hydraulic-Mechanical-Chemical disturbances of the host rock due to the excavation, operation and heat generation from the repository:** The excavation, operation of a repository and the long term heat generation from the emplaced wastes potentially can induce mechanical damage in the vicinity of the repository, enhance its permeability and disturb its geochemical regime. These factors can be a counter-argument to the central attribute of the geosphere (diffusion dominant transport) and the extent and nature of these disturbances have to be further understood.

4. **The study of the generation and migration of gas from the waste repository:** Gas could be generated from a geological repository from the degradation of the waste forms or the corrosion of the waste containers. The pressure in the generated gas

would build up, and induce the formation of either microcracks or macrocracks and would affect the integrity of the geosphere as a long-term contaminant barrier. It is thus important to understand the mechanisms of gas generation and migration and assess their effects on the mechanical (M) and hydraulic (H) stability of the geosphere.

### 9.2.3 Long Term Performance of Bentonite Seals for Geological Disposal Repositories

Geological disposal repositories rely on the host rock and the engineered barriers to contain and isolate the wastes from the biosphere for hundreds of thousands to millions of years. The acceptability of a DGD depends largely on the long term performance of these barriers. The host rock is a natural and major barrier for long term contaminant migration from the wastes. However, preliminary long term safety assessments performed by the NWMO have shown that the bentonite seals used to plug the galleries and shafts of the repository also play an important role to minimize preferential pathways for contaminant transport. This research program is to assess the long term performance of bentonite material used to seal galleries and shafts of deep geological repositories and to verify the full scale constructability of these seals. The project will be performed in collaboration with the IRSN (Institut de Radioprotection et de Sûreté Nucléaire, France). This proposed research program has three components:

- A program of in-situ tests called SEALEX to be performed at the IRSN's Tournemire underground research laboratory (URL).
- A program of laboratory tests at the Ecole Nationale des Ponts et Chaussées (ENPC), Paris.
- Mathematical modelling of the in-situ tests.

The SEALEX project consists of the construction of large boreholes (total of seven) at the Tournemire URL, the emplacement of prefabricated bentonite or bentonite/sand mixture blocks in the borehole, the injection of water to saturate the blocks with continuous monitoring of the hydraulic and mechanical evolution of the bentonite during water injection, and the dismantling of the boreholes for potential analysis using micro-analytical methods. The whole project will last more than ten years.

In the future, CNSC will continue to develop research projects based on issues and uncertainties identified from reviewing the safety case of the OPG DGD and the pre-licensing review of the NWMO APM, and CNSC's needs.

## 9.3 CZECH REPUBLIC

### 9.3.1 Safety strategy

The radiological protection requirements for the operational period of a disposal facility and the related safety criteria are the same as for any licensed nuclear facility. In radiological protection terms, the radiation source is under control during the operational period: releases can be verified, exposures can be controlled and actions can be taken if necessary. The engineering and practical means of achieving protection are well known. The optimisation of protection, that is, ensuring that radiation doses are as low as reasonable achievable (ALARA principle) is required to be considered in the design of the disposal facility. No releases, or only very minor releases, of radionuclides and no significant doses to members of the public are expected during the normal operation of radioactive waste disposal facilities. Even in the event of accidents involving the breach of a waste package, releases that would have impact outside the facility are unlikely. This fact has to be confirmed by means of safety assessment, which shall be sufficiently detailed and comprehensive to provide the necessary technical input for informing the regulatory body at each step. The doses associated with the transport of radioactive waste are required to be managed in the same way as the doses associated with the transport of other radioactive material complying with the requirements of the IAEA Regulations for the Safe Transport of Radioactive Material and Czech legislation.

Strategy for long-term safety is now based primarily on IAEA documents and NEA/OECD recommendation. NRI as a main contractor for preparation of the last safety case study based its strategy primarily on its participation in 6 FC EP PAMINA project, NEA/OECD project MeSA and on wealth knowledge published in Swedish and Finish technical reports due to great similarity of Czech and their disposal concepts. But contrary to Swedish or Finish safety strategy, the Czech concept cannot rely primarily on canister safety functions due to expected less corrosion resistance of steel than copper. The same importance is laid therefore on primary safety functions of buffer and backfill and granite host rock for retardation, dilution and dispersion of radionuclides as for their containment in canisters.

UJV Rez, a.s. (former NRI) is now one of the major subcontractors both to Czech regulatory body (SUJB) and to RAdioactive Waste Repository Authority (RAWRA) in fields of radiological protection and nuclear safety. The role of NRI as an independent technical safety organization must be therefore limited only on the issues, in which it will not be directly involved as a contracting organization for RAWRA. In these areas the role of technical support organizations will have to play other independent research organizations. This question of independency of NRI is now discussed in the Czech Republic not only for the evaluation of safety of radioactive waste repositories, but also for the evaluation of safety of nuclear power plants. There is under discussion that the role of independent TSO for some of issues where NRI cannot be considered to be independent could play recently established Centre of Research, Rez as a daughter company of NRI. The Czech Republic belongs to smaller countries in Europe and therefore it is not probable that it could be established only one TSO that would cover all safety case issues.

NRI will prepare for the role of TSO mainly due to the intention of RAWRA to increase their capabilities in conducting safety case by own staff.

### 9.3.2 Scenarios derivation

Disposal system in the mentioned safety case study was defined as the system characterised by all features, events and processes (FEPs), which determine scenarios that can lead to the release of radionuclides into the environment and to cause exposure of people. No systematic approach was, however, applied for identification and screening of FEPs in mentioned preliminary safety case report. The analyses of FEPs and possible screening were made only on the basis of NRI expert judgements. Instead of central/normal scenarios covering all the FEPs which can occur with high probability in a repository the following scenarios were derived and assessed:

1) Scenario caused by earthquake leading to the immediate failure of various numbers of the canisters in the time of mean lifetime of canisters
2) Scenario caused by denudation or erosion leading to the substantial shortening of radionuclide pathways to the environment
3) Scenario caused by the formation of preferential pathways in part of buffer in boreholes with immediate failed canisters due to earthquake
4) Intrusion scenario leading to the failure of one canister and buffer immediately after the end of institutional control of the repository (300 years)

Basic safety functions for safety important components (canister, buffer and backfill, host rocks, structure materials) were identified. The safety functions have not been, however, clearly arranged into a structure, which would allow better considerations of their importance.

No systematic approach was adopted in this issue. One of the main roles of TSOs is therefore to identify independently (in addition to implementer) possible FEPs and consequent scenarios that could lead to unexpected exposure of people or the environment.

### 9.3.3 Input parameters

It is generally accepted, that a safety assessment is such trustworthy as are the data applied for calculations. Needed input data for Czech safety case were divided into the following categories:

1) Inventory and characteristic parameters of waste
2) Input parameters for canisters
3) Input parameters for buffer and backfill
4) Input parameters for host rock
5) Input parameters for biosphere

All the data used in calculations were burdened by a higher or a lesser uncertainty caused either by their variability or some ignorance or by insufficient knowledge of their variability or ignorance. Both types of uncertainties must be addressed in safety assessment and audited by independent researchers, especially those which requires after measurements some interpretation or extrapolation. A good example form the Czech concept is the lifetime of canister and distribution of failures of individual canisters in a repository. These parameters are highly important for the results of safety assessment of this concept. The determination of this data usually requires sophisticated methodologies to enable extrapolation of short term laboratory or in-situ data to very long expected lifetimes of canisters. The interpretation of natural or anthropological analogues is not ambiguous as well. The wrong interpretation of data can significantly affect the results of safety assessments.

The R&D actions of TSOs should focus primarily on the issues with high importance for safety assessment for a given concept and with possible ambiguous interpretation. It is not necessary to conduct always fully independent experiments, but it is necessary to conduct independent evaluation of the results.

### 9.3.4 Inventory and characteristic parameters of waste

It is well known that inventory of wastes is a basic parameter for safety assessment, the most crucial being the inventory of instant release fraction (IRF) from spent fuel assemblies (e.g. $^{129}$I) The uncertainty increases with a high burn-up of spent fuels assemblies. It is difficult for most of non-specialized EU organizations to conduct own research in this area, because the experiments are very expensive and requires special equipment available only in several laboratories in the world. We consider that main role of TSOs in the area would be to get a good understanding of this issue from a comprehensive literature survey and possibly to participate in research activities of specialised laboratories.

The characteristic of waste important for safety assessment are highly waste specific. Different approaches must be applied for glass, bitumen, or cement matrices. The interpretation of the experiments is also not always straightforward and the results of the experiment might require different interpretation and possible independent experiments conducted by implementer and TSOs. Decision on conducting own research depends on the importance of the waste matrix in a safety case.

#### 9.3.4.1 Input parameters for canisters

In the Czech concept, multilayer steel based canisters with Ni alloy protective layer were proposed. The corrosion behaviour of materials in an anaerobic environment of deep repository of granite host rock is not very well known. The experimental results conducted in laboratories or in-situ requires not straight forward interpretation and extrapolation. Anaerobic corrosion of iron metals is in addition connected with gas generation that can be detrimental to other components of a repository. Especially here can be the results affected by wrong interpretation by researchers and implementers. For Czech concept, we consider this issue as very important for safety and therefore TSOs should be closely involved in experiments conducted by the implementer or implementer contracting organizations with independent evaluation of data used in safety assessments.

#### 9.3.4.2 Input parameters for buffer and backfill

In the Czech DGD concept bentonite was selected as a candidate material both for backfill and buffer. Bentonite has several safety functions in a repository related to retardation of bentonite and to safety functions of canisters as well. Local types of Ca, Mg bentonite are at the present time the main presumed buffering material for the concept solution of DGD in the Czech Republic. It would be applied in the form blocks/ prefabricated elements as the primary surrounding layer of canister as well as the secondary distance and sealing blocks in the remaining space of the disposal drift (both disposal concept variants). Uncertainties of parameters and behaviour of bentonite sealing can be divided into two large groups: 1) before an UOS failure and 2) after it. It is presumed, that bentonite will be sufficiently stable for a long time (minimum ten thousands of years).The first significant uncertainty is illitisation of smectite – a mineral transformation, which can essentially influence the retardation properties of bentonite barriers. It is a complicated long-term process depending on a number of factors (mainly it is potassium in porous water, high temperature and pressure). The second uncertainty of bentonite stability is in processes of mineral transformations in the alkaline front environment. Also in this case it is a complicated long-term process depending on a number of factors (it relates mainly to composition of penetrating alkaline waters, pH, single minerals transformation trend). The main source of the alkaline front is represented by structural materials (such as plugs of drifts or galleries, and/or structural reinforcing elements, as well as injecting of rock environment discontinuities). The third uncertainty is a change of chemical status of bentonite due to exchange of ions on clay minerals, e.g. ions from penetrating underground water. It can result except other to significant changes of geotechnical parameters (swelling, permeability). Another important uncertainty is a change in bentonite behaviour by influence of corrosion products of UOS. These uncertainties can be quantified only on the basis of a long-term systematic research directly with the candidate materials.

Influence of thermal or chemical load on local bentonite was examined up to now only in conditions which are widely different from the real conditions in DGD. In some aspects of THMC issue of bentonite buffering, a problem is also in not sufficient understanding of processes running in the close vicinity of UOS (e.g. gap homogenization).

The most important issue for Czech concept is, however, insufficient knowledge of local bentonite behaviour in a comparison with bentonite tested in Swedish or Finnish DGD programmes.

The evaluation of the results is also not ambiguous as in the case of evaluation of canister lifetime. The laboratory R&D action of TSOs should be closely connected with in-situ experiments conducted by implementer or implementer contracting organizations with independent evaluations of results in respect of data used in safety assessments.

#### 9.3.4.3 Host rock parameters

The most crucial data for evaluation of contribution of a host rock to safety of a repository are hydrogeological data and the data connected with migration of radionuclides (solubility, retardation coefficients, diffusion in matrix, wet surface of fractures, etc.). The characterization of candidate sites in the Czech Republic has not started, yet. For the preliminary safety assessment the available data from near surface granite sites and abroad data were mostly used. The interpretation of data obtained in site characterization programme is a necessary step to achieve data needed for safety assessments. Selection of rightly chosen geometry for evaluated region/area, its simplification into a network of geological elements and choice of representative transport routes is more or less connected with the subjective choice of a given expert. All consideration and assumptions are based on presumed facts, on which another person can have a different opinion. The involvement of independent experts of TSOs in evaluation of the results obtained is therefore very important for credibility of any safety assessments.

#### 9.3.4.4 Biosphere data

It is well that biosphere data depends primarily on local hydrological and climate evolution surface, which is strongly affected by human actions. All the data related to biosphere are therefore strongly affected by human society behaviour. There is therefore agreement not to consider that biosphere has any safety function in a disposal system. Despite this, biosphere plays a very important role for providing evidence for safety of any repository. It will have to be achieved some compromise between implementer and regulatory bodies, which data to use in safety assessments in order the results were not too much conservative or too much optimistic. TSO experts can provide invaluable independent opinions in this field.

### 9.3.5 Modelling strategy

Mathematical simulation of processes is a necessary tool for predicting and better imagining processes running in the disposal system for thousands of years. The credibility of safety assessment therefore depends on uncertainties of used conceptual, mathematical models and software codes. The uncertainties for all the models can be divided into the following categories:

1) Uncertainty of abstraction/simplification of the reality leading to conceptual models
2) Mathematical representation of processes included
3) Input data quality
4) Applied software and its verification and validation

Usually common analytical approached models cannot be used for description of real processes or their use is very limited. Therefore numerical model based on a thorough description of rock environment by in-situ and laboratory work comprising information from all scale levels (laboratory samples in size of centimetres, disposal facility in size of meters, and regional scale in size of tens up to hundreds of kilometres) are mostly utilized.

It is understandable that a complete description of rock environment in the scale of DGD is not achievable, but even perfectly described rock environment must be simulated with some simplification due to limited hardware capacity of present computers. As one of the important problem we see a problem with coupling of some processes, for example coupling of flow of water and transport and chemical behaviour of contaminants. Also THMC models are often solved without taking into account chemical changes.

Special attention requires criticality models, which must prove that the system will stay uncritical even after failure of canisters and penetration of water inside.

The use of all models requires comprehensive verification and validation at least against laboratory or in situ experiments. The conceptual, mathematical models and software used in Czech Safety study were not treated in such a way, usually significant assumptions and simplifications were applied, especially for modelling hydrogeology and transport of radionuclides in host rock.

The role of independent TSOs in the field of modelling is probably one of the most important in the process of preparation of safety case. All the models and their verification and validation should be carefully checked out by independent experts.

## 9.4 FRANCE

### 9.4.1 Context

#### 9.4.1.1 Legal and technical framework of the DGD safety assessment

IRSN (Radiation Protection and Nuclear Safety Institute) is the public body in charge of the scientific assessment of nuclear and radiation risks. It is mandated for advising the public authorities and contributing to public policies, for delivering services to other organisations and for developing the research activities necessary to support its scientific appraisal.

The key fields of research relate to safety of nuclear installations and waste, to severe accidents in nuclear reactors and emergency preparedness, to radioactivity and ecosystems and to radiation protection.

In the field of radioactive waste safety, IRSN develops a pluri-annual research programme so as to develop IRSN staff skills and anticipate the needs for new knowledge necessary to perform comprehensive safety reviews of high quality. This research programme, launched initially to support IRSN assessment of Andra’s file on the “feasibility of reversible geological disposal in clay” issued in December 2005, is now structured upon the new main steps related to the development until 2015 of the high-level and long-lived intermediate-level waste repository project as prescribed by the French Planning Act of 28 June 2006 on the sustainable management of radioactive materials and waste. This act plans a licence application to be submitted in 2015 for the creation of a deep geological repository. IRSN research programme is annually updated and periodically reviewed by a scientific committee and organised in order to addressing several “key safety issues” as presented below.

#### 9.4.1.2 Definition of safety research activities

The above mentioned “key” scientific and technical topics should also be of prime concern for the implementer since they relate to “key” safety issues for demonstrating the overall safety of the repository, and the level of funding that the implementer should afford to research activities of concern for safety should be naturally much higher than those of the regulator and technical safety organisation (TSO). This is fully justified by the different respective roles played by both entities but it is of assessor's duty to be able to cover all the safety case issues with care to make appropriate balance between topics that must be addressed by R&D programme or topics that do not require specific R&D development. In this last case, the regulator or TSO should be able to explain why it is not necessary to develop its own research capabilities. In this respect, some aspects are not addressed by IRSN R&D programme because either they relate to conception/construction demonstration tests that are of implementer responsibility or because IRSN considers that the scientific knowledge is sufficiently shared by different stakeholders and well managed by the operator. Considering the elements that justify IRSN R&D programme, 4 categories of major questions are addressed: the adequacy between experimental methods and data foreseen, the knowledge of complex coupled phenomena, the identification and confidence in components performances and the ability of the components to practically meet in-situ the level of performances required. Addressing these questions requires the research programme to be developed along the following lines:

- Test the adequacy of experimental methods for which feedback is not sufficient. The assessment of their validity allows addressing the consistency and degree of confidence of the data produced,
- Develop basic scientific knowledge in the fields where there is a need for better understanding the complex phenomena and interactions occurring all along the life of the repository and their influence on nuclear safety, so as to preserve an independent evaluation capability in these matters,
- Develop and use numerical modelling tools to support studies on complex phenomena and interactions so as to allow IRSN assessing orders of magnitudes of components performances and physico-chemical perturbations but independently than specified and estimated by implementers,
- Perform specific experimental tests aiming at assessing the key parameters that may warrant the performances of the different components of the repository. Such experiments are designed in particular to simulate the behaviour of components in altered conditions and allow IRSN delivering appraisal on the specifications of construction that are to be proposed by implementers.

These studies are carried out by the mean of experiments performed either in IRSN surface laboratories, or in the Tournemire Experimental Station (TES) operated by IRSN in the south-east of France.

In the particular field of DGD, responsibility of the design and construction of a DGD has been entrusted to Andra. The requested authorization directive to create the facility is foreseen in 2015 and its operation should start in 2025. During this process, IRSN is responsible for assessing, on behalf of the public authorities, the safety of the project that will effectively be proposed by Andra.

### 9.4.2 Identified key R&D issues

Because of time constraints, it is of crucial importance to be able to anticipate the development of knowledge and resources required to assess risks posed by nuclear facilities in the future, and in particular waste management safety. It is the reason why IRSN has identified very early in the French geological repository project development the scientific issues that had to be addressed in priority. This enabled IRSN to optimise the resources allocated to research. These resources are periodically assessed with respect of the progress made in studies, the new issues to be taken into account and duly planned, as well as the regulatory review agenda that requires to swap research and assessment activities.

The research activities carried out by IRSN are developed in consistency with conclusions drawn from the stepwise regulatory process that allows periodically addressing the remaining issues that must be dealt with to improve the safety demonstration. The expected outcomes of IRSN R&D programme are clearly identified with respect to the safety review approach, paying in particular a specific attention on which phenomena that must be studied by the TSO so as to ensure appropriate independent judgement of the level of safety that the repository may reach. It is also a duty for TSO to be able to deliver opinion on the consistency and degree of confidence of the data produced as well as on the ability of the implementer to realise, at industrial scale, components that will perform “as designed”.

Taking into consideration the feedback and main conclusions drawn from the regulatory review of the “feasibility of reversible geological disposal in clay” in 2005, IRSN has identified a number of important issues, grouped hereafter in "key safety issues", on which researches should be carried out with priority from 2006 to 2015. The issues presented hereafter, which relate only to the Meuse/Haute-Marne site, do not anticipate on the possible emergence of other issues of importance for establishing the safety demonstration during further steps of project development. However at this stage of the project, IRSN gives priority for examining:

- the confinement capabilities of the sedimentary host rock and the identification of possible fracturing in the host formation and the geological layers surrounding it,
- the perturbations due to excavation or due to the interactions between different components,
- the waste degradation,
- the uncertainties on corrosion rates of metallic components, due particularly to a lack of knowledge on transient environment conditions and their duration,
- the dimensioning hypotheses for the various repository components, with the aim at constructing containment barriers that are as effective as is reasonably possible,
- the construction/operational safety (accounting for reversibility) particularly with respect to the risk of explosion relevant to hydrogen produced by radiolysis in waste cells, the ability to remedy a situation caused by a package fall in cells and the possibility of retrieving waste,
- the sealing capabilities with the view to assessing the likely performances of a sealing engineered structure, taking into account the effects of potential disturbances over time or difficulties for emplacing seals at industrial scale,

- the long term performances of the repository with emphasis on hydrogeological modelling, integrated transfer of radionuclides and biosphere modelling. It is particularly important to be able to rule on whether or not localised preferential transfers exist and to assess their influence on the general flow patterns.

#### 9.4.2.1 Quality of data

Structures: 3D seismic methods; diffusion properties: "through" diffusion

- The diffusion mechanisms in stiff clay (origin of over-pressures and influence of pore size on water-rock interactions…). Many characterization methods (devoted to characterise movement of natural tracers…) have been tested,

- The hydraulic role of faults/joints : survey methods (seismic survey analysis combined with others methods…) used to identify fractures in clay and their potential as water pathway have been tested,

- The EDZ development: characterisation methods

- Characterisation of the parameters that will have to be specified and controlled in situ to warrant the performance of seals and concrete liners; a dedicated in-situ mock-up is under development and will be implemented in TES to study altered evolution of seals

#### 9.4.2.2 Understanding of complex processes

The TES is a former railway tunnel crossing a 150m meters thick Toarcian argillite formation and has been intensively used for some 20 years to perform in-situ experiments devoted to better understanding:

1. The differential fracturing phenomenon in clay and its high damping potential,

2. The EDZ development: characterisation methods and modelling have been used and developed taking advantages of, on the one hand the 100 years passed since tunnel construction, and, on the other hand the observation of new drifts recently drilled,

- The TES therefore make it possible to study the EDZ in galleries of different age, shape, dimension, excavation method, lining, orientation versus local stresses, tectonic history of the field. A thorough characterization of the EDZ is still in progress at the TES, using various geophysical and geomechanical methods and generating collaborations with e.g. the Canadian Nuclear Safety Commission (CNSC), Clausthal University of Technology (Germany), the French institute of science and technology for transport, development and networks (IFSTTAR), the HydrASA laboratory of Poitiers University (France), etc. It may be underlined that the fracture patterns and extents of the EDZ observed at the main level of the ANDRA's Meuse/Haute-Marne URL (France) also clearly show a dependence on the galleries orientation versus the in situ state of stress
- The modelling of the formation and evolution of an EDZ is a very challenging concern. This is intrinsically a three-dimensional problem involving the tunnel orientation and

the anisotropic rock properties and in situ stress state. Concerning the Tournemire site, this concern has been addressed in an especially devoted task of the DECOVALEX-THMC project (2004–2007)].

- Based on the observations conducted at Tournemire (visual observations, analysis of samples, EDZ permeability measurements…) a conceptual model of the EDZ initiation and propagation has been proposed as follows: at first, during excavation phase, no failure occurs around the opening because the induced new stresses are lower than the strength of the rock mass. Then, few months after the end of excavation, the desaturation/resaturation processes take place and induce a tensile failure around the new galleries allowing the development of the observed fissures parallel to the bedding planes. Later on, these seasonal processes can cause a gradual weakening of the rock which can induce a compressive failure leading to the creation of onion skin fractures as observed around the old tunnel.
- Numerical simulations of increasing complexity have been carried out in order to validate this model. In a first attempt, the response of the argillaceous rock to the excavation phase, followed by seasonal cyclic variations of temperature and relative humidity inside the opening, has been simulated by means of a purely mechanical analysis, using a simple elastic material model. The EDZ has been estimated by post-processing the calculated stress states using a Mohr–Coulomb failure criterion. The results show that no EDZ could be predicted unless adopting a low cohesion value for the rock mass. Moreover, the deferred nature of the EDZ formation in the TES could not be reproduced. These first results have been improved by using a coupled viscoplastic damaging mechanical model, the parameters of which have been identified from different laboratory experiments. With this model, a time evolution of the EDZ could be predicted but the EDZ pattern could not match the one observed in situ. Finally, in view of the importance of the hydraulic couplings, unsaturated hydro-mechanical calculations have been carried out to investigate the effect of the numerous seasonal variations cycles and the resulting shrinkage. A coupled unsaturated hydro-mechanical model has been implemented but still using a linear elastic mechanical model. As before, the EDZ formation has been evaluated through a post-processing according to a Mohr–Coulomb failure criterion. It has been shown that, when reduced rock cohesion is used, an EDZ appears directly after excavation by compressive failure mechanism on which a tensile failure mechanism is added during the time due to the desaturation process. Although this EDZ evaluation, by means of post-processing, is clearly an approximation, it leads to realistic predictions in the short term response of the rock mass, directly following the excavation period. However the long term prediction of the final EDZ patterns and extent differ from what is observed in situ. This is most probably due to the limitation imposed by a post processing analysis of the stresses in the determination of the EDZ, since in reality, irreversible changes in the material properties take place in the failure zone, which is not accounted for in the above simulations. A realistic simulation of the long term response of the rock mass should consider these degradations of strength.

3. The clayey materials evolution due to cement-clay / iron-clay interactions by characterisation and modelling of 10-year old in situ experiments (using a coupled transport/chemistry code Hytec developed by Ecole des Mines de Paris),

4. The chemical conditions during transient processes and the specific effects of the presence of micro-organisms or of redox conditions (characterisation of processes upon Tournemire data) on the waste or engineered components degradation over time,

5. The parameters that will have to be specified and controlled in situ to warrant the performance of seals and concrete liners; a dedicated in-situ mock-up is under development and will be implemented in TES to study altered evolution of seals.

From the feedback of the analysis that IRSN performed on Andra's Dossier 2005, a R&D programme was defined to make it possible for IRSN to give an expert advice on the overall questions related to the safety of a deep geological repository for radioactive waste. Part of this programme focuses on the efficiency of sealing systems (cell seals, gallery seals, shaft seals) for which two main questions arise:

(1) Will sealing systems effectively allow controlling the potential water fluxes? This relies in particular on characterizing the resaturation kinetics, the homogeneity of bentonite-based components and the water tightness of the seal/host- rock interface.

(2) What are the key mechanisms and parameters that govern the performance of sealing systems at an industrial scale under normal conditions, and what is the effect of some altered conditions on these performances?

To answer these questions, a series of in situ experiments have been purposefully designed for assessing the performance of sealing systems under conditions that are representative of long-term ones (isothermal and water- saturated). Their specific objectives are:

a) Quantifying the global performance in normal (non-altered) conditions, for different types of clay cores (pure MX80, sand/MX80 mixtures, either pre-compacted or in situ compacted);

b) Quantifying the impact of construction specifications (intra-core geometry, i.e. construction joints in the case of pre-compacted blocks) on the overall performance;

c) Investigating the concept robustness by considering altered scenarios, such as (i) an incomplete saturation of the swelling clay, and (ii) an incidental decrease of the swelling pressure (for instance originating from the failure of the confining plugs).

6. the physical and chemical properties of the concretes in their initial and altered state

Besides those activities mainly based on experimental aspects, specific studies are in progress in complementary scientific fields with the view to:

- better understanding the transient phenomena and in particular the behaviour of hydrogen generated by corrosion and radiolysis and its influence on water flow; these studies are addressed by experimental, theoretical and modelling developments,

- in the framework of the European FORGE IRSN has carried out irradiation experiments at 2 dose rates (100 and 50 Gy/h) using the new device developed during the previous year. Hydrogen production was monitored through continuous measurements during all the duration of the experiments (at least 4 weeks). According to the results obtained, it seems that irradiation (at these dose rates) influences corrosion processes and induces an enhancement of hydrogen production. However, this preliminary observation has to be further supported by additional data.

- better knowing of the waste performances,
- better knowing of the transfer properties of radionuclides and chemical elements under repository conditions (data base review),
- modelling flow and transport of radionuclides by developing computer models simulating the underground flow patterns at various scales in the vicinity of the Bure site as well as radionuclide migration from the waste packages to the biosphere (3D computer code MELODIE),
- modelling the biospheres of interest for the Bure site (existing and possible in future).
- In addition, the safety researches to be possibly undertaken related to operational safety and reversibility issues are in a preliminary phase devoted to the definition of targeted actions.
- the influence of industrial implementation conditions on their performances?
- verification of extent and intensity and importance of processes

#### 9.4.2.3 Identification of sharing and networking possibilities

Where no specific installations nor sufficient resources and specific skills are available or not considered as of core competencies of IRSN, it is decided to perform literature survey, to foster participation to national groups managing R&D on waste matrix development and characterisation. -Evaluation of waste matrix performances

Because of the complexity and large scope of issues to be addressed, IRSN promotes a multi-disciplinary approach integrating experimentalists, modellers and experts of safety who work together on each of the topics of interest for safety. This synergy between research engineers and experts in safety assessment is a valuable tool to ensure consistency and quality of technical assessment. Scientific partnerships with research facilities and universities is the preferred strategy of IRSN in order to be able to take benefit of high level scientific skills in different specialities and for a duration compatible with the planned time frames of the assessment process (several decades).

Part of IRSN research programme is integrated in the EURATOM Framework Programme related to radioactive waste management research. IRSN is involved in 6th and 7th Framework Programmes which offer a valuable framework for achieving results and for sharing experience among countries involved in waste safety. IRSN supports also international research programmes as the Mont Terri project as well as bilateral cooperation with homologous organisations in foreign countries.

**Relation to SRA**

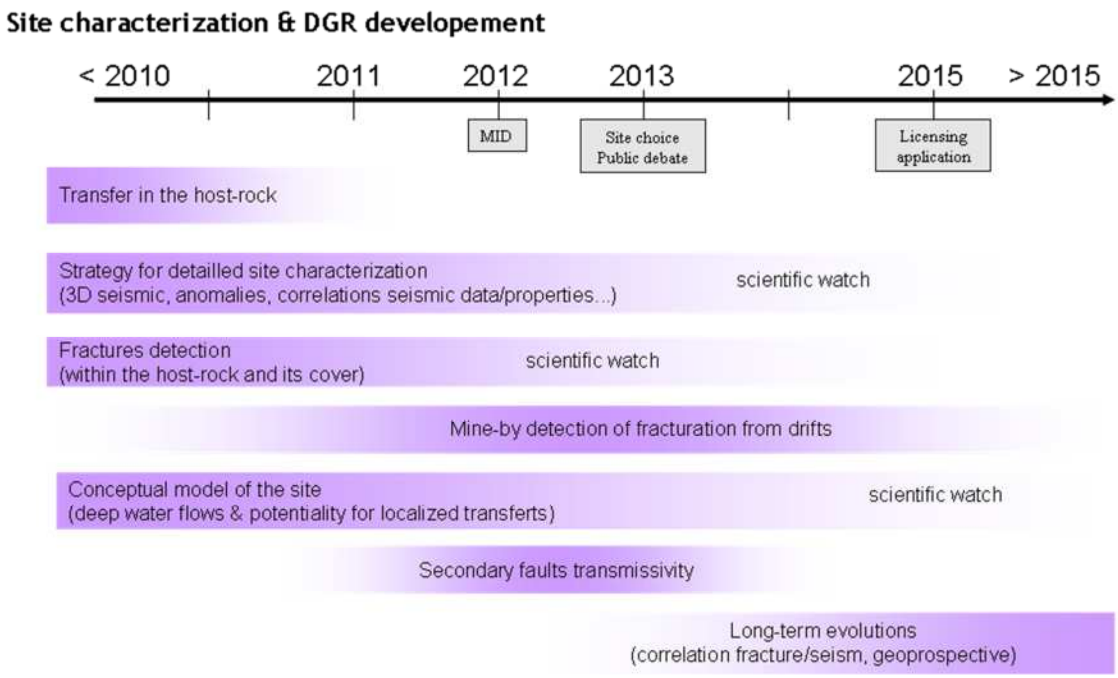
But the efficiency of the research carried out by the regulator or the TSO does not rely only on technical skills but also on its ability to promote synergy between experts in charge of assessment and researchers. This contributes highly in guiding research efforts that must be made for the purpose of maintaining the quality of the regulatory review. In complement, high scientific skills ensure efficient technical dialogue between the implementer and the evaluator which is also a necessary condition to achieve valuable assessments.

Quality and independency of research programme carried out by IRSN allow building and improving a set of scientific knowledge and technical skills that serves the public mission of delivering technical appraisal and advice. In particular they contribute in improving the decisional process by making possible scientific dialogue with stakeholders independently from regulator or implementer.

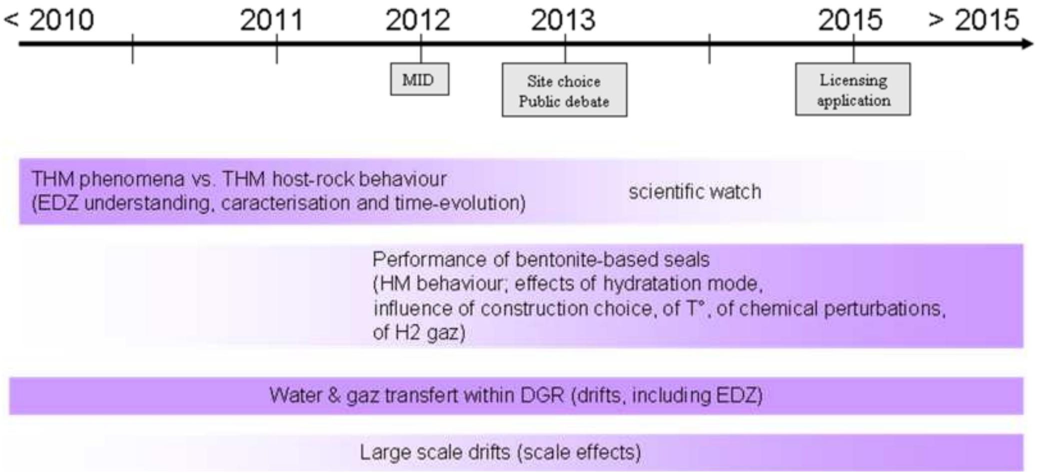
Given the present stage of development of the French DGD concept, the present IRSN key R&D issues cover the four main following topics:

- Site characterization & DGD development;
- THMC perturbations;
- Physico-chemical evolution;
- Global modelling of solutes and gas transfer from the DGD.

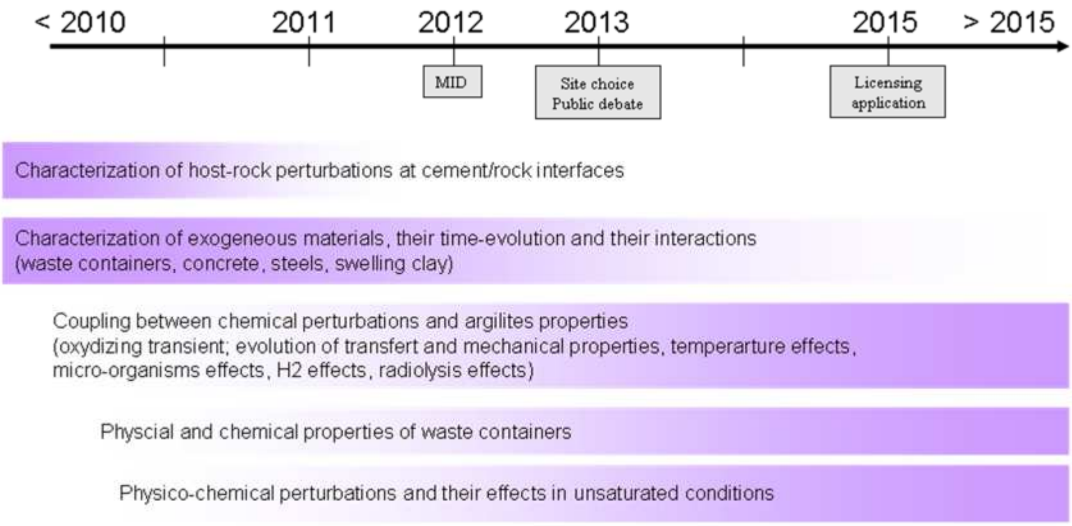
The following diagrams summarize the time-evolution of effort that have been spent in the past, and the amount of efforts per specific topics that planned over the next years.



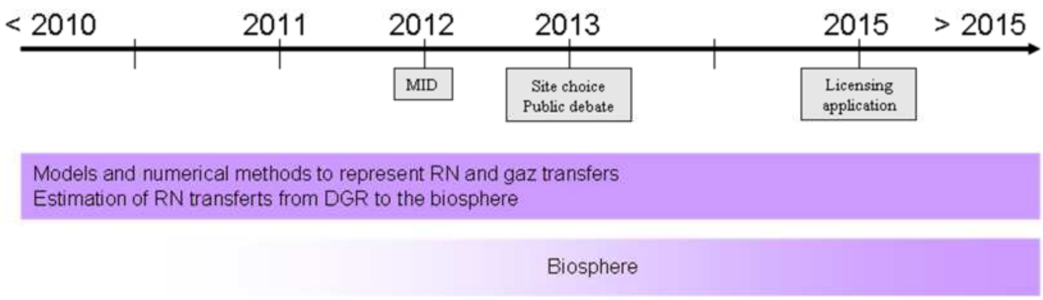
**THMC perturbations**




**Physico-chemical evolution**




**Global modelling of solutes and gaz transfer from the DGR**




#### 9.4.2.4 Site characterisation

The experimental work aims to study:

- Transport properties of water and natural substances in clay formation since its formation (180 million years ago);

- The damage induced by gallery excavation and/or rock desaturation due to natural ventilation;
- The alteration induced by the presence of concrete and metallic material – components that exist in a geological disposal;
- The potential of geophysical investigations (electric and 3D seismic surveys) to detect fault zones in argillites and limestone formations;
- To test long term hydraulic performances of sealing systems (in nominal and alerted conditions), quantify the impact intra core geometry of sealing systems and quantify the effect of altered conditions on its performance (the SEALEX project).

Measuring devices are emplaced and observation techniques are implemented to study the rock and its behaviour, the pore water and its flow rate. Part of the data acquired within the framework of experiments is used to develop models and validate models that will allow assessing the safety of future Andra disposal sites.

Here are some of the topics IRSN worked on since 2010:

- Assessment of a zone of interest for detailed reconnaissance
- Geological disposal facilities for high- and intermediate-level long-lived waste
- Sealing performance experiments (SEALEX project) on the performance of clay seals in a DGD at long term.

## 9.5 GERMANY

The maintenance of competence and independence are key requirements for any TSO and safety authority. Therefore R&D by the TSOs and regulators are obligations that might help to fulfil these requirements. In awareness of this role the GRS as a TSO, performs in a continued way respective R&D work. The R&D work comprises more or less four main areas:

- Development of tools and methods
- Elaboration of safety requirements and criteria
- Guidelines
- Maintenance of competence, education and training

In the following subsections the envisaged R&D work in the near future of the GRS in respect of the above mentioned four main areas are described.

### 9.5.1 Development of tools and methods

#### 9.5.1.1 Research and development of methods and tools for the safety case

I. Development of computer codes for performance assessments:

For the comparison of different disposal sites appropriate simulation tools are needed. In the past some relevant processes in salt like the compaction of crushed salt, corrosion and changes of temperatures are modelled and coupled with a computer code which simulates two phase flow. Those simulation tools have to be developed further especially in terms of other host rocks, e.g. clay. In this context the quality assurance is of particular importance.

Further, it has to be tested that necessary assumptions for the simulation of processes in safety analyses are admissible, in order to improve the underlying physical models.

Relevant objectives for this task are:

- Identification of needed R&D work concerning process simulation for different host rocks
- Development of computer codes
- Quality assurance of computer codes (e.g. benchmarks)

II. Advancement of the existing method for the derivation of scenarios:

In the frame of a former project the basis for a scenario development method with a strong connection to safety functions was compiled. This method shall be further developed for the application to a specific disposal site. It is expected that due to the related approach to safety functions this method is more useful for the development of calculation cases from derived scenarios than other methods.

Specific objectives for this task are:

- Analysis and evaluation of scenario development methods from other countries
- Further development of an existing method for scenario development
- Development of an approach for the derivation of calculation cases from scenarios

#### 9.5.1.2 Development and application of a method for a systematically derivation of operational accident scenarios

According to the safety requirements of the BMU, comprehensive safety analyses for the operational phase of a repository are needed. This includes an analysis of operational accidents. Such analyses are done so far on the basis of deterministic approaches and expert judgement. A general systemic approach like the one that is used for scenario development in the post closure phase is not available for an operational accident analysis. Therefore it is intended to develop a respective method which allows the systematic derivation of operational accident scenarios on the basis of gained experience with scenario development in the long term analysis. The specific objectives for this task are:

- Compilation and studying of existing international methods
- Compilation of relevant influencing factors (similar to FEPs)
- Evaluation of operational experience and incidents
- Evaluation of incident rates
- Distinction of different disposal concepts and host rocks
- Definition of safety functions for components of the disposal system
- Exemplarily derivation of accident scenarios on the basis of influencing factors

### 9.5.2 Elaboration of safety requirements and criteria

#### 9.5.2.1 Derivations of safety requirements relating to the retrievability of heat-generating radioactive waste from an operational perspective

The safety requirements of the BMU require the retrievability of the emplaced radioactive waste during the operational phase if necessary. Therefore, a possible retrievability will be an integral part of the licensing procedure like the disposal operations. At present, no safety related requirements to the retrievability exist.

In this context the specific objectives are:

- Description of correlations between disposal operations and retrievability
- Evaluation of the state of the art concerning retrievability
- Analysing of international disposal concepts with a retrievability option
- Derivation of safety related requirements to the retrievability during the operational phase
- Requirements on the disposal casks/ containers in view of retrievability actions

#### 9.5.2.2 Requirements on the construction of technical barriers from a regulatory point of view: Realisation and quality assurance

Technical barriers like drift seals and shaft seals have to provide essential safety functions in the post closure phase of a disposal system. It has to be proven in the safety case that these barriers can be constructed as planned and that requirements from a long term safety perspective are met. Of particular importance is the question of the feasibility of such technical barriers and the quality assurance during the construction.

Specific objectives for this task are:

- Compilation of requirements on technical barriers in different host rocks (salt, clay)
- Evaluation of the state of the art concerning the construction of technical barriers
- Evaluation of international approaches
- Determination of necessary requirements on the quality assurance to the construction of technical barriers
- Analyses and assessments of existing demonstration tests
- Identification of necessary R&D work

#### 9.5.2.3 Requirements and tools for the proof of integrity (containment) from a regulatory point of view

The integrity of geological barriers in connection with technical barriers shall be secured for a demonstration period of 1 million years. In recent years the requirements in context with the proof of integrity are strongly raised, especially in terms of the used tools, methods and calculation programs.

Relevant objectives for this task are:

- Compilation of the international status to the proof of integrity in different host rocks (salt, clay)

- Identification of processes which might jeopardise the integrity of barriers
- Derivation of requirements on a comprehensive and transparent proof of long term integrity of the barriers

#### 9.5.2.4 Further development of criteria for a comparison of disposal sites for heat-generating radioactive waste

In view of the expected comparison of disposal sites it is required to provide a set of criteria on a well-founded scientific basis. These criteria should be derived for sites in salt and clay.

Relevant objectives are:

- Derivation of criteria to compare different disposal sites in salt and clay
  - Temperature criteria for the surrounding rocks and aquifers adjacent to the repository
  - Site specific application of fluid pressure criteria and dilatancy criteria
  - Gas pressure compatibility of different rocks
  - Influence of hydrocarbon occurrences
- Derivation of minimum scope of investigations
- Dealing with different scopes and qualities of siting data
- Derivation of robustness criteria

### 9.5.3 Guidelines

The GRS supports the BMU in the development of guidelines in the field of radioactive waste management. These guidelines form the basis for the implementer in different aspects of radioactive waste disposal.

At present, the following issues are subject of intended guidelines:

- Calculations of radiation exposure in the long term
- Justification of the distinction between probable and less probable scenarios
- Treatment of future human actions at disposal sites in safety cases

### 9.5.4 Maintenance of competence, education and training

#### 9.5.4.1 Investigations to the radionuclide transport in fresh water/ salt water systems in consideration of differences in density and viscosity

The groundwater situation at disposal sites in Germany is characterised by saline conditions (density dependent transport). Furthermore, problems in dependence of disposal concepts appear especially in modelling of mixtures of solutions with different density and viscosity as well as in coupling of mine models and geosphere models.

Relevant objectives for this task are:

- Compilation of the international status of the state of the art relating to processes and modelling
- Evaluation of internationally applied mathematical/ physical models

- Adaption/ modification of numerical models for the simulation of density driven radionuclide transport in salt water systems
- Coupling of transport models (repository mine – overburden) for the simulation of the transfer of squeezed brine in deep saline water
- Development of instruments/ tools for modelling density driven transport processes

##### 9.5.4.2 Radionuclide mobilisation, reactive transport processes and kinetic of geochemical reactions

Chemical reactions take place both at salt sites and clay sites which essentially affect the radionuclide mobilisation and radionuclide transport. These reactions are therefore relevant for the long term safety.

Objectives are:

- Speciation of radionuclides
- Description of the source term
- Consideration of IRF
- Modelling of reactive transport processes e.g.
    - Coupling with reaction kinetics
    - Corrosion processes (redox reaction, kinetic, gas production)
    - Development of the composition of solutions at clay sites
- Geochemical influences on hydrocarbon occurrences

##### 9.5.4.3 Exclusion of nuclear criticality concerning disposal of spent fuel in non-saline host rocks without consideration of burn-up

According to the safety requirements of the BMU it has to be proven that nuclear criticality can be excluded for the disposal of spent fuel in the post closure phase. At present the waste producers do not consider the burn-up of each spent fuel element due to the effort for the respective verification. Therefore, it is assumed that nuclear criticality analyses take fresh fuel elements into account.

Relevant objectives are:

- Feasibility study for the exclusion of nuclear criticality on the basis of fresh fuel elements in non-saline host rocks
- Consequence analyses in consideration of nuclear criticality of disposal concepts with bentonite backfill in non-saline host rocks
- Compilation of the international status of the state of the art relating to nuclear criticality

## 9.6 LITHUANIA

### 9.6.1 Safety strategy

Lithuanian priorities in the nuclear R&D are best reflected in current National Energy Strategy. The following areas in scientific research of nuclear field are identified:

- thermonuclear and new generation nuclear reactors (by participating in respective international programmes);
- nuclear energy safety, reliability and durability of energy equipment and systems, and ageing of construction materials;
- management, storage and disposal of spent nuclear fuel and other radioactive materials.

In relation to the current situation in the field of geological disposal in Lithuania the Safety Case has not been prepared and reviewed yet. However in case of priorities in R&D in geological disposal the following aspects are identified by LEI:

As the Lithuania's national program on geological disposal is at the very beginning stage the main priorities would be the evaluation of safety functions (containment, isolation) of disposal system as well as those allocated on its components taking into account the potential geological environment in Lithuania (repository in the hard rock overlaid by the sedimentary rocks cover, in clayey formation). This is related to the selection of the engineered barriers (for example, less expensive steel canister might be suggested instead of copper canister for the disposal in crystalline rock).

The approach of managing the various activities related to the disposal facility development and implementation (such as sitting and design, safety assessment site characterization, management of uncertainties, waste form characterization, etc.) is a part of safety strategy based on [9]. This is an important issue as it is contributes to the input information for the cost evaluation related to the geological repository.

### 9.6.2 Data acquiring

For the preliminary assessment to be carried out during conceptualization phase the data on expected host rock, engineering structures and waste characteristics are required. Thus the priorities for R&D are to collect this type of information. For the evaluation of different design options the representative data on geological formations is needed. This means that the priority is in the performance of geological investigations (reliable methods, hydrogeological characteristics of geological formations, tectonics) and in the upscaling of data (data should be of representative nature with regard to spatial and temporal variability).

The extrapolation of short term experiment data to the long term data is also important (waste leaching rates, sorption). The changes of the properties of the natural and engineered barriers (through the changes in groundwater flow regime, chemical conditions) are strongly related to the climate conditions which are expected in the future. Thus the evaluation of the impact of climate change within long term post closure period and its incorporation in the safety assessment is an important issue. Features, events and processes that are potentially important for the safety of the disposal system should also be identified thus there is a priority suggested on the development of updated FEP databases, the most appropriate way/method for the scenario development.

We identified the priority in the radiological and disposal characteristics, the amount of SNF and long lived intermediate level waste as well. The Ignalina NPP has already been shutdown and its decommissioning is on-going thus the opportunity to verify the estimations by direct/indirect measurements (at least for a part of waste) occurs. It is known that in case of SNF disposal the radionuclide I-129, Cl-36, for graphite C-14 are the most important ones. As there is no experimental data on radionuclide release (congruent release, instant release fraction) from various radioactive waste from RBMK type reactor (graphite from reactor core, SNF assemblies and other waste considered for geological disposal), this type of research is important. However the importance of this type of research will depend on the repository concept and needs to be evaluated through the sensitivity analysis. Besides, it is valuable to analyse the scaling from the determined data for the materials from other reactors, its acceptability from regulatory point of view and the applicability during the development of Safety Case. In relation to the new nuclear power plant the assessment of the inventory of SF with high burn-up and its criticality becomes of high priority.

Measuring and verifying input parameters of canister, sealing system is of lower priority as the final decision on the disposal concept has not been done yet in Lithuania.

### 9.6.3 Modelling strategy

A priority for the research was also identified in the evaluation of coupled processes in the repository environment (THMC), which might contribute to the design of the repository and has to be assessed to certain degree. The analysis of the coupled processes is complex and might require a large computing resources and time. The priority might be in the development of scoping models (upscaling from small scale detailed models to the repository scale models).

Another important issue could be the evaluation of the priorities in further R&D at different stages of the repository based on the sensitivity and uncertainty analysis, i.e. what is the end-point of the assessment for which the sensitivity and uncertainty analysis is performed at each stage, is it necessary to include the biosphere and its related the uncertainties, etc. Important aspect related to the modelling strategy is treatment of the uncertainties in the biosphere which are strongly related to the climate change during post-closure period. It is also might be a priority for the analysis and definition of the acceptable level confidence after the performance the sensitivity and uncertainty analysis (how to decide is performed analysis enough or need more investigations). In case of probabilistic assessment the important issue is the definition of probability distribution function for the parameter which is poorly measured (not enough data for statistical description). The common recommendation would be valuable.

## 9.7 NETHERLANDS

In The Netherlands, areas where R&D actions conducted by TSOs are needed in priority are derived from the policy considering the Safety Strategy, and include:

- Short term safety
- Long term safety
- Siting and design strategy
  - National/International repository
  - Co-disposal of SNF/HLW and LL-LILW
  - Depth of disposal
  - Retrievability
  - Monitoring
  - Geology
  - Repository design concept

These issues are described in more detail in Section 6.7.1. The key technical issues are treated in Section 9.7.1.4.

### 9.7.1 Safety Strategy

A central policy consideration of the Dutch Government is a stepwise approach to finding waste management options that are feasible, suitable and acceptable, in both technological and societal respects, is. Based on three policy documents, published respectively in 1984 [10], 1993 [11] and 2002 [12], the current strategy can be summarized as follows:

- long-term interim storage in purpose-built stores at COVRA, the Dutch site for surface storage of radioactive waste, for at least 100 years;
- on-going research, preferably in international collaborative programs;
- eventually retrievable[6] deep geological disposal.

In 2001, the results of a Dutch national programme, the CORA programme, were published in which the possibilities were investigated for the deep geological disposal of radioactive waste in salt and clay layers [18]. Main conclusion of that study was that geological disposal in salt and clay layers with the option of retrievability would in principal be feasible. A deep geological repository is however not an option yet in the Netherlands because of the relatively small amounts of HLW that are produced on a yearly basis, and because it has been decided that surface storage in the interim facility COVRA for the period of at least 100 years is the current preferred option.

In the Netherlands it is has to be demonstrated through a safety report that for nuclear facilities risks and individual doses are below the regulatory limits. However, a license application will also include an EIS (Environmental Impact Statement), which follows more or less the ICRP principles for Radiation Protection, i.e.: (1) justification, (2) optimisation, and

[6] Retrievability means the deposition of radioactive waste in a way that it is reversible for the long-term by proven technology without re-mining.

(3) compliance with limits. The EIS uses the safety report to show compliance. For optimisation the EIS needs more indicators to be able to compare with alternative options. Presently the only indicators are dose and risk, for which there are reference values and constraints.

Isolation, control and surveillance form the main components of the Dutch radioactive waste policy. The IBC concept is applied to all kinds of waste material either until it is no longer radioactive or until it could be disposed of in such a way that the likelihood of unacceptable exposure to the biosphere is negligible. For the latter it should be noted that deposition of the waste must be performed in a retrievable manner until eventual closure of the repository, because, amongst others important issues like economical and societal aspects, an irreversible disposal process does not fully meet the IBC requirements.

One important aspect of isolation and control is the limitation of the quantity of the waste in terms of both activity and volume. The licensing system in the Netherlands requires that the use of radioactive materials is limited and hence contributes to a restrictive quantity of waste.

#### 9.7.1.1 Short term safety

An important objective of the Dutch radioactive waste policy is ensuring that humans and their environment are protected from the harmful effects of exposure to radiation. The Euratom basic safety standards [13], which followed in general the recommendations of the ICRP [14], form the basis of the Dutch national regulatory limits [15]. The dose limits for normal operation to both workers and members of the public are consistent with those listed in the Euratom basic safety standards. However the dose constraints, which could be used within the context of optimisation of radiological protection, are more stringent in the Netherlands than those listed by Euratom. Furthermore each licensee is required to work in accordance with the ALARA principle which is prescribed in its licence.

#### 9.7.1.2 Long term safety

The main components of long-term safety are the isolation, control and surveillance of waste material either until it is no longer radioactive or the amount of radioactivity finding its way into the biosphere is negligible. The radiological risk targets for facilities dealing with radioactive wastes is based on the Government policy statement dealing with radiation risks, in which the general radiation risk criteria are applied also to radioactive wastes. As a consequence of this policy the risk target for land-based disposal in respect to the calculated radiological risk of death or serious hereditary defect to an individual from the potentially most exposed group of people should be less than one in a million per year.

#### 9.7.1.3 Siting and design strategy

National/International repository

Although extensive studies, on national and international level, showed the technological feasibility of deep geological disposal no final decision has been made currently on that aspect in The Netherlands, following the policy that there should be an extended period of surface storage before disposal could take place. The process of selecting an underground site has been postponed indefinitely because of strong opposition to test drillings from local authorities and anti-nuclear organisations. The CORA study [16] concluded that prolonging the period of interim storage of long-lived waste by some hundreds of years is possible without major adaptations. On-going research is focused on technical and societal aspects of land-based geological disposal. The Netherlands has not excluded the possibility of a regional repository, though this option is presently not being promoted in The Netherlands.

Co-disposal of SNF/HLW and LL-LILW

In one of the preliminary studies on geological disposal possibilities in the Netherlands it was already stated that such a facility must have a capacity for all radioactive wastes originating from The Netherlands [17], from which co-disposal can be seen as a base option for a repository. However, no Governmental decision has been made on this issue.

Depth of disposal

Deep geological disposal, in either rock salt formations or clay layers, is important for long-term safety reasons. Previous studies have suggested that the minimum disposal depth should be at least 500 meters based on the demolishing effect on earth's surface by former ice ages [16]. For the Dutch situation the approximated depths for a repository in a rock salt formation is about 800 meters, whilst the disposal depth for clay will be between the 500 and 1000 meters.

Retrievability

Long-term retrievable disposal of all kinds of radioactive wastes has been official Government policy in The Netherlands, since 1993 [22]. For a host formation in rock salt experience from the mining industry shows that, with adequate maintenance, such a repository can be kept open for more than 100 years [15]. The period during which retrieval is feasible in deep clay layers is currently unclear, mainly due to the uncertainties over the lifetime of the supporting construction.

Monitoring

Surveillance (and thus monitoring) is one of the three main components of the Government's radioactive waste policy in The Netherlands. Surveillance and supervision is judged a 'societal' rather than strictly 'technical' requirement in gaining public acceptance for waste disposal. Hence, monitoring will form a key element in the waste management strategy. In the on-going research objectives monitoring issues play a very important role, because no (final) strategy has been developed yet. Furthermore it is essential to enhance the durability and reliability of the instruments according to specific disposal situations and monitor time frames. It is expected that a long-term in-situ monitoring approach will be adopted when taking into account the Dutch disposal requirements, especially the need to ensure the possibility of waste retrieval over a lengthy time period.

Geology

Two practicable host formations for deep geological disposal of radioactive waste exist in The Netherlands, namely rock salt formations mainly located in the north-east part of the country and (Boom) clay layers of which the minimum required depth can be found in the south near the Belgium border.

Repository design concept

Since no final repository design concept exists, the Netherlands collaborates in several international projects to expand their knowledge about different concepts. Beside the long-term safety requirements, aspects as economic considerations, retrievability and monitoring form the key elements in these investigations. Concerning SNF and HLW a disposal cell approach was analysed in the CORA program in both practicable host rocks: salt and clay formations [16]. The examined concept reached all required safety criteria like stability, criticality, environmental impact and retrievability.

The disposal cell approach is also followed in the present OPERA program, focusing on Boom clay as host rock. However, in respect to a retrievable disposal facility, the basic data for clay are less extensive compared to those of rock salt, but this lacuna will be addressed within the OPERA program [18]. In previous studies the concept of deep vertical boreholes were investigated too, but it was generally agreed that such a design did not facilitate retrievability. It is very likely that waste chambers or caverns will be the preferred option for LL-LILW, given the fact that this type of waste is already packaged in repository suitable drums. An overview of the Dutch DGD concepts is provided in Section 5.7.

#### 9.7.1.4 Key technical issues

The Dutch utilities and the Dutch Authorities jointly coordinate and finance the Dutch R&D program OPERA (“Onderzoeks Programma Eindberging Radioactief Afval” – Research Program for the Disposal of Radioactive Waste), which started in June 2011, runs for five years, and is organized in seven Work Packages comprising most if not all elements that are considered important in modern Safety Cases for geological disposal of radioactive waste. At this time it is not possible to distinguish between key technical issues identified by the WMO and key technical issues identified by the TSO. Therefore a short description of the OPERA research program is given below.

A detailed outline of the R&D efforts which will be addressed in the five year OPERA program is provided by [18]. The seven defined work packages comprise the most relevant topics of the Safety Case for geological disposal and are summarized below.

- WP1: Safety Case context. In this work package, all contextual and logistic boundary conditions for the OPERA Safety Case will be defined. This work package is split up in three parts: waste characteristics, political requirements and societal expectations, and communicating the Safety Case.
- WP2: Safety Case. This work package has a central role in the set-up and definition of both Dutch Safety Cases for Boom clay and rock salt. This WP elaborates the structure of the Safety Case and safety assessment methodology of the OPERA Safety Cases in more detail.

- WP3: Repository Design. In this work package, the principal feasibility of a disposal concept in Boom Clay in the Netherlands at 500 m depth is evaluated. In addition, possible design modifications can be investigated that may reduce uncertainties with respect to the safety assessment of the system concept.
- WP4: Geology and geohydrology. To evaluate the long-term safety of the geologic disposal, it is important to understand past and possible future evolutions of the geosphere. Geology and geohydrological behaviour of the geosphere, as well as geohydrological boundary conditions for the near-field are investigated.
- WP5: Geochemistry and geomechanics. In this work package, the geochemical and geomechanical properties of the engineered barrier system and the Boom Clay will be defined: geochemical behaviour of EBS, including HLW waste matrix corrosion processes, LILW degradation processes and products, metal corrosion processes, cementitious material degradation, and microbiological effects on the EBS and Boom Clay. Also properties, evolution and interactions of the Boom Clay will be studied: geochemical properties and long-term evolution, geochemical interactions and geomechanical properties and thermo-hydro-mechanical evolution.
- WP6: Radionuclide migration. This work package addresses the migration processes responsible for the transport of radionuclides from the waste container into the biosphere. The focus is on R&D on fundamental aspects of sorption, diffusion, and (gas) migration processes of species through Boom clay. In addition, radionuclide migration and uptake in the biosphere will be investigated.
- WP7: Scenario development and performance assessment. This WP comprises all activities to arrive to the radiological safety assessment of the generic repository, including scenario development, performance assessment model development and parameterization, and the execution of safety assessment calculations.

The research is performed by consortia consisting of Dutch companies, research institutes and universities, as well as several foreign organisations.

A topic that is treated outside the OPERA program, but which is considered essential in the Netherlands is monitoring. The Dutch licensing system requires monitoring of radiological conditions, the state of the waste (containment) and discharges from facilities, such as radioactive waste treatment plants and stores. No consideration has been made yet of the issue monitoring in respect to repositories. However, the need for on-going monitoring is realised, especially in the case of phased disposal concepts in which waste retrieval is explicitly accounted for. As part of fulfilling that need, NRG participates in the EU FP7 project MoDeRn[7].

The objective of the project MoDeRn is to provide an understanding of monitoring activities and available technologies that can be implemented in a deep geological repository, and to provide recommendations for related, future stakeholder engagement activities. NRG focuses for an important part on the further development and implementation of wireless, through-the-earth monitoring techniques, including modulation/demodulation techniques and signal analysis.

[7] Monitoring Developments for Safe Repository Operation and Staged Closure; http://www.modern-fp7.eu/

## 9.8 SLOVAKIA

The main objective for further development is to re-open the DGD development and implementation. According the currently elaborating and approving the updated strategy of the back-end of peaceful use of nuclear energy, the first activities should drive at two main directions:

- Continuing in geological investigation/siting activities where they have been interrupted,
- Establishing a system of stakeholders involvement and incentives

The expected nearest siting/geological investigation activities could be described as follows:

1. A detailed geological exploration of concerned sites with crystalline and clay host rock, that were identified during the preceding programme stages, based on exploration results obtained by means of light geological methods, but also by deep bore holes.
2. Narrowing the number of exploration sites (as well as the surface considered) and the selection of the candidate and stand-by sites

Later, it is expected to continue also in other aspects in DGD development as they were elaborated and interrupted in the past, particularly:

1. The draft concept of the deep underground disposal facility respecting the spent fuel and radioactive waste parameters after the storage, rock environment properties in the given sites, the long-term safety of the disposal site after the termination of its operation and closing based on a combination of disposal container properties, engineering barriers and geological environment and the minimisation of impacts on the environment.
2. Safety assessment for the draft concept of the deep underground disposal facility.

According the recent development, the first official involvement and licensing activities of the Slovak Nuclear Regulatory Authority within the DGD implementation and licensing can be expected, roughly, at the end of this decade, at earliest.