Nuclear Development

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Advanced Nuclear Fuel Cycles and Radioactive Waste Management

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NEA No. 5990

NUCLEAR ENERGY AGENCY
ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

5. WASTE MANAGEMENT AND DISPOSAL

5.1 Introduction

The different waste types described in Chapter 3 will require long-term management in a final repository. Different types of repositories are considered for the disposal of low- and intermediate-level waste and of high-level waste. This chapter focuses mainly on the disposal of high-level radioactive waste[1] because advanced fuel cycles are expected to have a considerable impact in this regard and the radionuclide inventory of high-level waste can be relatively well estimated from the results of neutronic calculations.

The information collected within this Study on the characteristics and amounts of low- and intermediate-level waste which will arise from the advanced fuel cycles was not very precise or reliable. As shown in Chapter 3, no major variations are expected on LILW volumes, and since activities and heat loads will remain at low levels, no sizeable impact is expected to result from these waste streams that might modify repository concepts. However, the impact of low- and intermediate-level waste will need to be assessed in future studies when information on these waste categories will improve. For example, if novel materials were proposed, their impact on geological repository safety would have to be evaluated early in the decision-making process when reactor designs were being developed and considered. The rigorous assessment processes developed by countries for assessing suitability of radioactive waste would need to be followed once the materials to be used in advanced fuel cycle schemes will be known.

5.2 Disposal of LILW

5.2.1 Disposal of short-lived LILW

Short-lived LILW can be disposed in a near-surface facility or in a mined repository at a depth of typically a few tens of metres (or in a deep geological repository in case of co-disposal with high-level and/or long-lived waste). Disposal facilities for short-lived LILW constructed near the surface or at moderate depth are in operation in various countries, e.g. in France, Spain, Sweden, Finland, United Kingdom, United States, and Japan. Waste acceptance criteria have been drawn up for these facilities.

For the LLW and ILW that are identified in the fuel cycle schemes it can be verified whether they can be considered as LILW-SL and hypothetically could be accepted at the existing disposal facilities or their future equivalents.

5.2.2 Disposal of long-lived LILW

Long-lived LILW has to be disposed of in a deep geological repository. Just two such facilities have been developed and were/are in operation: Morsleben (Germany, ceased accepting waste in

1. For the purposes of this chapter high-level waste includes spent nuclear fuel.

September 1998) and WIPP (United States). A third facility (Konrad, Germany) has been licensed but is not yet operational. Waste acceptance criteria have been drawn up for these facilities.

For the LLW and ILW that are identified in the fuel cycle schemes and that do not satisfy the waste acceptance criteria for short-lived LILW, it can be verified whether they can be considered as long-lived LILW that could hypothetically be accepted at the existing disposal facilities, or their future equivalents.

5.3 Geological disposal of HLW

5.3.1 Objective

Long-lived radioactive waste, such as spent nuclear fuel (if declared as waste) and high-level waste from fuel reprocessing must be contained and isolated from humans and the environment for many thousands of years [1]. Isolation means keeping the waste away from the biosphere by means which do not rely on active measures in the future and making deliberate human intrusion to the waste difficult without special technical capabilities. The avoidance of locations that may attract inadvertent human intrusion is typically a factor in repository siting. Since complete containment cannot be guaranteed for the whole period during which waste represents a potential hazard, any eventual releases from the repository system should present an acceptable risk. The long-term safety of the repository is provided by the protective functions of the geological environment and the engineered barriers placed around the waste, as well as by the stability of the waste form. The disposal of long-lived waste in deep and stable geological formations is a generally accepted option for the long-term management of radioactive waste.

5.3.2 Considered host formations

At present various types of geological formations are considered as possible host formations for deep disposal of long-lived radioactive waste. The main types considered are:

- hard rock formations, mainly granite; this option is currently studied by countries such as Canada, Finland, Japan, Spain, Sweden and Switzerland;
- argillaceous formations including plastic clays, over indurated clays and mudstones; this option is currently studied in counties such as Belgium, France, Japan, the Netherlands, Spain, Switzerland and more recently Germany;
- salt formations: salt layers as well as salt domes are being considered; disposal in salt domes is studied in Germany and the Netherlands;
- volcanic formations including tuff and basalt; the present United States programme for high-level radioactive waste disposal focuses on a welded tuff formation at Yucca Mountain.

5.3.3 Functioning of a geological repository

Repositories are typically sited in stable geological environments that offer favourable conditions in which the waste and engineered barriers are protected over a long time period [1]. This means that key characteristics that provide safety, such as mechanical stability, low groundwater flux and favourable geochemical conditions, should be unlikely to change significantly over relevant timescales.

Robustness of the repository system is favoured by the multi-barrier concept. The barriers should be complementary, with diverse physical and chemical components and processes contributing to safety, so that uncertainties in the performance of one or more components or processes can be compensated for by the performance of the others.

The safety functions of a repository system can be described in relatively simple terms. Different sets of safety functions have been developed within the main national safety cases and a number of international projects. For example, the safety functions that were identified for water bearing formations in the SPIN project [2] of the European Commission are:

- physical containment: a watertight barrier is isolating the radioactive waste from groundwater during the first phase after repository closure; as long as this safety function is effective, no release of radionuclides can occur from the waste form; physical containment makes the disposal system more robust and easier to analyse by preventing dispersion of radionuclides during the strongly transient initial phase of the repository history (re-saturation processes, heat release, strong radiation, pressure rebuilding, etc.);
- slow release: after container failure, when groundwater comes in contact with the conditioned waste, leaching of radionuclides from the waste matrix starts in combination with the degradation of the waste matrix; various physico-chemical processes, such as corrosion of and lixiviation from the waste matrix, precipitation, sorption or co-precipitation strongly limit the radionuclide releases into the surrounding layers;
- retardation: the radionuclides dissolved in the groundwater that is in contact with the waste will start to migrate through the bentonite buffer and the host formation; because of the very low groundwater fluxes in potential host formations, this transport will be very slow; furthermore, many radionuclides will be sorbed onto minerals of the buffer and the host formation; retardation delays the releases and drastically limits the amounts of radionuclides that are released into the biosphere per unit of time;
- dispersion and dilution: once the long-lived radionuclides leave the repository's barrier system, they are released into the overlying or surrounding aquifers and eventually into the accessible environment; the dispersion and dilution processes in the aquifers and surface waters will further reduce the radionuclide concentrations in the waters that are directly accessible by man.

5.3.4 Repository designs

In the case of geological disposal of long-lived radioactive waste, the geological barrier is complemented with a number of engineered barriers which depend on the characteristics of the host formation and of the disposed waste type. For a given repository site, disposal concepts are designed by taking into account waste characteristics such as radionuclide content, heat generation, criticality, radiation field, leaching rate, gas generation, etc.

Two main types of disposal configurations are considered for high-level waste:

- disposal in galleries, where the waste is placed along the axis of a gallery;
- disposal in boreholes, where the waste is placed in horizontal or vertical boreholes that are drilled from a gallery.

A distinction has to be made between repositories located in water bearing formations on the one hand, and repositories located in salt or unsaturated formations on the other hand.

Repository designs developed for water bearing formations, i.e. hard rock and argillaceous formations, typically consist of the following engineered barriers:

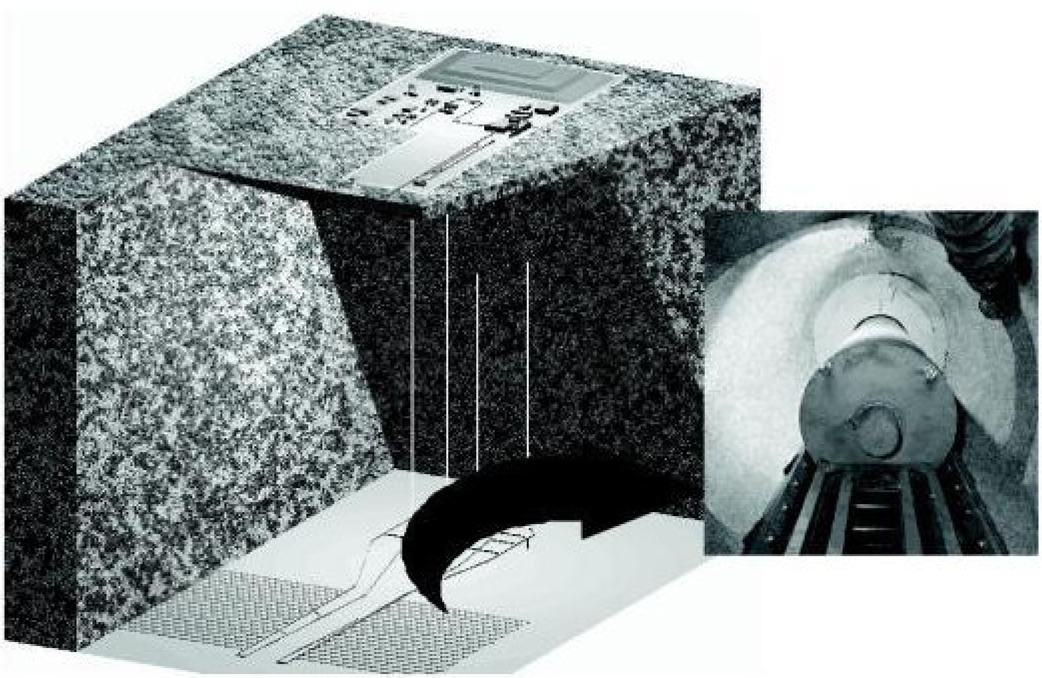
- a metallic container, often called canister or overpack, that has to remain intact during the initial gradient phase of the repository;
- a buffer surrounding the container, which consists in many designs of a swelling clay material, such as bentonite; the buffer has to fill possible voids and fractures and prevent advective water flow around the container;
- a backfill is used to fill transport and access galleries;
- seals and plugs are used to isolate the disposal gallery or borehole from the transport and access galleries and from the part of the host formation that have been disturbed by the excavation of the gallery or borehole.

For repositories located in salt or unsaturated formations several of the above mentioned engineered barriers are also used, but some may not be needed.

a) Example of a repository designed for a granite formation

The reference disposal concept, developed by ENRESA for a high-level waste repository in granite is based on the disposal of four spent fuel assemblies packed in carbon steel canisters in long horizontal galleries. The canisters are surrounded by high density bentonite. The disposal galleries are grouped in two symmetrically arranged disposal areas. Access is accomplished by means of transport galleries, which run perpendicular to the disposal galleries. The transport galleries start from a central area, which includes the required underground infrastructure. Transport between the surface and the central underground area is accomplished by 4 access shafts and a ramp. Figure 5.1 shows a view of the underground installations.

Figure 5.1. Underground installations for a reference repository concept in granite (Spain)



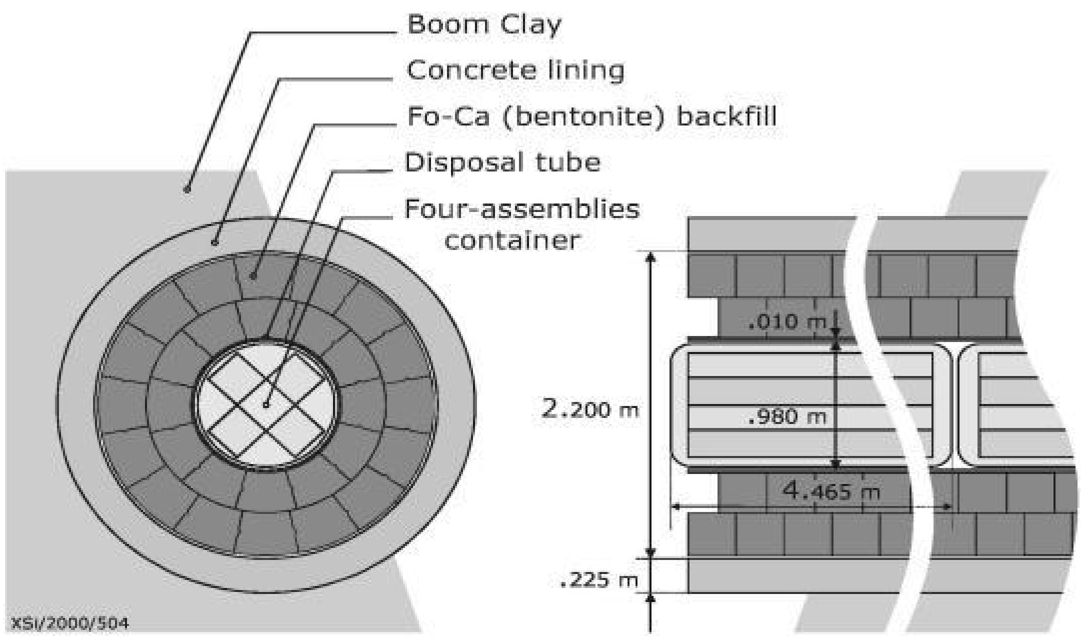
The 10 cm thick canisters are expected to provide a minimum period of containment greater than 1 000 years. The separation between canisters is mainly determined by thermal constraints. A separation of 2 m between canisters and of 35 m between disposal drifts is considered in order not to exceed a temperature of 100°C in the bentonite.

When a disposal drift is filled, it is sealed using blocks of bentonite and structures of concrete at the entry to the drifts. Galleries, shafts and other remaining rock cavities are backfilled, after completion of waste emplacement, with a mixture of bentonite and sand or an appropriate crushed material. The bentonite content of backfilling material is 10%, increasing up to 20% at the top of the galleries. Sealing plugs of galleries and shafts are made of compacted bentonite blocks piled over the whole cross section.

b) Example of a repository designed for an argillaceous formation

The repository configuration that has been developed by the Belgian radioactive waste management agency ONDRAF/NIRAS in the SAFIR 2 study for disposal of high-level radioactive waste in the Boom Clay formation in Belgium is similar to the configuration shown in Figure 5.2.

Figure 5.2. Scheme of the near field of a repository in clay (Belgium)



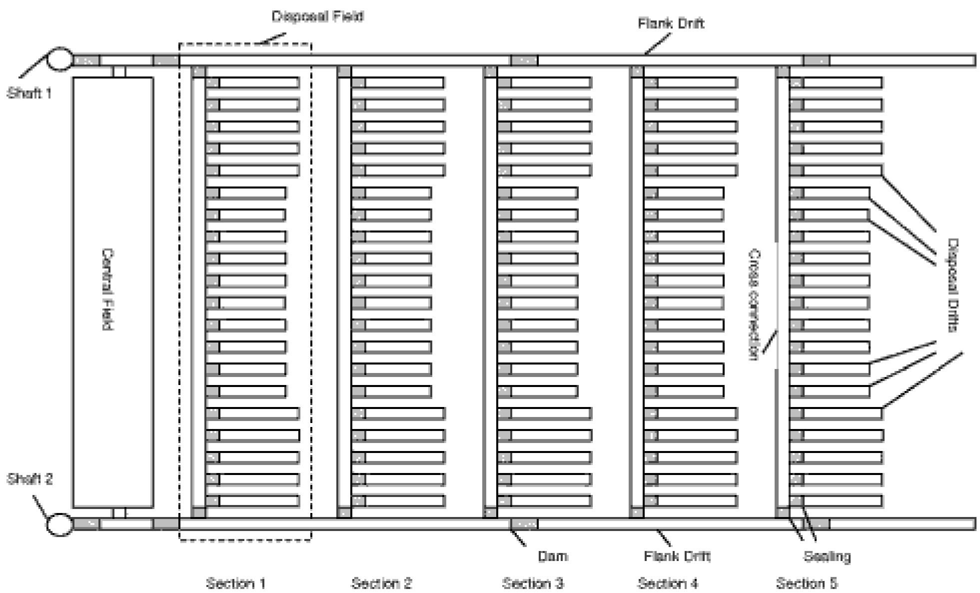
The repository is assumed to be constructed in the middle of the host formation. A central access facility consists of 2 or more vertical shafts connected by the main gallery, from which transport galleries start to the various disposal areas. The disposal galleries are perpendicular to the transport galleries. A scheme of a disposal gallery is shown in Figure 5.2. The spent fuel assemblies or vitrified high-level waste canisters are packed in stainless steel containers, which are placed in a disposal tube. The disposal tube is surrounded by a 60 to 80 centimetres thick bentonite-based buffer. The disposal galleries require the use of a concrete lining because of the high plasticity of the host clay formation.

c) Example of a repository designed for a salt formation

The disposal facility considered by GRS is assumed to be located in a salt dome at a depth of 870 m. The thickness of the overburden is about 300 m. The waste is disposed of in horizontal drifts in a single story (single layer) arrangement. The disposal facility consists of a central field, two flank drifts and five cross connections, with 100 disposal drifts (Figure 5.3). One cross connection and the adjacent 20 disposal drifts and the two segments of the flank drifts form one section. Access to the facility is provided by two shafts.

The volumes and dimensions of the near field assemblies such as flank drifts, disposal drifts, shafts and central field can be found in [14].

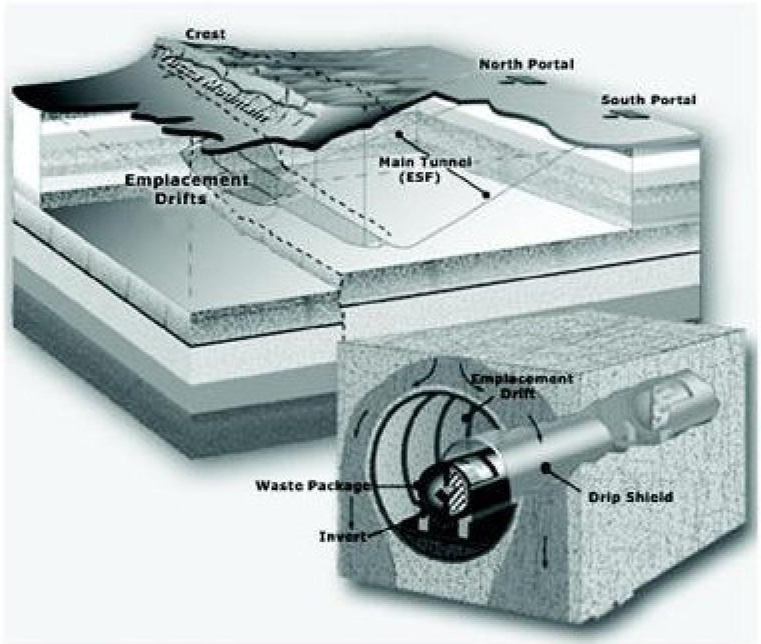
Figure 5.3. Sketch of a disposal facility in salt (Germany)



d) Example of a repository designed for a volcanic formation

The repository concept developed for a repository at the Yucca Mountain site in the United States is located in unsaturated, fractured volcanic tuff. As shown in Figure 5.4, the repository consists of access ramps from the surface and an array of emplacement drifts about 5 metres in diameter, about 2 km long and separated by about 80 metres. The emplacement area covers several km^2 and is located several hundred metres below the surface and several hundred metres above the water table. Both intact spent fuel and vitrified high-level waste would be emplaced in horizontal waste packages set on an engineered invert in the drifts. The waste packages are double shell, with a layer of corrosion resistant nickel-based alloy over a thick structural layer of stainless steel.

Figure 5.4. Conceptual design of a disposal facility in unsaturated tuff (United States)



The density of waste emplacement is constrained to meet thermal design objectives for maximum temperature at the drift wall and at the centre-point between drifts. The engineered system is designed to work with the natural host environment to limit and delay water contact with the waste and subsequent radionuclide release and transport to the accessible environment.

5.4 Evaluations of the impact of advanced fuel cycles on geological disposal of HLW

In a preceding NEA report [3], the impact of advanced fuel cycles on geological disposal was estimated by making extrapolations of the results from existing performance assessments. In the present study, specific evaluations were carried out to allow for a more accurate impact assessment. Modifications of the existing repository concepts for HLW disposal needed to accommodate the HLW types arising from advanced fuel cycles were evaluated and the performance of these concepts for the new waste types were assessed as well the resulting radiological consequences.

5.4.1 Fuel cycles selected for detailed analyses

In Chapter 2, one reference and twelve advanced fuel cycle schemes have been defined. A smaller number of representative schemes were selected for quantitative evaluations.

The schemes selected for the quantitative evaluations are:

- the reference Scheme 1a (PWR, open cycle, UO_2 fuel);
- Scheme 1b (PWR, PUREX reprocessing, single recycling of Pu as MOX) representative of current technology;
- one partially-closed fuel cycle, Scheme 2a (PWR, PUREX reprocessing, multi-recycling of Pu as MOX); and
- one fully-closed fuel cycle, Scheme 3cV1 (GFR, pyroreprocessing, carbide fuel).

For all the repositories considered the maximum dose resulting from the disposal of HLW from the fuel cycle schemes evaluated does not change significantly. The dose reduction factor resulting from reprocessing is at most 8 and mainly results from the removal of ^{129}I from the liquid HLW during reprocessing. Should ^{129}I be captured and disposed of in the HLW repository, the doses resulting from all schemes would be about equal.

The releases of solubility-limited fission products are somewhat lower in the case of advanced fuel cycle schemes, because the denser repository configuration amplifies the contribution of the solubility limitation to the repository performance.

Overall, disposal in reducing conditions results in very late, i.e. after a few million years, and very small releases of actinides into the biosphere. Reducing conditions exists in most hard rock or argillaceous candidate host formations.

In the very long-term, i.e. after a few million years, the total dose is somewhat lower in the case of the fully-closed fuel cycle schemes, because much smaller amounts of actinides have to be disposed of in the repository. However, this effect is not proportional to the amount of disposed actinides because most of them are strongly solubility limited.

The activity of the HLW arising from advanced fuel cycles decreases faster than that of the reference fuel cycle. This can considerably limit the eventual consequences of less probable repository evolution scenarios, such as future human intrusions, that might strongly perturb the functioning of the HLW repository.

Advanced fuel cycles based on fast reactors or accelerator driven systems may require the application of pyroprocessing. The composition of the conditioned HLW forms that will arise from pyroprocessing is not yet known. The presence of corrosion accelerating substances, such as fluorides or chlorides, can complicate the design of an appropriate engineered barrier system. There is also currently a lack of information on the amount and characteristics of LILW types that will arise from pyroprocessing.

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