7 RADIOACTIVE WASTE DISPOSAL

The Agency defines radioactive waste disposal as [5.1]:

*Emplacement of waste in an appropriate facility without the intention of retrieval.*

The key word in the definition is *intention*. Even in the absence of an *intention* to retrieve, waste could be recovered from a repository if the repository was mined. Mining waste could involve great costs and risks. In addition, even in the absence of an *intention* to retrieve, retrievability could be designed into a repository to facilitate waste retrieval should a future decision be made to retrieve it.

The purpose of this Section is to provide an overview of the scientific and technical basis for surface / near surface (see subsection 7.2) and geological disposal (see subsection 7.3) and to discuss issues and trends related to radioactive waste disposal in Agency Member States (see subsection 7.4).

7.1 The Principles of Radioactive Waste Disposal

7.1.1 Context

The safe disposal of radioactive waste, derived from the various stages of the nuclear fuel cycle (NFC) and waste arising from nuclear applications (NA) in medicine, research and industry, constitutes an important and integral component of the Agency’s programme on radioactive waste management. In particular, low and intermediate radioactive level wastes (LILW, see subsection 3.2) are produced in almost all countries and their safe management is of great importance. The potential hazards of radioactive waste to human health and the environment have long been recognized. As such, national and international standards and guidelines dealing with radiation protection and radioactive waste management, including disposal, have been developed and are continuously being improved.

With regard to radioactive waste disposal, the Agency has developed a radioactive waste classification system as part of its RADWASS [3] programme to provide a generic approach to radioactive waste management (see reference [3.1] and Section 3). Central to this approach is the identification of potential disposal options for various waste categories based on their specific characteristics, with the concentration of activity and half lives of the radioactive components being key determining factors.

There is international consensus that geological repositories are required for HLW, spent fuel, and long lived waste (LILW-LL), which may take tens of thousands to hundreds of thousands of years to decay to radiologically insignificant levels. However, surface disposal and near surface disposal (NSD) are suitable options for short lived low and intermediate level waste (LILW-SL) mainly containing radionuclides that decay to acceptably low levels within a few decades or centuries. Please note, the following discussion of NSD applies to surface disposal as well.

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7.1.2 Strategies

The two basic strategies for radioactive waste disposal are:

- **isolate and confine:** This strategy involves the emplacement of waste into a disposal facility that is intended to isolate the waste from humans and the environment and to prevent or limit releases of potentially harmful substances (toxic metals, radionuclides, organics) such that human health and the environment are protected, and

- **dilute and disperse:** This strategy involves deliberately dispersing the waste into the environment in a manner intended to dilute harmful contaminants to levels that are considered acceptable according to internationally agreed standards.

In most national radioactive waste management systems, priority is given to the first strategy. In fact, due to low release limits, the second strategy is practicable for small amounts of radioactivity only. This Section of the Status and Trends report is focused on the *isolate and confine* strategy. Examples of geological repositories being implemented (see subsections 7.4.4.3 and 7.4.4.4) and existing NSD and geological repositories (see Section 8) are provided. A brief overview of the role of underground research facilities (URF) in radioactive waste disposal programmes is given in subsection 7.4.4.2.

7.2 Scientific and Technical Basis for Surface / Near Surface Disposal

NSD is currently practised or planned in many Member States – see subsection 8.1 for an example. There is, therefore, a growing need for additional information and guidance in all aspects of NSD. To address this need, the Agency’s current programme is wide-ranging and encompasses a variety of activities including consideration of the various technology- and safety-relevant issues and approaches needed to develop, site, establish, and assess the safety and performance of NSD facilities.

A recent Agency Technical Report discusses the scientific and technical basis for NSD [7.1]. A synopsis is provided herein. Briefly, NSD is designed to isolate LILW-SL from the accessible environment for a sufficient time to allow decay of the shorter lived radionuclides and, in the longer term, to limit releases of the remaining longer lived radionuclides.

In order to achieve these objectives, a multibarrier approach is commonly employed in which the waste form, engineered barriers and the site itself all contribute to the isolation of the waste. Robust designs of engineered barrier systems can be employed in which a combination of physical barriers and chemical controls provide a high level of containment.

Previously, a multiple barrier system was viewed as a set of independent, individual barriers working sequentially but this concept is now viewed in a more integrated and synergistic manner, with complementary barriers operating in conjunction with one another. Defence-in-depth is provided by employing suitable engineered features in combination with favourable site conditions, controls on the physical form and content of the waste, appropriate operating procedures and institutional controls. The relative contributions of the various barriers to the overall safety of the disposal facility will depend on the characteristics of the waste, the site conditions, and the disposal concept. The relative importance of the barriers will also change with time.

The duration of post-closure institutional controls can be expected to last up to a few hundred years. A period of 300 years, for example, would correspond to around ten half lives of the
radionuclides Cs-137 and Sr-90, which are considered important radionuclides in LILW-SL. Even after that period, it can be anticipated that the degraded engineered barriers will continue to limit releases of longer lived radionuclides, largely by physical-chemical processes such as sorption and solubility control.

The NSD concept usually envisages continued monitoring and surveillance of the site as a part of active controls after repository closure for a period of several decades to a few hundred years. During this period, monitoring and surveillance represent an additional feature, contributing to confidence in the satisfactory performance of the facility. The acquisition of data from monitoring also contributes to general scientific and technical knowledge that can be used in the development and improvement of mathematical models for radionuclide transport and for assessing repository impacts.

Basic Disposal System Concepts

A range of technical solutions is feasible for the emplacement of LILW-SL in the near surface environment. The selection of a particular option depends on many factors, such as the source, characteristics and inventory of the waste, climatic conditions, characteristics of the site, national legislative requirements and radioactive waste management policies.

Generally, there are two main types of disposal systems: (a) shallow facilities consisting of disposal units located either above (mounds, etc.) or below (trenches, vaults, pits, etc.) the original ground surface; and (b) facilities where the waste is emplaced at greater depths in rock cavities or boreholes. In the first case, the thickness of the cover over the waste is typically a few metres, whereas, in the second case, the layer of rock can be some tens of metres thick. These depths can be contrasted with geological disposal for long lived radioactive waste, where the waste is emplaced at depths of hundreds of metres.

In some cases however, local environmental conditions give rise to novel approaches. For example, work has begun on a safety assessment for a LILW disposal facility that is proposed to be built on permafrost at Novaya Zemlya in the Arctic Ocean off the Kola Peninsula [7.2].

During the past 50 years, concepts for radioactive waste disposal have developed considerably. Most experience has been gained for NSD facilities. During this period, there have been many examples of successful repository development, but also of failures in repository performance. Examples of such failures include the rapid leaching of radionuclides from wastes and radionuclide releases due to the flooding of disposal trenches by rain-water or a rising water table (see Section 10, “Managing the Consequences of Past Practices”). Some of these past failures can be attributed to inadequate characterization of the site, unsatisfactory performance of engineered barriers and inadequate control of the nature and inventory of radionuclides and other toxic substances introduced into the repository.

The lessons learned from this experience have led to the development and adoption of improved disposal concepts and technologies. Good examples are the Centre de l’Aube in France, Rokkasho-mura in Japan, El Cabril in Spain (see subsection 8.1), Drigg in the UK, and Barnwell and Richland in the USA. There are also many smaller repositories, constructed in various countries, that have also adopted improved designs based on the multibarrier approach.

For rock cavity repositories, rock cavities can be either natural or excavated by man in various geological formations. A rock cavity repository for LILW has been constructed at Forsmark in Sweden in crystalline rock, about 60 metres below the sea. Two other rock cavity
repositories, located at a depth of 50 metres and similar to the Swedish repository in terms of both the design and type of host rock, are in operation in Finland at Olkiluoto (see subsection 7.4.4.4) and Loviisa. All of these repositories are located below the water table. An example of a rock cavity repository in the vadose zone is the Richard II disposal facility, located in an abandoned limestone mine in the Czech republic. This particular repository, in operation since 1964, has its disposal chambers about 50 metres above the water table.

The borehole concept has been developed for the disposal of spent/disused sealed radioactive sources (SRS) and LILW exceeding waste acceptance criteria for NSD. Some current borehole disposal designs are lined, for example, with reinforced concrete and stainless steel, and backfilled in situ with cement grout or a low melting point metal alloy. Although some aspects of the post closure safety of disposal in boreholes are still being evaluated, there is a great deal of interest in the borehole disposal concept because almost all Member States have some inventories of spent/disused SRS and require facilities for their safe disposal. Further details are provided in see subsection 7.4.3.

Most human activities that could lead to wastes being disturbed inadvertently, for example home construction, farming and road building, generally penetrate a few metres below the surface, indicating that NSD are susceptible to human intrusion. Therefore, an important aspect of NSD is the requirement to assure institutional control over the repository site for a period of time. The greater depth (tens of metres) of rock cavity and borehole facilities indicate that these disposal concepts are more intrusion resistant than NSD facilities and, therefore, they are likely to have less stringent requirements with regard to institutional controls to assure safety. Exceptions are human activities such as mining and drilling of wells where the depth of penetration can be expected to be much greater.

Repository Design

Good design is an important step towards ensuring operational as well as long term safety of LILW disposal. Recognizing the need of Member States to obtain specific information, the Agency has recently produced a report [7.3] with the objective of providing guidance focused on the design of NSD facilities. In particular the following recommendations were made:

- The overall waste management system (treatment, conditioning, storage, transportation, siting, construction, handling, operation and closure) is considered in its entirety to ensure the safety of the disposal system. Disposal design is an important aspect of the overall waste management system,
- Although the various steps described are not necessarily to be followed universally, the concepts shown in the Figure 7-1, if followed, are expected to contribute to a safe design,
- It is important to recognize that the design process is iterative and requires input information from site and waste characterization programmes and safety assessments,
- The designer ensures that, from the outset of the design process, there is a clear understanding of the regulatory requirements of the facility life cycle,
- The designer identifies and takes into account, as soon as practical, all of the data/parameters required to achieve the final design in a timely manner, and
- The designer takes into consideration the cost implications of the facility from conception to closure and the cost implications of institutional control after closure.
Non-Technical Issues - the need for an integrated approach

While the underlying scientific and technical issues in support of repository development and radiological safety are key to the safe disposal of radioactive waste, it is now recognized that an integrated approach that embraces a broad spectrum of non-technical issues has to be considered in the repository development process [7.4].

Fundamental in repository site selection is the need to develop public acceptance. Successful siting exercises have involved the establishment of clear national policies with gradual step-by-step approaches conducted in an open, inclusive and consultative manner. Complex technical aspects must be expressed in a manner that can be understood by members of the public and, while media involvement is important, the commercial aspects of journalism must be appreciated. The concept of risk often features in siting considerations and the associated complexities and perceptions of risk by different groups needs careful consideration.

Despite the complexity of the task, over one hundred near surface repositories for the disposal of LILW have been successfully developed in a number of countries. The provision of effective safety with reasonable assurance has been achieved by limiting the amount of long lived nuclides, by provision of natural and engineered barriers and by the establishment of monitoring programmes and institutional controls. Nevertheless, there remains resistance both
in a number of public interest bodies and in some political circles over the development of new near surface repositories.

The objective is to define, through negotiation, what kind of residual risk societies are ready to accept collectively and what corresponding burdens they are ready to accept for themselves and future generations. The political dimension of the issue is quite obvious and may not be reduced to a mere business transaction. The debate must therefore be carried out in a difficult context taking into account the extreme sensitivity of the public on all matters relating to nuclear energy and to the pending confusion generated by the issues at stake on the future of nuclear energy.

An Agency report has recently been prepared on this topic to inform managers and decision makers in Member States [7.5]. In addition, the Agency has either issued, or is in the process of preparing, a number of other reports dealing with the non-technical aspects of near surface disposal. These include both the socio-economic and environmental impacts and the application of quality management principles and requirements throughout the repository life cycle, including the process of repository development and implementation.

### 7.3 Scientific and Technical Basis for Geological Disposal

A recent Agency Technical Report discusses the scientific and technical basis for geological disposal [7.6]. A synopsis is provided herein.

A geological disposal system can generally be defined as a combination of conditioned and packaged solid wastes and other engineered barriers within an excavated or drilled repository located at a depth of some hundreds of metres in a stable geological environment. The geological formation in which the waste is emplaced, referred to as the ‘host rock’, generally constitutes the most important isolation barrier. The various barriers act in concert, initially to contain the radionuclides, therefore allowing them to decay, and then to limit their releases to the accessible environment. The combination of engineered and geological barriers constitutes the “multibarrier” system.

Geological repositories have the greatest potential for ensuring the highest level of waste isolation and are considered applicable to the disposal of the most demanding categories of radioactive waste, including HLW, SF and other long lived radioactive waste. The emplacement of waste can be carried out in different ways and various repository designs are possible. The different types of geological environments that have been considered for the disposal of radioactive waste can contribute in different ways to the overall objective of ensuring containment of the radionuclides for the necessary period of time.

A key precept of geological disposal is that the combination of natural and engineered barriers should contain the short lived, highly active radionuclide content of the wastes completely, i.e. until their radioactivity has decayed to insignificant levels. This period is generally on the order of a few hundred to a few thousand years. There is broad agreement, however, that the majority of repository concepts cannot be relied on to completely contain all the long lived radionuclides present in the waste. Consequently, for the very long term, the main function of the geological disposal system is to delay and limit the release of radionuclides from the progressively degrading engineered barrier system (EBS) into the geological environment and eventually to the biosphere.
Suitable geological environments

A well chosen geological environment will act as a cocoon for the repository EBS, protecting it from gross fluctuations in physical stress, water flow and hydrochemistry. Large fluctuations in these properties generally arise from the conditions in dynamic regions of the lithosphere, such as tectonically active regions and moderately deep rocks and groundwater systems, which are easily and rapidly affected by unavoidable changes in climate and unpredictable changes in land use.

Deeper rocks are generally sheltered from these latter effects. Increased depth acts as a buffer against near surface perturbations and smoothes their magnitude in time. This is an extremely important function of the geological barrier, as long term stability in the ‘boundary conditions’ enables the only part of the disposal system that can actually be designed and optimized (i.e. the EBS) to function predictably for long periods of time.

Suitable geological environments for disposal of long lived radioactive wastes exist widely throughout the world. They can vary considerably in their nature and, thus, provide the desirable features mentioned above in different combinations and to different extents. Typically, suitable environments can be found in:

- extremely low permeability rocks in which advective groundwater flow is essentially precluded
  
  These include massive evaporite deposits, such as salt domes and large formations of bedded salt, and some plastic clay and mudrock formations. In such host rocks, provided geological stability is maintained, there is no natural mechanism for water borne radionuclide release to surrounding geological formations other than extremely slow diffusion through pore waters and along crystal boundaries unless the presence of the repository itself adversely affects host rock stability. However, because such possibilities exist, the evaluation of such host rocks at potential repository sites also involves consideration of the surrounding wider geological environment, in which advective flow may occur (e.g. in overlying and/or adjacent aquifers),

- deep groundwater systems that have displayed stable extremely low natural advective fluxes for periods of hundreds of thousands of years or longer
  
  Typically, the groundwater in such systems would be saline, and possibly even dense brine, as a result of the largely stagnant nature of the groundwater system, which is isolated from significant fresh water recharge. It would also be chemically reducing, which minimizes the mobilization and transport potential of many radionuclides, and

- groundwater systems that have low fluxes combined with long transport paths away from the disposal zone to potentially accessible groundwater systems or to the biosphere
  
  Such environments might display thick (hundreds of metres) stable unsaturated zones (the region above the water table) and slow long distance migration pathways in deep groundwater bodies. They may also occur in saturated rocks in some coastal regions or in massive sedimentary basins, where infiltrating groundwater moves slowly to great depths before eventual discharge, perhaps with considerable mixing and extensive dilution in near surface waters.
In such environments, provided repository construction is feasible both practically and economically, and provided that safety standards can be met, the exact nature of the host rock is not a controlling factor in the choice of a site. Experience in many countries over the last twenty or thirty years [7.7] has shown that acceptable conditions can be found in such diverse rock types as granites, metamorphic basement rocks, plastic clays, more indurated claystones, bedded evaporites, salt domes, porous volcanic tuffs, highly compacted volcanic tuffs and various well lithified sedimentary or volcano sedimentary formations. This range of geological environments is illustrated by the various host rock types listed in Table I.

Table 7-1: Host Rock Types Considered by Member States for the Geological Disposal of Solid Radioactive Waste

<table>
<thead>
<tr>
<th>Host rock types</th>
<th>Examples</th>
</tr>
</thead>
<tbody>
<tr>
<td>crystalline rocks</td>
<td>• granite, gneiss</td>
</tr>
<tr>
<td></td>
<td>Canada, China, Finland, Sweden</td>
</tr>
<tr>
<td>argillaceous</td>
<td>• strongly consolidated clay:</td>
</tr>
<tr>
<td>formations</td>
<td>claystone, mudstone</td>
</tr>
<tr>
<td></td>
<td>Hungary</td>
</tr>
<tr>
<td></td>
<td>• consolidated clay:</td>
</tr>
<tr>
<td></td>
<td>shale, marl</td>
</tr>
<tr>
<td></td>
<td>France, Switzerland</td>
</tr>
<tr>
<td></td>
<td>• plastic clay</td>
</tr>
<tr>
<td></td>
<td>Belgium</td>
</tr>
<tr>
<td>rock salt</td>
<td>• bedded salt</td>
</tr>
<tr>
<td></td>
<td>USA</td>
</tr>
<tr>
<td></td>
<td>• salt dome</td>
</tr>
<tr>
<td></td>
<td>Germany</td>
</tr>
<tr>
<td>volcanic tuff</td>
<td>• above water table</td>
</tr>
<tr>
<td></td>
<td>USA</td>
</tr>
</tbody>
</table>

Basic Disposal System Concepts

The majority of geological disposal systems under investigation involve the excavation of a repository at a depth of several hundred metres in an appropriate host rock in a suitable geological environment. In the most common approach, vertical shafts or access tunnels, or a combination of these, are then excavated to the planned depth. At this depth, horizontal disposal galleries are excavated where the waste packages are emplaced to be surrounded by selected buffer material(s).

Although the main effort has gone into assessing this type of excavated repository, disposal in deep boreholes drilled from the surface could be considered as a viable option for geological disposal. However, much less effort has been spent on the controlled emplacement of waste packages in deep boreholes.

Repository design is clearly highly specific to waste type and to geological environment. Large volumes of long lived waste with negligible heat generation are usually conditioned in relatively bulky (~1 to 10 m$^3$) packages. Repository concepts for their emplacement normally involve the construction of caverns with height and width dimensions of ~5 to 20 m, provided the geotechnical properties of the host rock permit the excavation of such large openings. Disposal HLW or SF normally involves smaller waste packages (with waste volumes of the order of 1 m$^3$) emplaced in tunnels, or boreholes from tunnels, with diameters from ~1 m to a few metres only. Regardless of the waste type and even for small quantities of waste, construction of the access and emplacement shafts and tunnels will involve the excavation of a substantial underground facility involving the removal of some hundreds of thousands of
cubic metres of rock, to millions of cubic metres for larger waste disposal programmes. Geological repositories presently being considered have underground dimensions varying from a few square km to as much as about twenty square km depending on the inventory of waste, on its thermal output and on the repository design.

The major natural and engineered components of a geological disposal system can conveniently grouped as follows:

- the waste form, i.e. the waste in whatever form it is at the time of emplacement in the containers
  
  Some low level wastes can be packaged without any treatment or conditioning, or simply after compaction to reduce their volume, while other wastes, generally characterized by higher levels of activity, are conditioned by dispersion in a stable matrix such as cement, bitumen or glass,

- the waste package, i.e. the combination of the waste form and any surrounding containment components.
  
  The purpose of the container can vary from short term containment during transport and/or storage to shielding and longer term containment. Depending on management requirements, packages can consist of untreated or treated and conditioned waste in steel drums, simple concrete containers and casks or more sophisticated stainless steel or other metal containers, such as vessels for vitrified wastes,

- the engineered barriers
  
  These include any overpack on the waste container (e.g. steel or concrete multi package containers for some LILW, and copper or titanium outer canisters for SF in some disposal concepts), the backfill/buffer material emplaced immediately around the waste packages (such as cement for some LILW concepts and highly compacted bentonite for several HLW and SF concepts) and the repository mass backfill in and around the region used for waste emplacement (often a mixture of crushed rock and clay),

- the repository
  
  This includes, for performance assessment (PA) purposes, the rock immediately adjacent to the excavations and the backfilling and sealing systems back to the surface,

- the natural barrier system
  
  This includes the geological formations surrounding and protecting the repository, between the disposal zone and the geosphere–biosphere interface. Various processes act to retard released radionuclides as they pass through the natural barriers, and
the biosphere

Radionuclides released from the geosphere move through various regions, being subject to dilution and re-concentration processes, although at very dilute conditions, simultaneously causing radiological impacts to humans and other species.

The role of the various isolation barriers may differ significantly in different disposal concepts, as the essential requirement is the overall safety of disposal and not the performance of single barriers. However, some redundancy in isolation capacity among the various isolation barriers may be beneficial for the presentation of the safety case, by increasing the confidence that the isolation system is actually capable of meeting the safety related constraints.

Repository Development Phases

Building and operating a geological repository for long lived radioactive waste is a unique endeavour for which there are almost no precedents on which to base many of the decisions that will be required. Nevertheless, despite the lack of practical experience in geological disposal, much is known worldwide about every aspect of the conceptual approach, design, evaluation and engineering of such facilities. This knowledge is based on extensive in situ and laboratory investigations and experiments, the construction and operation of underground research laboratories (see subsection 7.4.4.2) [7.8 – 7.10] and repeated iterative assessments of the performance of individual barriers and of the safety of complete conceptual disposal systems. Within any national programme, the stages in developing a geological repository would include the following [4]:

• general concept development
  This is based on the precise nature and estimated quantity of the waste requiring disposal and the geological constraints and local availability of materials in the country concerned. The safety concept for the possible disposal alternatives is developed and discussed,

• general concept evaluation
  This uses available studies as baseline information and develops them in terms of the specifics of a national programme, including initial evaluations of likely geological environments for disposal. The principal alternative for the disposal concept, as well as its possible variants, is defined and the safety concept selected,

• definition of general siting requirements to guide a site selection programme
  These would probably include a combination of safety requirements (long term safety, operational safety and safety of transport) and waste transport, cost, social and planning considerations, with the greatest weight given to providing an adequate degree of radiological safety,

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4 It should be noted that these stages do not need to be developed sequentially, but may be accomplished in parallel.
- site selection
  This activity may involve the investigation and evaluation of a number of sites,

- detailed characterization of a selected site
  This includes both surface based and direct underground access exploration and experimental techniques. The need for characterization may also call for testing and demonstration of the important parts of the planned repository system,

- design of the repository
  This involves making the best use of the characteristics of the site. Design includes the fitting of the adits to the repository and repository galleries in suitable volumes of rock,

- construction of the repository
  Construction could occur in a phased manner such that emplacement of some waste takes place and is evaluated before the repository is completely built. It allows, if required and within determined time limits, the retrieval of emplaced waste packages, and

- operation
  Operation could take place over several decades. It would be followed by decommissioning of surface facilities and closure of underground openings. This may again be phased over a long period of time in order to demonstrate and obtain acceptance for the final closure.

On the basis of scientific and technical considerations, and after closure of the repository, no monitoring and/or active surveillance of the site are required. However, reasons have been suggested, either for the sake of public reassurance or to ensure safeguards, for some kind of monitoring and surveillance to continue for an undefined period of time. It is also generally accepted that preservation of records and continuation of institutional controls after repository closure, regardless of the reason, would have the beneficial impact of minimizing the risk of inadvertent human intrusion (see subsection 11.3).

The R&D programme, in practice, starts from overall requirements for the safe isolation of radioactive wastes as illustrated in Figure 7-2. On the basis of these requirements, a conceptual design for the geological disposal system is selected. Existing scientific knowledge and understanding of the natural system of the geological environment forms the basis for the development of functional requirements for designing the subsystems necessary for assuring the reliable performance of the multibarrier isolation system. The design basis developed in this manner will contain the specifications needed for testing and verification of the design solutions. Safety Assessments and Performance Assessments play important roles in giving reasonable assurance on the overall safety of the disposal system and performance of the system components, thus indicating the robustness of the system. Even assuming that performance has been found adequate, the system can be further optimized with evolving scientific knowledge and technology.
7.4 **Issues and Trends**

The purpose of this subsection is to provide an overview of issues and trends relative to radioactive waste disposal in Agency Member States.

7.4.1 **Disposal of Very Low Level Waste (VLLW)**

7.4.1.1 **General Considerations**

As discussed in Reference [7.11], an issue of current interest in many Member States is the management, and in particular, the disposal of large volumes of very low specific activity,
low level radioactive waste. VLLW\(^5\) may be generated in a wide range of activities associated with the nuclear fuel cycle, nuclear applications in hospitals, research and industry, and non-nuclear industries. In particular, decommissioning of nuclear facilities can also give rise to large volumes of VLLW. Presently, there is no internationally agreed definition of VLLW. The definition of VLLW can vary from one Member State to another but it is generally accepted that VLLW is a subset of LILW and has an activity at levels that some jurisdictions may class as exempt or cleared from nuclear regulatory control. Despite these differences, it has been possible to estimate the magnitude of potential VLLW arisings and their characteristics.

With regard to disposal, it is clear that VLLW does not pose a sufficient risk to warrant disposal in an engineered LILW disposal facility. However, in some Member States disposal has taken place, or is planned to be implemented, in national disposal facilities for LILW, dedicated repositories for VLLW, local facilities on existing nuclear sites, etc. There is, as yet, no clear international consensus on the management of VLLW. In some countries, such as Sweden, Japan and France, VLLW is being disposed of in dedicated NSD facilities with minimal engineering. However, it is recognized that many Member States are still waiting for further actions to be taken on the basis of future guidelines and direction, for example from the Agency, and steps other countries are taking towards the disposal of VLLW.

7.4.1.2 Illustration - ANDRA's VLLW Facility, Morvilliers, France

In France, VLLW is known by its French name and acronym (très faible activité, TFA). France's repository for VLLW is scheduled to open in the summer of 2003 \([7.12]\). The repository is designed to hold 650 000 cubic meters (about 750 000 metric tons) of VLLW with around 70% coming from decommissioning of nuclear industry installations and the rest from smaller producers. It is being built at Morvilliers, on land adjoining ANDRA's low- and intermediate level waste (LILW) disposal site at Centre de l'Aube. ANDRA received a construction permit and facility license in August 2002.

The Morvilliers centre will host three main categories of radwaste: inert waste such as concrete, gravel and soil; wastes falling into the category of so-called "banal" materials from nuclear facilities such as ventilation tubes and slightly contaminated piping; and waste resembling so-called "special" waste from non-nuclear facilities.

Some 25 000 metric tons per year of VLLW will be produced over the next 30 years in France, according to ANDRA’s projections. The average price will be about an order of magnitude lower than that for handling and disposing of LILW at Centre de l’Aube.

Other countries have planned to send decommissioning wastes to existing nuclear waste disposal sites or to recycle them in the general economy. French authorities, however, chose some years ago to establish a separate category for materials considered too radioactive to be placed in industrial landfills but not radioactive enough to require the precautions taken at Centre de l’Aube. Rather than set a universal contamination threshold below which waste

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\(^5\) Note: VLLW, LLW and ILW are not part of the Agency’s proposed waste classification scheme (see subsection 3.2). The waste classes cited in subsections 7.4.1.1 and 7.4.1.2 are part of France’s waste classification scheme.

\(^6\) Agence nationale pour la gestion des déchets radioactifs, the French national waste management agency
would no longer be classified as radioactive, French safety authorities instead dictated a process whereby producers determine the categories of waste and plan ahead for their future disposition.

Electricité de France, in particular, has been anxious to see the Morvilliers centre open for business because its old reactor sites have been storing wastes from decommissioning of first generation gas-cooled reactors and the legal status of those storage facilities was uncertain.

### 7.4.2 Upgrading Existing Near Surface Disposal

#### 7.4.2.1 General Considerations

Article 12 of the Joint Convention [2.1] states: “Each Contracting Party shall in due course take the appropriate steps to review: ... the results of past practices in order to determine whether any intervention is needed for reasons of radiation protection bearing in mind that the reduction in detriment resulting from the reduction in dose should be sufficient to justify the harm and the costs, including the social costs, of the intervention.”

As discussed in Reference [7.11], extensive experience has been gained in many countries in disposing LILW in NSD facilities. The development, construction and operation of NSD facilities have been generally implemented in accordance with regulations and procedures applicable at the time of initial licensing. However, circumstances can develop during the lifetime of the repository that may necessitate upgrading of the facility design or operational procedures to improve the overall repository performance and safety.

Given that many existing disposal facilities have been in operation for decades and are now subject to more stringent regulatory requirements, there is a current trend towards upgrading these facilities in order to meet the new regulatory criteria. Some disposal facilities are already implementing improvement measures ranging from installation of improved caps, slurry walls, monitoring and surveillance equipment, and erosion mitigation barriers to retrieval and repackaging of degraded waste packages and improved waste acceptance criteria.

Some examples of NSD facilities that are currently undergoing upgrading include the Novi Han repository in Bulgaria (see subsection 7.4.2.2) [7.13], the Püspökszilagy repository in Hungary (see subsection 7.4.2.3) [7.14], and the Ekores disposal facility in Belarus. There is, therefore, a wealth of relevant experience and knowledge already available on the approaches and technologies that have been employed, and this can be applied to existing NSD facilities that require upgrading to improve their performance and safety [7.11]. It is anticipated that many repositories are likely to be subjected to improvement measures in the next few decades, especially those currently in operation in Central and Eastern Europe and States of the former USSR. The Agency is currently preparing a document on the technical issues and experiences related to the upgrading of NSD facilities.

#### 7.4.2.2 Illustration - Upgrading of the Novi Han Repository, Bulgaria

The Novi Han Radioactive Waste Repository is the only national radioactive waste disposal site in Bulgaria. It is located in Losen mountain, 6.5 km from a small village named Novi Han. The distance to the capital Sofia is about 35 km. The repository accepts radioactive waste generated from nuclear applications in industry, medicine, research and education. The facility was constructed according to a modified Soviet design (type TP-4891). A license for
construction of the facility was issued in 1959 and for commissioning in 1964. The repository was specially built for the needs of the Institute of Physics, in support of its research reactor IRT-2000, and other institutions in the country that used sources of ionizing radiation and generated radioactive waste.

The repository was operated more than thirty years without any accident or release of radioactivity to the environment but also without any investment for upgrading. As a consequence, the Committee on the Use of Atomic Energy for Peaceful Purposes (CUEAPP) temporarily stopped the operation of the repository in October 1994 until measures for improvement could be undertaken.

In 1995, The Institute for Nuclear Research and Nuclear Energy (INRNE) initiated a program for upgrading the repository and developed an implementation plan. The activities are supported by the Bulgarian Academy of Sciences, the CUEAPP, the Agency with Model Technical Project BUL/4/005 “Increasing Safety of Novi Han Repository” for the period 1997 – 2000 BUL/4/005, and the Bulgarian Government with financing from the state budget in 1998, and subsequently from the State fund “Safety and storage of radioactive waste”.

A feasibility study for “Reconstruction and Modernisation of Novi Han Radioactive Waste Repository (NHRWR)” was contracted by INRNE at the end of 1999 to EQE Bulgaria AD. The initial results were reported in September 2000. The objectives of the study were the development of conceptual solutions for reconstruction of existing and construction of new facilities at the NHRWR site as follows:

- storage of unconditioned high level sources,
- temporary buildings over the existing radwaste storage vaults,
- a rainwater draining system,
- a plant for conditioning of high level spent sources,
- a plant for conditioning of low and medium level radioactive waste (these are Bulgarian classifications, see subsection 3.2), and
- a 150 m$^3$ water reservoir for external fire extinguishing.

In addition, the study had to propose engineering solutions for conservation of the existing vault filled with high level spent SRS (the gamma-well).

The study defined the principal technical solutions for each of the existing and new proposed facilities. It also identified the required new facilities and general site infrastructure systems, which were not requested by INRNE but are required or recommended in accordance with current safety and radiation protection requirements.

In addition, the study also identified areas where the currently available data and information was insufficient for contracting the technical design for the reconstruction and additional investigations were required. Administrative and regulatory problems related to the reconstruction, which may influence the time schedule, were also identified.

The NHRWR Reconstruction and Modernization Programme, developed at the end of the study, assumes all the above factors in order to present a realistic time frame for all investigations, design and licensing activities and construction work at the site. Currently the
Programme assumes activities for NHRWR will continue until 2007 when all new facilities at
the site can be commissioned. The repository is expected to be licensed in 2004.

A Cost Estimate (by activity and facility) and an Investment Plan complement the
Programme.

7.4.2.3 Illustration - Upgrading of the Püspökszilagy Repository, Hungary

In Hungary, application of open and sealed sources of radioactivity on a large scale began in
the second half of 1950’s. A research reactor was commissioned at 1959 in the Central
Research Institute for Physics in Budapest. The first nuclear power plant unit went into
operation in 1982.

Currently there are two main groups of institutions that generate LILW. The first group, small
scale or non fuel cycle producers, includes hospitals, laboratories and industrial companies.
The other main waste producer is the Paks Nuclear Power Plant with its four WWER-440
reactors.

Spent/disused SRS represent a major source of waste to be disposed (see subsection 9.3.4). It
is estimated that only 0.5% of SRS in Hungary have been collected and disposed. In addition,
Hungary is a significant exporter of SRS and recent licenses have included a commitment to
repatriate SRS originating from Hungary.

In 1960, a temporary waste disposal facility was set up at Solymár in the north-western
outskirts of Budapest. About 900 m³ of LLW were placed into 3-4 m deep concrete wells.
Once the site was determined to be inadequate, the Hungarian Atomic Energy Commission
(HAEC) decided to establish a new radioactive waste disposal facility for institutional wastes
close to the main production centre (Budapest). In December 1976, the Radioactive Waste
Treatment and Disposal Facility was commissioned some 40 km north-east of Budapest, near
the village of Püspökszilágy. In 1980, the Solymár site was cleaned up and closed by
transferring of all waste to the new facility.

The repository was commissioned in 1976 and was operated by the Budapest Branch of the
State Public Health and Medical Officer Services. Since July 2, 1998 the newly formed Public
Agency for Radioactive Waste Management (PURAM) has taken over the operation tasks.

The Püspökszilagy repository is now considered to be unsuitable for some of the waste it
contains. Based on safety assessments conducted, it was concluded that the long term safety
of the Püspökszilágy repository can be assured only with some technical or administrative
modifications to the facility.

In 1998, Hungary started systematic work on the safety upgrading of the Püspökszilagy
repository. During 1998 and 2002, the safety re-evaluation of the repository was the primary
focus with some basic modernization and refurbishment measures (replacing of the obsolete
equipment, supplementary site investigations, redefining the waste inventory, near-field and
far-field studies).

In 2003, a project was launched to decide on the most appropriate and acceptable methods for
enhancing the safety and to make the necessary preparations for the remedial measures.
Important elements of this new project include construction of the central store for long lived
waste, an inventory re-evaluation, a feasibility study, a detailed work programme, license
preparations and an application for international assistance.
The final step is the implementation of the safety measures that are envisaged to be started in 2004 and will last for years depending upon the measures selected.

The project will take into consideration all relevant Hungarian regulations, European Commission Directives, Agency Safety Series documents, expert recommendations, as well as other international recommendations regarding current best practice in this field.

When planning the forward programme, it should be recognized that the safety assessment process is an iterative cycle and further work is required to enhance and improve the Safety Analyses Report, which will be periodically updated and reissued during the operational phase. Decisions are required on which subject areas require more work and which are lower priority for the next iteration, and hence what tasks need to be done and the appropriate level of investment for each.

Based on the safety assessments carried out so far, some key recommendations, relating to the improvement of the site, can be formulated. Doses from future human disturbance vary according to the part of the facility that is assessed. It should be noted, however, that certain SRS are present that would give rise to very serious radiological consequences if they were ever handled.

Modifications to the facility might include, for instance, the introduction of grout into the vaults to provide an additional physical/chemical barrier to migration and potential intrusion. Alternatively, the modifications might specify an additional period for institutional control to be assured, to prevent inadvertent intrusion for a specified period of time. Recovery of some long lived waste, mainly SRS, has been seriously considered.

For any proposed intervention, the benefits, in terms of risk or dose averted, should be balanced against cost. In addition to the work on safety reassessments, it is necessary to develop short term and long term plans for providing storage and disposal capacity for all the waste types currently disposed at the site.

There are many issues that still need to be addressed subsequent to the safety assessments, which should not be regarded as final assessments of the performance of the facility. Rather, they should be taken as the initial assessments in a series. Having identified the key issues and uncertainties in the initial assessment, further work has been undertaken to resolve these issues and uncertainties, which will provide a good basis for the development of a more certain view in subsequent assessments. The results of the safety assessment are used to focus the subsequent research programme and to identify issues that require further consideration.

According to the plans of PURAM, the repository will be operational for an additional 40-50 years receiving the radioactive waste from small scale producers in the country. By the end of this period, a deep geologic repository is supposed to be available to receive those long lived wastes stored in Püspökszilágy facility that are not suitable for disposal in a near surface repository. Bearing this approach in mind, the initial work is to provide additional disposal capacity within the current site.

7.4.3 Disposal of Sealed Radioactive Sources (SRS)

In most Agency Member States, SRS are used extensively and widely in support of a broad range of medical, research and industrial applications [7.11]. The total inventory of SRS worldwide is estimated to be more than half a million. While many SRS have radionuclides
with a relatively low activity or a short half life, worldwide there are also many SRS with high activity or long lived radionuclides. Radium was commonly used in medical applications in the form of needles and tubes. Today, radium SRS constitute a significant problem because of the long half life and high toxicity of radium-226. The radionuclides used most commonly now to produce high activity sources include cobalt-60, cesium-137, and iridium-192. Once disused or spent, SRS that cannot be recycled, remanufactured or used for another purpose become radioactive waste that needs to be managed safely.

Considerable experience has been gained in the management of disused/spent SRS. In this regard, the Agency has an active programme dealing with various aspects of the safe management of disused SRS, ranging from the development of registry software to the conditioning and storage of SRS. However, experience in the disposal area has been rather limited. Many SRS containing short lived radionuclides can be stored until they decay to levels that no longer require nuclear regulatory control or they can be safely disposed in NSD facilities. However, SRS with long lived radionuclides and some short lived, high activity SRS are not suitable for disposal in conventional NSD disposal facilities. A greater degree of confinement is required for the safe disposal of these SRS.

In this context, work is currently under way at the Agency to assess the feasibility of using boreholes for the disposal of disused/spent SRS. The borehole disposal concept appears to be a particularly attractive and cost-effective option for the management of SRS in many countries, especially considering their relatively small volume and inventory, as well as their unique geometry. Current work is focusing on both the technology and safety issues associated with the development of a borehole disposal facilities. An Agency sponsored technical assistance project in South Africa is currently carrying out a field demonstration to assess the technical feasibility of the borehole disposal concept.

7.4.4 Development and Implementation of Geological Repositories

7.4.4.1 General Considerations

Radioactive waste of all types needs to be managed responsibly in facilities under institutional control to provide public safety, protection of the environment and security from accidental or deliberate intrusion. However, in the long term, long lived radioactive wastes need to be disposed of in a way that does not require continued institutional control. The concept of isolating long lived radioactive wastes from the human environment by placing them deep underground in repositories located in host rocks characterized by high stability and low or no groundwater flow, i.e. geological disposal, was proposed over 40 years ago. Geological disposal concepts have been developed to the present level after considerable thought, R&D and debate, including societal and ethical considerations.

Geological disposal is nearing implementation in several Member States and at least one geological repository located in bedded salt and destined to receive long lived waste with insignificant heat generation (WIPP in New Mexico, USA) is currently in operation. Nevertheless, support is being voiced by some sectors of society for postponement of disposal and for more review of different waste management strategies, including long term storage and partitioning and transmutation. While the debate is not yet closed on these issues, the progress that has been made on the scientific and technical aspects of geological disposal over recent decades gives assurances to the waste management community that geological disposal is a sound technical solution that is supported by good scientific understanding. This is a consequence of many years of scientific work carried out by numerous professional
institutions around the world, which have created the international basis for the demonstration and confirmation of the soundness of the geological disposal concept.

A large part of existing knowledge is generic in nature and is derived from both earth sciences and underground engineering work. However, much specific knowledge has been derived from the characterization of potential repository sites, from studies in underground research facilities and from the operation of underground repositories for the disposal of various types of waste, including hazardous wastes and LILW.

7.4.4.2 The Growing Role of Underground Research Facilities

Underground Research Facilities (URFs) have become an integral part of a number of national waste disposal programmes since they provide important and, at times, critical technical experience, knowledge, and confidence for the strategic elements on which the safety case of a final repository is to be based. The strategy for achieving and demonstrating safety is driven by science and engineering, and consists of three connected elements that need to be defined and developed:

1. **Facility siting and disposal system design**: – siting a repository in a rock mass with favourable isolation properties, developing durable, long lived waste containers compatible with the geologic environment, and developing robust engineered barriers,

2. **Underlying scientific and engineering support**: – organizing and conducting a rigorous programme of engineering and scientific investigations to provide the information necessary to design, verify the characteristics, and evaluate the performance of the disposal system, and

3. **Evaluation of safety**: – developing tools to carry out an analytical evaluation of the performance and safety of the repository for a variety of possible future scenarios. Although different programmes use somewhat different terminology, the activities carried out in URFs in support of the above goals can be broadly defined as:

   • **characterization** – *in situ* investigations to provide basic understanding of the geologic, hydrogeologic, geochemical, structural, and mechanical properties of the host rock, its response to imposed changes, and data required for safety assessments,

   • **testing** – a broad term including: the evaluation of the performance of characterization methods in order to judge their applicability and reliability during future investigations; tests of engineered materials, excavation methods, etc. which may be used in the development of a repository; and testing of the conceptual and numerical models that are used to assess the performance of the repository system and/or its component parts,

   • **technology development** – the development of equipment, techniques, and expertise for characterization, testing, repository construction, waste emplacement (and retrieval), construction of engineered barriers, and repository closure, and

   • **demonstration** – illustration, at full- or reduced-scale and under real and/or simulated repository conditions, of the feasibility of the repository design and of the behaviour and performance of various components of the
repository, including, for example, demonstrations of sealing and waste emplacement and retrieval techniques.

Under the last heading, demonstration could also include trial disposals of actual waste in facilities in which the necessary licences had been granted. A variety of activities may be carried out in URFs, which range from basic research up to development of a pilot waste disposal facility.

Although there is a continuum of possibilities for the manner in which a URF might be developed, at least two broad categories can be distinguished:

- facilities that are developed for research and testing purposes at a site that will not be used for waste disposal, but provide information that may support disposal elsewhere, here termed “generic URFs”, and
- facilities that are developed at a site that is considered a potential site for waste disposal and may, indeed, be a precursor to the development of a repository at the site, here termed “site-specific URFs”.

Generic URFs may be developed to gain general experience of underground construction techniques, model testing, and verification of measurement techniques. They may also be developed to gain information, understanding, and experience related to a specific rock type that is considered as a potential repository host rock at a site, or sites, elsewhere. The type of generic URF to be developed will depend on the stage of the repository development programme. For example, in Switzerland, the general investigations at the Grimsel Test Site (see Figure 7-3) were begun in advance of any site or host rock type selection and this site has continued to be the focus of international research for almost 20 years, whereas the investigations in the Mont Terri road tunnel were begun specifically because this tunnel intercepts a clay formation that is being considered as a potential host rock elsewhere in Switzerland.

URFs of this type, which take advantage of the geological and infrastructure opportunities that already exist, are useful to develop experience in techniques relevant to site characterization and to repository construction, operation, and closure, and to develop understanding and test models. In some cases, the facilities may be limited in how they represent conditions in and around an actual repository but they provide cost-effective opportunities, especially in the early stages of repository programmes.

The establishment of a URF requires a significant investment in infrastructure support, in terms of excavation, construction, and maintenance of underground services and safety. For this reason, most generic URFs developed in Member States have been developed within, or as extensions to, excavations that exist already, such as mines and tunnels. Using an existing mine or underground access makes use of the initial excavation and in-place mine maintenance and safety infrastructure. It may also be easier to get planning permissions to extend work in an existing mine or tunnel as opposed to the development of a new site.
Geological disposal programmes in Member States are in various stages of development. Construction of a geological repository is not contemplated for at least one/two or more decades more in most cases.

Under the auspices of the Agency, nationally developed URFs and associated facilities are being offered for use by other nations [7.15]. The facilities form a Network of Centres of Excellence (the Network) for training in and development of waste disposal technologies. Experience gained in the operation of the facilities, and through associated experimentation and demonstrations, will be transferred to participating Member States through hands-on work in these world-class facilities. This co-operative programme is intended to embrace countries that do not have sufficient resources to build a URF or participate significantly in the work needed to establish the high level of confidence in the concept of geological repositories for radioactive wastes. In so doing, it is envisioned that existing interactions between Member States with well developed disposal programmes will be encouraged further.
The Network will be developed without delay to assist lesser developed Agency Member States, but it will need a decade or more to establish the broad and robust network of international expertise needed to ensure the efficient development of safe nuclear waste isolation systems worldwide. The more important objectives of the Network are to:

- encourage the transfer and preservation of knowledge and technologies
  While participation in current URF activities is the primary interest for some Member States, and will be of benefit to all, numerous Member States (especially in developing countries) need assistance for earlier phases of geological isolation studies such as relative merits of various host media, site selection criteria, site characterization (surface and borehole studies), overall system safety assessment etc.,

- supplement national efforts and promote public confidence in waste disposal schemes
  Many attempts are being made internationally to address social acceptance. The Agency has considerable experience in presenting various aspects of nuclear technology to non-specialist audiences.

The possibility of producing a series of publications outlining the history of geological isolation technology from its initiation in the early 1950s to the present emphasis on URFs and plans for the future, such as monitoring and reversibility, remains an option. These publications would be largely non-technical but would include references to key technical documents. While directed primarily at the public, such publications could also be valuable in providing specialists new to waste isolation with an understanding of factors, technical and non-technical, that have influenced waste isolation technology, and
• contribute to the resolution of key technical issues.

Members of the Network have identified some key technical issues that could be proposed as co-operative projects, for example:

• A reversibility/retrievability demonstration test in salt medium,
• A comprehensive R&D project on the long term monitoring of geological repositories, and
• R&D studies and demonstration on the application of safeguards standards during the operational phase and partial closure of a geological repository in clay.

The Network consists of members (Network Members) and participants (Network Participants) willing to share co-operative activities in training and demonstration of waste disposal technologies in underground facilities. Network Members are owners of facilities in Member States that have offered their facilities to be part of the Network. Network Participants can potentially come from any interested Member State.

Network Members and Participants, representing 20 countries, met in Winnipeg, Canada in September 2002 [7.16]. The meeting objective was the evaluation of the interested Member States needs for training in URFs in the context of their national programmes for developing geologic repositories. The outcome of the meeting will serve as a valuable basis for formulating detailed cooperation programmes between Network Members and Participants.

7.4.4.3 Illustration – Yucca Mountain Repository, Nevada, USA
based on input provided by B. McKinnon,
Yucca Mountain Project, Las Vegas, Nevada, USA

Yucca Mountain: A remote, dry site

Yucca Mountain (see Figure 7-5) is located on land controlled by the United States Government in a remote area in the southern part of the State of Nevada, approximately 160 kilometres northwest of the Las Vegas urban area. Yucca Mountain is in an area commonly referred to as the Great Basin, which encompasses nearly all of Nevada, as well as adjacent parts of the states of Utah, Oregon, Idaho, California, and northwestern Arizona. The Great Basin is internally draining (i.e., precipitation that falls over the basin has no outlet to the Pacific Ocean). Yucca Mountain is located in one of the most arid regions of the United States. Measurements of the water level in boreholes at Yucca Mountain indicate that the water table is approximately 500 to 800 meters below the ground surface.
Phased Development Allows Stepwise Decision Making

The United States’ deep geological disposal program began more than 20 years ago with the passage of the Nuclear Waste Policy Act in 1982 (NWPA), which set forth the general process for site characterization, recommendation, selection, and licensing for the several sites. Figure 7-6 indicates the phases and completed steps.

**Screening:** Screening began before the passage of the NWPA and continued through 1986.

**Site Characterization:** Site characterization started on May 28, 1986, with the US Department of Energy (DOE) recommendation, and presidential approval, of three sites (in the states of Nevada, Texas, and Washington), in three different
geologic media. In 1987, the US Congress amended the NWPA and redirected the DOE to focus site characterization on Yucca Mountain, which continued through 2002. The DOE completed a Viability Assessment in 1998, and concluded “…that no show stoppers have been identified to date. …and that scientific and technical work should proceed to support a decision…”.

**Site Designation:** The site characterization phase concluded with issuance of site recommendation documents, including the Final *Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada*, the *Yucca Mountain Science and Engineering Report, Technical Information Supporting Site Recommendation Consideration, revision 1*, and the *Yucca Mountain Site Suitability Evaluation*. Together with the Secretary of Energy’s recommendation, the President transmitted these documents as the comprehensive statement of the basis of the Secretary’s recommendation of Yucca Mountain. Congressional and Presidential approval occurred in 2002.

**Licensing and Construction:** With official site designation in 2002, the DOE is now preparing a license application to be submitted to the US Nuclear Regulatory Commission (NRC) for authorization to construct the repository.

**Repository Operations:** Following construction of the repository facility, the DOE will seek a license to receive and possess waste and, upon receiving this license, will begin waste emplacement in 2010.

**Repository Closure, Sealing, and Decommissioning:** At some future point, the repository is expected to be closed, which could be from 50 to about 300 years after waste emplacement begins. The NRC has established requirements for the information to be submitted for approval of an amendment to the license to permit closure. The NRC regulations also establish requirements for the programs that the DOE must have in place to monitor and control access to the repository site following closure. Closing the repository would require sealing all shafts, ramps, exploratory boreholes, and other underground openings connected to the surface. Such sealing would discourage human intrusion and prevent water from entering through these openings. The Federal Government will maintain institutional control of the site following closure. Active and passive security systems and monitoring are planned to prevent deliberate or inadvertent human intrusion and other human activity that could adversely affect the repository.

**Repository Design Evolution**

Development of the current design for the proposed repository and the mode in which the repository would be operated has been an ongoing, iterative process over many years, involving scientists, engineers, and decision-makers. The design and mode of operations of the proposed repository at Yucca Mountain have evolved as more was learned about the site and about the performance contribution of design attributes and operational objectives.

Several design evolution studies were undertaken in 2002 to evaluate ways to phase construction and to allow greater operational flexibility. As a result, several changes have been made to the design of the proposed surface facilities, the underground layout of the repository “footprint” and the construction sequence for both of these.
Surface Facilities

The surface facilities are being designed to receive, package, and support emplacement of waste in the repository. Previous designs included one large, integrated Waste Handling Building (WHB) capable of processing 3,000 metric tons of heavy metal (MTHM) per year. The WHB design included three transfer lines to support receipt and unloading of shipping casks, wet transfer and staging or blending in four large pools capable of holding 5,000 MTHM, as well as dry transfer and staging.

The current conceptual design is for dry processing, with a smaller fuel pool (capable of holding approximately 200 spent fuel assemblies) for off-normal operations. This approach is similar to facilities at La Hague, France, that use the wet system for items that would disrupt and delay processing within the dry system.

The different building configuration needed for dry handling allows for a number of smaller waste handling facilities to be constructed in phases, thus reducing risks that any future constraints on funding could delay initial waste receipt.

Subsurface Facilities

The subsurface design evolution study focused on reducing uncertainties related to rock properties and post closure performance of the natural barrier components of the repository. The focus was also on phased development of adjacent, but independent, emplacement panels. The current conceptual design for the repository “footprint” consists of five emplacement panels; four panels located in the primary block (Panels 1, 2, 3, and 5), and one located in the lower block (Panel 4). Construction of the underground repository will also follow a phased development. Excavation will start in Panel 1 to support emplacement of waste by 2010. The order of excavation of the other panels has not been determined at this time. Panel 2 will be accessed by a new North Construction Ramp.

This design has the following potential advantages:

- faster availability of emplacement areas,
- greater ease in separation of construction and emplacement functions,
- flexibility to accommodate possible changes in design or thermal loading strategies,
- simpler ventilation design, and
- better utilization of the repository emplacement area.

Ongoing Exploratory Studies Facility Testing

The Exploratory Studies Facility (ESF) is a U-shaped tunnel (Figure 7-7) approximately 7.9 kilometres long and 7.6 meters in diameter, located about 304 meters below the crest of Yucca Mountain. The ESF was excavated between 1994 and 1997 with a tunnel boring machine to provide direct access to the proposed repository horizon. Additional areas were subsequently excavated to enable direct observation of the geologic and hydrologic conditions, the engineering properties of the rock, and its response to construction activities, and to conduct specific tests.
Since the start of active testing in the ESF in 1996, more than 20 major experiments have been completed or are in progress. These include studying the factors affecting performance of the Yucca Mountain unsaturated zone, one of the natural barriers to radionuclide release, such as the study described next.

Chlorine-36 (Cl-36) is a radioactive isotope produced in the atmosphere and carried underground with percolating water. High concentrations of this isotope were added to meteoric water during a period of global fallout from atmospheric testing of nuclear devices during the 1950s and 1960s. This “bomb-pulse” signal has been used to test for the presence of fast transport paths in the unsaturated zone at Yucca Mountain. Because of the important implications of the occurrence of “bomb-pulse” Cl-36 to the site-scale unsaturated zone flow and transport model, a study is ongoing to confirm the apparent Cl-36 signal detected in earlier studies, in the vicinity of the fault zones accessed in the ESF.

The elevated Cl-36 signature appears to be confined to the immediate vicinity of faults (i.e. where structural features provide continuous flow paths from surface to depth). Fifty 4 metre long boreholes have been drilled in the areas adjacent to the two faults and the core obtained is presently being analyzed for Cl-36 concentrations. Corroboration of these results would demonstrate the existence of fast flow paths (50 years or less) from the surface to repository depths. This study is still underway with final results expected during 2003.

Another important study involves thermal tests in the ESF to obtain data necessary to determine thermally coupled processes. Loading the waste into the repository will result in heating of the geologic system. In order to understand the effects of the heat on hydrologic, mechanical, and chemical processes, and to validate the models of those thermally-coupled processes, field scale thermal tests are being conducted.

The ongoing Drift Scale Test (DST) excavation consists of an Observation Drift, a Connecting Drift, and a Heated Drift that is separated from the other drifts by a thermal bulkhead door (Figure 7-8). The DST is the largest scale test with a Heated Drift 47.5 meters
long and 5 meters in diameter. It is also the longest duration test – 8 years – in the thermal testing strategy for Yucca Mountain. The heating stage of the test started in late 1997 and the heaters were turned off in January 2002 after slightly more than 4 years of heating. The cool down phase is expected to last 4 years, after which the test equipment will be removed and portions of the affected rock mass will be sampled for post-test observations and characterization.

![Figure 7-8: Plan View Showing the Thermal Test Facility (TTF)](image)

Figure 7-8 shows the TTF and associated thermal test regions located in Alcove 5 extending off of the SF main drift. For the Drift Scale Test, the dark line represents the bulkhead and indicates the beginning of the Heated Drift. The niche shown just outside this bulkhead represents the location of the associated Plate Loading Test.

To monitor and quantify the coupled thermal (T), hydrologic (H), mechanical (M), and chemical (C) processes that occur, almost 150 boreholes were drilled to house the wing heaters and instrumentation packages (Figure 7-9). There are approximately 4000 sensors located throughout the rock mass and within the heated drift to record temperature, relative humidity, gas pressure, mechanical changes in the rock, micro seismic events, changes to water saturation, moisture movement, and fracture permeability. In addition, instrumentation allows the collection of water, gas, and rock samples for analyses of bulk chemistry and isotopic composition of gas and water in the test and mineral alteration. Just outside the Heated Drift bulkhead, an associated niche includes a plate-loading test to determine bulk thermo-mechanical properties.

Results of the DST supply detailed measurements of the thermal, mechanical, hydrologic, and chemical processes within the heated drift and the surrounding rock. Many of these data on coupled processes are used to either validate, or in some cases calibrate, the coupled process models (i.e., the coupled TH, THC, THM, and, ultimately, THMC processes) to provide confidence that those models capture these processes appropriately for their application to the proposed repository system. See reference [7.17] for additional information.
7.4.4.4 Illustration – Olkiluoto, Eurajoki, Finland

In May 1999, pursuant to the Finnish Nuclear Energy Act, Posiva Oy, the organization responsible for radioactive waste management in Finland, filed an application to the Government for a policy decision on the disposal facility for spent nuclear fuel, planned to be located on the island of Olkiluoto in Eurajoki [7.18]. The Finnish Radiation and Nuclear Safety Authority, STUK, issued a statement of the application in January 2000.

According to STUK’s statement, there were no safety issues that would exclude Olkiluoto as the site for spent fuel disposal. After the STUK statement, the Municipal Council of Eurajoki made a decision that supported the selection of Olkiluoto for the disposal site. The decision was passed in the Council with 20 votes in favour and 7 against.

In December 2000, the Finnish Government made a policy decision in favour of the Olkiluoto disposal facility, concluding that it would serve the overall good of society. Parliament ratified the policy decision in May 2001 on the basis of the statements of the Commerce Committee and the Environment Committee.

The policy decision assesses the significance and necessity of the disposal facility as well as its operating and safety principles. It also evaluates the suitability of Olkiluoto for disposal of spent nuclear fuel.

The selection of Olkiluoto is based on the geological suitability of the area, the acceptance of local population and the feasibility of the selection in terms of the environment and technology. Most of Finland’s spent nuclear fuel is accumulated in Olkiluoto and the existing technical infrastructure as well as the availability of industrial and other services in this region also support the implementation of the project.
In addition to Olkiluoto, Posiva also performed site characterization investigations in Hästholmen in Loviisa, in Romuvaara in Kuhmo and in Kivetty in Äänekoski. Based on the safety assessment and site characterization, all four site alternatives would have met the requirements given in the decision made by the Government on the safety of disposal of spent nuclear fuel.

The next stage involves construction of an underground characterization facility in Olkiluoto. Research activities will be carried out at the actual final disposal depth to acquire additional information and verify the present conclusions on the site suitability.

The underground rock characterization facility in Olkiluoto will make it possible not only to investigate the construction of bedrock but also the chemical reactions of groundwater, the durability of the materials to be used as well as the influence of any cracks and fractures deep inside the rock on groundwater movements. One important aspect is how the rock reacts to heat, as the final disposal canisters will be quite hot during the first few years. This information is needed to determine the spacing of the canisters.

According to plans, the application for a construction license for the disposal facility will be filed with the Government after 2010. In addition to the construction license, an operating license granted by the Government is required before the planned commissioning of the facility in 2020.

7.5 Collection and Dissemination of Radwaste Disposal Information at the International Level

The Agency’s Net Enabled Waste Management Database (NEWMDB, see subsection 11.1) is used collect and disseminate information on national radioactive waste management programmes, plans and activities, relevant laws and regulations, policies and radioactive waste inventories. Specifically, the NEWMDB is used to collect information about waste disposal facilities and the inventories of wastes in these facilities. See Figure 7-11 and Figure 7-12.
At time of writing, the participation by Agency Member States in submitting information to the NEWMDB was low (see subsection 2.3). However, participation is expected to increase during future data collection cycles. The intent is for the NEWMDB to be the most comprehensive source of information about waste disposal facilities and disposed waste inventories. For up-to-date information about waste disposal in Agency Member States, please refer to NEWMDB reports, which are publicly accessible and cost free (http://www-newmdb.iaea.org/reports.asp).

Figure 7-11: Example of Disposal Facility Information in the NEWMDB
Figure 7-12: Example of Waste Disposal Inventory Data in the NEWMDB

References for Section 7


7.15 International Atomic Energy Agency “Network of Centres of Excellence”
http://www.iaea.or.at/worldatom/Programmes/Nuclear_Energy/NEFW/wts_l3_02.html

7.16 Contact Point, IAEA Network of Centres of Excellence
geoldispnetwork@iaea.org

7.17 Yucca Mountain Project, “Drift Scale Heater Test in Cooling Phase”
http://www.ocrwm.doe.gov/ymc/science/test.shtml

7.18 Posiva Oy home page
http://www.posiva.fi/