# Deliverable 9.10: Collection and Analysis of Actual Existing Knowledge about Disposal Options for SIMS

Work Package ROUTES

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| Authors | **Alfredo Baptista (IST-WMO), Anastasia Savidou (NCSRD), Eileen Langegger (NES), Grażyna Zakrzewska-Kołtuniewicz (INCT), Guido Bracke (former GRS), Hana Vojtechova (SÚRO), Ioannis Kaissas (EEAE), Isabel Paiva (IST-RE), Jens Wolf (GRS), Jitka Miksova (SURO), Joerg Feinhals (DMT), Leon Cizelj (JSI), Marie Charlotte Bornhöft (DMT), Miloš Tomić (JSI), Roman Beyerknecht (NES), Stefan Coninx (DMT).** |

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# Executive Summary

The aim of this report is to compile the existing knowledge on the various options for the final disposal of radioactive waste that could be used for small amounts of waste. For Small Inventory Member States (SIMS), which generate only a relatively limited amount of waste from medicine, industry and research, including Research Reactors (RRs) and/ or Nuclear Power Plants (NPPs) with limited production of energy, disposing of their small inventory is challenging. Adopting a disposal solution developed for large inventories might turn out to be economically unfeasible as the fixed (non-volume specific) disposal cost may be very high. Commensurate disposal solutions are mostly not available for these countries, regarding safety and costs. Within this report, the focus is on single state solutions. Shared solutions for the disposal of radioactive waste are analysed within deliverables D9.12 [1], D9.13 [2] and D9.14 [3] of the EURAD project, as well as in previous projects such as SAPIERR [4] and SAPIERR 2 [5].

The following European Union (EU) Member States (MS) are considered as SIMS: Austria, Croatia, Cyprus, Denmark, Estonia, Greece, Ireland, Latvia, Luxembourg, Malta, The Netherlands, Poland, Portugal, and Slovenia. In this report, the information about the waste streams generated in the following 8 representative SIMS, which participate in ROUTES, was analysed: Austria, Cyprus, Denmark, Greece, the Netherlands, Poland, Portugal and Slovenia. The typical waste streams, the inventories of radioactive waste, the facilities for treatment, the techniques for characterization of the waste, the availability of Waste Acceptance Criteria (WAC) and disposal facilities in these 8 SIMS is briefly presented, and completed, when necessary, by information of Large Inventory Member States (LIMS) in order to highlight comparisons on specific points.

The waste from SIMS consists largely of industrial and medical waste concerning mainly Disused Shield Radioactive Sources (DSRS). The radioactive waste in most of the SIMS is classified mainly as Very Low-Level Waste (VLLW) and Low-Level Waste (LLW). In some SIMS there are usually small quantities of Intermediate-Level Waste (ILW) from decommissioning of nuclear facilities, mainly from RRs or small NPPs, and in some SIMS, there are also limited quantities of High-Level Waste (HLW) (i.e. spent nuclear fuel). Typical volumes of conditioned waste in SIMS vary from some hundreds to a few tens of thousands of $m^3$. Regarding the treatment facilities, some SIMS have decontamination systems wastewater treatment systems, shearing and cutting installations, cementation stations, compactors and super-compactors, incineration plants, hot cells for treatment of High-Activity Shield Sources (HASS) and drum drying facilities. SIMS generally have limited capabilities for processing the existing waste streams. In the SIMS participating in ROUTES, the characterization of waste focuses mainly on the radiological characteristics. In terms of chemical and physical properties, Austria, Slovenia and The Netherlands perform characterization of the waste. In addition to the technical requirements, most of the SIMS lack adequate legal framework, including administrative guidelines, such as WAC considering necessary Radioactive Waste Management (RWM) routes. WAC for disposal in SIMS are only generic, preliminary or not implemented. In the EU, Slovenia and Poland have a disposal facility for radioactive waste (Naturally Occurring Radioactive Material (NORM), LLW, ILW).

Existing categorisation of final repository types varies depending on the publication and approaches applied. Common categorisation refers to the depth of the final repository, e.g., near surface disposal, whereby borehole disposal sometimes is handled as a separate category. Due to the number of categorisation options publicly available and based on discussions held with task participants, the authors of this deliverable have decided to use a strict depth-dependent categorisation, even if this approach results in getting types of facility included and discussed in multiple categories. The long-term management solutions for radioactive waste are presented in chapter 4 of this report applying the categorisation:

- On-surface long-term interim storage;
- Near surface disposal options: Near surface disposal facilities (NSDF), caverns and bunkers, tunnels and galleries, boreholes, as well as silos;
- Geological Disposal Facilities (GDF): mines (converted and new), caverns, tunnels, deep boreholes as well as very deep boreholes.

SIMS need disposal options that are adapted to their waste while providing a satisfactory level of safety (depending on whether it is VLLW, LLW, ILW or HLW) with a reasonable use of resources available. Since most of the disposal options shown in chapter 4 cannot address all radioactive waste, combinations of these disposal options are discussed in chapter 5.

Another option is long-term interim storage, as presented in chapter 4 with the intention of being able to free release large quantities of radioactive waste at a later date. Depending on the waste, the quantities can be released directly or parts can be released after treatment. In all cases, the long-term interim

storage with a corresponding treatment/measurement can reduce the amount of VLLW and LLW, but in general does not replace a disposal facility.

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# Glossary

**Activation:** a process through which a stable nuclide is transformed into a radionuclide by irradiating with particles or high-energy photons the material in which it is contained.

**Barriers:** A natural or engineered obstruction that provides safety functions.

− **Multiple barriers:** Two or more natural or engineered barriers.

− **Natural barrier**: Barrier provided by the host environment.

− **Engineered barrier:** Barrier provided by engineered components.

**Characterization:** (see waste characterization)

**Class:** (see waste classes)

**Clearance** (from IAEA safety glossary–edition 2018)**:**
Removal of regulatory control by the regulatory body from radioactive material or radioactive objects within notified or authorized facilities and activities.

- Removal from regulatory control in this context refers to regulatory control applied for radiation protection purposes.
- Conceptually, clearance — freeing certain materials or objects in authorized facilities and activities from further control — is closely linked to, but distinct from and not to be confused with, exemption — determining that controls do not need to be applied to certain sources and facilities and activities.
- Various terms (e.g. 'free release') are used in different States to describe this concept.
- A number of issues relating to the concept of clearance and its relationship to other concepts were resolved in IAEA RS-G-1.7.

**Clearance levels:** Values established by the competent authority or in national legislation, and expressed in terms of activity concentrations, at or below which materials arising from any practice subject to notification or authorization may be released from the requirements of this Directive.

**Commissioning** (from IAEA safety glossary–edition 2018)**:** The process by means of which systems and components of facilities and activities, having been constructed, are made operational and verified to be in accordance with the design and to have met the required performance criteria.

Commissioning may include both non-nuclear and/or non-radioactive and nuclear and/or radioactive testing.

The terms siting, design, construction, commissioning, operation and decommissioning are normally used to delineate the six major stages of the lifetime of an authorized facility and of the associated licensing process. In the special case of disposal facilities for radioactive waste, decommissioning is replaced in this sequence by closure.

**Component:** (see Structures, systems and components (SSCs))

**Conditioning:** Those operations that produce a waste or spent fuel package suitable for handling, transport, storage and/or disposal. Conditioning may include the conversion of the waste to a solid waste form, enclosure of the waste in containers and, if necessary, provision of an overpack.

**Construction** (from IAEA safety glossary–edition 2018)**:** The process of manufacturing and assembling the components of a facility, the carrying out of civil works, the installation of components and equipment, and the performance of associated tests.

The terms siting, design, construction, commissioning, operation and decommissioning are normally used to delineate the six major stages of the lifetime of an authorized facility and of the associated licensing process. In the special case of disposal facilities for radioactive waste, decommissioning is replaced in this sequence by closure.

**Containment:** Provisions of a disposal system that limit the release and the dispersion of radioactive substances.

**Contamination:** The unintended or undesirable presence of radioactive substances on surfaces or within solids, liquids or gases or on the human body.

**Control:** Function, power or means of directing, regulating or restraining. For example, control typically implies not only checking or monitoring something but also ensuring that corrective or enforcement measures are taken if the results of the checking or monitoring indicate such a need.

- **Regulatory control:** Control or regulation applied to facilities or activities by a regulatory body.

**Controlled area:** An area subject to special rules for the purpose of protection against ionizing radiation or preventing the spread of radioactive contamination and to which access is controlled.

**Closure:** The completion of all operations at some time after the emplacement of spent fuel or radioactive waste in a disposal facility, including the final engineering or other work required to bring the facility to a condition that will be safe in the long term.

**Competent regulatory authority:** An authority or a system of authorities designated in a Member State in the field of regulation of the safety of spent fuel or radioactive waste management as referred to in Article 6 of Directive 2011/70/Euratom

**Decommissioning:** Administrative and technical actions taken to allow the removal of some or all of the regulatory controls from a facility (except for a disposal facility or for certain nuclear facilities used for the disposal of residues from the mining and processing of radioactive material, which are 'closed' and not 'decommissioned'). For a disposal facility, the corresponding term is closure.

- **Decommissioning as applied to a radioactive waste disposal facility:** includes decontamination, dismantling and removal of ancillary surface facilities, such as waste handling equipment and stores, so that long term protection of the public and the environment is achieved with respect to these facilities. It also includes dismantling and removal of operational equipment from the facility itself, as considered appropriate. Decommissioning is here distinguished from closure which includes all measures necessary to bring a radioactive waste disposal facility to a condition that will be safe in the long term.

**Decontamination:** The complete or partial removal of contamination by a deliberate physical, chemical or biological process.

**Discharge, authorized:** Planned and controlled release of (usually gaseous or liquid) radioactive material into the environment in accordance with an authorization.

**Disposal:** The emplacement of spent fuel or radioactive waste in a facility without the intention of retrieval.

**Disposal facility:** Any facility or installation the primary purpose of which is radioactive waste disposal. Such a facility may include natural and engineered components. The disposal system is composed of the part of the facility where the waste is emplaced, those parts of the host environment whose properties and behaviour contribute to post-closure safety and the auxiliary facilities needed for its construction, operation and closure.

**Disused source or disused sealed radioactive source (DSRS):** A sealed source which is no longer used or intended to be used for the practice for which authorization was granted but continues to require safe management.

**Dose limit:** The value of the effective dose (where applicable, committed effective dose) or the equivalent dose in a specified period which shall not be exceeded for an individual.

**Effective dose (E):** It is the sum of the weighted equivalent doses in all the tissues and organs of the body from internal and external exposure. It is defined by the expression:
$E = \Sigma_T w_T H_T = \Sigma_T w_T \Sigma_R w_R D_{T,R}$
where
$D_{T,R}$ is the absorbed dose averaged over tissue or organ T, due to radiation R,
$w_R$ is the radiation weighting factor and
$w_T$ is the tissue weighting factor for tissue or organ T.
The values for $w_T$ and $w_R$ are specified in Annex II. The unit for effective dose is the Sievert (Sv).

Usually the effective dose is mentioned simply as dose.

**Exempt Waste (EW)** (from IAEA SS No. GSG-1)**:** Waste that meets the criteria for clearance, exemption or exclusion from regulatory control for radiation protection purposes as described in IAEA SS No. RS-G-1.7.

**Facility:** An installation and its associated land, buildings and equipment in which radioactive materials are produced, processed, used, handled, stored or disposed on such a scale that consideration of safety is required. Processing facilities are normally stationary facilities but there are also licensed mobile facilities.

**High-activity sealed source (HASS):** A sealed source for which the activity of the contained radionuclide is equal to or exceeds the relevant activity value laid down in Annex III.

**Historical (or Legacy) waste:** In this document, the term historical or legacy waste means the waste generated as a result of past practices, which are often lacking sufficient physico-chemical-radiological characterization data. This waste has either: 1) not yet been processed into a form suitable for safe storage and disposal, conditioned, stored or disposed in a form that no longer complies with the regulatory requirements and re-treatment/re conditioning processes in line with current regulatory requirements and/or checking compliance with WAC of storage/disposal facilities is needed.

**Host environment:** The local geology (rock) within which a radioactive waste disposal facility is directly located.

**Isolation:** Provisions of a disposal system that ensure that the waste is protected from both natural and human external disturbances.

**Legacy waste:** (see historical waste)

**Licence:** Any legal document granted under the jurisdiction of a Member State to carry out any activity related to the management of spent fuel or radioactive waste, or to confer responsibility for siting, design, construction, commissioning, operation, decommissioning or closure of a spent fuel management facility or of a radioactive waste management facility.

**Licensee:** The licensee is the person or organization having overall responsibility for a facility or activity (the responsible organization).

Remark: WGWD recognizes that this organization may change as the facility passes to the decommissioning phase according to national strategies.

**Members of the public:** Individuals who may be subject to public exposure.

**Monitoring:**
1. The measurement of dose or contamination for reasons related to the assessment or control of exposure
2. Continuous or periodic measurement of radiological or other parameters or determination of the status of a system, structure or component. Sampling may be involved as a preliminary step to measurement.

**Natural radiation source:** A source of ionizing radiation of natural, terrestrial or cosmic origin.

**Nuclear facility:** A facility and its associated land, buildings and equipment in which nuclear materials are produced, processed, used, handled, stored or disposed of on such a scale that consideration of safety is required. (see also facility)

**Nuclear safety:** (see safety)

**Operation:** All activities performed to achieve the purpose for which an authorized facility was constructed.

**Operational limits and conditions:** A set of rules setting forth parameter limits, the functional capability and the performance levels of equipment and personnel approved by the regulatory body for safe operation of an authorized facility.

**Owner:** A body having legal title to waste or spent fuel including financial liabilities (it is usually the waste and spent fuel producer).

**Passive Safety Feature:** A safety feature which does not depend on an external input and/or continuous supply of media.

**Post-closure** (based on IAEA SSR-5)**:** Phase beginning at the time when all the engineered containment and isolation features have been put in place, operational buildings and supporting services have been decommissioned and the disposal system is in its final configuration.

**Practice (i.e. waste management practice):** A human activity that can increase the exposure of individuals to radiation from a radiation source and is managed as a planned exposure situation.

**Processing:** Chemical or physical operations on radioactive material including the mining, conversion, enrichment of fissile or fertile nuclear material and the reprocessing of spent fuel.

**Product (i.e. waste product):** The result of any processing step in the treatment and conditioning of radioactive waste.

**Protection and safety:** The protection of people against exposure to ionizing radiation or radioactive materials and the safety of radiation sources, including the means for achieving this, and the means for preventing accidents and for mitigating the consequences of accidents should they occur. Safety is primarily concerned with maintaining control over sources, whereas radiation protection is primarily concerned with controlling exposure to radiation and its effects. Clearly the two are closely connected: radiation protection is very much simpler if the source in question is under control, so safety necessarily contributes towards protection. Sources come in many different types, and hence safety may be termed nuclear safety, radiation safety, radioactive waste safety or transport safety, but protection (in this sense) is primarily concerned with protecting humans against exposure, whatever the source, and so is always radiation protection.

- **Radiation protection**: The protection of people from the effects of exposure to ionizing radiation, and the means for achieving this.
- **Nuclear safety**: The achievement of proper operating conditions, prevention of accidents or mitigation of accident consequences, resulting in protection of workers, the public and the environment from undue radiation hazards.

**Radiation protection:** (see protection and safety)

**Radioactive waste:** Radioactive material in gaseous, liquid or solid form for which no further use is foreseen or considered by the Member State or by a legal or natural person whose decision is accepted by the Member State, and which is regulated as radioactive waste by a competent regulatory authority under the legislative and regulatory framework of the Member State.

**Radioactive waste management:** All activities that relate to handling, pretreatment, treatment, conditioning, storage, or disposal of radioactive waste, excluding off-site transportation.

**Radon:** The radionuclide $^{222}Rn$ and its progeny, as appropriate.

- **Exposure to Radon:** means exposure to radon progeny;

**Regulatory body:** An authority or a system of authorities designated by the government of a State as having legal authority for conducting the regulatory process, including issuing authorizations, and thereby regulating nuclear, radiation, radioactive waste and transport safety.

**Regulatory control:** Any form of control or regulation applied to human activities for the enforcement of radiation protection requirements.

**Remediation** (from IAEA safety glossary–edition 2018)**:**
Any measures that may be carried out to reduce the radiation exposure due to existing contamination of land areas through actions applied to the contamination itself (the source) or to the exposure pathways to humans.

- Complete removal of the contamination is not implied.
- The use of the terms cleanup, rehabilitation and restoration as synonyms for remediation is discouraged. Such terms may be taken to imply that the conditions that prevailed before the contamination can be achieved again and unconditional use of the land areas can be restored, which is not usually the case (e.g. owing to the effects of the remedial action itself). Often remediation is used to restore land areas to conditions suitable for limited use under institutional control.
- In some contexts (e.g. the wider chemical industry), the terms remediation and restoration are used to describe different parts of overall recovery.
- The term cleanup is used in the context of decommissioning.

**Representative person**: An individual receiving a dose that is representative of the more highly exposed individuals in the population, excluding those individuals having extreme or rare habits.

**Reprocessing:** A process or operation, the purpose of which is to extract fissile and fertile materials from spent fuel for further use.

**Retrievability:** The ability to retrieve waste that has been disposed of.

**Safety:**

- **Operational safety:** The protection of people and the environment against radiation risks, and the safety of the facility and activities that give rise to radiation risk as a result of operations (such as construction, waste emplacement, decommissioning and closure) with due consideration to potential exposures during normal operation, anticipated operational occurrences and possible accidents.
- **Post-closure safety:** Protection of people and the environment against radiation risks after closure of the disposal system taking into account the various possible ways it might evolve.

**Safety analysis:** Evaluation of the potential hazards associated with the conduct of an activity.

**Safety assessment:** Assessment of all aspects of the site, design, operation and decommissioning of an authorized facility that are relevant to protection and safety. Note: assessment should be distinguished from analysis. Assessment is aimed at providing information that forms the basis of a decision on whether or not something is satisfactory. Various kinds of analysis may be used as tools in doing this. Hence an assessment may include a number of analyses.

**Safety case:** A collection of arguments and evidence in support of the safety of a facility or activity. This will normally include the findings of a safety assessment and a statement of confidence in these findings.

**Safety policy:** A documented commitment by the licensee to a high nuclear safety performance supported by clear safety objectives and targets and a commitment of necessary resources to achieve these targets. The safety policy is issued as separate safety management document or as visible part of an integrated organization policy.

**Sealed source:** A radioactive source in which the radioactive material is permanently sealed in a capsule or incorporated in a solid form with the objective of preventing, under normal conditions of use, any dispersion of radioactive substances.

**Siting (from IAEA glossary – edition 2007):** The process of selecting a suitable site for a disposal facility, including appropriate assessment and definition of the related design bases. The siting process for a disposal facility is particularly crucial to its post-closure safety; it may therefore be a particularly extensive process and can be divided into the following stages: concept and planning, area survey, fundamental site characterization, site confirmation.

**Source or radiation source:** an entity that may cause exposure, such as by emitting ionizing radiation or by releasing radioactive material.

**Spent fuel:**
1. Nuclear fuel removed from a reactor following irradiation that is no longer usable in its present form. The adjective 'spent' suggests that *spent fuel* cannot be used as fuel in its present form (as, for example, in *spent source*). In practice, however (as in (2) below), *spent fuel* is commonly used to refer to fuel which has been used as fuel but will no longer be used, whether or not it could be (which might more accurately be termed 'disused fuel').
2. Nuclear fuel that has been irradiated in and permanently removed from a reactor core.

**Spent fuel management:** All activities that relate to the handling, storage, reprocessing, or disposal of spent fuel, excluding off-site transportation.

**Standards:** Measures of quality or suitability for a specified purpose recognized by authority or by general consent and expressed in terms of quantitative and/or qualitative rules or criteria. Examples are quality standards and safety standards.

**Storage:** The holding of spent fuel or of radioactive waste in a facility that provides for their/its containment, with the intention of retrieval.

**Structures, systems and components (SSCs):**
A general term encompassing all of the elements (items) of a facility or activity which contribute to protection and safety, except human factors.

- **Structures** are the passive elements: buildings, vessels, shielding, etc.
- A **system** comprises several **components**, assembled in such a way as to perform a specific (active) function.
- A **component** is a discrete element of a system.

- **System:** (see Structures, systems and components (SSCs))

**Treatment (or Waste treatment):** Operations intended to benefit safety and/or economy by changing the characteristics of the waste. Three basic treatment objectives are:

- volume reduction,
- removal of radionuclides from the waste, and
- change of composition.

Treatment may result in an appropriate waste form.

**Waste:** Material for which no further use is foreseen (see also radioactive waste).

**Waste (or spent fuel) acceptance criteria (WAC):** Quantitative or qualitative criteria specified by the regulatory body, or specified by an opera-tor and approved by the regulatory body, for radioactive waste or spent fuel to be accepted by the operator of a storage facility. WAC might include, for example, restrictions on the activity concentration or the total activity of particular radionuclides (or types of radionuclides) in the waste or the spent fuel or criteria concerning the waste form or the packaging of the waste or the spent fuel.

**Waste acceptance criteria (for disposal):** Criteria applicable to waste packages and unpackaged waste accepted for emplacement in a disposal facility. Such criteria must be fully consistent with the safety case for the disposal facility in operation and after closure. They may include criteria introduced for operational as well as for safety reasons. They may be specified by the regulatory body or by an operator. If specified by an operator, they may be approved by the regulatory body.

Additional in this document:

- **Specific WAC** apply to specific facilities (Planned or existing)
- **Generic WAC** are sometimes used if specific WAC are not yet defined for facilities and are applicable to all facilities performing a certain function including those envisaged for the future.
- **Preliminary WAC** if they are not yet adopted or in use.

**Waste classes** (from IAEA safety glossary–edition 2018)**:** are those recommended in IAEA GSG-1.

- This classification system is organized to consider matters considered of prime importance for the safety of disposal of radioactive waste.
- The term “activity content” is used because of the generally heterogeneous nature of radioactive waste; it is a generic term that covers activity concentration, specific activity and total activity.
- The class “heat generating waste (HGW)” mentioned in the IAEA safety glossary–edition 2018 is sometimes used, for example in national classification systems, and are mentioned in the safety glossary to indicate how they typically relate to the classes in GSG-1.
- Other systems classify waste on other bases, such as according to its origin (e.g. reactor operations waste, reprocessing waste, decommissioning waste and defense waste).

**Waste, radioactive:** For legal and regulatory purposes, radioactive waste is material in gaseous, liquid or solid form that contains or is contaminated with radionuclides at concentrations or activities greater than clearance levels as established by the regulatory body, and for which no further use is foreseen.

**Waste characterization:** Determination of the physical, chemical and radiological properties of the waste to establish the need for further adjustment, treatment or conditioning, or its suitability for further handling, processing, storage or disposal.

**Waste form:** Waste in its physical and chemical form after treatment and/or conditioning (resulting in a solid product) prior to packaging. The waste form is a component of the waste package.

**Waste owner:** The waste owner means a body having legal title to waste or spent fuel including financial liabilities. It is usually the waste and spent fuel producer.

**Waste (or spent fuel) package:** The product of conditioning that includes the waste or spent fuel form and any container(s) and internal barriers (e.g. absorbing materials and liner), as prepared in accordance with requirements for handling, transport, storage and/or disposal.

# Acronyms and Abbreviations

| | |
|---|---|
| ADN | The European Agreement concerning the International Carriage of Dangerous Goods by Inland Waterways |
| ADR | The European Agreement Concerning International Carriage of Dangerous Goods by Road |
| COTIF | The Convention concerning International Carriage by Rail |
| DBD | Deep Borehole Disposal |
| DGD | Deep Geological Disposal |
| DIDO | A Heavy Water Type Reactor |
| DSB | The Directorate for Civil Protection and Emergency Planning |
| DSRS | Disused Sealed Radioactive Source(s) |
| DUMP | The Deep Underground Melt Process |
| EC | European Commission |
| EU | European Union |
| ENSDF | Engineered Near-Surface Disposal Facility |
| EW | Exempt Waste |
| FTIR | Fourier Transform Infrared Spectrometer |
| GDF | Geological Disposal Facility |
| HASS | High-Activity Sealed Source |
| HEPA | High-efficiency particulate air filter, also known as high-efficiency particulate absorbing filter and high-efficiency particulate arrestance filter |
| HLW | High Level Waste |
| HPGe | High Purity Germanium Detector |
| ICAO | Technical Instructions on the Safe Transport of Dangerous Goods by Air from the International Civil Aviation Organization |
| ICP-MS | Inductively Coupled Plasma Mass Spectrometry |
| ILW | Intermediate Level Waste |
| IMDG | The Safety of Life at See (SOLAS) Convention with the International Maritime Dangerous Goods Code |
| ISO | International Organization for Standardization |
| ISO/TC | ISO Technical Committee |
| LAP | Less Advanced Programs |
| LILW | Low and Intermediate Level Waste |
| LILW-LL | Low and Intermediate Level Waste – Long Lived |
| LILW-SL | Low and Intermediate Level Waste – Short Lived |
| LIMS | Large Inventory Member States |
| LLW | Low Level Waste |

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|---|---|
| LSC | Liquid Scintillation Counter |
| MAP | Most Advanced Programs |
| MOX | Mixed oxide nuclear fuel that contains more than one oxide of fissile material, usually consisting of plutonium blended with natural uranium, reprocessed uranium, or depleted uranium. |
| MOZAIK | Type of steel container |
| MS | Member States |
| NORM | Naturally Occurring Radioactive Material |
| NPP | Nuclear Power Plant |
| NPPs | Nuclear Power Plants |
| NSDF | Near Surface Disposal Facility |
| OES | Optical Emission Spectroscopy |
| PWR | Pressure Water Reactor |
| QMS | Quality Management System |
| R&D | Research & Development |
| RID | Regulation Concerning the International Carriage of Dangerous Goods by Rail |
| RR | Research Reactor |
| RWM | Radioactive Waste Management |
| SF | Spent Nuclear Fuel |
| SHARS | Short-lived High Activity Radioactive Source(s) |
| SIMS | Small Inventory Member States |
| SNF | Spent Nuclear Fuel |
| TENORM | Technological Enhanced Naturally Occurring Radioactive Material |
| TGA | Thermal Gravimetric Analysis |
| TRIGA | Training, Research, Isotopes, General Atomics (a type of light water research Reactor) |
| UNF | Used Nuclear Fuel |
| VDBD | Very Deep Borehole Disposal |
| VLLW | Very Low Level Waste |
| WAC | Waste Acceptance Criteria |
| XRD | X-Ray Diffractometric Analysis |
| XRF | X-Ray Fluorescence Measurement |

# 1. Introduction

The production of radioactive waste might be significantly different depending on the development of nuclear programmes. Even though the technical issues for LIMS and SIMS are often similar, the boundary conditions to be considered for the management of radioactive waste may be completely different.

SIMS can be defined as countries without nuclear power programme (thereafter SIMS with Less Advanced Program (LAP)) or with a small number of nuclear power plants (thereafter SIMS with Most Advanced Program (MAP)). These countries have small amounts of waste from research reactors and from medicine, industry and research but low volume from nuclear power plants. The following EU member states are considered as SIMS: Austria, Cyprus, Denmark, Greece, The Netherlands, Poland, Portugal and Slovenia (these 8 SIMS participating in ROUTES), as well as Croatia, Estonia, Ireland, Latvia, Luxembourg, Malta.

The objectives of ROUTES-Task 5 are notably to:

- Collect, analyse and compare the actual existing knowledge about disposal options for small amounts of waste.
- Describe the necessary predisposal routes for the disposal options.
- Evaluate the possible small-scale disposal solutions and describe their positive and negative aspects.
- Disseminate the results to other SIMS and describe the spin off for countries with large amounts of radioactive waste.
- Identify R&D gaps.

A country producing only a relatively limited amount of waste from medicine, industry and research, including Research Reactors (RRs) and/or Nuclear Power Plant (NPPs) with limited production of energy, the management of a small inventory can be challenging. Commensurate solutions are mostly not available for these countries, regarding safety, time and costs, whereas RWM in LIMS can be supported by existing facilities, infrastructures and knowledge. A key concern for SIMS, which typically have limited infrastructures and facilities (in operation or planned) is how to temporarily and safely manage existing waste streams while maintaining compatibility with disposal options . These steps include all the predisposal activities (characterization, classification/categorization, treatment, conditioning and storage). But moving forward for SIMS with programmes for treatment and packaging wastes in the absence of a disposal strategy and facilities is not only risky but the small volume to be managed can make the development of specific characterization and treatment capabilities disproportionately expensive [6].

Moreover, adopting a disposal solution developed for large inventories might turn out to be economically unfeasible as the volume-specific disposal cost may be very high. The downscaling of disposal concepts typically developed for LIMS for small quantities is failing and special concepts for SIMS are needed; such small inventories require safe and cost-effective disposal solutions. The disposal concepts for small inventories of intermediate and high-level radioactive waste are presented in IAEA TECDOC-1934 [7]). This report addresses the concepts of disposal for small quantities of radioactive waste belonging to all classes. However, disposal strategies established by SIMS are often less advanced, as these countries have not the expertise for planning, licensing, erection, operation and closure of a disposal facility.

Although Task 5 mainly explores the disposal strategies and options for SIMS, those disposal options might also be of interest to LIMS in case of challenging waste with specific properties, such as graphite, reactive metals, toxic materials/mixed waste and waste with excess plutonium. Some challenging waste in LIMS possibly cannot be disposed of because they don't fulfil national WAC for disposal within the existing or planned disposal facilities or after closure of an existing disposal facility. For such waste streams a smaller-scale disposal concept could offer a solution.

SIMS (and especially those with LAP) need to have an overview of existing disposal solutions and existing pre-disposal management solutions and practices. The scope of this Task is to support SIMS with LAP in developing their RWM strategy and to assist SIMS with Most Advanced Programs (MAP) in moving forward and developing an comprehensive/integrated disposal strategy. In this context, the work performed in subtask 5.1 and presented in this report aims at collecting and analysing actual existing knowledge about disposal options for SIMS. This work was performed complementary to the relevant IAEA initiative [7] [8]. The work presented in this report relies on:

- The data from the questionnaire of Task 2 regarding SIMS inventories for challenging waste, Link with other tasks 2, 3, 4, 6 + inputs of deliverable 9.4 (task 2), draft of D9.5 (task 2) and draft of D9.7 (task 3).
- Case studies of SIMS; notably in some countries, like Denmark, Slovenia and Norway, first conceptual designs to operational facilities exist. These options have been brought together with studies of other options being considered.

Two workshops were organised by Task 5 leaders in March 2020 and in January 2021. The outputs of these workshops which are relevant to Subtask 5.1 are compiled in this deliverable. The data from the questionnaire of Task 2 regarding SIMS inventories for challenging waste were collected and presented in chapter 2 of this report “Typical waste streams generated in SIMS”. These data were complemented by data for non-problematic waste for these countries to get a reliable basis for the required size of the disposal facilities. Information from other National Reports of these SIMS (i.e. Joint Convention, EC) was used and presented in this chapter as well, when necessary.

Chapter 3 “General differences in RWM between SIMS and LIMS” of this report was developed using inputs from the ROUTES questionnaire and in cooperation with tasks 3, 4 and 6, as well as with the expert team of PREDIS WP2 on Waste acceptance systems. The inputs used in this deliverable were verified by ROUTES participants from targeted SIMS countries. Several options for final solutions are in principle suitable for SIMS (Chapter 4). Existing categorisation of final repository types varies depending on the publication and approaches applied. Common categorisation refers to the depth of the final repository, e.g., near surface disposal, whereby borehole disposal sometimes is handled as a separate category. Due to the number of categorisation options publicly available and based on discussions held with task participants, the authors of this deliverable have decided to use a strict depth-dependent categorisation, even if this approach results in getting types of facility included and discussed in multiple categories. These options are:

- Long-term interim storage on surface for decay to reduce the amount of waste;
- Near surface disposal e.g. in silo-type facilities, shallow borehole (some tens meters), bunkers and caverns underground;
- Geological disposal in old mines, tunnels, deep shafts, boreholes adjusted to amount and activity (some 100 meter) and VDBD (some km).

In addition to the individual solutions, a consideration of combinations of the above-mentioned disposal options are presented in Chapter 5.

Case Studies describing the results of this work in more detail will be produced as part of Task 8 and delivered in the associated report for that task.

The expected outcomes of this deliverable are:

- SIMS (and especially those with LAP) have an overview of possible final solutions for small waste inventories to be supported in developing their radioactive waste disposal strategy;
- SIMS with MAP to be assisted in moving forward to develop a comprehensive/integrated disposal strategy.

# 2. Typical waste streams generated in SIMS

Information on the waste streams generated in the following SIMS participating in ROUTES was analysed: Austria, Cyprus, Denmark, Greece, the Netherlands, Poland, Portugal and Slovenia. The analysis was based on the description of the existing inventory of radioactive waste in these countries; the starting point was the classification adopted by the different SIMS. The ROUTES Task 2 Deliverable 9.4 [6] provides the approaches to radioactive waste classification based on questionnaires collected from all project partners, including SIMS. In this report, all analyses have been made based on the responses of these countries to the questions of the survey prepared in the framework of the ROUTES WP (answering the questionnaire prepared in Task 2), as well as national reports, such as reports for the Joint Convention on the Safety of Spent Fuel Management and on the Safety of RWM.

As reported in the questionnaires, most of SIMS adopted activity and half-life approach for classification; some directly the IAEA methodology [9]. All SIMS have the national inventory of radioactive waste described in national reports and other government documents; most of them have estimated the future occurrence of radioactive waste and simulated time span. Some of the countries emphasize roughness of the estimation.

Current and future inventories in SIMS include waste from nuclear decommissioning. Most countries already have such waste in their inventories, others provide estimations; it is mainly waste from the decommissioning of nuclear facilities like nuclear power reactors and RRs. In Denmark, large part of inventory originates from the decommissioning activities which have been ongoing since 2003 and will be finalized in 2025 according to the current plan.  The decommissioning waste are from three research reactors, a Hot Cell facility – cell row with five cells, a Fuel Fabrication Plant and the RWM Plant. Poland, a country without nuclear power reactors, has experienced from the decommissioning of the RR Ewa. The waste from decommissioning is disposed in the NSDF in Rozan (short lived waste) and stored in the premises of RWM Plant (ZUOP) (long-lived waste). Rozan is a disposal site for short-lived radioactive waste and storage facility for long-lived waste~~.~~

Even countries that have decommissioning waste and/or have assessed its volume, do not have the full characteristics of this waste, as mentioned in the questionnaire prepared under Task 2 as well as in Deliverable 9.4 [6].

## 2.1 The waste from Nuclear Power Plants (NPPs) and Research Facilities including Research Reactors

Among the countries selected for analysis, only Slovenia and the Netherlands have nuclear power reactors (*Table 1*). The other SIMS had RRs in the past, which now are at different stages of decommissioning, or still under operation. Only Cyprus does not have a RR, limiting its activities to the use of radioisotopes in research and medicine.

Typical waste generated in nuclear power plants includes spent ion exchange resins, ventilation filters (i.e. charcoal filters, HEPA filters), organic waste like personal protective items and laboratory waste (gloves, tissue paper etc.), activated and contaminated metals from maintenance, liquid and gaseous effluents, spent fuel - if it is declared as waste -, and waste from decommissioning. The origin of the decommissioning waste is mostly activated and contaminated solid parts of structural materials, moderator and coolant. Both activation products as well as contamination from the corrosion of the activated parts and fission products arising from the fuel are present in these components.  Non-radiological hazards of nuclear power plant waste are related to the toxicity of heavy metals and possible exothermic chemical reactions leading to the formation of flammable gases and heat [10].

The Krško NPP in Slovenia, connected to the power grid in 1981, was built jointly by Slovenia and Croatia which were then both part of Yugoslavia. The plant equipped with Westinghouse PWR with 1,994 MWt and 696 MWe runs on enriched to 5% $^{235}U$ fuel. The end of this NPP operation is scheduled for 2023, although an application for lifetime extension for 20 years has been made to the URSJV - Regulatory Body of Slovenia. The spent nuclear fuel assemblies will be stored in the spent fuel pool until the end of the plant life. Half of the waste must go to Croatia by 2025.

The only operating NPP in the Netherlands is Borssele (485 MW), which operation started in 1973 and now produces 4% of the country's electricity. According to the government's decision it will remain under operation until 2034. The older NPP, Dodewaard, was shut down in 1997. The policy for spent nuclear fuel is reprocessing [11]. In 2003 the interim storage facility for high-level waste was commissioned.

| Country | Nuclear power reactors | Nuclear research reactors |
|---|---|---|
| AUSTRIA | Zwentendorf NPP, closed in 1978, never entered service | • Austrian Research Centres at Seibersdorf— 10 MW ASTRA pool RR in use 1960–1999 (dismantled)<br>• **Atomic Institute of the Technical University Vienna — 250 kW TRIGA Mark II RR (in use since 1962)**<br>• Reactor Institute of the Technical University in Graz — 10 kW Siemens Argonaut RR (operated from 1965–2004, dismantled) |
| CYPRUS | No | No |
| DENMARK | No | • Risø – DR-3 DIDO class experimental reactor (shut down permanently in 2000) (inner parts dismantled, concrete<br>• shielding under decommissioning) Risø – DR-2 experimental reactor (shut down in 1975) (dismantled)<br>• Risø – DR-1 experimental reactor (shut down permanently in 2001) (dismantled) |
| GREECE | No | **GRR-1 – 5 MW open pool RR at National Centre for Scientific Research “Demokritos”, Athens, (extended shutdown)** |
| THE NETHERLANDS | Borrsele NPP in operation until 2034<br>Dodewaard shut down in 1997 | • **Reactor Institute Delft (2MW), part of Delft University of Technology (temporary shutdown)**<br>• **LFR, nuclear reactor in Petten**<br>• **HFR, nuclear reactor in Petten (shut down in 2010, dismantled)**<br>• Biologische Agrarische Reactor Nederland, part of Wageningen University, shut down in 1980<br>• ATHENE nuclear reactor, at the Eindhoven University of Technology, shut down<br>• KEMA Suspensie Test Reactor, test reactor at KEMA, Arnhem, disassembled 2003 |
| POLAND | No | • **Maria reactor - 30 MW RR, in operation**<br>• Ewa reactor - 10 MW VVR-SM RR (dismantled in 1995)<br>• Anna reactor - 10 kW RR (dismantled)<br>• Agata reactor - 10 W zero-power RR (dismantled)<br>• Maryla reactor - 100 W zero-power RR (dismantled)<br>• UR-100 reactor - 100 kW training reactor (dismantled) |
| PORTUGAL | No | • **Portuguese RR (RPI) - 1 MW open pool type (shut down)** |
| SLOVENIA | **Krško NPP (shared with Croatia), closure in 2043, in operation** | • **Ljubljana - 250 kW TRIGA Mark II RR, Jozef Stefan Institute (supplied in 1966 by the US); operational** |

*Table 1 – Nuclear reactors in SIMS (IAEA RRs database). Installations in operation are marked in bold*

Research facilities are very important tools for the further development of nuclear science and technology and for testing new systems and materials. RRs play a major role in such studies. Their operation contributes to the research, development and service provided to existing NPPs for the development of new technologies of the fuel cycle. Furthermore, they provide irradiation services for the industrial sector, agriculture and medicine.

Research facilities also include laboratories in institutions supporting both the work for nuclear energy and the application of nuclear methods in various areas of the national economy. All analysed SIMS have such institutes that conduct research and development works in the field of nuclear and atomic sciences; research in this field is also conducted at universities.

RRs require far less fuel and produce much less fission products from their use than NPP. On the other hand, their fuel often requires uranium which is more enriched, typically up to 20% $^{235}U$, although some older ones still use 93% $^{235}U$.

The radioactive waste generated in RRs contains mainly parts of experimental installations and packaging of irradiated samples. These are solid waste like irradiated target cans, used irradiation rigs and reactor components (e.g. thermocouples), neutron beam guide tubes, control rods as well as waste arising from the pool service area: charcoal filters, HEPA filters, spent ion exchange resins, organic waste like used personal protective items and laboratory waste (gloves, tissue paper, disposable glassware, etc.). Typical liquid and gaseous wastes are also generated in RRs [10]. There are radiological and non-radiological hazards, as in power reactors, only on a smaller scale.

## 2.2 Medicine and industrial facilities

Radioactive materials are used in a variety of medical applications for diagnostic, therapeutic and research purposes. Medical applications are now well-established arena for the use of radionuclides and DSRS. The production and subsequent handling of these materials results in the generation of a wide range of radioactive waste. The amount and types of these wastes depend on the specific medical application which involves different radionuclides, but they are in general short-lived, except for DSRS [12].

A wide variety of DSRS is used for industrial applications. The quantities of other types of waste from industry are relatively small, although different approaches to handling these various waste streams sometimes create difficulties.

Both types of radioactive waste, medical and industrial, are generated in all the SIMS participating in ROUTES.

## 2.3 Disused Sealed Radioactive Sources (DSRS)

DSRS are usually the main issue for SIMS.

DSRS are used in all SIMS. The main DSRS contain radionuclides: $^{60}Co$ $^{137}Cs$ $^{192}Ir$ $^{226}Ra$ $^{241}Am$ $^{90}Sr$ ($^{90}Y$) $^{75}Se$ $^{125}I$ or are neutron sources: $^{241}Am$, $^{238}Pu$, $^{239}Pu$ and $^{226}Ra$ to induce (α, n) reactions with light elements, e.g. beryllium, boron, lithium or fluorine [13] [14]. DSRS have been used widely in industry, medicine, and research. The potential hazard resulted from such waste depends on the type of the source and amount of radioactive material sealed in the capsule.

Some examples about DSRS management from the ROUTES questionnaire are presented below.

<u>Denmark</u>

DSRS originating mainly from medical and industrial applications in Denmark (originating from more than 50 years) has generally been packed in drums after removal of the original shielding. Stronger DSRS are stored in their original shielding. Dekom is considering recycling. Suitable containers for long term storage are under consideration.

<u>Greece</u>

The major elements of the national policy relating to the management of DSRS:

• For DSRS, repatriation is the preferred management option (back–end solution).

• For very short lived DSRS the decay and clearance options may be applied.

• On a 10-year periodic basis, if necessary, withdrawal projects are carried out to export DSRS to foreign authorized recycling facilities.

The state through the relevant legislation put a certain effort in order the DSRS not to be managed as radioactive waste. Therefore, the number of the DSRS that will not be exported and remain in the country will be as low as possible and it is currently not known.

DSRS mainly from industry, medicine and research are located at the NCSRD or at the premises of the waste owners under regulatory control.

<u>Poland</u>

Most of spent high activity gamma DSRS are transported back to the supplier abroad, but several of them, mainly of Soviet origin, still remain at the user's premises, or is stored at ZUOP storage facilities in Swierk Centre.

Some DSRS (mostly dismantled smoke detectors) have been disposed in Rozan disposal facility.

Slovenia

In the Central Interim Storage from small producers (CIS) in Brinje mainly ionised smoke detectors (146 packages) 7 $m^3$, containing $^{241}$Am and $^{226}$Ra; other DSRS (223 packages), 7 $m^3$, containing $^{241}$Am/Be, $^{226}$Ra, $^{60}$Co, $^{238}$U,$^{232}$Th, $^{63}$Ni, $^{55}$Fe, $^{90}$Sr, $^{106}$Ru, $^{152}$Eu, $^{137}$Cs, $^{85}$Kr, $^{133}$Ba, $^{241}$Am, are stored.

## 2.4 Naturally Occurring Radioactive Material (NORM)

A wide range of industrial activities, involving minerals and raw materials that are not even part of the nuclear fuel cycle, generate NORM. NORM is generally produced by industrial, mining and manufacturing processes. In general, when the concentration of natural radionuclides increases during a human activity, the material, classified as NORM, requires special handling. Recycling the NORM is the first option to consider [15] when its management is planned. Recycling NORM residues or using them as by-products depend on a variety of factors. They can be used in places where they do not present a risk, such as in the construction of tailings embankments, bedding material in roads, additive to cement (e.g. phosphogypsum). It also can be used as a cover and liner material for conventional landfill disposal facilities and in marine applications such as coastal protection [15]. When it is not possible, the material is managed as waste or disposed at specific facilities for NORM. Some data about the NORM in selected European countries provides the report [15]; information is also presented in the ROUTES Survey:

Austria

Potential industries-sources for NORM in Austria were identified as mining and milling of ores, phosphate and fertilizer industry, coal mining and burning, oil and natural gas industry, rare earths industry, zirconium industry, radium industry and thorium industry [16]. Legacy sites of processing pitchblende residues for the production of radium as well as monazite sand for the production of thorium and subsequent manufacturing of incandescent gas mantles were catalogued. The survey of these sites showed concentrations of radionuclides of natural origin, especially isotopes of thorium, uranium and their progeny. Some parts of the legacy sites were remediated, the rest has to be disposed of in a landfill suitable for NORM or classified as radioactive waste [16] [17].

Cyprus

The NORM waste originates from phosphate fertilizer industry.

Denmark

In general, storage of NORM waste is the responsibility of the operator. The number of cases involving disposal of NORM has so far been relatively limited in Denmark. The sources of NORM originate primarily from the oil and gas industry and production platforms located in the Danish sector of the North Sea. The disposal of NORM waste is mentioned in the Parliamentary Resolution - the NORM may be disposed of in the Geological Disposal Facility (GDF).

Greece

According to the published data for Greece (indoor radon concentrations, γ- dose rates etc.), the main work activities that may lead to a significant increase in the exposure of the workers or the public are: (a) mines and quarries, (b) thermal spas, (c) phosphate industry, (d) cement industries, (e) oil and gas industry and (f) caves visited by tourists. Until now, Greek Atomic Energy Commission, EEAE, has identified the following workplaces with significant increase in exposure due to natural radiation sources, in accordance with the Title VII Art 40 paragraph 2 of the 96/29/Euratom Directive: 2 fertilizer production industries, located in northern Greece and places where aero-engines constructed from Th-Mg alloy are repaired. Such alloys may have a $^{232}$Th activity up to 160 Bq/g.

Until recently, the means of waste disposal were as follows: a) Phosphogypsum disposal in stacks; b) Phosphogypsum disposal in open land; c) Use of phosphogypsum for agriculture purposes (saline soil improvement). Many old aero-engines, such as J33 (170 Bq/g of $^{232}$Th) and J79 (240 Bq/g $^{232}$Th), are disassembled. The disposal of their components containing Th-Mg alloys is under consideration [18].

The Netherlands

Dutch industries with known or potential significant discharges or residues of NORM were specified as follows: oil and gas, phosphoric acid (closed down), phosphorus production, iron and steel production, $TiO_2$ pigment, cement production, mineral sands and fertilizer production [18]. There are controlled releases to air and water from these industries, as well as solid residues from production. Some types of residues are reused (e.g. slag from phosphorus production), the others are stored. There are NORM landfills in Mineralz (near Rotterdam) and Nauea (Alkmaar).

Poland

There are some amounts of naturally occurred radionuclides present in the waste and by-products from copper and fertilizer industries, they are partly recycled and partly dumped.

There are also some 100 dumps resulted from past uranium mining activities, mostly abandoned, of waste rock and ore, reaching approximately 1.4 $x10^6$ $m^3$ as well as one tailing pond, which has been the object of a remediation project partly funded by the European Commission.

Portugal

The installations that produce significant quantities of NORM: phosphate industry, spas, uranium mining, coal burning. However, this would require appropriate investigation and methods of management [18].

Slovenia

Mining waste is a main problem related to NORM. All mining waste from numerous other mining waste piles has been moved to the Jazbec mining waste disposal site and disposed of. The total amount of disposed material on this site is 1,910,425 tons, with a total activity of 21.7 TBq. At the Boršt uranium mill tailings disposal site, 610,000 tons of hydrometallurgical waste, 111,000 tons of mine waste and 9,450 tons of material collected during decontamination of the ore mill have been disposed of, with a total activity of 48.8 TBq. Closure works at the Jazbec disposal site have been completed and the administrative procedure is in its final stage. The closure of the Boršt disposal facility has been delayed due to the activation of a landslide.

## 2.5 Challenging wastes

So-called "challenging wastes" are wastes for which there are no safe, efficient and cost-effective methods of characterization and processing. Different terms are used for challenging waste, like non-standard, special, unique, or problematic waste. An overview of the categorization and classification of radioactive waste is given in D9.4 of the EURAD project [6].

Usually, “problematic wastes” are defined as those for which:

- The technologies for the treatment of the waste types had not been established for routine RWM, or
- The type of waste does not meet the acceptance criteria for processing by generally available local technologies [19].

Problems related to specific waste can be identified not only by the characteristics of the waste, but can be facility-specific, site or even country specific. These problems may be related to the current regulatory framework in the country, existing experience in RWM, available disposal option, etc. The problematic waste could be very diverse, and what is considered problematic now at one site may become more or less routine later or at another site or facility [20]. These radioactive waste are often defined as challenging: DSRS; sludge; organic waste; ion exchange resins; bituminized waste; graphite; uranium/radium/thorium bearing waste; some types of decommissioning waste (soil, rubble etc.); waste containing chemotoxic material such as beryllium, mercury, asbestos, lead; historical (or legacy) waste which are the major issue in most SIMS.

# 3. General differences in RWM between SIMS and LIMS

## 3.1 Inventory of radioactive waste

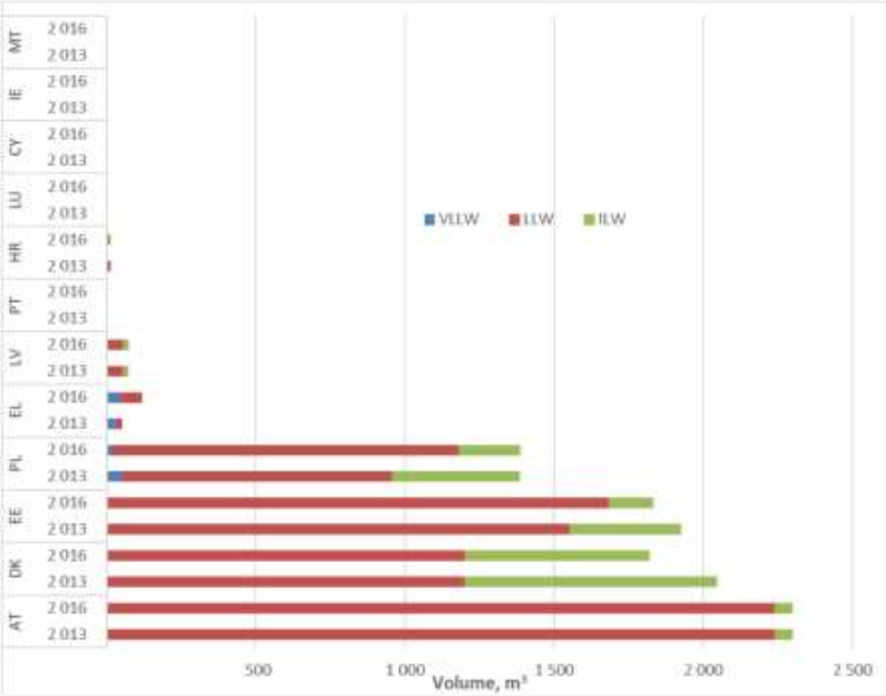
During the evaluation of the questionnaires in the framework of the EURAD programme to identify the inventory and future accumulation of radioactive waste in the MS, it became apparent that the information provided by each MS is difficult to compare. On the one hand, information about national inventories provided by the MS were in many cases unprocessed data in extensive text documents. The documents could not be evaluated since the posed status and situation of radioactive waste disposal in the MS were too complex. Furthermore, some of the national documents were only available in national language. On the other hand, there are different radioactive waste classification schemes in the MS, which differ from the radioactive waste classification of the IAEA. MS radioactive waste categories are based not only on activity levels (low-level, medium/intermediate level, high level, spent nuclear fuel) and on half-life (exempt waste, transition radioactive waste, short-lived, medium-lived, long-lived), but also on processing status (conditioned, unconditioned), physical state (solid, liquid, gaseous) and heat generation (heat generating waste, waste with negligible heat generation). Some of the categorisation schemes are also country-specific (Cat A, B and C or Cat 1, 2 and 3). In the end, the unit of measurement was both the weight in tons (t, Mg) and the waste volume in cubic meters ($m^3$). These different classification categories (and units) complicate the comparability of the inventories since there are no conversion factors to the IAEA classification.

At this point, a need for research and development can be directly deduced: For a comprehensive analysis and evaluation of radioactive waste, uniform or at least comparable classification schemes are necessary throughout Europe. These schemes need to consider among others:

- The classes of waste (e.g. according to IAEA: VLLW, LLW, ILW, HLW),
- The material composition (metal, concrete, resin etc.),
- The condition (residue, raw waste, packed or conditioned, non-conditioned),
- A EU wide uniform unit ($m^3$ or Mg) for each material.

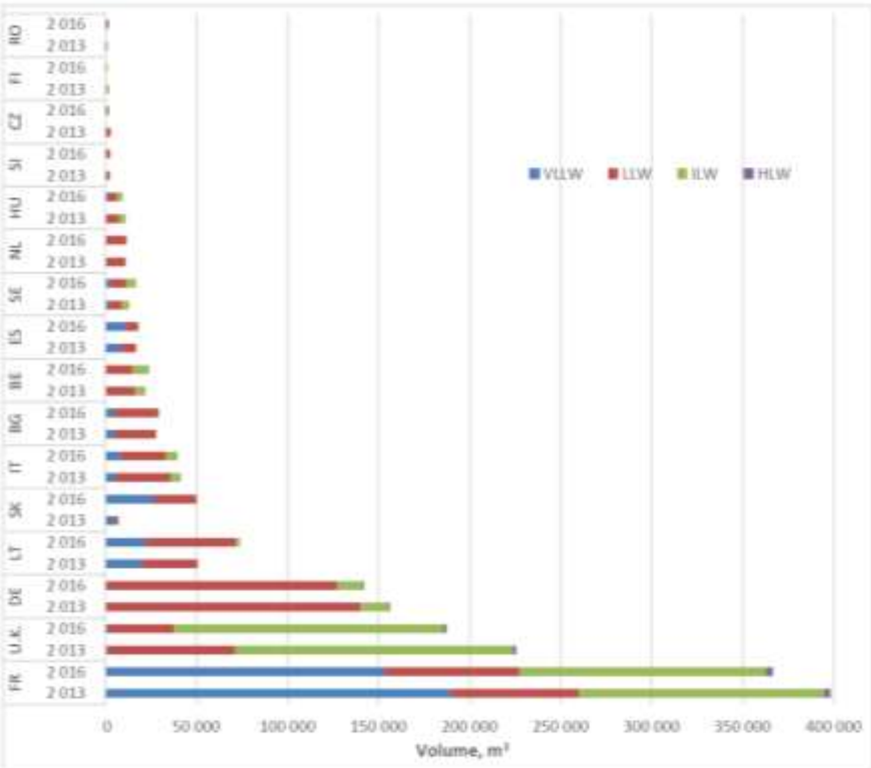
### 3.1.1 Existing radioactive waste in member states

A survey of existing radioactive waste was already carried out by the EC in 2017 [21]. The *Figure 1* and *Figure 2* are from this report and clearly show the differences between MS with and without nuclear reactors or rather nuclear power programme. The two figures show the accrued radioactive waste until 2016. In many MS with nuclear power programmes, the future radioactive waste is still permanently installed in the existing nuclear reactors and is therefore not included in these figures. Thirteen MS (both with and without NPP) already have a disposal site for their radioactive waste. However, in some cases the disposal of waste at several sites is being reconsidered. Hence, there are plans for the retrieval of the waste disposed of in the past. This waste volume is also not included in these figures and needs to be considered in future inventories.




*Key: MT: Malta; IE: Ireland; CY: Cyprus; LU: Luxembourg; HR: Croatia; PT: Portugal; LV: Latvia; EL: Greece; PL: Poland; EE: Estonia; DK: Denmark; AT: Austria*

*Figure 1 – Volumes of stored radioactive waste by class in MSs without nuclear power programme, end of 2013 and 2016* [21]




*Key: RO: Romania; FI: Finland; CZ: Czech Republic; SI: Slovenia; HU: Hungary; NL: The Netherlands; SE: Sweden; ES: Spain; BE: Belgium; BG: Bulgaria; IT: Italy; SK: Slovakia; LT: Lithuania; DE: Germany; UK: United Kingdom; FR: France*

*Figure 2 – Volumes of stored radioactive waste by class in Member States with nuclear power programmes, end of 2013 and 2016* [21]

### 3.1.2 Classification of SIMS and LIMS by their inventory

A clear distinction between LIMS and SIMS cannot be made. However, the figures show that there are clearly MS that belong to the SIMS group and others to the LIMS group.

All the MS in *Figure 1* belong to the SIMS group.

Most of the MS in *Figure 2* belong to the LIMS group. Even if some countries, such as Finland or the Czech Republic, only report very small quantities of radioactive waste, it must be considered that in all of the countries in *Figure 2* at least two nuclear power reactors must be dismantled and disposed of in the future. Since the power plants are still in operation or have not yet been dismantled, these masses are not yet included in the waste volume of *Figure 2*. The dismantling of a NPPs produces about 4600 Mg of radioactive waste (LLW, ILW) [22]. In addition, a running NPP produces several 10 Mg up to several 100 Mg (depending on work during revision) of operational waste per year. In some countries also the operational and dismantling waste from RRs needs to be considered. Due to the wide variety of designs and operating methods, it is not possible to estimate the amount of radioactive waste produced. Also the producers of clearance differ from country to country, so the amounts of exempted waste and VLLW can differ. Decommissioning waste from some RRs were examined in the [23]. *Figure 3* and *Figure 4* give an overview of different RRs and their decommissioning waste. The building structures represent the largest share of total dismantling waste. Therefore, the proportions may also vary significantly depending on whether the reactor buildings are also dismantled and disposed of or whether these buildings are further used.

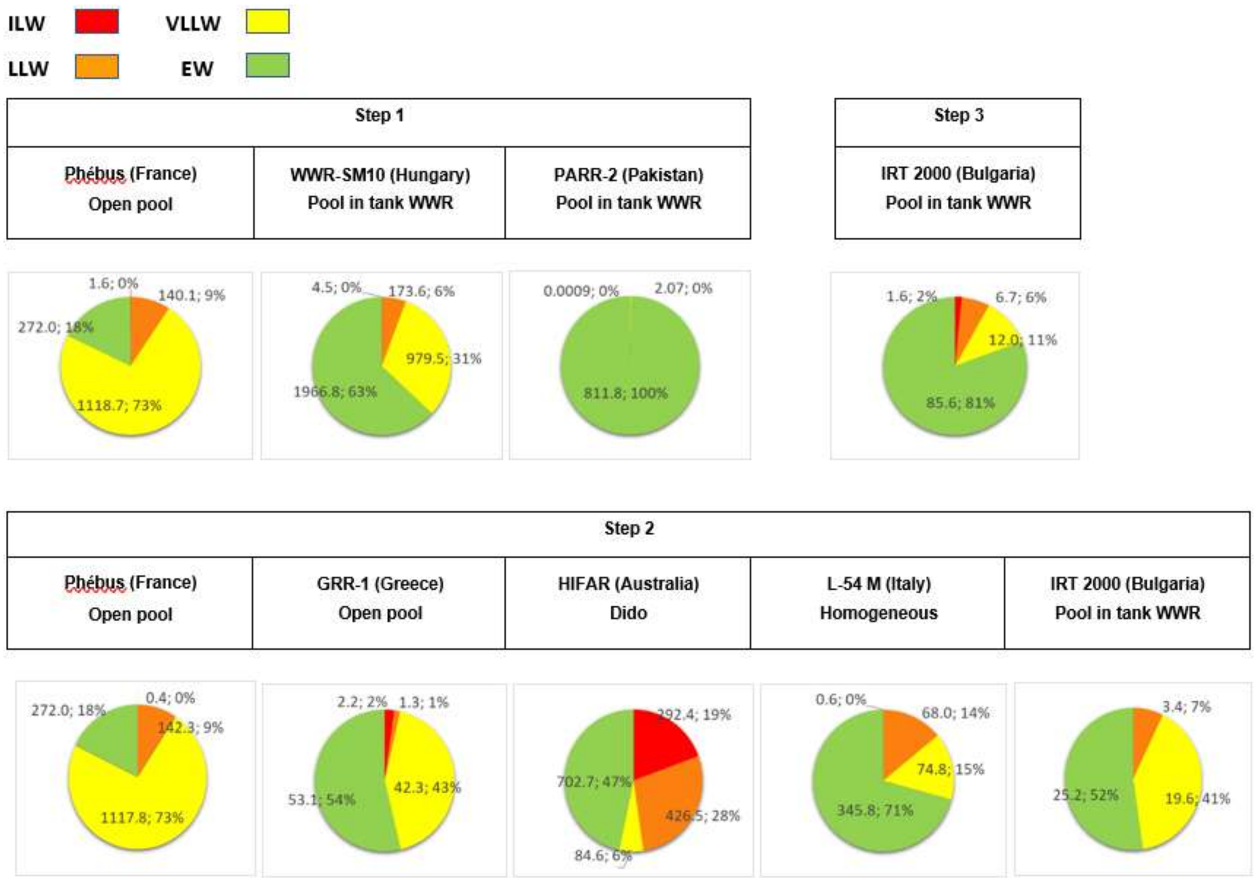



*Figure 3 – Mass in tons and distribution of dismantling waste for various types of RRs. Step 1: estimation based on similar reactors; Step 2: estimation based on radiological characterization results; Step 3: completed decommissioning projects* [23]


*Figure 4 – Mass in tons and distribution of dismantling waste for TRIGA of RRs. Step 1: estimation based on similar reactors; Step 2: estimation based on radiological characterization results; Step 3: completed decommissioning projects* [23]

| COUNTRY | Radioactive waste in the national inventory/ time span of estimation | References |
|---|---|---|
| DENMARK | Spent fuel:<br>4.9 kg solution of 20% enriched uranyl sulphate in light water<br>233 kg uranium oxide pellets mostly in zircalloy tube | [24] |
| THE NETHERLANDS | Spent fuel on 31st December 2016, stored at the COVRA facilities:<br>• 6.2 m$^3$ of RRs<br>• 1.2 m$^3$ Uranium targets<br>HLW at 31st December 2016, stored at the COVRA facilities:<br>• 91 m$^3$ HLW (vitrified waste from spent fuel of the Dodewaard NPP)<br>Future reprocessed spent fuel from Borssele NPP | [25] |
| POLAND | Spent fuel (150 Fuel assemblies) on 30th of June 2017, stored in MARIA RR pool:<br>• 21.836 kg U-235, average burn-up 60 % max<br>• 110.219 kg Uranium, average burn-up 60 % max | [26] |
| SLOVENIA | Number of spent fuel assemblies at Krško NPP: 884 (burn-up 18.6 to 53.2) | [27] |

*Table 2 – HLW and Spent Fuel in SIMS participating in WP ROUTES*

Almost all of the countries shown in *Figure 2* can be classified in the LIMS group. The exceptions to this rule are the Netherlands and Slovenia MS. These states operate NPPs but with limited energy production.

The MS are divided into SIMS and LIMS as follows:

- SIMS→ Austria, Croatia, Cyprus, Denmark, Estonia, Greece, Ireland, Latvia, Luxembourg, Malta, The Netherlands, Poland, Portugal, Slovenia.
- LIMS → Belgium, Bulgaria, the Czech Republic, France, Finland, Germany, Hungary, Italy, Lithuania, Romania, Slovakia, Spain, Sweden, United Kingdom.

The MS division in SIMS and LIMS is based on the information provided by the EC and on further assumptions and rough estimations.

It should be also mentioned that some countries considered as SIMS have also small quantities of HLW and spent fuel (see *Table 2*).

As already mentioned in paragraph 3.1, there is a need for research to develop a uniform classification scheme. Based on this uniform classification scheme, the MS may be able to quantify their national radioactive waste inventories in a comparable way. Moreover, there is a R&D gap in compiling the findings from previous dismantling and disposal programmes in order to make reliable estimates for the radioactive waste that be generated in MS in the future (e. g. dismantling of reactors, yearly operational waste etc.). This is where LIMS with their advanced decommissioning and disposal programmes can provide reliable information.

## 3.2 Treatment facilities at SIMS and LIMS

A non-exhaustive list of different treatment facilities in LIMS is given in *Table 3*. Treatment options in some SIMS are also mentioned. The information was shared by the countries or gathered from the organisations' websites. Table 3 shows the broad variety of treatment facilities offered. In LIMS, the treatment of all kinds of waste is possible. In France, in addition, spent fuel can be treated. Regarding SIMS, they have some treatment facilities which are already narrowed in potential applications, and as discussed in Task 6, shared solutions for treatment of radioactive wastes are necessary.

| Company | Location | Facilities for treatment of radioactive waste | Availability of Mobile solution (Yes/ No) | References |
|---|---|---|---|---|
| ARAO | Slovenia | NPP:<br>Different methods used for treatment of RW in NPP: existing treatment procedures reduce volume, extract radionuclides, change the composition of waste and minimise the discharge of radioactive substances into the environment (predisposal management for RW in NPP is done at the Krsko plant, either own it or outsource it)<br>CISF:<br>The radioactive waste inventory at the CSF is subject to treatment and conditioning with the aim of reducing its volume.<br>SF is not treated. | No | ARAO (p.19-21 and p. 27) |
| Belgoprocess | Belgium | • Treatment and conditioning of all types of radioactive waste<br>Plasma thermal technologies | No | Belgoprocess |
| Billfinger | Germany | • Broken down, sorted and compacted<br>• Depending on the level of contamination, subjected to varying waste treatment processes<br>• Cementing systems<br>• Uncapping facilities<br>• Safety housing<br>• Double-cover locking<br>• Manipulators<br>• Decontamination<br>• Mobile water purification plants are also available | Yes | Billfinger |
| COVRA | The Netherlands | • Separator for organic/inorganic liquids<br>• Dedicated incinerator for biological wastes<br>• Dedicated incinerator for organic liquids<br>• Shearing and cutting installation<br>• Cementation station<br>• Wastewater treatment system | | COVRA |

| | | | | |
|---|---|---|---|---|
| Cyclife | France, Sweden, UK | • Metal treatments: segmentation, decontamination, melting, clearance and recycling of produced metal ingots, and characterisation and conditioning of secondary waste for return to customer<br>• Thermal treatment of organic waste by incineration or pyrolysis<br>• Mobile units to condition ion exchange resins from NPPs<br>• Mobile encapsulating unit conditions radioactive concentrates<br>• Provides a range of services for handling and treating waste on site (decontamination, remediation of circuits and containers…) | Yes | Cyclife |
| Eckert & Ziegler | Germany | • DSRS recycling<br>• Waste transport services<br>• Volume reduction by compaction or incineration<br>• Packaging and conditioning of waste in approved containers | No | Eckert & Ziegler |
| ENRESA/El Cabril | Spain | • High-pressure compactor (super compactor)<br>• Facility for the treatment of effluents<br>• Volume reduction by compaction<br>• Incineration of organic liquids and small quantities of solids combustible<br>• Treatment/Conditioning of ashes<br>• Mortar system for both blocking and selling the concrete container having inside the packages from producers<br>• Radiochemical laboratory of samples from NPP<br>• Hot cell for testing real packages<br>• | No | ENRESA/<br>El Cabrilhttps://www.enresa.es/eng/index/activities-and-projects/el-cabril |
| GNS | Germany | • Vacuum drying facility PETRA for solid and/ or liquid radioactive waste<br>• Drying facilities for liquid waste (evaporator concentrate, sludge)<br>• Hydraulic super compactor FAKIR<br>• Powder-resin-transfer-facility PUSA<br>• Waste conditioning plant FAFNIR<br>• De-watering facility NEWA<br>• Cutting and packaging facility ZVA<br>• Underwater scrap shear UWS<br>• Drying facility KETRA | Yes | GNS |
| Hinneburg | Germany | • Metal scrap melting | Yes | Hinneburg |
| JAVYS-Bohunice Radioactive waste | Slovakia | • Solid radioactive waste segregation facility<br>• High-pressure press for reducing the volume of incombustible waste<br>• Incineration facility for solid and liquid combustible radioactive waste<br>• Liquid radioactive waste concentration facility<br>• Cementation facility for reinforcement and stabilization of concentrated waste | No | JAVYS |

| | | | | |
|---|---|---|---|---|
| Treatment Centre | | | | |
| NES | Austria | • High-pressure compactor with handling system (super compactor)<br>• Drum drying facility<br>• stainless steel processing chambers for breakdown, crushing, decontaminating, and processing radioactive waste<br>• Hot cell with underground storage for the treatment of HASS<br>• Source processing centre for treatment of DSRS<br>• Decontamination room with high pressure water jets<br>• Incineration plant<br>• Water purification including inlet pool, collection tanks, filtration systems for decontamination/purification, sludge filling station, storage pools for collection before discharge | No | NES |
| NRG, Petten | Netherlands | • Decontamination facilities | | NRG |
| Orano La Hague | France | • Reprocessing and recycling spent fuel | No | Orano |
| Siempelkamp – CARLA* | Germany | • Recycle contaminated metals from operation and decommissioning of nuclear facilities<br>• RWM, residual waste treatment and decontamination | No | Siempelkamp |

*Table 3 – Examples of treatment facilities identified by Member States*

*Siempelkamp GERTA melting services for mercury and/or NORM contaminated metals from oil- and gas industries, this unit is going step by step out of operation.

## 3.3 Characterisation techniques in SIMS

The characterization of radioactive waste concerns the radiological, chemical and physical properties. For these analyses, a set of high-tech equipment is used in LIMS to determine waste treatment, , conditioning and disposal requirements and to confirm WAC compliance for storage and disposal.

For SIMS participating in ROUTES, the characterization of waste focuses mainly on the radiological characteristics. Gauges for photons and neutron dose rate as well as contamination monitors are widely used. In addition, the destructive and non-destructive gamma spectrometry are utilized for radiological characterization of waste. The more sophisticated segmented γ-spectrometry is used only in Austria and in the Netherlands and gamma tomography in Austria. Sampling and radiochemical analyses in combination with alpha spectrometry, LSC, ICP-MS and X-ray spectroscopy are carried out in most SIMS. Nevertheless, the capabilities for analysis of radionuclides and materials are limited in each of the examined states.

In terms of chemical and physical properties, only Austria and the Netherlands perform characterization of the waste. The techniques used for chemical characterization of the waste are XRF, OES and photometer for elemental analysis; ion chromatography, analyser of carbon & sulphur in organic samples, FTIR spectrometer for organic compounds analyses. For study of physical phenomena, flash point measuring apparatus, combustion calorimeter, XRD, TGA are used. Press for testing stability and non-destructive strength testing are used for concrete testing. Other equipment for physical properties determination is stamping volumeter for determination of the bulk density and equipment for sieve analysis for particle size distribution.

It should be mentioned that these techniques which are used for chemical and physical characterization of the waste, are available in all SIMS for research purposes, which are mostly not connected to the radioactive waste research, as well as for food and water control and testing of concrete stability.

More detailed information regarding requirements for waste characterization in connection with WAC is provided in ROUTES deliverable 9.7 developed under Task 3 [28].

## 3.4 Availability of WAC in SIMS and LIMS

This chapter describes the current status of WAC in selected MS, with a focus on SIMS. The aim is to provide SIMS a deeper insight and comparison of their RWM approaches, commonalities in associated needs, and related WAC issues. It is based on the overview of the use of WAC in MS and some associated countries produced under ROUTES Task 4.

The results of the analysis presented in this chapter are based primarily on the national responses to the ROUTES questionnaire, the second part of which concerned WAC, and also on the MS88 report [29] prepared within ROUTES Task 4. National responses to the ROUTES questionnaire as well as MS88 report are "living" documents, i.e., they reflect an evolving situation where progress in waste management is continually being made. Consequently, the information has been reviewed and updated during the implementation of Task 5 to reflect more recent discussions with ROUTES partners.

These sources of inputs are complemented by publicly accessible information (e.g. from national reports etc.) and by outputs from Task 5 workshops [30]. The acquired information was compared with those reported under ROUTES Task 4, in particular the WAC needs in SIMS, and verified by involved EURAD ROUTES WP participants. Separate analyses are presented for SIMS (i.e. Austria, Cyprus, Denmark, Greece, The Netherlands, Poland, Portugal and Slovenia) and LIMS (i.e. Belgium, Bulgaria, The Czech Republic, France, Germany, Hungary, Lithuania, Romania, Slovakia, Spain, Sweden, Ukraine and the UK) in Sections 3.4.1 and 3.4.2 respectively, and a discussion of the differences in WAC availability is provided in Section 3.4.3.

This part of the report provides information about the availability of WAC at different stages of the waste lifecycle separately where relevant, i.e.:

- Retrieval and pre-treatment of raw waste (e.g. sorting, drying,…)
- Waste treatment (e.g. thermal treatment)
- Waste conditioning and packaging
- Interim storage
- Transport (may take place at multiple stages)
- Disposal

As already mentioned in Chapter 2 *Table 1*, different countries apply different approaches to waste classification, and therefore the national responses have been compared through application of the IAEA's system of radioactive waste classification, which defines radioactive waste classes based primarily on the combination of activity content and half-life of the radionuclides present in the waste, as is set out in the General Safety Guide No. GSG-1 [9]. This means that all low or medium active waste containing predominantly short-lived radionuclides is considered to be low-level waste (LLW), whilst low or medium active wastes that are predominantly composed of long-lived radionuclides are considered to be intermediate-level waste (ILW). Division of waste into classes according to this system is as below:

- Very low-level waste (VLLW)
- Low-level waste (LLW)
- Intermediate-level waste (ILW)
- High-level waste (HLW) (including spent / used nuclear fuel (SNF / UNF) if considered as waste)

Nevertheless, for the purpose of mapping the availability of WAC for individual waste classes at different stages of the waste lifecycle, the HLW and SNF / UNF classes are presented separately. One reason is the existence of different approaches to the definition of SNF/USF as waste. Furthermore, in some countries, the WAC for HLW and the acceptance criteria for SNF are defined separately, especially in countries where SNF is not defined as "waste" under national legislation/law. In such cases, the approach or system of SNF management may be different from the management of other types of radioactive waste, including the division of competencies and responsibilities between individual actors (producers, regulator, waste management organization etc.).
The use of different waste classification systems is one of the key factors hindering the development of common WAC and their implementation through uniform procedures across different countries.

As regards the use of terminology such as "generic WAC", "preliminary WAC" and "specific WAC" in this report (D9.10), the intention was to apply the same terminology used in MS88, recognising that different countries have different understanding of these terms and use them in various ways. The issue of different understandings of this terminology was discussed in detail within the MS88 report [29]. The D9.10 report does not address this issue in detail, but describes only individual national cases of the use of these terms.

### 3.4.1 Availability of WAC in SIMS

This subchapter summarizes information about availability of WAC in SIMS that have been surveyed (a total of eight SIMS). The information in the following *Table 4* is based on the findings given in MS88 (more specifically, in Table 1 of MS88) [29] and is extended by other information such as the use of international regulations and conventions concerning the transport of dangerous goods by road, sea or air in connection with transport of radioactive waste and derivation of WAC for transport. *Table 4* also distinguishes between two basic reasons for the absence of WAC for different classes of waste: the first is when a country has the relevant radioactive waste class but not WAC ("No" is used) and the second reason is that the classification is not used, or no waste of the class is in the country ("n/a" is used).

| **SIMS** | Stage of waste lifecycle / Waste class | *Retrieval and pre-treatment of raw waste (e.g. sorting, drying…)* | *Waste treatment (e.g. thermal treatment)* | *Waste conditioning and packaging* | *Interim storage* | *Transport* | *Disposal* |
|---|---|---|---|---|---|---|---|
| Austria | *VLLW* | n/a | | | | | |
| | *LLW* | Yes | | | Yes | ADR/RID/ADN/ICAO | No |
| | *ILW* | | | | | | |
| | *SNF / UNF* | n/a | | | | | |
| | *HLW* | n/a | | | | | |
| Cyprus | *VLLW* | No | | | Yes | *ADR/RID/ADN* | No* |
| | *LLW* | No | | | Yes | *ADR/RID/ADN* | No* |
| | *ILW* | n/a | | | | | |
| | *SNF / UNF* | n/a | | | | | |
| | *HLW* | n/a | | | | | |
| Denmark | *VLLW* | n/a | | | | | |
| | *LLW* | No | | | Yes | ADR/RID/ADN | No |
| | *ILW* | No | | | Yes | ADR/RID/ADN | No |
| | *SNF / UNF* | No | | | Yes | ADR/RID/ADN | No |
| | *HLW* | n/a | | | | | |
| Greece | *VLLW* | No | | | Yes | ADR | No |
| | *LLW* | No | | | Yes | ADR | No |
| | *ILW* | No | | | Yes | ADR | No |
| | *SNF / UNF* | n/a | | | | | |
| | *HLW* | n/a | | | | | |
| Netherlands | *VLLW* | n/a | | | | | |
| | *LLW* | No | Yes | Yes | Yes | ADR/RID/ADN/ICAO/IMDG | No |
| | *ILW* | No | Yes | Yes | Yes | ADR/RID/ADN/ICAO/IMDG | No |
| | *SNF / UNF* | No | | Yes | Yes | ADR/RID/ADN/ICAO/IMDG | No |
| | *HLW* | No | | Yes | Yes | ADR/RID/ADN/ICAO/IMDG | No |
| Poland | *VLLW* | n/a | | | | | n/a |
| | *LLW* | Yes | | | | ADR/RID/ADN/ICAO | Yes |
| | *ILW* | Yes | | | | ADR/RID/ADN/ICAO | Yes |
| | *SNF / UNF* | No | | | | ADR/RID/ADN/ICAO | No |
| | *HLW* | No | | | | ADR/RID/ADN/ICAO | No |
| Portugal | *VLLW* | No | | | | ADR/RID | No |
| | *LLW* | Yes | No | Yes | Yes | ADR/RID | No |
| | *ILW* | Yes | No | Yes | Yes | ADR/RID | No |
| | *SNF / UNF* | n/a | | | | | |
| | *HLW* | n/a | | | | | |
| Slovenia | *VLLW* | No | | | | ADR/RID/ADN | No |
| | *LLW* | No | | | Yes | *ADR/RID/ADN* | Yes |
| | *ILW* | | | | | | |
| | *SNF / UNF* | No | | | Yes | *ADR/RID/ADN* | No |
| | *HLW* | | | | | | |

*Table 4 – Availability of WAC at different stages of the waste lifecycle in SIMS surveyed*

Notes: n/a - not applicable (the classification is not used, no wastes of the class), No – WAC don't exist, Yes - WAC exist (for example a country doesn't have the class VLLW, we put n/a but when a country has VLLW but not WAC for VLLW, then we put No).

- The European Agreement Concerning the International Carriage of Dangerous Goods by Road (ADR) [31],
- The Regulation Concerning the International Carriage of Dangerous Goods by Rail (RID) [32],
- Annex C to the Convention Concerning the International Carriage by Rail (COTIF), the European Agreement concerning the International Carriage of Dangerous Goods by Inland Waterways (ADN) [33],
- The SOLAS Convention with the International Maritime Dangerous Goods (IMDG) Code [34],
- The International Civil Aviation Organization (ICAO) Technical Instructions on the Safe Transport of Dangerous Goods by Air [33]

ADR Member states: Albania, Andorra, Austria, Azerbaijan, Belgium, Bosnia and Herzegovina, Bulgaria, Croatia, Denmark, Estonia, Finland, France, Georgia, Germany, Greece, Iceland, Ireland, Italy, Kazakhstan, Latvia, Liechtenstein, Lithuania, Luxembourg, Macedonia, Malta, Montenegro, Morocco, Netherlands, Nigeria, Austria, Belarus, Norway, Poland, Portugal, Republic of Moldova, Romania, Russian Federation, San Marino, Serbia, Slovak Republic, Slovenia, Spain, Sweden, Switzerland, Tajikistan, Turkey, Tunisia, Czech Republic, Ukraine, Hungary, United Kingdom and Uzbekistan

* However, Cyprus has some acceptance criteria for "disposal" in the form of permitted releases to the environment, which are not normally considered as disposal.

Concerning transport: The transport of radioactive waste comprises all operations and conditions associated with and involved in their movement; these include the design, manufacture, maintenance and repair of packaging, and the preparation, consigning, loading, carriage including in-transit storage, unloading and receipt at the final destination of loads of radioactive waste and packages (see e.g. 1.7.1.3 ADR). In many countries, WAC for transport are not specified, however, maximum values of radionuclides (or their mixtures) in the transport container and other transport conditions (such as dose effective rate) are determined. The transport container must be type approved on the basis of the prescribed tests.

<u>Context notes and observations on RWM and WAC for each SIMS</u>

The country-specific notes below are based on the knowledge in MS88, which has been revised, updated, and further expanded, to reflect additional discussion with partners and status updates since the MS88 report was produced. For this reason, a certain overlap of these two documents (MS88 and D9.10) can be observed. Some of the information below can also be found in the document MS88 [29], however, it has been decided to allow a degree of duplication in order to maintain the required context and to ease the orientation of readers, without the need to study both documents simultaneously.

<u>Austria</u>

- The VLLW classification is not used.
- There is no HLW and SNF requiring long-term management – SNF from the RRs (ASTRA and TRIGA) is repatriated to the USA.
- **Specific WAC** for specific waste categories (e.g. solid non-combustible, solid combustible, liquids, DSRS) which define the RWM processes and specific facilities which are currently in operation – two sets of WAC:
    - Set of WAC concerns the raw radioactive waste to enter the Nuclear Engineering Seibersdorf (NES) – WAC for raw radioactive waste were established alongside a pricelist for the pre-treatment, treatment and conditioning services at NES (the WMO). They therefore include the requirements of the WMO, its site and its facilities. They also go through an approval process with the regulatory authority (the ministry of environment)
    - NES created and implemented internal WAC for interim storage. Take-over to interim storage (compliance to WAC double checked internally):
    - Since there is no solution for disposal conditioning should be carried out "as safe as necessary" but "as flexible as possible"; for example, 200L drum strategy, no grouting of high force compacted waste, sort of retrievability for DSRS ((e.g. 200 litre drums with high-pressure compacted pellets, homogenously cemented 200 litre drums, 200 litre drums with a container/capsule of DSRS cemented inside).

- For development of WAC comparable international regulations and documents were considered (interim storages, final disposal facilities).

The WAC for interim storage are included in the IMS (QM) system of NES (in German) and are an internal document.

- WAC regarding transport – the European Agreement Concerning the International Carriage of Dangerous Goods by Road (ADR), the Regulation Concerning the International Carriage of Dangerous Goods by Rail (RID), Annex C to the Convention Concerning the International Carriage by Rail (COTIF), the European Agreement concerning the International Carriage of Dangerous Goods by Inland Waterways (ADN), the SOLAS Convention with the International Maritime Dangerous Goods (IMDG) Code, the Convention on International Civil Aviation with its Annex 18 and the ICAO-Technical Instructions for the Safe Transport of Dangerous Goods by Air -- all these regulations are applicable for national and international transport of dangerous goods in Austria [35].

Cyprus

- **Generic WAC** for interim storage of VLLW and LLW. Cyprus follows the guidelines of IAEA [29] regarding the definition and classification of radioactive waste.
- No facilities that could treat, process, reprocess, condition etc. either spent fuel or radioactive waste exist in the country and currently, there is no centralised waste storage facility.
- The main origins of radioactive waste in the Cyprus are from activities in the field of medicine, industry, and research. All sources or other radioisotopes used in Cyprus are produced abroad.
- Radioactive waste is produced in low volumes in liquid or solid form, in facilities such as medical laboratories for nuclear medicine applications.
- NORM was produced in the past due to the activities of a now decommissioned fertiliser plant at Vasilikos area in the southern coast of Cyprus. NORM from decommissioning and phosphogypsum are kept at the site of the plant, properly stabilised and covered with plastic liner and soil, under the supervision and monitoring of RICS/DLI. [36]
- Disposal is in the form of authorised discharges to the environment - Radioactive waste (primarily composed of short-lived radionuclides from medical uses) is discharged as normal waste once it meets exemption levels provided for by law [36] . For DSRS, a condition imposed to the license holders is to return back to the supplier/manufacturer any DSRS. National policy regarding the safe management of radioactive waste foresees the establishment of WAC, if necessary.
- There is no disposal facility, none is currently planned.
- WAC regarding the transport - It also applies the relevant international standards for transport of radioactive materials, by road, sea or air, namely: the IAEA Safety Regulations for the Transport of Radioactive Materials, the United Nations Recommendations on the Transport of Dangerous Goods, ADR, IMDG ICAO etc. [31].

Denmark

- No WAC for VLLW - the VLLW classification is not used.
- HLW is in the form of spent fuel from RRs (no reprocessing is carried out).
- National policy is for storage until 2073 followed by disposal of all radioactive waste to a GDF.
- **Specific WAC** apply to specific storage facilities, and relates to the types of waste containers allowed in the specific facility:
    - There are 4 interim storage facilities operating in Denmark. Routing of wastes to these facilities distinguishes between waste from operations at Dansk Dekommissionering (Dekom) and external users, and decommissioning waste (which may be larger items).
    - WAC for these facilities differ in terms of container types accepted; levels of shielding; radionuclide inventory (particularly with respect to the fissile material content); and dose rates one metre from the surface of the waste package. One of the storage facilities use

selective emplacement of waste packages to manage units with a higher associated dose rate (by surrounding these packages with other waste units to provide shielding).

    - Different dose rates (at 1m distance of each waste container) apply to different storage facilities (concerning three of the facilities). For all storages apply: no free liquids are allowed in waste containers. Pressure build-up must be avoided.
    - The WAC do not describe the waste in terms of its origin, material content or classification.

- Dekom is responsible for waste from external users. External users are responsible for arrangements to transport wastes they have produced to Dekom. Transportation to storage facilities from the decommissioning sites at Risø is carried out according to agreement with the Danish authorities.

- The volume of radioactive waste requiring storage and disposal must be minimised, e.g. by incineration, melting and reuse, or when possible by clearance. Combustible waste is sent to an operator abroad for combustion (the Studsvik pyrolysis - steam reformation facility in Sweden) [37].

- WAC for transport - comply with the ADR, RID and ADN (e.g. Danish Acts on Sea Transportation and executive orders regarding the transportation of hazardous materials, the Danish Air Navigation Act and executive orders regarding the transportation of hazardous materials [38].

<u>Greece</u>

- There is no HLW in Greece. All SNF from the Greek RR has been returned to the USA.

- VLLW, LLW and ILW are stored separately at the NCSR “Demokritos” storage facility.

- **Specific WAC** for interim storage at the NCSR “Demokritos” of VLLW, LLW and ILW:

- All wastes are in their raw form. For storage, waste packages must satisfy WAC setting out dose rate limits (they are lower than the dose rate limits for transport). Some wastes are sent to the centralised storage at the NCSR “Demokritos”, but at present, most wastes are stored at the producer sites.

- Criteria for storage at the storage reception area of the NCSR “Demokritos” facility include limits on dose rates at the package surface (and at 0.7 m) and limits on the waste package mass, to enable handling using a forklift truck.

- Waste packages are placed in the storage reception area of the facility to optimise worker safety (e.g. by placing the waste packages with the lowest dose rates at the edges of an array of waste packages) [39].

- New WAC for the dose rate are foreseen for the future configuration of the drums in the new interim storage facility at the NCSR “Demokritos”.

- Disposal in near-surface facilities, surface trenches and/or boreholes is envisaged but no plans have been developed yet.

- WAC for transport (ADR) applies to all shipments of radioactive waste or materials within the country as well as to all shipments abroad.

<u>The Netherlands</u>

- The VLLW classification is not used in the Netherlands. It is used in the Dutch questionnaire response to set out WAC relating to NORM waste that is less than a factor of ten times above the clearance (exemption) level, which is suitable for landfill disposal. NORM wastes with higher levels of radioactivity are transferred to COVRA for storage pending geological disposal.

- There are various **specific WAC** for different types of waste and also for different types of processing steps. This also involves the packaging in which waste is packed.

- Specific WAC for waste treatment & conditioning - these have been developed for specific treatment installations in operation. Exceptions may apply. COVRA defines packaging and dose rate, sometimes specific activity (fissile material for example).

- There are also specific WAC for storage of LLW, ILW, HLW (conditioned and non-conditioned) and SNF. Waste treatment depends on the facility in which the waste will be stored and the given WAC. Not all waste stored is treated. The requirements placed on waste to be accepted for storage by COVRA are intended to enable direct transfer of waste packages to the DGR, once available, without further processing (such that all stored waste is ready for final disposal). Some resin waste streams are stored for short periods (~5 years) in packages that don't meet the dose rate WAC for the relevant storage facility. This allows time for COVRA to identify and deploy suitable conditioning solutions. In the meantime, "smart packaging and stacking" is employed to ensure safe storage. For example, some waste packages might have an extra concrete shielding package placed around them for a period of time, thereby enabling dose rate criteria to be met.
- Dose rate is the most important "generic" WAC. Every storage facility at COVRA has a dose rate limitation. Not all waste complies with this limit at first. Smart packaging (extra concrete shielding package, when needed) and stacking are applied to make sure the dose rate limits of a specific storage location are respected.
- Deep geological disposal is planned for all other types of radioactive waste in the Netherlands except exempt waste1 and very short-lived waste (VSLW).
- HLW includes vitrified HLW (from reprocessing of SNF from the Borssele and Dodewaard NPPs), SNF from RRs and waste from molybdenum production. SNF from Dutch NPPs is presently reprocessed in France and was previously done in the UK).
- There are no WAC regarding disposal (except for very short-lived waste), but in general it is considered that many WAC relevant to storage can also be applied to disposal.
- The term '**generic WAC**' is used in the Dutch response to refer to WAC that are not specific to a particular disposal facility/site or to acceptance of particular waste types. Rather, COVRA defines WAC for each of its storage facilities (e.g. dose rate limits) and these criteria are currently considered applicable as "generic WAC".
- WAC for transport - for transport (which COVRA does also) COVRA always comply with the ADR [31].

<u>Poland</u>

- The VLLW classification is not used in Poland.
- LLW and ILW is processed and stored by the RWM Plant (ZUOP) at Świerk before transfer to the National Radioactive Waste Disposal facility (KSOP) in Różan.
- Short lived waste is disposed of in surface/near-surface facilities (both terminologies are used in Poland);
- long-lived waste (mainly alpha-bearing waste) is stored pending availability of a deep GDF.
- Some nuclear materials (e.g. depleted uranium) are also stored at Różan.
- The only WAC developed in Poland are for disposal of the waste in State Radioactive Waste disposal facility in Różan (KSOP) which also concern packaging, preparation for disposal and disposal in the disposal facility. No WAC for HLW and SNF - Two RRs have operated in Poland: EWA (closed in 1995) and MARIA (in operation). Some SNF has been returned to Russia; other SNF is stored at Świerk, pending geological disposal. Some SNF elements have been 'encapsulated' and are stored in dedicated containers inside the shaft of the decommissioned EWA reactor [40] [41].
- In addition to a GDF, a new NSDF is planned to be constructed, to receive short-lived waste (LLW/ILW) once the KSOP at Różan is full. The new WAC will be elaborated for this facility.

[1] Exempt waste with concentrations of radionuclides small enough not to require provisions for radiation protection. Such material can be cleared from regulatory control and does not require any further consideration from a regulatory control perspective.

- There are WAC regarding transport of LL/ILW. The Polish provisions of law are based on the following international modal regulations: ADR, RID, ADN, IMDG, ICAO Technical Instructions and IATA DGR (International Air Transport Association – Dangerous Goods Regulation) [42].

Portugal

- No WAC for VLLW, however the VLLW classification exists in and is mentioned in the Portuguese National programme There is no HLW in Portugal; SNF from shutdown of the Portuguese RR (RPI) was transferred to the USA.
- There is one national RWM facility in Portugal named Pavilhão de Resíduos Radioativos (PRR) that exists for over 50 years as an interim facility. This facility was legally licenced in 2016, to receive LILW. The PRR is located at the Nuclear and Technological Campus (CTN) where the RPI is located, and it's a Pole of Instituto Superior Técnico (IST). IST (Instituto Superior Técnico) is also responsible for the collection, segregation, conditioning and storage of solid and liquid LLW and ILW produced in the country.
- Concerning medical applications in general, and nuclear medicine in particular, Decree-Law 180/2002 of August 8th establishes that solid and liquid radioactive waste with a very short half-life (VSLW) may be stored on site until it decays or is subject to authorized discharge.
- All the solid radioactive waste received from private and public entities from across the country is stored at the PRR elimination facility, after appropriate segregation and conditioning is carried out. Solid and liquid waste contaminated with H-3 and C-14 is also stored at the PRR
- The national legal framework ensures that, at all stages of RWM, individuals, society and the environment are adequately protected against radiological and other hazards but it doesn't include WAC yet.
- No WAC was yet established for the interim storage facility (PRR), however there are, procedures for the management of wastes (considering the national classification), radionuclide characteristics (solid, liquid, compressible, etc.), drums max activity, dose rate limits and surface contamination.
- The procedure mentioned before is implemented for medical and research waste producers (solid, liquids), lightning rods, smoke detectors and short lived DRSS that are delivered at the interim storage facility.
- WAC regarding transport – WAC criteria for the transport preparation of radioactive waste (LLW and ILW) follows the requirements of the European Agreement Concerning the International Carriage of Dangerous Goods by Road (ADR) and the Regulation Concerning the International Carriage of Dangerous Goods by Rail (RID). For example, maximum values of radionuclides (or their mixtures) in the transport container and other transport conditions (such as dose effective rate) are determined.

Slovenia

- The Regulation on RWM and Classification of Radioactive Waste takes into account, with some modifications, the radioactive waste categorisation system recommended in the "EC Recommendation on a Classification System for Solid Radioactive Waste" (OJ L 265, 13 October 1999, p. 37) [43] and [44]
- No WAC for VLLW - The VLLW classification is assigned to radioactive waste that may be cleared by the regulatory authority for release as exempt waste in future. A separate disposal route for VLLW is therefore not planned.
- WAC development for storage and disposal is a legal requirement defined in the Slovenian rules on radioactive waste and spent fuel management (JV 7, Off. Gaz.49/2006). The **adopted WAC** are therefore in use for all facilities in operation.
- Two storage facilities for both LLW and ILW are in operation: The Solid Radioactive Storage Facility (SRSF) at Krško NPP and CIS Brinje (for small producers).
- There are also SNF storage facilities at the TRIGA reactor and Krško NPP.

- Planed Vrbina LILW disposal is intended for short-lived LILW disposal. The LILW originates from the electrical power generation (at Krško NPP) and also from the use of radiation sources in medicine, industry and scientific research. Long lived waste will be moved to the LILW disposal facility location for storage, for planned LILW disposal facility the **WAC** are considered **to be preliminary** as they are not yet approved by Slovenian Nuclear Safety Administration (SNSA).
- The construction of a disposal facility for low- and intermediate-level radioactive waste (LILW) named Vrbina will start in 2021, with its trial operation planned in 2023. Two mines have also been used for the disposal of radioactive waste from uranium mining.
- Deep geological disposal is planned for long lived ILW, HLW and SNF
- WAC for transport - ADR, the Road Traffic Safety Act, and other legislative acts such as the Explosive Substances Act, the Chemicals Act, the Rules on the Approval of Packaging for the Transport of Dangerous Goods and the Rules on the Tasks of the Safety Adviser in Transport of Dangerous Goods. Such legal acts lay down requirements concerning the means of transport, equipment, personnel qualifications, package suitability, and labelling of packaging and transport means [43] [29].

<u>Summary of main findings regarding WAC availability and application in SIMS</u>

*Table 5* and *Figure 5* summarise how SIMS apply WAC for individual classes of waste for each stage of the waste lifecycle. As mentioned above, eight SIMS were surveyed:

- ✓ Austria (AT)
- ✓ Cyprus (CY)
- ✓ Denmark (DK)
- ✓ Greece (GR)
- ✓ Netherlands (NL)
- ✓ Poland (PL)
- ✓ Portugal (P)
- ✓ Slovenia (SI)

| Stage of waste lifecycle / Waste class | *Retrieval and pre-treatment of raw waste (e.g. sorting, drying,)* | *Waste treatment (e.g. thermal treatment)* | *Waste conditioning and packaging* | *Interim storage* | *Transport** | *Disposal* |
|---|---|---|---|---|---|---|
| | WAC availability in SIMS (8 SIMS surveyed) *(SIMS with WAC / SIMS with a given waste class/radioactive waste(SF))* | | | | | |
| *VLLW* | *(0/4)* | *(0/4)* | *(0/4)* | CY, GR *(2/4)* | CY, GR, P, SI *(4/4)* | *(0/4)* |
| *LLW* | AT, NL, PL *(3/8)* | AT, NL, PL *(3/8)* | AT, NL, P, PL *(4/8)* | AT, CY, DK, GR, NL, PL, P, SI *(8/8)* | AT, CY, DK, NL, GR, PL, P, SI *(8/8)* | P , SI *(2/8)* |
| *ILW* | AT, PL, P *(3/7)* | AT, NL, PL *(3/7)* | AT, NL, PL, P *(4/7)* | AT, DK, GR, NL, PL, P, SI *(7/7)* | AT, DK, GR, NL, PL, P, SI *(7/7)* | AT, PL, *(2/7)* |
| *SNF / UNF* | *(0/4)* | *(0/4)* | NL *(1/4)* | DK, NL, SI *(3/4)* | DK, NL, PL, SI *(4/4)* | *(0/4)* |
| *HLW* | *(0/3)* | *(0/3)* | NL *(1/3)* | NL, SI *(2/3)* | NL, PL, SI *(3/3)* | *(0/3)* |

*Table 5 – WAC availability and utilisation in SIMS for different stages of the waste lifecycle*

**SIMS* using ADR and IAEA standards as "generic" WAC for transport of radioactive waste/SNF

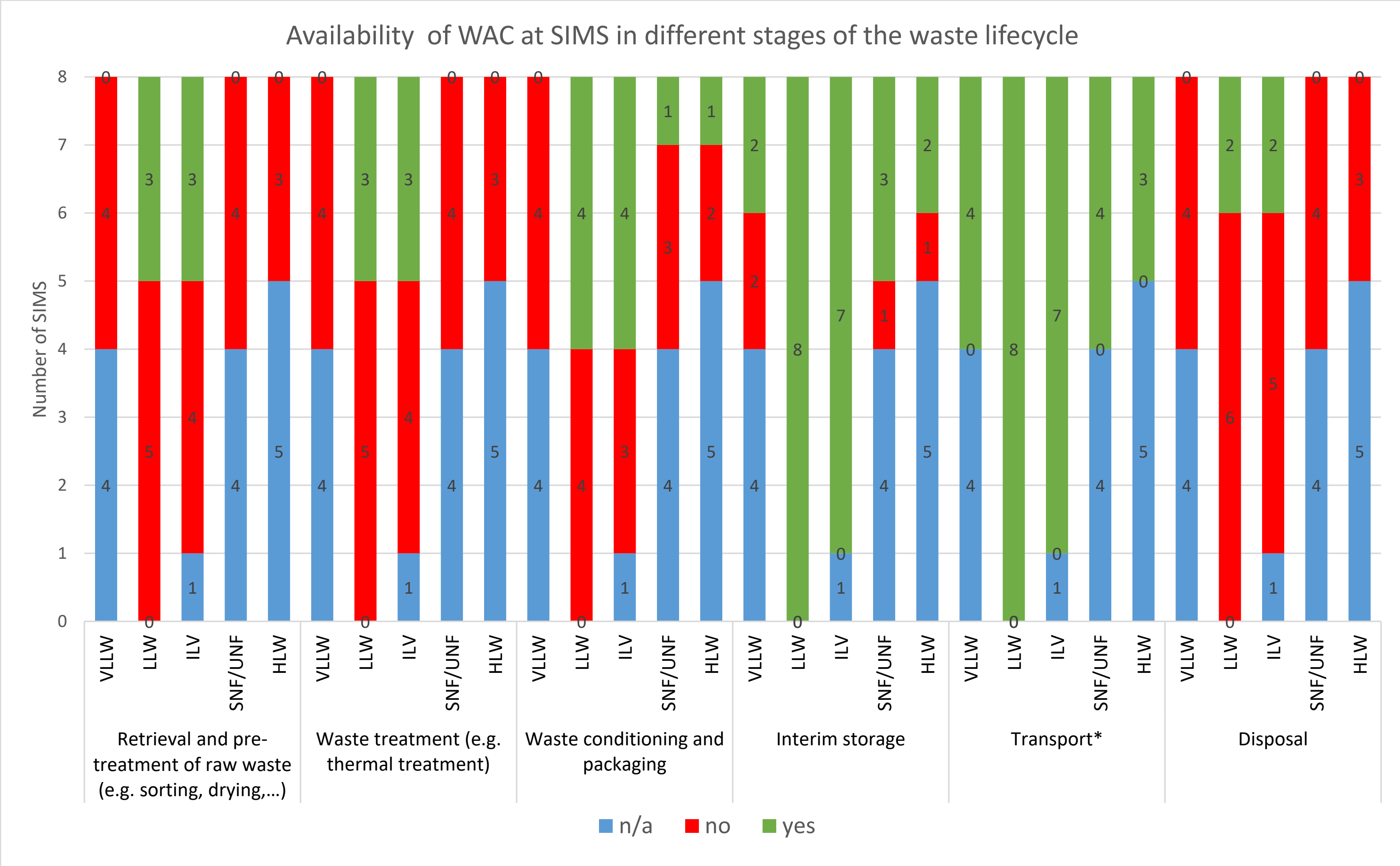



*Figure 5 –* Availability *of WAC in SIMS in different stages of the waste lifecycle for various waste classes (A total of eight SIMS countries were surveyed through ROUTES questionnaire).*

*Notes: n/a - not applicable* (classification is not used, no wastes of the class), *No – WAC don't exist, Yes - WAC exist; *SIMS* using ADR and IAEA standards as "generic" WAC for transport of radioactive waste/SNF

- The reasons for the absence of WAC for some waste classes or for certain stages of their life-cycle are described in detail in the MS 88 report [29], e.g. the executive summary of this report provides an overview of these reasons. Briefly, these reasons include:
  - RWM programmes are still at an early stage.
  - WAC are not relevant for the lifecycle stage in question for certain waste classifications the given waste class is not defined within the national radioactive waste programme or through national legislation (this often applies to VLLW), the life cycle of waste is not at a given stage ((this often applies to the disposal stage).
  - WAC are not defined for facilities that accept all wastes regardless of their characteristics. This reflects the implicit nature of WAC as facilitating the ability to also reject certain wastes or waste packages.
  - WAC are not always reported for management activities conducted outside of the country responding to the ROUTES questionnaire – e.g. Austria, Greece and Portugal repatriation of SNF from the RRs to the USA; Danish combustible waste is sent for combustion to the Sweden (the Studsvik pyrolysis); SNF from Dutch NPPs is reprocessed in France; Poland - some SNF has been returned to Russia.
  - WAC are not always defined for lifecycle stages where there is no transfer in the ownership or liability for the waste.
- Only four countries out of eight have the waste classified as VLLW and two of them have WAC for interim storage. None of these countries has WAC for the disposal of VLLW.
- In the case of SIMS, the WAC are most often applied in the field of low-level and intermediate-level waste management, especially in the stage of interim storage – most countries have a specific WAC derived for a given storage facility. Responses to the questionnaire showed that WAC for interim storage of LLW are implemented in all eight surveyed member states with small inventories, a similar situation is also true for of ILW, where the seven countries that have ILW also have relevant storage acceptance criteria.
- There are four countries out of eight, which have SNF; one of them has WAC for conditioning and packaging, whilst three have WAC for SNF interim storage.
- Only three countries out of eight have waste classified as HLW; one of them has WAC for conditioning and packaging, whilst two have WAC for interim storage.
- (WAC for interim storage of HLW and SNF are derived and in use in the Netherlands (HLW+SNF) Slovenia (HLW and SNF) and Denmark (SNF)).
- WAC for transport – for transport between nuclear licensed sites and different users, most countries apply IAEA transport regulations and international agreements concerning the carriage of dangerous goods as WAC for transport, such as e.g. IAEA's transport regulations [45] and agreements developed under the auspices of the United Nations concerning the carriage of dangerous goods (there are separate documents for carriage by road (the 'ADR') [31], Convention concerning International Carriage by Rail (the 'RID') [32] etc. These regulations are typically reflected and implemented in the national legislative framework. However, sometimes they are applied directly without such implementation in national legislation or such promulgation. Transport between facilities on the same site is often not subject to formalised specific WAC, although there may still be export and/or receipt requirements at the original and destination facilities respectively.
- In general, most of the countries surveyed with a small inventory do not have a system of acceptance criteria for the final stage of the waste lifecycle – disposal. Most countries do not have any disposal WAC because planning facilities for disposal is still at an early stage (especially in the case of geological disposal).
- In most SIMS surveyed, specific WAC are used / applied, which are derived for different types of waste and also for different types of processing steps and facilities.

### 3.4.2 Availability of WAC in LIMS

This subchapter summarizes information about the availability of WAC in LIMS that have been surveyed (a total of thirteen LIMS). The information in the following *Table 6* is again based on the findings given in MS88 (in Table 1 of MS88) [29] and is extended by other information such as the use of international regulations and conventions concerning the transport of dangerous goods in connection with transport of radioactive waste and derivation of WAC for transport. Furthermore, *Table 6* again distinguishes between two basic reasons for the absence of WAC for different classes of waste: the first is when a country has the relevant radioactive waste class but not WAC ("No" is used) and the second reason is that the classification is not used, or no waste of the class is in the country ("n/a" is used).

| **LIMS** | Stage of waste Lifecycle / Waste class | *Retrieval and pre-treatment of raw waste (e.g. sorting, drying,…)* | *Waste treatment (e.g. thermal treatment)* | *Waste conditioning and packaging* | *Interim storage* | *Transport* | *Disposal* |
|---|---|---|---|---|---|---|---|
| Belgium | *VLLW* | n/a | | | | | |
| | *LLW* | Yes | Yes | Yes | Yes | ADR/RID/ADN | Yes (draft) |
| | *ILW* | Yes | Yes | Yes | Yes | ADR/RID/ADN | No |
| | *SNF / UNF* | No | No | No | No | ADR/RID/ADN | No |
| | *HLW* | No | No | Yes | Yes | ADR/RID/ADN | No |
| Bulgaria | *VLLW* | Yes | No | Yes | Yes | ADR/RID/AND/ICAO | Yes |
| | *LLW* | Yes | Yes | Yes | No | ADR/RID/AND/ICAO | Yes |
| | *ILW* | Yes | Yes | Yes | No | ADR/RID/AND/ICAO | Yes |
| | *SNF / UNF* | No | No | No | No | ADR/RID/AND/ICAO | No |
| | *HLW* | Yes | No | Yes | No | ADR/RID/AND/ICAO | No |
| Czech Republic | *VLLW* | Yes | Yes | Yes | Yes | ADR/RID/AND/ICAO/ IMDG | Yes |
| | *LLW* | | | | | | |
| | *ILW* | | | | | | |
| | *SNF / UNF* | No | No | Yes | Yes | ADR, RID, ADN, ICAO IMDG | No |
| | *HLW* | No | | | | ADR, RID, ADN, ICAO IMDG | No |
| France | *VLLW* | No | Yes | *Yes* | Yes | ADR, RID, ADN, ICAO IMDG | Yes |
| | *LLW* | No | Yes | *Yes* | Yes | ADR, RID, ADN, ICAO IMDG | Yes |
| | *ILV* | No | Yes | *Yes* | Yes | ADR, RID, ADN, ICAO IMDG | Yes |
| | *SNF/UNF* | No | | | | ADR, RID, ADN, ICAO IMDG | No |
| | *HLW* | No | Yes | Yes | Yes | ADR, RID, ADN, ICAO IMDG | Yes |
| Germany | *VLLW* | n/a | | | | | |
| | *LLW* | No | Yes | Yes | Yes | ADR/RID/ADN | Yes |
| | *ILW* | | | | | | |
| | *SNF / UNF* | No | No | No | No | ADR/RID/ADN | No |
| | *HLW* | Yes | No | Yes | Yes | ADR/RID/ADN | No |
| Hungary | *VLLW* | No | | | | ADR/RID/ADN/ICAO | No |
| | *LLW* | No | Yes | No | Yes | ADR/RID/ADN/ICAO | Yes |
| | *ILW* | No | Yes | No | Yes | ADR/RID/ADN/ICAO | Yes |
| | *SNF / UNF* | No | | | | ADR/RID/ADN/ICAO | No |
| | *HLW* | No | | | | ADR/RID/ADN/ICAO | No |
| Lithuania | *VLLW* | Yes | Yes | Yes | Yes | ADR/RID/ICAO/IMDG | Yes |

| **LIMS** | Stage of waste Lifecycle / Waste class | *Retrieval and pre-treatment of raw waste (e.g. sorting, drying,…)* | *Waste treatment (e.g. thermal treatment)* | *Waste conditioning and packaging* | *Interim storage* | *Transport* | *Disposal* |
|---|---|---|---|---|---|---|---|
| | *LLW* | Yes | Yes | Yes | Yes | ADR/RID/ICAO/IMDG | Yes |
| | *ILW* | Yes | Yes | Yes | Yes | ADR/RID/ICAO/IMDG | Yes |
| | *SNF / UNF* | No | No | Yes | Yes | ADR/RID/ICAO/IMDG | No |
| | *HLW* | No | | | | ADR/RID/ICAO/IMDG | No |
| Romania | *VLLW* | Yes | Yes | Yes | Yes | ADR/RID/ADN/ICAO | Yes |
| | *LLW* | | | | | | |
| | *ILW* | No | | | Yes | ADR/RID/ADN/ICAO | |
| | *SNF / UNF* | No | | | | ADR/RID/ADN/ICAO | No |
| | *HLW* | No | | | | ADR/RID/ADN/ICAO | No |
| Slovakia | *VLLW* | Yes | No | Yes | Yes | ADR/RID | Yes |
| | *LLW* | Yes | Yes | Yes | Yes | ADR/RID | Yes |
| | *ILW* | Yes | Yes | No | Yes | ADR/RID | No |
| | *SNF / UNF* | No | | | Yes | ADR/RID | No |
| | *HLW* | No | | | | ADR/RID | No |
| Spain | *VLLW* | Yes | Yes | Yes | No | ADR/RID/ICAO/IMDG | Yes |
| | *LLW* | Yes | Yes | Yes | Yes | ADR/RID/ICAO/IMDG | Yes |
| | *ILW* | Yes | Yes | Yes | Yes | ADR/RID/ICAO/IMDG | Yes |
| | *SNF / UNF* | No | No | Yes | No | ADR/RID/ICAO/IMDG | No |
| | *HLW* | No | No | No | No | ADR/RID/ICAO/IMDG | No |
| Sweden | *VLLW* | Yes | | | | ADR/RID/ICAO/IMDG | Yes |
| | *LLW* | Yes | Yes | Yes | Yes | ADR/RID/ICAO/IMDG | Yes |
| | *ILW* | Yes | Yes | Yes | Yes | ADR/RID/ICAO/IMDG | No |
| | *SNF / UNF* | No | | | | ADR/RID/ICAO/IMDG | Yes |
| | *HLW* | n/a | | | | | |
| Ukraine | *VLLW* | No | | | | *ADR/RID/ICAO/IMO* | *No* |
| | *LLW* | Yes | Yes | Yes | *Yes* | *ADR/RID/ICAO/IMO* | *Yes* |
| | *ILW* | Yes | Yes | Yes | *Yes* | *ADR/RID/ICAO/IMO* | *Yes* |
| | *SNF / UNF* | No | | | | *ADR/RID/ICAO/IMO* | No |
| | *HLW* | No | Yes | Yes | *Yes* | *ADR/RID/ICAO/IMO* | No |
| United Kingdom | *VLLW* | No | | Yes | No | ADR/RID/AND/ICAO/IMDG | Yes |
| | *LLW* | No | Yes | Yes | No | ADR/RID/AND/ICAO/IMDG | Yes |
| | *ILW* | No | Yes | No | Yes | ADR/RID/AND/ICAO/IMDG | No |
| | *SNF/UNF* | No | | | | ADR/RID/AND/ICAO/IMDG | No |
| | *HLW* | No | | | Yes | ADR/RID/AND/ICAO/IMDG | No |

*Table 6 – Availability of WAC in different stages of the waste lifecycle in LIMS (large inventory member state) surveyed*

*Notes: n/a - not applicable (the classification is not used, no wastes of the class), No – WAC don't exist, Yes - WAC exist (for example a country doesn't have the class VLLW, we put n/a but when a country has VLLW but not WAC for VLLW, then we put No).*

- The European Agreement Concerning the International Carriage of Dangerous Goods by Road (ADR),

- The Regulation Concerning the International Carriage of Dangerous Goods by Rail (RID),
- Annex C to the Convention Concerning the International Carriage by Rail (COTIF),The European Agreement concerning the International Carriage of Dangerous Goods by Inland Waterways (ADN),
- The SOLAS Convention with the International Maritime Dangerous Goods (IMDG) Code,
- The International Civil Aviation Organization (ICAO) Technical Instructions on the Safe Transport of Dangerous Goods by Air.

*ADR Member states: Albania, Andorra, Austria, Azerbaijan, Belgium, Bosnia and Herzegovina, Bulgaria, Croatia, Denmark, Estonia, Finland, France, Georgia, Germany, Greece, Iceland, Ireland, Italy, Kazakhstan, Latvia, Liechtenstein, Lithuania, Luxembourg, Macedonia, Malta, Montenegro, Morocco, Netherlands, Nigeria, Austria, Belarus, Norway, Poland, Portugal, Republic of Moldova, Romania, Russian Federation, San Marino, Serbia, Slovak Republic, Slovenia, Spain, Sweden, Switzerland, Tajikistan, Turkey, Tunisia, Czech Republic, Ukraine, Hungary, United Kingdom and Uzbekistan.*

Context notes and observations on WAC for each LIMS

The country-specific notes below are again based on the knowledge in MS88, which has been revised, updated, and further expanded, to reflect additional discussion with partners and status updates since the MS88 report was produced. LIMS are covered here in order to provide a basis for comparison with the status of WAC use in SIMS, as presented in Section 3.4.3.

Belgium

- The VLLW classification is not used.
- The **WAC for disposal** of LLW in a surface disposal facility is preliminary but takes account of the specific surface disposal facility planned in Dessel.
- The **WAC for disposal** of LLW currently only consider conditioning via cementation[2] - WAC are based on homogeneous or heterogeneous cementation. All WAC based on an organic matrix (bitumen or polymer) have been removed a few years ago. The compatibility with surface disposal of certain families of waste in an organic matrix will be demonstrated at a later stage.
- The **specific WAC** for non-conditioned waste and conditioned waste are adopted and in use for pre-disposal activities:
    - WAC for non-conditioned LLW and ILW for retrieval and pre-treatment, treatment and interim storage.
    - WAC for conditioned LLW and ILW for retrieval and re-conditioning, packaging and interim storage.

  However, a thorough revision of these WAC is currently being prepared. This revision takes account of the safety report of the planned Dessel surface disposal facility.
- WAC for conditioning, packaging and storage of HLW are specific to vitrified HLW produced in La Hague (CSD-V). These WAC have only been used in the context of a specific reprocessing contract. The WAC have not been used anymore since more than a decade ago, because reprocessing of SNF from Belgian NPPs is suspended.
- There are no WAC for SNF
- WAC regarding transport - comply with the ADR/RID/ADN. Generally, regulations follow the technical requirements of IAEA Regulations for Safe Transport of Radioactive Substances but for certain of shipment, the transport license may specify special condition, such as provision of

[2] The composition of the conditioning mortar is not directly prescribed in the WAC. The mortar only has to meet certain acceptance criteria. However, the composition of the conditioning mortar IS prescribed by the qualification of the conditioning process. This qualification requires extensive testing of the mortar that will be used.

escorts. All shipments, by whatever means, must be authorized in advance by the Federal Agency for Nuclear Control under the supervision of the federal Minister of the Interior [46].

Bulgaria

- **Specific WAC** are in use for all types of radioactive waste and are specific to the facilities in operation.
- SNF is not classified as radioactive waste in Bulgaria.
- There are WAC for VLLW, LLW, ILW
- The acceptance criteria at SE radioactive waste are established according to technical, mechanical, chemical, physical, radiological, biological and special organizational requirements which ensure that the received radioactive waste are in full compliance with the specific legislation and general approach for RWM in Bulgaria.
- Practically the establishment of WAC system in Bulgaria is regard to pre-treatment, treatment and conditioning, and storage, but still is under further development. The approach to this is intended for all kind of facilities with applicability to various waste classifications and reflecting the widespread nature of RWM practices and the value of WAC to be develop as a management tool overall applicable in the country.
- For the existing and designed facilities there are defined specific WAC. These are developed in the design of the relevant facility and are included also in the safety case. WAC are in use for all types of radioactive waste and these are specific for each facility in operation.
- For the planned facilities, the WAC as **preliminary WAC** are included in the relevant design and safety case. Two facilities are planned Two facilities are planned:
  - The National Disposal Facility for short-lived low and medium active waste (i.e. LLW, according to IAEA waste classifications), which is under construction,
  - Facility for interim long-term storage of HLW and long-lived low and medium active waste (i.e. ILW, according to IAEA waste classifications).
- **Preliminary WAC** for the planned (geological disposal for high-and long-lived wastes) facility are included in the relevant design and safety case.
- WAC regarding transport - the safety requirements for radioactive waste and SF transport are defined in the Regulation on the Terms and Conditions for Transport of Radioactive Material, which was developed in accordance with the requirements of IAEA document "Regulations for the Safe Transport of Radioactive Materials" TS-R-1, as well as with the requirements of the relevant international regulations for the transport of dangerous goods ADR/RID/AND/ICAO etc. [47].

Czech Republic

- Only facility **specific WAC** have been developed and are in use:
  - WAC for LLW disposed of in Dukovany site
  - WAC for LLW and ILW disposed of in Bratrství site
  - WAC for LLW and ILW disposed of in Richard site
  - WAC for LLW and ILW stored in Richard site
  - WAC for all categories of waste for storage (i.e. SNF interim storage), for installations of waste generators

Relevant WAC are derived from safety analysis under the main requirement not to exceed the dose limit for the representative person.

- Existing facilities are operated according to approved OLCs (Operational Limits and Conditions) including WAC, which do not distinguish radioactive waste classes. All radioactive waste must meet the given acceptance criteria in order to ensure safe management. In connection with radioactive waste disposal into existing disposal facilities, it has to comply the disposal WAC specific for the sites / disposal facility.

- WAC for deep GDF (ILW, HLW) were not derived yet
- WAC are a part of operational documentation of each licensee.
- Facilities carry out activities through all stages of RWM and WAC are used for each process.
- There are WAC for transport for all waste classes and categories. They comply with the requirements of ADR, RID, ADN, ICAO IMDG etc. [48].

France

- WAC for conditioning and storage facilities are in use - they are defined on a case-by case basis for each facility and each waste package in order to meet safety and technical requirements - Each treatment and conditioning facility is designed and operated in order to be compliant with safety standards. Conditioned LILW-LL and HLW are temporarily stored in storage facilities usually located close to the production site, pending disposal in the appropriate facility.
- WAC for conditioning and packaging of VLLW and LLW - there are specific guide for small producers
- WAC for disposal facilities:
  - **CSA site** – both **generic** and **specific WAC are** adopted and used: generic - Basic Safety Rules at national level, for the management of LLW emitted by French regulator; specific national license for the CSA Site and ANDRA specific WAC
  - **CIRES** – **specific WAC** for VLLW disposal are adopted and used - local decree to authorize construction of the site and ANDRA specific WAC
  - **Cigéo: WAC** for deep geological disposal **are still preliminary,** nevertheless precise guidelines are already available. Preliminary WAC for geological disposal have been developed, adapted and specified over the years on the basis of the results of the R&D work carried out in the framework of the Law 2006-739.
- The Industrial Centre for Geological Disposal (Cigeo) will serve for disposal of long-lived HLW waste produced by France's current fleet of nuclear facilities, until they are dismantled, as well as from reprocessing of spent fuel from nuclear power plants. Waste to be disposed of in the future Cigéo facility are conditioned under the general requirements of the “Safety guide on the permanent disposal of radioactive waste in deep geological repositories”. The safety guide defines general requirements which are being transposed in Specific WAC - still preliminary.
- In France, spent nuclear fuel is not considered as ultimate waste and is reprocessed for recovery of reusable materials. Commercial reprocessing is carried out at the La Hague plant operated by AREVA.
- WAC for transport - the transport of radioactive waste is governed by IAEA Standards and comply the requirements of ADR, RID, ADN, ICAO, IMDG etc. [49].

Germany

- No WAC for VLLW - the VLLW classification is effectively irrelevant – disposal to landfill is not carried out or planned.
- WAC are individual for every facility (storage, treatment or disposal).
- In Germany, disposal in deep geological formations is intended for all types of radioactive waste. Accordingly, there is no need to differentiate between wastes containing radionuclides with comparatively short half-lives and those with comparatively long half-lives. Radioactive wastes are therefore subdivided based on whether or not they generate significant quantities of heat.
- At the moment there is only one final disposal site, for which WAC were formulated - WAC for LLW and ILW are reported for routing to disposal at Schacht Konrad (Asse and Morsleben are no longer operating).

- Final disposal of wastes with negligible heat generation (i.e. LLW and ILW) is planned in Schacht Konrad, a former iron ore mine at Salzgitter (Lower Saxony). The Konrad disposal facility is expected to become operational in 2027.
- Planning for retrievals from Asse is currently based on the WAC for Schacht Konrad since these represent the state of the art for conditioning and packaging of radioactive waste.
- Disposal in near-surface facilities, surface trenches and/or boreholes is envisaged but no plans have been developed as yet.
- In Germany, all the radioactive waste, including HLW and SNF, which does not fulfil the WAC for the KONRAD disposal facility are left in above ground interim storage facilities at various sites. Waste categories for which a disposal route was not decided yet were identified as the radioactive waste retrieved from ASSE salt mine facility, the heat generative waste, historical/problematic waste, most of irradiated carbonaceous materials, pebble bed fuel and depleted uranium [50].
- There are WAC for transport (LLW, ILW and HLW) based on IAEA Standards. Transboundary movement within the EU is regulated by Council Regulation No. 1493/93/EURATOM. 0 The WAC for transport comply with the requirements of ADR/RID/AND [29] [51].

<u>Hungary</u>

- No WAC for VLLW - new waste class since 2018. Introducing this new class of waste (VLLW) produce some benefits, it gives possibility to dispose it in the NSDF. A planned NSDF that guarantees the safe disposal of short-lived, VLLW is a magnitude cheaper than the more sophisticated disposal facilities with the engineering barriers for LILW. Additionally, site selection and licensing procedure might be much easier as well.
- LILW disposal facilities:
- Radioactive Waste Treatment and Disposal Facility (RWTDF), Püspökszilágy - surface disposal facility for institutional LILW
- National Radioactive Waste Repository (NRWR), Bátaapáti - geological facility (granite host rock) for LLW and LILW-SL from the operation and decommissioning of the Hungarian nuclear power plant (Paks NPP)
- **Specific WAC for RWTDF** – a set of criteria was derived for final disposal in the early 2000s (activity content, leach ability limit of the waste form volume of void etc.) At the beginning of the operation of the RWTDF there was no WAC. To improve the safety, PURAM started a safety upgrade program whose goal was to retrieve segregate, condition and redispose legacy waste using appropriate WAC [52]. In 2016 the WAC for RWTDF became a multi-level system: There are separate WAC sets in the RWTDF for receiving waste, storage and disposal. The bulk of the waste producer does not have any waste conditioning capacity, so the waste received from these producers cannot be in compliance with the criteria for disposal. The waste received from these institutions is going to be treated and conditioned in the RWTDF appropriately. For sets of WAC were developed in case of RWTDF:
    - WAC for long term disposal – Complying with this criterion enables the immediate disposal in the disposal vaults. This criterion contains the most rigorous limits for activity content, physical-chemical properties and packaging.
    - WAC for mid-term disposal – The only alteration from the previous WAC is the activity content. This set of criteria is applicable for the waste that require some amount of time in the interim storage facility before complying with the WAC for long term disposal.
    - WAC for interim storage – This set of criteria is more permissive regarding the activity content, physical – chemical forms and packaging. This criterion is suitable for the waste that can be stored on the site only it can be because of the content of isotopes with long half-life.
    - WAC for takeover – Most permissive criteria suitable for those producers that cannot fulfil the stricter criteria. In this case the treatment and conditioning of the radioactive waste relies on the RWTDF.

- **Specific WAC for NRWR** - In the mid 2012 application for operational license of the first chamber was submitted. According to the schedule, in the beginning of 2013 the first concrete containers are going to be disposed there. According to the schedule, the first concrete containers were stored there at the beginning of 2013. Final disposal began with the oldest, so called "historical" waste. This kind of waste was generated before the WAC, and its application to the NPP: waste form specification entered into force [52].
- The disposal routes for LLW and ILW are determined by the origin of the waste (either from institutional use or from NPPs) and do not vary depending on whether the waste contains predominantly short-lived / long-lived radionuclides, except that long-lived institutional waste is interim stored at the RWTDF, pending final disposal in a deep GDF.
- No WAC for HLW and SNF
- Deep GDF is planned for HLW (including SNF and waste with a high heat output)
- WAC for transport - The Hungarian regulation on transport of radioactive material is based on the relevant international conventions (such as such as ADR, RID, ADN and ICAO). The same general rules should be applied to the transport of radioactive waste as to radioactive material. In addition, the transport of radioactive waste across national borders should take place in accordance with the European Union Directive on transboundary shipment of radioactive waste. The Hungarian Atomic Energy Authority is the competent authority for licensing of transport packages and transport arrangements [53].

<u>Lithuania</u>

- According to the Lithuanian Law on Nuclear Energy, spent fuel is classified as radioactive waste. HLW is in the form of spent fuel (no reprocessing is carried out) [54]
- Storage facilities [55]
  - Two dry type spent nuclear fuel storage facilities (the first facility in operation since 1999, the second one – since 2017)
  - Storage facility for cemented liquid waste (in operation since June 2020)
  - Solid radioactive waste processing and storage facilities ("hot trials" of facilities have started in October 2017 and it is planned start of the industrial operation at the end of 2021).
  - Very low-level radioactive waste buffer storage facility (in commercial operation since 2013

  There are WAC for storage derived for each of the above storage facilities.
- Long-lived radioactive waste, which remains dangerous for thousands of years, is currently stored at the Ignalina plant's interim storage facilities
- Planned disposal facilities [55]:
  - There is one storage facility (bitumized LRW storage facility) which is under consideration for transformation to disposal facility. It means that radioactive waste already are in place, no change in volume or activities is foreseen, but legal status of these wastes now storage facility.
  - **Landfill** disposal facility for VLLW start of operation is planned for 2022
  - **NSR** for LILW SL start of operation is planned for 2024
  - **DGR** for HLW and LILW-LL - start of operation is planned for 2066
- Preliminary WAC have been developed for planned specific facilities in the specific site as follows: for Landfill facility for VLLW and for NSDF (NSR - near surface repository) for short-lived LLW and ILW:
  - Preliminary WAC for VLLW disposal in the Landfill facility were established in 2012.
  - Preliminary WAC for short-lived LLW and ILW disposal were established in 2017.
- These **specific WAC** for disposal are derived for specific disposal facility which is relevant to certain radioactive waste activity content in accordance to the national waste classification

scheme taking into account radionuclide inventory (operational and post-closure safety) as well as nuclear criticality.

- **Preliminary WAC** for disposal of VLLW and sort lived LLW have been derived based on the Technical Design (TD) and Preliminary Safety Assessment Report (PSAR) developed for planned disposal facilities and agreed with the national regulator (VATESI). Once construction of these facilities is completed, the Final Safety Assessment Report (FSAR) will be prepared and associated Final WAC will be derived.
- Specific WAC exist for the interim storage of spent fuel and long lived ILW, but not yet for disposal since there is no GDF [39].
- WAC for transport - requirements for transport described in national legislation ( [56]) are based on the international transport regulations given by IAEA. They comply with ADR, RID, ICAO and IMDG [57].

<u>Romania</u>

- VLLW exists in Romania, but its management is not currently segregated from that of LLW.
- National Repository Baita Bihor (DNDR) – It is NSDF for institutional radioactive waste and takes place in galleries at a depth of ~50 m below ground level (the elevation of the former uranium mine is 840 m above sea level). According to the WAC institutional LILW–SL and limited activities of LILW-LL are disposed at Baita Bihor disposal facility.
- Planned disposal facilities:
  - a NSDF - for LILW-SL generated from Cernavoda NPP operation, refurbishment and decommissioning, but also for the institutional LILW-SL after Baita Bihor disposal facility closure (in ~2040)
  - a deep GDF for LILW-LL and CANDU SNF, foreseen to be in operation in 2055
- Operational LLW generated at the Cernavoda NPP is treated by incineration or melting at a European operator.
- In the meantime, ILW is stored on site. No requirements for the treatment or conditioning of long-lived waste have been defined as yet, so there are no associated WAC.
- The HLW refers mainly to the spent fuel that in Romania is considered radioactive waste (CANDU SNF from the Cernavoda NPP).
- No WAC for SNF - SNF from RRs is stored in the reactor storage pool (for TRIGA SNF) and then returned to the country of origin (Russia / USA). SNF from NPPs is stored in dedicated on-site facilities (in ponds for the first six years and then in dry storage). There are no WAC for these facilities because all of the associated SNF is accepted.
- Existing **Specific WAC** developed for specific facilities:
  - WAC for the predisposal activities on institutional radioactive waste - there are requirements for waste acceptance developed for the two research institutes involved in institutional RWM: IFIN-HH Bucharest and RATEN ICN Pitesti;
  - the WAC for institutional radioactive waste have been developed for National Repository Baita Bihor;
  - for the low-level waste generated by Cernavoda NPP operation that are treated by incineration or by melting at international operators, specific WAC are imposed by the treatment operators;
- WAC adopted and in use for the operational radioactive waste generated at Cernavoda NPP, treated by incineration or melting at a European operator. The operational radioactive waste generated by Cernavoda NPP are stored in Solid Radioactive Waste Interim Storage Facility – DIDR that consists of three above ground concrete structures. Compactable and non-compactable solid radioactive waste are stored in a concrete warehouse structure, the spent filter cartridges are stored in a concrete cylindrical structure and the spent ion exchange resins

are stored in concrete vaults lined with epoxy, segregated as non-fuel contact resins and fuel-contact resins.

- The **preliminary WAC** for LILW-SL were developed for the future near-surface disposal facility, that is proposed to be located in the exclusion zone of Cernavoda NPP (Saligny is the preferred site);
- There are no specific WAC for transport in Romania; however, there are specific norms for transport of radioactive materials set out in national legislation. Romania has been ratified and implemented the provisions of international agreements and conventions regarding the transport of dangerous goods such as ADR, AND, RID, ICAO.

<u>Slovakia</u>

- **Specific WAC** have been developed for all nuclear facilities in operation according to legislative provisions. **Generic WAC** are not used at the moment but may be helpful as an inspiration when developing preliminary WAC for a deep GDF. Generic WAC could provide a tool to facilitate RWM (incl. treatment, conditioning, packaging, storage and evaluation of disposability, etc.) for radioactive waste where no disposal route is currently available.
- All procedures that might have an influence on nuclear safety or radiation protection are applied in WAC including specific treatment, conditioning and/or packaging approaches.
- Specific WAC for VLLW and LLW disposal: One surface disposal facility is currently in operation (the national radioactive waste disposal facility in Mochovce). This accepts VLLW and LLW (predominantly comprising short-lived radionuclides) for disposal in separate areas.
- Preliminary WAC might be considered only for deep geological disposal, which shall be specified in updated version of feasibility study. A deep GDF is planned for the disposal of ILW (predominantly comprising long-lived radionuclides) and SNF. SNF is not reprocessed.
- There are WAC for interim storage of SNF.
- The Slovakian questionnaire response notes that only very small volumes of ILW will be consigned to a deep GDF (e.g. reactor internals and activated / heavily contaminated components). Decommissioning waste will be routed to Mochovce.
- No WAC for HLW
- There are WAC for transport – The shipment of radioactive waste is performed in certified transportation equipment on means of transport meeting the conditions of the European Agreement on international carriage of dangerous goods (ADR), or the Regulation concerning international carriage of dangerous goods (RID), Act No. 541/2004 Coll. and the Decree of ÚJD SR No. 57/2006 Coll.In SR the transboundary movement of spent fuel and radioactive waste, imports, exports are governed by Act No. 541/2004 Coll. and by the ÚJD SR Decree No. 57/2006 Coll., transposing the Council Directive 2006/117/Euratom on the supervision and control of shipments of radioactive waste and spent nuclear fuel, which is based on the IAEA recommendations formulated in the documents of TS-R-1 series.

<u>Spain</u>

- The WAC are specific for each facility or site from their safety assessment. The WAC are developed for Nuclear and Radioactive Waste in the whole waste life cycle of NPPs and from facilities or activities out of the nuclear waste cycle.
- LLW and VLLW are disposed of at the El Cabril near-surface disposal facility (there are two different repositories at El Cabril Disposal Centre: the LLW disposal facility and VLLW disposal facility). Treatment, conditioning and storage (of LLW) and a small amount of transitional storage (of VLLW) are also carried out at El Cabril [58].The El Cabril facility accepts only radioactive wastes with very low levels of long-lived alpha emiters (half-life > 30 years). Enresa has established a set of WAC linked to the safety assessment for operational and post-closure phases. The official document “WAC” establishes the limits and criteria that need to be accomplished by the Disposal Unit in order to be accepted and consequently disposed of in the disposal cells. (El Cabril is designed for direct disposal of disposal unites and not for direct

disposing waste packages; therefore, the disposal units are the entities that must actually fulfil the WAC. By this reason, specific WAC for the waste packages are derived from the relevant WAC for disposal unite (DU) in such a way that the waste packages automatically meet the DU WAC.)

- SNF discharged from NPPs is considered to be waste according to Spanish regulations. It is not reprocessed. There are WAC for SNF interim storage.
- A surface centralised storage facility (the CTS) is planned to provide temporary storage for all HLW (mainly SNF) from Spanish NPPs (for the next 60-100 years). Construction of the CTS has been originally planned to begin in 2019 but the permission has currently been blocked.
- Deep geological disposal is the preferred end point for management of HLW and SNF in Spain (along with special wastes and ILW that are not suitable for disposal at El Cabril). Provisional plans for the development of a deep GDF are set out in Spain's General Radioactive Waste Plan (GRWP).
- There are WAC for transport (LLW and ILW). All transport regulations base the requirements applicable to radioactive material on the IAEA's Regulations for the Safe Transport of Radioactive Material, SSR-6 (last edition SSR-6 (Rev.1) Edition 2018). Nevertheless, the requirements incorporated in the international regulations for the different modes of transport (ADR, RID, IMDG Code and ICAO technical instructions), which are currently applicable in Spain, are given in the 2012 edition of SSR-6. The CSN has published a guide for the application of the regulatory requirements on the transport of radioactive material, G.S-06.05 (Guía de Seguridad 6.5 - only Spanish version) [59].

<u>Sweden</u>

- VLLW is cleared for disposal at municipal landfills or else disposed of in shallow land burials at Studsvik and at the Forsmark, Oskarshamn and Ringhals NPPs. There are no WAC for VLLW. However, the permits for landfill include requirements that are equivalent to WAC which have to be met in order to dispose of waste.
- LLW will be disposed of at the SFR shallow GDF (~100 m below ground). The SFR facility accepts only radioactive wastes with very low levels of long-lived alpha emitters (half-life > 30 years). Construction of an extension to SFR is planned to start in 2024. **Specific WAC** are adopted and in use for the existing areas of SFR (these are specified conditioning and packaging approaches). **Preliminary WAC** have been developed for the SFR extension.
- Long-lived waste (i.e. ILW, according to IAEA waste classifications) is planned to be disposed of at the SFL GDF at a depth of 300-500 m (construction is planned to start in 2037). In the meantime, ILW is stored at the NPP sites. Plans for interim storage of some long-lived ILW in the SFR extension were withdrawn by SKB in 2017 [60]. Some acceptance criteria are established, linked to the safety assessment of the planed SFL facility. No WAC for disposal of ILW exist for SFL but Preliminary WAC are planned for the next few years.
- No reprocessing of Swedish SNF is carried out. SNF is planned to be disposed of in a deep GDF at a depth of 500 m. Construction is planned to start in 2027. There are no WAC for SNF management, but SKB, the Swedish nuclear fuel and RWM company ensures compliance with all handling steps before a new fuel type can be used.
- WAC apply to specific treatment, conditioning and/or packaging approaches - the WAC for the final disposal facility specifies the packaging approaches for different parts of the disposal facility as well as approaches for waste form:
    - ISO-container for non-conditioned LLW (maximum dose rate 2mSv/h)
    - Concrete- or steel moulds for LLW (maximum dose rate 500 mSv/h).
    - Waste conditioned with concrete or bitumen
- There are **Preliminary WAC** for the future geologic disposal facility for spent nuclear fuel. (For the future KBS-3 geologic disposal facility, the SNF must be in oxide form and have low dissolution rate in the disposal facility, while the waste consisting in structural parts, including radioactive activation products, shall have a certain corrosion resistance. The content of

radionuclides in the canister shall be determined and documented. The total decay power of the encapsulated SNF in each canister must not exceed a specified limit. Regarding the nuclear criticality, keff shall be less than 0.95 for the possibly occurring most reactive cases identified. For cases that can be deemed to be unrealistic but that would act to increase the reactivity, the criterion keff < 0.98 is applied. Also, a maximum allowed dose rate at the canister surface is set with respect to the possible impact of radiation on the functions of the engineered barriers of the final disposal facility.)

- WAC for transport – For transports on Swedish territory or otherwise under Swedish jurisdiction, the ADR, RID, IMDG-code and the ICAO Technical Instructions are applicable for the respective modes of transport. Shipments of radioactive waste or spent fuel, into or from Swedish territory, shall also be subject to handling within the scope of Directive 2006/117/EURATOM [61].

<u>Ukraine</u>

- For all existing and designed facilities (both predisposal and disposal), **specific WAC** shall exist.
- A new radioactive waste classification system aligned with that of the IAEA is currently being implemented - Besides of two types of radioactive waste (short-lived for disposal in the near-surface disposal facilities and long-lived for disposal in stable deep geological formations), four **classes** are established: VLLW, LLW, ILW and HLW – depending on acceptability for disposal in one of four types of disposal facilities: surface, near-surface, at intermediate depth and deep geological disposal facilities. These changes will be put **into force in 2021**.
- Currently, there are no WAC for VLLW
- Facility-**specific WAC** are required for all existing and designed facilities (both pre-disposal and disposal). They are developed initially during the design of the facility and form part of the safety case. These '**design WAC' are preliminary**; they are subsequently finalised in preparation for facility operation. In addition, technical specifications for waste packages are developed taking into account WAC for the subsequent lifecycle stages of RWM.
- At the moment, WAC are in use, in particular, for the following RWM facilities:
    - Engineered Near-Surface Disposal Facility (ENSDF) for disposal of LILW-SL
    - Buryakivka disposal facility for Chernobyl origin radioactive waste
    - Radioactive waste processing plants at NPPs including WAC for retrieval and pre-treatment
    - Centralized Storage Facility for DSRS - for a stage of "hot" tests.
- Also, technical specifications for radioactive waste packages are developed taking into account WAC for the subsequent stages of radioactive management.
- Two disposal facilities for LLW are currently in operation in Ukraine: the engineered near-surface disposal facility (ENSDF) at the Vector complex in the Chernobyl exclusion zone and the Buryakivka disposal facility for waste arising from the Chernobyl nuclear accident. The permissible content of long-lived activity is limited at both facilities through application of WAC.
- Various legacy storage / disposal facilities also exist at Radon enterprises and in the Chernobyl exclusion zone, for which a decision on whether to retrieve waste for disposal elsewhere will be made based on the results of safety assessment. Two more near-surface disposal facilities are also under construction at the Vector complex.
- A roadmap for development of a deep GDF for ILW and HLW disposal is under development – the Chernobyl exclusion zone is considered as the preferred location based on regional screening.
- SNF is not considered to be a waste in Ukraine. It is currently stored on site at NPPs. The Zaporizzhya NPP has its own long-term storage facility in operation. A centralised long-term storage facility for SNF from three other NPPs is under construction and a long-term storage facility for SNF from the Chernobyl NPP is under commissioning. Some SNF is being sent back to Russia [39].

- International transport of radioactive materials is carried out in accordance with the IAEA Standards and international agreements which Ukraine has joined including: Convention on International Civil Aviation Organization (ICAO); Convention on International Maritime Organization (IMO); European Agreement concerning the International Carriage of Dangerous Goods by Road (ADR); Convention on International Carriage of Dangerous Goods by Rail (RID) etc. [62].

United Kingdom

- The UK waste classification approach does not distinguish between short-lived and long-lived wastes – wastes are classified according to the type and quantity of radioactivity they contain and how much heat is produced.
- WAC are in place for the disposal of VLLW and LLW.
- Three landfill sites are permitted to receive VLLW - WAC have been adopted and are in use for waste acceptance at three facilities: East Northants Resource Management Facility (ENRMF), operated by Augean, Clifton Marsh, operated by Suez, and Lillyhall, operated by FCC Environment.
- The various WAC present the requirements for waste to be accepted at the disposal facilities, and associated requirements for packaging and transport. The WAC don't specify particular treatment or conditioning approaches.
- Two facilities are in operation for the disposal of LLW: the LLW Disposal facility (LLWR) in Cumbria and the LLW Facilities (LLWF) at Dounreay.
- The LLW facilities at Dounreay are authorised by the Scottish Environment Protection Agency (SEPA) to operate under an Environmental Authorisations (Scotland) Regulations 2018 Permit, which sets limits and conditions on the waste to be disposed of. Only waste generated at Dounreay or the Vulcan Nuclear Reactor Test Establishment (Vulcan NRTE) sites may be disposed of at these facilities; therefore the facility WAC are not publicly available and are not discussed in detail in the ROUTES questionnaire.
- A GDF is planned for the disposal of HLW, ILW and some LLW that is unsuitable for disposal to LLWR (collectively referred to as Higher Activity Waste, or HAW).
- There are currently no WAC developed for the disposal of Higher Activity Waste (HAW). However, Radioactive Waste Management Ltd produces packaging specifications as a means of providing a baseline against which the suitability of plans to package waste for geological disposal can be assessed. These are **generic in nature** (i.e. not specific to a particular site or disposal concept) and may also be regarded as preliminary WAC for the GDF.
- WAC for transport - There are regulatory frameworks for transportation of spent fuel and radioactive waste by road, rail, inland waterway, sea and air, which implement international agreements and regulations such as ADR, RID, ADN, IMDG, ICAO and other IAEA standards, directly by referring to the current version as revised or reissued from time to time [63].

Summary of main findings regarding WAC availability and application in LIMS

The following *Table 7* and *Figure 6* provide overview of LIMS applying WAC for individual classes of waste for each stage of their life cycle. As mentioned above, thirteen LIMS were surveyed:

- Belgium (BE)
- Bulgaria (BG)
- Czech Republic (CZ)
- France (FR)
- Germany (DE)
- Hungary (HU)
- Romania (RO)
- Slovakia (SK)
- Spain (ES)
- Sweden (SE)
- Ukraine (UA)
- United Kingdom (UK)

- Lithuania (LT)

| Waste class | *Retrieval and pre-treatment of raw waste (e.g. sorting, drying,…)* | *Waste treatment (e.g. thermal treatment)* | *Waste conditioning and packaging* | *Interim storage* | *Transport** | *Disposal* |
|---|---|---|---|---|---|---|
| | WAC availability in LIMS (13 LIMS surveyed)<br>*LIMS with WAC / LIMS with a given waste class/radioactive waste(SF)* | | | | | |
| *VLLW* | BG, CZ, LT, RO, SK, ES, SE, *(7/11)* | CZ, FR, LT, RO, ES, SE *(6/11)* | BG, CZ, FR, DE, LT, RO, SK, ES, SE, UK *(10/11)* | BG, CZ, FR, LT, RO, SK, SE, *(7/11)* | BG, CZ, FR, HU, LT, RO, SK, ES, SE, UA, UK *(11/11)* | BG, CZ, FR, LT, RO, SK, ES, SE, UK *(9/11)* |
| *LLW* | BE, BG, CZ, LT, RO, SK, ES, SE, UA *(9/13)* | BE, BG, CZ, FR, DE, HU, LT, RO, SK, ES, SE, UA, UK *(13/13)* | BE, BG, CZ, FR, DE, LT, RO, SK, ES, SE, UA, UK *(12/13)* | BE, CZ, FR, DE, HU, LT, RO, SK, ES, SE, UA *(11/13)* | BE, BG, CZ, FR, DE, HU, LT, RO, SK, ES, SE, UA, UK *(13/13)* | BE, BG, CZ, FR, DE, HU, LT, RO, SK, ES, SE, UA, UK *(13/13)* |
| *ILW* | BE, BG, CZ, LT, SK, ES, SE, UA *(8/13)* | BE, BG, CZ, FR, DE, HU, LT, SK, ES, SE, UA, UK *(11/13)* | BE, BG, CZ, FR, DE, LT, ES, SE, UA *(9/13)* | BE, CZ, FR, DE, HU, LT, RO, SK, ES, SE, UA, UK *(12/13)* | BE, BG, CZ, FR, DE, HU, LT, RO, SK, ES, SE, UA, UK *(13/13)* | BG, CZ, FR, DE, HU, LT, RO, ES, UA *(9/13)* |
| *SNF/UNF* | *(0/13)* | *(0/13)* | CZ, LT, ES *(3/13)* | CZ, LT, SK *(3/13)* | BE, BG, CZ, FR, DE, HU, LT, RO, SK, ES, SE, UA, UK *(13/13)* | SE, *(1/13)* |
| *HLW* | BG, DE (2/12) | FR, UA (2/12) | BE, BG, FR, DE, UA (5/12) | BE, FR, DE, UA, UK (5/12) | BE, BG, CZ, FR, DE, HU, LT, RO, SK, ES, UA, UK (12/12) | FR (1/12) |

**LIMS* using ADR and IAEA standards as "generic" WAC for transport of radioactive waste/SNF

*Table 7– WAC availability and utilisation for different stages of the waste lifecycle in LIMS*

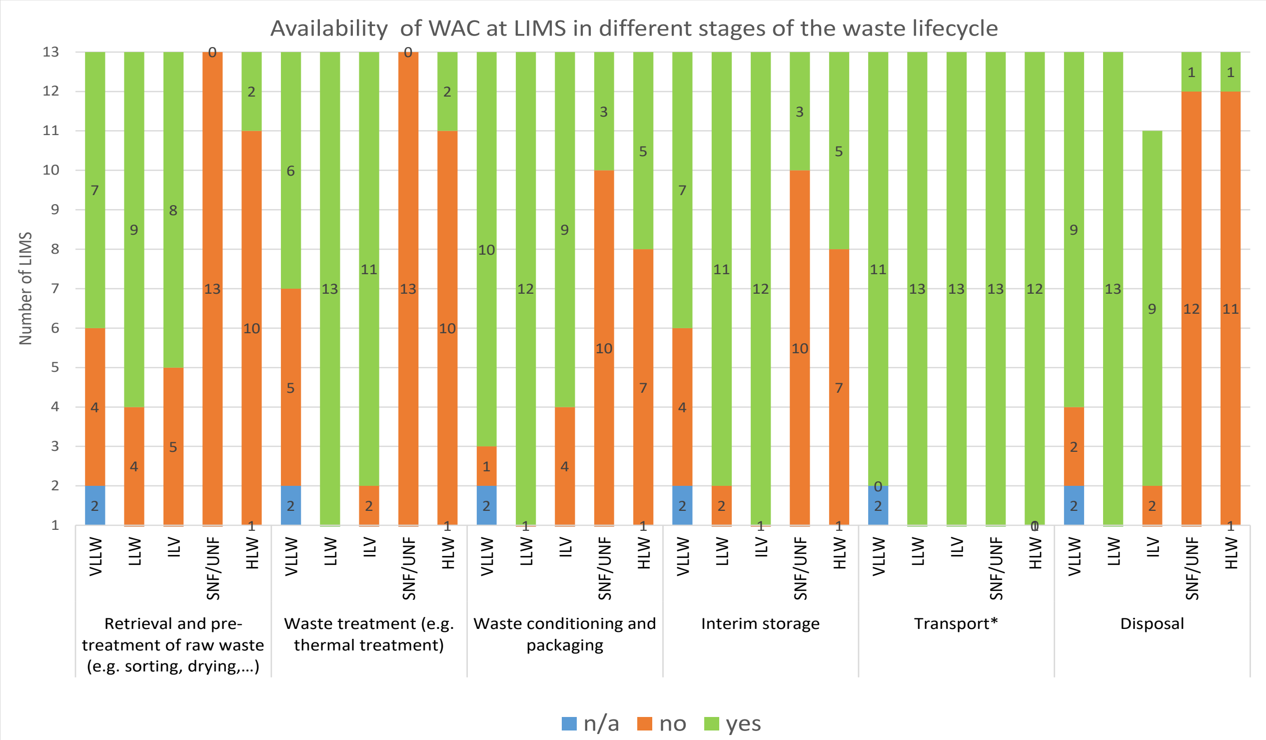



*Figure 6 – Availability of WAC in LIMS in different stages of the waste lifecycle for various waste classes*

*Notes: n/a - not applicable (the classification is not used, no wastes of the class), No – WAC don't exist, Yes - WAC exist; *LIMS using ADR and IAEA standards as "generic" WAC for transport of radioactive waste/SNF*

- WAC are most often and most comprehensively applied in the field of low-level waste management, more or less, across all stages of the waste life cycle. A similar situation exists in the use of WAC for ILW.
- Disposal WAC are typically more established for VLLW and LLW because it is generally easier to progress plans for the disposal of these waste classes. However, many countries don't make use of the VLLW classification and instead, in some cases, a common system of acceptance criteria for VLLW and LLW is derived and in use. Elsewhere, where the VLLW classification has only recently been introduced, the VLLW management programme is currently under development and associated WAC may not yet exist.
- SNF management is one of the least developed areas of RWM in terms of WAC application. There are several reasons why this is (or may be) the case:
  - SNF isn't categorized and classified as radioactive waste.
  - Repatriation of SNF to the country of origin (e.g. Romania – SNF returned to Russia / USA; Ukraine - some historic SNF is being sent back to Russia).
  - SNF from NPPs is stored in dedicated on-site facilities and there are no WAC for these facilities because all of the associated SNF is accepted.
  - Characterization of SNF is not done in the same manner as for waste for disposal because at the present time interim storage of SNF is conducted.
  - SNF is generally stored without treatment or conditioning therefore there are no WAC; reprocessing of nuclear fuel is not part of the spent fuel management programme / strategy.
  - Absence *of* WAC for SNF disposal - there are no appropriate disposal facilities for SNF

- There are no WAC for SNF however, responsible organisation (Waste Management Organizations) shall ensure that all requirements for the safe management of spent fuel are met as well as compliance with all handling steps before a new fuel type can be used.

In some countries surveyed, WAC for interim storage of SNF are adapted and in use (the Czech Republic, the Netherlands and Slovakia).

Preliminary WAC for future geologic disposal facility for SNF are developed (e.g. Sweden KBS-3 GDF).

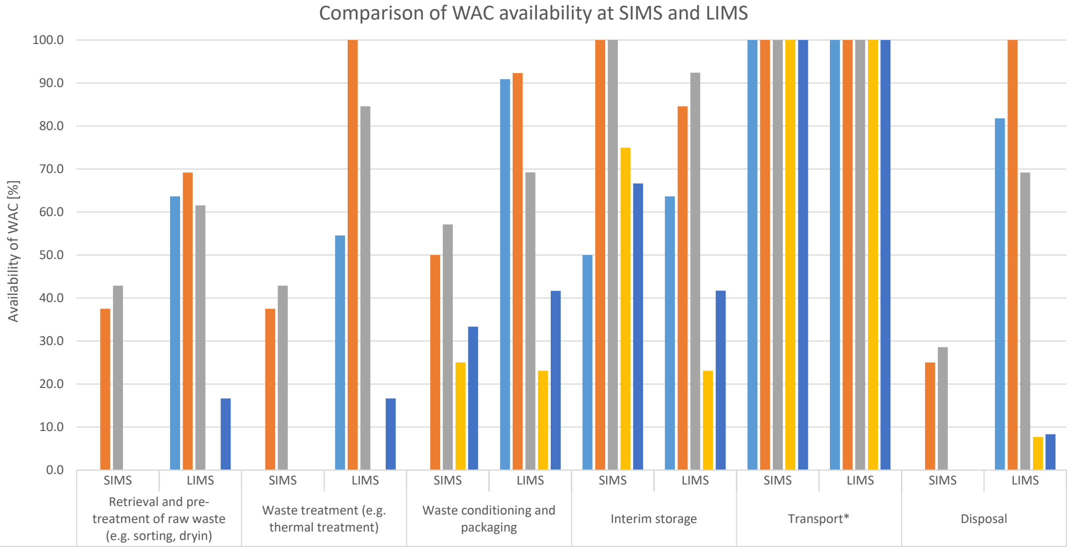
- Another special area where WAC are currently applied only sporadically is the HLW (may also include SNF/UNF, where declared as waste as already explained above) management. Some countries give reasons why storage WAC do not apply for HLW, e.g.:
  - There is no HLW
  - HLW management programme is in the early stage
  - A roadmap for development of appropriate disposal facility for HLW is under development

Preliminary WAC are derived for planned interim storage and disposal facilities which are/will be considered in the safety case (e.g. Bulgaria – WAC for Facility for interim long-term storage of HLW and long-lived LLW and ILW; France – WAC for Cigéo deep GDF)

- In several cases, treatment, conditioning or packaging requirements can be part of WAC set covering multiple lifecycle stages, however, they are primarily focused on disposal.
- WAC for transport - for transport between nuclear licensed sites and different users, most countries apply IAEA transport regulations and international agreements concerning the carriage of dangerous goods as WAC for transport. These regulations are usually reflected and implemented in the national legislative framework. However, sometimes they are applied directly without such implementation in national legislation or such promulgation. Transport between facilities on the same site is often not subject to formalised specific WAC, although there may still be export and/or receipt requirements at the original and destination facilities respectively.

### 3.4.3 Differences in WAC availability at SIMS and LIMS

*Figure 7* compares WAC availability in SIMS and in LIMS for different waste classes across all stages of the waste life cycle.




| | SIMS | LIMS | SIMS | LIMS | SIMS | LIMS | SIMS | LIMS | SIMS | LIMS | SIMS | LIMS |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | Retrieval and pre-treatment of raw waste (e.g. sorting, dryin) | | Waste treatment (e.g. thermal treatment) | | Waste conditioning and packaging | | Interim storage | | Transport* | | Disposal | |
| VLLW | 0.0 | 63.6 | 0.0 | 54.5 | 0.0 | 90.9 | 50.0 | 63.6 | 100.0 | 100.0 | 0.0 | 81.8 |
| LLW | 37.5 | 69.2 | 37.5 | 100.0 | 50.0 | 92.3 | 100.0 | 84.6 | 100.0 | 100.0 | 25.0 | 100.0 |
| ILW | 42.9 | 61.5 | 42.9 | 84.6 | 57.1 | 69.2 | 100.0 | 92.4 | 100.0 | 100.0 | 28.6 | 69.2 |
| SNF / UNF | 0.0 | 0.0 | 0.0 | 0.0 | 25.0 | 23.1 | 75.0 | 23.1 | 100.0 | 100.0 | 0.0 | 7.7 |
| HLW | 0.0 | 16.7 | 0.0 | 16.7 | 33.3 | 41.7 | 66.7 | 41.7 | 100.0 | 100.0 | 0.0 | 8.3 |

*Figure 7 – Comparison of WAC availability in SIMS and in LIMS*

*The percentages given here mean the number of SIMS/LIMS with WAC per number of SIMS/LIMS, which have the specific class of waste.*

- In general, the analysis has confirmed that LIMS have much more comprehensive WAC systems. WAC for disposal are in far more widespread use in LIMS, although such WAC generally only cover the disposal of VLLW and LLW (and in the Czech Republic and Hungary, also ILW), while in the case of SIMS, the focus is mainly on interim storage of LLW and ILW and to a lesser extent to their conditioning.
- For both SIMS and LIMS, the determination and application of WAC systems is focused mainly on the area of low-level or intermediate-level waste management.
- The biggest differences between SIMS and LIMS are in the area of waste disposal. While most LIMS have developed WAC for VLLW, LLW and ILW, in the case of SIMS there are only two out of eight countries that have WAC for LLW and two out of seven with WAC for ILW.
- In a number of cases, treatment, conditioning or packaging requirements can be part of WAC covering multiple lifecycle stages. However, they are primarily focused on storage and/or on disposal (if a disposal facility is available).
- In many countries, especially in the case of SIMS, VLLW classification is not defined within the national radioactive waste programme or through national legislation. In some cases, a common system of WAC for VLLW and LLW is derived and in use. This is also the main reason why WAC for very low-level waste are far less progressed in SIMS than in LIMS. The introduction of a classification of VLLW and a separate disposal route for ones can bring some benefits - a disposal facility that guarantees the safe disposal VLLV is significantly less costly than a more sophisticated NSDF and additionally site selection and licensing procedure might be a lot easier as well, and may also be easier for public acceptance.
- On the other hand, in some countries (e.g. Germany, Netherlands, Denmark), disposal in deep geological formations is intended for all types of radioactive waste. Hence it is not necessary to distinguish between wastes containing radionuclides with comparatively short half-lives and those with comparatively long half-lives. Radioactive wastes are therefore subdivided based on whether or not they generate significant quantities of heat. This combined approach could be useful in the case of SIMS. However, the combination of a NSDF and second one for LILW-LL needs detailed characterization that means more radiochemical analyses and higher cost. It should also be noted that it is not just a question of a few technical aspects and financial cost. The reasons why a country decides on certain disposal option, e.g. to dispose only geologically, are generally a combination of many and very diverse aspects, such as e.g. socio-economical, geographical and, last but not least, political aspects.
- SNF management is one of the least developed areas of RWM in connection with the WAC application in LIMS as well as SIMS and the common reason why WAC for SNFs have not yet been derived could be for example: the absence of SNF in the country, SNF isn't categorized and classified as radioactive waste, repatriation of SNF to the country of origin, there are no appropriate disposal facilities for SNF or there are no WAC for SNF however, responsible organisation (Waste Management Organizations) shall ensure that all requirements for the safe management of spent fuel are met as well as compliance with all handling steps before a new fuel type can be used.
- There are no significant differences between LIMS and SIMS in terms of conditions and requirements for the transport of radioactive waste and related WAC. As already mentioned, for transport between nuclear licensed sites / users, most countries simply apply IAEA transport regulations/requirements and international agreements concerning the carriage of dangerous goods as WAC for transport, without specifying additional requirements. Elsewhere, transport WAC are heavily based on these international regulations and requirements, with extensions to reflect the national context. Transport between facilities of the same site is often not subject to formalised WAC, although there may still be requirements regarding export and receipt of waste at the premises of origin and destination respectively.
- Similarly, for both SIMS and LIMS, the transport of radioactive waste comprises all operations and conditions associated with and involved in their movement; these include the design, manufacture, maintenance and repair of packaging, and the preparation, consigning, loading, carriage including in-transit storage, unloading and receipt at the final destination of loads of

radioactive waste and packages (see e.g. 1.7.1.3 ADR). In many countries, WAC for transport are not specified, however, maximum values of radionuclides (or their mixtures) in the transport container and other transport conditions (such as dose effective rate) are determined. The transport container must be type approved on the basis of the prescribed tests.

- During transport, the shielding function of the container and the entire packaging assembly is essential, while during storage and disposal of radioactive waste the life of the packaging assembly is important, sufficient shielding can be ensured otherwise when needed. There is no connection between the transport conditions and the WAC for disposal. Some transport conditions concerning the container/packaging assembly may also apply to storage. However, in the case of storage, the value of activity at the workplace may be limited by specific WAC. From the point of view of safety assessment, the derivation of the WAC for disposal (storage) is not related to the derivation of the transport conditions, i.e. the fulfilment of transport conditions does not guarantee the fulfilment of WAC for disposal (storage) and vice versa.
- As discussed in the MS88 report [29], the large number of countries with preliminary or generic disposal WAC reflects the relatively long lead times for siting, developing, implementing and licensing disposal facilities, particularly geological repositories for ILW and HLW. Preliminary and generic WAC therefore persist for longer periods and help to facilitate timely RWM even when disposal facilities are not available, as well as contributing to documentation that underpins approval of the facility by the relevant safety authority and other stakeholder groups.
- Also as discussed in the MS88 report, generic WAC may be precursors to site-specific WAC (e.g. in Ukraine and the UK), but can also be final (i.e. in use) requirements for an operation to be performed at multiple sites (or by multiple organisations / licensees), as is the case for generic WAC for storage and 'disposal' (discharge) of radioactive waste in Cyprus. In the case of a French CSA disposal facility, both specific and generic criteria are adopted and used at the same time (generic - Basic Safety Rules at national level, for the management of LLW emitted by French regulator; specific national license for the CSA Site and ANDRA specific WAC.

## 3.5 Disposal facilities in SIMS

Every MS that uses nuclear technology generates radioactive waste, which needs special management to protect human health and the environment. Waste disposal facilities must provide a safe, secure and permanent solution for future generations. Most of the disposal solutions were developed for LIMS. For MS with no significant waste inventories, especially for small amounts of ILW and HLW/SNF, developing a dedicated waste disposal facility poses a significant challenge due to the relatively high fixed costs for construction of disposal facilities, particularly those at depth. Therefore, SIMS need alternative disposal options with an equal protection level compared to more conventional disposal options but with lower programme costs. To demonstrate safety performance of alternative disposal options, extensive safety cases of disposal concepts are required, which are not available in most cases. Hence, most SIMS do not have a final disposal solution.

Nevertheless, some examples of disposal facilities that are in operation in SIMS in Europe, as well as outlining joint initiatives towards the implementation of disposal solutions in SIMS already exist and are described in this subchapter.

### 3.5.1 Disposal facility in Poland – NRWR in Różan

The National Radioactive Waste Repository (NRWR) in Różan (district of Maków, 90 km from Warsaw) is a near-surface type disposal facility covering 3.045 hectares, being the first and only radioactive waste disposal facility in Poland operating since 1961. It is located at a former military fort constructed in the years 1905-1908 (see *Figure 8*).


*Figure 8 – Location of nuclear site and radioactive waste disposal facility in* Różan, *Poland* [64]

Geology of Różan site is characterized by boulder and sandy clays with groundwater table at the depth 15-20 m. There are four concrete structures with roof and wall, thickness of 1.2 to 1.5 m and the floor layer of 30 cm (Układ Tytuł (iaea.org)).

NRWR in Różan serves for disposal of LILW-SL (with half-life of radionuclides being shorter than 30 years). The facility is also used to store long-lived, mainly alpha radioactive waste, as well as DSRS waiting to be placed in a deep geological disposal facility. At present, the following facilities located at NRWR area are used (see *Figure 9*):

- No. 1 (partially) – storage for LILW-SL,
- No. 3a – disposal of LIL-SL (Low and Intermediate Level, short lived) DSRS,
- No 8 – disposal of LILW-SL,
- No 8a – storage for LILW-LL.

Historical LILW-SL and LILW-LL (non-segregated, only partially conditioned and packed) are stored in the facilities No 1, 2, 3.


*Figure 9 – National Radioactive Waste Repository in Różan* [64]

According to WAC, the waste can be disposed of in Rozan repository only in solid or conditioned form. Since 1968, LILW-SL has been disposed of in the part of the dry moat in the Rozan fort. The bottom and slopes of the moat have been covered with 20 cm thick concrete layer. The waste is arranged in the 'layer by layer' mode, and the free space between waste packages is filled with concrete. Long-lived waste is placed in separate facility with the intention of retrieval. [41]

The inventory of radioactive waste and nuclear materials deposes and/or stored at NRWR Rožna is shown in *Table 8* and *Table 9*.

| Facility No. | Volume [m3] | Activity on 31.12.2019 [GBq] | Waste Class |
|---|---|---|---|
| 1 | 805.5 | 14,004.7 | LLW-LL |
| 2 | 46.9 | 342.9 | |
| 3 | 530.5 | 2,449.0 | |
| 8a | 3.7 | 756,8 | |
| 3a | 1.9 | 5,627.5 | DSRS LIL-SL |
| 8 | 2,706.7 | 24,456.3 | LILW-SL |
| TOTAL | 4,095.2 | 47,637.2 | - |

*Table 8 – Waste disposed or stored at NRWR in Różan site (01.01.1961 – 31.12.2019)* [64]

| Nuclear materials | Mass |
|---|---|
| Sources Pu-Be | 42.6 g |
| Depleted U | 3,504.2 kg |

*Table 9 – Nuclear materials stored at the National Radioactive Waste Repository in Różan site (01.01.1961 – 31.12.2019)* [64]

*Notes*: For the activity of particular isotopes present in the waste stored/disposed at the Różan repository in the period of time 1.01.1961 – 31.12.2019 see Annex III of [64].

Due to the fact that the disposal space is running short, the repository is scheduled to be prepared for closure in the years 2021–2024 and for ultimate closure in 2025-2029 (as stipulated in the National Plan for Radioactive Waste and Spent Nuclear Fuel Management). The aim of the undertaken actions is therefore preparation for closing of the NRWR in Rozan and then its final closing and long-term monitoring.

### 3.5.2 Disposal facilities in Slovenia

There are two permanent waste disposal sites in Slovenia: the Jazbec mine waste pile and the Boršt mill tailings site at the closed uranium mine Žirovski vrh (see *Figure 10*). A further near-surface facility for disposal of LLW and some ILW is planned at Vrbina (see subchapter 4.2.5.3).


*Figure 10 – Disposal sites for waste and residual material produced during uranium ore mining, processing and decommissioning are situated at the settlement of Todra*

The Žirovski vrh Uranium Mine was in operation between 1984 and 1990. Its lifetime production was 610,000 tons of ore, from which 452.5 tons of $U_3O_8$ was produced. The Žirovski vrh Uranium Mine ceased regular operations in 1990. All entrances to the underground mine are closed. The uranium mill is decommissioned, and the resulting wastes are disposed of on the mining waste disposal site Jazbec. At this site, all mining waste from numerous other mining waste piles has been moved and disposed of. The total amount of disposed material on this site is 1,910,425 tons with a total activity of 21.7 TBq. The disposal site of Jazbec is located approximately 3 km from Gorenja vas, a settlement with about 1000 inhabitants and its area (the area inside safety fence of the mine waste pile) is about 74,239 m$^2$. The mine waste pile is situated on the slope and covers an area of 67,325 m$^2$, at the elevation of 427 to 509 m. The remediation works set the inclination of the pile to 24 – 27 degrees. Three-metre-wide berms were constructed every 12 m of height. The remediation work on Jazbec disposal site is finished. The restored disposal site is covered with 2 metres of a complex covering layer, including 0.3 metre of clayey radon barrier directly over the waste material, and a humus soil layer on the top. The surface of the disposal site is covered with grass. Ecological succession to forest is prevented by regular mowing and removal of shrubs to prevent damage to the protective layers by ingrowth of the roots. Lateral surface drainage channels were constructed along each side of the waste pile and leaching water from the disposed material is collected in the drainage tunnel in the lower part of the waste pile (see *Figure 11* & *Figure 12*). Its outflow and the surface drainage channels are joined at the bottom of the waste pile and flow into the water stream of Brebovščica. [65] [66].

On the uranium mill tailings disposal site Boršt, 610,000 tons of hydrometallurgical waste, 111,000 tons of mine waste and 9,450 tons of the material collected during decontamination of the mill tailings site Boršt vicinity have been disposed of with a total activity of 48.8 TBq (total amount of disposed material is 730,450 t). The area of Borst is 42,000 m$^2$. Remediation work at the Boršt disposal site has been delayed due to the activation of a landslide. After the final administrative closure of the sites, the long-term surveillance and maintenance will, according to the law, be entrusted to ARAO.

*Figure 11 – Slope of the Jazbec disposal site* [65] [66] *.*

*Figure 12 – Lateral drainage channel leads to the main channel of the Jazbec stream (red arrow)* [65] [66]

### 3.5.3 Disposal facility in Norway

Norway is not a member of the EU but has, like many EU states, only a small inventory of radioactive waste. Unlike other SIMS (except for e.g. Poland, Slovenia), Norway has a disposal facility for radioactive waste generated by RR operations and by the use of radioactive materials in medical, research and industrial applications. In general, most of these wastes contain only low or medium levels of activity consisting mainly of short-lived radionuclides and are classified as LILW-SL. There are also LILW-LL in Norway as well as spent nuclear fuel from the RRs.

The Himdalen facility is the national disposal and storage facility for LILW; it is located 40 km east of Oslo. The facility consists of four large halls excavated in a mountain, three of them for disposal and one for storage (see *Figure 14*). The total capacity is equivalent to 10,000 drums (210 litres).

The Himdalen facility has four caverns (halls) for waste packages and one slightly inclined 150-metre-long access tunnel for vehicles and personnel. All caverns and the access tunnel have a monitored water drainage system. A service and control room with service functions for personnel and a visitor's room are located along the tunnel. The rock caverns are excavated in such a way that about 50 meters of rock covering remains. This natural geological cover is intended for protection against intruders, plane crashes and other untoward events, although it is not intended to act as a main barrier in long-term safety calculations. Long-term safety will rely on the engineered barriers. In each cavern, two solid sarcophagi have been constructed with a concrete floor and walls. When a section of the sarcophagus has been filled, it is planned that a roof will be constructed. The roof of the sarcophagus will be shaped to shed infiltrating ground water, and a waterproof membrane will be affixed to the concrete roof. Three caverns (*Figure 13*) will be used for waste disposal, with drums and containers stacked in four layers. When one layer in a sarcophagus section has been filled with waste packages, it will be encased in concrete. One of the caverns is used for storage for certain waste packages (166 drums of the old, retrieved waste packages containing some plutonium). The decision whether to retrieve the waste in the storage cavern or dispose of it by encasing it in concrete will be made based on experience during the operational period and the safety reports to be prepared for closure of the facility, expected by the year 2030. There are no plans to retrieve any of the waste placed into the storage facility during operation [67] [68].

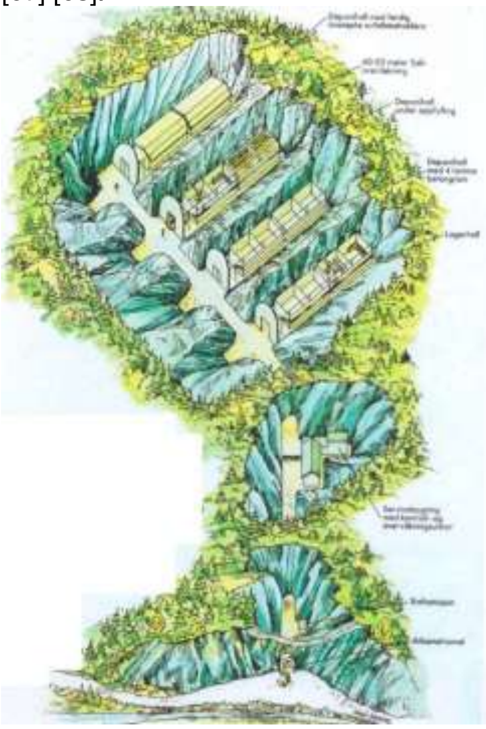


*Figure 13 – Cavern for waste disposal at the Himdalen facility* [69]

*Figure 14 – Himdalen Disposal and Storage Facility for LILW* [69]

<u>The ERDO Association</u>

Since the availability of disposal facilities for SIMS is limited and only a few countries have developed a final solution for the most hazardous and long-lived radioactive wastes, the ERDO association was initiated by organisations from seven countries (Slovenia, Netherlands, Denmark, Italy, Croatia, Poland and Norway). Every country has a national responsibility to establish a programme and schedule for the safe management and disposal of radioactive wastes and spent fuel. This requires the investment of significant financial, technical, and human resources. The costs are high relative to the quantities of materials involved, and independent disposal facilities would be a poor use of limited national resources. The objective of the project is to work together to address the common challenges of safely managing the long-lived radioactive wastes in their countries and to provide the necessary groundwork to enable the establishment of one or more operational and shared multinational RWM solutions.

Since many challenges are common to different countries, sharing knowledge, technology and facilities can bring benefits. Hence, the ERDO association has the advantage of reducing the costs associated with national programme development by shareing multinational approaches to common problems. Cooperative RD&D has resulted in optimized technical solutions,implementation of common solutions and facilities, and cost optimization for RWM [70] [71]**.**

# 4. Long-term management solutions for SIMS

Based on the SSR-5, IAEA, 2011 [72] a number of design options for disposal facilities have been developed and various types of disposal facility have been constructed in many states and are in operation. These design options have different degrees of containment and isolation capability appropriate for the radioactive waste they will receive.

The specific objectives of the disposal are:

- To contain the waste;
- To isolate the waste from the accessible biosphere and to reduce substantially the likelihood of, and all possible consequences of, inadvertent human intrusion into the waste;
- To inhibit, reduce and delay the migration of radionuclides at any time from the waste to the accessible biosphere;
- To ensure that the amounts of radionuclides reaching the accessible biosphere due to any migration from the disposal facility are such that possible radiological consequences are acceptable low at all times.

There are several types of disposal options adopted around the world, corresponding to specific classes of radioactive waste, as follows:

(a) **On Surface**: Two options for disposal of waste on surface level are available: specific landfill disposal and long-term interim storage, if the waste can be released after a specific period.

Disposal in a specific landfill disposal facility similar to a conventional and fill facility for industrial refuse, is possible if measures to cover the radioactive waste are incorporated. Such a facility may be designated as a disposal facility for VLLW with low concentrations or quantities of radioactive content [9]. Typical waste disposed of in a facility of this type may include soil and rubble arising from decommissioning activities. Another waste type is mining and mineral processing waste which is usually disposed of on or near surface, but the manner and the large volumes in which the waste arises, its physicochemical form and its content of long lived radionuclides of natural origin distinguish it from other radioactive waste. The waste is generally stabilized in situ and covered with various layers of rock and soil.

Long-term interim storage can reduce the activity content of VLLW and LLW to a point where all or part of the waste can be released. This usually requires an appropriate treatment facility. Depending on the type of waste in SIMS, this can be an important addition to the disposal facility or, if the activity permits, it can even be the only option needed and a disposal facility is not required at all.

(b) **Near Surface**: Disposal in a facility consisting of engineered trenches or vaults constructed on the ground surface or up to a few tens of meters below ground level. Such a facility may be designated as a disposal facility for LLW [9], but depending on its characteristics (i.e. low concentration or quantities of long lived radionuclides), ILW might be as well suited to be disposed of in near surface disposal facilities [6] [9]. Trench disposal systems are best suited for the disposal of VLLW. These systems are similar in many aspects to conventional municipal or industrial waste disposal facilities. These types of facilities can also be constructed above ground as mounds. Both trench and mound systems are considered suitable for the disposal of waste with limited isolation requirements, generally for several decades. The trench is usually designed and constructed with an overall slope that provides drainage to a single collection point, i.e. a sump. The base of the trench is completed with a permeable aggregate layer allowing drainage of water to the collection sump. The aggregate material is specifically selected to minimize clogging of pores, thereby maintaining the permeability of the drainage layer. A maintenance shaft provides access to the sump. The sump is installed in a concrete footing that provides adequate stability and support for the access shaft.

(c) **Geological**: Disposal in a facility constructed in tunnels, vaults or silos in a particular geological formation (e.g. in terms of its long-term stability and its hydrogeological properties) at least a few hundred meters below ground level. Depending on its characteristics, ILW has to be disposed of in a geological disposal facility [6] [9]. Disposal could then be by emplacement in a facility constructed in caverns, vaults or silos up to a few hundred meters below ground level. It could include purpose-built facilities and facilities developed in or from existing mines. It could also include facilities developed by drift mining into mountainsides or hillsides, in which case the overlying cover could be more than 100 m deep. Such a facility could be designed to receive HLW [9], including spent fuel if it is to be treated as waste. However, with appropriate design, a GDF could receive all types of radioactive waste.

Geological disposal can concern the disposal in a facility consisting of an array of boreholes, or a single borehole, which may be between a few tens of meters up to a few hundreds of meters deep. Such a borehole disposal facility is designed for the disposal of only relatively small volumes of waste, in particular DSRS. A relatively new option for very deep geological disposal storage is drilling technology.. A design option for very deep boreholes, several kilometers deep, has been examined for the disposal of solid HLW and spent fuel, but so far this option has not been adopted by any State.

The division of the sections in this chapter represents the above-mentioned types of radioactive waste disposal options with a special look at disposal sites in planning or operation. Due to the recent developments in the field of deep geological disposal utilising boreholes, this part will be discussed in greater detail.

## 4.1 On Surface Long-term Interim Storage

### 4.1.1 Long-term interim storage in Austria - Nuclear Engineering Seibersdorf GmbH

From the end of the 1960s, the Seibersdorf site (*Figure 15*) began collecting and temporarily storing radioactive waste on behalf of the Republic of Austria. Over time, more and more treatment and conditioning facilities were built (e.g. incinerator, high-force compactor, drying systems etc.)

At the beginning, it was assumed that a final disposal facility would be available by the 1980s, but the date was postponed several times afterwards. There is currently a trilateral contract between the Republic of Austria, the Seibersdorf municipality and Nuclear Engineering Seibersdorf that NES will take over RWM until 2045 and thus also operates the interim storage facility until then.

*Figure 15 – NES site*

Since there are no NPPs in operation in Austria, no fuel cycle systems exist and the spent fuel elements from the existing RRs have been and are being returned to the US, there is no spent nuclear fuel and no HLW, but only LILW. The majority of them are short-lived waste (half-life <= 30 years), only 2 to 3 % must be considered and treated as long-lived waste.

Since there is still no specific plan for a disposal facility (i.e. a NSDF is expected to be developed for the short-lived waste, other solutions such as a borehole for the long-lived waste), there are no WAC for disposal.

<u>Interim storage</u>

In the past, the waste drums were conventionally stacked close together in the interim storage facility (*Figure 16*). The (older) drums, which were at the back and below, were no longer accessible and an inspection was hardly possible. On older drums, some very complex inspections (e.g. with endoscopy systems) also revealed signs of corrosion, which was easily explained due to the fact that such a long interim storage period was never planned. In addition to the corrosion, old containers contained e.g. also flammable material, free liquids and thus no longer corresponded to the modern acceptance requirements of interim storage facilities. The documentation of the old containers (e.g. with regard to radiological or chemical parameters as well as information on treatment steps and conditioning procedures) was incomplete or inadequate.

*Figure 16 – Former storage concept*

During the extension of the interim storage contract to 2030 (and subsequently 2045), the requirement arose to ensure the long-term stability of the containers and to enable an inspection of each individual container.

For this purpose, a new interim storage concept had to be developed and it became necessary to subject older waste packages to a reconditioning, during which they were brought up to the state of the art with regard to long-term stability for interim storage and acceptance criteria for final disposal. The documentation also had to be followed up.

Since there is no (advanced) disposal facility project in Austria and therefore no WAC for disposal exist, explicit and detailed WAC for the interim storage facility of the NES were developed and implemented. The focus was on the one hand on safety requirements for interim storage and on the other hand on international documents and guidelines as well as WAC for repositories for comparable radioactive waste (LILW). In doing so, the principle was followed to carry out the conditioning as safely as possible (e.g. exclusion of chemical processes in the waste container, prevention of possible pressure build-up, safe enclosure, long-term stability, resistance, ...), but to remain as flexible as possible in order to be able to respond to future regulations and WACs for the final disposal facility with the least possible effort. The concept of a 200L drum as a standard waste container was used as much as possible. For example, high-force compacted pellets are put into the drums, but not grouted/cemented, which means that the flexibility is retained to be able to transfer the pellets later to other final disposal containers or to be able to feed the drums as a whole for final conditioning (e.g. grouting/cementing into cuboid final disposal containers).

In principle, all conditioned waste packages - if technically feasible - are subjected to drying (*Figure 17*) before they are brought to the interim storage. This removes the residual moisture from the waste and minimizes the risk of subsequent chemical reactions or corrosion phenomena as much as possible.

*Figure 17 – Drum drying facility*

At NES 200litre steel drums (*Figure 18*) with a high-quality, multi-layer coating are used as standard containers. The drums are closed with a screwed flange lid; inside the drums there is a glass fiber reinforced inner liner, which on the one hand protects the drum coating from damage(e.g, protects when the waste is brought in) and, on the other hand, prevents contact corrosion between the drum and metallic parts in the waste.

*Figure 18 – Not radioactive sample of NES standard waste drum with pellets*

To prevent a possible pressure build-up in the waste drums during interim storage, sintered metal filters are installed in all the drum lids (this is strictly necessary for waste that can change like soil or sludge), which on the one hand prevent the escape of aerosols, but on the other hand enable pressure equalization with the environment and thus prevent pressure build-up (see *Figure 19*). In the case of drums with radioactive waste, in which the emanation of radioactive substances through the sintered metal filter cannot be ruled out with certainty (e.g. in the case of waste containing radium like soil containing radium), a tightly fitting septum made of rubber is used instead of the filter. This can be pierced with a needle for taking and analyzing a gas sample and measuring the differential pressure, which makes it possible to detect chemical reactions (e.g. reactions of Aluminium with concrete, corrosion of metals, reactions with organic substances) and any pressure build-up early on and to initiate the necessary countermeasures immediately.


*Figure 19 – Sinter metal drum filter, rubber septum*

In the acceptance criteria for the NES interim storage facility, the permitted drum and container types are described and specified in detail, for example the following standard types are described:

- Pellet drums: drums with high force compacted radioactive waste.
- Drums with homogeneously cemented waste: In these drums, liquid, dusty etc. radioactive waste such as sludges, ashes or salts were homogeneously distributed and bound in a concrete matrix.

Ash drums: The ashes from the NES incineration plant are tightly welded in thick-walled stainless steel cartridges; high force compaction or grouting of the ashes are (for the time being) refrained from. Homogeneous cementation is no longer carried out due to bad experiences in the past (the cement cylinders showed that they are not stable).

- Drums with DSRS: Depending on the radionuclide, the radioactive sources are conditioned in different container types and structures regarding the necessary shielding.

All other requirements such as maximum mass, maximum activity inventory, permissible dose rate, required chemical characterization, etc. are also specified in the acceptance criteria.

When the conditioned waste containers are taken over by the interim storage facility, compliance with the acceptance criteria is checked according to the dual control principle.

In order to ensure that the drums can be inspected, a storage system was developed and optimized which enables the drums to be accessed with minimal overall space requirements (*Figure 20*). The drums are stored horizontally on specially developed storage racks, between which there are inspection aisles that allow access to the containers. The storage racks also ensure that the drums are not subjected to stress in order to rule out damage or deformation.

*Figure 20 – New interim storage concept*

In order to exclude the appearance of condensation and the resulting risk of corrosion, the interim storage buildings (v) are heated (minimum temperature + 10 ° C) and the air is dehumidified (maximum humidity 60%).

In the design and construction of the storage halls (*Figure 21*) and the storage racks, aspects of earthquake security were included, and the consequences of an airplane crash on the interim storage facility were considered.

The halls themselves are built in steel girder construction with insulated profile panel walls in areas (walls) where no shielding effect is necessary, where shielding is necessary (e.g. towards the outer fence of the NES company premises) the walls consist of up to 60cm reinforced concrete.

*Figure 21 – Interim storage buildings*

The halls are equipped with all facilities for air conditioning (heating, dehumidification), ventilation and monitoring of the room air for radioactive contamination. Since the drums and racks are stacked with forklifts and manipulation of the containers is not expected during the interim storage period, no cranes are installed in the interim storage.

All waste packages are inspected (*Figure 22*) at regular intervals (at least every 5 years). A visual check is carried out, the freedom from contamination is checked by means of a wipe test and, if the drums are tightly closed by a rubber septum, a gas sample is taken (by a random choice process) and analyzed from the inside of the drum (using gas chromatography).

*Figure 22 – Inspection of waste containers*

The documentation of all waste data (e.g. activity inventory, chemical information) as well as all treatment steps carried out from receipt of the waste to interim storage takes place in a specially developed database called DOKURAD. A wide variety of evaluations with regard to interim storage, decay calculations and visualizations of the interim storage can be carried out by DOKURAD (*Figure 23*). This database represents an essential aspect for the subsequent disposal of the waste packages since all relevant data come together here. In addition to electronic storage, the essential data and information are also stored in paper form for each container.

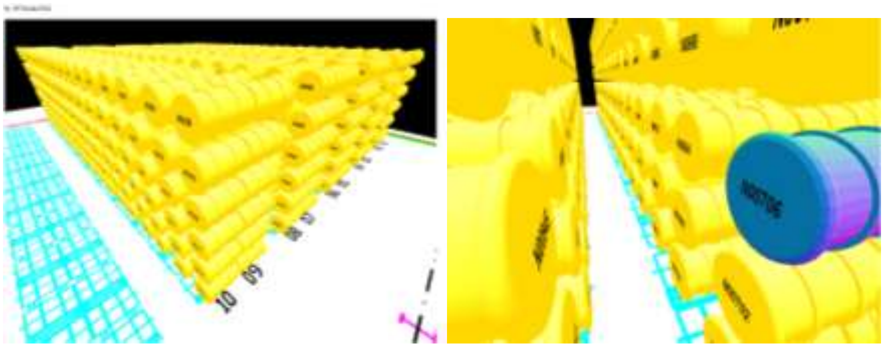


*Figure 23 – 3D-Visualisation of interim storage (DOKURAD)*

In addition to standard containers (200L drums) there are also some special containers in the NES interim storage facility, e.g. these contain bulky waste or waste with a higher dose rate from the decommissioning of the RR in Seibersdorf (completed in 2006).

The NES interim storage facility has a storage capacity that, with some reserves, will last until 2045, the end of the current contract with the Republic of Austria.

<u>Reconditioning old waste containers</u>

Since, as already described above, the older waste containers do not meet the current requirements and, moreover, the documentation is insufficient, a reconditioning project was initiated a few years ago; in the meantime several thousand drums have already been processed. The procedures used depend on the type and condition of the container, but here are some examples of standard procedures:

> **Pellet drums**: Basically, it is a matter of transferring the high-force compacted pellets to new standard drums (*Figure 24*), drying the waste, comprehensive characterization using gamma spectroscopy and updating the documentation.

*Figure 24 – Reconditioning of pellet drum*

- **Homogeneously cemented waste**: Here the old drums are removed in a specially developed drum dismantling plant and the diameter of the concrete cylinder is milled to match the geometry of the new standard drums (*Figure 25* & *Figure 26*). The resulting milling dust is also used for a comprehensive re-characterization of the waste and supplements the data from the gamma measurement of the drums, which is also carried out. The concrete cylinders are lifted into new drums and the documentation is followed up.

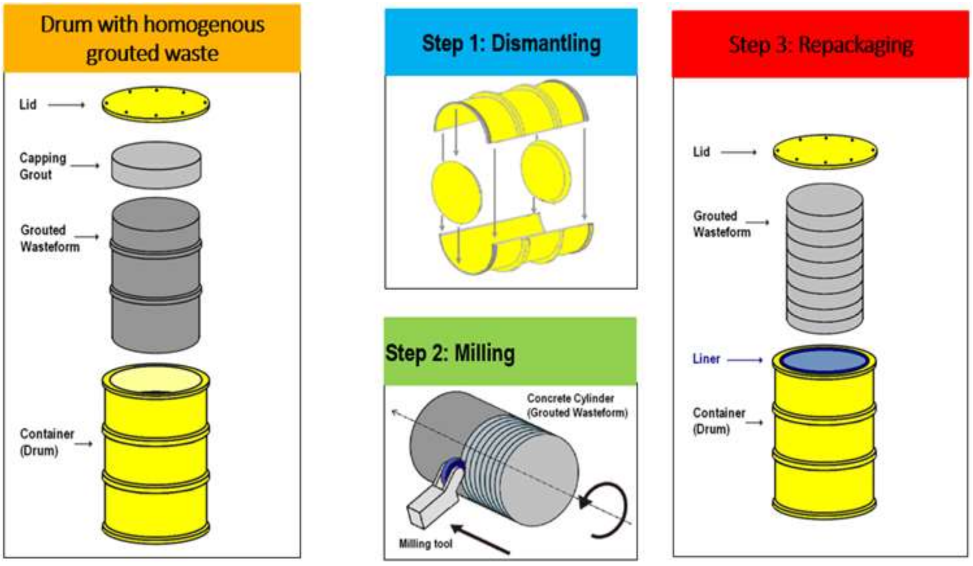



*Figure 25 – Principle of reconditioning of homogeneously cemented drum*

*Figure 26 – Reconditioning of homogeneously cemented drum*

- **Inhomogeneously cemented waste**: In the past, waste was partially filled into 100L drums (partially precompacted) and these drums were placed in 200L drums and enclosed within inactive concrete. Since this type of waste does not meet the current requirements and in particular the acceptance criteria of the NES interim storage facility, these drums have to be completely dismantled and the waste they contain has to be reconditioned using the systems available today (*Figure 27* & *Figure 28*). Combustible waste is fed to the incinerator and non-combustible waste is high force compacted. A comprehensive characterization is also carried out for this type of waste and the documentation is followed up. These works are carried out inside special "caisson", 2 big stainless-steel boxes that are completely equipped for the dissemination of those drums (all the pictures are taken in those boxes). Personnel work with full protective clothing there.

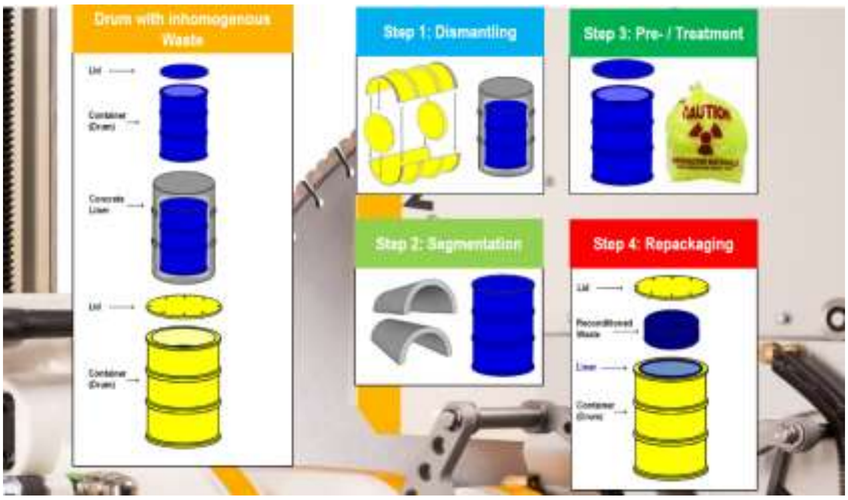



*Figure 27 – Principle of reconditioning of inhomogeneously cemented drum*

*Figure 28 – Reconditioning of inhomogeneously cemented drum*

During reconditioning, the conditioning systems and processes available today lead to a significant reduction in volume and thus to a considerable reduction in the number of containers that have to be brought to the interim storage facility for further storage. In addition, waste that has meanwhile decayed can be sorted out and sent for clearance measurement and clearance as inactive waste (discussions with the responsible supervisory authority are currently in progress on this procedure).

### 4.1.2 Long-term interim storage in the Netherlands – COVRA N.V.

In the Netherlands, the decision to opt for long-term interim storage of radioactive waste was made very early on. One speaks of a period of at least 100 years. Since safe interim storage of the waste is possible with the facilities available at COVRA (*Figure 29*), the technical / scientific progress in the disposal of radioactive waste as well as any multinational solutions can be awaited and, moreover, the amount of waste to be disposed of later can be minimized by the decay during the interim storage period. Due to the difficult geographical / geological situation in the Netherlands (large areas below sea level, very high groundwater level), the high population density and the high level of environmental awareness, the Netherlands relies on a later disposal in a deep GDF if the concept of shared repository is not realised. In this case, other disposal solution could be chosen by the Netherlands.

*Figure 29 – COVRA site*

At COVRA, the national organization for the management of radioactive waste, the waste drums are stored horizontally in the interim storage facility (*Figure 30*) and the individual containers can be inspected. In contrast to NES, however, the waste is already cemented, and the drums can therefore be stored without a drum lid.

*Figure 30 – COVRA LILW interim storage*

The principle of "air conditioning" in the storage halls is identical to NES, so a minimum temperature and a maximum humidity of 60% are maintained with COVRA.

| | **NES** | **COVRA** |
|---|---|---|
| Storage period | until 2045 (according contract) | min. 100y |
| Containers | drums, other containers | drums, other containers |
| Storage concept for drums | horizontal on racks | horizontal on racks |

| | | |
|---|---|---|
| Air conditioning | >10°C, <60% humidity | a minimum temperature, <60% humidity |
| Technical equipment | minimized | minimized |

*Table 10 – Comparison interim storage for LILW*

*Table 10* gives a comparison between NES and COVRA. In contrast to Austria, there is also HLW and spent nuclear fuel in the Netherlands. These are also interim stored at COVRA for a long time, where they rely on dry storage in vaults.

## 4.2 Near Surface Disposal Options

### 4.2.1 Near surface disposal facilities (NSDF)

Introduction

Commonly accepted international management options for LLW include near-surface disposal at or below ground level, or placement in rock cavities below ground level [73] [74] [9] [75]. A range of technical solutions exists for the emplacement of radioactive waste in near surface environments. The selection of a disposal option depends on many factors, both technical and administrative, such as the radioactive waste management policy; national legislative and regulatory requirements; waste origin, characteristics and inventory; climatic conditions and site characteristics; public opinion; etc. [74]. All types of near surface disposal rely on a system of passive engineered and natural barriers to prevent, or very drastically delay, the transport of radionuclides into the environment. The specific features and characteristics of the protective barriers may differ significantly between various facilities. Initially, many facilities started with research on previous facilities such as caves, abandoned mining facilities, below and above ground vaults and bunkers.

#### 4.2.1.1 Disposal of VLLW in France - CIRES Facility

The CIRES facility (Le Centre Industriel de Regroupement, d'Entreposage et de Stockage or Industrial facility (*Figure 31*) for grouping, sorting and disposal), hosts multiple activities. The cross-section of the trench disposal cells of the CIRES Facility is shown in *Figure 32*.

Located in Aube in the municipalities of Morvilliers and La Chaise, the CIRES hosts several activities such as storage, segregation, processing and disposal. The disposal facility opened in 2003 and is managed by ANDRA (French National Agency for Radioactive Waste Management). [76] The management of VLLW in France  comes mainly from the operation or dismantling of nuclear facilities, from conventional industries using radioactivity, and from the remediation of polluted sites.

CIRES started receiving and disposing of VLLW on this site. This category of wastes is managed at CIRES (light metallic and plastic waste can be compacted on-site, to optimize disposed volumes). The waste is prepared on the production sites and packaged, for the most part, in metal boxes or in large plastic bags called "big-bags". Almost all VLLW packages are transported from producers to the CIRES disposal facility by road, following stringent international regulations. About 85% of delivered waste is already treated and packaged for direct disposal. Every year, around 30,000 m$^3$ of waste is disposed of in the trenches dug out in the clay. At the arrival of the wastes to CIRES, measurements are performed to verify the radioactivity level (dose rate and contamination) on transport containers and radiological characteristics of the waste packages. Treatment and conditioning at the CIRES implies that certain waste packages undergo additional conditioning before disposal, that certain plastic or scrap waste need to be compacted to reduce its volume and that sludge and liquid waste must first be solidified.

Since autumn 2012, CIRES has been expanded with two new activities : grouping and temporary storage of non-nuclear waste, or waste produced outside of the nuclear electricity production sector. The grouping building act as a multimodal platform. The waste packages are sorted out and grouped by category before being oriented towards processing, storage or disposal. Two other activities were added in 2015, namely the sorting and treatment of non-nuclear waste. The operations include checking packages using an x-ray scanner, blending and packaging solvent, oil-based or aqueous liquid waste, processing scintillation vials and disassembling lightning conductor heads. The storage building hosts waste packages for which disposal solutions are currently under study.

*Figure 31 – The disposal facility CIRES* [77]

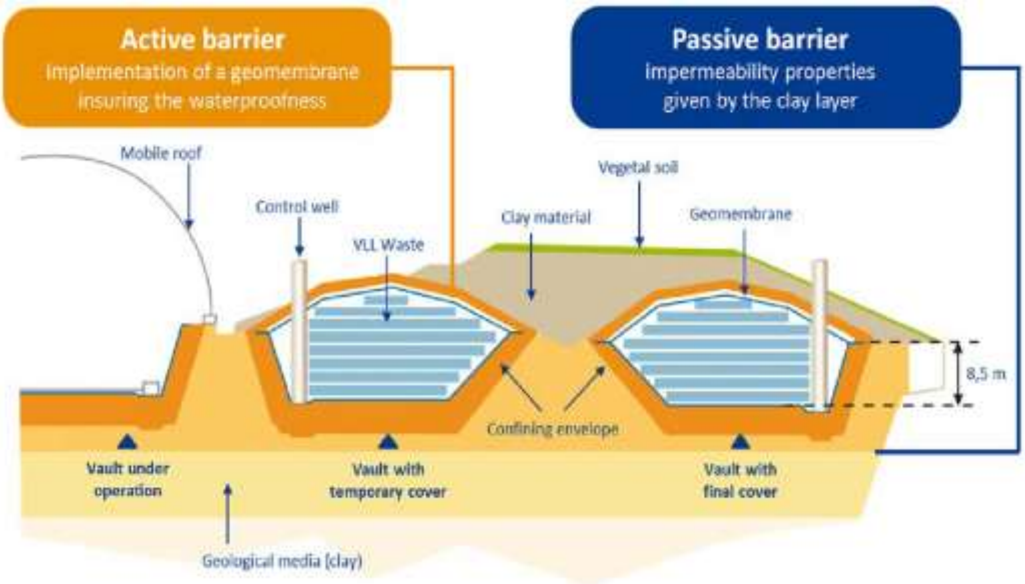



*Figure 32 – Cross-section of the trench disposal cells implemented at CIRES* [78]

#### 4.2.1.1 Disposal of LILW in France - The MANCHE disposal facility (CSM)

*Figure 33 – Centre de la Manche LILW Disposal facility (ANDRA 2011)*

*Figure 34 – Emplacement of wastes at CSM, France.* [79]

At the very top of the Cotentin peninsula in Normandy, ANDRA monitors the first French disposal facility built for LILW [79]. The CSM is the first radioactive waste disposal facility in the world to enter the post-closure monitoring stage (*Figure 33* - *Figure 35*). This disposal is now closed; it has been operating for 25 years and it received a total 527,225 m$^3$ of LILW, mainly containing short-lived radionuclides. Most of the waste emplaced at the CSM comes from nuclear facilities operated by EDF, Orano and the CEA. This waste is the result of operating and maintaining nuclear power plants, research laboratories and from dismantling nuclear facilities. It includes, for example, work clothes worn by employees; filters used in ducts at nuclear facilities, concrete and metal waste. The CSM was the first radioactive waste disposal facility ever operated and closed in France. Its design features were used to help define the major principles applicable to radioactive waste disposal facilities.

Disposal structures

The 25 years of the facility's operation were marked by major changes in disposal design. Following the commissioning, the waste packages were emplaced directly in excavated trenches. This practice was then abandoned. However, one trench containing waste packages was left as it is.

New disposal design methods were implemented providing a waste package complying an adequate level of safety that could be placed on a slab called a "disposal Platform". Concrete packages were first placed along the edge of the structure to create the shape of a pyramid, and then the space in the middle was filled with other packages and gravel to immobilize them. The final disposal structure was therefore in the form of a "tumulus". Waste packages that required additional protection, depending on their activity levels were placed in reinforced disposal structures called "concrete-lined trenches" or "monoliths". Waste packages were placed inside in successive layers. Concrete was poured between each layer to envelop and stabilize the packages.

*Figure 35 – Cap covering the top of the disposal vaults installed between 1991 and 1997* [79]

<u>The cap</u>

The final cap covering the top of the disposal vaults was installed between 1991 and 1997 (*Figure 35*). It is designed to protect the packages and minimize water infiltration. Water run-off is collected and checked before being discharged back into the environment. The cap also protects the waste disposal facility from human, animal or plant (tree root) intrusion. The cap construction operation was a world first. It is made up of alternating drainage/impermeable layers, including a bituminous membrane chosen for its elasticity and its ability to adapt to ground movements. The material is soaked in bitumen to make it watertight. The cap also has a multi-layer drainage system to collect rainwater that would otherwise permeate down through the cap. Water is collected at the CSM through a series of networks (drifts, pipework and drains) including a surface network that collects runoff water from the cap and access roads, a drainage network at the level of the membrane that collects any water that has infiltrated down through the cap and an underground drainage known as the "underground gravity-assisted separation network" that lies beneath the disposal structures at the base of the buried disposal vaults, supporting walls and drifts.

<u>Closing of the facility</u>

The CSM was closed in 1994, and the challenge now is to determine its final form, and especially concerning the final cover. The CSM is the first radioactive waste disposal facility in the world to enter the post-closure monitoring stage. The facility no longer receives waste packages but improvements and adaptations are still being made with a view to the facility's final closure within the next fifty years. ANDRA is developing tools and reports to ensure that the memory of this site is preserved and passed down to future generations.

#### 4.2.1.2 Disposal of LILW in France - The Aube waste disposal facility (CSA)

The CSA is in the department of Aube on the communes of Soulaines-Dhuys and Ville aux Bois, it is located about 200 km southeast of Paris and 60 km east of the city of Troyes. Waste disposed at CSA (*Figure 36* & *Figure 37*) is conditioned in concrete or metal packages. Waste packages are placed in reinforced concrete repository structures of 25 m² and 8 metres high, that are constructed as needed. Once they are filled, the structures are closed with a concrete slab and then sealed with an impermeable coat. At the end of operations, a cap formed mainly of clay will be placed over the structures to ensure long-term waste containment. The CSA waste disposal facility will then be monitored for 300 years. [80]

*Figure 36 - Aube CSA LILW Disposal facility (ANDRA 2011)*

*Figure 37 – Disposal facility at l´Aube, France* [77]

#### 4.2.1.3 pDisposal of radioactive waste in Spain - El Cabril, Córdoba

Empresa Nacional de Residuos Radiactivos SA (Enresa), is responsible for the management of radioactive wastes in Spain as well as for the national inventory with the data received from all waste producers being also responsible for the preparation and management of the National Inventory for Spent Fuel and Radioactive Wastes. The inventory includes all spent nuclear fuel and radioactive waste, as well as estimates of future amounts, including from decommissioning, and information about the siting and amount of spent nuclear fuel and radioactive waste, according to a classification that takes into account the scheduled final management. The final management route is key in the classification of radioactive waste. Although there are many types of radioactive wastes, these wastes are categorized in Spain according to the management facilities authorized for a volume, radiological inventory and specific activity concentration limits, depending on the nature of the different radionuclides present. These categories are LILW, Special Waste and HLW.

LILW-SL, whose activity is mainly due to the presence of radionuclides beta-gamma emitters, with a short-medium disintegration period (below 30 years) and whose content in long-life radionuclides is very low and limited. This group integrates wastes that can be temporary stored, treated, conditioned and finally disposed in the disposal facility for LILW in El Cabril including the class of VLLW.

<u>El Cabril facility</u>

El Cabril (Instalación Nuclear de Almacenamiento de Residuos Radiativos Sólidos de Sierra Albarrana, C.A. El Cabril, located in the province of Córdoba), is the Spanish disposal facility for VLLW, LILW managed (*Figure 38*). In 1988, an application for the construction permit and authorization for commissioning of the expansion of El Cabril was carried out and in 1990 the new facilities were constructed. An extension to El Cabril for VLLW was added in 2007. The El Cabril Centre is a permanent solution for the storage of LILW and VLLW, being the main classes type of radioactive waste produced in Spain (health and industrial sectors, and NPP's that includes contaminated scrap metal and rubble mostly from decommissioning of NPPs). The facility has two areas: Buildings and disposal, which allows for a better performance of activities, facilitating their monitoring and control, and distinguishing the areas with regard to radiation regulations. Licensing of El Cabril is for 165.000 $m^3$ total of radwastes (35.000 $m^3$ for LILW and 130.000 $m^3$ for VLLW).

Building area includes the conditioning buildings, where waste treatment activities take place, and the control room, where operations, monitoring and supervision are carried out. It is also equipped with laboratories for verifying the condition of the waste.

*Figure 38 – General view of El Cabril site for LILW and VLLW.* [81]

Management of LILW Wastes

The disposal system is based primarily on the interposition of engineered barriers and natural barriers that safely confine the deposited materials, ensuring the protection of people and the environment. There are three types of barriers interposed between the environment and the waste: The first barrier, consisting of the conditioned waste and the cask; The second barrier, made up of the engineered structures that house the waste and the third and final barrier, formed by the natural terrain in which the facility is sited and the covering layers placed over the structures once they are full to capacity.

LILW transported to El Cabril (ENRESA is responsible for collecting, inspecting and transporting radioactive waste and the transfer of ownership of materials takes place at the facility where the waste is produced. All waste is fully inspected, both physically and radiologically, at the place of collection in order to verify that it corresponds to the technical specifications ENRESA requires), is directed to the conditioning area or, if already conditioned in drums as it is the case of most of the waste produced by Spanish NPP's, to one of the interim storage areas (*Figure 39* shows LILW disposal area includes north and south platform with 16 and 12 disposal structures respectively).

*Figure 39 – LILW disposal areas* [81]

Radioactive waste from hospitals, research centers and industry are treated in the El Cabril facility. All packages received are deposited in concrete casks and when these ones reach full capacity, they are filled with injected mortar forming concrete blocks whose, after drying, are deposited in the disposal vault. When disposal vaults get full, they are sealed with slab made of a mix of reinforced concrete and waterproofed. When all the structures composed by various sealed vaults in a platform are also full, another sealing takes place with a final cover of multiple waterproofing layers. A final layer of topsoil is put over the ensemble ready to be revegetated. Monitoring and surveillance of the site starts at this point and it will last for 300 years.

Management of VLLW

VLLW arrives at the facility in large sacks, drums or casks and it is directly deposited in the specific disposal structure (*Figure 40*). If treatment is required, it is sent to the corresponding area. Once all structures reach their full capacity, they are also covered with multiple waterproofing layers and topped with a final topsoil layer for revegetation. Monitoring and surveillance of the site starts at this point and it will last for 60 years.

VLLW disposal area includes one platform with 2 disposal structures, from the four planned, that are already finished.

*Figure 40 – VLLW disposal area* [81]

<u>Treatment, conditioning and disposal of radioactive waste at El Cabril</u>

Solid waste - Solid waste is segregated according to its level of contamination and its physicochemical properties. The main objective is to reduce the volume requiring treatment. To this end, decontamination, shredding, crushing and compacting techniques are used. It is subsequently immobilized creating a block made of cement. Organic waste is incinerated to solidify it, and then mortar is used to create blocks from the ashes.

Liquid waste - Liquid waste is segregated according to its aqueous or organic nature. Subsequently, it is treated using physical and chemical methods in order to reduce both contamination and volume. Physical methods include filtration, centrifugation and evaporation. The most common chemical methods are precipitation and ion exchange. Finally, it must be solidified, since this is the safest state for transport and disposal. This involves mixing it evenly with concrete, mortar or cement.

<u>Disposal of LILW and VLLW</u>

At the end of December 2018, 21 of the 28 disposal structures for LILW (*Figure 41*) were full, with 33,602 m$^3$ stored, which represents an occupation of 77.2 %. Regarding the complementary installation for very low activity waste, at the end of December 2018, 15,491 m$^3$ of radioactive waste had been stored.

In 2018, the inventory date of VLLW and LILW disposed in vaults or structures of C.A. El Cabril facility, counted on primary waste packages received on site is shown in *Figure 42*.

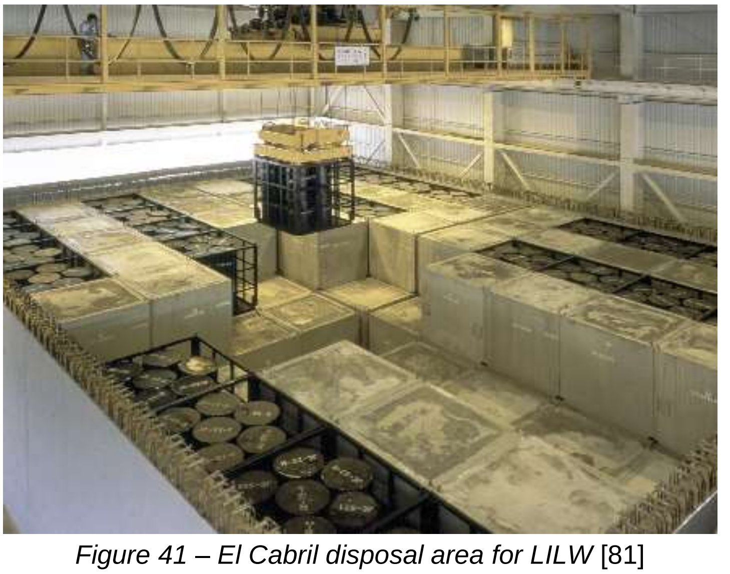


*Figure 41 – El Cabril disposal area for LILW* [81]

| VLL Wastes (m³) | LIL Wastes (m³) |
|---|---|
| 15,491 | 33,602 |

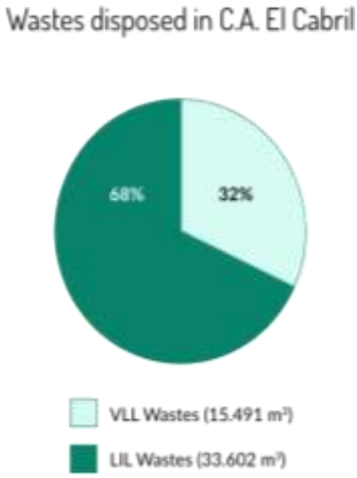



*Figure 42 – Inventory of LILW and VLLW disposed of at El Cabril in 2018.* [82]

### 4.2.2 Cavern and bunker

LLW covers a very broad range of waste. It may include short lived radionuclides at higher levels of activity concentration, as well as long lived radionuclides at relatively low levels of activity concentration and comprises some ninety percent of the volume but only one percent of the radioactivity of all radioactive waste. It ranges from radioactive waste with an activity content level just above that VLLW (waste not requiring shielding or particularly robust containment and isolation), to waste with a level of activity concentration that shielding and more vigorous containment and isolation are necessary (typically suitable for disposal in engineered NSDF) for periods up to several hundred years. A definitive or precise boundary between LLW and ILW cannot be specified in a general manner with respect to activity concentration levels, as limits on the acceptable levels of activity will differ between individual radionuclides or radionuclide groups.

Research has been carried out for a long time [83] about the possibility to use abandoned mines and tunnels for the disposal of radioactive waste such as ILW containing long-lived radionuclides that requires the same rigorous isolation from groundwater in the host rock of repositories as HLW waste [84]. ILW, which does not generate heat, can be disposed of in big caverns in abandoned mines together with LLW [85]. A generalized rock structure model with 1st, 2nd and 3rd order discontinuities in the form of fracture zones and discrete fractures of different size in crystalline rock were important features in studies in the USA, Spain and Switzerland to examine how sufficient isolation can be achieved. Discontinuities of 1st to 3rd orders can cause difficulties in the construction of repositories by having relatively low shear strength, and by controlling regional and local groundwater flow after closing the disposal facility [85].

*Figure 43* shows a sheared zone in crystalline rock illustrating the importance of rock structure for groundwater flow and tunnel stability. The high hydraulic conductivity in the tunnel direction plays a major role for transporting groundwater in the host rock of a disposal facility. The schistosity gives the zone a potential to undergo further splintering and loss in stability if the tunnel is left unfilled for a long time.

*Figure 43 – Large shear strain can give the rock mass strongly anisotropic hydraulic and mechanical properties* [85].

The impossibility to have easy transport and emplacement of the waste in structures such as underground mines and tunnels is due quite often to the need to avoid intersection with unsuitable oriented fracture zones, among other rock factors. According to [85], crossing different systems (parallel deposition tunnels and parallel fractures), creates large channels with high permeable rock zones (*Figure 44*)

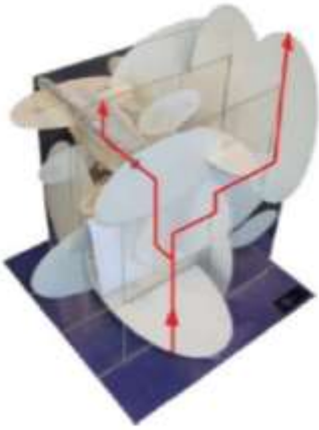


*Figure 44 – Pipe-like channel system resulting from interacting fractures* [85]

These zones resulting from shearing ancient weaknesses can reveal discontinuities of 1st and 2nd order, therefore, not suitable to host disposal of radioactive wastes.

Studies about the prediction of stability and percolation in order to evaluate if an abandoned mine or tunnel can be used for the disposal of radioactive waste needs the definition and application of predictive models. Parameters such as type of mine, topology and geometry, local chemical environment, geology, rock structure, groundwater flow and pressure, besides waste types and characteristics and others, need to be taken into account in the models.

### 4.2.3 Tunnel and galleries

In this subchapter, only tunnels and galleries to the depths at about 100m below the surface are considered. Although most of disposal facilities which belongs to the near surface disposal type for LLW are categorized as trenches, vaults and/ or in deeper horizons as caverns/silos, some examples of disposal options in horizontal tunnels and/or galleries can be found. In most cases, tunnels and galleries serve only as access and service within other type of repositories which ensure higher containment and isolation of the waste.

#### 4.2.3.1 Disposal of radioactive waste in Switzerland - Wellenberg

Low- and intermediate-level waste from the nuclear power plants is processed into a form suitable for disposal either at sites of origin or at ZZL Würenlingen (*Figure 45*). It is packaged into suitable containers and then stored in facilities at the power plants or at ZZL. Two smaller interim storage sites for this waste have been operating since 1993: the government's BZL associated with the Paul Scherrer Institute at Würenlingen, and Zwibez at Beznau, which also has a storage hall for dry cask storage of spent fuel and high-level waste [86].

A proposal for a low- and intermediate-level waste disposal facility at Wellenberg was blocked by a cantonal referendum in 1995. A federal working group reviewed the proposal and recommended in 2000 that it proceeds, though modified to allow for retrieval. A further cantonal referendum blocked it in 2002. The revised Nuclear Energy Act removes the cantonal veto right, but requires a national referendum.

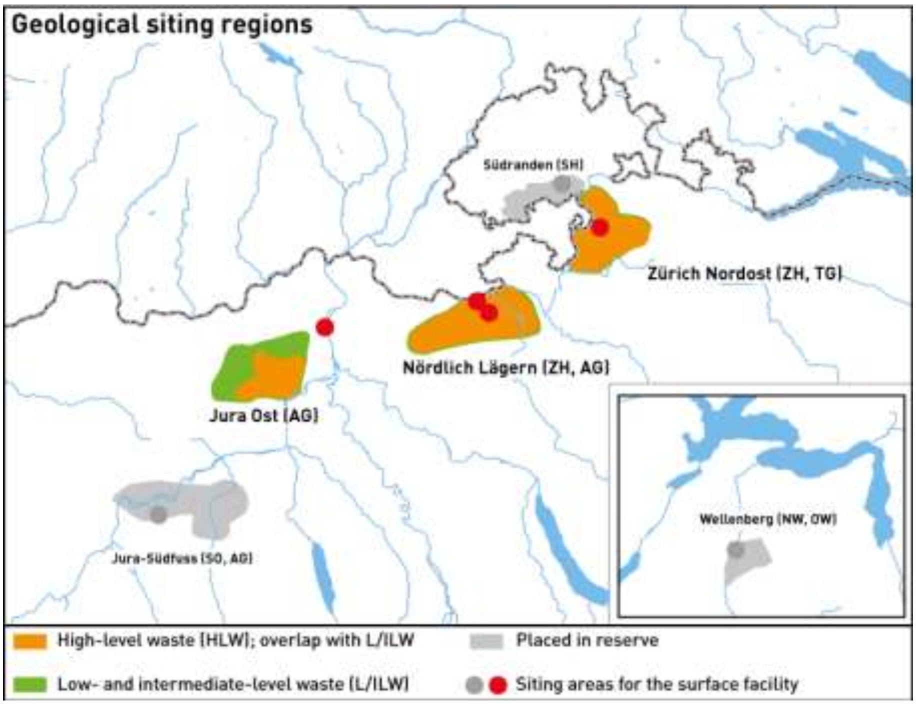



*Figure 45 – The siting regions in grey have been placed in reserve. The siting regions in green and orange will undergo further investigation in stage 3* [87].

Although, the decision made by the Federal Council on November 21st 2018, determined that the siting regions Jura-Südfuss, Südranden and Wellenberg should be placed in reserve; Wellenberg's technical solution intended for this kind of disposal facility (disposal in tunnels) could be adopted for SIMS as an example for their LLW disposal solution.

Some of the most relevant information can be found in the Nagra report TBR 93-26[3]. This report was prepared as part of the documentation justifying the choice of Wellenberg as the proposed site for a low- and intermediate-level waste disposal facility. The aim of the report is to provide information which will serve as a basis for comparing Wellenberg with the other three potential sites which were evaluated, and to document the deliberations and calculations which are part of the safety analysis of Wellenberg as a potential disposal facility site (*Figure 47*) [88].

Another report is part of the supporting documentation for the LILW disposal facility site selection procedure. The aim of the report is to present the site-specific geological data, and the geosphere database derived therefrom, which were used as a basis for evaluating the long-term safety of a disposal facility at Wellenberg. These data are a key component of other reports appearing simultaneously with the present one, first on the intercomparison of the four potential sites (Bois de la Glaive, Oberbauenstock, Piz Pian Grand and Wellenberg; (NTB 93-02[4]) and, second, on the safety assessment of the Wellenberg site itself (NTB 93-26) (*Figure 46*) [89].

[3] d_ntb93-026.pdf (nagra.ch)
[4] https://nagra.ch/wp-content/uploads/2022/08/d_ntb93-002-Textband.pdf

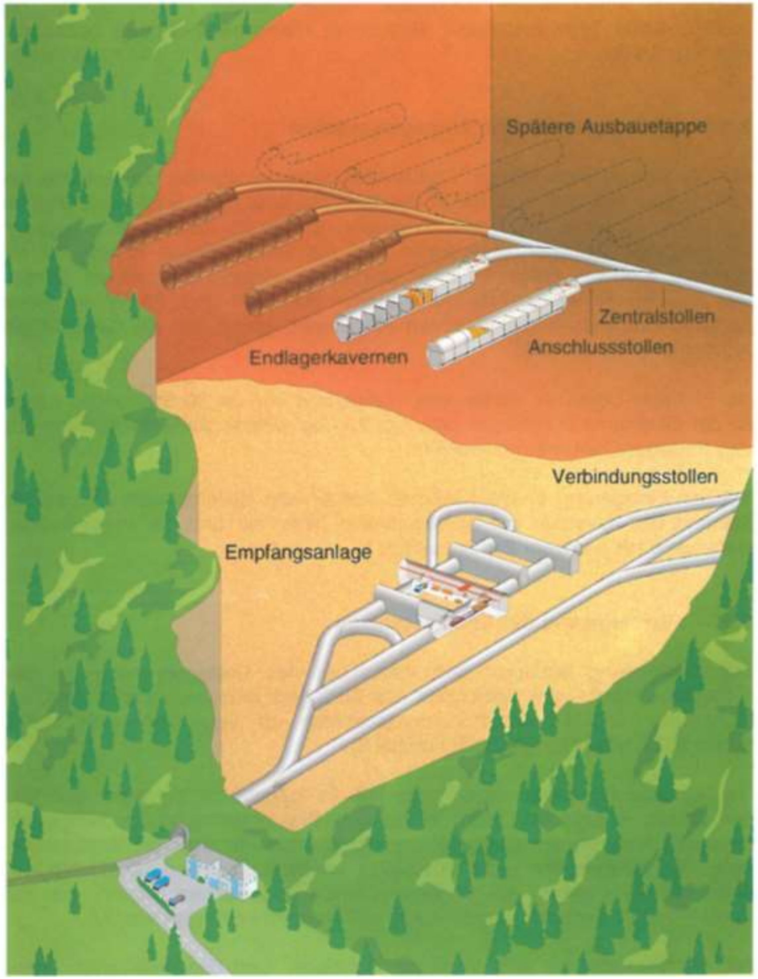



*Figure 47 – Intended disposal facility layout,* [82]

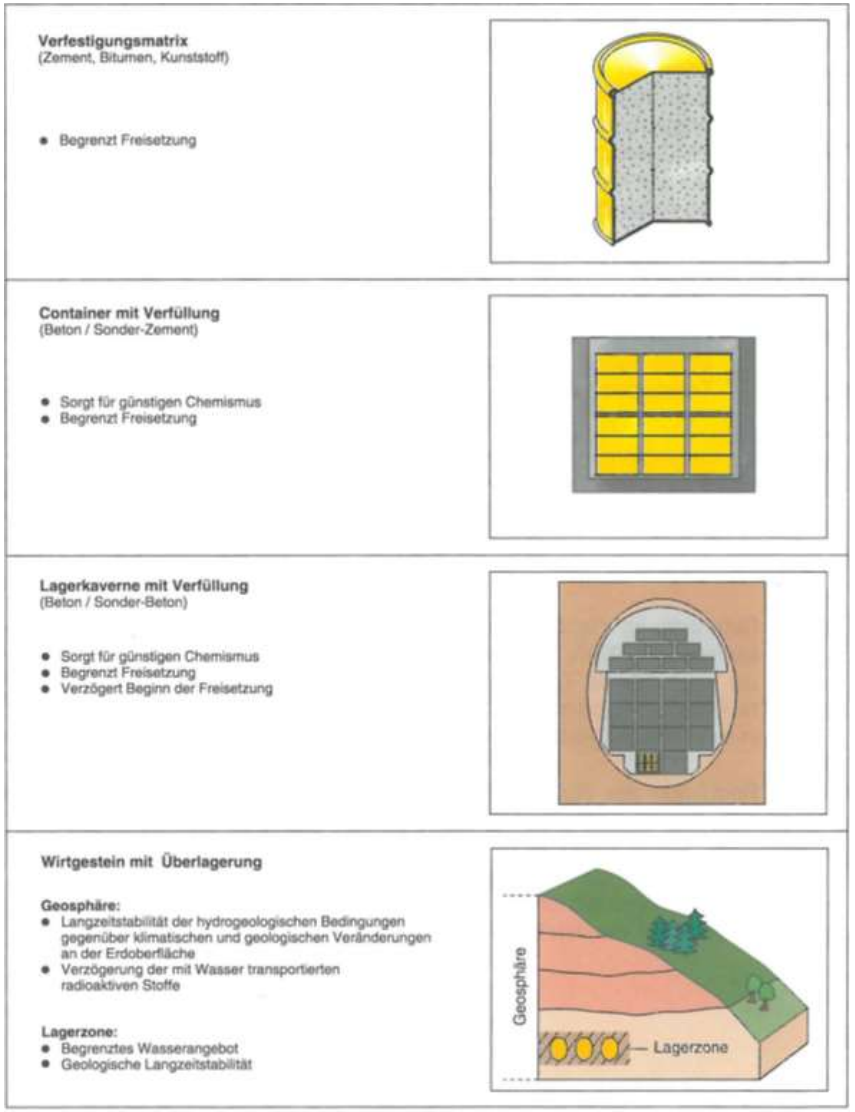



*Figure 46 – System of safety barrier* [83]

#### 4.2.3.2 Disposal of radioactive waste in the Czech Republic – URAO Bratrství

Radioactive Waste Disposal Facility Bratrstvi in Jachymov (*Figure 48*) (*Table 11*) to dispose radioactive waste consisting of or contaminated with natural radionuclides of the radium and thorium series is in operation since 1974. Its closure is planned between 2020 – 2025.

| | |
|---|---|
| **Beginning of operation** | 1972 |
| **Scheduled end of operation** | after 2025 |
| **Repository depth under the surface** | more than 50 m |
| **Total volume adapted for the disposal facility** | 3 600 $m^3$ ( from which 1 200 $m^3$ for disposal) |
| **Filled volume of disposal chambers** | 954 $m^3$ (volume of disposed RAW 381.6 $m^3$) |
| **Free volume** | 2460 $m^3$ in corridors, 0 $m^3$ in disposal rooms |

*Table 11 – Summary data on Radioactive Waste Disposal Facility Bratrstvi (as on December 31, 2019)* [90]

Design and Construction of Disposal Facility Bratrstvi

Radioactive Waste Disposal Facility Bratrstvi in Jachymov is designed to dispose of radioactive waste consisting of or contaminated with natural radionuclides of the uranium and thorium series. The disposal facility was developed particularly to dispose of leaking and DSRS from healthcare facilities.

The Bratrstvi in Jachymov disposal facility has been developed from part of abandoned underground premises of the former uranium mine Bratrstvi.

Two factors are specific for the disposal facility operation:

- High humidity in the underground premises and a substantial flow rate of mine water nearby the disposal chambers,
- High concentration of radon decay products (not generated by the disposed radioactive waste but by natural activity of the host environment) which makes it necessary to maintain a special regime.

The mine work is stable from the geotechnical viewpoint. Based on the extensive prospecting work carried out previously , regular hydrological and geotechnical monitoring of the site was initiated in 1992 in the location and focuses on the safety of the disposal facility in terms of its stability.

A concept has been approved for the disposal facility's decommissioning and closure.

The disposal facility was developed by adaptating a gallery in a former uranium mine, while five chambers were adapted for waste disposal with a total volume of nearly 1200 $m^3$. The disposal facility started operating in 1974. The mine is situated in a water-bearing crystalline complex and therefore a drainage system has been built in the surroundings of the disposal facility area with a central retaining tank and flow-through retaining tanks. The removed water is monitored.

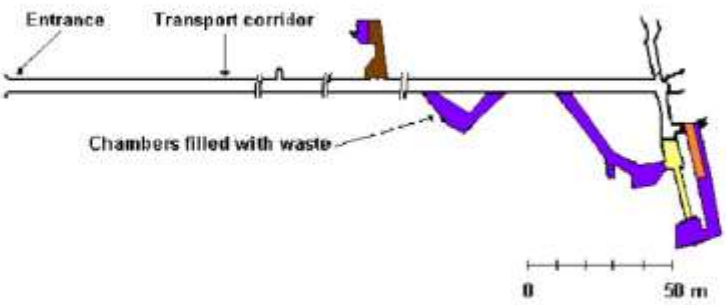



*Figure 48 – Radioactive Waste disposal facility Bratrství - layout* [90]

Safety Assessment of the Disposal Facility Bratrstvi

The safety analyses performed in 2003-2013 aimed to verify the disposal facility capacity and to propose limits and conditions for its operation. The efforts included safety evaluations for options with and without a backfilling material in the disposal facility premises, taking into account the updated information on the source term, including radioactive waste inventory and employment of different types of filling materials, particularly bentonites and materials on cement basis. The safety analyses evaluate individual personal doses in the following scenarios: transport of radionuclides in the disposal facility and underground water in the case of barrier damage, scenario in which a person enters the disposal facility and scenario with a person staying in the location. The transport of radionuclides was considered in two variants - with and without a backfilling material. The scenarios were anticipated to take place after termination of institutional control (i.e., 120 years after the operation of the facility is finished). Individual doses calculated for the real disposal facility system (inventory, construction design, and host rock environment) were compared with applicable limits and the acceptance criteria for radioactive waste in the Bratrství disposal facility have been proposed based on the comparison.

In 2019, a revision of the safety analyses was prepared, including a revision of the hydrogeological model. [90]

Inventory of radioactive waste in the Disposal Facility Bratrstvi

The disposal facility's safety has been assessed using requirements of the Act No. 263/2016 Coll. and its implementing regulations.

The utilization of underground premises for radioactive waste disposal is classified as a special interference in the earth's crust and Decree No. 99/1992 Coll., Decree of the Czech Mining Authority on the establishment, operation, securing and disposal of waste storage facilities in underground spaces establishes basic obligations for its operation. These requirements extend requirements resulting from the Atomic Act, including the following:

- Monitoring of geotechnical parameters of the underground premises,
- Monitoring of airstreams.

A standard container used for radioactive waste disposal has been a "sandwich" disposal unit (100 l drum in 200 l drum, see *Figure 49*) with the volume of 200 l with anticorrosion finish. The drums are laid down flat in layers up to about 2 m.

The radioactive waste disposed in the Bratrství disposal facility is mostly $RaSO_4$ in platinum cases (medical DSRS), Ra-Be neutron DSRS, laboratory waste containing natural radionuclides, depleted uranium and natural thorium (mostly as $Th(NO_3)_4.5H_2O$ and $ThO_2$).

The overall inventory of selected radionuclides disposed of in the disposal facility shall not exceed $1.10^{13}$ Bq of natural radionuclides (see also *Table 12*).

| Radionuclide | Total activity [Bq] |
|---|---|
| $^{226}Ra$ | $1.36x10^{12}$ |
| U | $6.38x10^{11}$ |
| $^{232}Th$ | $3.20x10^{9}$ |

*Table 12 – Inventory of radioactive waste disposal in the URAO Bratrstvi at December 31, 2019)* [90]

Waste form for disposal in the URAO Bratrstvi

Homogeneous and uniform solidified waste containing natural radionuclides have been disposed of at the disposal facility Bratrství. The waste is usually solidified in drums of 210 liters each, using cement as solidification medium or in a sandwich container (100 l drum in 200 l drum) with a non-active cemented interlayer) or in the MOZAIK steel container. WACs are approved by SÚJB in the frame of licensing process.

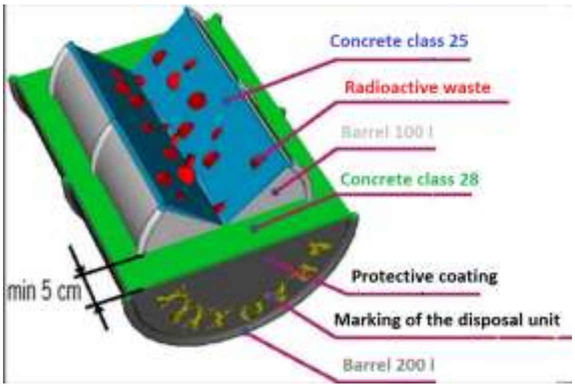



*Figure 49 – Radioactive waste disposal container – "sandwich" disposal unit (author: SÚRAO)*

#### 4.2.4 Borehole

There are several vertical disposal options, based on the design and depth of the facility (see *Figure 50*):

1) Boreholes

- Borehole (depth < 100 m)
- Deep borehole (100 m < depth < 1000 m)
- Very deep borehole (1000 m < depth < 5000 m)

2) Shaft (deep borehole with large diameter)

3) Silo (very short shaft with large diameter)

4) Deep cavern (silo with borehole on top)

5) Converted mine (combination of shaft, tunnel and horizontal cave)

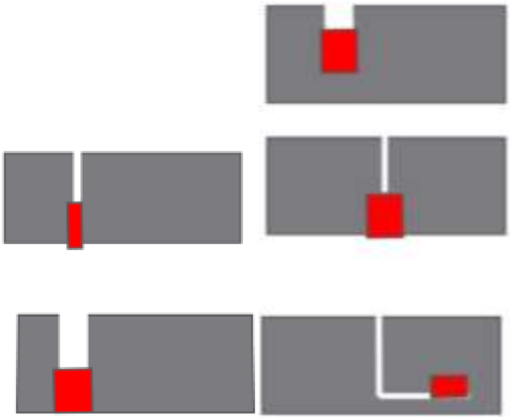


*Figure 50 – borehole (top left), shaft (bottom left), silo (top right), deep cavern (middle right), converted mine (bottom right)* [91].

The borehole disposal concept (*Figure 51* and *Figure 52*) entails the emplacement of DSRS and small volumes of LILW in an engineered facility bored or drilled and operated directly from the surface. Borehole disposal is envisaged mainly as a small-scale activity that can be carried out without a large programme of scientific and site investigation.

Consideration is being given to determining the type of radionuclides and/or waste for which borehole disposal is most likely to be appropriate:

(a) Too long lived for decay storage (e.g. a half-life higher than a few years).

(b) Too long lived and/or too radioactive to be placed in a NSDF.

(c) Small volume waste for which no other disposal facility is cost-effective.

From a safety perspective, borehole disposal is not conceptually different from either near surface disposal or geological disposal of radioactive waste. Indeed, because the range of depths accessed by borehole disposal approaches the depths normally associated with both near surface disposal and geological disposal. Like near surface disposal and geological disposal, a combination of natural barriers and engineered barriers contribute to safety for borehole disposal. In combination, these barriers are designed to contain radioactive material until it has decayed to insignificant levels and to provide sufficient isolation and containment to ensure an adequate level of protection for people and the environment.

Only very minor releases of radionuclides (such as small amounts of gaseous radionuclides) may be expected during pre-disposal activities and during the operation of a borehole disposal facility. The design should ensure that, in the event of an accident involving the breach of a waste package, releases are not likely to have any impact outside the facility. Relevant considerations should include the packaging, the waste form, the radionuclide content of the waste and the control of contamination on packages and equipment [92].

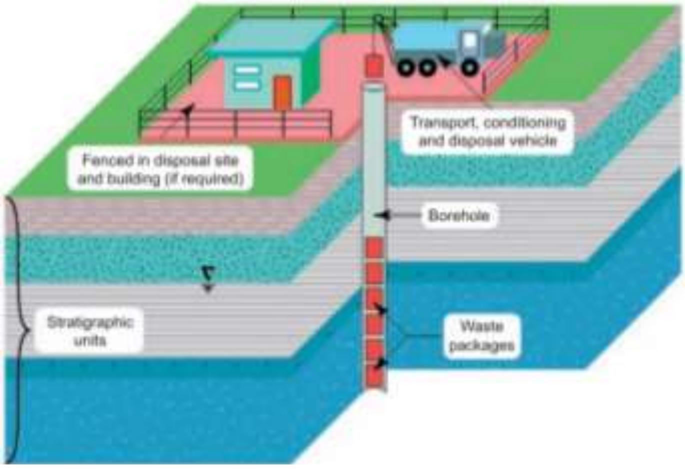



*Figure 51 – Schematic layout of borehole disposal facility* [86]

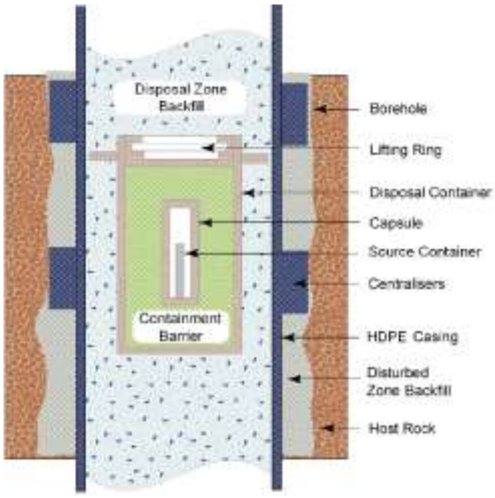



*Figure 52 – Cross-section identifying the multiple barriers in the IAEA borehole disposal concept* [87]

#### 4.2.4.1 Examples of boreholes in Russia and Hungary

*Figure 53* shows a shallow borehole disposal facility in Russia. These boreholes are between 4 m and 40 m deep. Steel disposal compartment at the bottom of the borehole can contain up to $10^{15}$ Bq quantities of DSRS. Such large quantities can generate significant heat. To help with heat dissipation, molten lead is poured into the compartment [93].

Russia is also building deeper boreholes (between 75 m and 200 m) at crystalline rock disposal site Yeniseisky (Krasnoyarsk region, Siberia), but those are intended to hold ILW and HLW, such as SNF [94].

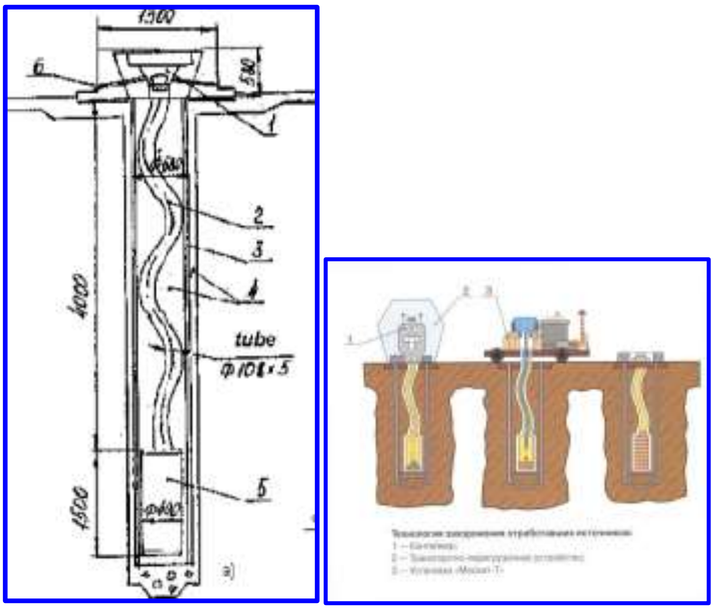



*Figure 53 – Scheme of shallow borehole disposal facility in Russia* [93].

In Hungary, shallow borehole disposal facility is built as part of larger disposal site in the village of Bátaapáti. There are 2 types of boreholes at this site. First one is for SHARS, as seen in *Figure 54* & *Figure 55*. In this case, boreholes are 6 m deep wells, with a diameter of 40 to100 mm. They are lined with stainless steel and located in a concrete monolithic structure [93].


*Figure 54 – Shallow boreholes for SHARS in Hungary* [93].

Second type of shallow boreholes is built for storing long-lived DSRS. It is fairly similar to the first type. It is a 6 m deep well, stainless steel lined, located in concrete monolithic structure, but with larger diameter (200 mm).

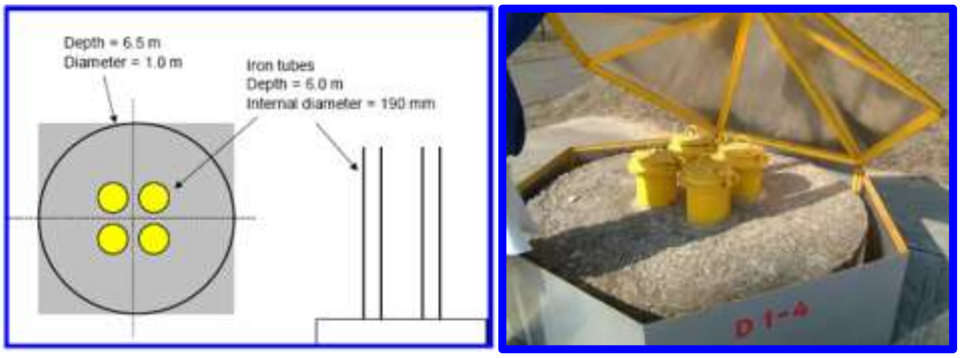



*Figure 55 – Shallow boreholes for DSRS in Hungary* [93].

### 4.2.5 Silos

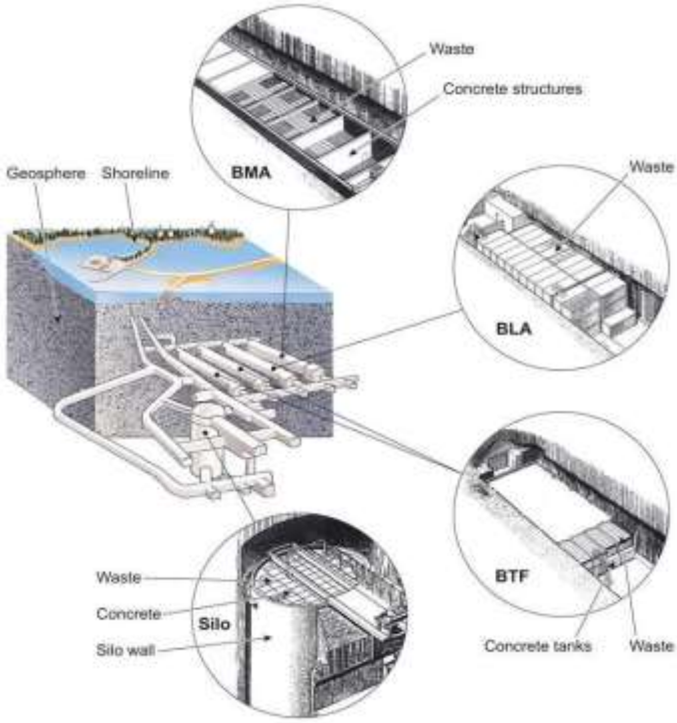
A NSDF, like a silo, is constructed to prevent water from entering the disposed waste and thus ensure that diffusion is the only transport mechanism for radionuclides. Diffusion is an extremely slow process. In such a disposal facility, the waste is emplaced above the groundwater table and the waste stays dry as long as the protective barriers are intact which may be hundreds of years. An advantage with this type of disposal facility is that the requirements on the conditions at the site are moderate and it is therefore normally easy to find a suitable place that conforms to the technical requirements. The disadvantage is that the protective cover and barriers are exposed to weathering, especially erosion that can endanger the integrity of the disposal facility. However, after the institutional control period, , the geohydrological conditions at the site are important.

The impact of processes like weathering and erosion is however lower than for the vault type of disposal facility but a location below or close to the groundwater table may imply higher risk for corrosion and degradation of structures.

Waste can also be emplaced in underground cavities with access through ramps or shafts. In this case the waste is normally placed below the ground water table which means that the environment outside the engineered barriers is water saturated soon after closure of the disposal facility [95].

#### 4.2.5.1 Disposal of LILW in Sweden – Forsmark

Nuclear waste management in Sweden is defined in Nuclear Activities Act (1984) [96]. SFR, the Swedish




final disposal facility (*Figure 56 -*

*Figure 58*) for LILW-SL (it does not have to be cooled and is relatively short-lived), was taken into operation in 1988 [95]. It was the first facility of its type when it was commissioned [97]. It serves as final disposal facility for the operational LILW that is generated from the Swedish nuclear power plants, as well as radioactive waste from medical care, research and industry [95].

SFR was built and is operated by SKB, Swedish Nuclear Fuel and waste Management Company. More than 30 people work at the facility [96].

The disposal facility is located close to the nuclear power plant at Forsmark, 50 metres beneath the seabed in crystalline bedrock. Currently, the area above the disposal facility is covered by the sea. However, the ongoing shore-level displacement (the land rising after the latest glacial period) at the site will lead to substantial changes of the geohydrological conditions and the surface ecosystem during the next coming 10,000 years. This is considered in the safety assessments [95].

The original licence comprised a total waste volume of 63,000 $m^3$ with the possibility to expand the facility. In 2010, SFR had received 33,900 $m^3$ of wastes. About 1,000 $m^3$ of waste is added every year [95].

The facility consists of four 160-metre long rock vaults and a 50-metre high concrete silo which is surrounded by a layer of bentonite. The silo is intended for the ILW with the highest radioactivity that may be deposited in the SFR. Two parallel kilometre-long access tunnels link the facility to the surface [97].

The radioactive waste is kept in various types of waste containers that are protected by one or more barriers. When the SFR is sealed in the future, the disposal facility will slowly fill with groundwater that seeps in from the surrounding rock. The task of the barriers is to slow down and prevent as far as possible any movement of dangerous substances to the surrounding environment. This movement is so slow that most of the radioactivity should disappear before the substances reach daylight [97].

The waste in the silo consists mainly of solidified filter resins classed as ILW and contains the majority of the activity in the facility (it is licensed for 92% of the total activity content in SFR) [95].

In the silo, the waste is kept in containers of steel or concrete which after placing are surrounded by a layer of concrete. The next barrier is the concrete wall of the silo, which is almost one metre thick. Between the outer wall of the silo and the bedrock, there is a thick layer of bentonite clay. After sealing, the clay will swell and prevent groundwater from flowing freely through the silo. It also acts as a filter to slow down and prevent radioactive substances from moving out from the silo. The bentonite also protects the silo from mechanical movements in the rock. The final barrier is the rock in which the disposal facility has been built. This barrier is capable of slowing down the movement of radioactive substances. Even if some of it escapes, its movement should be considerably slower than the movement of the groundwater.

After about 500 years most of the radioactivity will decay. SKB has to show that what is left can be stored safely for 10,000 years. This is done with the help of continuous safety analyses [97].

The Swedish regulations state that the following undertakings must be included in the safety assessment for SFR:

• "The annual risk of harmful effects after closure does not exceed $10^{-6}$ for a representative individual in the group exposed to the greatest risk" resulting in a dose of 1.4 $10^{-5}$ Sv/year.

• Description of the effects on biota.

• Consequences of intrusion.

• More detailed assessment for the first 1,000 years after closure.

• Collective committed dose integrated over 10,000 years from releases during the first 1,000 years.

In the safety assessments, which have been prepared at intervals since 1983, SKB has considered potential impacts through the analysis of several possible developments in the disposal facility. There is an expected "main scenario" where different possible and plausible variants have been taken into consideration. In addition, the authorities require the evaluation of some fewer probable scenarios and so-called "residual scenarios", which are scenarios that are selected and studied independently of probabilities in order to shed light on the importance of individual barriers and barrier functions [95].

*Figure 56 – Forsmark, location of the SFR – Final Disposal facility for Short-lived LILW Radioactive Waste – lies about 50 metres deep in the rock below the sea* [97].

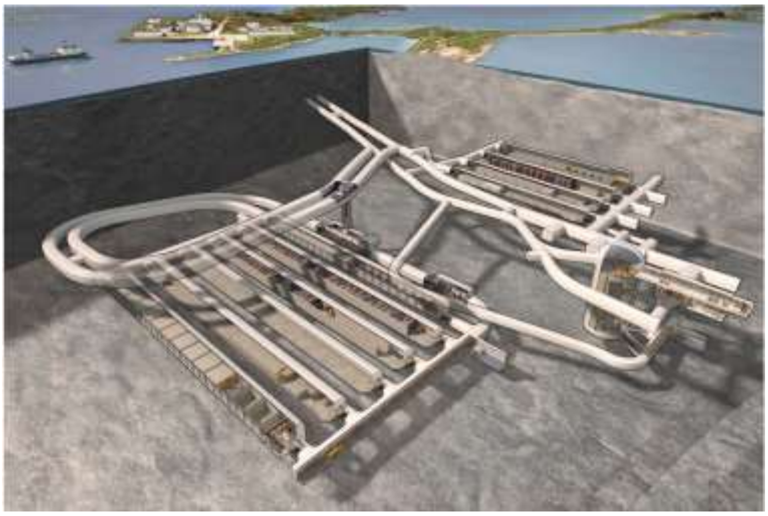


*Figure 57 – Overview of the SFR facility* [96]

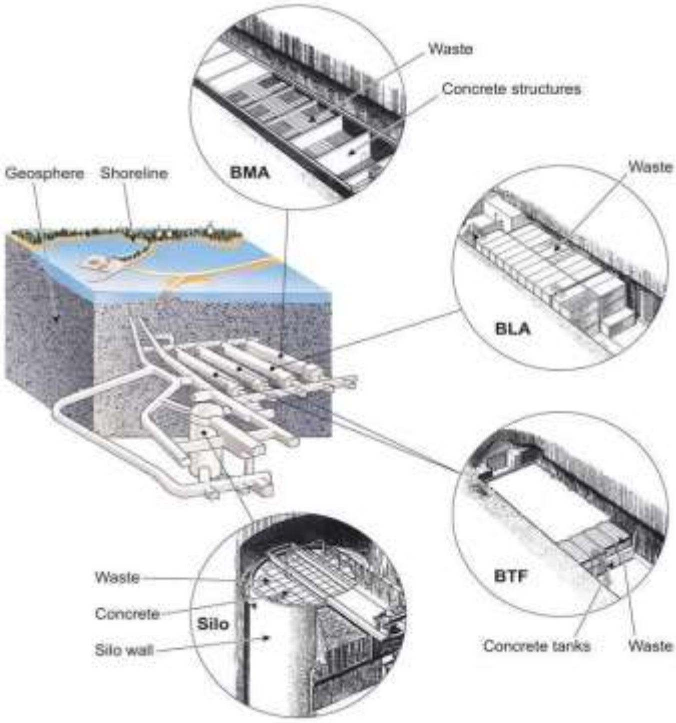



*Figure 58 – Illustration of components that have a safety function in SFR* [95]

#### 4.2.5.2 Disposal of LILW in Finland – Olkiluoto

In Finland, operational LILW has been generated since 1977, when the first NPP was started. The accumulation of LILW from other sources e.g. universities, hospitals, industry etc is only about one percent of that from the NPPs.

The Finnish RWM policy is based on the disposal of LILW into rock cavity repositories located at the NPP sites, Olkiluoto and Loviisa. The design basis of the disposal facility in Olkiluoto, where there are two units, is geological disposal, the safety of which rests on natural and engineered barriers. The disposal system isolates the waste for a few hundred years. Therefore, all LILW generated at Olkiluoto NPPs will be disposed in the on-site facility commissioned in 1992 (VLJ Disposal facility; VLJ is an abbreviation of the Finnish word "voimalaitosjdte": equal to "reactor operating waste"). The disposal facility, as well as the NPP units, is operated by Teollisuuden Voima Oy (TVO) [95].

The disposal facility (*Figure 59* & *Figure 60*) consists of two silos, 24 m in diameter and 34 m high, excavated at a depth of 60–100 meters in the bedrock. The silo for LLW is a rock silo with no internal structure, only shot created walls. The silo for ILW has a reinforced concrete silo building inside the rock silo. The disposal facility is constructed so that its long-term safety is based on several consecutive barriers, engineered barrier system (EBS) and the natural barrier system. The engineered barrier system consists of the waste matrix and the concrete boxes, the silo structures; backfilling material as well as closure and sealing arrangements with the function to retard radionuclides and protect the waste mechanically. The main backfill materials are crushed rock and concrete, which keeps the geochemical changes due to the backfilling moderate. If necessary, the crushed rock can be replaced with sand or moraine, but also in this case the local mineralogical material is to be preferred [95].

The operational low-level waste includes fireproof fabrics, protective plastic sheeting, protective clothing used in power plant maintenance and machine components and pipes removed from the power plant. The LLW is compacted in 200 l steel drum and the drums are packed into concrete boxes. Metal scrap is packed without treatment into concrete boxes. The ion exchange resins used to clean the process water at the power plant are intermediate-level waste. They are mixed with bitumen and cast into drums which are placed in the ILW part of the disposal facility. In addition to the bituminised waste some solid waste in concrete boxes are disposed of in the ILW silo.

The activity inventory for the performance assessment calculations is based on the waste accumulation experience and a reasonable margin, which is introduced to take the related uncertainties and the future unexpected evolution into account. Hence, it does not directly represent the expected activity accumulation of the operational waste but is to be used as a basis for the performance assessment of the final system [95].

At closure, the lower parts of the disposal facility will be backfilled with crushed rock. The gap between the rock and concrete silo will also be filled with crushed rock and the lower part mixed with bentonite. Tunnels and shafts will be plugged and closed with concrete at the ground surface [95].

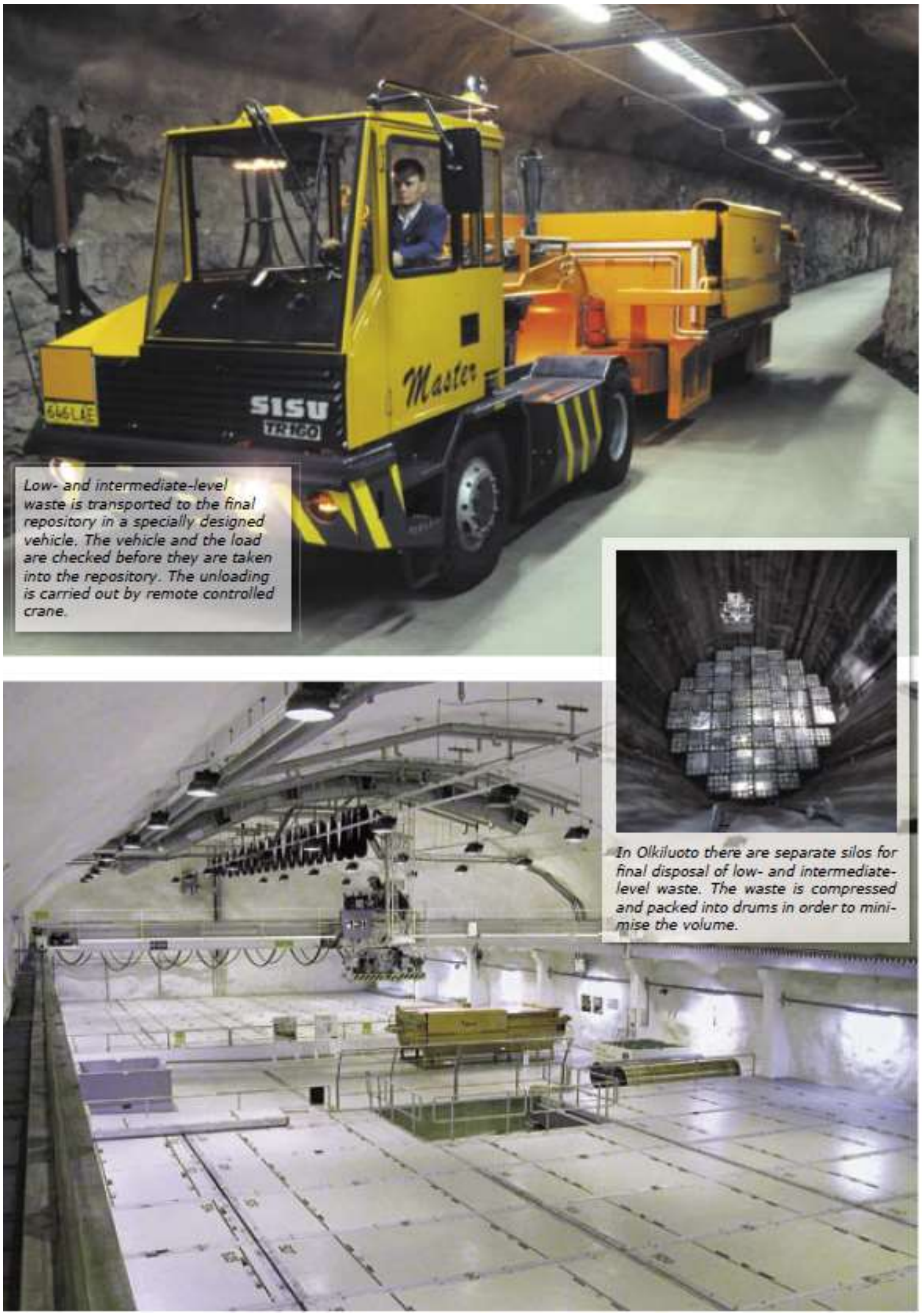



*Figure 59 – The disposal disposal facility for LILW in Olkiluoto* [98]

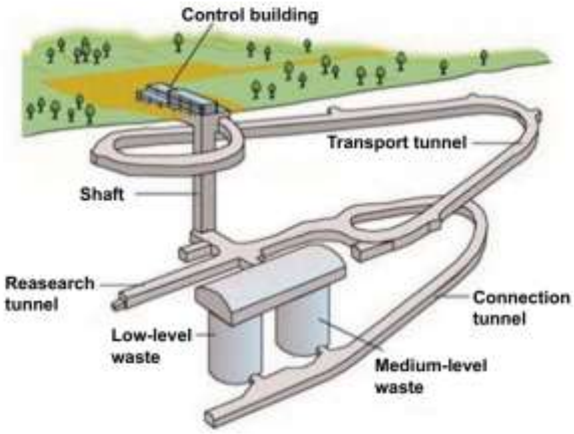



*Figure 60 – Sketch of the TVLJ disposal facility at Olkiluoto* [95]

#### 4.2.5.3 Disposal of LILW in Slovenia – Vrbina

The LILW disposal facility designed by Slovenia will be of the near-surface type, with a disposal unit in the form of a silo and will be built at Vrbina in the Municipality of Krško (see *Figure 61*,*Figure 62*, and *Figure 63*). The type and site of the disposal facility have been approved under the National Spatial Plan. Work will comprise the construction of an administration/service and technological facility, a disposal silo, a hall above the silo, and road and utilities infrastructure. The external areas will be landscaped.

The disposal facility is intended for the disposal of LILW produced at Krško Nuclear Power Plant (Krško NPP) during its operation and its subsequent decommissioning as well as for institutional radioactive waste produced by medical, research and industrial activities in Slovenia.

Construction of the LILW disposal facility will establish the conditions for the reliable long-term operation of Krško NPP, which contributes significantly to the secure supply of electricity in Slovenia, and for the safe, cost-effective and stable use of radioactive sources in science, medicine and industry. The ethical aspect is also important: the generation that derives benefits from the use of nuclear energy should also be responsible for disposing of radioactive waste and should not place an unnecessary burden on future generations [99].

LILW disposal site at Vrbina, Krško is planned as a surface area landfill in the form of an underground silo to be filled from above. The fenced area of the landfill will occupy 10 hectares of land and it will be 650 meters away from Krško NPP. Overhead landfill facilities and top of silo will be built on a flood embankment. Depending on the amount of LILW, it will be possible to build one or more storage silos (if necessary). Initially, one silo will be constructed, which will provide adequate disposal volume for all existing and forecasted waste and support Slovenian disposal needs. Additional silos can be added to account for future waste arising. After closing, the above-ground facilities will be removed.

The safety approach of the facility consists of a combination of natural and engineered barriers (existing geological formations and reinforced concrete structures with specially designed water management and drainage systems). The disposal concept relies on minimizing the access of water to the waste by a combination of measures.

The silos in the Slovenian concept act as very large disposal vaults with direct access to the surface during the operational phase. The silos will be constructed completely below ground. The silo concept is shown in cross-section in *Figure 61*.

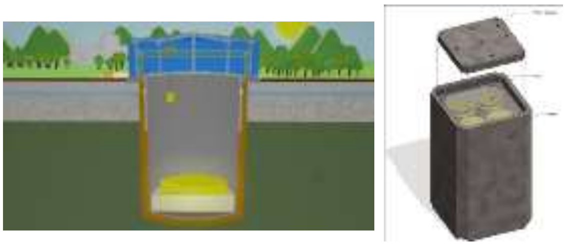


*Figure 61 – Vrbina disposal facility, Slovenia: vertical cross-section of the silo during the operational period and an illustration of a waste container (courtesy of the Agency for Radioactive Waste, Slovenia–ARAO)* [7]

The safety of underground silo will be provided via construction and geological environment. The entire silo will be in layers of silt, which is under a layer of gravel with groundwater. The silt is saturated with water, but the water in it moves a million times slower than in gravel. That's why it is a suitable natural barrier that insulates the disposal site and prevents the escape of radionuclides into the environment.

The dimensions of the silo are sufficient to deposit 10 layers with 99 LILW storage containers (each installed in the silo with a crane). Intermediate spaces between the containers will be constantly filled with filler, and every other layer will be covered with a concrete slab. The concrete plate above the last, tenth layer will be one meter thick. The space above it will be almost to the surface filled with clay to prevent water seepage from above.

Project has started in 2005 with environmental studies, the construction was scheduled to start in 2020 and the site should be in operation from 2025 to 2050 (25 years total) [100].

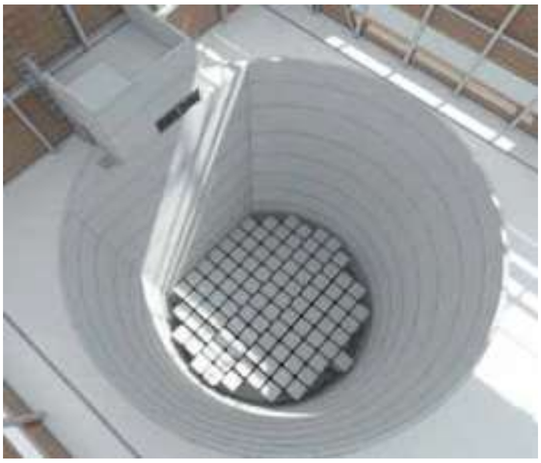


*Figure 62 – View of the silo from a bird's eye view* [100]

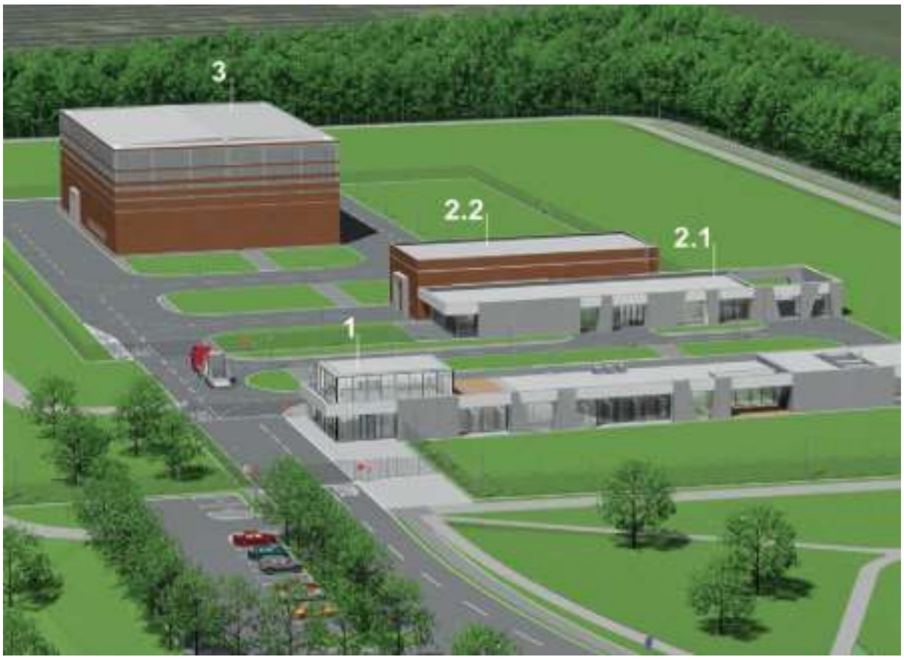



*Figure 63 – Above-service facilities: 1. administrative-service facility, 2. technological facilities (2.1 and 2.2), 3. hall above the storage silo* [100]

## 4.3 Geological Disposal Options

### 4.3.1 Converted mine

#### 4.3.1.1 Geological disposal facilities in Czech Republic – URAO Richard

Radioactive Waste Disposal Facility Richard (URAO Richard)

Radioactive Waste Disposal Facility Richard in Litoměřice to disposal of institutional waste; in operation since 1964. Planned reconstruction to extend the capacity 2018-2020, Disposal to 2100 (see also *Table 13*).

| | |
|---|---|
| **Beginning of operation** | 1964 |
| **Scheduled end of operation** | Not before 2025 |
| **Repository depth under the surface** | 70 - 90 m |
| **Total volume adapted for the disposal facility** | 18 900 $m^3$ |
| **Filled volume of disposal chambers** | 8 201 $m^3$ |
| **Free volume** | 2 047 $m^3$ |
| **Access tunnel and other corridors (including that to Richard I)** | 8 652 $m^3$ |

*Table 13 – Summary data on Radioactive Waste Disposal Facility Richard (as on December 31, 2019)* [90]

Design and Construction of Disposal Facility Richard

The Richard Disposal facility (*Figure 64* & *Figure 65*), in the North of the Czech Republic, is a near surface underground facility for LILW of institutional origin.

Radioactive Waste Disposal Facility Richard is designed to dispose of radioactive waste containing artificial radionuclides. The disposal facility is situated on the north-western edge of the Litoměřice cadaster area under the Bidnice hill. There used to be three limestone quarries in the location (now called Richard I - III) and there was an underground factory construction during World War II. Limestone had been quarried there until 1960s by a company called "Čižkovicke cementarny a vapenky". In the early 1960s the mine work Richard II was identified as a potential disposal facility for LLW.

The disposal facility is situated in a carbonate bank, with overlying and underlying clayey rocks. URAO Richard has been developed in a complex of the former limestone mine Richard II (inside the Bidnice hill - 70 m under the ground level). Its communication corridor is 6 – 8 m wide and 4 – 5 m tall. Individual disposal chambers are accessible from the corridor. Since 1964, the disposal facility has been used to dispose of institutional waste (radioactive waste from utilization of radioisotopes in medical care, industry and research). The mine premises and disposal rooms are dry. The only leakage of underground water in the disposal facility premises occurs in the entrance portal and from ventilation chutes. Additional water gets into the disposal facility by condensation of water from forced ventilation. The seeping and condensing water in the disposal facility are drained into the mine drainage system. The mine water from the Richard disposal facility (in orders of tenths of liters per second) is drained through a system of retaining tanks into a public sewerage system. The mine water is monitored before it is discharged into the sewerage system. Moreover, 13 drills have been made in the Disposal Facility Richard to monitor hydrogeological conditions in the concerned area, 9 of which for monitoring purposes and the remaining ones for prospecting purposes. From the geotechnical viewpoint the mine can be considered as stable.

Based on the extensive prospecting work carried out previously, regular hydrological and geotechnical monitoring of the site was initiated in 1992 in the location and focuses on the safety of the disposal facility in terms of its stability

A concept has been approved for the disposal facility's closure and decommissioning.

The total volume of adapted underground premises exceeds 17,000 $m^3$, while the capacity for waste disposal is about a half of the volume and the rest are service galleries.

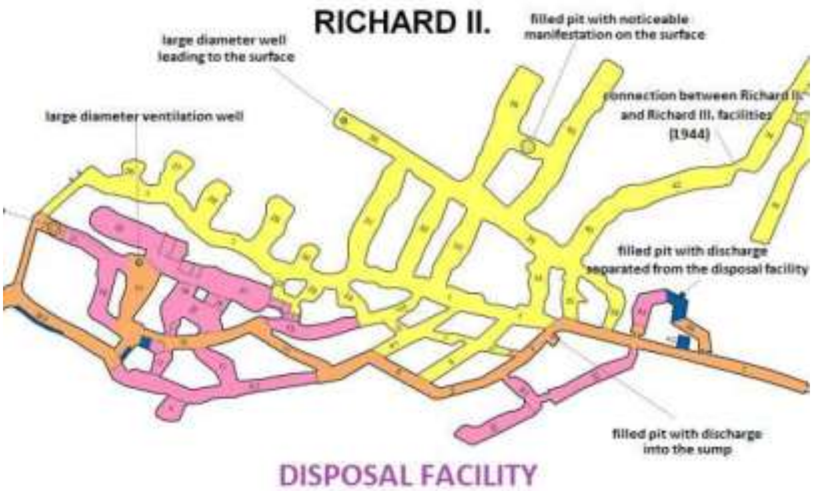



*Figure 64 – Rad. Waste Disposal Facility Richard layout (disposal chambers marked in violet)* [90]

<u>Safety Assessment of Radioactive Waste Disposal Facility Richard</u>

A revision of safety analyses for Radioactive Waste Disposal Facility Richard was prepared in 2016, which is a continuation of safety analyses, and their revisions performed in 1995 – 2013, and it has been used as a supporting document for the application for a license to operate the disposal facility.

The safety analyses performed in 2003-2016 were supposed to verify the disposal facility capacity and to reassess the already proposed closure and decommissioning method. The efforts included safety evaluations for options with and without a backfilling material in the disposal facility premises, taking into account the updated information on the source term, including radioactive waste inventory and employment of different types of filling materials, particularly bentonites and materials on cement basis. The transport model has been updated using data from the newly made drill holes to further specify hydrogeological data in the location.

Safety analyses evaluate the individual doses received by people in the following scenarios:

- transport of radionuclides in the disposal facility and underground water in the case of barriers damage,
- people enter the disposal facility,
- people stay in the location.

The transport of radionuclides was considered in two variants - with and without a backfilling material. The scenarios were anticipated to take place after termination of institutional control, i.e. 300 years after the operation of a facility is finished. Individual doses calculated for the real disposal facility system (inventory, construction design, host rock environment) were compared with applicable limits and the acceptance criteria for radioactive waste in the disposal facility Richard Litoměřice have been proposed based on the comparison.

In 2019, a revision of safety analyses was prepared, including a revision of the hydrogeological model. Furthermore, the new safety analysis will consider the optional extention of disposal capacity to previously unused premises in the northern part of the disposal facility.

<u>Inventory of radioactive waste in the Disposal Facility Richard</u>

This disposal facility is used to mainly dispose of institutional radioactive waste containing artificial radionuclides. Separately from disposed waste, there is also waste that does not comply with WAC for disposal and are waiting to be disposed of in a respective disposal facility. They mainly include DSRS, collected radionuclide sources from fire detectors and nuclear materials.

The maximum observed effective dose of a worker due to radon inhalation was 2.29 mSv in 2019. The effective dose is also evaluated for the contractor's workers who perform work on the first stage of the reconstruction of ÚRAO Richard and show a much longer period of stay in controlled zone compared to the normal operation of the disposal facility.

In connection with the limits and conditions for safe operation, a verification is performed of electric equipment operability, forklift truck operability, pass ability of the drainage system and operability of the instrumentation. Since the beginning of the operation, radioactive waste has been always disposed of in agreement with the WAC valid in the given period. When disposing of the waste, the operator checks for:

- damage of the container,
- surface contamination of the container,
- dose rate on the container surface,
- content of radionuclides.

The individual containers are placed in disposal rooms. Individual containers are disposed of to maximize utilization of the space in the rooms, up to 5 layers (from the viewpoint of strength capacity up to 8 layers may be stacked without damage of the bottom layer of the casks).

In addition to the monitoring of parameters important from a radiation protection perspective, also basic climatic and hydrological data and geotechnical parameters are measured in the location.

Radioactive waste in which the content of radionuclides exceeds the acceptance criteria for disposal is, in accordance with the limits and conditions for storage of radioactive waste, stored separately from the disposed radioactive waste (they include particularly radionuclides $^{60}Co$, $^{137}Cs$, $^{241}Am$, $^{238}Pu$ and $^{239}Pu$) (see *Table 15* & *Table 14*).

<u>Waste Form for Disposal in the Richard Radioactive Waste Disposal Facility</u>

Homogeneous and uniform solidified waste and a non-solidified institutional waste containing artificial radionuclides have been disposed of at the disposal facility Richard. Waste is usually solidified in drums each of 210 litres using cement as solidification media, in a sandwich container (100 l drum in 200 l drum) with a non-active cemented interlayer) or in the MOZAIK steel container. WAC are approved by SÚJB in the frame of licensing process.

| Radionuclide | Total activity [Bq] |
|---|---|
| $^{3}H$ | $3.09 \times 10^{13}$ |
| $^{14}C$ | $1.05 \times 10^{13}$ |
| $^{36}Cl$ | $9.05 \times 10^{9}$ |
| $^{90}Sr$ | $7.68 \times 10^{12}$ |
| $^{99}Tc$ | $4.34 \times 10^{9}$ |
| $^{129}I$ | $2.13 \times 10^{7}$ |
| $^{137}Cs$ | $3.31 \times 10^{14}$ |
| $^{239}Pu$ | $3.78 \times 10^{12}$ |
| $^{241}Am$ | $1.03 \times 10^{13}$ |
| Total activity of other radionuclides α | $1.11 \times 10^{12}$ |

*Table 15 – Inventory of radioactive waste disposed in the URAO Richard recalculated as at December 31, 2019 (incl. disposed DSRS)* [84]

| Radionuclide | Total activity [Bq] |
|---|---|
| $^{3}H$ | $3.87 \times 10^{7}$ |
| $^{14}C$ | $6.17 \times 10^{7}$ |
| $^{36}Cl$ | 0 |
| $^{90}Sr$ | $2.40 \times 10^{11}$ |
| $^{99}Tc$ | 0 |
| $^{129}I$ | $1.21 \times 10^{4}$ |
| $^{137}Cs$ | $4.19 \times 10^{14}$ |
| $^{239}Pu$ | $9.12 \times 101^{2}$ |
| $^{241}Am$ | $9.08 \times 10^{12}$ |
| Total activity of other radionuclides α | $3.49 \times 10^{11}$ |

*Table 14 – Inventory of radioactive waste stored in the URAO Richard as at December 31, 2019 (incl. disposed DSRS)* [84]

*Figure 65 – A view down the operating URAO Richard radioactive waste disposal facility and disposal chambers* [90]

#### 4.3.1.2 Geological disposal in Germany – Konrad

History

The ore deposit of the Konrad mine was discovered in 1933 at a depth below 800 meters during the search for crude oil. The Second World War interrupted the exploration of the ore deposit, which began in the 1940s and was not completed until the 1950s. Shaft Konrad 1 reached its final depth of 1,232 m at the beginning of 1960 after a construction period of almost two and a half years. The construction of Shaft Konrad 2 (*Figure 66*) then began around 1.8 km southeast. The actual ore mining started at the beginning of 1964 and ended in 1976, as the mining at the Konrad site was no longer commercial.

From 1976 to 1982, the mine was investigated for its suitability as a possible disposal facility for LILW (according to the licence: radioactive waste with negligible heat generation) due to its favourable geology. After the investigations indicated that the mine was suitable as a disposal facility, the planning started. In 1982, licensing procedures were started, in which more than 70 authorities and nature conservation organisations submitted their comments. The complete planning documents were submitted to the licensing authority in 1989 and were available to the public for two months. This resulted in about 290,000 objections, which were discussed in 75 days of negotiations until March 1993.

In May 2002, the approval authority approved the Konrad mine for the disposal of radioactive waste after a complete review of all documents and after taking into account the objections and other requirements. Eight lawsuits were filed against the decision by municipalities, districts, churches and private individuals. In March 2006, the court in Lüneburg dismissed the claims and did not allow an appeal. The Federal Administrative Court finally confirmed the decision in March 2007.

About 30 years passed between the completion of the mining operation and the final approval as a disposal facility, although the suitability test on geological disposal was started one year after the end of ore mining. Only the approval procedure last about 20 years.

Expansion to a disposal facility

In 2007, the detailed design for the construction of the disposal facility started. During the construction of the disposal facility, the current legal requirements had to be implemented and the planning documents, some of which were more than 20 years old, had to be processed. Substantial parts of the required infrastructure had to be built and prerequisites for the awarding of construction contracts had to be created.

In the end of 2009, underground construction work started. The completion date is delayed and based on an external expert opinion, construction work is expected to be completed in the first half of 2027. In addition to the time which was required to approve the mine as a disposal facility, another 20 years will pass until the Konrad disposal facility is completed and operational. Among other aspects, new suitable cavities have to be excavated for the expansion to a disposal facility [101].

*Figure 66 – Extension of the emplacement transport section as a tunnel tube (connection of the Konrad 2 shaft with the emplacement chambers)* [102]

Purpose

Depending on the deposit and the mining method, cavities, which are suitable for final disposal, rarely arise during ore mining. Therefore, in the context of production mining, no cavities for final disposal were generated during mining in Konrad.

At the Konrad disposal facility, solid and solidified radioactive waste with negligible heat generation (low and intermediate active waste), which were produced in the peaceful use of nuclear energy, must be stored.

The cumulative volume of waste packages to be finally stored by the year 2080 amounts to about 303,000 m$^3$. The cavities must be excavated for the corresponding volume of radioactive waste.

The work at the Konrad mine allows for the excavation of up to 1.1 million m$^3$ for disposal cavities, corresponding to a waste package volume of up to 650,000 m$^3$. This maximum possible waste package volume suitable for final disposal has been used by the applicant as a basis for the mining planning and technical design of the plant. The licensing authority set the limitation unilaterally to 303,000 m$^3$.

Primarily after production mining, the suitability of the deposit for final disposal is checked. Furthermore, an inventory of the cavities must be made in order to assess whether the mine is lucrative for a disposal facility. [103]

Concept for final disposal

Old mining chambers are not available for final disposal. A final disposal of waste in sections outside of disposal chambers is not planned either, especially since underground cavities for the planned disposal volume have not been available at the Konrad mine so far. Cavities actually required for the disposal facility will be excavated successively and according to demand.

The hoisting devices of Konrad 1 shaft are going to be rebuilt. The hoisting equipment at shaft Konrad 2 will be renewed and designed for waste packages up to 20 tons. The waste containers and the muck will be conveyed and transported in separate sections and shafts.

In field 1, pre-prepared sections are extended to disposal chambers and are adapted to the geological conditions, the technical and safety requirements of the operation. Due to adequate dimensioning of the fixed and anchor support structures, they are stable even over longer operating periods. The debris resulting from the cavity expansion is partly used for backfilling of cavities, for the remaining backfilling of the mine workings and for the final structures.

The need-based and successive driving of the emplacement cavities enables the separation of the driving operation from the emplacement operation, so that a safety-related impairment of the underground waste container transport and the driving is not given. In addition, wastewater flows from the emplacement areas do not affect the operating points occupied by personnel. The facilities for conveying, transporting and handling waste packages are designed for two-shift operation with 6,800 transport units per year. The maximum mass of a transport unit is 20 t.

After completion of the emplacement of the radioactive waste and the residual filling of the disposal facility, the shafts are backfilled and sealed. [103]

Package concept

Various standardised containers or round casks made of concrete or cast materials are used as containers for final disposal. The permissible total weight of a container is 20 Mg. The containers and round casks are equipped with corresponding load attachment points (e.g. ISO corners castings) or transport systems are available to quickly store the various containers..

Direct storage of drums is not permitted. However, drums can be used as inner containers in the containers or round casks.

The standardised containers, round casks, and their respective radiological limits are specified in the WAC for the KONRAD disposal facility [104].

The waste packages have to fulfil the requirements of the WAC regarding the properties of the waste product, the packaging and the activity limitation. The requirements for the packaging and the waste product aim at limiting the activity release from the containers, avoiding chemical reactions of the waste product and ensuring the mechanical stability required for handling. The requirements derived from the investigation of the accident analyses for mechanical and thermal loads on the waste packages lead to a limitation of the activity contents of the waste packages.

Compliance with the requirements from the final disposal conditions is checked within the so-called product control. This includes general requirements for waste packages as well as specific requirements for waste products and waste packages and activity limitations for individual radionuclides.

The radiological limits for each package depend on the package type itself and on the waste contained or its conditioning. Basis for the derivation of the limit values are accident analyses and long-term safety analyses (protection of the biosphere).

The general requirements for waste packages include requirements regarding the permissible local dose rates and surface contamination as well as pressureless delivery to the disposal facility. For the use of fixatives, additional basic requirements have to be fulfilled.

Regarding transport, handling and stacking in the disposal facility, the residual cavities in the waste container must be filled as completely as possible and mass distributions in the waste containers must be as uniform as possible.

In addition, requirements for documentation and the delivery of waste packages are also included in the WAC.

According to the Atomic Energy Act, both conventional protection goals (e.g. surface, groundwater) and radiological goals (e.g. protection against radionuclide input to personnel and the biosphere) are defined. To fulfil the radiological objectives, technical barriers are needed in addition to the closure by geotechnical and natural barriers.

These technical barriers are defined in the form of requirements for suitable casks for the containment of nuclear waste.

Mine buildings and emplacement areas

The mining work at the Konrad disposal facility consist of the day shafts Konrad 1 and 2, the roadways, spirals and ramps, the emplacement chambers and the mine rooms. From above ground, the ore deposit is accessed through the two shafts Konrad 1 and 2. Shaft Konrad 1 is located in the northern part of the mine field; it has a depth of approx. 1,232 m and is designed as a main haulage, man way and material shaft. Shaft Konrad 2 is located in the eastern part of the mine workings; it has a depth of approx. 998 m and is mainly used as a ventilation and shaft for the emplacement of the radioactive waste. Both shafts have a clear diameter of 7 m.

During the conversion of the shaft system to a disposal facility and during operation, parts of the mining work will be excavated anew, other parts will be discarded, whereby the development and the layout of the mine workings will essentially be preserved. For the emplacement of radioactive waste, max. nine emplacement fields can be excavated as required. These fields are developed from six main levels and in which max. 59 emplacement chambers with a maximum emplacement cavity of approx. 1,150,000

$m^3$ can be developed. The emplacement chambers will all be newly excavated and are located in a part of the deposit that has not yet been exploited. Existing mine workings will be used during operation but not for emplacement.

The mine workings of the Konrad mine allow for the additional excavation of up to 1.1 million $m^3$ of disposal cavity. According to the project, this volume allows the emplacement of a waste package volume of up to 650,000 $m^3$.

Consequently, twice as much disposal cavity volume has to be planned for each emplaced waste package volume. [103]

Backfilling of the emplacement chambers

After emplacement, the remaining cavities are backfilled section by section. In sections of approx. 50 m, the emplacement cavity is closed with shotcrete. The shotcrete is applied layer by layer with a specially designed vehicle by spraying the container stack wall starting from the bottom, filling the remaining cavities in the section profile with shotcrete and thus building up a concave curved shotcrete wall. After the shotcrete has set, the cavity filled with containers in the emplacement section behind the wall is backfilled via pipelines running through the shotcrete wall. The thick matter used as backfill material consists of approx. 70 wt.% of prepared heap material, approx. 10 wt.% of cement with additives and approx. 20 wt.% of water.

This sealing technology will be used to meet the protection goals. The backfilling and sealing concept guarantee the safe storage of the waste and a maintenance-free post-operational phase. [103]

#### 4.3.1.3 Geological disposal in Germany – Morsleben

The Morsleben disposal facility for radioactive waste (ERAM) was established in 1971 in the former potash and rock salt mine Bartensleben. Between 1971 and 1991 and from 1994 to 1998, a total amount of about 37,000 cubic metres of LILW was disposed. In addition, radioactive waste was temporarily stored. Today the disposal facility is in decommissioning phase.

In this region, salt mining began over a hundred years ago. The sinking of Marie shaft began in May 1897 and ended in August 1898 at a depth of 370 metres. The Bartensleben shaft was sunk from 1910 to 1912. The mine workings of Bartensleben and Marie are connected at several points.

After the Second World War, chickens were fattened in the Marie shaft complex (1959 to 1984) and toxic chemical waste was temporarily stored (1987 to 1996).

This was executed in parallel with the storage of LILW in the Bartensleben mine (from 1971).

History

The former Germen Democratic Republic (DDR) started to operate nuclear power plants in the mid-1960s. Therefore, a disposal facility for radioactive waste was required. The highly radioactive waste was returned to the Soviet Union. The selection of a disposal facility for LILW was made after an assessment of ten other mines. Three of these were shortlisted, including the shafts "Bartensleben" (Morsleben) and "Marie" (Beendorf). The decision in favour of Morsleben as the site of the later "Central Disposal facility Grube Bartensleben" (ZEGB) was made in 1965.

Apart from the disposal facility medium salt, important criteria were the size of the available cavities and the mine's soon usability. In 1971, the Bartensleben mine in Morsleben was licensed as a disposal facility. The first radioactive waste was emplaced within the framework of a second licence on a trial basis from 1971 onwards. *Figure 67* shows the construction of the main building during the transformation phase.

*Figure 67 – Construction of the main building in 1977 during the transformation of the production mine into a disposal facility from 1974 to 1978* [105]

Later, the disposal facility was permanently licensed and prepared for emplacement under a so-called permanent operating licence. A three-year trial operation started in 1978. After the licence for limited continuous operation in 1981, the licence for unlimited continuous operation (permanent operating licence) was issued in 1986. Today, this permanent operating licence is still the essential basis for the operation of the disposal facility, which will be kept open until its final decommissioning.

As a result of the unification treaty, the Morsleben disposal facility was transferred to the area of responsibility of the Federal Republic of Germany. Legal disputes repeatedly determined the further emplacement operation, which only ran for a few years from 1994 onwards. The Higher Administrative Court of Magdeburg stopped the emplacement in one part of the disposal facility (Ostfeld) in 1998.

In 2001, after a re-evaluation of the site, the operator irrevocably waives the acceptance and final disposal of further radioactive waste in Morsleben. In 2005, the operator submitted the "Decommissioning Plan" with all legally required documents to the competent licensing authority.

In the years 2003 to 2011, the legacy of the past was clearly demonstrated at the Morsleben disposal facility. In the central part of the Bartensleben mine, stabilisation activities were necessary. 27 mines were backfilled with special concrete (salt concrete) in order to ensure a permanent decommissioning capability. Without these activities, the progressive deformation of the rock will cause long-term damage to the water-impermeable cap rock between disposal facility and the overburden [105].

Concept for final disposal

During the conversion of the mine into a disposal facility between 1974 and 1978, the mine was designed for an emplacement operation for radioactive waste. The radioactive waste was to be emplaced in several emplacement fields. The disposal facility Morsleben was equipped with:

- a controlled area with the two sub-areas above and below ground,
- the associated monitoring facilities,
- a special sewerage system,
- object security in a triple barrier system including associated technical facilities,
- buildings for administration, social and technical facilities for supply and disposal,
- technical facilities for navigation, dewatering, air supply and supply of auxiliary and operating materials. [106]

Mine buildings

The mine buildings of the Bartensleben and Marie shafts are accessible from the surface through the two shafts "Bartensleben" and "Marie" and are connected to each other by connecting passages on the 2nd and 3rd levels. The emplacement areas are located in the Bartensleben mine workings. During emplacement operation, Bartesleben shaft served as a disposal facility shaft with all conveying, supply and disposal functions. Shaft Marie served and still serves only as escape and ventilation route for the disposal facility and has supply and disposal functions for the Marie mine area.

The existing mine workings (with the exception of the shafts above the salt deposit) extend for a total of about 5.6 km. The largest transverse extension is about 1.7 km and is located in the Bartesleben mine. The volume of cavities excavated amounts to about 8.7 million m³ of which about 6.2 million m³ are open cavities. Of this total, approx. 5.2 million m³ are located in the Bartensleben mines and approx. 1.1 million m³ in the Marie mines . [107]

Radioactive wastes in the disposal facility Morsleben

During emplacement operation, almost 37,000 m³ of LILW were emplaced. About 60 percent of the currently emplaced waste volume was disposed of underground after German reunification. This waste contains 40 percent of the radioactivity emplaced. The radioactive waste mainly originates from the operation of nuclear power plants and from the decommissioning of nuclear facilities. Typical wastes are evaporated and solidified radioactive liquids, filters, scrap metal, paper, laboratory waste, construction waste, sludge or mixed waste.

In general, the radioactive waste was packed in standardised containers, e.g. 200 to 570-litre drums and cylindrical concrete containers. The standardised containers were made of non-combustible material and were protected against corrosion on the inside and outside. Certain types of waste such as bulky waste or filters were stored in approved special packaging. Only a very small proportion of the waste was emplaced unpackaged.

Some of the liquid concentrates were directly dumped and then solidified with lignite filter ash. This solidification only succeeded partially, which is the reason for the presence of a sump on the 7th level.

From 1971 to February 1991, a total amount of 14,432 m³ of radioactive waste was disposed of in the disposal facility. This can be divided into:

- 6,174 m³ solid radioactive waste,
- 8,258 m³ liquid waste,
- 6,223 pieces of DSRS.

In the Morsleben disposal facility, radioactive waste with a total activity of about 1.8 $10^{14}$ Bq was disposed of from 1971 to 1991.

After resumption of emplacement operation, comparatively small quantities of waste were disposed of in the first year (1364.40 m³). From about mid-1994, there was a gradual increase in the monthly and thus also the annual quantities of waste disposed of. In 1995, the volume of waste disposed of was 4326.35 m³. In 1996 and 1997, the quantities disposed of were 5471.02 m³ and 6081.30 m³ respectively. Most recently, the volume of waste disposed of in 1998 fell to 5077.30 m³. In summary, from 1994 to 1998 a total volume of 22,320.37 m³ radioactive waste was emplaced.

During this period, radioactive waste with a total activity of approx. 9.1 $10^{13}$ Bq was emplaced in the disposal facility. Of this total activity, 8.1 $10^{10}$ Bq was attributable to the activity of alpha emitters. [108] [109] [110]

Emplacement areas

The radioactive waste in the disposal facility is located about 480 metres below the surface in the vicinity of the 4th level (bottom) of the Bartensleben mine.

The following emplacement areas exist:

- North field: In the north field, LLW with a total volume of 1,701 m³ was disposed of using the stack technique between 1971 and 1981.
- West field: With a waste volume of 18,637 m³, most of the waste is stored here. Solid LLW was stacked here from 1974 to 1991 and from 1994 to 1998. Even today, smaller quantities of company-owned radioactive waste are still stored in the West Field (*Figure 69*).
- South field: In the South field, waste with a total volume of 10,119 m³ was emplaced from 1978 to 1991 and from 1994 to 1998. The solid and liquid LILW and DSRS were mainly emplaced, but between 1978 and 1991 liquid LLW was also solidified on site, i.e. directly in the disposal area.

- Central section: Between 1983 and 1990, LLW with a total volume of 157 m³ was emplaced in the central section. The solid and liquid waste and DSRS were stacked, solidified on site or dumped.
- East field: After German reunification, the East field was used for emplacement between 1997 and 1998. 6,140 m³ of solid LLW were stacked here (*Figure 68*).

*Figure 69 – Waste in the west field on the 4th level of the Morsleben disposal facility* [106]

*Figure 68 – Emplacement cavity in the east field: Disposal of radioactive waste between 1994 and 1998. (Coverage of waste with salt)* [106]

Emplacement technology

In the disposal facility, the following emplacement techniques have been used depending on the type of waste:

- Solid LLW was stacked in drums or in cylindrical concrete containers in emplacement cavities on the 4th level. The waste was dumped on the 4a level. Filled emplacement cavities were backfilled with lignite filter ash or salt breeze and closed or covered. A wall was set to secure the waste in the North Field but was not covered due to the licensing situation at that time.
- Solid LILW and DSRS were dumped in the South Field from the 4th level via locks into underlying cavities of the 5a level. After completion of the emplacement, the remaining cavity was filled with salt breeze and then sealed with steel lids or plates.
- Also in the south field from the 4th level, liquid LLW was mainly deposited in a mine and solidified on site with lignite ash.

DSRS and small amounts of solid ILW were packed into seven special containers (steel cylinders) with a volume of 4 litres each. These special containers were emplaced in boreholes on the ground for interim storage. Furthermore, a concrete container with Ra-226 wastes is temporarily stored in a specially prepared cavity in the east field. [108] [109] [110]

Decommissioning of the disposal facility Morsleben

The aim of the decommissioning concept is to keep radioactive waste away from the environment as long as possible. In doing so, the legally prescribed protection goals for the release of radioactivity have to be ensured for a period of one million years. This is verified in a long-term safety analysis. The elaboration of the decommissioning concept is accompanied by extensive investigation programmes, which have to provide both mathematical and structural evidence for the planned operations.

The decommissioning concept provides three central actions:

1. extensive backfilling of the mine,
2. the construction of sealing structures and
3. the closure of the Marie and Bartensleben shafts.

The extensive backfilling of the mine should stabilise the disposal facility and the overlying rock and reduce the risk of water ingress. Should water nevertheless penetrate the disposal facility, sealing structures should separate the waste from the rest of the mine and additionally isolate it from the environment. Finally, the shaft seals should also ensure that no gases or solutions can enter or escape into or from the disposal facility for long periods of time.

The decommissioning of the mine was applied in a nuclear licensing procedure. In 1997, the licensing procedure was limited to the decommissioning of the disposal facility, and in 2005, all legally required documents were submitted to the competent licensing authority with the "Plan for Decommissioning". A decision is still pending.

Only the extensive backfilling of the mine workings with seals in the vicinity of the emplacement areas and shaft seals fulfil the actual nuclear and mining law requirements for decommissioning. Accordingly, this concept was applied for planning approval. The possibility of retrieval of the radioactive waste was also evaluated several times. Since retrieval would be associated with an additional radiation exposure that can be avoided if the waste remains, this alternative would not lead to a qualified safety gain according to the present state of knowledge. [110] [107] [111]

<u>Risks</u>

The hazards of the Morsleben disposal facility can be summarised in the following points:

- Danger of collapse and water inflows: The danger of collapse and water inflows were already known in 1969, i.e. before the first licence was granted.
- Lack of closure from the biosphere: There is no closed cap rock above the disposal facility, the requirement for the thickness of the cap rock is not fulfilled.
- Gas formation: Considerable gas formation is to be expected in the disposal facility after decommissioning. Therefore, large cavities are to be kept free. However, the resulting pressure may drive radioactive liquids upwards faster than expected due to new gaps and cracks.
- In view of the complicated geological conditions and the previous use of the shaft Bartensleben/shaft Marie mine, the computer simulations for the demonstration of long-term safety were based on insufficient information.
- Failed in-situ solidification: During the operating period in the DDR, liquid waste should be solidified underground when it is brought to the disposal facility. However, this did not work to the desired extent. Since then, liquid waste has been vagabonding through the mine.
- Fate of unlicensed materials: A radium drum and the DSRS whose permanent emplacement is not covered by the licence to be simply left in Morsleben. A retrieval would halve the emplacement activity. [112]

#### 4.3.1.4 Geological disposal in Germany – Asse

In the Asse II mine, potash salt was mined from 1909 to 1925 and rock salt from 1916 to 1964. On the 31st of March 1964, salt mining was stopped. In 1965, the decommissioned shaft was purchased by the Federal Government and, after a first test phase, was used for the emplacement of radioactive waste - mainly from plants of today's nuclear power plant operators. From 1967 to 1978, a total of 125,787 drums and casks containing LILW were emplaced in the Asse II mine.

<u>History</u>

Around 1900, salt mining began on the Asse mountain range north of the Harz Mountains. In the course of time a total of three shafts were built. The Asse II mine is the only one that is still accessible today.

Asse I (the mine had to be abandoned in July 1906 due to water ingress) and Asse III (the shaft was drilled from 1911 to 1921, work on it was stopped in 1924) have been "flooded" (i.e. filled with water) for decades.

In the Asse II mine the mining of potash salt began in 1909. But mining was stopped again as early as 1925 for economic reasons. By then the miners had extracted around 1 million $m^3$ of the salt rock. The moist residues produced during the production of potash fertiliser were returned to the mine to backfill the mining chambers.

In 1916 the mining of rock salt in the southern flank began. The miners worked their way up from the 750 m level to the 490 m level. In this way, 131 mining chambers were created on 13 levels by 1964. In this time, 3.75 million m³ of rock salt were extracted. In the central part of the mine, additional 450,000 m³ of so-called Staßfurt rock salt were mined from 1927 onwards.

Since hardly no residues remain from the rock salt extraction, the created cavities were left unfilled.

In 1965, the Federal Ministry for Scientific Research and Technology commissioned the Gesellschaft für Strahlenforschung to investigate the final disposal of radioactive waste in the decommissioned production mine. After corresponding modifications, the experimental emplacement of radioactive waste began in 1967.

From 1971 onwards, Asse II was de facto no longer used as an experimental emplacement facility but as a disposal facility in order to dispose the majority of the LILW of the Federal Republic of Germany here. By 1978, a total of 125,787 drums and containers with LILW had been emplaced in the mine.

The radioactive waste was emplaced in 13 chambers: Ten are located in the southern flank of Asse II at a depth of 750 metres and two in the central section at depths of 750 and 725 metres. From 1972 to 1977, ILW was emplaced in a chamber located at a depth of 511 metres.

Disposal ended in 1978 after the Atomic Energy Act was amended in 1976. As a prerequisite for the final disposal of radioactive waste, a plan approval procedure under nuclear law was now prescribed, but the legal basis for the operation of the Asse II mine remained mining law. At that time, there was no decommissioning concept for the time after disposal. According to today's law, a decommissioning concept with long-term safety demonstration is an essential prerequisite for a nuclear licence for a disposal facility. [113]

Geological Situation

The Asse mountain (*Figure 70*) range consists of different Zechstein, a salt rock that was formed about 230 million years ago. It is chracterisitc for Salt rocks to rise in weak zones of the mountain range (e.g. fault zones) in case of larger overlaps with cover layers.

In the case of the Asse, the layers were folded up to form a saddle below the younger overburden, which is why they are referred to as a saddle structure). The core of this saddle consists of the older Staßfurt rock salt (dark blue). Above it is the potash seam Staßfurt (carnallite), also a salt rock (pink). Further layers are the younger Leine rock salt (violet) and anhydrite salts (light blue), a mixture of rock salt and the mineral anhydrite.

The overlying and secondary mountain ranges surrounding the salt structure consist mainly of various layers of red sandstone (clay, sandstone and limestone) as well as shell limestone and the Keuper rock, which was formed later in the earth's history. Directly above the centre of the saddle, the rock layers are interrupted and shifted against each other: The overburden here has fallen away. In the south-western flank of the saddle (left part of the illustration) there are also numerous faults, which can be imagined as cracks in the rock. They are shown as narrow dotted lines in the illustration.

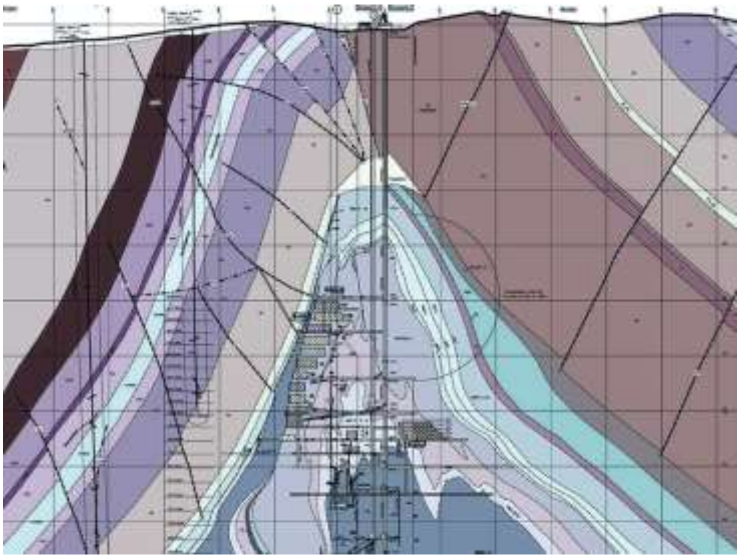


*Figure 70 – Cross-section of the mountain* [114]

During salt mining, numerous mining chambers were created in the Asse II mine, which are located close together on the south-west flank of the mine. In order to give away as little of the raw material as possible, the salt deposit was mined with small distances between the mining chambers and partly up to the surrounding mountains. In some places the mining chambers in the salt layers reach up to five metres from the foothills. The distances between the mining chambers are twelve metres and the intermediate ceilings (teeter) are sometimes only six metres thick.

Today, the high excavation ratio and the proximity of the former mining chambers to the adjoining rock cause the biggest problem in the Asse. Fissures have formed through which salt-saturated groundwater can flow, in the area of the salt structure. Since at least 1988, these waters have been penetrating the mine in the upper part of the southern flank, at a depth of about 500 to 575 metres.

In addition, the mining chambers themselves become unstable due to the movement of the rock (*Figure 71*). In some cases, the intermediate ceilings (teeter) between the mining chambers have already collapsed. The movements of the mountain can create new fissures (cracks) at any time, through which additional water could penetrate. In extreme cases, the inflow of groundwater from the overburden could increase to such an extent that the mine could fill up before the end of the decommissioning measures and thus groundwater could come into contact with radioactive waste.

As a former salt mine, the Asse was not planned from the beginning as a disposal facility for radioactive waste. The original objective was to exploit the salt deposits in the Asse as effectively as possible. In the process, mining chambers were created which reach right up to the outermost edge of the salt layer. Today, the high excavation ratio and the proximity of the former mining chamber to the surrounding mountains cause solution inflows and stability problems. [114]

*Figure 71 – Loose rocks and drip points in mining chamber 5 at the 700-metre level (1993) and arching of the ground in mining chamber 6 at the 553-metre level (1967)* [114]

<u>Radioactive wastes in the disposal facility Asse II</u>

From 1967 to 1978, 125,787 packages with LILW waste were emplaced in 13 chambers of the Asse II mine. Two chambers are located in the central part and ten in the southern flank of the mine at a depth of 725 to 750 metres below ground. Only ILW was emplaced in one chamber at a depth of 511 m from 1972 to 1977. Most of the waste comes from the operation of nuclear facilities and, to a lesser extent, from the use of radioactive materials in research, industry and medicine. Typical wastes are filters, scrap metal, liquids, sludges or mixed waste.

On the basis of documents, it can be reconstructed how many casks are in the Asse II. However, there are uncertainties whether the radionuclide and material inventory of the emplaced radioactive waste is correctly indicated in the documents. The waste declaration at that time does not meet today's standards and is partly incomplete and incorrect. In its retrieval plans, the Federal Agency for Final Disposal ("Bundesgesellschaft für Endlagerung") assumes that even incorrectly declared waste is stored in the Asse II mine. Since the plant is under nuclear law, considerable efforts have been made to eliminate uncertainties in waste documentation. Even after very extensive examinations, there is no evidence that HLW is stored at Asse.

The total inventory amounts to 125,787 packages. The packages are divided into:

- 1,293 packs with ILW
- 14,779 packages of LLW in "lost concrete shielding" (VBA),
- 109,715 packs with LLW

More than 25 % of the packages originate from the last storage year 1978, at a time when the end of storage was already foreseeable. [115] [116] [117]

Emplacement technology

The radioactive waste was emplaced in 13 former mining chambers. Two chambers are located in the central part and ten in the southern flank of the mine at a depth of 725 and 750 metres. Another chamber with ILW is located at the 511-metre level.

At the beginning of emplacement, the waste containers are stacked upright (*Figure 73*). In order to make better use of the hollow space, the former operator switches to horizontal stacking (*Figure 72*). The necessary individual handling of the waste containers results in a higher radiation exposure of the personnel and higher costs for emplacement.

Since 1971, the waste has been dumped mainly with a wheel loader (*Figure 75* & *Figure 74*. The simultaneous handling of several casks leads to lower costs and to a lower radiation exposure of the personnel. With the application of this technology, it becomes clear that the waste should remain in the Asse II mine. A retrieval was not planned. Possible damage to the waste containers was neglected. Long-term protection should have been provided by the surrounding salt rock.

In the emplacement chamber at the 511-metre level, only ILW is emplaced. Due to the significantly higher radiation exposure compared to LLW, a greater distance to the waste package and additional shielding was necessary during the emplacement. The packages were dropped with remote controlled winch from a chamber above into the emplacement chamber (511 metres). Lowering through a central opening in the intermediate ceiling results in the cone of tilted packs shown [115] [116].

*Figure 73 – Vertical emplacement of drums with radioactive waste (LLW) (1967)* [117]

*Figure 72 – Horizontal storage of drums (LLW) with a lost concrete shielding (1978)* [116]

*Figure 75 – Dumping of drums (LLW) with a wheel loader (since 1971)* [116]

*Figure 74 – Drum cone in an emplacement chamber (ILW)* [116]

Retrieval and decommissioning of the disposal facility Asse II

There is no decommissioning concept for the time after emplacement. Today, a decommissioning concept with prior long-term safety analysis is a prerequisite for the nuclear licensing of a disposal facility and its decommissioning. For the Asse II mine, on the other hand, the necessary decommissioning concept and the analysis of the long-term safety under today's conditions have to be worked out retrospectively. Actually, the search for the best solution employs numerous experts.

In 2007, possible decommissioning options were presented. In a so-called option comparison, three decommissioning variants were worked out on the basis of previously defined criteria:

1. retrieve the waste from the mine and store it safely elsewhere (retrieval)
2. create new cavities in a deeper part of the salt dome and transfer the waste there (relocation) or
3. completely backfill the remaining cavities in the mine with concrete and leave the radioactive waste in the mine (full backfilling)

In 2010, the result of the option comparison was presented. According to the current state of knowledge, long-term safety of man and the environment can only be ensured by retrieving the waste from the Asse. In 2013, the retrieval of the waste was laid down in law by a broad political majority of the Bundestag. Today, Section 57b of the Atomic Energy Act states: "The Asse II mine is to be decommissioned without delay. Decommissioning is to take place after retrieval of the radioactive waste." [118]

Risks

In summary, there are three main risks associated with ASSE II:

- Water inflows: Even before the start of operation, on 15 April 1965, the Lower Saxony mining authority warned of water ingress in Asse II. In February 1979 there were water ingresses in the area of the emplacement chamber, which were stopped after a long period of effort. Since 1988, approx. 12 m³ of inflowing water have been collected daily. In 2008 it became known that some of the waters have been in contact with the nuclear waste in chamber 12 for many years.
- Lack of stability: The stability of Asse II is at risk. Since 1995, the southwest flank has been partly backfilled and other actions have been taken to increase its stability. Problem: Since mid-August 2013 the operator has backfilled further sections on the 725-m level and the 750-m level with salt concrete. Here almost all chambers are filled with nuclear waste. At the same time, there is no concept on how the waste can later be retrieved.
- Unknown inventory: The documentation of the emplaced radioactive waste is incomplete and partly deliberately wrong. In August 2009 the Federal Environment Ministry had to report that instead of the 9 kg of plutonium assumed up to then, at least 28 kg had been stored. The tritium values are also considerably higher than the known inventory allows concluding. [117]

Summary

Construction of repositories for LLW and ILW in abandoned mines (caverns and tunnels) in crystalline rock environments requires bolting and concrete casting, removal of extra rock, ventilation systems, electric power systems, lightning and transports, pipe and drainage systems and the whole area needs to be protected, fenced and controlled. This is expensive but many countries have been developing old mines in repositories for LLW and ILW in various rock materials such as granite, salt and mixtures with clay. Most research carried out for the last 40 years in terms of characterizing possible sites to host a disposal facility for radioactive wastes, started with the idea of getting advantage of old mines as a ready place to start with and the huge amount of data collected is still today a disposal facility of precious information. However, many abandoned mines restart exploitation due to market changes (ex.: iron) and others have shown to have chemical conditions that will have strong impacts in the waste containers and in the environment with costs much higher than expected ; but research carries on.

Radioactive material must remain isolated from the biosphere after final disposal. This requires an intact barrier system. The suitability of a converted mine as a disposal facility mine is not determined by the geological properties alone. Rather, the previous use of the mine and the damage caused by mining in the rocks has to be taken into account in the suitability assessment. The geological situation and the previous use as an extraction mine are at least equally important for the suitability as a disposal facility mine. Possibly, the damage in the rock caused by mining restricts the use far more than the geological properties of the rock.

The further back the extraction in the mine is, the less suitable the operating resources will be for the high standards of accident prevention in case of emplacement. They have to be adapted to the technical state of the art - as for example the hoisting equipment in a shaft. Such a high effort is also necessary if a small mine is used due to the small amount of waste. The conveying and infrastructure facilities are certainly appropriate for production mining from the point of view of safety, but are not sufficient for disposal facility mining and need to be adapted.

Recognising and addressing these deficiencies requires a great deal of time. In addition, there are the time periods for licensing and the proof that a modification has improved the situation to such an extent that it is suitable for the handling of radioactive waste for emplacement. In this respect, a modification is often more time-consuming than a new construction. In Germany, modifications often have to be co-ordinated with the population so that each modification initiates a new licensing procedure which again delays emplacement.

Thus, an acceleration and simplification of the emplacement of radioactive waste cannot be assumed if a converted mine is used. The use of a former extraction mine as a disposal facility mine requires the same tests as for a new mine. Proof of the integrity of the geological barrier system in the case of converted mines must, for example, take into account the use already made of it, which is no easier than in a deposit that is not affected by mining.

### 4.3.2 Deep cavern

The term "cavern" describes a large underground cavity that have been created naturally or by mining activities. Caverns are created, for example, during the extraction of salt by brining. In Germany, they are used commercially following salt extraction, e.g. as storage facilities for gas or oil, or even specially produced for this purpose.

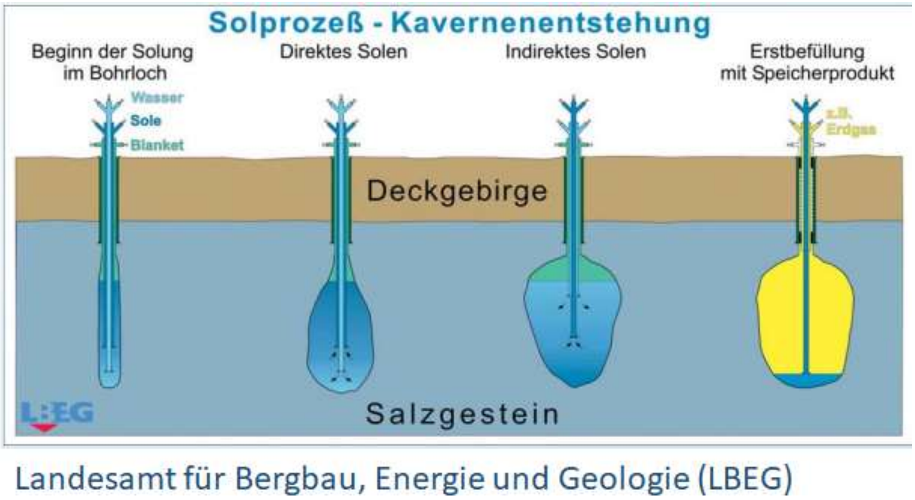



*Figure 76 – Preparation of caverns in salt rock* [119]

To create a cavern, a borehole with a diameter of approx. 20-45 cm is drilled to approximately the planned cavern depth (*Figure 76*, left). Then the salt is dissolved by flushing with a brine set and pumped to the surface. In this way, the cavern is created from bottom to top by expansion. The cavern is constantly filled with brine, which is necessary to stabilise the cavern. The construction period of a cavern used commercially for energy storage is approx. 3 years. Depending on the deposit, it takes approx. 450 m and has a cavity volume of approx. 1.7 million m³ with a diameter of approx. 70 m. Occasionally, caverns are up to 700 m high; there are also caverns with a height of 100 m – 200 m. After production, the brine is replaced by the storage medium, which is also under pressure and further stabilises the cavern.

The next picture shows caverns in 3-D representation.

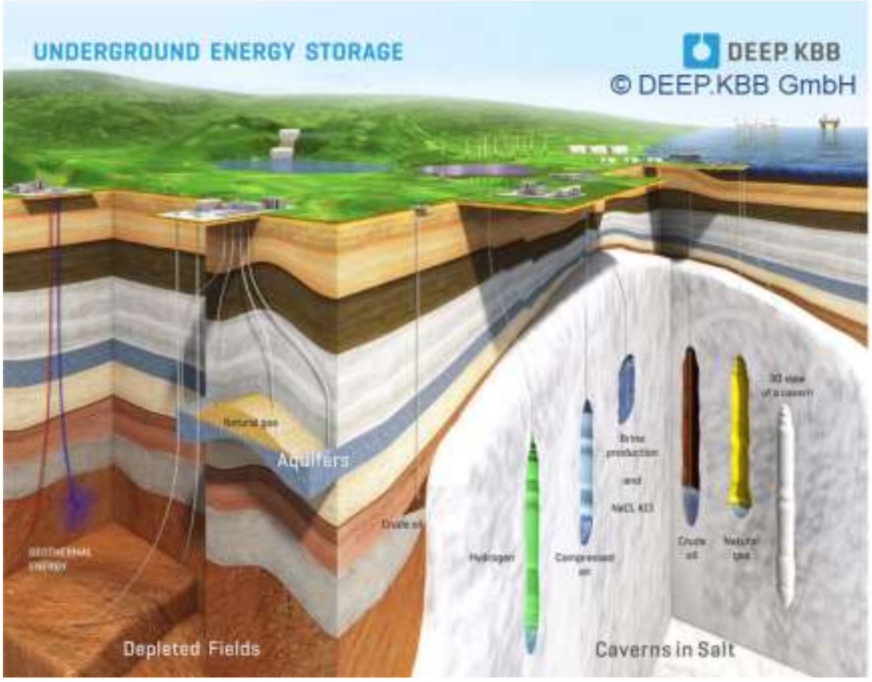



*Figure 77 – Different types of cavern*

Caverns with a height of only 100 m are not stable without counter-pressure from brine or a storage medium. According to current estimates, commercially used caverns cannot be used after their use as final storage facilities without counterpressure due to the lack of stability.

It is conceivable that new, smaller cavern adapted to the requirements of final disposal could be constructed, which is stable without counterpressure and was e.g. brined in a spherical shape (exemplary *Figure 77*, 4th cavern from the right). After widening the borehole, the required infrastructure can be installed like a shaft for final disposal. Caverns which are used for the disposal of non-radioactive waste can serve as a template. However, development work is required in any case, even if existing techniques can be partially used.

Alternatively, a slender cavern with a small diameter where only slight rock deformation is to be expected can be brined. The shape is similar to a shaft.

For small amounts of waste, shafts are suitable and sufficient. The prerequisite for storage is an intact geological barrier. The importance of newly excavated emplacement cavities increases with increasing waste volumes.

Suitable emplacement techniques for emplacement in shafts can be developed from shaft construction.

Large caverns are not stable without counterpressure. Therefore, they are less suitable for the emplacement of small quantities and only with great effort. Smaller caverns which are stable due to their shape and size (spherical or slender caverns) are suitable for storage but have to be constructed for final disposal.

### 4.3.3 Use of new mines

<u>Reasons for new mines</u>

As explained by the examples in the previous chapter, old mines cannot be used as disposal facility mines without modification. On the one hand, a shortening of the licensing and planning periods is not to be expected. On the other hand, extraction mines are mostly developed and built according to the requirements of an economic and safe production. If the requirements for a disposal facility mine differ from those of an extraction mine, the extraction mines have to be converted into a disposal facility. This mainly refers to the question which of the cavities created by the extraction mining are suitable for a final disposal.

The shape of the deposit itself and the used mining methods are important in this question. There are a numerous of mining methods that have been adapted in detail to the local deposit conditions and their presentation would fill entire textbooks. Therefore, the following considerations are not intended to represent a complete evaluation of the usability of existing extraction mines for final disposal. Rather, the aim is to provide an overview of what can be expected from the suitability of existing mines as disposal facility.

The usability evaluation starts with the central question of the suitability of cavities for disposal. In the area of the mine workings, fractures and rock movements may have taken place, which have damaged the barrier required for emplacement. Although extraction mining is also interested in producing as few fractures as possible for economic production, the criteria for an intact barrier are often much higher in mining for a disposal facility than in extraction mining. In addition, the geological-geotechnical properties of the rock play a major role in the assessment of the suitability of mines.

The Morsleben disposal facility for radioactive waste (ERAM) and the Asse II mine are former salt mines. Due to its geotechnical properties, salt permits mining methods that allow large mining cavities. These large mining cavities are available over long periods in the time scaling of extraction mining if the rock mechanics are dimensioned correctly. Whether they are suitable as a deep GDF for radioactive waste must, however, be demonstrated using criteria different from those used for regular mining and its extraction. The entire mine workings can be used from the surface to the cavities for disposal. If necessary, mine workings that are not available and necessary for disposal can be supplemented.

After mining in the Konrad mine, no cavities were available that could be used for final disposal. The cavities for final disposal have to be newly excavated almost independently of the former extraction mine. If the former mining cavities are unsuitable for final disposal, the routes leading to them are also dispensable. Therefore, a large part of the roadways used for extraction mining were dumped.

Depending on fracture formation and rock movement, final disposal in the part of the mine that was used for extraction is even excluded. Final disposal must then take place in a part of the mine where there is no extraction and thus no rock deformation. In this case, only the shafts of the mine remain usable, from which a new roadway is excavated into the resulting disposal facility as a new operating area. The difference between the extended use of a former extracted mine for final disposal and the construction and use of a disposal facility constructed specifically for this purpose is in such a case and with a highly abstract view limited to the further use or sinking of the necessary shafts.

<u>Use of shafts for direct disposal</u>

As explained above, shafts are not only valuable but also important mining structures in an extraction mine, as they are often the only surface connections through which the mine can be operated. The aim of extraction mining is to completely mine a deposit. Since complete extraction can damage shafts to the point of loss of functionality, they are protected from rock movements during extraction mining. In most cases, areas where mining was not allowed to take place were defined around the shafts, the so-called shaft pillar. The rock deformations in shaft pillar are lower than in the extraction area, but the rock stresses are higher.

Shafts have a diameter of approx. 6-8 m and thus an area of approx. 30-50 m². For 10,000 m³ of waste, at least 200 m to 600 m of shaft is required to store the waste. For small amounts of waste it may therefore be practical to store the waste directly in the shaft instead of far outside the shaft in cavities (*Figure 78*).

The suitability of shafts was investigated in a study [120] and e.g. the arrangement of the packages in a shaft was sketched.

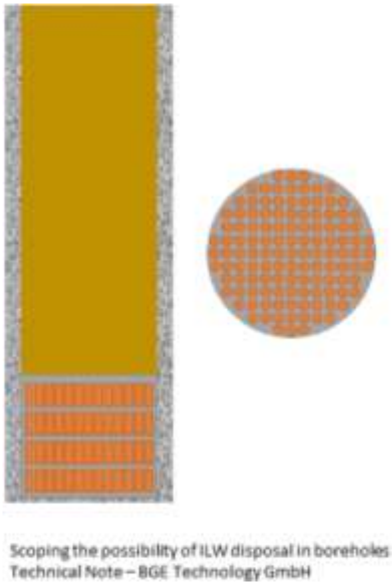



*Figure 78 – Shaft Disposal Concept* [120]

The study concludes that the loading and insertion of the waste packages could probably be faster than with the borehole concept, as the shaft offers more space than a borehole and the space is limited within a borehole. The next *Figure 79* shows a possible arrangement of waste packages in the shaft cross-section.

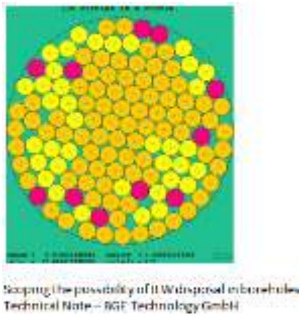


*Figure 79 – Maximum number of drums per layer* [113]

In order to lower the containers to great depths, existing shaft hoisting techniques can be used. For this purpose, e.g. high-performance drum hoisting systems are available, in which the waste packages are attached to a rope and lowered. In contrast to boreholes, the handling of the waste containers in shafts is considerably more difficult. A study was therefore supplemented by some aspects. The elastic elongation of the rope can be reduced by appropriate dimensioning, the rope rotation with suitable ropes. Furthermore, the point of rope departure must be variable so that the waste packages can be positioned in the shaft cross-section. An inclined position of the shaft makes storage more difficult and can be improved by a rope deflection. If this rope deflection is located in the area of the top edge of the ground, the rope will still swing and a precise positioning of the packages is not possible. The oscillation is even increased by corrections of the rope deflection. From this point of view it is hardly possible to place the waste packages directly at the intended location from above ground. It is easier to store the waste packages on a short rope. For this purpose mobile platforms can be installed in the shaft.

A platform is understood to be a working platform. Fixed and mobile platforms are common in shaft construction. The waste containers from above ground are delivered and deposited on a mobile platform which is guided laterally along the shaft. In addition, the platform is provided with operating equipment for the emplacement, which picks up the waste packages and places them precisely in the designated place.

The waste packages can be stacked considering that the weight load on the lower layer of waste packages, does not exceed their strength. In order to reduce the weight load, fixed platforms (*Figure 80*) can be prepared and installed and a method commonly used for shaft filling can be used.

*Figure 80 – Vertical disposal* [120]

On a fixed platform, backfill material from a mobile platform that has previously been delivered through the shaft is dropped. The backfill material can also be used to seal the stored waste packages from the biosphere. The weight of this material is initially carried away solely from the platform and via its attachment in the shaft into the shaft lining and the rock. As a function of the height, the weight force increases and with it the transverse forces on the filled material and finally on the shaft wall. Higher fillings are therefore only self-supporting, so that the fixed platform does not need to be constructed in a massive way. The first layer of waste packages is then placed on the backfill base for the next storage section.

There are extensive techniques and experience with shaft construction, operation and storage. However, these techniques cannot be applied without adaptation and innovation. Although these are available techniques, the effort required for their application in the final disposal of radioactive waste is not insignificant. The above-mentioned study also points out that a more detailed assessment of the construction and emplacement time is required, and a cost calculation is necessary to substantiate these initial assumptions of suitability.

This is of particular importance when new shafts are excavated for final disposal. On the one hand, it is a good option especially for small amounts of waste where it is not necessary to create additional cavities for final disposal. Routes connecting these cavities with the shaft and a second shaft, which is necessary for a mine operation, are not necessary. On the other hand, the construction of a new shaft for final disposal is a high investment additionally to the cost of the equipment necessary to store the waste in the shaft. However, the primary prerequisite for the use of shafts is that the rock area intended for final disposal is suitable for disposal.

#### 4.3.4 Tunnel

In this subchapter tunnels and galleries at the depths deeper than at about 100 m below the surface are considered.

#### 4.3.4.1 Geological disposal in Hungary – Bátaapáti


*Figure 81 – Bátaapáti disposal facility location* [121]

Hungary is one of the first countries in the world to operate a purpose designed and built underground disposal facility for low- and medium-level nuclear waste. The Bátaapáti complex (*Figure 81*) is designed and has started the fully entombed long-term storage of low- and medium-level radioactive waste generated by the operation of Paks, one nuclear power plant in Hungary which has four operating nuclear power reactors and is planning for installation of another two. The first reactor at Paks began operation in 1982 with the fourth coming into service in 1987.

The underground disposal facility has six emplacement chambers and will have a capacity for up to 20,0000m$^3$ of containerised low- and medium-level radioactive waste. This waste is comprised of tools and personal protective equipment used by employees in the management and operation of the power plant. The facility will also store low- and intermediate-level radioactive waste from the eventual decommissioning of the Paks reactors and some ion exchanger resin [122].

While Hungary is planning a separate facility for high-level and spent nuclear fuel waste, these levels of waste are stored in pool storage at the Paks site (*Figure 81*) and in an interim dry storage facility, also at the site, that has been in operation since 1997.

The purpose-built underground disposal facility for the low- and medium-level waste at Bátaapáti is located about 60km from Paks. Excavation began in February 2005 and placement of the specially prepared containerised waste began in December 2012. Since then, the first entombing chamber has been filled. The second is in the process of being filled. The third and fourth chambers are excavated and excavation of two further chambers is being prepared.

The facility is established in an identified outcrop of granite. It has two access decline adits of about 1.7km long to about 250 m below the surface where the disposal facility chambers are aligned. One of the adits is used for access to the excavation operations and the other for the vehicles that transport the prepared waste to the emplacement chambers.

As one of few repositories currently operating in its specific purpose of storing radioactive nuclear waste, Bátaapáti (*Figure 83* & *Figure 82*) has received international recognition for its design, construction and method of waste encasement and storage. Storage at Bátaapáti is designed to be permanent.

*Figure 83 – Complete access roadway [115]*

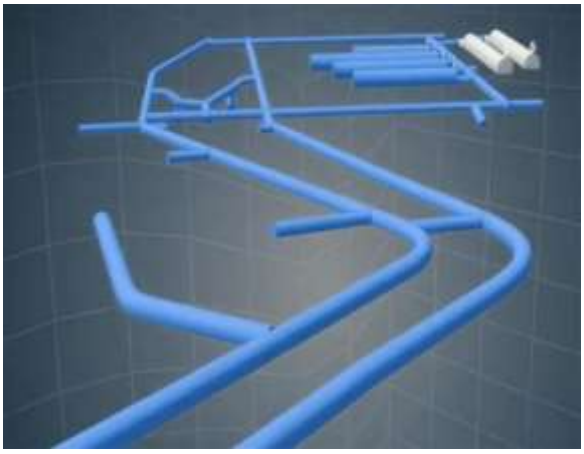


*Figure 82 – Facility has six chambers for storage of containerised low- and medium-level radioactive waste* [115]

#### 4.3.4.2 Geological disposal in Finland – Loviisa

There are not so many repositories over the world which could be identified as of "Tunnel" category. This category is not explicitly identified by IAEA publications. One of them could serve as an example from Finland.

The Finnish RWM safety policy is based on the disposal of LILW into rock cavity repositories located at the NPP sites (*Figure 84*). The design basis is geological disposal, the safety of which rests on natural and engineered barriers. The disposal system isolates the waste for a few hundred years. Therefore, all LILW generated at Olkiluoto NPP will be disposed of in the on-site disposal facility commissioned in 1992. The disposal facility, as well as the NPP units, is operated by Teollisuuden Voima Oy (TVO). A similar facility has been in operation since 1998 at the other Finnish NPP, the Loviisa plant (*Figure 85*). The designs of Olkiluoto and Loviisa repositories are different due to the difference in the local geological conditions. The disposal facility at Olkiluoto has two vertical silos, whereas the disposal facility at Loviisa has horizontal tunnels [95].

At closure the lower parts of the disposal facility will be backfilled with crushed rock. The gap between the rock and concrete silo will also be filled with crushed rock and the lower part mixed with bentonite. Tunnels and shafts will be plugged and closed with concrete at the ground surface. [95].

The Loviisa disposal facility is a rock-cavity type facility at intermediate depth in crystalline rock. The bedrock at the Loviisa site consists of homogeneous Rapakivi granite with low permeability and two major fractured zones with high permeability. The disposal facility is constructed at a depth of 120 m in the crystalline rock. Three tunnels are used for the disposal of dry maintenance waste from the power plant, two separate tunnels are used for combustible and non-combustible waste and one cavern (silo) is used for the disposal of all solidified waste. The total disposal capacity is 5,400 m3. After operation, the disposal facility will be backfilled and sealed in order to prevent ingress of groundwater into the disposal cavern and tunnels and to prevent inadvertent intrusion into the disposal facility during the post-closure phase [21].


*Figure 84 – Nuclear waste disposal facility, Loviisa, Finland* [95]

Operating waste is waste generated during the operation and maintenance of the nuclear power plant. Low-level waste includes, for example, protective plastic sheets, protective clothing, tools and sheets used in maintenance work. Intermediate-level waste consists of liquids, slurries and the ion-exchange resin used to purify process water, for example.

The power companies plan all work carefully and try to avoid generating any unnecessary decontamination or waste. All materials and people are monitored by the radiation protection personnel when leaving the controlled area. All waste is sorted and classified according to the activity level.

The Radiation and Nuclear Safety Authority has issued strict guidelines for the processing, storage and handling of different types of operating waste. Low-level waste is compressed with a hydraulic press to half its original volume and packed in barrels. Liquid radioactive waste is dried and mixed with a solid agent, such as bitumen, and then cemented in barrels.

In Loviisa, the final disposal of operating waste started in 1998.

The waste containers are transported to the disposal facility by a special shielded vehicle (*Figure 86*). All containers are appropriately marked and recorded.

Loviisa produces 100-150 m$^3$ of operating waste annually. The disposal facility will accommodate all the operating waste generated during the lifetime of the power plant. The plant site disposal facility has been designed to serve also as final disposal sites for decommissioning waste once the power plant reaches the end of their lifetime [98].

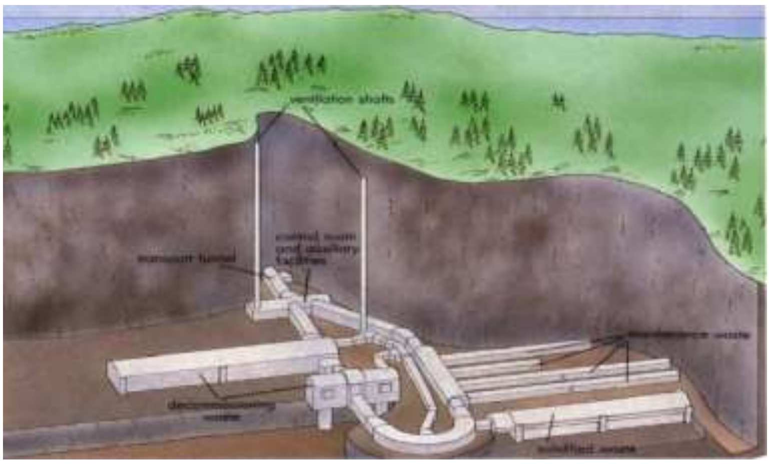


*Figure 85 – Loviisa disposal facility layout* [123]

The Finnish power companies TVO and FPH currently operate four nuclear reactors, two each at the Olkiluoto and Loviisa sites. Both companies are responsible for the safe management of nuclear wastes. The underground disposal facility program for low- and intermediate-level waste was commissioned in the late 1970s. From the start, the design basis has been geological disposal, the safety of which rests on natural and engineered barriers. The disposal system should isolate the waste for a few hundred years. During this time, the radiotoxicity of the waste will decline significantly. The critical radionuclides are $^{14}C$, $^{239}Pu$/$^{240}Pu$, $^{59}Ni$, $^{90}Sr$, and $^{137}Cs$ [124].

The Loviisa site consists entirely of Rapakivi granite, which is coarse grained and porphyritic. The disposal facility layout was constrained by the local geology. At Loviisa, the subhorizontal fracture zones above the planned disposal depth favored horizontal drifts 120 m below the surface. Thorough safety assessments were conducted during the licensing process. The Loviisa disposal facility was built between 1993 and 1997 and, it was commissioned in 1998.


*Figure 86 – Low- and intermediate-level waste is transported to the final disposal facility in a specially designed vehicle. The vehicle and the load are checked before they are taken into the disposal facility. The unloading is carried out by remote controlled crane* [95]

### 4.3.1 Deep borehole

By definition, deep boreholes extend to a depth of more than 100 m and up to 1000 m. They have not yet been used for final disposal. Thus, there are no techniques, procedures or rules for the use of boreholes as means of geological disposal of radioactive waste available, which can be used. Nonetheless, borehole technologies or other methods used, e.g., within the oil and gas industry, can provide transferrable techniques for the use of boreholes for final disposal. Technologies for drilling boreholes up to a depth of 1000 m are commercially available for different diameters. Even diameters of several meters, e.g. for shafts, can be realised.

Many technologies can and must therefore be compiled, tested for suitability and, if necessary, adapted. A detailed description of the borehole technology for boreholes up to 1000 m and an analysis of deep borehole disposal is printed in Appendix A.

### 4.3.2 Very Deep Borehole

A disposal option with boreholes of depths greater than 1000 m is considered as very DGD. These depths are far beyond borehole disposal concepts provided in a guide for borehole disposal of DSRS [92]. A possible gain in radiological safety for the human biosphere should outweigh the increased effort in exploration and construction compared to shallower depths. Using very deep borehole for disposal of small volumes or even larger amounts of HLW is considered [125].

An interesting option to reach out for deeper depths is the use of drilling technology instead of mining. Disposal in these depths can be more costly compared to simple standard drilling technology from oil and gas industry because of the larger diameter of the borehole. Although there are other limitations, this is currently considered to be a feasible technology for disposal. Therefore, this option is added in ROUTES as a disposal option for SIMS. A recent general overview of pros and cons is given by [126]. It refers to disposal depths greater than two or three kilometers, although pointing out that this is strategic matter and not a safety or technology requirement.

A generic safety assessment for disposal concepts for (spent) radioactive sources in deep but not very deep boreholes is published [127]. The approach may be similar for very deep boreholes, but the concept may vary. Numerical simulation to support generic safety analyses for VDBD concepts were published [128] [129].

Also, the model regulations for boreholes which was developed [130] may apply for very deep boreholes. The model regulations propose a "*multi-barrier concept*" comprising the waste and its container, the rubble and the borehole casing, and, finally, the host rock. These successive barriers are designed as obstacles against the transfer of radionuclides to the biosphere.

A detailed description of very deep boreholes, the history of deep borehole concepts, followed by an outline of the available technologies, waste streams, discussion on advantages / disadvantages and on the current status on pursuing deep borehole for disposal of HLW can be found in Appendix A.

# 5. Different combinations of disposal facilities options

Most of the disposal options shown in chapter 4, except for deep geological and very deep geological facilities, cannot accept all radioactive waste because radioactive wastes with higher specific radioactivity require greater safety barriers to prevent transfer of radioactive substances to the human environment.

SIMS have a much smaller amount of radioactive waste compared to LIMS. However, both SIMS and LIMS have a great variety of radioactive waste with different radiotoxicity, such as low-level tritium contaminated waste from a research institute or spent fuel from a RR. In almost all cases, the amount of LLW is much greater than to the amount of ILW, and the amount of ILW is often much greater than the amount of HLW and SF. In addition to volume and specific activity, there is also a high variety of wastes with different lifetime. It is obvious that in many cases one type of disposal option is not technically and/or economically suitable for all the radioactive waste existing in the member states– but a combination of different disposal options may be sufficient.

Not all combinations are reasonable. For example, it doesn´t make sense to have a borehole about ten meters deep for short-lived DSRS, if a large NSDF already exist and can also accept such DSRS. A combination of different disposal options is more effective, if there is one option for larger quantities of LLW on the one hand and another disposal option for smaller quantities of ILW and HLW/SF on the other. We must therefore look for combinations of disposal options with different safety barriers. The combinations presented in the following subsections were selected and discussed by a subset of the author group.

Chapter 4 sections (from on surface interim storage to geological disposal) are indicating these different safety barriers, geological and engineered. Therefore, combinations of disposal options with different safety barriers can be effective. For economic and logistical reasons, but also for public acceptance, it is assumed that such a combination is located on a single geologically suitable site.

In the following chapter, some of these possible effective combinations are described in more detail:

1. NSDF (landfill type) together with cavern, bunker, tunnel & galleries, boreholes and silo.
2. Silo & tunnels together with galleries & boreholes.
3. Deep boreholes/Very deep boreholes together with other disposal options.

Some of these combinations are already in use, some of them are planned in detail, and some of them are under discussion.

## 5.1 Near surface disposal with cavern, bunker, tunnel & galleries, boreholes and silo

There are many NSDF in operation worldwide, so the concept of near surface disposal seems to be the most available option for countries with a limited amount of radioactive waste (SIMS), especially low-activity and short-lived radionuclides. Since the SIMS are in some cases handling not only VLLW or short lived LILW suitable for typical NSDF, but also a small amount of long lived ILW and HLW and/or even SNF, the NSDF concept is not sufficient, at least for safety reasons and a combination of different disposal concepts may be of interest to them. This combination should take into account, among other things, their overall inventory (existing and future), site/geological conditions, available technology as well as the status of their programme and the requirements of the legal framework, all while taking into account their limited resources.

The costs of near-surface disposal are relatively well known around the world, but the cost of the specific design of an underground disposal solution can vary significantly, thus financial matters and available budget could be a limiting factor for implementing such a disposal concept in SIMS. One option considered could be a combination of NSDF with simple underground facilities or borehole disposal.

In this chapter, a possible combination of NSDF with silos, caverns, bunkers and boreholes is considered. Depth from the surface to an intermediate depth of max tens meters below the surface are hypothesized for NSDF. Different disposal concepts are discussed taking into account that different types of waste require different isolation and isolation time and thus different disposal solutions.

As presented in Chapter 4, the NSDF is a radioactive waste disposal facility consisting of engineered trenches or vaults constructed within a few tens of meters of the Earth's surface. This facility does not require an engineered cover but can also be constructed under a natural cover [72].

Depending on the geological conditions, a combination of different disposal concepts could be carried out at one site or at a very near distance to implement complementary barrier systems to comply with the safety concept. This safety concept, including all safety requirements, ensures the highest feasible level of a safe isolation of the waste from the biosphere with respect to the long-term disposal scale. In addition to the safety requirements, the feasibility of the different disposal concepts/options combined should also be investigated.

Trench disposal systems can be a disposal solution for SIMS dealing with a large volume of VLLW. These systems are similar in many aspects to conventional municipal or industrial waste disposal facilities. These types of facilities can also be constructed above ground in the form of mounds. Both trench and mound systems are considered suitable for the disposal of waste with limited isolation requirements, typically for several decades. However, additional design considerations allow for longer isolation periods. These facilities are best suited for the disposal of VLLW and, if site properties are favourable and adequate lining and capping systems are provided, may also be suitable for LLW. Examples of simple trench and mound disposal systems are shown in *Figure 87* [78].


**a**.

**b.**

*Figure 87 – VLLW disposal: trench at El Cabril, Spain (a) and mound at Oskarshamn Nuclear Power Plant, Sweden (b)* [78]

Near surface engineered structures are constructed with more engineered barriers than earthen trenches, and thus provide a higher level of containment and isolation of waste. They can provide a robust solution with a wider range of application than simple earthen trenches. A typical solution is the concrete vault. These systems are generally constructed at the surface and subsequently covered with an engineered cap [78].

The choice between a trench or a more complex concrete structure for disposal will depend on many factors, including climate, site conditions, the hazard posed by the waste, public acceptance, and national policy, as well as funding considerations. Each of these factors will need to be evaluated for their relevance to the disposal need and appropriateness for application [78].

Near surface engineered structure repositories can also be implemented combining two basic design concepts, such as the one developed in Slovenia for the Vrbina-Krško disposal facility, intended for the disposal of LILW-SL: first and most common as concrete disposal vaults, and additional direct access, near surface concrete silo (see section 4.2.5.3).

If SIMS also deal with limited amounts of waste consisting of long-lived radionuclides or even HLW/SNF, it is obvious that such waste cannot be sufficiently isolated in NSDF but must be disposed of in underground facilities. Moreover, if there is no strategy in a country to develop a GDF, a disposal concept of NSDF, resp. trenches and/or silos can be complemented by disposal in boreholes, the depth of which will depend on the existing inventory and waste amount.

From a safety perspective, borehole disposal is not conceptually different from near surface or geological disposal of radioactive waste. As with near surface and geological disposal, a combination of natural and engineered barriers contribute to the safety of borehole disposal. Together, these barriers are designed to contain radioactive materials until they have decayed to insignificant levels, and to provide sufficient isolation and containment to ensure an adequate level of protection for people and the environment [130] [92]. (see section 4.2.4).

Many NSDF for radioactive waste are in operation worldwide, several examples of disposal in greater depths of LLW and ILW exist as well. No facilities are being operated today for the disposal of HLW and/or SNF, but several countries are making significant progress in developing, planning and constructing such facilities. Advanced programmes can be found in Finland, Sweden, France and Switzerland. These MS are planning for the disposal of their ILW and HLW/SNF in mined geological repositories [7]. Their experience in developing and implementing disposal concepts could be evaluated in SIMS to find the most effective combination of appropriate disposal concepts.

## 5.2 Silo & tunnels with galleries & boreholes

In the following chapter, combinations of disposal facility options - silo with tunnels and galleries as well as boreholes - are investigated. In the previous chapter, a combination of concepts of open NSDF is considered with e.g. boreholes. The chapter 5.3 focuses on the combination of different disposal facilities with deep geological boreholes. Since these disposal options might not be suitable for all states with inventories of radioactive waste, this chapter focuses on combinations of the disposal option in intermediate depth (silos with tunnels and galleries as well as boreholes).

Disposal options at intermediate depth are assumed to ba at depths of tens up to around one to two hundred metres (silo, tunnels and galleries) as well as around one to several hundreds of meters (boreholes). In general, it is not possible to define depth constraints for the disposal facility concepts of silos, tunnels and boreholes. The essential depth will depend on the siting environment and the details of the developed engineered barrier system. The construction of underground silos makes sense up to a certain depth, since no additional safety and shielding can be achieved by the cylindrical design in deeper areas . In this case, the construction of tunnels with galleries is more feasible.

The combination of silo disposal facility concepts with boreholes as well as tunnels and galleries is suitable for SIMS, dealing with a wide range of different categories of radioactive waste. The combinations of these concepts (silo/tunnels with galleries/borehole) allows the disposal facility to be tailored to the amount of radioactive waste in SIMS and its different waste categories (LLW; LILW; ILW; HLW). Hence, the combination of the three different disposal facility concepts enables the disposal of radioactive waste from a wide range of operations. For example, waste from the operation of nuclear facilities, from external users like hospitals, industry and universities as well as decommissioning waste from the nuclear facilities, including spent fuel, could be disposed of in the different parts of the disposal facility.

Intermediate depth underground caverns and silos are, unlike surface disposals, entirely enclosed in the host geological formations and accessible by a tunnel or shaft. These facilities can be found in a variety of topographic settings. The construction of caverns under hills takes advantage of the isolating thickness of the overlying rock formations. Underground caverns are particularly suited to construction in hard, competent host rock and may be lined or unlined, depending on the host rock formation. Waste packages are emplaced by stacking in the caverns, with or without backfill. Wastes that requires a higher level of containment should be emplaced in underground silos with a more substantial engineered barrier system, typically comprising concrete silo walls with grouting between waste packages. Upon closure, the access tunnels and shafts are backfilled and sealed. The two combinations of disposal concepts (silo with borehole and silo with tunnel and galleries) are shown in *Figure 88*.

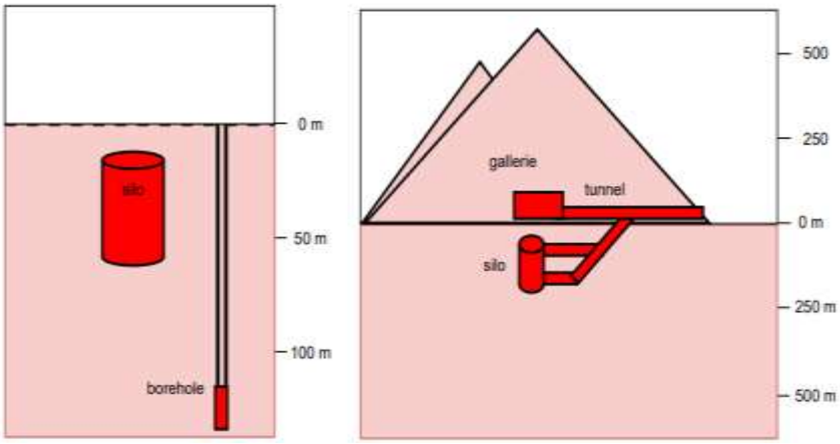



*Figure 88 – Disposal facility concepts LLW; ILW and HLW*

Underground caverns/galleries or silos are already operated in SIMS like Republic of Korea, Sweden, Hungary, Norway and Finland (see *Figure 89*) and have been considered in detail in Switzerland. *Table 16* shows an overview of the combinations of the disposal facility concepts of silos, tunnels and caverns, which are already in operation [131].

| Disposal facility | Depth | Waste volume | Activity | Disposal facility concept |
|---|---|---|---|---|
| SFR (Sweden) | 50 m below the bed of the Baltic Sea | 63,000 $m^3$ | | Tunnel with four vaults/caverns and one silo |
| Wolsong LILW Disposal Centre KORAD (Republic of Korea) | 80 – 130 m below sea level | 800,000 waste drums, 16,700 waste drums per silo | $10^3$ TBq | Tunnel with silo |
| Disposal facility in Loviisa (Finland) | 110 m below surface | | 21.000 TBq | Tunnel and shaft with two silos |

*Table 16 – Repositories for LLW and ILW* [131]

The Swedish Final Disposal facility for Radioactive Waste (SFR) is excavated in granite rock and currently comprises four vaults/caverns and a single silo. The facility is located 50 metres below the bed of the Baltic Sea and is linked to the surface by two parallel access tunnels of about 1 km length. The silo is 30 m in diameter and about 70 m high (about 50 m is intended for waste disposal). The radioactive waste is emplaced in concrete or steel boxes (referred to as moulds) and steel drums. The caverns have dimensions of about 160 m length, 19.5 m width and of 16.5 m height. The waste is packaged in ISO containers (lowest activity waste) or more robust waste packaging, i.e., steel drums, large concrete containers (called tanks), or moulds for higher activity LLW. The waste is segregated and emplaced into designated vaults based on waste type and activity content. Three different vault designs are currently in use at the SFR. The total capacity of the current facility is about 63,000 $m^3$ [131].

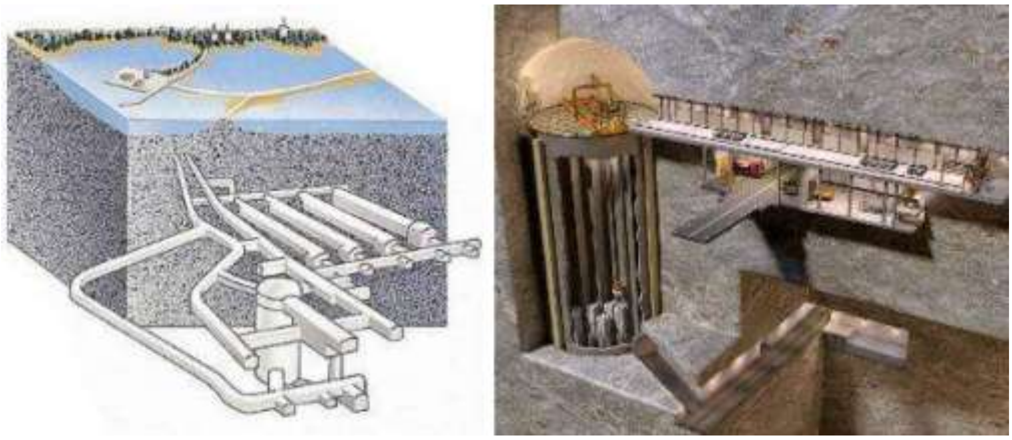


*Figure 89 – Swedish Final Disposal facility for Radioactive Waste (SFR) in Forsmark* [131]

The Wolsong LILW Disposal Centre in the Republic of Korea (KORAD) was constructed to dispose both LLW and ILW in six underground silos. In a second phase, near surface concrete vaults will be constructed for the disposal of LLW and the silos will then be used primarily for ILW disposal. The waste drums are placed in circular or rectangular concrete containers, which are emplaced in the underground silos. The first six silos at the facility are constructed approximately at 80-130 meters below sea level. They are about 25 m in diameter and 50 m in height, with a total disposal capacity of 100.000 waste drums (approximately 16.700 waste drums per silo). Until the end of operation, 800,000 waste drums with a total activity of 103 TBq will be disposed of in the underground facility [131].

The disposal facility at the Loviisa NPP site in Finland is excavated in crystalline rock and consists of three halls for operational waste, one hall for solidified liquid waste; two halls for decommissioning waste and one hall for primary circuit components beneath which 2 vertical silos for the reactor vessels are located (future extension). The facility is at a depth of 110 m below the current sea level. The disposal facility is connected to the surface by a 1170 m long access tunnel and two shafts. The facility is designed for the disposal of LILW with a total activity of 21.000 TBq [131].

The cavern or silo-type disposal facility offers a safe disposal solution for most waste in a national inventory, apart from some long-lived DSRS and high active wastes. If the concept of silos and tunnels with caverns is combined with the disposal concept of boreholes, the disposal facility also provides sufficient isolation for HLWs. Considerations could be given to drilling a separate disposal borehole on the same site or even within a cavern or silo-type disposal facility. By drilling boreholes in deeper areas, it is possible to obtain conditions close to those of geological disposal facilities. Especially for SIMS with only small inventories of HLW, the concept of boreholes provides a safe, cost effective and efficient solution. It requires only a limited area and infrastructure for the construction and closure [131].

## 5.3 Combinations of very deep boreholes and other disposal options

Disposal of radioactive waste in very deep boreholes is a concept that has long been pursued. Many more recent studies have demonstrated general feasibility, including the technical feasibility of drilling vertical and horizontal boreholes to depths of several thousand meters (see chapter 4.3.6 and subsection b in Appendix A). Generic safety cases for DBD have been provided for example by [132] [133]. However, the application of the concept to radioactive waste disposal has not yet been tested and considerable RD&D work is still needed. Therefore, it remains to be demonstrated that a safety case for DBD can be licensed [7].

The use of very deep boreholes as a disposal facility option is still at a conceptual level; there are currently no examples of actual application. Thus, there are no examples of options combined with other disposal options, and the following discussion of possible combinations of VDBD with other options is more at a strategic level. The following discussion can only highlight some general aspects. Any suitable or preferable disposal concept generally depends on factors such as predisposal management, the legal and regulatory framework, financial aspects and preferences of stakeholders. An evaluation or ranking of the disposal concepts cannot therefore be done separately from the general RWM strategy.

The inherent advantage of very DBD in fulfilling the primary safety function of isolating waste from the biosphere should not be ignored a priori when planning a disposal program [92]. Very DBD could be of particular interest for programs (in LIMS or SIMS) that need to manage specific materials independently of other disposal options. This could be the case if the implementer has to provide a customized solution for a particular waste stream, e. g. separated PU wastes or other waste types from research and prototype reactor decommissioning. Very DBD could be an option when joint disposal of certain waste types is unfavourable, but single-site disposal is preferred. The great depth and the very limited footprint of the borehole can be a strategic choice for preventing proliferation and human intrusion.

It is especially in combination with other disposal options that very DBD can be very interesting. In principle, combinations of very DBD with other options can be performed in the same way as other, shallower boreholes (see chapter 5.2). Combinations of these concepts (silo/tunnels, galleries/tunnels or a mined disposal facility with very deep boreholes) allow the characteristics of the disposal facility to be matched to the characteristics and quantity of radioactive waste in the SIMS. Hence, as with other boreholes, the combination of the different disposal concepts enhances the flexibility of the disposal program. Especially for SIMS, a combination with very deep boreholes could enable disposal of the waste at one site, if the geology is suitable.

*Figure 90* illustrates possible combinations of very DBD with other disposal options.

A simple option would be a cavern or shaft combined with a borehole. The very deep part of the borehole could be used for high-level waste, whereas as the upper part of the borehole is sealed, the shaft could be used for LILW for example (Option A in *Figure 90*).

On the left-hand side of *Figure 90* (Option B) several near surface disposal options are combined with a (nearby) very DBD. This option could be of interest for SIMS that have mainly LILW, which do not require high standards regarding isolation from the biosphere, but need to dispose of additionally high-level waste such as spent fuel from an old RR. Often, high uncertainties exist regarding the chemical behaviour of such spent fuel elements, making their handling difficult and may justify a higher standard regarding isolation.

A combination of very DBD can also be an appropriate solution for programs with larger amounts of waste and when a mined disposal facility is a viable option (Option C, *Figure 90*, right-hand side). The boreholes can potentially be drilled from the disposal level of the mine instead of drilling from the surface. A feasibility study and a cost-benefit analysis would be necessary to decide on the preferred option and would strongly depend on the amount and the specific characteristics of the high-level waste to be disposed of in boreholes.

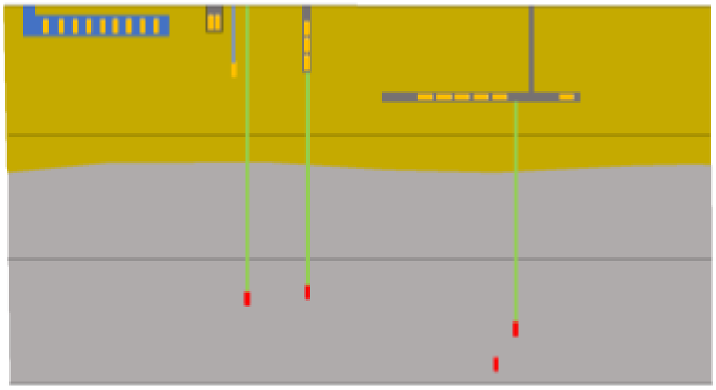


*Figure 90 – Possible combinations of very DBD with other disposal options*

**For which wastes are these combinations suitable?**

Given the size and volumetric constraints, the very DBD, in combination with other disposal facilities, could be considered a solution primarily for small quantities of long-lived, high hazard potential radioactive materials that are expensive and complicated to deal with [126]. Chapman (2019) examined in detail the technical attributes that would make a waste type suitable or unsuitable for very DBD:

- High concentration of long-lived radionuclides: Would motivate a solution that guarantees a high degree of isolation for a very long period (millions of years)
- High specific activity: For example, very high specific activity wastes, even though they contain only short-lived radionuclides
- Small total volume: Only a few tens to hundreds of cubic meters: volumes of thousands of cubic meters would require several to many boreholes
- Small package size: Maximum diameters of useable borehole space at several kilometers depth are around 400 to 500 mm
- Separated fissile material: Nuclear safeguards requirements would motivate guaranteed total isolation with no real prospect of retrieval and misuse

Balancing the advantages and disadvantages of very DBD (chapter 4.4, [7], [126]), it seems obvious that very DBD is probably not a stand-alone disposal option for SIMS. But in combination with other options, it is a promising alternative, especially for problematic waste streams.

# 6. Conclusion

The objectives of this report are to present the challenges of RWM in 8 representative SIMS in Europe participating in the ROUTES WP and to compile the existing knowledge on the various options for final disposal of radioactive waste that could be used for small amounts of waste. The sharing of experience and knowledge on RWM routes in all ROUTES WP tasks between EU countries, with advanced or less advanced RWM programmes and for different types and amounts of waste, has enabled the situation in the SIMS to be clearly identified and presented in this report.

For member states with only small inventories of radioactive waste, the issue of final disposal is not the most urgent one, but it is a challenge that will have to be addressed in the future. In addition to technical feasibility issues of , the question of costs plays a greater role, as do other socio-economic conditions and the political decision.

Thus, most of the expertise and exploration regarding radioactive waste disposal is in LIMS. This report highlights the existing knowledge gap between LIMS and SIMS about radioactive waste disposal. In order to ensure a necessary transfer of knowledge and know-how between SIMS and LIMS, the EURAD project is an important steppingstone, especially in the frame of ROUTES Task 5 and also via the Knowledge Management WPs (WP11, WP12, WP13).

The large-scale solutions such as deep geological repositories developed in Germany, France or Sweden are not necessarily suitable solutions for SIMS, that have smaller amount of HLW and ILW to manage. However, this report presents alternative solutions for the disposal of different categories of waste.

As the waste produced by SIMS may consist in DSRS, institutional and decommissioning waste of all categories, SIMS countries need disposal options that are appropriate to their waste quantity, while ensuring a satisfactory level of safety with a reasonable use of resources.

Regarding disposal options that exist in Europe, there is a wide range of options for constructing safe disposal facilities for radioactive waste, depending on the volume and classification of radioactive waste as well as the geological and technological conditions available. For example, VLLW and LLW consisting mainly of short-lived radionuclides can be disposed of in NSDF and for ILW consisting mainly of long-lived radionuclides and HLW, disposal facilities further from the biosphere are being considered. In addition to NSDF, other final disposal alternatives used or under development in the EU are: bunkers, caverns, tunnels, galleries, boreholes as well as geological disposal solutions like mines (either mine type facilities which sometimes are called new mines and former mines) or deep and very deep boreholes. Depending on the situation, existing mine workings are converted into disposal facilities or new facilities are built.

Based on the current state of knowledge on the status of SIMS disposal, it can be stated that the first missing aspect is an integrated disposal strategy in each country. The knowledge gathered in this report on the possible disposal options represents only one important component for the development of such a strategy.

When developing the disposal strategy, the existing waste with all its characteristics and diversity, the treatment options and the possible forms of disposal options (including long-term interim storage) shall be taken into account. These aspects are interdependent, so that a single consideration does not make sense.

With regard to these boundary conditions, three possible combinations of these solutions that could be suitable for SIMS were presented in this report, considering i) complementary near surface disposal concepts, ii) combination of silos and tunnels with galleries and boreholes and iii) a combination of very deep boreholes and other disposal options.

# Appendix A.

In the following steps, the subtask 5.2 (Deliverable D9.11) will describe the waste treatment routes and the new Task 8 of ROUTES will describe the relationships and interdependencies between these components (waste characteristics, treatment, conditioning and disposal options). Borehole disposal concepts.

## a. Deep Boreholes

Introduction

By definition, deep boreholes extend to a depth of more than 100 m and up to 1000 m. They have not yet been used for final disposal. Thus, there are no techniques, procedures or rules for the use of boreholes as means of geological disposal of radioactive waste available, which can be used. Nonetheless, borehole technologies or other methods used, e.g., within the oil and gas industry, can provide transferrable techniques for the use of boreholes for final disposal. Technologies for drilling boreholes up to a depth of 1000 m are commercially available for different diameters. Even diameters of several meters, e.g. for shafts, can be realised.

Many technologies can and must therefore be compiled, tested for suitability and, if necessary, adapted. This section describes existing and adaptable knowledge. The explanations are based on considerations for borehole disposal which were made in the past in various countries such as Norway and the USA and which are flanked by elaborations for Germany. Topics covered within this section about deep boreholes cover "drilling", "keeping open" and "emplacement" as well as aspects of "retrieval".

Status of drilling technology

Boreholes are used in different industries. These include the oil and gas industry, geothermal energy, mining, the military and also the geosciences for the investigation or documentation of rock layers. The borehole diameters used for these purposes are mostly standardised (approx. 8'' (210 mm) up to 14 ¾'' (340 mm)). The diameters are drilled as small as possible and as large as necessary, whereby attention is paid to the smallest possible diameter.

- Vertical drilling holes

Currently, rotary drilling, rotary drilling with down hole motor/turbine, hammer drilling, large hole drilling/shaft drilling, mining drilling are used in industry and can be examined for their suitability for drilling a disposal facility. The following pictures (*Figure 91* to *Figure 96*) show the drilling methods in principle.

In rotary drilling, a drill bit crushes the bottom of the borehole on a rotating drill rod driven from the earth's surface.

The drill pipe is gradually extended and the drilling process is continued. The cuttings must be extracted from the bottom of the borehole. In rotary drilling, a direct flush drilling method is used for this purpose, in which the cuttings are conveyed to the surface in the annular space between the drill pipe and the borehole wall with a liquid or a suitable flushing liquid. Different bits are available. The next picture shows a cross-roller bit.

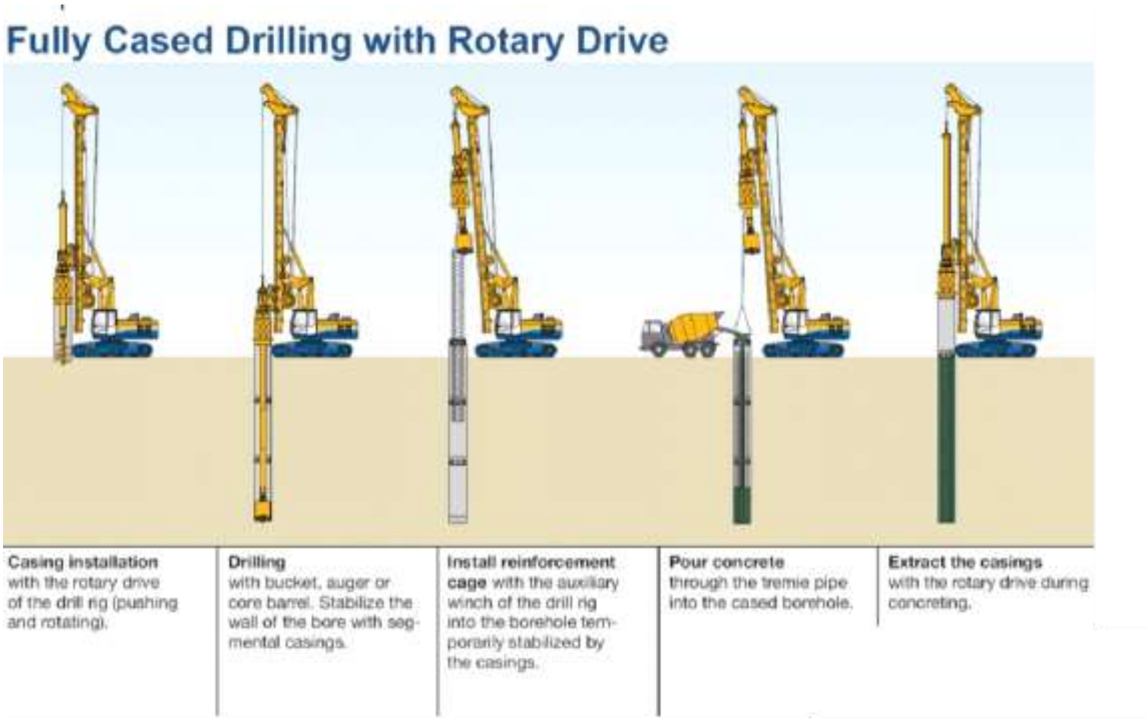



*Figure 91 – Rotary Drilling with cases* [134]

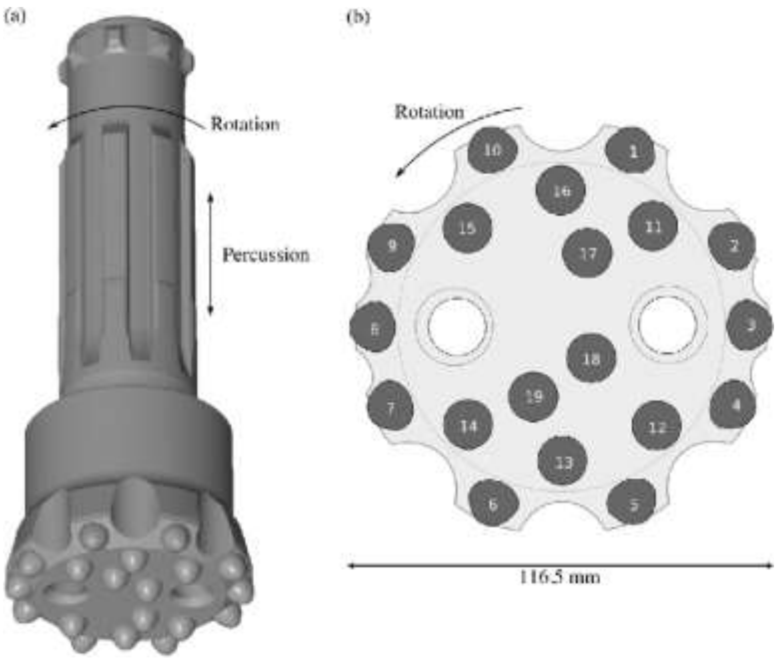



*Figure 92 – Percussion Drilling bit* [135]

The rod to which the bit is attached is not only lengthened, but the borehole wall is gradually reduced in sections.

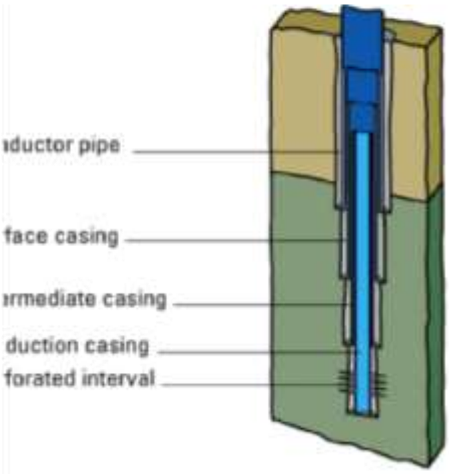



*Figure 93 – Casing design* [136] *(Original from* [137]*)*

The rods reach weights of several 100 t. A 20 inch standard rod (50.8 cm) has a weight of about 700 t at a length of 3500 m, which is lower at the planned 1000 m, but increases again with a larger diameter. A standard drill rig can handle these weights.

In hammer drilling, the energy for loosening the rock is introduced with impulse-like blows. The resulting fine cuttings are also flushed to the surface with air between the drill pipe and the annulus. A larger borehole diameter requires more air with increasing technical effort.

- Angled und horizontal drilling holes

In rotary drilling with hole motor, the drill bit has its own drive, which is suitable for deflected drill holes. Direct (described above) or indirect flushing methods can be used.

Indirect flushing methods include the suction drilling method without using any flushing liquid or the airlift method. Although the name suggests otherwise, the airlift method also uses a liquid; the liquid is only lifted with the aid of compressed air. If boreholes exceed a certain diameter (e.g. when hammer drilling) or depth, the cuttings can hardly be lifted without liquid.

It shows a possible arrangement of several boreholes from a 100 m diameter drill field with straight boreholes at an angle of e.g. 10° to the vertical.

It shows a borehole arrangement from a stationary drilling site with deflected boreholes. The course of a borehole can be controlled with high precision using today's state-of-the-art technology. The target accuracy with regard to the depth is in the tenth of a per mile range.

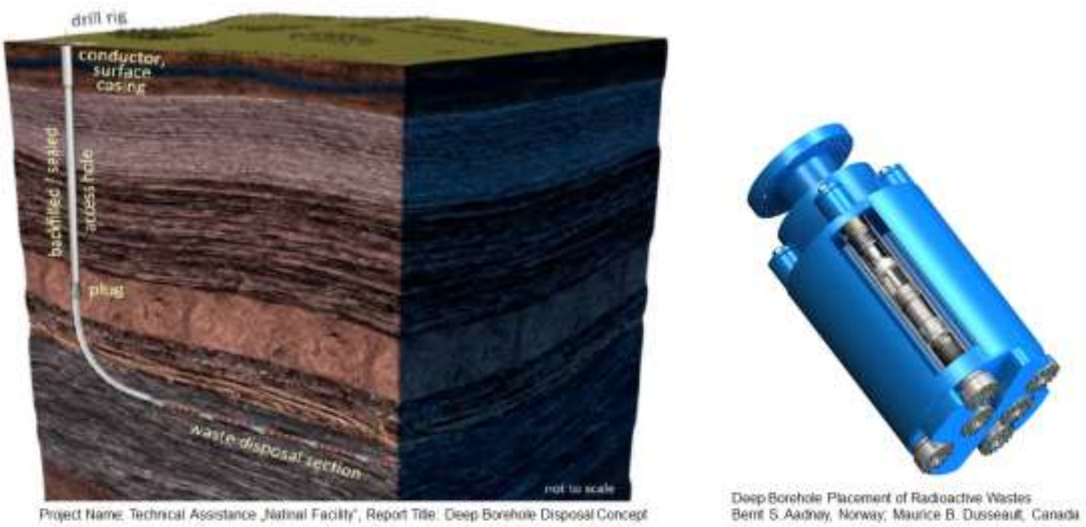



*Figure 94 – Rotary drilling with hole motor (left:* [136]*; right* [135]*)*

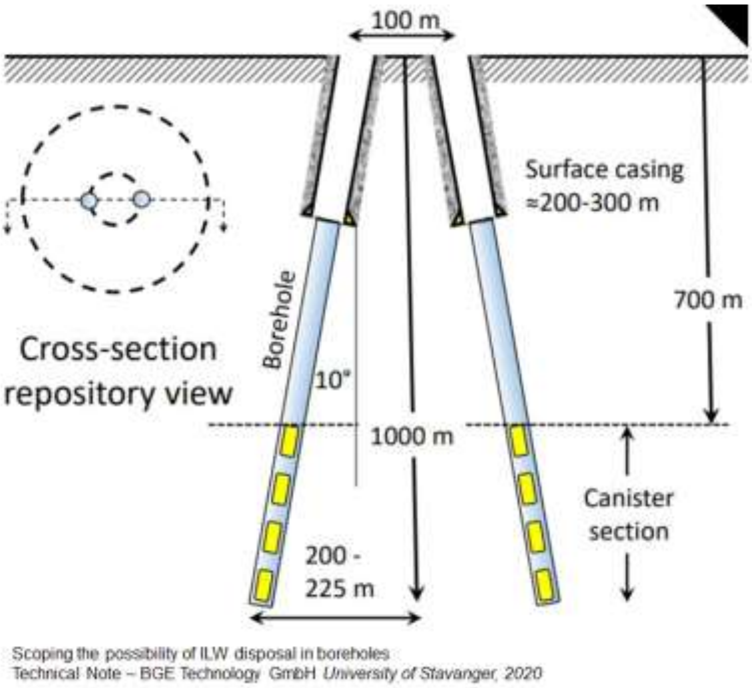



*Figure 95 – Tilted borehole option* [120]

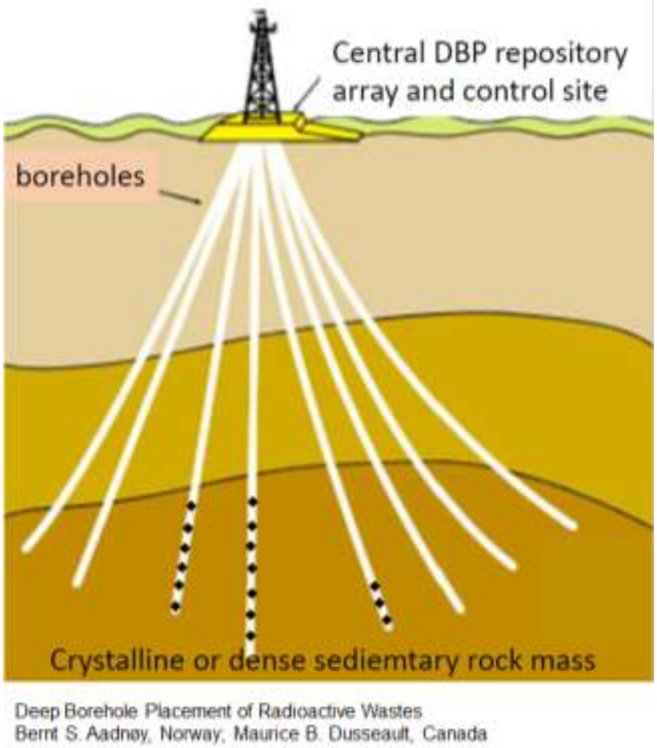
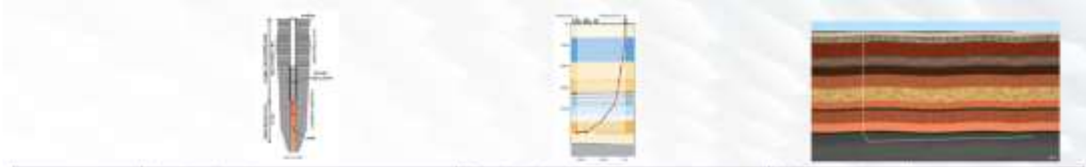



*Figure 96 – Deflected drilled boreholes from a central position* [135]

- Evaluation of drilling hole wells

Assessments of the advantages and disadvantages of different borehole orientations are already available and can be used (*Table 17*). This is independent from the use.



| | Vertical wells | Deviated wells | Horizontal wells |
|---|---|---|---|
| Positive aspects | • Easiest to drill and complete<br>• Possibility to minimize casing program<br>• Disposal and retrieval geometry relatively straight<br>• Easy sealing and backfilling operations | • Several wells from one surface location to different subsurface targets possible<br>• Reduced pressure on canisters<br>• Possibility to sidetrack<br>• Reduced rig hook load | • Longest total borehole length<br>• Reduced potential flow path through the borehole (if the horizontal section is slightly upward)<br>• No pressure put on canisters by other canisters |
| Negative aspects | • Short disposal zone<br>• Potential flow path to the surface through the borehole<br>• Great pressure on the lowest canister by overlying canisters<br>• Only one borehole from one surface location | • Potential flow path to the surface through the borehole<br>• Increased torque and drag during the drilling operation<br>• Diameter might be limited in some geologies | • Disposal geometry more difficult<br>• Complicated drilling operation (torque and drag etc.)<br>• Backfilling of horizontal part might be challenging<br>• Diameter might be limited in some geologies |

**The ERDO borehole project;** Håvard Kristiansen, Norwegian Nuclear Decommissioning(NND)

*Table 17 – Evaluation of drilling hole wells* [138]

- Large diameter bore hole drilling

Various methods are available for large-hole drilling. *Figure 97* shows different equipment for the drilling of large diameter holes. The two rods on the left are used for top to bottom drilling and the equipment on the right is used for bottom up drilling. However, these methods both require a pilot hole to remove the cuttings.

In the drilling direction "from top to bottom", gravity causes the drill cuttings to pass through the pilot hole into a cavity below the borehole (centre), from which the resulting cuttings are removed. In the right part, the borehole is drilled from bottom to top using the raise-boring method (see *Figure 98*). The drill head is driven from the surface by a rod which is guided through the pilot hole. Afterwards the cuttings have to be brought to the surface through an existing shaft or stored underground.

In both cases, a section must pass under the borehole to remove the cuttings. This requires another access point such as a shaft. It is not state of the art to drill large boreholes "from the solid" - i.e. without a pilot hole - although this was already attempted in the 1970s. Techniques would first have to be developed or adapted to enable drilling "from the solid".

Scoping the possibility of ILW disposal in boreholes
Technical Note – BGE Technology GmbH, **Source: Herrenknecht**

*Figure 97 – Shaft drilling equipment* [120]

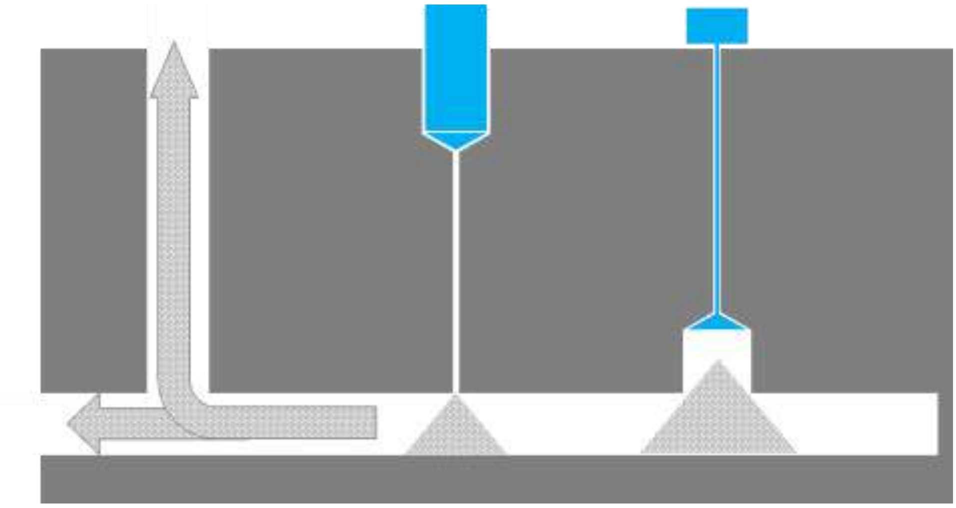


*Figure 98 – Drilling of large diameter boreholes*

For the methods not yet used in practice, such as the electro-pulse method or spallation drilling, the state of further developments remains needs to be observed. In the electro-pulse method electrical discharges crush the rock while in spallation drilling heat is used.

The description of drilling methods covers all aspects of a system, including:

- Drill bit, drill motor/turbine,
- Drill pipe,
- Drilling mud, solids control, disposal,
- Casing, cementation,
- Drilling rig, drilling site,
- Borehole measurements, mud logging, measurement/ logging while drilling,
- Hydraulic borehole tests,
- Directional drilling,
- Safety gear.

For common applications, the required drill hole diameters are comparatively small (for gas and oil wells approx. 0.2 m). If a borehole is used to store waste, the diameter should be as large as possible (e.g. 1 m), which does not contradict the previous strategy for dimensioning borehole diameters (as small as possible, as large as necessary), but the need is shifted towards larger diameters. The evaluation if a large borehole diameter is feasible for a target location, requires the analysis of various aspects. It is known from mining and tunnelling that in cavities underground, depending on rock properties and rock stress situations, sudden or even long-lasting deformations can occur. Support measures counteract these deformations. In order to control deformations in boreholes, measures are required which have to be adapted to the use of the borehole or a borehole section with corresponding effects on the requirements for the geological condition of possible drilling locations, the procedures for the safe emplacement of radioactive waste, and the reversibility and retrievability. Whether a borehole deformation above the emplacement area is appropriate as part of a borehole closure or contributes to the closure needs to be considered and shall not occur unplanned.

- Stability of boreholes

Problems of stability do not only occur in mines and tunnels, but also regularly in boreholes in both sedimentary and crystalline rocks. An example is the well documented KTB main borehole (Continental Deep Drilling Programme of the Federal Republic of Germany) [137], in which borehole edge damages with the size up to twice the bit diameter occurred. These borehole instabilities can be controlled by an appropriate casing programme and by adjusting the flushing properties. Depending on the depth, the rock stresses acting on the borehole and especially on the borehole wall increase. Depending on the rock properties such as the rock strength, the existing interfaces (such as fissures and layer changes) as well as their position in relation to the borehole, deformation can occur. Boreholes can be deformed elastically, plastically or by fracture depending on the rock properties and stress conditions. These deformations are less extensive in granite, for example, than in thinly layered clays of low strength. However, it must be assumed that deformations of the borehole also occur in crystalline rock (e.g. granite).

Extensive empirical values exist for boreholes up to 1000 m depth with diameters of 0.2 m to about 0.5 m. A possible utilisation for the final disposal of radioactive waste needs diameter of 1.0 m for a borehole. The diameter has to be 2 to 5 times larger. Therefore, the experiences with smaller borehole diameters must be evaluated accordingly - an assessment that was expressed in 2008 [138] and recently confirmed [139]. If deformations of the borehole have to be expected, these will not only occur immediately after production but will continue to occur at a reduced rate of deformation until the borehole is sealed again.

These creep deformations can clamp packages or drums in the open borehole after comparatively short periods of time. This influence needs to be considered in any retrieval strategy, which is described in more detail in the last chapter. On the one hand, the period of experience regarding the stability of boreholes is only decades; on the other hand, as already mentioned, the diameter of use must be increased with not easily foreseeable consequences for the deformation of the borehole. Therefore, an extrapolation does not seem sufficient at present if a retrieval has to be considered.

Practical experience in the production and use of 1.0 m diameter boreholes at great depths does not exist at present. The question of what is required to achieve a borehole diameter of one metre goes beyond an assessment and the necessary measures will not be deducible from assessments. Rather, it is necessary to calculate in advance the rock behaviour and its effect on the borehole.

For the prediction of the deformations of shafts with diameters of several metres and in comparative depths of 2,500 m, numerical model tests have proven to be effective, are state of the art and can be scaled back to smaller depths and diameters. In order to determine the expected borehole deformations and to develop measures against them, the calculation methods used so far for larger diameters can be scaled down to small calibres of boreholes. From today's point of view, this procedure seems to be more suitable than extrapolating a known behaviour with small borehole diameters, without assessable extrapolation results of a comparable situation.

The assessment of the stability of existing boreholes must always take into account that a pressure in the borehole (e.g. the hydrostatic pressure of the flushing fluid) counteracts a borehole deformation. In addition, a slight oval cross section of the borehole does not play such a major role in the production of gas or oil as it does in emplacement of packages. The wall thickness of a casing - comparable to a closed lining in shafts - can influence the force acting against rock deformation. In rock mechanics, this value is referred to as lining support pressure. A pressure in the pipe therefore has the effect of increasing the support pressure. If the supporting effect is dispensed, greater pipe wall thicknesses are required to prevent the pipe from deforming inadmissibly.

This reinforcement must also be determined in advance, since it entails an increase in pipe weight with effects on the tensile strength of the case. In the USA the concept of slotted casing with the same internal and external pressure ratios is followed. On the other hand, it is necessary to calculate wall thickness in advance so that a greater required wall thickness is not at the expense of the useful diameter. Maybe a larger diameter needs to be drilled. If the requirements with regard to long-term safety should completely or partially exclude a casing, high stresses act on the borehole wall of the uncased borehole wall. A rock must have suitable properties for this purpose, as known methods for reducing borehole deformation are not used.

Once these preparatory considerations, preliminary work and calculations have been completed, it is possible to specify the requirements for boreholes with a usable diameter of one metre that result from the rock mechanics point of view. Safe storage and retrieval in boreholes can only be determined by taking into account the manufacturing process of the boreholes, their construction, their filling with packages and the geological/geotechnical condition of the rock.

**Assuming a borehole diameter of one metre and a storage length of 500 m, a volume of approx. 400 m³ is roughly available.** This shows that numerous boreholes are required to emplace the estimated volume of 10,000 m³. Since a sufficient distance between waste and biosphere must be ensured in each borehole, the actual borehole length per borehole is considerably greater than 500 m. Regardless of the length of this distance, it is clear that such a volume of waste cannot be stored in one borehole. Several boreholes are required.

The distance between the boreholes required for drilling reasons is determined according to the drilling technique. In addition, mutual interference of boreholes even at depths of up to 1,000 m must be excluded with regard to the damages of the rock and heat transport, so that the underlying initial state remains unchanged. This requires e.g. existing numerical models and possibly a necessary research and development effort.

Requirements for the technical condition of boreholes

For safe emplacement and retrieval of the packages, the change of direction and inclination at one point in the borehole is important (dogleg severity). Depending on the package dimensions and the emplacement technique, it is necessary to examine which maximum changes of the borehole course are permissible. The measuring technology available on the market for determining the course of the borehole are checked and, if necessary, adapted with regard to its measuring accuracy and its applicability in boreholes with large diameters.

Even at depths of up to 1000 m, which are small in comparison to other technologies, borehole stability cannot be completely assumed, depending on the rock properties. However, the stability of boreholes at a depth of 1000 m and with a diameter of one metre can be determined with existing calculation methods. The measures for stability must be adapted to the geological properties on the one hand, and to the methods for emplacement and reversibility/retrievability on the other.

Furthermore, damage during the construction of deep boreholes cannot be completely prevented. This includes the damage risks due to:

- Borehole instabilities, sinking,
- Fluid inflows (water ingress), flushing losses,
- Breakage of drill pipes and drill string components,
- differential sticking due to pressure differences and
- Clamping of the packages during installation.

Procedures must be developed which are possible to repair or prevent the damage.

Requirements for the geological nature of possible drilling locations

With regard to the suitability of geological sites, there is a broad knowledge, numerous references and an established network regarding DGD of radioactive waste (see IGD-TP platform https://igdtp.eu/), e.g. through the work for the disposal programmes in Germany, France, Sweden, Finland and Switzerland. Based on these fundamentals and with additional focus on the latest developments worldwide (e.g. a current review by Neil Chapman [140] provides an initial overview) and on the disposal concept in deep boreholes in the USA [141], the relevant geological facts regarding the requirements for the characteristics of potential drilling sites can be worked out.

The description of the requirements is methodically derived from the elements of a safety analysis of a potential site:

- Closure and security concept,
- FEP (Features, Events, Processes) and scenario analysis,
- Geoscientific long-term prognosis,
- Consequence analysis,
- uncertainty considerations.

However, such a procedure has proven methodically to be useful for the long-term safety assessment of disposal facilities in different host rocks and for different emplacement concepts. With regard to the geological aspects, the first step is to list and evaluate the results:

- Which natural barriers are used and how are they characterised?
- How can these barriers be impaired or endangered by the emplacement itself and which technical barriers prevent or reduce this?
- Which processes can impair the long-term safety of the geological and engineered barriers?

- Which protection goals have to be ensured during the operational and post-operational phase and how can a demonstration procedure for long-term safety be conducted?

These aspects show that for a long-term safety assessment, the entire disposal facility system must be considered and, in particular, geological and engineered barriers can only be considered separately to a limited extent. Thus, the general requirements regarding long-term safety of the engineered barriers and, in particular, the borehole plug and the borehole in general must also be worked out. With regard to the procedures for the safe emplacement of radioactive waste, these requirements can be taken up for more detailed discussion, e.g. with regard to transport paths due to water columns in the borehole after emplacement or with regard to borehole plugging materials and techniques.

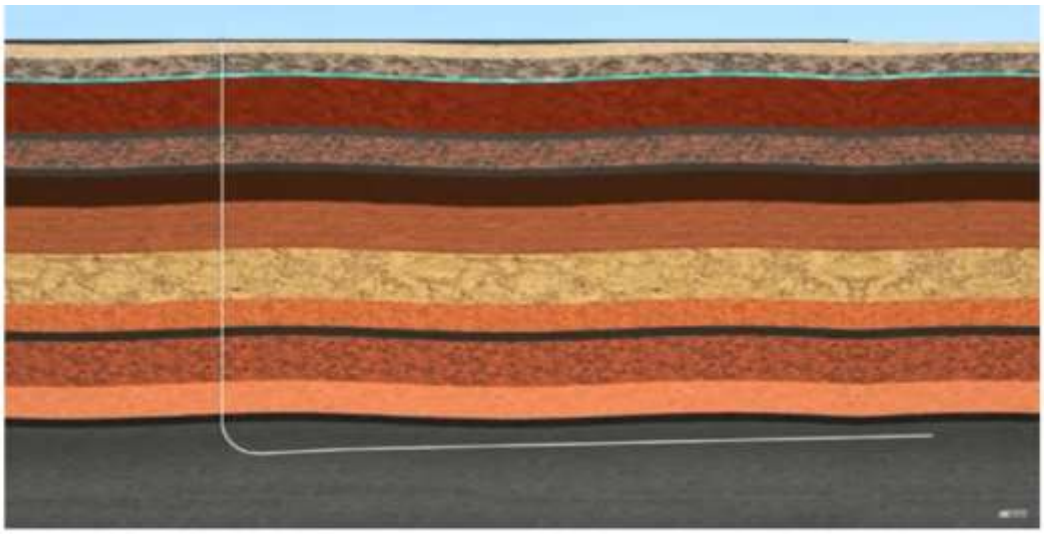
As an example of the methodological procedure, the first aspect (radiological long-term safety) is briefly discussed with regard to the requirements on rock formation (host rock): A potential transport of radionuclides to the biosphere after loss of integrity of one or more packages and mobilisation of the radionuclides from the wastes is based on:

- advective transport in the well, e.g. in dissolved form with the water in the well due to density or pressure differences or in the gas phase along the well between the container and the casing or well wall,
- advective-diffusive transport through the rock surrounding the well. This is favoured by high permeability of the rock, high transmissivity of fissure and fault zones or natural large-scale flow systems driven by infiltration and exfiltration as well as geothermal convection if necessary. This transport can be counteracted by e.g. sorption on fissure surfaces, matrix diffusion, or dilution by uncontaminated water.

The search for suitable host rocks is therefore guided by:

- geological and tectonic characteristics such as storage conditions, direction and distribution of fault zones or the absence of overpressure areas,
- hydrogeological characteristics such as permeability distribution or natural flow conditions,
- geochemical and biochemical characteristics such as reducing conditions or geochemical stability,
- geomechanical characteristics,
- geothermal characteristics to exclude thermally induced convective transport,
- relevant properties of the surrounding host rock.

The characteristics and course of a suitable host rock deposit also influence the course of the borehole (see *Figure 99*).



**The ERDO borehole project;** Håvard Kristiansen, Norwegian Nuclear Decommissioning(NND)

*Figure 99 – Borehole pathway depending on deposit of the host rock* [138]

In addition to the above mentioned characteristics, it is also necessary to point out and discuss existing detection methods. A first review of publications from Great Britain [142], Sweden [143] and the USA [144] will provide an overview. Where possible and meaningful, comparisons with characteristics and demonstration methods for classical deep geological repositories can be used, especially when estimating and comparing costs or uncertainties.

However, the borehole path (depending on the strength and direction of the rock layers) also influences the expected deformation of the borehole and its usability. Therefore, analogous considerations must be made with regard to the requirements for the condition of the drilling route. The extent of the expected borehole damaged zone and the stability (avoidance of water ingress, low stresses) are especially relevant. Special attention needs to be paid to the interface of the procedures for safe emplacement and the state of the art of drilling technology.

<u>Procedures for the safe emplacement of radioactive waste</u>

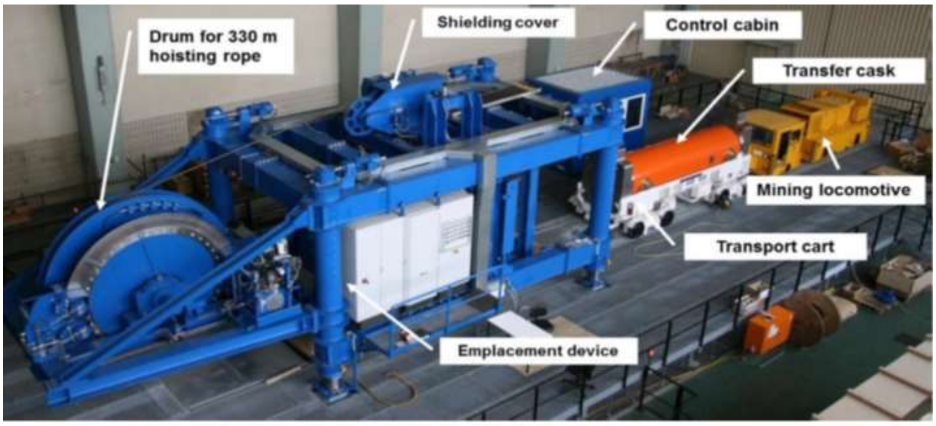
For the procedures for safe confinement, the emplacement procedure, the requirements on the engineered barriers such as casks or well seal as well as possible incidents must be considered.

- Conveyor systems

The procedures for the safe emplacement of radioactive waste include the lowering of the containers to the intended depth. For this purpose, e.g. conveyance via rods or ropes is considered. In a borehole, the transport can be carried out with cable-bound conveying technology, i.e. with a winch (see *Figure 100* and *Figure 101*). In conventional shaft hoisting technology, a similar task, travel distances of about 3,000 m are currently the technical limit. It shows a rope with cables which make it possible, for example, to control coupling processes of packages even in the borehole or to enable visual transmission.

For rod usage, the packages can be inserted and removed, e.g. on a drill pipe. For cable haulage there is a comprehensive and proven set of regulations and a state of the art for comparable applications (e.g. shaft hoisting technology, offshore technology, crane technology). In addition, these rules and regulations have already been applied to mining of disposal facilities. In addition, a concept and prototype developed for mining is available for container transport up to the point of licensing support.


*Figure 100 – Heavy-duty cable with electrical power carriers for special operations* [135]




Project Name: Technical Assistance „National Facility"
Report Title: Deep Borehole Disposal Concept

*Figure 101 – Demonstration test stand of the emplacement device for deep vertical boreholes up to 300 m* [136]

The following points need to be addressed:

- requirements on materials, rope construction and design (static rope safety, operational strength under dynamic load) of the ropes,
- requirements on the structural design of the rope drums and the rope winding systems of winches with long travel distances,
- requirements on the safe design of brake systems and other safety-related equipment of the winches under consideration of the special risks resulting from the payload of highly radioactive waste materials.

In addition, there is the preparation of the state of the art with regard to the possible control technology and sensor technology for rope-bound or rope-free data transmission and energy supply of actuators at the winch/package interface. This point concerns gripper technology and release mechanisms between the rope and the packages to disconnect them from each other when the emplacement position in the borehole is reached.

A comprehensive set of rules and regulations for the use drill pipes is not available. If necessary, it has to be checked whether existing technologies from the rope technology are transferable to the rod technology or whether there is still a need for development.

In this respect, the aspects described in the following and related to the rope haulage technology will initially apply to haulage by drill pipe or other rods. Following, there are alternative methods of haulage which require development.

The complete consideration also includes conceivable incidents, such as e.g. a cable or rod break and the resulting "package crash" is an assumed accident. Comparative analyses of regulations in other fields of technology with similarly high requirements on the operational safety of ropes (e.g. shaft hoisting technology, cableway technology, lift technology, crane technology) can be used to execute the safety analysis. It must be evaluated which requirements arise for the function and dimensioning of safety gears in case of failure. If such safety gears are required, it must be examined whether safety gears exist in other technologies (e.g. in drilling technology) which can be optimised to the geometric characteristics of boreholes and the properties of the borehole walls.

Alternatives to this are preventive measures which ensure a safe rope and rod condition and therefore an incident can be safely excluded. These include, for example, the design, regular inspection and suspension of the vessels using multi-rope technology.

Other incidents are e.g. handling errors on the surface when inserting the packages into the drill hole and an "irreversible winding failure". The danger is known as "suspended cable" or by tilting of the package when it is lowered. Techniques are available regarding a safe detection sensor system for the formation of a longitudinal rope or jamming of a package in other fields of application, e.g. in shaft hoisting technology, lift technology and cable car technology. Among other aspects, preventive measures can also influence the design of the outer shape of the packages,

Therefore, the package design will also be influenced by the requirements of the conveyor technology. Here, the shape and surface quality of the borehole, (long-term changes) as well as the guiding devices in the borehole are important.

These aspects must be coordinated with each other according to the state of the art. After assessing the state of the art for applicability, the expected development requirements must be described. When describing possible methods of production for emplacement, it is also taken into account that the packages and production technology are intended for retrieval. If a stored package is to be safely retrieved after many years, e.g. with a winch in the borehole, the requirements to be met by the sensor and actuator technology (such as the gripping systems) of the winch equipment are much higher than if the package only has to be set down after reaching the disposal position. When designing the package, suitable interfaces must be provided, for example, shaped elements at the front sides, which can be securely gripped with a gripper even under the conditions of a drilled hole (possible presence of dust, rocks, inclined positions).

This has considerable effects on the entire emplacement concept, in particular on the question how many containers can be stored in a borehole.

- Borehole seals

Borehole seals are well established in a wide range of industries such as oil and gas, $CO_2$ sequestration and gas storage. In addition, in connection with the final disposal of radioactive waste, e.g. in classical DGD, the closure of access or ventilation shafts or access ramps is an issue that is considered in the corresponding safety demonstrations.

In a first step, the usual techniques and materials from the various branches of industry can be compiled and synthesised. In a second step, the techniques can be evaluated with regard to long-term safety, whereby analogies to the procedure for classical DGD can be used in some cases.

Particularly relevant points that need to be considered, concern the integrity of a closure over long periods of time, both in terms of mechanics and chemical properties. The technical feasibility of a closure or of backfilling around the packages requires that the corresponding closure or backfilling material can be transported to the locations of the closure or backfilling.

- Conveying the sealing or backfilling materials

In case the annular space between the borehole and the packages has to be filled, material has to be conveyed vertically over a maximum distance of approx. 1000 m. With deviated boreholes the production distance is even longer. Depending on the requirements for safe containment, different techniques can be considered. Different techniques are available for vertical conveyance of different materials, whereby experience shows that the conveying length is a significant influencing variable. The conveying technology must be examined to determine whether these are suitable as backfilling techniques. If necessary, the recipes of the backfill material can be changed on the basis of existing experience with conveying technology in such a way that conveying is possible.

The technical barriers must then be monitored. Based on the state of the art in science and technology in the field of gas storage, monitoring of old mines with residual waste emplacement, and classical DGD (e.g. through the results of the EU project MODERN/MODERN 2 [145]), analogies are possible and transfer possibilities must be considered. A critical point that needs to be examined are the potentially different depth locations. In addition to these analogies, the most recent results on monitoring from international specific deep well disposal programmes are analysed, with a focus on the USA.

<u>Packages</u>

Package for emplacement must ensure the safe confinement of radionuclides during surface handling, emplacement and operation of the installation. Furthermore, they should maintain their integrity as long as possible. In the concept of disposal in deep boreholes, the geological barrier has the essential radionuclide retention function. **Thus - from long-term safety aspects - a package integrity beyond the closure of the disposal facility is not necessary.** On the other hand, requirements for retrievability may require this integrity. In addition, there is criticism that this concept does not follow the "Defence-in-Depth" principle, with various barriers. An study of MIT from 2009 (Sapiie & Driscoll, 2009) therefore considers a copper lining (comparable to Swedish KBS-3 containers) and filling of the free spaces of the packages with a cementitious mortar with protective and retaining chemical properties to increase the barrier effectiveness of the packages. An appropriate interface for reversibility has to be taken into account.

<u>Reversibility and retrievability</u>

A sustainable disposal concept must ensure long-term safety and containment of the waste. The protective functions are taken over by the geological host rock, which is proven to be stable over the long term, and the engineered barriers. Within the framework of the public discussion on the subject of final disposal, there have recently been many calls in Germany for final disposal concepts with the option of decision reversal and retrievability. These demands are often connected with the view that technical measures for a later retrieval of the waste from the disposal facility could increase the safety of a disposal concept. Other arguments hope for new technical procedures in the future for a better dispose of the waste or their use as an energy source.

In order to ensure the long-term safety of a disposal facility, it should be ensured that radionuclides cannot penetrate the biosphere or can only penetrate it to a very small extent. For this purpose, preferential flow or release paths must be prevented and therefore all cavities must be filled as completely as possible and sealed tightly. In contrast, technical concepts that are intended to enable or facilitate the retrieval of waste necessarily require that parts of the disposal facility be kept open. This basically increases the probability of water access to the available cavities and thus also to the emplaced waste.

In order to be able to evaluate the technical feasibility of emplacing radioactive waste in deep boreholes, it is also necessary to discuss the possibilities and the associated expenditure for reversing the decision and for retrieval. Technical measures have to be taken in order to be able to emplace waste in a retrievable manner and clarified which consequences these have for a disposal concept and its long-term safety.

The time periods to be considered play an important role here: the measures can be planned in such a way that the waste can be retrieved for several decades as long as a disposal facility is still in operation, or the planning extends to retrievability over hundreds of years or even longer. This time aspect has a considerable impact on the technical design, but also on the long-term safety of a disposal facility.

Periods for ensuring retrievability vary from country to country. For the possible correction of errors, retrievability measures in a disposal facility should be taken into account in Germany. The safety requirements of the competent ministry require a retrievability from the decommissioned and sealed disposal facility for a period of 500 years including the manageability of the waste packages.

In the operating phase up to the closure of the wells, it must be possible to reverse the decisions and thus to retrieve the waste containers. Measures taken to ensure the possibilities for retrieval must not impair the passive safety feature and thus long-term safety.

<u>Summary</u>

No drillings for the special requirements of a disposal facility have been carried out so far. However, extensive experience can be drawn upon, as with a smaller calibre, boreholes of more than 1000 m are state of the art.

With regard to final disposal, requirements and strategies were formulated from these experiences. The interrelationships that are effective for successful well production and final disposal are summarised exemplarily in the following *Figure 102*.

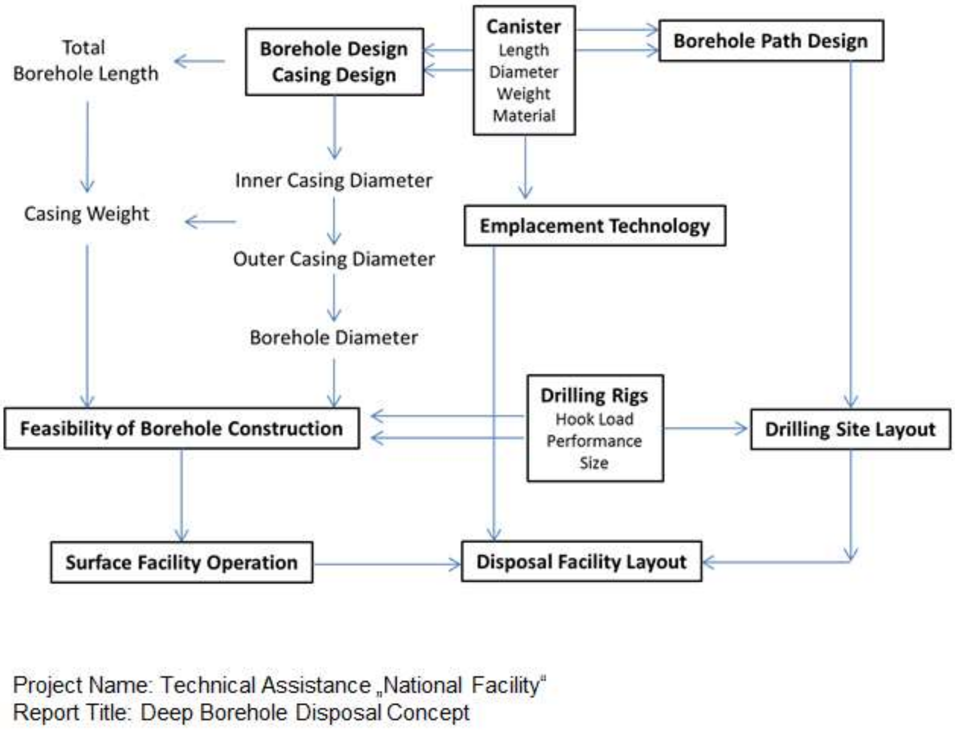
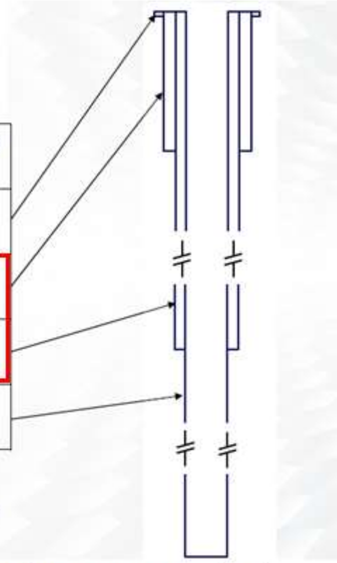



*Figure 102 – Planning stages for a first design of a DBD facility in Norway* [136]

For utilization for disposal, the largest possible borehole diameter is of interest. It is possible to refer to existing considerations which show that diameters of approx. 1.00 m are also feasible at depths of up to 1000 m (see *Figure 103*).



| | Casing outer diameter (m) | Casing setting depth (m) |
|---|---|---|
| Conductor pipe | 1.500 | 15 |
| Surface casing | 1.250 | ~ 400 |
| Intermediate casing | 0.900 | ~ 1800 |
| Disposal casing | 0.700 | 3500 |



*Figure 103 – Vertical borehole design* [138]

For rock mechanical reasons, however, the effort required to maintain the cross section with a large borehole diameter increases with increasing depth due to the greater rock stresses. The course of the borehole also influences the stability of the borehole wall depending on its position in relation to the stratification of the rock.

The today's drilling technique enables the creation of a suitable cavity even at depths of up to 1000 m. More decisive is the question whether, how and over which periods of time these cavities in rocks must or should be kept open. It should be possible to reconcile the requirements for a host rock for safe isolation from the biosphere with the requirements for the low deformations of the cavities required for emplacement (here boreholes). With regard to these considerations, geotechnical planning tools are available which can be scaled down from large-calibre cavities - such as shafts. However, such planning must be carried out in order to ensure the chances of stability of boreholes whose diameters exceed the previous extensive horizon of experience (using borehole technologies). It is not so much the technology for carrying out boreholes that holds imponderables, but rather the plannable verification of stability, which can impair if not prevent the operative production of a borehole.

Once the borehole has been successfully produced in a ready-to-operate state, significant hurdles are overcome for successful disposal. If drill pipes or ropes are intended for handling weights of several 100 t, there is no fundamental obstacle to the lowering of the packages in the casing. On the contrary, existing techniques can be suitably adapted.

Aspects of retrievability depend decisively on the stability of the rock and - if necessary - on the casing. The interactions between the required stability of the borehole during emplacement, the isolation from the biosphere and the reversibility and retrievability are the major tasks for final disposal in boreholes. The task cannot be planned in general but for each geology, each borehole diameter and direction, each casing and each sealing technique prior to emplacement and under consideration of numerous events.

### b. Very Deep Borehole

A disposal option with boreholes of depths greater than 1000 m is considered as Very DGD. These depths are far beyond borehole disposal concepts provided in a guide for borehole disposal of DSRS [92]. A possible gain in radiological safety for the human biosphere should outweigh the increased effort in exploration and construction compared to shallower depths. Using very deep borehole for disposal of small volumes or even larger amounts of HLW is considered [125].

An interesting option to reach out for deeper depths is the use of drilling technology instead of mining. Disposal in these depths can be more costly compared to simple standard drilling technology from oil and gas industry because of the larger diameter of the borehole. Although there are other limitations, this is currently considered to be a feasible technology for disposal. Therefore, this option is added in ROUTES as a disposal option for SIMS. A recent general overview of pros and cons is given by [126].

It is referring to disposal depths greater than two or three kilometers although pointing out that this is strategic matter and not a safety or technological requirement.

A generic safety assessment for disposal concepts for (spent) radioactive sources in deep but not very deep boreholes is published [127]. The approach may be similar for very deep boreholes, but the concept may vary. Numerical simulation to support generic safety analyses for VDBD concepts were published ( [128] [129].

Also, the model regulations for boreholes which was developed [130] may apply for very deep boreholes. The model regulations propose a "*multi-barrier concept*" comprising the waste and its container, the rubble and the borehole casing, and, finally, the host rock. These successive barriers are designed as obstacles against the transfer of radionuclides to the biosphere.

The following chapters provide a short overview on the history of deep borehole concepts, followed by an outline of the available technologies, waste streams, discussion on advantages / disadvantages and on the current status on pursuing deep borehole for disposal of HLW.

Short history of DBD concepts

In 1957, the U.S. National Academy of Sciences (NAS) Committee on Waste Disposal considered the deep disposal of liquid radioactive waste brines in permeable formations at depths of 1,500 m or greater using boreholes [146]. Deep disposal of liquid wastes was revisited from time to time (e.g. [147]) but is not pursued currently anymore and is banned. Research that focused on deep disposal of solid waste forms in boreholes did not receive serious attention until the late 1970s, following advancements in deep drilling technology by the petroleum industry.

A feasibility study performed in 1979 by Berkeley [148] provided a thorough analysis of the very deep borehole concept, highlighting several key uncertainties and technological hurdles, many of which are relevant today and highlighted in the following sections of this subchapter. This was followed by more detailed study [149]. Very deep boreholes were also considered for the disposal of militarily usable fissile material [150].

Other countries that have conducted some research on the VDBD concepts include Switzerland [151], Denmark [152], Sweden [153] [154] [145], and the United Kingdom [144].

Deep borehole concepts were reviewed for the U.S. NRC in 2011 [155]. This report came to the conclusion that the construction of boreholes in hard rock to the depths necessary for waste isolation is technologically possible and likely could be economically feasible. Significant technological uncertainties and potential safety challenges remain, however, in the areas of waste handling and procedures for lowering waste into the disposal boreholes and those uncertainties limit the ability to reliably evaluate post closure performance of the DBD concept.

A report published in 2004 [144] thoroughly summarizes previous international studies and concepts for VDBD. Issues related to hydrogeology, hydrochemistry, geomechanics, canister construction, canister retrievability, and drilling technology, among others, have been identified and, to varying degrees, examined. In 2009 an in-depth analysis of VDBD by Sandia National Laboratories was published [156], including a formal scenario analysis and performance assessment.

VDBD is considered as an alternative option in several countries, e.g. in Germany [157]; it is under observation by the German regulatory authority BASE. [158]

Some types of borehole disposal concepts are not pursued or subject to studies any longer:

- The Deep Underground Melt Process (DUMP) concept, developed by the Lawrence Livermore National Laboratory in the 1970s [159] [160]. This project consisted in introducing exothermic waste into a cavity made from a borehole, either using conventional explosives or by means of a nuclear explosion, at a depth between 2 and 4 km. The goal is to place a large amount of waste in the same cavity;
- The Deep Self Burial (DSB) concept, originally developed by Sandia National Laboratories [161] [162]. This concept considers waste disposal down to a depth of 2 km, enclosed in high-density containers, possibly cooled in cased boreholes, then, if necessary, in disconnecting the cooling system so that the waste melts as it sinks deep into the liquefied rock, under the effect of its own weight. Renewed interest in an option derived from this concept emerged in the 1990s, particularly in Russia, China and the United Kingdom, aimed at eliminating small quantities of

high-level waste [163] [164]. The new option aimed at partial melting of the host rock, the waste being protected by a container not intended to melt and mix with the molten rock;

- The Deep Rock Disposal (DRD) concept, also developed by Sandia National Laboratory [165] [166]. The concept combines elements of the two previous concepts: the targeted wastes are essentially liquid effluents from spent fuel reprocessing plants. The project does not envisage them being placed in containers, but rather directly injected in liquid form into deep boreholes, where they, due to their exothermicity, would melt the surrounding rock;
- The solidified waste in-situ melting concept [167] consists in mixing the solid HLW with rubble inside a cavity. The average thermal load of the waste-rubble unit should allow the melting of the rubble while avoiding a thermal degradation of the rock beyond the molten rock zone. Likewise, the voids in the embankment are considered to allow the expansion of the rock during the melting, thus reducing the risk of causing the surrounding formations to fracture in the future.
- An option derived from DSB is still under study in the United States until recently [168] [169] [170] [171] [172]. In particular, it is considered as a disposal solution for Hanford caesium and strontium sources. Its principle is based on the partial melting of a granite-based material ("rock welding") located around or above the waste packages. The heat released by the waste is, for this concept, high enough to cause the melting of the material that surrounds it but remains low enough not to deteriorate the containers in which it is enclosed. When cooling, it is expected that the granite-based material will naturally seal the area around the waste in the deep boreholes. Current research focuses on materials (especially crushed granite) that would allow effective recrystallisation for sealing elsewhere in the borehole.
- Direct injection of waste in liquid form (or incorporated in a solution after crushing in the manner of a cement grout) in a geological layer requires that the rock be sufficiently porous and permeable to allow injection. Its characteristics must also make it possible to limit the horizontal or vertical migration of radionuclides. The choice of a porous and permeable geological formation, but with weak hydraulic gradients and framed by formations of low permeability, is therefore favourable. In the United States, after initial tests in 1959 and 1960, the Oak Ridge National Laboratory in Tennessee has regularly conducted radioactive effluent injection operations. Between 1966 and 1979, about 7,500 $m^3$ of waste was injected in the form of cement mortar at a depth of about 300 m [173] [174]. In 1972, high-level waste injection in crystalline rock under the Savannah River site (South Carolina) was abandoned prior to its implementation, due to public concern [175]. In the United Kingdom, the injection of liquid radioactive waste into rocks, after examination based on the work done in the United States and Russia [176], is now considered to be contrary to UK regulations in force, the latter requiring in particular that the waste be managed exclusively in solid form. At the moment no country considers rock injection as a definitive management option for intermediate or high-level waste.
- A planned field test in the U.S. [174] was suspended because of political issues in 2018 [177]. In 2016, the company Battelle was commissioned by the DOE to drill atest-borehole of a depth of 4,880 m in a crystalline basement of North Dakota for the disposal of caesium and strontium sources as well as possibly high-level calcinates and salts resulting from the electrometallurgical treatment of sodium fuels [178]. The site identified in North Dakota was abandoned as a result of protest movements. The DOE and Battelle successively considered several other sites for this test-borehole (South Dakota, New Mexico, Texas, etc.), but had to abandon them in turn due to local opposition [179]. The DOE announced the termination of its test-borehole project on the 23rd of May, 2017 [178].

More detailed information of these disposal concepts can be found in a IRSN Report of 2019 [180].

<u>Technology</u>

The detailed design or choice of borehole drilling and disposal technology is depending on the overall disposal concept. Other limitations or restrictions on using drilling and disposal technologies may logically arise from the waste and container characteristics and site specific conditions. **If boreholes larger than 1 m in diameter at a final depth larger than 1 km are envisaged, then shaft sinking is likely a more appropriate technology than drilling.** This technology will not be discussed here, since this is regarded as conventional mining.

State-of-the-art of drilling

The state-of-the-art technologies of the drilling process include directional control, minimizing drill pipe vibrations, down-hole motors, automated drilling systems, and polycrystalline diamond compact drill bits, which will be required to drill and maintain a stable borehole to the necessary specifications. The state-of-the-art in oil and gas industry aims and develops for smallest boreholes as larger boreholes are more costly. Therefore, custom-designed drilling fluid program will be needed to maximize borehole diameter, stability and borehole cleaning in the basement rock. The experts on drilling [181] agreed that, although no 43.2-cm borehole has been drilled to 5-km depth in crystalline basement rock, there appear to be no insurmountable technological barriers to completing such a borehole. At lower depths (3.5 km) larger diameters (e.g. 90 cm) are also possible.

Any drilling technology will use cemented conductor pipes, casing, liner and fluids to stabilize the borehole, to reach the required depth and to facilitate the emplacement of waste containers. The drill bits can be adapted to the rock type.

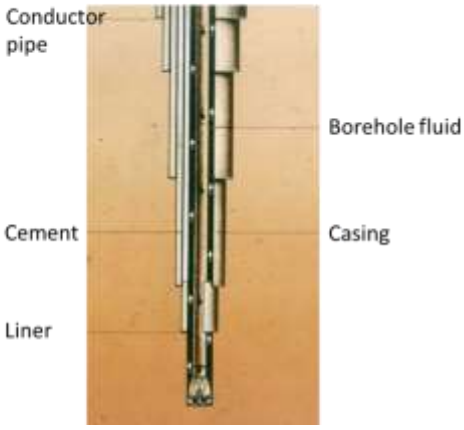



*Figure 104 – Layout of the KTB borehole design* [182]

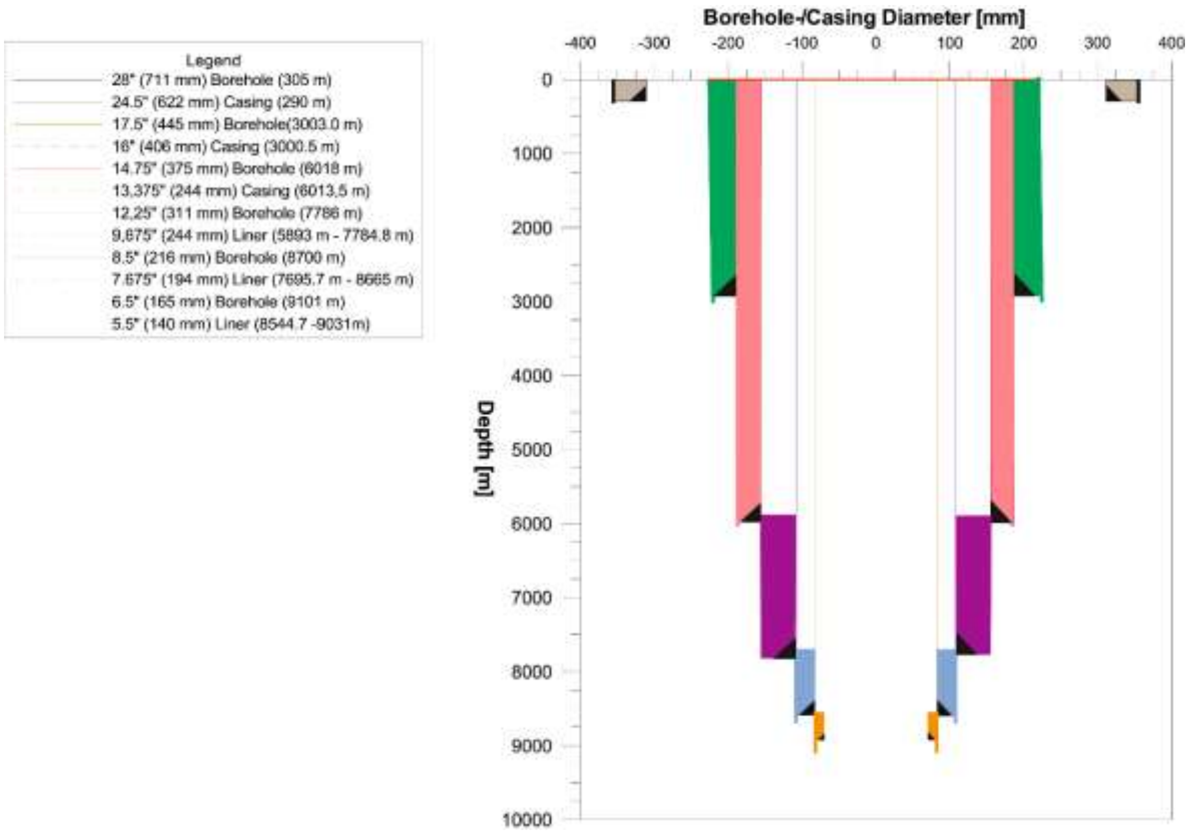



*Figure 105 – Layout of the KTB borehole design* [182]

A rig will be required to perform the drilling and the emplacement of containers. The size of the rig depends on the desired depth and the weight of the containers. Any borehole will start with a larger diameter than needed for the disposal of containers. The layout of the casing design depends on the desired diameter and depth. The borehole diameter as well as the internal diameter of the casing versus depth for the German Continental Deep Drilling Program (KTB) borehole [182] are shown in Figure 104 and Figure 105.

Emplacement

The emplacement of waste canisters must be considered [183] while maintaining radiation protection at operation (*Figure 106*). Possible emplacement modes are [181]:

**Wireline** emplacement -lowering single waste packages on an electric cable.

**Coiled steel tubing** which refers to a very long metal pipe that is supplied spooled on a large reel. The small diameter pipe, normally 25 to 83 mm (1 to 3.25 in) in diameter, is hydraulically driven into the hole, in contrast to the wireline emplacement method, which relies on gravity to lower the package.

**Drill-string** emplacement -lowering a string of packages connected by threaded joints at the end of a standard drill pipe.

**Freefall** or gravity emplacement (briefly discussed in [183]. With a guidance liner running from the surface to total depth, and the borehole filled with an emplacement fluid with controlled properties, it could be possible to allow waste packages to sink freely and slowly into disposal position.

**Conveyance liner** is a large-diameter casing that is sealed at the bottom and held in place at the well head. Waste packages are stacked inside the conveyance liner using a wireline. Then the entire casing is lowered into place using a drill string. To maintain a casing path for the conveyance casing, using the same size waste packages, a larger diameter borehole is needed in the disposal zone than for the other emplacement modes [184].

If deviating or inclined horizontal boreholes (*Figure 107*) are used, free fall (sinking in the fluid) cannot be used as an emplacement mode as containers must be pushed into place. Free fall is also excluded because of concerns that it could be difficult to manipulate a stuck container.

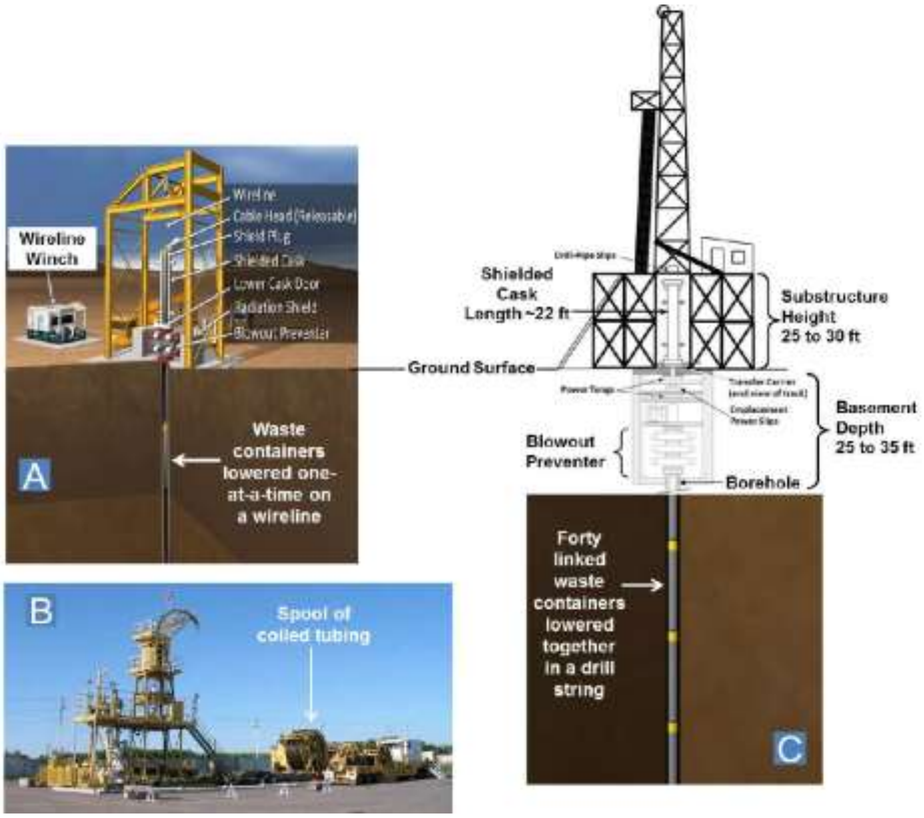



*Figure 106 – Rig and examples for emplacement modes: wireline (A), coiled tubing (B) and drill string (C)*

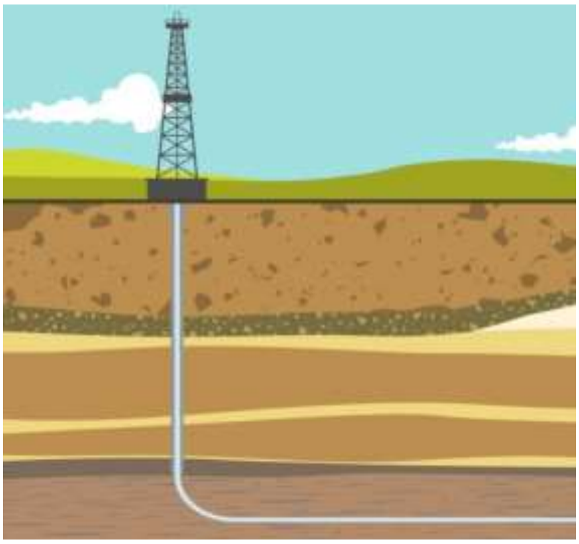


*Figure 107– Horizontal borehole*

<u>Operational and environmental safety</u>

Operational and environmental safety is an important issue and some considerations were taken.

Depending on the waste characteristics and transfer container shielding measures (e.g. by a hydraulic gate, see *Figure 90*, chapter 5.3) may be necessary, if the container with the radioactive waste does not provide sufficient shielding from radiation when it is handled. Once the container is lowered in the fluid of the borehole, the fluid will provide sufficient shielding regarding the disposal facility. After disposal of all container the borehole fluid or other sealing material (e.g. bentonite, clay, salt grit) can provide a sealing of the borehole.

Any disposal concept will have a designated disposal zone. The container will have to pass other rock formations. There is a small risk that a container gets stuck in the casing before reaching the disposal zone or the zone where retention of radionuclide is provided. The installation of a removable liner is proposed as a safety measure for those transfer zones to ensure the retrievability of the stuck container with the removable liner.

Another risk is that a leakage of the container contaminates the borehole fluid with radioactivity. Regular measurements are required to detect any contamination. Using a (removable) liner with a hydraulic gate and a control pipe would provide a further safety measure to recover any contaminated borehole fluid.

A pre-closure (operational) safety assessment still requires R&D.

<u>Wastes Streams</u>

The types of waste that might sensibly be directed to VDBD is similar with all other geological disposal concepts. Any type of waste under consideration should be solid and sufficiently physically and chemically stable to allow its emplacement in a borehole. Some technical attributes that would make a waste type suitable or unsuitable for VDBD is explained in chapter 5.3.

<u>Advantages / Disadvantages</u>

On one hand there are a number of advantages to argue in favour of VDBD, on the other hand, there are considerations that would argue against VDBD.

- Advantage: Speed

VDBD could be developed faster than a GDF and operated on a much reduced timescale (e.g. [185]). A GDF requires at least a decade of site investigations at surface and from underground excavations, and development proceeds slowly as new volumes of rock are entered and characterised. This might be faster for a VDBD option, since operational safety for underground staff is not required. Furthermore, once the access and the underground work areas are constructed, the GDF should remain operational until its eventual closure, even if disposal takes place in campaigns. This could take up to a hundred years in some national programmes. Investigation requirements for rock at some kilometres depth are not as demanding: the VDBD safety concept is based on large-scale properties of the geological environment (e.g., presence of a dense, ancient, isolated deep groundwater system) that require a detailed characterisation by exploration boreholes to exclude uncertainties. A conventional GDF safety concept at lower depth and safety case, for example, can depend on detailed understanding of fracture network properties and the ability to pre-qualify each small volume of rock holding a waste package and additional geotechnical and technical barriers. Constructing a borehole, emplacing waste and sealing it might be achieved within one to two years, after which the disposal borehole does not need to remain operational until a further borehole is required: the concept is suited to modular operation in short campaigns.

- Advantage: Volume

An agency or nation might possess small amounts of waste that would require a GDF and could go to a very deep borehole. This applies for countries with even modest nuclear power programmes. Almost all nations without NPPs do have some wastes that will require geological disposal. If a small amount of SF could go to VDBD, those other wastes might be disposed of in a less heavily engineered, less deep, and less expensive GDF than would otherwise be required, where SF also to be accommodated.

A similar situation could arise if a nation were to be in a partnership with other countries to dispose of wastes in a common GDF. Although there is no real example of a common GDF, the WAC for that GDF might preclude it being used for certain wastes of small volumes or characteristics (military) for which VDBD elsewhere might then be appropriate. This seems an unlikely scenario: any common, shared GDF would be expected to have broad enough WAC to accept all wastes from its users (subject to meeting requirements for safe disposal), otherwise it does not absolve them from having to develop their own independent facilities as well.

- Advantage: Adaptability to waste characteristics

Some types of waste are more difficult to fit into the disposal programme of a conventional GDF and could require either design adaptations or longer operational schedules. For example, the fissile wastes, if emplaced in a GDF that will remain open to permit long operational periods or to satisfy retrievability requirements, will require design and operational adaptations to meet safeguards requirements. This issue affects few national disposal programmes. A more common problem will arise if there is increasing use of MOX fuel, especially with high burn-ups. Spent MOX fuel emits higher quantities of heat for longer periods than conventional UOX SF and requires either design adaptations in the engineered barrier system of a GDF, protracted operational periods involving programme schedule adaptations, or prolonged storage prior to disposal. Availability of a VDBD solution, which is much less sensitive to thermal factors, could become attractive for countries that move to a MOX fuel cycle.

- Advantage: Politics and society

VDBD has not only technical aspects [66]. A variety of political and societal reasons might be envisaged why it could be advantageous, or even required, to dispose of wastes **at or close to the location where they have been generated or are stored**. VDBD is intrinsically rather flexible in terms of site suitability: the kilometres-deep geological environment **required for VDBD is considerably less sensitive to location than the shallow environment requirements for a GDF**, where siting is much more critical. Consequently, VDBD could be countenanced at many locations. There are some attractions in having the waste generated at a site, being disposed at that site and in the area that has benefitted from the presence of the NPP or other nuclear facility. Where there have been obvious disbenefits for the nuclear facility, the argument runs in the opposite direction. The strategic and political factors that might argue in favour of opting for VDBD are considerably more complex and potentially constraining than the purely technical attributes.

- Advantage: economics

For some waste management programmes, an important driver could be the overall economics of disposal. For the scheduling and operational scale reasons discussed above, as well as the limited excavation and engineering involved, VDBD is advanced as being a considerably less expensive option than a full-scale GDF. If some of the other factors discussed above also apply (e.g., no GDF will be available for decades), then opting for the less expensive and earlier solution might be attractive. One of the other economic factors that comes into play for a country having a mixture of disposal option is that it could be possible to close the different facility at different times. A VDBD could accept HLW or SF that is less cooled and therefore not suitable for a GDF. Disposal could be implemented earlier—a GDF that had been used mainly for decommissioning ILW would not then be 'waiting' (possibly for many decades) until SF was ready to be emplaced (see, also, the comments above on MOX fuel). Of course, the discounted costs of protracted storage of wastes until some indefinite time in the future when a GDF does come on-line, are arguably small compared to any form of geological disposal, including VDBD. At this point, other considerations come into play: who is responsible for the costs; what is the policy on intergenerational equity and passing on burdens to future generations etc.

- Advantage: Operational and long-term safety

All of the borehole disposal operations are carried out from the surface, from the digging and the handling of the packages to the closing operations. The increased depth and additional geological barrier may contribute to a high-quality long-term isolation of the waste from the biosphere.

- Disadvantage: Existing programme

If a nation is already planning a GDF, has a functioning implementation programme to ensure that it will eventually have one and the design concept and WAC are able to accept all relevant wastes in the national inventory, then it could be an easy decision to wait until the GDF is available. Accepting this argument requires trust in the institutions charged with managing the wastes — in particular, their longevity (will they be here to get the job done), the long-term availability of funds to complete the work (the government has not decided to use the funds for something else) and the stability of the national legal and planning framework that permits the GDF over many future decades and changes of government.

- Disadvantage: Retrievability

Some national policies require that wastes remain retrievable for some period after they are disposed of. The origin of this requirement is entirely societal. In order to progress with public engagement and public approval to proceed with GDF development, a retrievability/ back-stop has been accepted into policy, in case the GDF does not function to expectations or in case a preferable solution becomes available. There is a distinction between practical retrieval from a GDF and a VDBD facility. In a GDF, SF and HLW are emplaced together with the engineered barrier system and backfills and are isolated by plugs and seals into disposal tunnels and boreholes. They are in their intended final state with respect to the disposal and safety concept. Sections of the GDF are left open to allow access through plugs and backfill and make retrievability achievable (if difficult and potentially hazardous). In a VDBD facility, retrieval is only possible if the borehole is left unfinished and the wastes are not properly isolated by plugs and seals (although periodic plugs might be essential anyway within the disposal region of the hole). A not properly sealed borehole seems difficult to justify or license and would not be a sensible choice. While it has been shown that a package can be emplaced and then, shortly afterwards, removed from a deep borehole using oilfield technology, this demonstration is a far cry indeed from the concept of being able to retrieve all the waste many decades after it has been disposed. One might find the emplaced waste containers using modern drilling technology and available data, but corrosion processes on containers and casing make likely it impossible to render a retrieval of containers.

Retrievability is diametrically counter to the concept of VDBD, which aims at the maximum isolation possible by specifically making it practically impossible to retrieve waste once a borehole is sealed. **Consequently, if retrievability is an immutable part of national policy, VDBD is not an option for consideration.**

- Disadvantage: Siting

Siting is not only a technical issue. As almost every national GDF programme has found out to its cost, in terms of delays and project cancellations, siting is dominated by societal concerns and thus enters the sphere of local and national politics. Even though VDBD can provide an apparently higher level of isolation, is much more remote from the surface and has smaller environmental impacts (e.g., spoil, operational period) than a GDF, there is little reason to believe that a VDBD facility would be any easier to site than a GDF. In several countries, society is also now sensitised to activities in deep boreholes by hydraulic fracturing of boreholes for natural gas production. Siting a centralised national VDBD facility using a nationwide screening and volunteering approach would likely encounter exactly the same problems as siting a GDF. However, a key aspect of the flexibility of VDBD siting is the potential to locate a facility within the boundaries of an existing nuclear site where waste is stored. In principle, and although notoriously unpredictable, the socio-political process of siting and licensing ought to be more straightforward and less contentious than a 'new', centralised site for a VDBD facility or a GDF. This factor again points towards VDBD, if and when implemented, likely being a 'boutique' solution — handling small volumes of waste in one or two boreholes, at or close to the site of waste generation. Where this site is a major, historical nuclear legacy facility undergoing extensive remediation, then this approach seems potentially realistic. On the other hand, the model advanced by some proponents in the USA for local disposal of SF at each NPP location, whilst attractive from the viewpoint of logistics, would seem likely to run into exactly the same problems as siting any GDF — multiplied ten-fold across thirty states.

- Disadvantage: Feasibility and Demonstration

Liquid radioactive wastes were disposed into the very deep geological underground, but this technology is discarded. For solid wastes no known feasibility studies or demonstrations were done for VDBD. Therefore, there is a lack of practical experience.

Current status

Very deep boreholes for disposal of radioactive waste still seems to be a sensible approach to construct a GDF. There are realistic perspectives of a feasible technology to drill larger boreholes with larger diameters in greater depths to accommodate waste packages, when there is sufficient interested. The oil and gas industry aims for smaller boreholes because of cost issues. A current overview on the status of VDBD is provided in Energies 2020 [186]. The content covered topics on:

- waste types as outlined above [126]
- engineered barrier performance for vertical [187] and horizontal [188] boreholes. Borehole seals in salt rock were not presented but could be interesting alternative [189]
- performance assessments, including the thermal evolution around heat-generating waste for a wide range of material properties and disposal configurations [190], geochemical analyses of deep brines focussed on fluid-rock interactions [191] and post-closure performance assessment calculations for the DBD of Cs/Sr capsules [192],
- the safety case, including deep horizontal drill-holes in sedimentary, metamorphic or igneous rocks [193] and safety case aspects for deep vertical boreholes [132].

This provides a sufficient knowledge base to plan and execute field-based demonstration projects that address key elements of borehole disposal, including deep drilling of wide-diameter holes, waste emplacement testing and seal emplacement and performance monitoring.

VDBD gets still attention in several countries: Australia [194], United States [195], Sweden. The ERDO-Working group (www.erdo-wg.com), Australia[5] and Norway[6] are currently (2020) planning studies or studying VDBD in more detail for low- and intermediate-level radioactive waste.

Australia has to dispose some vitrified radioactive waste returning from France and some small amount of other radioactive waste. A recent estimate of volume showed that one borehole of 2 km could be sufficient for disposal of the waste.

Norway is decommissiong the Halden reactor and also has small amounts of low-level waste in drums, which could be disposed in boreholes.

Summary

VDBD is considered in some countries but has not come to realization yet. It has some advantages that makes this option interesting for SIMS. Depths greater than approx. 3,500 m seems not to be sensible approach as the technical effort and uncertainty in geological characterisation may outweigh the possible gain in safety.

[5] Personal communication with Dirk Mallants, CSIRO.
[6] Personal communication with Håvard Kristiansen, Norwegian Nuclear Decommissioning.

# References


[1] EURAD, „D9.12 - ROUTES - Studies and plans for developing shared solutions for radioactive waste management in Europe," EURAD, 2020.

[2] RAD, „D9.13 - ROUTES - Case studies of shared development and use of technologies and facilities," EURAD, 21.

[3] EURAD, „D9.14 - ROUTES - Report on the feasability of developing further European shared solutions for waste management from cradle to grave," EURAD, 2022.

[4] European Commission - CORDIS, "Support action: pilot initiative for european regional repositories (SAPIERR)," [Online]. Available: https://cordis.europa.eu/project/id/509071. [Accessed 12 04 2022].

[5] European Commission - CORDIS, "Strategy action plan for implementation of european regional repository - Stage 2 (SAPIERR-2)," [Online]. Available: https://cordis.europa.eu/project/id/35958. [Accessed 12 04 2022].

[6] V. W. u. Tokarevskyi, „Overview of existing work on categorization/classification of RWs in participating states. Final version as of 19.05.2021 of deliverable D9.4 of the HORIZON 2020 project EURAD. EC Grant agreement no: 847593," EURAD, 2021.

[7] International Atomic Energy Agency, "Underground Disposal Concepts for Small Inventories of Intermediate and High Level Radioactive Waste. TECDOC-1934," IAEA, Vienna, 2020.

[8] International Atomic Energy Agency, „Selection of efficient options for processing and storage of radioactive waste in countries with small amounts of waste generation, TECDOC-1371," IAEA, Vienna, 2003.

[9] International Atomic Energy Agency, "Classification of Radioactive Waste, IAEA Safety Standards Series No. GSG-1," IAEA, Vienna, 2006.

[10] International Atomic Energy Agency, „Predisposal management of radioactive waste from nuclear power plants and research reactors, IAEA Safety Standards, Specific Safety Guide No. SSG-40," IAEA, Vienna, 2016.

[11] International Atomic Energy Agency, "Country Nuclear Waste Profile - NEWMDB - IAEA, Netherlands," IAEA, [Online]. Available: https://newmdb.iaea.org/profiles.aspx?ByCountry=NL. [Accessed 22 03 2022].

[12] International Atomic Energy Agency, „Management of radioactive waste from the use of radionuclides in medicine, TECDOC-1183," IAEA, Vienna, 2000.

[13] International Atomic Energy Agency, „Modular design of processing and storage facilities for small volumes of low and intermediate level radioactive waste including disused sealed sources," IAEA, Vienna, 2014.

[14] International Atomic Energy Agency, „Management of disused sealed radioactive sources," IAEA, Vienna, 2014.

[15] International Atomic Energy Agency, „Management of NORM Residues - TECDOC-1712," IAEA, Vienna, 2013.

[16] F. Rechberger, E. Lindner, M. Dauke, M. Tatzber, C. Landstetter und C. Katzlberger, „Evaluation and survey of NORM legacy sites in Austria," in *Naturally occuring radioactive material (NORM VIII) Proceedings of an international symposium Rio de Janeiro, Brazil, 18-21 October 2016*, Brazil, 2018.

[17] M. Dauke und C. Katzlberger, „Remediation of a NORM contaminated site in Austria," in *Naturally occuring radioactive material (NORM VIII) Proceedings of an International Symposium Beijing, China, 22-26 April 2013*, Vienna, 2015.

[18] European Commission, „Effluent and dose control from European Union NORM industries assessment of current situation and proposal for a harmonised community approach volume 2: Apprendices," Directorate-General for Energy and Transport Directorate H - Nuclear Safety and Safeguards Unit H.4 - Radiation Protection, 2003.

[19] International Atomic Energy Agency, „New developments and improvements in processing of 'problematic' radioactive waste, Results of a coordinated research project 2003-2007 - TECDOC-1579," IAEA, Vienna, 2007.

[20] R. Robbins und P. Ormai, „Management of problematic low and intermediate level radioactive wastes - Introduction," in *Technical meeting of IPN and DISPONET Networks on the management of radioactive waste streams that present specific challenges, 28 Nov - 02 Dec 2016*, Vienna, 2016.

[21] European Commission, „Inventory of radioactive waste and spent fuel present in the community's territory and the future prospects - SWD 2019 - 435 final," EC, Brussels, 2019.

[22] VGB PowerTech e.V., „Entsorgung von Kernkraftwerken - Eine technisch gelöste Aufgabe," VGB PowerTech e.V., 2016.

[23] International Atomic Energy Agency, „Data analysis and collection for costing of research reactor decommissioning - Final report of the DACCORD collaborative project - Nuclear Energy Series Report - No. NW-T-2.12,“ IAEA, Vienna, 2021.

[24] „National Report of Denmark - Joint Convention,“ 2018. [Online]. Available: https://www.sst.dk/-/media/Udgivelser/2017/Joint-Convention-6-IAEA/Joint-convention-on-the-safety-of-spent-fuel-management-safety-of-radioactive-waste-management.ashx?la=da&hash=32BFD42810393E617AC992562C296E949DB8FDA5.

[25] „National report of the Netherlands-Joint Convention,“ 2017. [Online]. Available: https://www.iaea.org/sites/default/files/national_report_of_netherlands_for_the_6th_review_meeting_-_english.pdf.

[26] „National report of Poland - Joint Convention,“ 2017. [Online]. Available: https://www.iaea.org/sites/default/files/national_report_of_poland_for_the_6th_review_meeting_-_english.pdf.

[27] „National report of Slovenia - Joint Convention,“ 2017. [Online]. Available: https://www.iaea.org/sites/default/files/national_report_of_slovenia_for_the_6th_review_meeting_-_english.pdf.

[28] ROUTES, "D9.7 Review of radioanalytical characterisation of selected radioactive wastes and wastes with complex chemical and toxic properties," EURAD, 2022.

[29] EURAD, „Milestone 88: Current use of waste acceptance criteria (WAC) in European Union Member-States and some Associated Countries, Work Package 9 (ROUTES) Task 4 (sub-Task 4.1),“ EURAD, EC grant agreement No. 847593.

[30] EURAD, „Workpackage 9 ROUTES, Task 5: Minutes of the Workshop, March 5-6 Athens, Greece“.

[31] United Nations Economic Commission for Europe (UNECE) Inland Transport Commitee, „European agreement concerning the international carriage of dangerous goods by road - ADR,“ UNECE, 2019.

[32] Intergovernmental Organisation for International Carriage by Rail (OTIF), *Convention concerning International Carriage by Rail (COTIF) Appendix C – Regulations concerning the International Carriage of Dangerous Goods by Rail (RID),* OTIF, 2019.

[33] United Nations Economic Commission for Europe (UNECE), “European Agreement Concerning The International Carriage of Dangerous Goods by Inland Waterways, effective from February 2008 (ADN),” UNECE, 2008.

[34] International Maritime Organization, “International Maritime Dangerous Goods Code 2018 Edition (incorporating Amendment 39-18) IL200E* (IMDG),” International Maritime Organization, 2018.

[35] Federal Ministry of Austria Climate Action, Environment, Energy, Mobility, Innovation and Technology, “Seventh National Report of Austria on the implementation of the obligations of the Joint Convention on the Safety of Spent Fuel and on the Safety of Radioactive Waste Management,” Vienna, 2020.

[36] Republic of Cyprus, Ministry of Labour, Welfare and Social Insurance, Department of Labour Inspection, Radiation Inspection and Control Service, „National report on the implementation of the obligations under the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management,“ 6th Review Meeting on the Convention Vienna, 21 May - 1 June 2018, Vienna, 2018.

[37] G. Krause, A. Stenmark and B. Amcoff, “Waste Acceptance Criteria – Incineration Services, Cyclife Sweden AB Report C-16/001,” Cyclife, 2016.

[38] Danish Health Authority - Radiation Protection, “Joint convention on the safety of spent fuel management and on the safety of radioactive waste management from Denmark and Greenland,” Sundhedsstyrelsen, 2017.

[39] IRSN, „Waste management routes in Europe from cradle to grave, WP ROUTES Y1, Minutes of workshops held in Athens, 2-6 March 2020,“ EURAD, 2020.

[40] National Atomic Energy Agency of Poland (PAA), „National report of Republic of Poland on compliance with obligations of the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management, Polish 6th national report,“ PAA, 2017.

[41] Nuclear Energy Agency, „Radioactive waste management programmes in OECD/NEA member countries: Poland,“ NEA, 2011.

[42] National Atomic Energy Agency (PPA), „Annual report - Activities of the President of the National Atomic Energy Agency and assessment of nuclear safety and radiological protection in Poland in 2016,“ PPA, 2016.

[43] Slovenian Nuclear Safety Administration, „Sixth Slovenian Report under the Joint Convention on the Safety of Spent Fuel Management and on the Safety Radioactive Waste Management," Slovenian Nuclear Safety Administration, Ljubljana, 2017.

[44] *Radioactive Waste Management and Decommissioning in Slovenia – National Framework for Management and Regulation of radioactive waste and Decommissioning,* OECD-NEA, 2020.

[45] International Atomic Energy Agency, *Regulations for the Safe Transport of Radioactive Material - Specific Safety Requirements No. SSR-6 (Rev. 1),* Vienna: IAEA, 2018.

[46] *Radioactive waste management programmes in OECD/NEA member countries – Belgium,* OECD-NEA, 2010.

[47] The Republic Bulgaria, *Sixth National Report on Fulfilment of the Obligations under the Joint Convention on the Safety of Spent Fuel Management and on Safety of Radioactive waste Management,* Sofia, 2017.

[48] SÚJB, „BEZPEČNOSTNÍ NÁVODY SÚJB - bezpečné využívání jaderné energie a ionizujícího záření: Přeprava radioaktivních látek, jaderná bezpečnost, BN-JB-TR-1.1(Rev. 1.0)," Prague, 2018.

[49] International Atomic Energy Agency, „IAEA SAFETY STANDARDS APPLICATIONS - TRANSAS-6: Appraisal for France of the Safety of the Transport of Radioactive Material," IAEA, Vienna, 2004.

[50] C. Bucur, A. Dodaro, B. Ferruci, G. Meskens, D. Ricard, H. Tietze-Jaensch, P. Thomas und C. Turcanu, „Synthesis of commonly used methodology for conditioned radioactive weaste characterization - Deliverable D2.2 - Work Package 2 - CHANCE project (Contract-Number 755371)," CHANCE, 2019.

[51] Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU), Division S III, „Report of the Federal Government for the Seventh Review Meeting in May 2021 on the fulfilment of the obligations of the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management," BMU, Bonn, 2020.

[52] B. Nös, P. Molnár und A. Baksay, „Disposal of low and intermediate level waste in Hungary," Public Agency for Radioactive Waste Management, Rud.-geol.-naft. Vol. 24, Budaörs, 2012.

[53] T. Katona, K. Horvath und J. Safar, „New safety and security requirements for the transport of nuclear and other radioactive materials in Hungary," Hungarian Atomic Energy Authority, Budapest, 2016.

[54] Lithuanian State Nuclear Power Safety Inspectorate (VATESI), „Lithuanian National Report Under the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management, Fifth Lithuanian National Report ISSN 2351-6445," VATESI, 2017.

[55] "ENSREG - Lithuania," [Online]. Available: http://www.ensreg.eu/country-profile/Lithuania. [Accessed 2021].

[56] Lithuania, „Rules on the Issue of Certificates for Transport of Nuclear Fuel Cycle, Nuclear and Fissionable Materials - Nuclear Safety Requirements - BSR-4.1.1-2017," 2019.

[57] Minister of Health - Lithuania, „Order of the Minister of Health and the Head of the State Nuclear Power Safety Inspectorate No. V-1271/22.3-139 on the Rules on Import, Export, Transit and Transport of Radioactive Material, Radioactive Waste and Spent Nuclear Fuel," VATESI, 2020.

[58] U. N. E. C. f. E. (UNECE), „European Agreement Concerning The International Carriage of Dangerous Goods by Inland Waterways (ADN)," UNECE, 2008.

[59] Consejo de Seguridad Nuclear (CSN), „Transport of radioactive materials, Practical documentation for professionals, Revision 9.0," CSN, 2019.

[60] Ministry of the Environment Sweden, „Sweden's sixth national report under the Joint Convention on the safety of spent fuel management and on the safety of radioactive waste management," 2017.

[61] T. Nilsson, "Information on legal requirements for Transport of Radioactive Material under Swedish jurisdiction, Document: 14-1066," Swedish Radiation Safety Authority, 2019.

[62] Integrated Regulatory Review Service (IRRIS), „Report to Government of Ukraine," IRRIS, Kiev, 2008.

[63] Department for Business, Energy and Industrial Strategy, „The United Kingdom's Sixth National Report on Compliance with the Obligations of the Joint Convention on the Safety of Spent Fuel and Radioactive Waste Management," 2017.

[64] „National report of Republic of Poland, Polish 7th national report as referred to in Article 32 of the Joint Convention," National Atomic Energy Agency, 2020.

[65] M. J. Krizman und M. Kralj, „Long-term Surveillance of the Disposal Site for Uranium Mine," in *27th International Conference Nuclear Energy for New Europe*, Portoroz, 2018.

[66] Nuclear Energy Agency, "Radioactive Waste Management and Decommissioning in Slovenia - Retrieved from Radioactive waste management and decommissioning programmes in NEA member countries," NEA, 2021.

[67] IAEA, „Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management - National Report from Norway to the sicth review meeting," IAEA, 2008.

[68] Nuclear Energy Agency, "Radioactive waste management and decommissioning programmes in NEA member countries: Norway," [Online]. Available: https://www.oecd-nea.org/jcms/pl_33756/norway-profile-web. [Accessed 26 04 2021].

[69] Norwegian Radiation Protection Authority, „Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management - National Report from Norway," 2021.

[70] ERDO working group, "ERDO: Working on a shared solution for radioactive waste - Retrieved from ERDO Latest Activities and News," [Online]. Available: http://www.erdo-wg.com/. [Accessed 26 04 2021].

[71] ERDO, "Association for Multinational Radioactive Waste Solutions," [Online]. Available: https://www.covra.nl/app/uploads/2021/01/ERDO_brochure.pdf. [Accessed 26 04 2021].

[72] International Atomic Energy Agency, „Disposal of radioactive waste for protecting people and the environment - No. SSR-5," IAEA, Vienna, 2011.

[73] International Atomic Energy Agency, „Technical Considerations in the Design of Near Surface Disposal Facilities for Radioactive Waste - TECDOC-1256," IAEA, Vienna, 2001.

[74] International Atomic Energy Agency, „Disposal Aspects of Low and Intermediate Level Decommissioning Waste: Results of a coordinated research project 2002–2006 - TECTDOC-1572," IAEA, Vienna, 2007.

[75] World Nuclear Association, "Storage and Disposal of Radioactive Waste," WNA, [Online]. Available: https://www.world-nuclear.org/information-library/nuclear-fuel-cycle/nuclear-waste/storage-and-disposal-of-radioactive-waste.aspx. [Accessed 11 2021].

[76] ANDRA, "The CIRES," ANDRA, [Online]. Available: https://international.andra.fr/operational-facilities/industrial-facility-grouping-sorting-and-very-low-level-waste-vllw-disposal-facility-cires/cires. [Accessed 01 01 2022].

[77] ANDRA, "Our Missions," ANDRA, [Online]. Available: https://international.andra.fr/about-andra/our-missions. [Accessed 01 01 2022].

[78] International Atomic Energy Agency, „Design Principles and Approaches for Radioactive Waste Repositories, IAEA Nuclear Energy Series NW-T-1.27," IAEA, Vienna, 2020.

[79] ANDRA, "Manche disposal facility (CSM)," ANDRA, [Online]. Available: https://international.andra.fr/existing-facilities/manche-disposal-facility-csm. [Accessed 01 01 2022].

[80] ANDRA, "Operational facilities," ANDRA, [Online]. Available: https://international.andra.fr/operational-facilities. [Accessed 01 01 2022].

[81] Enresa, "History of El Cabril," Enresa, [Online]. Available: https://www.enresa.es/eng/index/activities-and-projects/el-cabril. [Accessed 01 01 2022].

[82] Enresa, "National Inventory," Enresa, [Online]. Available: https://www.enresa.es/eng/index/activities-and-projects/national-inventory. [Accessed 01 01 2022].

[83] R. D. Bennett, W. O. Miller, J. B. Warriner, P. G. Malone und C. C. McAneny, „Alternative Methods for Disposal of Low-Level Radioactive Wastes - NUREG/CR-3774-Vol.1, DE84/901100," NRC, 19984.

[84] R. Pusch, R. N. Yong und M. Nakano, Geologic Disposal of High-Level Radioactive Waste, Boca Raton: CRC Press, 2018.

[85] V. Popov, „Disposal of radioactive waste in abandoned mines," *Journal of Earth Sciences and Geological Engineering, Vol. 9, No. 3,* pp. 1-38, 2019.

[86] World Nuclear Association, "Country Profiles - Nuclear Power in Switzerland," 03 2020. [Online]. Available: https://www.world-nuclear.org/information-library/country-profiles/countries-o-s/switzerland.aspx. [Accessed 07 2020].

[87] NAGRA, "Site Selection Process," [Online]. Available: https://www.nagra.ch/en/site-selection-process. [Accessed 2020].

[88] NAGRA, „Technischer Bericht NTB 93-26, Endlager für kurzlebige schwach- und mittelaktive Abfälle (Endlager SMA) Beurteilung der Langzeitsicherheit des Endlagers SMA am Standort Wellenberg (Gemeinde Wolfenschiessen)," NAGRA, 1993.

[89] NAGRA, „Technischer Bericht NTB 93-28, Untersuchungen zur Standorteignung im Hinblick auf die Endlagerung schwach- und mittelaktiver Abfälle, Geologische Grundlagen und Datensatz zur Beurteilung der Langzeitsicherheit des Endlagers SMA am Standort Wellenberg," NAGRA, 1993.

[90] State Office for Nuclear Safety (Státní úřad pro jadernou bezpečnost), "National Report under the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management," State Office for Nuclear Safety, Prague, 2020.

[91] J. Feinhals und J. Mikšova, *Notes from RWM solutions for small amounts of waste, ROUTES Workshop, March 6th 2020,* Athens, Greece, 2020.

[92] International Atomic Energy Agency, „Borehole disposal facilities for radioactive waste, specific safety guide, IAEA Safety Standards series No. SSG-1," IAEA, Vienna, 2009.

[93] N. Železnik und J. Mikšova, *Notes from RWM solutions for small amounts of waste, Near surface disposal, ROUTES Workshop, March 6th 2020,* Athens, Greece, 2020.

[94] N. Laverov, S. Yudintsev, B. Kochkin und V. Malkovsky, „The Russian Strategy of using Crystalline Rock as a Repository for Nuclear waste," *Elements (2016) 12 (4),* pp. 253-256, 2016.

[95] U. Bergström, K. Pers und Y. Almén, „International perspective on repositories for low level waste, R-11-16," Svensk Kärnbränslehantering AB SKB, Stockholm, 2011.

[96] Svensk Kärnbränslehantering AB SKB, "Extending the SFR," 01 2018. [Online]. Available: https://skb.se/upload/publications/pdf/Fact-sheet_Extending_the_SFR.pdf. [Accessed 08 2020].

[97] Svensk Kärnbränslehantering AB SKB, "SFR - Final repository for short-lived radioactive waste," 11 2016. [Online]. Available: https://skb.se/upload/publications/pdf/SFR_folder_engelsk.pdf. [Accessed 08 2020].

[98] Finnish Energy Industries, „Nuclear Waste Management in Finland," 2007. [Online]. Available: https://cienciaensocietat.org/wp-content/uploads/2018/10/14-Nuclear-Waste-Management-in-Finland.pdf.

[99] ARAO, "Planning of a low- and intermediate-level radioactive waste disposal facility," ARAO, 12 2018. [Online]. Available: https://www.arao.si/index.php/en/activities/planning-of-a-low-and-intermediate-level-radioactive-waste-disposal-facility. [Accessed 08 2020].

[100] ARAO, "Odlagališče nizko- in srednjeradioaktivnih odpadkov (NSRAO) v Sloveniji ZAKAJ, KAKO, KJE IN KDAJ? Informacije za radovedne," 12 2016. [Online]. Available: https://www.arao.si/images/pdf/ARAO_NSRAOzgibanka_WEB_corr%202018(1).pdf. [Accessed 08 2020].

[101] Bundesgesellschaft für Endlagerung, "Geschichte des Endlagers Konrad," BGE, [Online]. Available: https://www.bge.de/de/konrad/kurzinformationen/geschichte-des-endlagers-konrad/. [Accessed 17 11 2020].

[102] Bundesgesellschaft für Endlagerung, "Wichtige Ergebnisse der Überprüfung," BGE, [Online]. Available: https://www.bge.de/de/konrad/themenschwerpunkte/themenschwerpunkt-fertigstellung-des-endlagers-konrad/ergebnisse-der-ueberpruefung-und-umsetzung/. [Accessed 17 11 2020].

[103] Niedersächsisches Umweltministerium, "Planfeststellungsbeschluss für die Errichtung und den Betrieb des Bergwerkes Konrad in Salzgitter als Anlage zur Endlagerung fester oder verfestigter radioaktiver Abfälle mit vernachlässigbarer Wärmeentwicklung vom 22. Mai 2002," 22 05 2002. [Online]. Available: https://www.bge.de/fileadmin/user_upload/Konrad/Wesentliche_Unterlagen/Genehmigungsunterlagen/Planfeststellungsbeschluss_Endlager_Konrad_vom_22_Mai_2002.pdf. [Accessed 17 11 2020].

[104] Bundesamt für Strahlenschutz, "Anforderungen an endzulagernde radioaktive Abfälle (Endlagerungsbedingungen, Stand Dezember 2014), Endlager Konrad," BfS, 2014.

[105] Bundesgesellschaft für Endlagerung, "Geschichte des Endlagers Morsleben," BGE, 2020. [Online]. Available: https://www.bge.de/de/morsleben/kurzinformationen/geschichte-des-endlagers-morsleben/. [Accessed 17 11 2020].

[106] Bundesamt für Strahlenschutz, "Planfeststellungsverfahren zur Stilllegung des Endlagers für radioaktive Abfälle Morsleben: Plan für die Umrüstung und den Offenhaltungsbetrieb des Endlagers für radioaktive Abfälle," 2005. [Online]. Available: https://archiv.bge.de/archiv/www.endlager-morsleben.de/SharedDocs/Downloads/Morsleben/DE/planfeststellungsverfahren/lfdnr111-901-00-v01b1dd.pdf?__blob=publicationFile&v=1. [Accessed 17 11 2020].

[107] Bundesamt für Strahlenschutz, "Planfeststellungsverfahren zur Stilllegung des Endlagers für radioaktive Abfälle Morsleben: Verfüllplan zur Stilllegung des ERAM nach vorgezogener Verfüllung von Grubenbauen des Zentralteils, Konzeptplanung," 2005. [Online]. Available: https://archiv.bge.de/archiv/www.endlager-morsleben.de/SharedDocs/Downloads/Morsleben/DE/planfeststellungsverfahren/lfdnr066-220-00-v01-p220b1dd.pdf?__blob=publicationFile&v=1. [Accessed 17 11 2020].

[108] Bundesamt für Strahlenschutz, "Planfeststellungsverfahren zur Stilllegung des Endlagers für radioaktive Abfälle Morsleben: Entsorgung radioaktiver Abfälle im Endlager für radioaktive Abfälle Morsleben (ERAM), Einlagerungszeitraum 1971 bis 1991," 2006. [Online]. Available: https://archiv.bge.de/archiv/www.endlager-

morsleben.de/SharedDocs/Downloads/Morsleben/DE/planfeststellungsverfahren/lfdnr091-252-00-v01-p252b1dd.pdf?__blob=publicationFile&v=1. [Accessed 17 11 2020].

[109] Bundesamt für Strahlenschutz, "Planfeststellungsverfahren zur Stilllegung des Endlagers für radioaktvite Abfälle Morsleben: Entsorgung radioaktiver Abfäle im Endlager für radioaktive Abfälle Morsleben (ERAM), Einlagerungszeitraum 1994 bis 1998," 2007. [Online]. Available: https://archiv.bge.de/archiv/www.endlager-morsleben.de/SharedDocs/Downloads/Morsleben/DE/planfeststellungsverfahren/lfdnr092-253-00-v01-p253b1dd.pdf?__blob=publicationFile&v=1. [Accessed 17 11 2020].

[110] Bundesgesellschaft für Endlagerung, "Radioaktive Abfälle im Endlager Morsleben," BGE, 2020. [Online]. Available: https://www.bge.de/de/morsleben/kurzinformationen/radioaktive-abfaelle-im-endlager-morsleben/. [Accessed 17 11 2020].

[111] Bundesgesellschaft für Endlagerung, "Stilllegung des Endlagers Morsleben," BGE, 2020. [Online]. Available: https://www.bge.de/de/morsleben/kurzinformationen/stilllegung-des-endlagers-morsleben/. [Accessed 17 11 2020].

[112] Arbeitsgemeinschaft Schacht Konrad e.V., "Atommüllreport "ERA Morsleben"," 24 09 2020. [Online]. Available: https://www.atommuellreport.de/daten/detail/era-morsleben.html. [Accessed 17 11 2020].

[113] Bundesamt für Strahlenschutz, "Asse II: Vom Salzbergwerk zum Atomlager - Die wechselvolle Geschichte der Schachtanlage Asse II," BfS, 09 08 2016. [Online]. Available: https://archiv.bge.de/archiv/www.asse.bund.de/Asse/DE/themen/was-ist/geschichte/geschichte_no. [Accessed 17 11 2020].

[114] Bundesamt für Strahlenschutz, "Asse II: Das Bergwerk - Geologie und Stabilitätsprobleme," BfS, 27 02 2017. [Online]. Available: https://archiv.bge.de/archiv/www.asse.bund.de/Asse/DE/themen/was-ist/bergwerk/bergwerk_node.html. [Accessed 17 11 2020].

[115] Bundesgesellschaft für Endlagerung, "Radioaktive Abfälle in der Schachtanlage Asse II," BGE, 2020. [Online]. Available: https://www.bge.de/de/asse/kurzinformationen/radioaktive-abfaelle-in-der-schachtanlage-asse-ii/.

[116] Bundesamt für Strahlenschutz, "Asse II: Die radioaktiven Abfälle in der Schachtanlage Asse II," BfS, 15 08 2016. [Online]. Available: https://archiv.bge.de/archiv/www.asse.bund.de/Asse/DE/themen/was-ist/radioaktive-abfaelle/radioaktive-abfaelle_node.html. [Accessed 17 11 2020].

[117] Arbeitsgemeinschaft Schacht Konrad e.V., "Atommüllreport "Asse II"," 24 09 2020. [Online]. Available: https://www.atommuellreport.de/daten/detail/asse-ii.html.

[118] Bundesgesellschaft für Endlagerung, "Rückholung der radioaktiven Abfälle," BGE, 2020. [Online]. Available: https://www.bge.de/de/asse/kurzinformationen/rueckholung-der-radioaktiven-abfaelle/.

[119] Landesamt für Bergbau, Energie und Geologie - Geozentrum Hannover, "Kavernen - die unsichtbaren Energiespeicher," Abt. Wirtschafts- und Umweltgeologie, Referat Rohstoffwirtschaft, Steine, Erden, Salz, Hanover, 2017.

[120] T. Fischer, J. Engelhardt und T. Wanne, „Scoping the possibility of ILW disposal in boreholes (Technical Note)," BGE Technology GmbH, 2020.

[121] M. Martell und G. Ferraro, „Radioactive Waste Management Stakeholders Map in the European Union, JRC Report," JRC, 2014.

[122] S. Wallis, „Excavation and operation of Hungary repository - TunnelTalk," 09 07 2020. [Online]. Available: S. Wallis: Excavation and operation of Hungary repository, 9 Jul 2020, , TunnelTalk, https://www.tunneltalk.com/Hungary-09Jul2020-Construction-and-operation-of-Ba%CC%81taapa%CC%81ti-low-and-medium-level-nuclear-waste-repository.php.

[123] P. Ormai, „Joint ICTP-IAEA Workshop on Radioactive Waste Management - Solutions for Countries without Nuclear Power Programme," in *Disposal options of LILW and IAEA technical documents for radioactive waste disposal*, Trieste, Italy, 2015.

[124] T. Äikäs und P. Anttila, „Repositories for low- and intermediate-level radioactive wastes in Finland," *Reviews in Engineering Geology Volume XIX: Deep Geologic Repositories,* 2008.

[125] International Atomic Energy Agency, „Underground Disposal Concepts for Small Inventories of Intermediate- and High-Level Radioactive Waste (Draft)," IAEA, Vienna, 2020.

[126] N. A. Chapman, „Who Might Be Interested in a Deep Borehole Disposal Facility for Their Radioactive Waste?," *Energies, Vol. 12,* pp. 1-13, 2019.

[127] International Atomic Energy Agency, „Generic Post-closure Safety Assessment for Disposal of Disused Sealed Radioactive Sources in Narrow Diameter Boreholes," IAEA, Vienna, 2017.

[128] Deep Isolation, „Spent Nuclear Fuel Disposal in a Deep Horizontal Drillhole Repository Sited in Shale: Numerical Simulations in Support of a Generic Post-Closure Safety Analysis, DI-2020-01-R0," Deep Isolation, 2020.

[129] G. Freeze, E. Stein, L. Price, R. MacKinnon und J. Tillman, „Deep Borehole Disposal Safety Analysis, SANDIA Report SAND2016-10949R," SANDIA National Laboratory, 2016.

[130] International Atomic Energy Agency, „Model Regulations for Borehole Disposal Facilities for Radioactive Waste, TECDOC No. 1827," IAEA, Vienna, 2017.

[131] Danish Decommissioning, „Pre-feasibility study for final disposal of radioactive waste. Disposal concepts. Denmark: COWI," Danish Decommissioning, 2011.

[132] G. Freeze, E. Stein, P. Brady, C. Lopez, D. Sassani, K. Travis, F. Gibb und J. Beswick, „Deep Borehole Disposal Safety Case," *Energies, Vol. 12,* p. 2141, 2019.

[133] Deep Isolation, „Spent Nuclear Fuel Disposal in a Deep Horizontal Drillhole Repository Sited in Shale: Numerical Simulations in Support of a Generic Post-Closure Safety Analysis, DI-2020-01-R0," Deep Isolation Inc., Berkeley, 2020.

[134] AUSIPILE Foundation Contractors and Engineers, "Fully cased drilling with rotary drive," Ausipile, [Online]. Available: https://ausipile.com.au/double-rotary-case-piles/fully-cased-drilling-with-rotary-drive. [Accessed 01 01 2020].

[135] B. S. Aadnøy und M. B. Dusseault, „Deep Borehole Placement of Radioactive Wastes," [Online]. Available: https://www.norskdekommisjonering.no/wp-content/uploads/2020/10/Deep-Borehole-Placement-of-Radioactive-Wastes-A-Feasibility-Study-1.pdf.

[136] T. Fischer, J. Engelhardt und T. Wanne, „Deep Borehole Disposal Concept," BGE-TEC, 2020.

[137] Schlumberger, „Schlumberger oilfield glossary - Casing string," 2020. [Online]. Available: https://glossary.oilfield.slb.com/en/terms/c/casing_string.

[138] H. Kristiansen, *The ERDO borehole project - Presentation 2020-06-10,* 2020.

[139] B. Engeser, „The continental deep drilling project of the Federal Republic of Germany - Documentation of the used drilling technology," Niedersächsisches Landesamt für Bodenforschung, 1996.

[140] J. Beswick, „Status of Technology for Deep Borehole Disposals," EPS International for the Nuclear Decommissioning Authority (NDA), 2008.

[141] J. Beswick, K. Travis und F. Gibb, „Deep borehole disposal of nuclear waste: engineering challenges," in *Proceedings of the Institution of Civil Engineers*, 2014.

[142] N. A. Chapman, "Deep Borehole Disposal of Spent Fuel and Other Radioactive Wastes, NAPSNet Special Reports," 25 07 2013. [Online]. Available: http://nautilus.org/napsnet/napsnet-special-reports/deep-borehole-disposal-of-spent-fuel-and-other-radioactive-wastes/. [Accessed 17 11 2020].

[143] P. B. Leslie, "Summary of NWTRB Deep Borehole Disposal Workshop," 16 12 2015. [Online]. Available: https://www.energy.gov/sites/prod/files/2015/12/f27/2%20Bret%20Leslie.pdf. [Accessed 17 11 2020].

[144] United Kingdom Nirex Limited, „A Review of the Deep Borehole Disposal Concept for Radioactive Waste - Nirex Report N/108," United Kingdom Nirex Limited, 2004.

[145] K. Ahäll, „Final Deposition of High-level Nuclear Waste in Very Deep Boreholes: An evaluation based on recent research of bedrock conditions at great depths," Swedish NGO Office of Nuclear Waste Review (MKG), 2006.

[146] National Academies Press (US), „The Disposal of Radioactive Waste on Land," National Academies Press (US), Washington (DC), 1957.

[147] W. J. Drescher, „Hydrology of Deep-Well Disposal of Radioactive Liquid Wastes," in *Fluids in subsurface environments: A symposium*, Tulsa, Oklahoma, U.S.A., 1965.

[148] M. O'Brien, L. Cohen, T. Narasimhan, T. Simkin, H. Wollenberg, W. Brace, S. Green und H. Pratt, „The Very deep hole concept: evaluation of an alternative for nuclear waste disposal, LBL-7089," Lawrence Berkeley Lab, Berkeley (USA), 1979.

[149] Woodward-Clyde Consultants, Battelle Memorial Institute - Office of Nuclear Waste Isolation, „Very Deep Hole Systems Engineering Studies: Technical Report," National Technical Information Service, 1983.

[150] J. Swahn, „The Long-Term Nuclear Explosives Predicament - The Final Disposal of Militarily Usable Fissile Material in Nuclear Waste From Nuclear Power and From the Elimination of Nuclear Weapons," 1992.

[151] J.-P. Dobrath, A. Rey-Grange, S. Cordier und B. Maurice, „Feasibility study for large diameter boreholes for the deep drilling concept of a hight-level waste repository, Nagra Technischer Bericht 80-04," Nagra, 1980.

[152] Elsam, Elkraft, „Disposal of High-level Waste from Nuclear Power Plants in Denmark. Salt Dome Investigations, 5 vols.," 1981.

[153] C. Juhlin und H. Sandstedt, „Storage of nuclear waste in very deep boreholes: Feasibility study and assessment of economic potential. Part I Geological considerations Part II Overall facility plan and cost analysis, SKB Technical Report 89-39,," 1989.

[154] C. Juhlin, T. Wallroth, J. Smellie, T. Eliasson, C. Ljunggren, B. Leijon und A. Beswick, „The Very Deep Hole Concept – Geoscientific appraisal of conditions at great depth, Technical Report TR-98-05," 1998.

[155] J. Winterle, R. Pauline und G. Ofoegbu, „Regulatory Perspectives on Deep Borehole Disposal Concepts, NRC–02–07–006," U.S. Nuclear Regulatory Commission, 2011.

[156] P. Brady, B. Arnold, G. Freeze, P. Swift, S. Bauer, J. Kanney, R. Rechard und J. Stein, „Deep borehole disposal of high-level radioactive waste, SAND2009-4401," Sandia National Laboratory, 2009.

[157] Kommission Lagerung hoch radioaktiver Abfallstoffe, „Abschlussbericht. Verantwortung für die Zukunft - Ein faires und transparentes Verfahren für die Auswahl eines nationalen Endlagerstandortes, K-Drs 268," Drucksache der Kommission Lagerung hoch radioaktiver Abfallstoffe, 2016.

[158] Bundesamt für die Sicherheit der nuklearen Entsorgung, "Stand von Wissenschaft und Technik bei alternativen Entsorgungsoptionen für hochradioaktive Abfälle," BASE, [Online]. Available: https://www.base.bund.de/SharedDocs/Kurzmeldungen/BASE/DE/2020/0803_alternative-entsorgungsoptionen.html. [Accessed 01 01 2020].

[159] L. Schwartz, J. Cohen und R. Braun, „High level radioactive waste isolation by incorporation in silicate rock, UCRL-78746," 1976.

[160] J. Cohen und T. Steinborn, „The Rock-Melt Approach to Nuclear Waste Disposal in Geological Media," in *Scientific Basis for Nuclear Waste Management: Volume 1 Proceedings of the Symposium on "Science Underlying Radioactive Waste Management", Materials Research Society Annual Meeting*, Boston, Massachusetts, 1978.

[161] H. Hardee und W. Sullivan, „Approximate solution for self-burial rates of radioactive waste containers," 1974.

[162] S. Logan, „Deep Self-Burial of Radioactive Wastes by Rock-Melting Capsules," *Nuclear Technology,* p. 111–124, 1974.

[163] V. Kascheev, A. Nikiforov, P. Poluektov und A. Polyakov, „Towards a theory of self-disposal of high-level waste," *Atomnaya Energiya, Vol. 73,* p. 215–221, 1992.

[164] W. Chen, J. Hao und Z. A. Chen, „Study of Self-Burial of a Radioactive Waste Container by Deep Rock Melting," *Science and Technology of Nuclear Installations,* pp. 1-6, 2013.

[165] R. Klett, „Deep rock nuclear waste disposal test: design and operation," 1974.

[166] L. Bertram, „Mechanics analysis for deep rock disposal of radioactive wastes," 1976.

[167] J. Angelo, „Nuclear waste management by in-situ melting," 1976.

[168] F. Gibb, „High-temperature, very deep, geological disposal: a safer alternative for high-level radioactive waste?," in *Waste Management 1999*, 1999.

[169] F. Gibb und P. Attrill, „Granite recrystallization: The key to the nuclear waste problem?," *Geology (2003) 31,* pp. 657-660, 2003.

[170] P. Attrill und F. Gibb, „Partial melting and recrystallization of granite and their application to deep disposal of radioactive waste: Part 1 - Rationale and partial melting," *Lithos (2003) 67,* p. 103–117, 2003.

[171] P. Attrill und F. Gibb, „Partial melting and recrystallization of granite and their application to deep disposal of radioactive waste: Part 2 - Recrystallization," *Lithos (2003) 67,* p. 119–133, 2003.

[172] P. Yang, Y. Wang, M. Rodriguez und P. Brady, „Rock-welding materials development for deep borehole nuclear waste disposal," *Materials Chemistry and Physics (2019) 221,* p. 178–187, 2019.

[173] R. Sun, „Selection and investigation of sites for the disposal of radioactive wastes in hydraulically induced subsurface fractures - Professional Paper No. 1215," 1982.

[174] International Atomic Energy Agency, „Disposal of radioactive grouts into hydraulically fractured shale," IAEA, Vienna, 1983.

[175] National Research Council, „An evaluation of the concept of storing radioactive wastes in bedrock below the Savannah River Plant Site," National Academy of Sciences, 1972.

[176] United Kingdom Nirex Limited, „Review of CoRWM Document No. 623, Direct injection No. 472939,“ 2005.

[177] E. Hardin, A. Clark, J. Su und F. Peretz, „Preclosure Risk Assessment for Deep Borehole Disposal - SANDIA Report SAND2019-1827,“ Sandia National Laboratory, 2019.

[178] Department of Energy, “Studying the Feasibility of Deep Boreholes,” Energy.gov, 19 12 2016. [Online]. Available: https://www.energy.gov/articles/studying-feasibility-deep-boreholes. [Accessed 28 08 2020].

[179] P. Voosen, „Protests spur rethink on deep borehole test for nuclear waste“.

[180] M. Rocher, „International panorama of research on alternatives to geological disposal of high-level waste and long-lived intermediate-level waste - IRSN Report 2019-00318,“ IRSN, 2019.

[181] Nuclear Waste Technical Review Board, „Technical Evaluation of the U.S. Department of Energy Deep Borehole Disposal Research and Development Program - A Report to the U.S. Congress and the Secretary of Energy,“ 2016.

[182] R. Emmermann und J. Lauterjung, „The German Continental Deep Drilling Program KTB: Overview and major results,“ *Journal of Geophysical Research (1997) 102,* p. 18179–18201, 1997.

[183] J. Cochran und E. Hardin, „Handling and Emplacement Options for Deep Borehole Disposal Conceptual Design - Revision 9,“ 2015.

[184] Sandia National Laboratories, „Deep Borehole Field Test Specifications - FCRD-UFD-2015-000132 Rev. 1,“ Sandia National Laboratories, 2015.

[185] F. Gibb, A. Beswick und K. Travis, „Deep borehole disposal of nuclear waste: Engineering challenges,“ in *Proc. Inst. Civ. Eng.-Energy 2014, 167, 47–66*, 2014.

[186] M. Driscoll, R. Lester, K. Jensen, B. Arnold, P. Swift und P. Brady, „Technology and Policy Aspects of Deep Borehole Nuclear Waste Disposal,“ *Nuclear Technology (2012) 180,* p. 111–121, 2012.

[187] N. Collier, N. Milestone und K. A. Travis, „Review of Potential Cementing Systems for Sealing and Support Matrices in Deep Borehole Disposal of Radioactive Waste,“ *Energies (2019) 12,* p. 2393, 2019.

[188] J. Payer, S. Finsterle, J. Apps und R. Muller, „Corrosion Performance of Engineered Barrier System in Deep Horizontal Drillholes,“ *Energies (2019) 12,* p. 1491.

[189] W. Minkley, „Integrität von Salzbarrieren und Endlagerung Wärme entwickelnder radioaktiver Abfälle in tiefen Bohrlöchern in Salzformationen,“ in *Proceedings of the Workshop “Final Disposal in Deep Boreholes Using Multiple Geological Barriers: Digging Deeper for safety"*, Berlin, 2015.

[190] S. Finsterle, R. Muller, R. Baltzer, J. Payer und J. Rector, „Thermal Evolution near Heat-Generating Nuclear Waste Canisters Disposed in Horizontal Drillholes,“ *Energies (2019) 12,* p. 596, 2019.

[191] P. Brady, C. Lopez und D. Sassani, „Granite Hydrolysis to Form Deep Brines,“ *Energies (2019) 12,* p. 2180, 2019.

[192] G. Freeze, E. Stein und P. Brady, „Post-Closure Performance Assessment for Deep Borehole Disposal of Cs/Sr Capsules,“ *Energies (2019) 12,* p. 1980, 2019.

[193] R. Muller, S. Finsterle, J. Grimsich, R. Baltzer, E. Muller, J. Rector, J. Payer und J. Apps, „Disposal of High-Level Nuclear Waste in Deep Horizontal Drillholes,“ *Energies (2019) 12,* p. 2052.

[194] D. Mallants, K. Travis, N. Chapman, P. Brady und H. Griffiths, „The State of the Science and Technology in Deep Borehole Disposal of Nuclear Waste,“ *Energies (2020) 13,* p. 833.

[195] Deep Isolation, “Deep Isolation,” [Online]. Available: https://www.deepisolation.com. [Accessed 2020].

[196] INTRI d.o.o., *Visualization of LILW disposal site at Vrbina, Krško,* INTRI d.o.o., 2018.

[197] L. Börgesson, R. Pusch, A. Fredriksson, H. Hökmark, O. Karnland und T. Sandén, „Final report of the Rock Sealing Project – sealing of zones disturbed by blasting and stress release. Stripa Project technical report 92-21,“ Swedish Nuclear Fuel and Waste Handling Company (SKB), Stockholm, 1992.

[198] R. Pusch, „Waste disposal in rock,“ *Developments in Geotechnical Engineering, Vol. 76,* pp. ISBN: 0-444-89449-7, 1994.

[199] L. Harvey, C. De Bock, M. Maitre, R. Strange, E. Leoni und C. Bucur, „Charing experience on waste management with/without WAC available, EURAD Milestone 144, EC Grant agreement no.: 847593,“ EURAD, 2021.

[200] L. Nachmilner, „Deliverable 2.4 - International approaches to establishing a waste acceptance system,“ PREDIS, 2021.

[201] V. Wasselin und Tokarevskyi, „Overview of existing work on categorization/classification of RWs in participating states. Final version as of 19.05.2021 of deliverable D9.4 of the HORIZON 2020 project EURAD. EC Grant agreement no: 847593.," EURAD, 2021.

[202] K. K. I. H.-K. G. Zakrzewska-Kołtuniewicz, „Possible Options for Management of Spent Nuclear Fuel From Future Power Reactors in Poland," in *Technical Meeting on Strategies and Opportunities for the Management of Spent Fuel from Power Reactors in the Longer Timeframe*, Bahadurgarh, India, 2019.