RESEARCH REPORT

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KYT SURFACE
Near Surface Repositories in Finland

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Summary

Near surface waste deposition is used for short-lived very low (VLLW), low (LLW) and intermediate (ILW) level nuclear waste. Common to all near surface repositories is that the waste is deposited close to ground surface, either in the sediment layer located above the bedrock or above the ground surface. The specific design of a near surface repository depends on the waste to be deposited and the expected lifetime of the repository. In the simplest scenario, waste (VLLW) may be buried in an underground rift with engineered structures used to limit waste interaction with surrounding groundwater, or in a landfill-type repository located above the groundwater table. ILW is generally deposited in a vault type repository with heavy concrete structures limiting the dispersion of radionuclides into the surrounding environment. Engineered barrier systems including waste form, waste packaging and barrier materials are typically used to limit the infiltration of precipitation into the repository and the release of radionuclides from the repository.

Near surface repositories for the deposition of nuclear waste is a new concept in Finland. Near surface repositories can accommodate short-lived VLLW, e.g., operational waste from nuclear power plants or decommissioning waste, near to the ground surface rather than deep underground in a cavern. Finnish geological and meteorological conditions are different from many of the other European locations where near surface repositories exist. Particular considerations for Finland include the relatively thin layer of sediment overburden above the bedrock, the generally close proximity of the groundwater table to the ground surface and the potential impacts of freeze-thaw cycles during winter months. In addition, the longer-tem effects of climate change need to be accounted for in the design of robust near-surface repositories suited to Finnish conditions. Potential near surface repository design options include landfill and vault style repositories.

The performance and evolution of a near surface repository will be assessed in a safety case for the repository.

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Preface

This report was compiled at VTT during 2019. The work performed was part of the KYT 2022 Programme (Finnish Research Programme on Nuclear Waste Management 2019–2022) with the aim to ensure that the authorities have sufficient high-level nuclear engineering expertise and other facilities at their disposal to effectively evaluate various nuclear waste management options. The emphasis of the KYT programme is on nationally important research topics; the particular focus of the KYT 2022 programme is to maintain and enhance national know-how and infrastructure in nuclear waste management and to promote collaboration between authorities, nuclear industry and scientists.

The work summarised in this working report was part of the KYT 2022 project "Near Surface Repositories in Finland" (KYT SURFACE) funded with a 70% share by TEM (Ministry of Economic Affairs and Employment in Finland) and with a 30% share by VTT. Because the concept of near surface repository use for nuclear waste deposition is new in Finland, the objective of the KYT SURFACE project was to explore near surface repository design based on Finnish regulations and international guidelines, site conditions in Finland, repository concepts, engineered barriers, monitoring, principles of the safety case and life-cycle of the repository. A secondary goal of the project was to ensure continuity of expertise within Finland by educating new Finnish experts on the topic. The project manager of this project was Paula Keto, VTT (design basis, site conditions, repository concepts and engineered barriers). Other VTT experts engaged closely in this project were Heidar Gharbieh (safety case, repository lifetime and engineered barriers) Leena Carpén (metallic barriers), Ville Rinta-Hiiro (geomaterials as barriers), Migurel Ferreira (concrete barriers), Mervi Somervuori (metallic barriers), Shila Jafari (metallic barriers), Arto Laikari (monitoring) and Minna Vikman (repository concepts). Erika Holt, Laura Wendling, Henrik Nordman and Kari Rasilainen took part as advisers to this work. The overall review of this report was performed by Laura Wendling, VTT. The legislation and description of the process for applying licence for operations in Finland have been reviewed by Arto Isolankila and Jarkko Kyllönen (STUK).

External advisers of the project included: Arto Isolankila, Jarkko Kyllönen, Ville Koskinen and Petri Jussila from STUK; Jere Tammela, Annukka Laitonen and Pasi Iivonen from TVO; and, Heikki Hinkkanen from Fennovoima. The authors of this report thank all advisors for valuable input on the content of the project. Valuable information was also contributed by Anders Wiebert and Pernilla Sopher from SSM, and Peter Flyhammar (SGI). Special thanks to Anita Bogh, David Rossiter, Sam Usher and Lynne Sutherland from AECOM and Camille Espivent ISRN for visiting Finland and sharing unique knowledge of the LILW waste management experience in UK and France. Extra special thanks to Dan Aronsson, Anna Larsson and Petter Larsson from Vattenfall for granting members of the SURFACE team the opportunity to exchange knowledge via a visit to the Ringhals site in December 2019.

As part of this work, the first KYT Seminar "Near Surface Repositories in Finland" was held on 26 September 2019. The SURFACE project team extends sincere appreciation to all presenters, as well as to the participants of the seminar. The presentations delivered during the seminar are publicly available from http://kyt2022.vtt.fi/kyt2022_seminar_sept_2019.htm.

Espoo, Finland, 3.2.2020

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Contents

1. Introduction

The fundamental safety objective of all nuclear facilities and activities throughout their lifetime or duration is to protect people and the environment from harmful effects of ionizing radiation (IAEA, 2006a).

Disposal is the endpoint of a nuclear material for which no further use is intended and that has subsequently been declared as waste. Disposal is defined as the emplacement of radioactive waste into a facility or location without the intention of later retrieval. In contrast, storage of radioactive waste is a temporary measure following which some future action is planned (IAEA, 2011a). In comparison to nuclear installations such as nuclear power plants, a disposal facility needs to provide safety for long periods of time and, therefore, emphasis lies on passive means of ensuring safety (IAEA, 2014). The main safety functions of a disposal facility are the containment of the waste and its isolation from the accessible environment to the extent demanded by the waste's hazard (IAEA, 2014).

Based on the legislative framework, national waste management policies set out possible disposal routes for different types of waste (IAEA, 2003). A generic framework for appropriate disposal options is defined by the IAEA radioactive waste classification system (IAEA, 2009a), which classifies radioactive waste based on its characteristics and radionuclide content, i.e., the specific activity and half-life, as the key determining features. Accordingly, high level waste (HLW) and long lived low and intermediate level waste (LILW-LL) require disposal at geological depths, i.e., several hundred meters below ground surface. LILW containing mainly short lived radionuclides (LILW-SL; radionuclide half-lives of less than approximately thirty years) represents only a small fraction of the total activity but more than 90% of the total volume of all radioactive waste worldwide. Thus, LILW-SL is considered an important waste category (IAEA, 2003). Relevant radionuclides in LILW-SL are ^{137}Cs and ^{90}Sr, which will decay to radiologically insignificant levels in about 300 years, corresponding to approximately 10 half-lives (IAEA, 2002). Very low level waste (VLLW) does not require a high level of containment and isolation and is suitable for the disposal in near surface landfill type repositories (IAEA, 2009a). IAEA's classification system serves only as a recommendation, however, whilst binding legislation needs to be established on a national level. In Finland, this is done through the "Radiation and Nuclear Safety Authority Regulation on the Safety of Disposal of Nuclear Waste" (STUK Y/4/2018).

In Finland, the majority of LILW consists of operational waste from nuclear power plants. In the future, radioactive waste will also be created during decommissioning of currently operating and newly-constructed nuclear power plants. LILW is also generated by hospitals, some industrial applications, and in mining and metallurgical activities where the quarried rock contains a relative abundance of radioactive elements (e.g., uranium and thorium).

In contrast to deep geological disposal facilities, the capacity of near surface disposal facilities (NSDF) to contain and isolate waste can be maintained over a substantially shorter timescale due to the location at or near the surface and the related degradation processes and events occurring in the biosphere (IAEA, 2014). In particular, the considerably higher possibility for inadvertent or intended human intrusion after an intuitional control of a maximum assumed period of few centuries after closure limits the capacity and the scope of NSDF (IAEA, 2014). Although institutional control after closure can be considered an additional safety measure, post-closure safety must not depend on such active means but rather relies on the passive engineered and natural features of the disposal system (IAEA, 2014). Therefore, NSDF are a suitable option only for LILW-SL, and the quantity of long lived radionuclides contained in the waste must be limited (IAEA, 2002; IAEA, 2014).

Options for NSDF can be distinguished with respect to their location relative to the ground surface (IAEA, 2002, 2003, 2014):

a. Above the original ground surface, e.g., mounds/hill type facility

b. Below the original ground surface at shallow depth, e.g., trenches, vaults, and pits

c. At intermediate depth (up to a few tens of meters underground), e.g., rock caverns, silos and tunnels. Borehole type repositories may all fall within this category

The choice of the optimal disposal option depends on various factors, such as the waste characteristics and the conditions at a specific site, and the variety of possible engineered structures is manifold (IAEA, 2014).

As of 1997, over one hundred LILW disposal facilities were, or had been, operating and more than 40 repositories were at different stages of development in member states of the International Atomic Energy Agency (IAEA) (Han et al., 1997). Finnish practitioners possess decades of operational experience in LILW disposal in NSDF at intermediate depth in rock caverns and silos. The final repository in Loviisa was designated for LILW arising from the operation and the decommissioning of the LO1&2 power plants, with a depth of 110 m. The Loviisa NSDF initiated operation in 1999 and has an operational licence valid until the end of 2055. In 1992, a repository in Olkiluoto (VLJ) at a depth of 60–100 m designated for LILW from the operation and decommissioning of OL1 and OL2 (and the future OL3) received a licence to operate with validity through 2051 (Posiva, 2012, 2016).

Some European countries, such as France and Spain, have disposed of VLLW and LILW in near surface repositories constructed in areas with varying geological conditions as an alternative to geological disposal. Advantages of near surface repositories for deposition of radioactive waste include relatively lower investment costs and increased ease of operation above ground. In order to reserve a greater proportion of underground repository space for LILW, there are currently plans in Finland to emplace VLLW within a near surface repository above or close to the original ground surface. The design principles of a near surface repository can be also applied to radioactive waste from other sources, e.g., from mining and processing of uranium- and thorium-containing ores.

1.1 Scope, structure and limitations of this document

The purpose of this report is to establish a basis for investigating the possibility of other types of NSDF in Finland for the disposal of VLLW, in particular but not limited to mounds/hill type facilities located above the original ground surface. Taking into account the national legislative framework and waste management strategy, the research focuses on advantages and limitations of the different facility types and possible design options in the light of waste generation and environmental conditions in Finland.

The different topics discussed in the report are:

- International guidelines and Finnish legislation regulating disposal of radioactive waste in NSDF (Section 2)
- The safety strategy inherent to NSDF (Section 3)
- The life-cycle of NSDF (Section 4)
- Radioactive waste suitable for NSDF (Section 5)
- Repository site conditions in Finland (Section 6)
- The different concepts and designs of NSDF (Section 7)
- The multi-barrier system comprising the engineered barrier system (EBS) and the geological barrier (Section 8). This chapter includes the related site conditions to be taken into account in Finland and processes relevant for the performance of the NSDF.
- Monitoring of a NSDF and different monitoring options (Section 9)

- The safety case of a NSDF and related safety assessments (Section 10)

Outside the scope of this work is to discuss underground repositories constructed in the bedrock. The focus of the work will be on repositories suitable for deposition of VLLW and to some extent also for deposition of LLW.

2. Safety requirements/legislation

2.1 International legislation and guidelines

In Finland, the regulations to be followed in nuclear waste management are based on Finnish legislation (see section 2.2). However, the Finnish legislation has been prepared and updated taking into account EU-level regulations given by EURATOM (European Atomic Energy Community), international treaties (for example Treaty of the Non-Proliferation of Nuclear Weapons) and guidelines and recommendations given especially by IAEA (International Atomic Energy Agency), ICRP (International Commission on Radiological Protection) and WENRA (Western European Nuclear Regulators Association).

The management of radioactive waste in the European Union is steered by the Radioactive Waste and Spent Fuel Management Directive (Council directive 2011/70/EURATOM of 19 July 2011). This directive requires that:

- The EU countries have a national policy and programmes for spent fuel and radioactive waste management.
- The EU countries have a robust framework and competent and independent regulatory body, as well as financing mechanisms to ensure that adequate funds are available.
- Public information on radioactive waste and spent fuel and opportunities for public participation are available.
- EU countries carry out self-assessments and invite international peer reviews of their national framework, competent authorities and/or national programme at least every ten years (by August 2023).
- The export of radioactive waste for disposal in countries outside the EU is allowed only under strict conditions.

Another important EU directive is the Council Directive 2013/59/Euratom (December 2013) laying down basic safety standards for protection against the dangers arising from exposure to ionising radiation. This directive is also known as the revised EU BSS "*European Basic Safety Standard*".

Considering management of other than nuclear waste in the European Union, the EU-directive "2008/98/EC Waste Framework Directive" shall be taken into account. This directive may be relevant for the waste that is no longer considered as radioactive waste. This directive defines:

- The basic concepts of waste management including waste, recycling, recovery and secondary raw material
- That the waste management should be done without harming humans and the environment, and
- The concept of waste hierarchy to minimise the amount of waste to begin with and to re-cycle or recover the material so that as little as possible of the waste is deposited in a landfill.

IAEA has a specific structure for nuclear safety requirements, including:

- Fundamental Safety Principles (IAEA, 2006a).
- General Safety Requirements.
- Specific Safety Requirements. Concerning management of radioactive waste, specific safety requirements are given in the IAEA safety standard SSR-5 "Disposal of Radioactive waste". (IAEA, 2011).
- General and Specific Safety Guides. For example:
  - IAEA Safety Guide SSG-23 "The Safety Case and Safety Assessment for the Disposal of Radioactive Waste" (IAEA, 2012).
  - IAEA Safety Guide SSG-29 "Near Surface Disposal Facilities for Radioactive Waste" (IAEA, 2014).

In addition, WENRA has a report series on Safety Reference Levels. Concerning disposal of radioactive waste the relevant document is: "WENRA Radioactive Waste Disposal Facilities Safety Reference Levels" (WENRA, 2014).

2.2 Finnish legislation and guidelines

2.2.1 Legislation linked to management of radioactive waste generated in nuclear power plants

Considering that the radioactive waste has been generated in nuclear power plants during production of nuclear energy, the final deposition of the waste is regulated under the Nuclear Energy Act (990/1987). The hierarchy of the Finnish regulations concerning nuclear safety is shown in Figure 2-1.

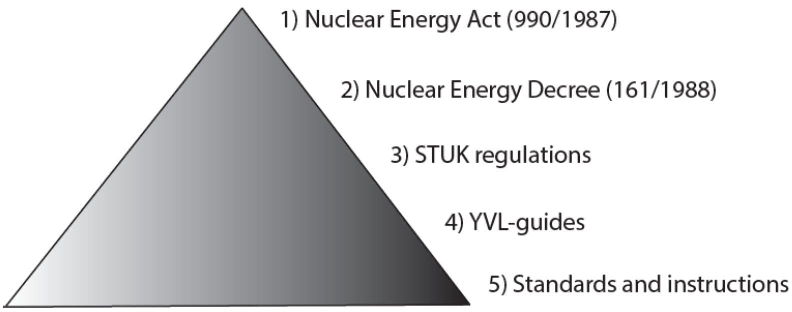

Figure 2-1. Hierarchy of Finnish regulations concerning nuclear safety (according to Routamo 2019).

Considering management and final disposal of very low (VLLW), low level (LLW) and intermediate level (ILW) nuclear waste, the regulations that should be taken into account include:

- Nuclear Energy Act (990/1987) (In Finnish: Ydinenergialaki)
- Nuclear Energy Decree (161/188) (In Finnish: Ydinenergia-asetus)
- STUK Y/4/2018 Radiation and Nuclear Safety Authority Regulation on the Safety of Disposal of Nuclear waste. (In Finnish: Säteilyturvakeskuksen määräys ydinjätteiden loppusijoituksen turvallisuudesta). In addition to this regulation there is STUK Explanatory memorandum 4/0007/2017, 4.12.2018 available (unofficial translation from Finnish).

- STUK YVL Guides, for example:
  - YVL D.4 Predisposal management of low and intermediate level nuclear waste and decommissioning of a nuclear facility. (In Finnish: Matala- ja keskiaktiivisen jätteen käsittely ja ydinlaitoksen käytöstäpoisto).
  - YVL D.5 Disposal of Nuclear Waste. (In Finnish: Ydinjätteiden loppusijoitus)

The content of these regulations are described briefly below based on legislation available in 2019. Some sections that may be relevant for near surface repositories are highlighted. It should be noted that when using this legislation one should always check the actual up-to date regulations from following sources:

- https://www.stuklex.fi/fi
- https://www.finlex.fi

Nuclear Energy Act (990/1987)

Nuclear energy act (990/1987) gives e.g. general principles for safe production of nuclear energy and implementation of nuclear waste management. In addition, it is applied also to mining of uranium and thorium. Following definitions are given in the Nuclear energy act (990/1987):

- *"Nuclear material* means special fissionable materials and source materials, such as uranium, thorium and plutonium, suited for obtaining nuclear energy;"
- *"Nuclear waste* means: a) radioactive waste in the form of spent nuclear fuel or in some other form, generated in connection with or as a result of the use of nuclear energy; and b) materials, objects and structures which, having become radioactive in connection with or as a result of the use of nuclear energy and having been removed from use, require special measures because of the danger arising from their radioactivity; (1420/1994)"
- *"Nuclear waste management* means: all measures necessary to recover, store and handle nuclear waste and permanently dispose of it (final disposal), including measures pertaining to the decommissioning of a nuclear facility"
- *"Nuclear facilities* shall refer to facilities necessary for obtaining nuclear energy, including research reactors, facilities performing extensive disposal of nuclear wastes, and facilities used for extensive fabrication, production, use, handling, storage of nuclear materials or nuclear wastes; nuclear facilities, however, shall not refer to: a) mines or enrichment plants intended for the fabrication of uranium or thorium, or the premises and places, including their precincts, where the nuclear wastes derived from such facilities are stored, or their repository; or to b) premises permanently shut down which contain nuclear wastes, enclosed there in a manner approved as permanent by the Finnish Centre of Radiation and Nuclear Safety;"

According to Nuclear energy act (990/1987), "*Nuclear waste generated in connection with or as a result of use of nuclear energy in Finland shall be handled, stored and permanently disposed of in Finland*".

According to Nuclear energy act (990/1987), "*Nuclear waste shall be managed so that after disposal of the waste no radiation exposure is caused, which would exceed the level considered acceptable at the time the final disposal is implemented. The disposal of nuclear waste in a manner intended as permanent shall be planned giving priority to safety and so that ensuring long-term safety does not require the surveillance of the final disposal site.*"

Nuclear Energy Decree (161/188)

The Nuclear Energy Decree (161/188) includes detailed regulations concerning laws presented in the Nuclear Energy Act (990/1987) on e.g. definitions of nuclear material and nuclear waste, import, export, transport and storage of nuclear materials/waste and licencing of nuclear facilities (construction & operation). It also defines in detail the licencing process for mining and enrichment of uranium and thorium (chapter 9).

According to section 5, nuclear waste as referred in paragraph 3 of section 3 (1) of Nuclear Energy Act shall not include:

- "radioactive materials that have spread into the environment that do not exceed the limits set for emissions;"
- "radioactive materials manufactured for commercial, industrial, agricultural, medical, scientific operations that are not part of nuclear waste management;"
- "radioactive waste generated when processing raw material containing uranium and thorium, if the annual production of uranium or thorium is < 10,000 kg,"
- "samples taken from a nuclear facility for research purposes."

According to section 6, extensive final disposal of nuclear waste (as referred to in paragraph 5, section 3 (1) of the Nuclear energy act), is meant by "final disposal of nuclear waste with total activity of radioactive materials is higher than 1 TBq or the alpha activity is higher than 10 GBq (excluding natural uranium, thorium or depleted uranium)." "The extensive manufacture, production, use, handling or storage of nuclear materials or nuclear waste means that the facility at given moment contain 1) more than 1 effective kilogram of nuclear materials or; 2) an amount of nuclear waste in which the total activity or alpha activity of radioactive materials exceeds the limits sets for extensive final disposal of nuclear waste."

Radiation and Nuclear Safety Authority Regulation on the Safety of Disposal of Nuclear waste STUK Y/4/2018

STUK Y/4/2018 is applied "to the disposal of spent nuclear fuel and other nuclear waste into nuclear facilities to be constructed in bedrock and facilities constructed into the ground." The regulation applies also "nuclear facilities intended for handling and storage of spent nuclear fuel and other waste that are not part of a nuclear power plant and in which the amount of spent nuclear fuel at any given time is not more than 100 tonnes of uranium." This means that these regulations apply for a near surface repository for nuclear waste from nuclear power plants, but also waste from other sources (e.g. radioactive medical waste). The safety of the production of uranium and thorium and disposal of processing waste and other nuclear waste generated during the production process is presented separately in STUK Y/5/2016.

STUK Y/4/2018 defines very low level waste (VLLW), low level waste (LLW), intermediate level waste (ILW) and high level waste (HLW) as follows:

- "*Very low-level waste* shall refer to nuclear waste whose average activity concentration of significant radionuclides does not exceed the value of 100 kBq per kilogram and the total activity does not exceed the values laid down in Section 6(1) of the Nuclear Energy Decree (161/1988) (total activity is higher than 1 TBq, or the alpha activity, excluding natural uranium, thorium and depleted uranium, is higher than 10 GBq)".
- "*Low level waste* shall refer to nuclear waste that, because of its low level of activity, can be processed without any special radiation protection arrangements. The activity concentration of such waste is usually not more than 1 MBq/kg."
- "*Intermediate level* waste shall refer to nuclear waste that, because of its high level of activity, requires effective radiation protection arrangements when processed. The activity concentration of such waste is usually 1 MBq/kg–10 GBq/kg".

- "*High level waste* shall refer to waste that, because of its high level of activity, requires highly effective radiation protection arrangements when processed and usually also cooling. The activity concentration of such waste is usually more than 10 GBq/kg."

STUK Y/4/2018 also defines the difference between short-lived and long-lived waste:

- "*Short-lived waste* shall refer to nuclear waste, the calculated activity concentration of which after 500 years is below the level of 100 MBq/kg in each disposed waste package, and below an average value of 10 MBq/kg in waste in one emplacement room".
- "*Long-lived waste* shall refer to nuclear waste the calculated activity concentration of which after 500 years is above 100 MBq/kg in a disposed waste package, or above an average value of 10 MBq/kg in waste placed in one emplacement room."

The STUK Y/4/2018 defines following general principles for the design of the disposal system:

- "Principle of *defence in depth* shall be taken into account in the design, construction and operation of a nuclear facility. (section 13)"
- "The design in accordance with the principle of structural defence-in-depth shall use *consecutive technical barriers* for limiting the dispersion of radioactive substances into the environment. (section 14)"
- "The long-term safety of disposal shall be based *on long-term safety function***s** achieved through *mutually complementary barriers* so that the degradation of one or more long-term safety function of foreseeable change in the bedrock or climate will not jeopardise the long-term safety." (section 30) According to the Explanatory memorandum 4/0007/2017, "Safety functions related to long-term safety refer to the functions brought about by the physical and chemical characteristics and processes of engineered and natural release barriers, which are intended for isolating the nuclear waste from bedrock or the living environment". In addition, the components shall be classified on the basis of their safety significance.
- "The radiation levels in a nuclear facility shall be monitored as well as discharge of radioactive substances and their concentrations into the environment." (Section 28)
- "The characteristics of the disposal site shall be favourable for isolation of radioactive substance from the environment." (Section 31)
- Considering engineered barriers for LILW: "The characteristics of engineered barriers shall be such that they *effectively prevent the release of radioactive substances into the bedrock* surrounding the underground emplacement rooms for a duration of time that is sufficient in relation to the half-life of the radioactive elements contained in the waste." (Section 32). According to explanatory memorandum 4/0007/2017, for *short-lived waste, this period shall be at least several hundreds of years*. In addition**,** the activity concentration after 500 years will be below the level of 100 MBq per kilogram in each disposed waste package, and below an average value of 10 MBq per kilogram of waste in one emplacement room. In practice, all low- and intermediate level and most of the decommissioning waste (except heavily activated metal waste, e.g. reactor pressure vessel) fall within this category.
- Considering VLLW: "In the case of very low-level waste in the ground, the entry of radioactive substances into the environment must be prevented effectively. For short-lived waste, this period shall be at least several hudreds of years, and for long-lived waste, at least several thousands of years." (Section 32)
- "*The engineered barriers shall slow down the passage of radionuclides*". (Section 32)
- "Engineered barriers shall not be constructed of materials or combination of materials that have a clearly unfavourable characteristics in terms of long-term safety or whose

operability may be reduced under the conditions in the emplacement rooms in a manner that jeopardises long-term safety of disposal." (section 32)

- Considering research and monitoring it is stated that: "In order to ensure the performance of barriers, a research and monitoring programme shall be established and implemented for the operating stage of the disposal facility." (section 33)
- "An *adequate protection zone* shall be reserved around the disposal facility as a provision for the prohibitions on measures referred to in Section 63 (1)(6) of the Nuclear Energy Act. According to the explanatory memorandum STUK 4/0007/2017 (4.12.2018, Radiation and Nuclear Safety Authority Regulation on the Safety of Disposal of Nuclear Waste, explanatory memorandum): A protection zone shall be reserved for the disposal site, for which prohibitions on measures pursuant to section 63 (1)(6) of Nuclear Energy Act may be placed in order to ensure the safety of the disposal. Pursuant to Section 85 of the Nuclear Energy decree, the *prohibitions on measures will be entered in the property register, land register or list of titles* in order to ensure that, if ownership of the area is transferred, the new owner will be made aware of land use limitations. The licensee should be in possession of a land area corresponding to the protection zone during the implementation of the disposal."
- Chapters and 2 and 8 describe principles for general safety and 8 long-term safety:
  - Section 3 describes how the safety of the operations shall be demonstrated to be according to safety requirements.
  - According to section 4, "the long-term safety of disposal of nuclear waste shall be assessed based on the applicable principles when selecting the disposal site and applying for a decision in principle." "*The long-term safety shall also be assessed when applying for a licence of operations for a very low level waste (VLLW) disposal facility*, a construction licence and operating licence for a disposal facility and a decommissioning licence for a nuclear waste facility as during periodic safety assessments." According to section 7e (2) of the Nuclear Energy Act, "the overall safety of a nuclear facility shall be assessed every 10 years and the overall safety of a facility for the large-scale disposal of nuclear waste at least once every 15 years." Safety assessment shall be also updated before closure. The time-period discussed in the safety assessment shall extend as far into the future as the disposed waste can be seen to constitute a risk to the safety of people and the living environment.
  - *"Ageing management* related to safety shall be controlled throughout the life cycle of the nuclear facility." (Section 6) According to Explanatory memorandum (4/0007/2017), "the systems, structures and components are subjected to stresses and environmental effects that may reduce their operability." "For this reason *monitoring, periodic inspections, tests and service* (replacement if needed) of the engineered barriers is required." This applies to the operational period.
  - *"Management of human factors* shall be taken into account in the design and operation of a nuclear facility." (Section 7)
  - "The long-term safety shall be assessed on the basis of safety research results and it shall be presented in the *safety case*." (Section 4)
  - "The safety functions for the operation of the nuclear facility and the long-term safety functions shall be defined, and the systems, structures and components performing them and related to them shall be classified." (Section 5)
  - "Compliance of the engineered barriers and the disposal site with the requirements shall be demonstrated by means of a safety case, including possible evolutions of the disposal system and rare events impairing long-term safety." (Section 35)
  - "Compliance with the dose constraints for the most exposed people, as referred in the Nuclear Energy Decree, shall be demonstrated by considering a community

that derives nourishments from the immediate surroundings of the disposal site and is most exposed to radiation. Impacts on flora and fauna shall be also analysed." (Section 35)

- The reliability of the safety case is defined in section 36. "The safety case and the methods, data and models used in it shall be based on high-quality research data and expert judgement, and they shall be documented in a traceable manner. The basis for the calculational analyses shall be that the actual amounts of radioactive substances released and the actual radiation exposure shall be, with high degree of certainty, lower than the results received from the safety analyses."
- Chapter 9 describes requirements for the organisation and personnel of a nuclear facility.

YVL D.4 Predisposal management of low and intermediate level nuclear waste and decommissioning of a nuclear facility (15.12.2019)

YVL D.4 gives guidelines for management of low- and intermediate level waste and decommissioning of a nuclear facility (nuclear power plants and research reactors). The guide gives requirement for planning and implementing the sorting, processing, storage, activity determination and record keeping of the (LILW) operation waste arising from a) the operations of nuclear facilities and b) decommissioning. The guide also addresses the clearance of nuclear waste, including recyclable materials arising from operations or decommissioning.

YVL D.4 (15.12.2019) defines following design basis for storage and processing of operational waste and decommissioning:

- "Under Section 22 b of the Nuclear Energy Decree, the annual dose constraint for the member of the public arising from the normal operation of a nuclear power plant and other nuclear facility equipped with a nuclear reactor shall be 0.1 mSv. The annual dose constraint for the member of the public arising from the planned decommissioning of a nuclear power plant and other nuclear facility equipped with a nuclear reactor shall be 0.01 mSv. (YVL D.4/301)."
- "Pursuant to Section 22 d of the Nuclear Energy Decree, the processing and storage of operational waste and the decommissioning of the nuclear facility shall be so designed that the annual dose constraint for the member of the public arising from planned processing and storage is 0.01 mSv. (YVL D.4/302)."
- "The annual dose constraints for the most exposed member of the public arising as a result of an operational occurrence or accident where nuclear waste is processed or stored shall be, under Section 22 d(2) and Section 22 b(2–6) of the Nuclear Energy Decree: a. 0.1 mSv as a result of an anticipated operational occurrence b. 1 mSv for a Class 1 postulated accident c. 5 mSv for a Class 2 postulated accident. (YVL D.4/303)."

YVL D.4 also defines the regulations concerning *clearance of nuclear waste*:

- "According to Section 27 d of the Nuclear Energy Act clearance levels shall be set in such a way that the exposure caused to members of the public is of minor significance. The basic radiation protection requirement for the clearance of nuclear waste is that the annual dose constrain to any member of the public or worker handling the waste is 0.01 mSv, and that the radiation exposure arising from the cleared waste is otherwise kept as low as reasonably achievable. This dose constraint applies to the clearance of materials arising from the operation or dismantling of a single nuclear power plant or other nuclear facility (Section 3.2/308)."
- As a general guideline nuclear waste may be cleared from regulatory control following a general or case specific procedure:

  - "In a general clearance procedure, the destination of the material released from the facility need not to be designated, or is only designated as its outline, and the activity levels to be applied are fixed" (4.1/409a). What this requires in practice, that the *nuclide specific activity levels* presented in APPENDIX 1 are complied with (section 411). "Alternatively, if the annual amount of the waste to be cleared *does not exceed 100 tonnes for one nuclear power plant of nuclear facility, the activity levels in APPENDIX 2 may be applied* for waste that is disposed of in a public landfill or dispatched to be melted as metal (section 411)". However, it should be noted that when the levels specified in the appendices are applied to several nuclides, "*the sum of the ratios between nuclide specific activities and the respective activity levels shall be less than one*" (4.1/411). In addition, when necessary, an assessment of the nuclide composition and activities of the waste may be used (section 411).
  - "In a *case specific clearance procedure*, the recipient of the material and the maintenance process must be defined; the activity levels will be imposed based on case-to-case consideration (409b)". In this case, the activity levels approved by STUK shall be compiled with, taking into account the provisions contained in Section 10(1) of Nuclear Energy Decree:
    - "The total activity of nuclear waste in the possession of the transferee shall be lower than 1 GBq and the alpha activity lower than 10 MBq (section 412a)."
    - "The Annual effective dose caused by the transferred nuclear waste to any individual shall not exceed 0.01 mSv (section 412b)."
    - "The Radiation exposure caused by the transferred nuclear waste shall also otherwise be as low as reasonably achievable (section 412c)."
  - The general clearance procedure is not applied for volatile or flammable waste or waste that is otherwise prone to cause radiation exposure (section 410).
- YVL D.4 also regulates the storage and handling of the nuclear waste prior to disposal. For example in the storage of nuclear waste and waste packages, the waste type, origin and amount shall be specified along with the activity inventory data, special characteristics and location (section 419). The activity of dominant nuclides in packed waste shall be determined before disposal (section 420).

YVL D.5 Disposal of nuclear waste

YVL D.5 defines (D.5/101) that low and intermediate level waste arising from the operation of nuclear power plants and other nuclear facilities are to be processed and disposed of in bedrock repositories constructed at an intermediate depth. This applies also for low- and intermediate level waste arising from decommissioning. However, "*very low level waste (VLLW) can be disposed of in repositories constructed near surface* (2018/2/13)". In annex "Nuclear waste classification" of the explanatory memorandum for D.5 it is further defined that "part of the low level waste may be classified as very low level waste (VLLW) that can be disposed of in the ground". In this case, *the average activity of the concentration for the waste is not more than 100 kBq/kg. In addition, it is defined in the same appendix that all VLLW falls within the category of short-lived waste where the majority of the nuclides have half-lives of not more than 30 years or so.* Short-lived waste refer to nuclear waste with the calculated activity concentration of which after 500 years is below the level of 100 MBq/kg in each disposed waste package, and below an average value of 10 MBq in one emplacement room.

Majority of the regulations YVL guide D.5 are addressed on extensive disposal of nuclear waste constructed inside the bedrock (YVL D.5/201), and are therefore not applied on near surface repositories for VLLW.

2.2.2 Legislation linked to NORM

According to IAEA (2017b) "*Waste that contains naturally occurring radioactive materials is known as NORM waste. It occurs as a by-product, residue or waste from activities such as mining and processing of uranium, extraction of rare earth elements, production and use of thorium and its compounds, mining of ores other than uranium ore, production of oil and gas, and from the phosphate industry*". In mining it is typical that large quantities of NORM is generated and the NORM also contains toxic substances such as heavy metals (IAEA 2017b).

From Finnish legislation point of view, the NORM produced in mining of uranium and thorium is regulated by the Nuclear Energy Act (990/1987) and the other NORM (or radioactive materials) by Radiation Act (859/2018). These cases are described briefly below.

Legislation linked to production of uranium and thorium

The production of uranium thorium and deposition of the processing waste and waste rock is regulated by:

- Nuclear Energy Act (990/1987) In Finnish: Ydinenergialaki
- Nuclear Energy Decree (161/188) (In Finnish: Ydinenergia-asetus)
- STUK Y/5/2016 Radiation and Nuclear Safety Authority regulation on the safety of Mining and Milling Operation Aimed at Producing Uranium or Thorium.
- Mining act (621/2011).

Regulation STUK Y/5/2016 is applied to safety of mining and milling activities aimed at producing uranium and thorium insofar as the activities fall within the scope of application of the Nuclear Energy Act (990/1987). The Nuclear Energy Act is applied to mining and enrichment operations of uranium and thorium (2§). However, according to Nuclear Energy Degree (161/188, 9b §), the Nuclear Energy Act is not applied when:

- "The annual production capacity of thorium and uranium is less than 10 t (Nuclear Energy Decree 161/1988, 9b)," or
- "If the average activity concentration is less than the activity concentration presented in 2§ (of the Nuclear Energy Act)", and
- "When the enriched product has total activity concentration for uranium and thorium less than 0.5 kg/t."

The regulation STUK Y/5/2016 is also applied to handling and disposal of the radioactive waste generated by these activities.

Following definitions linked to deposition are presented in the STUK Y/5/2015:

- "Mineral processing waste refers to tailings and other waste generated during the process of separating uranium and thorium from ore."
- "Waste rock shall refer to rock that is excavated from a mine but not forwarded to a production plant."
- "Production waste shall refer to nuclear waste generated during the production of uranium and thorium where the average activity concentration of the isotope uranium-238, radium-226, lead-210, thorium-232 or radium-228 exceeds the value of one Becquerel per gram (Bg/g)."
- "Nuclear waste shall refer to a) radioactive waste in the form of spent nuclear fuel or in some other form, generated in connection with or as a result of the use of nuclear energy; or b) materials, objects and structures which, having become radioactive in

connection with or as a result of the use of nuclear energy and having been removed from use, require special measures because the danger arising from their radioactivity."

STUK Y/5/2016 gives regulations for limiting radiation doses, planning of mining and processing operations, commissioning and operations and management, organisation and personnel. Some of the regulations to be taken into account in the deposition are presented below:

- "The licensee shall use computational analyses to estimate the radiation doses caused to the population in the vicinity of the production unit by the releases of radioactive substances as a result of normal operation, anticipated operational occurrences or accident situations (4§). The limits of releases of radioactive substances are presented in the Nuclear Energy Decree (161/1988) (3§)".
- "The possible releases of radioactive substances from the production unit shall be monitored and their concentrations in the environment shall be observed (10§)".
- "Nuclear waste generated from mining and milling activities shall be processed and disposed of in a manner that can be considered safe in terms of long-term isolation, while taking into account the amount of the waste, its activity concentration, the other factors affecting radiation exposure and the local conditions (14§)."
- "Adequate protection zone shall be reserved around the disposal area (15§)."
- "Waste rock shall be covered in a manner where the strength of the external radiation and radon concentration in air will not exceed the natural levels prevalent in the area. The protective layers shall withstand degradation caused by natural phenomena and effectively limit the release and dilution of radioactive substances (15§)."
- "Mineral processing waste classified as production waste shall be processed in a manner that will make the long-lived radioactive substances within the waste chemically stable within their disposal environment. A mineral processing waste area located near the ground shall be covered in a manner where the strength of external radiation and radon concentration in air will not exceed the natural levels prevalent in the area (15§)."
- "If necessary, the mineral processing waste shall be isolated by means of protective layers that act as release barriers. They prevent the filtration of rainwater and the flow of surface water and groundwater through the waste area, the resulting passage of radioactive substances from the waste are to the environment and, on the other hand, limit the penetration of plant roots into the mineral waste. The protective layers shall withstand degradation due to natural phenomena (15§)."
- "Structures, items, components and materials contaminated by radioactive substances that cannot be decontaminated shall be disassembled and disposed of in a manner approved by the STUK (16§)."
- "The licensee shall arrange the records concerning disposed production waste and other nuclear waste into a file that includes information concerning the location of the waste area, the characteristics of the waste and the amounts of radioactive substances within the waste. The information shall be continuously kept up to date for as long as the mining or milling activities continue. The information shall be regularly submitted to STUK."

Other NORM/radioactive materials produced e.g. in mining and in industry

The radioactive materials generated for example in industry or in mining of other minerals than uranium and thorium are regulated based on the Radiation Act (859/2018). According 145§ of the Radiation Act (859/2018), notification to STUK is needed prior to operations including "a) mining activities as specified in the Finnish Mining Act, b) working underground more than 100 h/years, c) any handling, used, storage or production of materials or waste containing U-238, Th-232 or their decay products in excess of 1 Bq/g or d) aviation". In addition, all activities requiring notification are obliged to make dose assessment (859/2018, 146§). Dose assessment is also required from industries listed in Council Directive 2013/59 EURATOM Annex VI, unless the regulator is provided with appropriate evidence proving that the all relevant materials in all relevant stages of handling have activity concentrations < 1 Bq/g (uranium and thorium series) (Kallio, 2019). This is true also for other activities where it is possible to exceed reference levels set for the workers or the public (Kallio, 2019). According to (Kallio, 2019) reference levels are for effective dose in addition to natural background radiation (excluding radon and cosmic radiation) and are 0.1 mS/a for the public and 1 mSv/a for the worker. Considering building materials the limits for the public is 1 mSv/a and 0.1 mSv/a for Cs-137 (Kallio, 2019). If these reference levels are exceeded (even after dose limiting measures) the licencing by STUK is required (Kallio, 2019).

According to Kallio (2019), the waste generated in these activities is not classified as radioactive waste, but waste causing exposure to natural radiation. Typical for this type of waste is high volumes and presence of other impurities. Majority of this type of waste can be deposited in industrial landfills after case by case consideration (Kallio, 2019). A licence is needed for the disposal. The discharges from the site shall be monitored and if the discharges exceed STUK clearance levels, authorisation by STUK is required (Kallio, 2019). This is also needed when re-using or recycling the material (Kallio, 2019).

For the basis of these regulations see also Government Degree on Ionizing Radiation (2018/1034) and "*Säteilyturvakeskuksen määräys luonnosäteilylle altistavasta toiminnasta,* STUK S/3/2019".

2.2.3 Cleared waste / normal waste

Cleared nuclear waste or other waste that is not regulated either by Nuclear Energy Act (990/1987) or Radiation Act (592/1991) will be regulated by the Waste Act (1072/1993) and Waste Decree (1390/1993). Also to be considered is the Environmental Protection Act (527/2014) and its prohibition against soil contamination (525/2014, 16§) and groundwater pollution (525/2014, 17§). Environmental permit is required in activities referred to in Annex 1 of the Environmental protection act, including treatment of waste on a professional basis including treatment of hazardous waste, and disposal of non-hazardous waste with capacity exceeding 50 tonnes per day. In addition, the permit is required for landfills receiving more than 10 tonnes of waste per day in an overall capacity exceeding 25 000 tones (excluding landfills for inert waste). Enviromental permit is issued by ELY. The environmental impact assessment linked to the environmental permit is regulated by Enviromental Impact Assessment Act (252/2017).

2.2.4 Licencing and applying the existing legislation for a near surface repository

The waste deposited in a near surface repository will be in practice VLLW operating or decommissioning waste from a nuclear power plant. Since this waste is generated in production of nuclear energy, it will be regulated under the Nuclear Energy Act.

According to STUK (STUK 2019) licencing for a VLLW repository will be granted by STUK, provided that the disposal is not large scale disposal. The disposal is considered large scale when the disposal facility contains nuclear waste in which the total activity is higher than 1 TBq, or the alpha activity, excluding natural uranium, thorium and depleted uranium, is higher than 10 GBq (Nuclear Energy Decree, 6§). The average activity concentration of significant radionuclides should not exceed 100 kBq/kg and the total activity of significant radionuclides does not exceed the values mentioned in Nuclear Energy Decree (6§).

The licencing documents shall include the basic information specified in sections 42 and 43 of the Nuclear Energy Decree:

- "The application for a licence for operations shall contain at least the following information: (1) the applicant's name or the firm name used in business operations, and domicile; (2) the operations for which the licence is applied for; (3) the site where the operations are intended to be carried out; (4) the extent of the intended operations; (5) the timetable of operations, especially the starting time planned for the operations and their duration; (6) the quantity and quality of any other nuclear materials possibly possessed by the applicant; and (7) the licences granted to the applicant in accordance with the Nuclear Energy Act, and the supervisors in charge approved for the various operations (Nuclear Energy Decree, 42§)."
- "The application for a licence for operations shall be supplemented with an extract from the population register or an extract from the trade register or some other equivalent document about the applicant or about his nationality and also, unless when obviously unnecessary because of the nature of the planned operations, with: (1) documents equivalent to the documents referred to in Nuclear Energy Decree section 36; (2) a description of the organisations planning and implementing the operations; (3) proof of the applicant's right to use the site or areas required by the operations; (4) a description of the settlement and other activities and town planning arrangements at the site of the operations and in its immediate vicinity; (5) a description of the environmental impact of the operations and a description of the design criteria that will be observed by the applicant to avoid environmental damage and to restrict the burden on the environment; (6) a description of the applicant's plans and available methods for arranging the management of nuclear waste resulting from the operations, including the disposal of nuclear waste, and a description of the timetable of nuclear waste management and its estimated costs; (7) a description of the types and amounts of nuclear materials and nuclear waste that the operations have been planned to involve; and (8) any other information considered necessary by the authorities (Nuclear Energy Decree, 43§)."

Considering the type of the waste, the regulation given in the STUK Y/4/2018 on Safety of Disposal of Nuclear Waste shall be applied graded (STUK, 2019). This means for example that the multi-barrier system designed taking into account the containment period. According to STUK (2019), guidance for VLLW disposal are currently being prepared by STUK and should be taken into account in future. This document will include requirements for the site, natural and engineered barriers, monitoring and the safety case.

3. Safety strategy

Safety should be the primary consideration of a waste disposal facility, and the safety strategy sets out how safety is achieved during throughout the facility's lifetime. The safety strategy can be considered a high level approach integrating all aspects relevant for safe disposal of radioactive waste, and therefore must thoroughly describe how all requirements related to long-term (Section 3.1) and operational (Section 3.2) safety are fulfilled. The safety strategy should also explain the applied management system (Section 3.3) ensuring the requisite quality of all works carried out and decisions made. The general safety philosophy provides for an iterative and step-wise approach, thus allowing flexibility, learning and optimisation during the process of implementing disposal facilities for radioactive waste. Accordingly, the safety strategy should not only be produced at the very early stages of a disposal program but should also be periodically revised (IAEA, 2014). The safety strategy constitutes a fundamental part of the safety case concept, as described in more detail in Section 10.

3.1 Long-term safety

In accordance with the fundamental safety principles presented in Section 2, radioactive waste must be managed in such a way as to avoid imposing an undue burden on future generations

(IAEA, 2006a). Therefore, the long-term safety of radioactive waste disposal should rely on passive engineering and natural features of the disposal system to be developed in a step-wise and iterative manner. Additional active measures to support passive safety should be minimised with reliance on these measures limited by the confidence in institutional and financial stability (up to a few centuries) (IAEA, 2014). This is consistent with the safety requirements presented in SSR-5 (IAEA, 2011a) stating that "[t]he long term safety of a disposal facility for radioactive waste has not to be dependent on active institutional control."

Passive post-closure safety is ensured through the disposal system's main safety functions of containment and isolation. These main safety functions are provided by a series of engineered and natural barriers with multiple complementary and compatible lower-level safety functions preventing the release of harmful substances or retarding their migration into the biosphere to an acceptably low level set by applicable regulatory limits. In this type of multibarrier system, a number of safety functions may be assigned to a single barrier, whilst a particular safety function may be performed by a number of barriers (IAEA, 2014). The various safety functions may be effective over different timescales in the post-closure period and thus, the relative importance of the barriers and assigned safety functions will vary with time (IAEA, 2002; IAEA, 2014). In the context of a multibarrier system, compatibility means that the engineered and natural components of the disposal system mutually favour, or at least do not unduly hamper, the fulfilment of their assigned safety functions.

Containment of the radioactive waste can be achieved by a combination of physical and chemical means. Physical barriers like metal containers, concrete overpacks or geotechnical barriers with low permeability can prevent the release of radionuclides in a gaseous state (e.g., ^{3}H, ^{14}C and ^{129}I) or dissolved in the liquid phase (i.e., water as primary transport medium) and, thus, mitigate radionuclide egress. Chemical containment can be provided by retarding the radionuclide migration by controlling solubility limit (e.g., through the use of cementitious materials) and through different sorption mechanism onto immovable substance, such as clay mineral surfaces (IAEA, 2014). In the context of sorption phenomena, the importance of radionuclide transport facilitated by colloids needs to be assessed.

Isolation refers to the separation of the hazardous waste from the accessible biosphere by an appropriate repository design and the suitability of the chosen site, also taking into account external events potentially detrimental for the safety performance of the disposal system. Regarding external events, floods, freeze-thaw cycles and human intrusion might be of particular relevance for NSDF due to their location at or near ground surface. Human intrusion can be impeded by placing the NSDF at greater depths up to a few tens of meters, as it is the case for the currently existing Finnish repositories for LILW (IAEA, 2014, Posiva, 2016). Institutional control measures in the post-closure phase can be supportive in this regard by limiting human activities at the disposal site for a specified period of time.

The described redundancy and diversity of assigning multiple compatible safety functions to a series of barriers form the basis for the defence in depth concept. Defence in depth implies that safety does not unduly rely on a single feature of the disposal system and that a degradation or a loss of a safety function provided by one or multiple barriers is compensated by other safety functions for the relevant timescale (IAEA, 2014). In addition to the interplay of favourable engineered and natural features of the disposal system and the set of multiple complementary and compatible safety functions, defence in depth is provided by controlling the inventory and the waste form (e.g., the concentration of long-lived radionuclides), operating procedures and institutional and administrative controls (IAEA, 2011a, 2014). Again, reliance should not be placed excessively on active means and can only support the system of passive measures for ensuring long-term safety.

Active safety measures in the post-closure phase can be taken within the framework of institutional control, which is presumably limited to few hundred years. However, the long-term safety must not be dependent on active institutional control. Institutional control comprises monitoring and surveillance activities to verify the continuous fulfilment of safety functions in order to detect a possible deterioration of the barriers and to ensure the functionality of the post-closure monitoring system. Maintenance and repair of engineered structures should be kept to a minimum and an appropriate facility design should ensure the required longevity of the barriers and other components (IAEA, 2002, 2014). Other active measures include land use restrictions and access controls to the site in order to prevent human intrusion. After an active institutional control period, human intrusion cannot be excluded and therefore must be addressed in the long-term safety assessment of a near surface disposal facility (IAEA, 2011a, 2014). The use of durable warning markers at the site and the preservation of records in archives can reduce the risk of human intrusion beyond the period of active institutional control (IAEA, 2014).

With this in mind, it can be summarized that the long-term safety of a NSDF is to be achieved by (IAEA, 2011a, 2014):

- The containment and isolation capacity provided passively through the disposal system consisting of the disposal facility with its engineered components and the site's natural features contributing to the containment and isolation.
- Controls on the radiological inventory and the waste form. This applies in particular to the concentrations of long-lived radionuclides in the waste, since the capacity and scope of NSDF is limited due to their location at or near the surface and the related degradation processes and events occurring in the biosphere. Particular attention needs to be given to the increased likelihood of human intrusion after loss of institutional control.
- Supporting the passive safety approach by active safety measures like institutional control without placing unduly reliance on them. Institutional control is further discussed in Section 4.3.

3.2 Operational Safety

During the operational phase, the safety of workers, the public and the environment should be ensured by active and passive means (IAEA, 2014). With respect to radiological safety the ALARA principle applies and accordingly, radiological consequences arising from the operation of disposal facility must be **a**s **l**ow **a**s **r**easonably **a**chievable taking into account relevant economic and social factors (IAEA, 2001). Active measures contributing to operational safety include (IAEA, 2014):

- control of surface dose rate and contamination of waste packages;
- implementation of operational activities such as waste emplacement and the installation of engineered barriers by defined quality controlled procedures (see also Section 3.3. Management systems);
- monitoring of radioactive releases from the disposal facility in liquid form and airborne in the form of gases or dust.

Wherever possible, passive safety systems, e.g., shielding during waste handling, should be employed (IAEA, 2014).

Operational safety provisions need to take into account normal operation mode and anticipated operational occurrences as well as incident and accident conditions, which need to be assessed in the context of the specific disposal system. In this connection, internal events (e.g., drop of waste package during handling operations) and external events (e.g., floods, earthquakes or extreme weather conditions such as strong winds) need to be considered (IAEA, 2014).

Unauthorized access to the site and removal of radioactive material need to be prevented by appropriate security systems, which should be developed together with the nuclear safety measures in an integrated manner (IAEA, 2011b). Nuclear safeguards are most likely not relevant for the NSDF considered for development in the Finnish waste management program and as such are not further discussed herein.

3.3 Management systems

Essential to both operational and long-term safety is the use of appropriate management systems to ensure the required quality of all activities carried out and all decisions taken throughout the entire lifecycle of the disposal facility including the pre-operational, operational and post-closure periods (see Section 4). At all times, the management systems are part of the iterative development of the safety case, the central tool to demonstrate and provide confidence in the safety of a disposal facility, and are subject to periodic safety assessments which constitute the core of the safety case (see Section 10 for details) (IAEA, 2012). Whenever possible, management systems should be in accordance with national or international recognised standards, codes and regulations and should be evaluated regularly by independent bodies. (IAEA, 2014).

With respect to the implementer, the management system needs to ensure the effective coordination of the different organisational units and the various interrelated scientific and technical disciplines involved in the implementation of a disposal facility. This is particularly critical with respect to the communication between researchers (i.e., experts) and safety assessors (i.e., generalists) (IAEA, 2003; NEA, 2013). Furthermore, the management system should indicate the way of interacting with external organisations, such as the regulator, technical service providers, consultancies and inspection companies, the scientific community, NGOs, and the affected and general public (IAEA, 2001, IAEA, 2014). Establishing the organisational framework is a fundamental basis for the development of a high-level safety culture with safety as the core consideration. The organisational structure requires a clear definition of roles, responsibilities and authorities of the various individuals and organisational units involved in all safety relevant activities (IAEA, 2014). For establishing a good safety culture and for ensuring that the management systems are appropriately applied everywhere in the organisation, the roles of the management and leadership, and in particular at the senior level, are crucial. Managers and leaders act as role models and provide guidance to ensure that their commitment to safety is shared with all employees (IAEA, 2016; NEA, 2016).

Quality control (QC) and quality assurance (QA) programmes constitute important elements of the management system. These include staffing requirements, training, education and certification of workers, all of which contribute to a healthy safety culture and ensure the relevant competences to perform all activities with the necessary quality in a safe working environment. Concomitant with a questioning and learning attitude to be developed as part of a healthy safety culture, the effectiveness of the training and certification programmes are regularly assessed and updated to take into account new insights and gained experiences, e.g., from the operation of the disposal facility (IAEA, 2016; NEA, 2016). Regular reviews are also necessary to ensure that relevant information is updated (e.g., ongoing research on component behaviour and the suitability of chosen materials) and made available to the affected persons and groups forming the basis for a well-founded iterative decision-making process (IAEA, 2011, 2014). Adequate QC and QA programmes are used to ensure that the design, manufacturing/production and construction/installation of all systems, structures and components relevant for the safety of the disposal facility comply with the requirements and specifications set in the safety case, so that ultimately the assigned safety functions can be fulfilled (IAEA, 2014). This is naturally interlinked with the site characterisation and selection process, in which QC and QA programmes should be applied to ensure the quality, usability and availability of data (IAEA, 2014). In the context of the established organisational framework, an effective QC and QA programme indicates the type and means of internal and external reviews, checks and verifications of all activities during the entire repository lifecycle to be undertaken by independent and skilled individuals or organisations (IAEA, 2001).

As mentioned in the previous chapter, operational activities, e.g., the receipt, transport/handling and possible storage of waste packages, waste emplacement, installation of barriers and closure of the disposal facility (see Section 4.2), should be carried out in accordance with procedures defined in the QC and QA framework, which implies that those procedures are well documented and accessible (IAEA, 2014). The operational period may last several decades; thus, adequate ageing and maintenance management should be employed for both the active and passive systems relevant for the safety of the facility. In this context, suitable aging management aims to minimise the need for maintenance of relevant safety (e.g., waste handling equipment, shielding) and support systems (e.g., civil and electrical systems) by means of regular inspections, testing and, when needed, restoration of functionality in accordance with the procedures defined in the QC and QA programme (IAEA, 2014). In addition to operational safety considerations, the aging management programme aims to detect occurrences during the operational period with negative implications for long-term safety, which would remain undiscovered until after closure (IAEA, 2014). In the post-closure phase (see Section 4.3), the need for maintenance of, for example, components of the post-closure monitoring system or engineered structures like the cap in case of a mould/hill type NSDF, might occur. However, as emphasized previously, the safety at all stages of the development of the disposal facility should rely on passive rather than active measures (IAEA, 2001).

Information used in safety assessments and in supporting the decision-making should be obtained from a range of various sources including field, laboratory and theoretical studies (NEA, 2013). The long-lasting process of the repository development is accompanied by an accumulation of a vast quantity of different types of data and other information. Therefore, adequate information management and documentation should be part of the overall management system and be implemented early in the site development process to enable traceable and transparent decision-making and future reassessments, if desired (IAEA, 2014). The record keeping includes inventory data and waste package information, site investigation and monitoring data, design documents, information on construction work and on operational and closure activities. The applied QC and QA systems themselves are part of the record preservation indicating how the required quality has been achieved and all requirements and specifications set in the safety case have been met (IAEA, 2003, 2014a). Additionally, the applicable laws and regulations, licence documents, the safety case and safety assessment documentation and the provisions and measures during the institutional control period are to be thoroughly documented (IAEA, 2003). All changes or modifications to component or equipment specifications, procedures and conditions established in the safety case and license documents must be clearly documented, justified and assessed in the light of operational and long-term safety. Remaining uncertainties and their treatment are of central concern in the safety case (see Section 10) and the management system must ensure the documentation (IAEA, 2014). Similarly, an assumption management system should be instituted and its documentation taken into account by the record keeping provisions. This is of particular importance experts carrying out simulation modelling and safety assessments (Section 10.1), as the underlying assumptions determine the choice of input data and profoundly impact the assessment outcomes. In conformity with the iterative approach, the safety case and affected licencing documents must be updated accordingly. Records should be maintained in both physical and electronic forms in appropriate archives (IAEA, 2014). International co-operations may ease the technical and administrative challenges related to long-term record keeping (IAEA, 2003).

Finally, the QC and QA programme provides assurance for sufficient and secure financial resources throughout the decade-spanning disposal programme (IAEA, 2014).

4. Development (life-cycle) of a near surface repository

The implementation of a final disposal facility for radioactive waste is a long-term process that is influenced by several factors, including (IAEA, 2003, 2011a):

- National legislation
- Organisational framework defining the roles, responsibilities and authorities of the different actors involved in the disposal programme
- Technical and scientific basis
- Waste inventory (Section 5)
- Availability of potential sites and site characteristics including geological, hydrogeochemical and climate conditions (Section 6)
- Choice of disposal concepts and designs (Section 7 and Section 8)
- Socioeconomic factors, e.g. public acceptability, cost, site ownership, existing infrastructure and transport routes
- Involvement of stakeholders.

Repository development can be described as a sequence of interrelated steps, which require integrative and iterative process management (IAEA, 2003). Typically, the disposal programme includes formal key decision points at which the development is reviewed, safety assessments are conducted and the safety case is prepared or updated and provided to the regulatory body for review, before eventually receiving a licence to proceed to the next stage (IAEA, 2011a, 2014). As a general rule, the interaction between the implementer and the regulator and other stakeholders should be initiated at the beginning of the disposal programme. This interaction becomes particularly important when taking major decisions. Stepwise and iterative process management enables flexibility in the development so that new technical and scientific insights can be incorporated and helps to ensure that all relevant information has been taken into account. Additionally, the step-wise process with its iterations facilitates optimisation of the disposal facility in terms of safety performance through, for example, comparisons of design alternatives, and further provides for reversibility of decisions. This generally increases the quality of and confidence in the decision-making regarding, e.g., the direction taken in the programme, allocation of resources and prioritisation of research needs (IAEA, 2014). The latter requires balancing of confidence in the present scientific and technical understanding with the prospects of additional research, and is in the focus of the implementer-regulator duality. During all stages of repository development, uncertainties need to be systematically identified, assessed with regard to safety and, if possible, reduced or accounted for otherwise. Uncertainties need to be fully considered in decision-making and are of central importance in the safety case (IAEA, 2014).

It is convenient to divide the repository development in three stages, namely the pre-operational, operational and post-closure periods (IAEA, 2011a, 2017a):

- **"Pre-operational period –** activities that may be undertaken during the pre-operational period include the decision for action, development of the disposal concept and the safety strategy, site investigation, environmental impact assessment, site selection, initial facility design studies, the development of plans for R&D and monitoring, and the development of the detailed facility design. Construction and licensing of the facility also take place during this period."
- **"Operational period –** the operational period begins when waste is first received at the facility and continues up to the closure of all parts of the facility. During the operational period, construction activities and modifications to operations and remediation of

existing disposals may take place at the same time as waste emplacement and closure of other parts of the facility."

- **"Post-closure period** – the post-closure period begins after the facility is closed. After closure, no further waste disposals occur and all engineered barriers are in place. Active (e.g. monitoring) and/or passive (e.g. restrictions on land use) institutional controls may contribute to the safety of certain disposal facilities before and after closure of the facility."

However, in line with the iterative nature of the development process, the different phases can overlap and activities can run in parallel if the facility is planned, built and operated in a modular manner (IAEA, 2001, 2003). For example, while a part of the disposal facility is in closure phase, waste might be emplaced in other parts or new disposal units may be under construction. In addition, cross-cutting activities such as monitoring of the site (Section 9) or regulatory oversight are undertaken throughout the lifecycle of repository development (IAEA, 2003). Similarly, the safety case (Section 10) is continuously developed in an iterative manner during the different stages of the repository life-cycle. The repository development process, related key-decision points and responsibilities of different actors involved are illustrated in Figure 4-1.

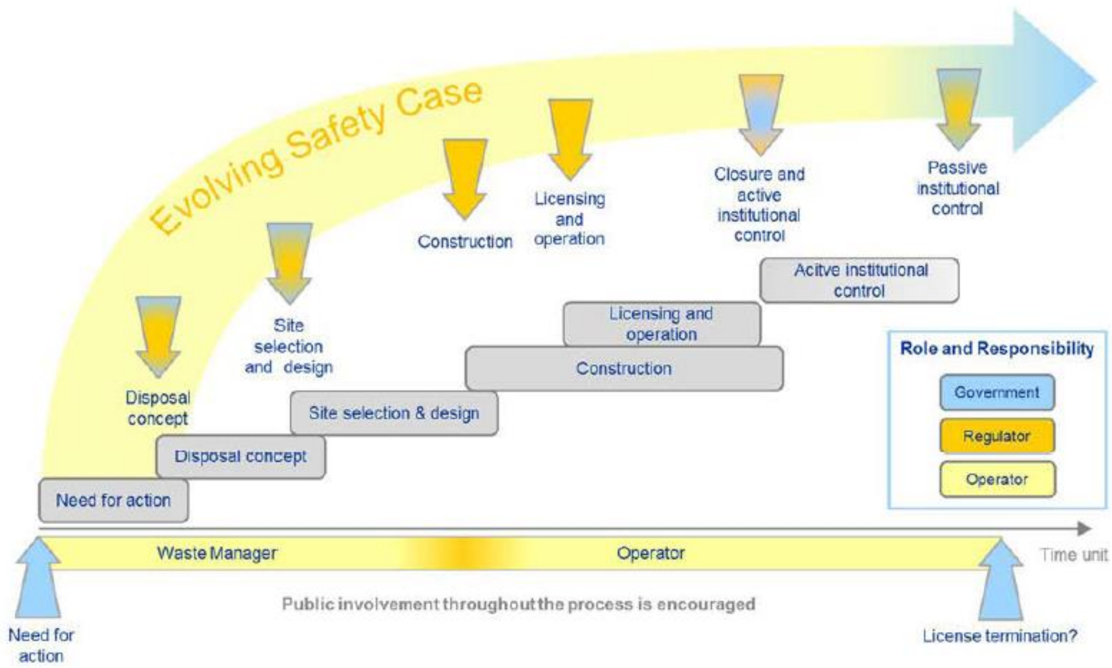

Figure 4-1. Typical sequence of key-decision points for a repository for nuclear waste (IAEA, 2017a).

The key decision steps (IAEA, 2017a) and their adaptation to the Finnish context is presented in Table 4-1. Since the near surface repository concept is applied only for deposition of VLLW, the process differs from the decision steps presented in IAEA (2017a). There will be no separate construction and operation licences, rather the only licence to be applied by the operator is termed "Licence for operation" (toimintalupa in Finnish). The authority granting the licence in Finland is STUK (Radiation and Nuclear Safety Authority in Finland). TEM (Ministry of Economic Affairs and Employment in Finland) is the contact authority for issuing Environmental Impact Assessment. The licence terminates after the closure of the repository and this is when the responsibility is shifted from the operator to the state of Finland. The regulations and guidelines concerning institutional control period remain to be specified in Finland.

The pre-operational, operational and post-closure periods are described in greater detail in the following chapters.

Table 4-1. Key-decision steps for a near surface repository (IAEA, 2017a) and adaption to Finland. STUK = Radiation and Nuclear Safety Authority in Finland; TEM = Ministry of Economic Affairs and Employment in Finland.

Decision step (IAEA, 2017a)	**Examples of decisions (IAEA, 2017a)**	**Decision steps in Finland**
1. Need for action	- Decision: go for disposal - Decision to reassess an existing facility	- Operator decides to apply for a licence for operations from STUK (*toimintalupa in Finnish*).
2. Disposal concept	- Decision on the broad disposal concept and safety strategy for given environment and set of conditions (e.g. relating to the waste). For example, engineered vaults, simple mounds or trenches, boreholes	- Description of the repository concept and safety strategy shall be included in the application.
3. Site selection and design	- Decision: choose a site and a corresponding design	- Documents and information to be included in the application are specified in the Nuclear Energy Decree, sections 42§ and 43§. - Environmental impact assessment (YVA) is addressed by the operator to TEM.
4. Construction	- Decision to proceed with construction (operator) - Decision: authorisation and/or licence for construction (authorities)	- The operator can start the construction after the licence for operations has been approved by STUK.
5. Licencing and operation	- Decision to start operations (operator) - Decision: authorization and licence for operation (authorities)	- Operator asks acceptance from STUK to start operation. - STUK gives the acceptance (note, this is not termed as an operational licence).
6. Closure and continued institutional control	- Decision to close a facility - Decision to initiate period of active institutional control	- Operator asks acceptance from STUK to close the facility. After closure the licence terminates and active institutional control period begins (responsibility of the waste is shifted to the state of Finland). The regulations concerning the active institutional period are not currently yet available.
7. Passive institutional control	- Decision to cease active institutional control	The regulations concerning the passive institutional period are not currently yet available.
8. License termination	- Decision to release a facility from regulatory control	Licence by the operator is terminated already at closure of the facility. The regulations concerning terminating passive institutional control period are not currently yet available.

4.1 Pre-operational-period

In the pre-operational phase, the site characterization and selection process is undertaken accompanied by the development of suitable facility designs and the establishment of waste acceptance criteria and ultimately followed by the construction of the repository.

4.1.1 Site characterization and selection

Depending on the hazard associated with and available options for conditioning and transportation of the radioactive waste in question, the disposal site can be located close to the source of waste generation (e.g., at the nuclear power plant site) or alternatively, a centralized facility that receives waste from several places can be considered (IAEA, 2003). As mentioned in Section 3.3, a management system should be employed to ensure the quality, usability and availability of data obtained during the site characterization programme (IAEA, 2014). During the site characterization, a baseline record of the original site conditions should be produced against which the future site evolution affected by the construction and operational activities can be compared. Section 6-1. provides detailed information about safety relevant site features, which need to be studied in the site characterization programme (IAEA, 2014).

4.1.2 Design

The disposal facility must be designed in a way that the operational and post-closure safety requirements are met and the design process is mutually interdependent with the site characteristics (Section 7.1) and the waste inventory (IAEA, 2014). In context with the operational phase, auxiliary facilities, e.g., (secondary) waste conditioning and storage facilities, buildings for electricity and water supply, and other engineered systems, need to be included in the repository design (IAEA, 2002). Operational safety requires that the facility design takes into account the normal operation mode and anticipated operational occurrences as well as incident and accident conditions. In addition, the design should be developed considering (operational and post-closure) safety and nuclear security matters in an integrated manner. Furthermore, it needs to allow for monitoring, inspection and maintenance activities and, if required, reversibility and retrievability. In line with the IAEA definition of disposal (IAEA, 2011a), waste retrieval in the post-closure phase is not intended and as such should be considered as an exceptional occurrence. Provisions facilitating waste retrieval should neither compromise safety, in particular by distracting from the passive safety approach, nor place undue burdens to the present generation or future generations (IAEA, 2014).

As described for post-closure safety in Section 3.1, the facility design should be consistent with the concept of intrinsic and passive safety, robustness and the defence-in-depth principle resulting in the assignment of complementary and independent safety functions to the different barriers of disposal system. The feasibility and constructability need to be demonstrated, particularly for novel repository concepts (IAEA, 2014). The repository design needs to ensure that the disposal systems' main safety functions containment and isolation are preserved for the required duration, which in the case of NSDF primarily corresponds to the institutional control period (a few hundred years) (IAEA, 2003). As mentioned previously, no reliance with respect to post-closure safety should be placed on active measures during the institutional control period subsequent to repository closure (e.g., repair of the cap in case of mound/hill type NSDF) and their use should be minimized (IAEA, 2014).

With the stepwise and iterative approach to repository development in mind, the design process can further be sub-divided into the following three stages (IAEA, 2001):

1. Generic, conceptual design phase

   At a very early stage, no detailed site characterization data is available and only estimates of the waste inventory and characteristics exist. Therefore, a range of disposal options with different conceptual designs can be studied with regard to their safety performance by conducting preliminary safety assessments. The preliminary safety assessments are based on generic site characteristics and the estimated waste inventory and eventually lead to the selection of a favoured disposal option and design, for which general waste acceptance criteria (Section 4.1.3) can be established. Based on the results, the requirements related to the site, waste and design can be further developed. From a management point of view (Section 3.3) it is crucial to enhance already at this stage the co-ordination of the different groups and organisations involved in undertaking the described tasks.

2. Basic engineering design phase

   After concluding the characterisation programme and the eventual selection of a disposal site, more information is available, which, together with a refined repository design and additional data on waste inventory and characteristics, can be used to improve the safety assessments. As a result, the selected disposal option and design can be confirmed and further improved, considering operational safety aspects (Section 3.2), the eventual reversibility of operations and the possibility for future extensions of the disposal facility.

3. Detailed engineering design phase

   At this stage, the design of the repository including ancillary and auxiliary facilities is finalized and the possibility for its safe construction, operation and closure is demonstrated. The final waste acceptance criteria are defined and the safety assessments and the safety case are completed and provided for licence application. The documentation includes construction and commissioning plans and specifications on operational procedures including staff requirements related QC and QA programmes (Section 3.3).

Different concepts for NSDF are presented in Section 7. Detailed information on the disposal system consisting of the geological features of the site and the engineered features of the facility contributing to the containment and isolation of the waste are presented in Section 6, together with the description of various design and material alternatives (sections 7 and 8).

4.1.3 Waste acceptance criteria

Having the various interdependencies of the different aspects and stages of the disposal facility development in mind, the establishment of waste acceptance criteria (WAC) is conditioned by many factors in order to serve several purposes. With the help of WAC, it is ensured that the safety functions assigned to the waste packages (i.e., consisting of the waste form and the container, Section 7.2) are fulfilled for the operational phase, and where applicable, for the post-closure phase. This includes the aspect of compatibility discussed in Section 3.1, meaning that the waste package characteristics do not have an undue negative effect on the safety functions of other repository components (IAEA, 2014). As the containment and isolation capacity of a NSDF is limited and as such is only a suitable option for radioactive waste, including mainly short-lived radionuclides (Section 5), the WAC need to ensure that the amounts of long-lived radionuclides are restricted. For the operational phase, it is of particular importance that the established WAC enable a safe waste handling during both, normal operation mode and anticipated operational occurrences (IAEA, 2014). The WAC specify the following safety relevant aspects of the waste packages (IAEA, 2014):

- Permissible activity levels in individual waste packages and the entire facility including long-lived radionuclides;
- Allowable dose rate and contamination on the surface of the waste packages;
- Dimensions, mass and other manufacturing specifications of individual waste packages;
- Allowable chemical and physical properties of the waste and the waste form and specifications on substances in and properties of the waste (form) to be excluded for disposal. This includes allowable uncertainties with regard to the waste characterization;

- Requirements for accompanying documentation. See Section 3.3 for quality control and quality assurance measures.

Generic and later site specific WAC, as previously described in the context of the staged process of designing the disposal facility, can be derived with the aid of safety assessments for the operational and the post-closure phase (Section 10) (IAEA, 2004). Regarding limitations applicable to long-lived radionuclides, activity levels/concentrations of individual waste packages or the entire disposal facility may be complemented by operational considerations, e.g., controlling the distribution of waste packages with higher activity levels/radionuclide concentrations within the different parts of the repository (IAEA, 2009a).

4.1.4 Construction

As described in Section 3.3, an appropriate QC and QA programme in connection with an efficient document management system are essential at all stages of the repository development, including the construction phase (IAEA, 2014). Special attention has to be given to meeting the specifications (e.g., material choices, construction plans) set in the safety case and licence documents in order that the safety functions of the different components and structures are fulfilled during both the operation and the post-closure period. Deviations in the ‘as-built’ facility need to be evaluated in terms of safety and if a significant impact is identified, the safety case needs to be updated accordingly (IAEA, 2014).

4.2 Operational period

The operation of a NSDF usually lasts between 30 and 40 years and comprises the following activities (IAEA, 2002; IAEA, 2014):

- Commissioning of the facility;
- Waste receipt and eventual conditioning and temporary storage;
- Waste emplacement;
- Installation of barriers (e.g., backfills, covers and sealing) and closure of the facility,
- Operational monitoring (Section 9.2);
- Surveillance and maintenance;
- Nuclear security provisions preventing unauthorized access and removal of radioactive material; and,
- Eventual emergency activities.

During the operational phase it might be necessary to install temporary covers or sealing structures (e.g., geo barriers like cut-off walls to change the hydraulic conditions of the site) to protect the not yet closed parts of the disposal facility from infiltrating groundwater and rain or snowfall (IAEA, 2002).

Deviations in the disposal facility of the original licensed design may be undertaken as result of, for example, improved information of the site characteristics gained from the monitoring programme during the operational phase (Section 9.2), improvements of materials or emplacements techniques or new insights obtained from periodic safety assessments (IAEA, 2003, 2014). As already discussed for the construction phase in the previous section, all changes require a clear and encompassing documentation within a record keeping system (Section 3.3) and a thorough evaluation with respect to safety. The recorded information may be relevant for eventual reassessments or to decide about corrective actions, if required in the future (IAEA, 2014).

Active safety measures during the operational phase are surveillance and maintenance. These activities include an appropriate aging management as part of the QC and QA programme helping to reduce the need for maintenance of repository components and structures by regular visual and physical inspections (IAEA, 2002, 2014). With regard to NSDF near or at the ground surface (Section 6), phenomena particularly relevant for the integrity of covers or caps of already closed disposal units are erosion, cracking, subsidence, deflation and damages induced by burrowing animals (IAEA, 2002). However, as emphasized earlier, post-closure safety must not rely on active means but the disposal system should provide safety passively by its site and engineering features contributing to the confinement and isolation of the radioactive waste.

4.2.1 Closure

During the operational phase, a closure plan should be prepared taking into consideration (IAEA, 2014):

- Waste characteristics, as well as timing and location of waste emplacement activities;
- A description of the types of barriers, e.g., backfills or final cap (Section 8), including materials used and installation techniques as well as the expected barrier performance;
- Duration of eventual temporal covering of individual disposal cells and timing of their final closure. The final closure of parts of the disposal facility should be undertaken early after the completed waste emplacement. The impact of the sequential closure of the disposal facility on the post-closure safety needs to be well understood and considered in the safety case (Section 10);
- Decommissioning of no longer needed parts of the facility (e.g., administrative buildings, temporary storage units, etc.) and environmental restoration;
- Transfer to institutional control (Section 4.3) including the type and duration of planned actions, e.g., installation of durable site markers, site access and land use restrictions to prevent human intrusion, post-closure monitoring and surveillance programmes (Section 9.3), and the responsible organisations.

The closure plan, activities conducted and eventual deviations in the actual closed facility need to be thoroughly documented within a document management system (Section 3.3). The closure plan can also serve as a communication tool informing interested parties and the affected and general public about the post-closure plans and their possible effects in the local community (IAEA, 2014).

4.3 Post-closure period

With the objective to not place undue burdens on future generations, the long-term safety of a disposal facility has to be ensured passively by a system of complementary, independent and compatible safety functions assigned to a series of barriers contributing to the containment and isolation capacity of the disposal facility (Section 3.1. Accordingly, active measures can only support and enhance the passive safety approach but no overly extensive reliance can be placed upon them with regard to post-closure safety (IAEA, 2002). In particular, active measures taken during the institutional control period subsequent to facility closure cannot justify any reduction in the performance of the described passive safety concept (IAEA, 2002, 2006a). The required length of the institutional control period is determined by the activity levels and concentrations of long-lived radionuclides contained in the emplaced waste and are assumed to last a few hundred years at the longest, which in turn influences the amounts of long-lived radionuclides to be possibly disposed of and so the waste acceptance criteria (Section 4.1.3) (IAEA, 2014).

Institutional controls can be divided in active and passive measures, i.e., requiring or not requiring future actions by the operator or others, respectively (IAEA, 2003). Active measures comprise access controls, post-closure monitoring (Section 9.3), surveillance of barriers and monitoring systems and eventual corrective actions, while passive measures include durable site markers, land use restrictions, archiving of information in local, national or international records and knowledge transfer to future generations (IAEA, 2002, 2003, 2014). As previously discussed, the actions planned during the institutional control period need to be described already in the closure plan and be thoroughly recorded in a document management system facilitated by a QC and QA programme (Section 3.3). Institutional control measures, such as post-closure monitoring, can also contribute to the confidence of stakeholders in the safety of the disposal system (IAEA, 2003).

After the institutional control period, the safety of the disposal facility relies fully on the passive system to ensure containment and isolation of the waste. Of particular significance for post-closure safety, human intrusion can no longer be excluded and requires special consideration in the safety assessments performed as part of the safety case (Section 10) (IAEA, 2014). In case of NSDF at intermediate depth and thus with a greater isolation capacity, e.g., rock caverns or borehole type repositories, institutional control is less important for the prevention of human intrusion (IAEA, 2002).

5. Waste types suitable for a near surface disposal facility

In contrast to geological disposal facilities, the capacity of NSDF to contain and isolate waste can be maintained over a substantially lesser timescale due to the location at or near the earth's surface and the related degradation processes and events occurring in the biosphere (IAEA, 2014). In particular, the considerably higher potential for human intrusion, inadvertent or intended, following cessation of intuitional controls (maximum assumed period of a few centuries after closure; Section 4.3), limits the capacity and the scope of NSDF (IAEA, 2014). Therefore, NSDF are a suitable option only for short-lived (i.e., half-lives of less than approximately thirty years) low and intermediate level radioactive waste (LILW-SL), and limits on the total activity of long lived radionuclides contained in the waste must be established in the waste acceptance criteria (Section 4.1.3) (IAEA, 2002, 2014). LILW-SL represents only a small fraction of the total activity but more than 90% of the total volume of all radioactive waste worldwide. Thus, LILW-SL is considered an important waste category (IAEA, 2003). Relevant radionuclides in LILW-SL are ^{137}Cs and ^{90}Sr, which will decay to radiologically insignificant levels in about 300 years, after approximately ten half-lives (IAEA, 2002). This length of time determines the necessary duration of the institutional control period subsequent to facility closure (Section 4.3).

LILW is generated by a number of different activities. During the operation of a nuclear power plant (NPP), LILW is generated in the form of ion exchange resins, solid concentrates from the treatment of liquid wastes (e.g., organic solvents and liquid scintillators), activated components and contaminated equipment and tools such as papers, rags, working clothes and gloves (IAEA, 2003, 2009b). Large volumes of operational waste from NPP contain only low quantities of long-lived radionuclides and are generally suitable for disposal in NSDF (IAEA, 2002). At nuclear research facilities, sources of LILW are solid concentrates from liquid waste treatment and laboratory equipment and items of various materials including metal, glass, plastics and cotton (IAEA, 2003) .The use of radioisotopes in medicine and in medical and biological research produces LILW including animal carcasses, contaminated blood, small plastic bottles, needles and syringes and glass (IAEA, 2002). The suitability of these wastes for disposal in NSDF depends on the characteristics of the specific radioisotopes involved. Similarly, disused sealed radioactive sources originating from medical, research and industrial activities include many different radionuclides, e.g., ^{60}Co, ^{90}Sr, ^{137}Cs, ^{226}Ra, ^{241}Am and ^{252}Cf. Only those wastes containing short-lived radionuclides can typically meet the waste acceptance criteria for deposition in NSDF (IAEA, 2002, 2003).

Another source of significant volumes of LILW is the decommissioning and dismantling of nuclear facilities. Approximately 95% of the total volume of radioactive waste generated in reactor decommissioning falls into the category of LILW (IAEA, 2002).This includes decontamination solutions and materials, contaminated building materials, pieces of metalor wood, electric wires, etc. (IAEA, 2003). A significant fraction of wastes arising during decommissioning activities (or also from the clean-up and remediation of contaminated sites) can be categorized as very low level waste (VLLW) and thus, the proximity of the disposal facility may be an important factor in the site selection process for practical considerations such as transportation (IAEA, 2014).

An additional category of radioactive waste comprises naturally occurring radioactive materials (NORM) including radioactive substances occurring in nature (e.g., in sediments) such as uranium (^{238}U) and thorium (^{232}Th) and their decay products (daughters), as well as radioactive potassium (^{40}K). Also to be taken into account are radioactive substances accidentally released from nuclear facilities (especially ^{137}Cs) that may occur in sediments or in other natural materials. When peat, coal, wood, biomass from woods or fields or by-products from the pulp and paper industry are burned, the ash may also include some radioactive substances. Sediments such as gravel and sand as well as ash may be used in construction, e.g., as a raw material for concrete.

According to Finnish legislation (YVL D.5/101), LILW wastes arising from the operation and decommissioning of nuclear power plants and other nuclear facilities are to be disposed of in bedrock repositories constructed at an intermediate depth. However, VLLW as defined in Section 2.2.1 can, according to Finnish legislation (STUK YVL D.5), be disposed of in repositories constructed near the surface (2018/2/13).

STUK Y/5/2016 regulates the management of radioactive waste originating from mining of uranium and thorium (e.g., tailings, processing waste) and allows for the possibility of disposal of such waste in NSDF of similar type to those used for disposal of VLLW arising from the operation and decommissioning of nuclear power plants. It should be noted, that toxic substances as heavy metals are often present in the mining waste and this should be also taken into account in the deposition of the waste (IAEA, 2017b). In other mining related processes where radioactive waste is generated, the disposal of the waste will be considered case by case. Similar applies to industrial processes (e.g., production of steel and production of concrete), where some radioactive waste may be generated depending on the characteristics of the used raw materials (e.g., ashes containing NORM). Besides radiological considerations, favourable characteristics of waste to be disposed of in a NSDF are a solid or solidified form, homogeneity, low leaching rate and small contents of non-degradable chemical substances (IAEA, 2001, 2003). Requirements on the waste characteristics are set in the waste acceptance criteria (Section 4.1.3). Waste acceptance can be facilitated by waste (pre-) treatment, conditioning, volume reduction and appropriate packaging (IAEA, 2001).

According to the IAEA (2017b) the treatment of the nuclear waste is usually preceded by characterisation of the waste and possibly also by pre-treatment of the waste. Pre-treatment includes operations such as collection, segregation, decontamination and chemical adjustment (IAEA 2017b). The waste treatment and conditioning processes aim at improved safety or economy by changing the characteristics of the waste including volume reduction, radionuclide waste removal from the waste or change of physical or chemical properties (IAEA 2017b). Considering VLLW from nuclear power plants, the treatment depends on the type of the waste. In case of organic operational waste (e.g., contaminated clothing, plastics, paper, wood and other organic matter), the waste could be treated, e.g., by compaction into pallets to reduce the volume of the waste. Another alternative for this type of waste could be thermal treatment (e.g., incineration, pyrolysis) to reduce the volume of the waste followed by immobilisation of the ash produced (IAEA, 2002, 2017b). For example, the advantage of incineration would be very high volume reduction of the processed waste and high throughput of the process. The disadvantage of thermal treatment is high investment costs and the need to meet environmental requirements for discharges (IAEA, 2017b). There are restrictions applicable to the gaseous effluent generated with thermal treatment methods and monitoring of radioiodine (^{131}I), tritium (^{3}H) and radiocarbon (^{14}C) is usually required, in addition to monitoring of chemical substances such as oxygen (O_2), carbon monoxide (CO), carbon dioxide (CO_2), sulphur dioxide (SO_2) and nitrous oxides (NO_x) (IAEA, 2017b). The metallic VLLW waste (waste metal scrap) can be cut into pieces and compacted to reduce the volume of the waste. Thermal treatment by melting may also be possible (IAEA, 2017b), but is not likely an economically feasible alternative. A new alterative could be molten salt oxidation (i.e., salt oxidation and thermochemical treatment), where the radionuclides are captured by the salt (IAEA, 2017b).

In general, waste conditioning aims at producing a stable waste package wherein hazardous substances are immobilised (IAEA, 2003, 2017b). Waste forms, including different conditioning materials, are further discussed in Section 7.2.1.

6. Repository site conditions in Finland

6.1 Geological conditions

6.1.1 Sediment types

Sediments and soil at the site are considered part of the natural barrier system. The formation of sediments in Finland has been strongly affected by multiple glaciation cycles, the latest ending roughly 10 000 years ago (Haavisto-Hyvärinen & Kutvonen, 2007). The most abundant sediment type in Finland is glacial till consisting of old sediments re-organized under a glacier (see Figure 6-1 and Haavisto-Hyvärinen & Kutvonen, 2007). Glacial till (especially basal till/moraine) has an even grain size distribution from fine particles (<0.063 mm) up to large stones (Haavisto-Hyvärinen & Kutvonen, 2007). Glacial till can be considered a suitable sediment type underneath a near surface repository with relatively good sorption capacity (e.g., illite as a clay mineral in the fine fraction), good bearing capacity and relatively low hydraulic conductivity. The downside of the glacial till with fine fraction is that the material is susceptible to frost heaving because water cannot escape the material rapidly and when frozen, ice lenses tend to form under bigger stones heaving the stones upwards. These stones then can, e.g., rupture a synthetic liner if this is not taken into account by taking measures to prevent ground frost. The thickness of the glacial till formations vary (and requires on-site study with core-drilling), but is typically some meters on top of bedrock.

Other sediment types abundant in Finland are sand-rich formations (e.g., eskers) formed as result of running water underneath or in front of a glacier, and clay beds formed at the bottom of different glacial lake and sea phases after the previous glaciation (Haavisto-Hyvärinen & Kutvonen, 2007). Sand-rich sediments are not necessarily optimal at a near surface repository site as these sand-rich formations generally have high hydraulic conductivity, good hydraulic connection with the surrounding environment, and may be important for water supply (classified as groundwater areas class I, II or E, see section 6.4 groundwater conditions). Clay formations, on the other hand, have good cation sorption capacity and low hydraulic conductivity (clay minerals and fine grain size distribution), but they tend to have poor bearing capacity with respect to construction and use of the repository. Peatland areas that are typically former lakes dried by the land-uplift (Haavisto-Hyvärinen & Kutvonen, 2007) are not considered optimal, since these areas also have poor bearing capacity and they function as groundwater discharge areas.

The sediment map of Finland is available on-line at: http://www.gtk.fi/tietopalvelut/varmuus/julkaisut/kartat/index.html

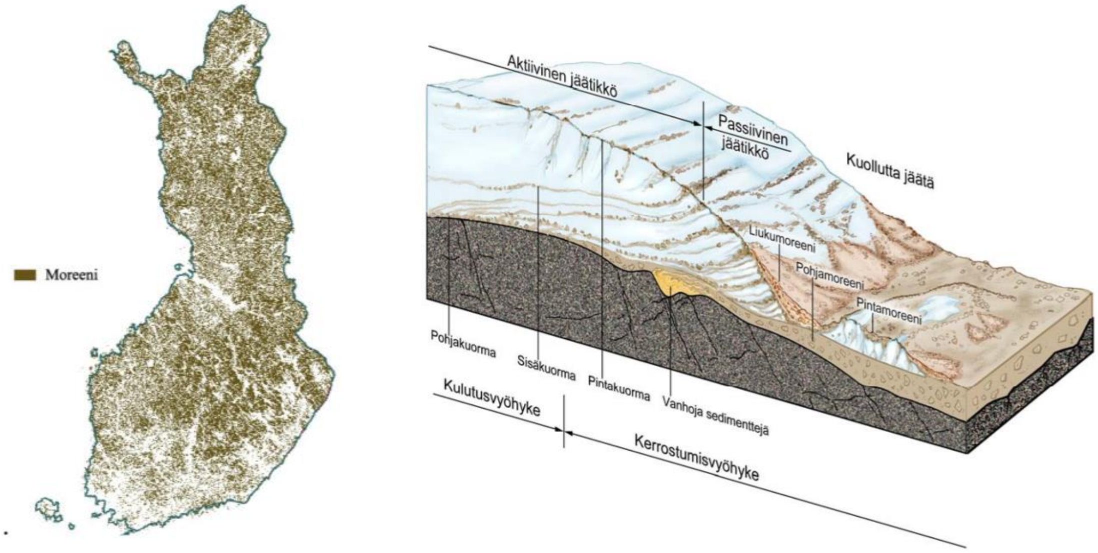

Figure 6-1. Abundance of glacial moraine (moreeni) in Finland (on the left) and formation of glacial till under a glacier (images reproduced from Haavisto-Hyvärinen & Kutvonen, 2007). Basal till (pohjamoreeni) is formed underneath the ice sheet from former sediments (vanhoja sedimenttejä).

6.1.2 Bedrock

Bedrock is also considered part of the natural barrier system at a repository site. At some sites, there may be limited sediment overburden on the bedrock, exposing the rock surface. Finnish bedrock consist primarily of tectonically stable crystalline bedrock forming approximately 1/3 of the Fennoscandian shield area (see Figure 6-2). The rock types vary in different locations, but in in the Swecofennian province (southern part of Finland) the bedrock consist typically of granitoids or gneisses of metamorphic origin (Nironen, 2017). The bedrock maps in different locations are available on-line at: https://gtkdata.gtk.fi/Kalliopera/index.html

Considering a near surface repository in Finland, more relevant than the rock type of the bedrock is the fracturing of the rock and location of fracture zones and potential faults at the site. A fracture, and especially a fracture zone, represents a potential pathway for radionuclide transport (Hölttä et al., 2004). Thus, major fracture zones are to be avoided in the location of the repository. At the same time, the fracture minerals are known to also function as a natural barrier for certain nuclides through sorption (e.g., smectites; (Andersson et al., 1983, Won-Seok et al., 2019).

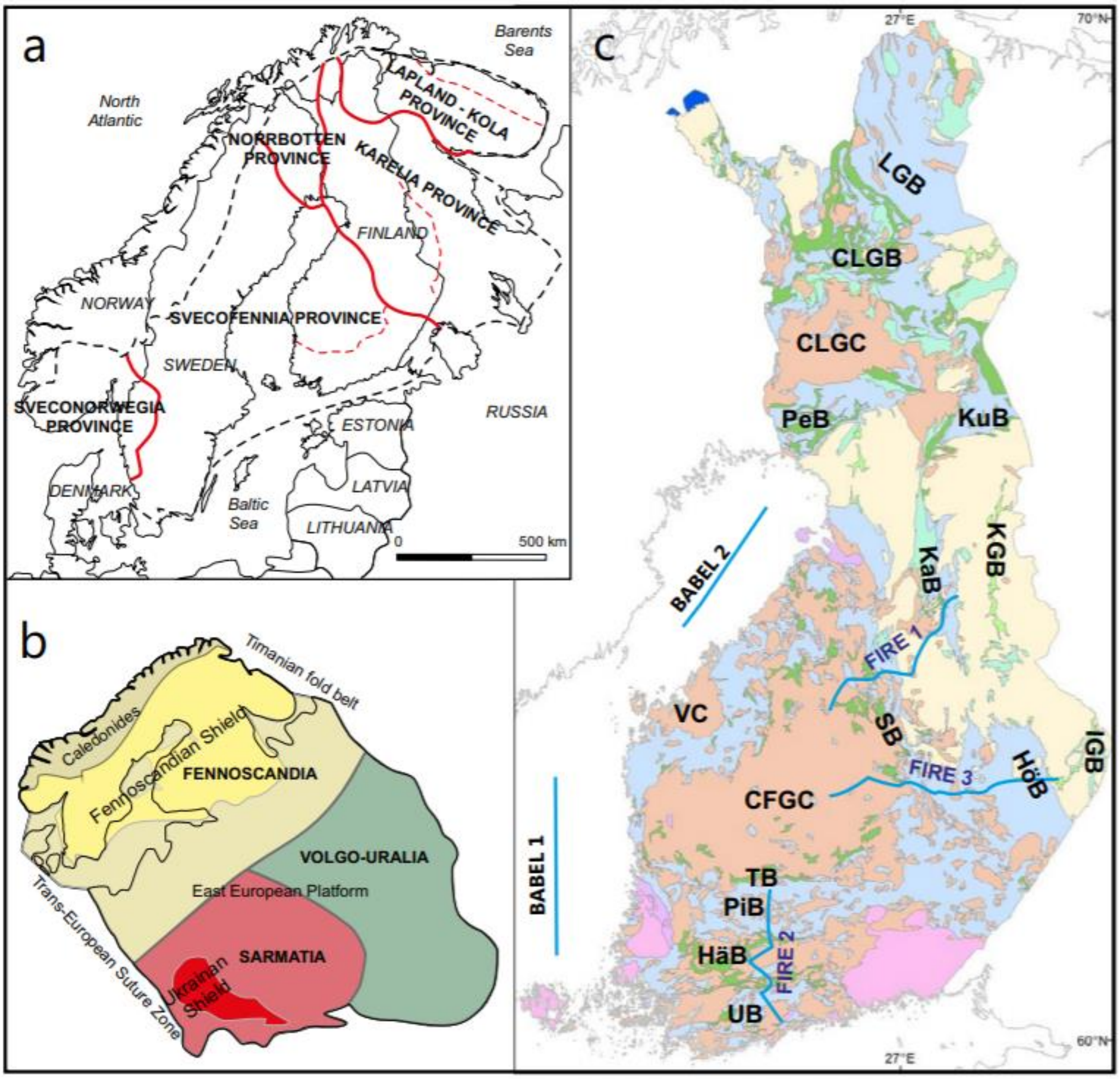

Figure 6-2. Tectonic provinces and generalised bedrock map of Finland (images reproduced from Nironen, 2017). a) Tectonic provinces in the Fennoscandian Shield. Province boundaries are shown by a solid red line, and sub-province boundaries by a broken red line. b) The East European Craton (Baltica) with crustal segments Fennoscandia, Sarmatia and Volgo-Uralia. c) Generalized bedrock map of Finland. The locations of deep seismic reflection profiles are shown by blue lines. LGB = Lapland granulite belt, CLGB = Central Lapland greenstone belt, CLGC = Central Lapland granitoid complex, PeB = Peräpohja belt, KuB = Kuusamo belt, KGB = Kuhmo greenstone belt, KaB = Kainuu belt, SB = Savo belt, IGB = Ilomantsi greestone belt, HöB = Höytiäinen belt, VC = Vaasa complex, CFGC = Central Finland granitoid complex, TB = Tampere belt, PiB = Pirkanmaa belt, HäB = Häme belt, UB = Uusimaa belt.

6.2 Ground frost

Ground frost is a seasonal phenomenon in Finland occurring during winter months when the temperature decreases to less than 0 °C (Farouki, 1992). The depth where the ground frost is able to penetrate at a near surface repository site can be estimated based upon:

- Freezing index F (*pakkasmäärä* in Finnish). The freezing index represents the amount of frost over a year and it is calculated from the degrees of Celsius below 0 °C and total sum of hours when these conditions prevail (Farouki, 1992). The freezing index usually varies from year to year, and therefore a suitable F_d value needs to be defined as a design basis for the application, usually meaning a maximum F-value for a certain number of years. In Finland, the F_d used is commonly F_{50} (Farouki, 1992; Liikennevirasto, 2018); example shown in Figure 6-3). The exact formula and starting data for calculating F in Finland is presented in RIL (2013).

- Depth of the snow coverage (Farouki, 1992). Snow coverage acts as an insulator and decreases the depth of ground frost. Therefore, changes in snow cover as a result of climate change may affect the depth of ground frost. The average and maximum depth of snow coverage in Finland can be found in the statistics of Ilmatieteen laitos: https://ilmatieteenlaitos.fi/lumitilastot.
- The prevailing sediment type on the site (Farouki, 1992). The frost depth is highest for clays, silts and silt-rich sediments (e.g., fine-rich moraine) and lowest for coarse sediments such as gravel and sand with superior ability to drain water (see Figure 6-4).

The correlation of frost depth (m) and freezing index F (h °C) is shown in Figure 6-4 for different types of sediments and varying snow coverage. The frost-free depth in frost susceptible sediments in Finland is presented in Figure 6-5.

Considering a near surface repository, the ground frost has risks linked to maintaining the stability and properties of the engineered barriers made of natural geo-materials (e.g., sealing layer consisting of a mixture of bentonite and aggregate) and frost upheave of, e.g., larger stones in a fine-rich moraine leading to puncture of, e.g., a synthetic geomembrane or a bentonite mat. For a landfill-type repository, the guidelines for dimensioning structures against ground frost are presented in detail in SYKE (2002, 2008) and in RIL (2013) (see also Figure 7-2). In practice this means that the uppermost layer on top of the repository must be sufficiently thick (>1 m) to shelter the critical layers (e.g., sealing layer) from ground frost. Considering sealing layers below the waste, the possibility of ground frost should be taken into account if the operations take place during the cold season (T <0 °C).

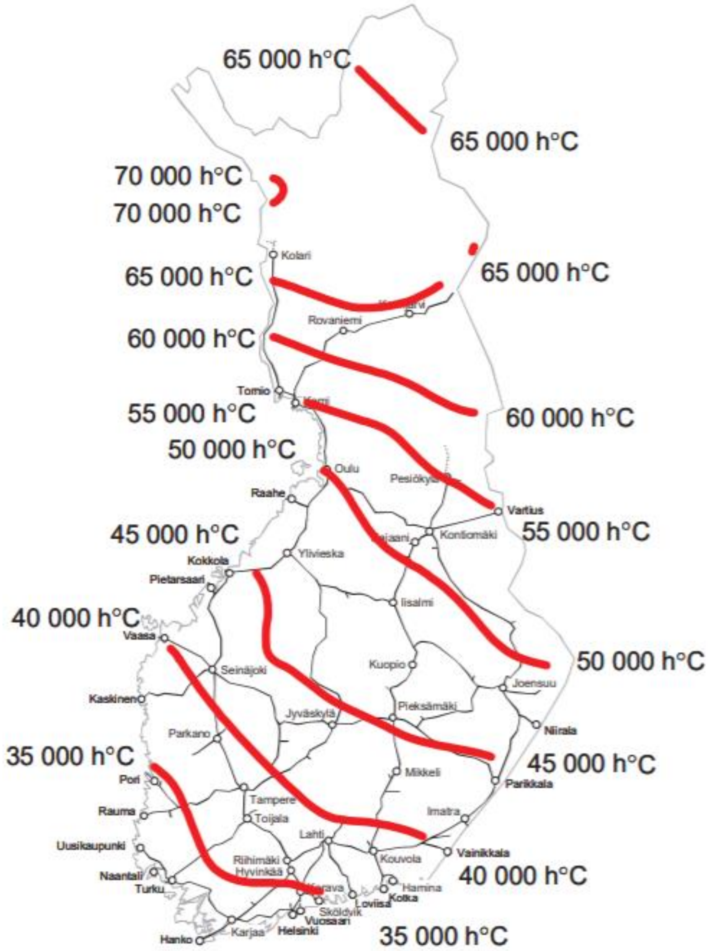

Figure 6-3. Freezing index F_{50} values in Finland (Liikennevirasto, 2018).

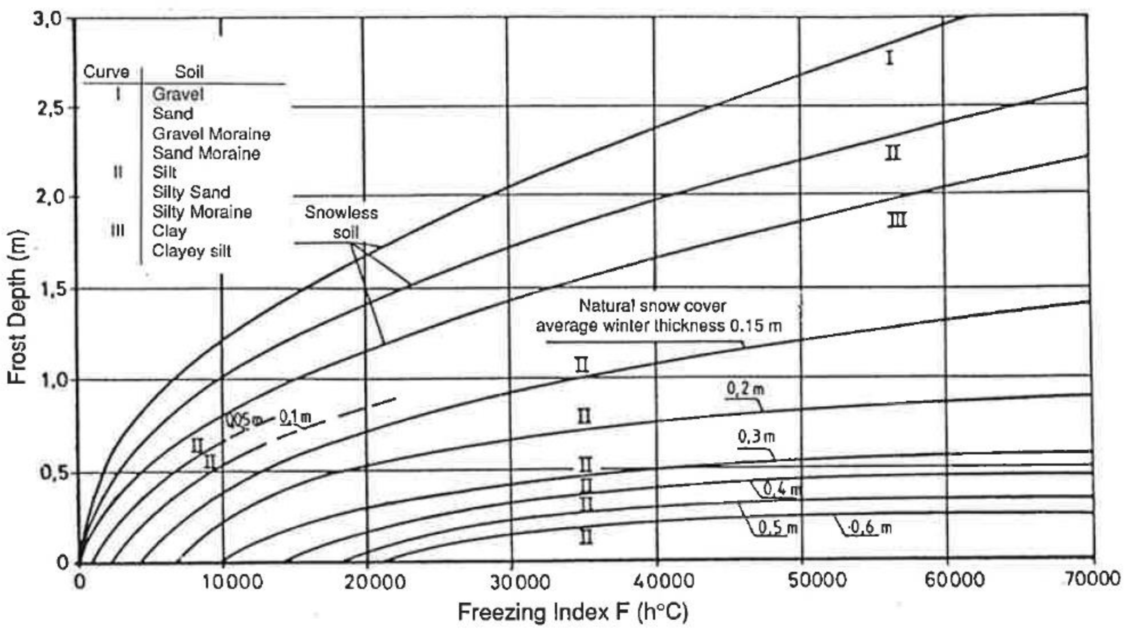

Figure 6-4. Influence of freezing index and snow cover thickness of frost depth (reproduced from Farouki, 1992, originally presented in Soveri & Varjo, 1977).

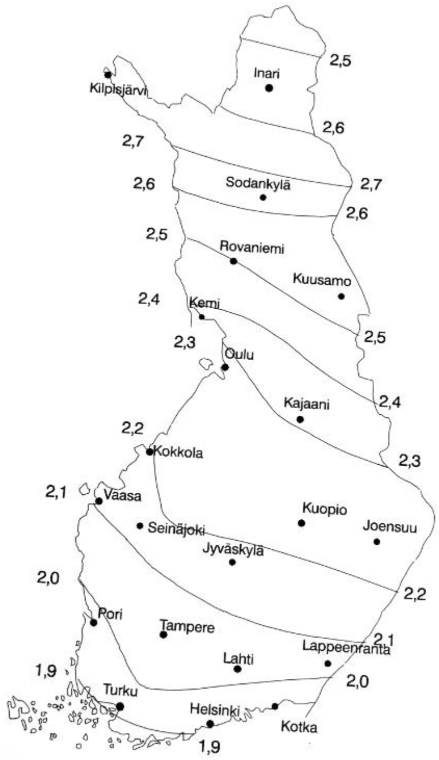

Figure 6-5. Average frost-free depth (m) for foundations of unheated structures on frost-susceptible soil in Finland (neglecting snow cover effect). Figure published with the permission of RIL ry, published originally in RIL (2013).

6.3 Background radiation

Prior to the start of operations, the background radiation needs to be determined at the site to establish a base level for the background radiation. The natural background radiation includes: 1) cosmic radiation; and, 2) radiation from ground (or building materials) caused mainly by decay of parent nuclides ^{40}K, ^{232}U and ^{238}U (Muikku et al., 2014). The radiation from the ground may vary locally due to differences in the geology of the site (Muikku et al., 2014). Cosmic radiation may also have some variation due to changes in the activity of the sun. In addition, spatial differences in the conditions (e.g., snow coverage) can have an effect on the measured background radiation levels (Muikku et al., 2014). Therefore, it is recommended that the monitoring of the background radiation levels of the planned site should be initiated, for example, a few years before the start of operations.

It should be noted that background radiation yields only part of the mean annual effective dose (3.2 mSv) for the Finnish population (see Figure 6-6), with roughly 50% of the average background radiation dose coming inhalation of radon in indoor environments (Muikku et al., 2014). In vault type repositories that would resemble an indoor facility prior to repository closure, radon should be monitored during the operations.

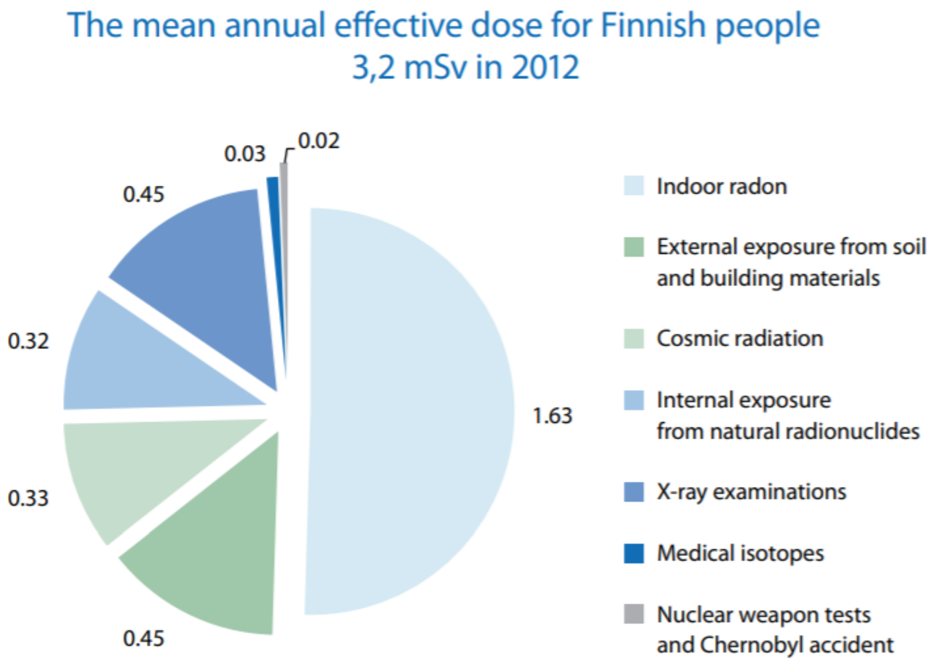

Figure 6-6. Mean annual effective dose for Finnish people in 2012 (Muikku et al., 2014). Natural background radiation marked in green.

6.4 Groundwater conditions and site hydrology

Groundwater consists of water that fills the voids in soils and sediments and in fractures of bedrock, forming a continuous entity and moving gravitationally (Britschgi et al., 2018). Groundwater is in principle formed everywhere, but how much groundwater is formed locally depends upon: a) the topography of the site (the steeper the topography, the more water is leaving the area by surface runoff and less is infiltrated into the subsurface); and, b) the geological conditions of the site (mainly properties of the sediments prevailing on the site and type and fracturing of the underlying bedrock). Important groundwater forming areas in Finland are located in areas where sand and gravel are the main sediment types formed during the previous glaciation (see section 6.1 Geological conditions) (Britschgi et al., 2018). In these areas, the sediment thicknesses are typically greater than the Finnish average and the groundwater surface can be located tens of meters below the ground surface (Britschgi et al., 2018). Elsewhere, e.g., in areas with glacial moraine as the prevailing sediment type, the groundwater surface is typically 2–4 m below the ground surface (Britschgi et al., 2018). Considering a near surface repository, the location of the groundwater surface in the repository area needs to be continuously monitored (see section 9, Monitoring).

The groundwater areas in Finland are classified as I, II or E based on their importance as a water resource, with I representing the relatively most significant resources (Britschgi et al., 2018). A near surface repository cannot be built on a classified groundwater area (I, II and E) based on the prohibition of groundwater pollution (section 17 in the Environmental Protection Act 527/2014, see also chapter 2.2.2 Legislation linked to environmental protection). In other areas, possible connections via groundwater to the surrounding environment and to, e.g., wells used for drinking water needs to be considered in the environmental impact assessment and safety case for the near surface repository.

According to Britschgi et al. (2018), the quality of Finnish groundwater is generally good, with a high level of oxygen, mild acidity and low level of dissolved substances. Considering the quality of the water at a repository site, the initial state (baseline characterisation) of the groundwater quality needs to be assessed in the environmental impact assessment prior to the start of the operations. This is important in order to be able to monitor the changes in the quality of the water during the operations and after closure of the repository. The quality of groundwater is also relevant information for the safety case. In addition to monitoring the groundwater, the discharged waters from the drainage systems need to be monitored. The monitored parameters (e.g., electric conductivity) are to be defined based on the characteristics of the site and the waste. This is consistent with monitoring of the waters from landfills defined in the Council of State Decision on Landfills 861/1997. Monitoring is discussed further in section 9 (Monitoring).

Considering the site, the local water balance also needs to be assessed in the environmental impact assessment and for the safety case. The most important factor for the safety of a near surface repository is to estimate the amount of water infiltrating through the waste during the operations and after closure. The amount of infiltrated water can be calculated via a water balance based on measured or calculated precipitation, evaporation, surface runoff, amount of drained water, etc., based upon principles presented, e.g., in Leppäranta et al. (2017) and for landfills in Christensen et al. (1989). The main principles of water handling in a typical landfill are to maintain separation between surface runoff water and water infiltrating through the waste, limit the quantity of water infiltrated into the waste, and to monitor, collect and manage the leachate water appropriately (Niemi, 2009). These principles should also be considered for a near surface repository for VLLW. In general, a water management plan is needed to address drained and leachate water, and to ensure that the collected leachate water does not contaminate local groundwaters.

In addition to water balance calculations, the surface hydrogeology at site is recommended to be modelled with suitable numerical models for the safety case purposes.

6.5 Terrestrial and aquatic ecosystems

It is known that the terrestrial ecosystem including soil, flora and fauna can have an effect on the radionuclide transport and distribution (Coughtrey & Thorne, 1983). As part of safety case for the repository (biosphere assessment and radionuclide transport analysis) the prevailing flora, fauna and soil types should be comprehensively characterised as part of the environmental impact assessment of the repository site. Monitoring during operation of the repository may be necessary to ensure that the integrity of the local ecosystem is maintained.

6.6 Climate conditions

The present climatic conditions in Finland can be described by the climate statistics from a 30-year long period starting from 1981–2010 (Pirinen et al., 2012; note that updated statistics for 1991–2020 are anticipated to become available in 2021). Considering the loadings to a near surface repository, the design should at a minimum take into account: a) mean annual temperatures and lowest temperatures at the site; b) total annual precipitation; and, c) weather and climate related risks, especially those linked to future climate scenarios. These are discussed briefly below.

6.6.1 Temperatures

The mean annual temperatures during 1981–2010 are shown in Figure 6-7. The monthly mean average temperatures in Finland are lowest during the mid-winter months (January-February) when negative temperatures and snow typically prevail in Finland (see Figure 6-8). The warmest months are during June, July and August (Pirinen et al., 2012).

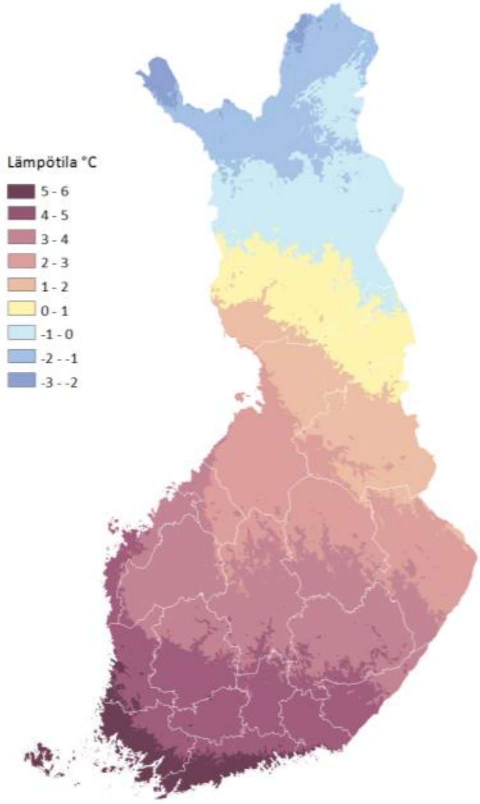

Figure 6-7. Mean annual temperatures in Finland (1981–2010) (Pirinen et al., 2012).

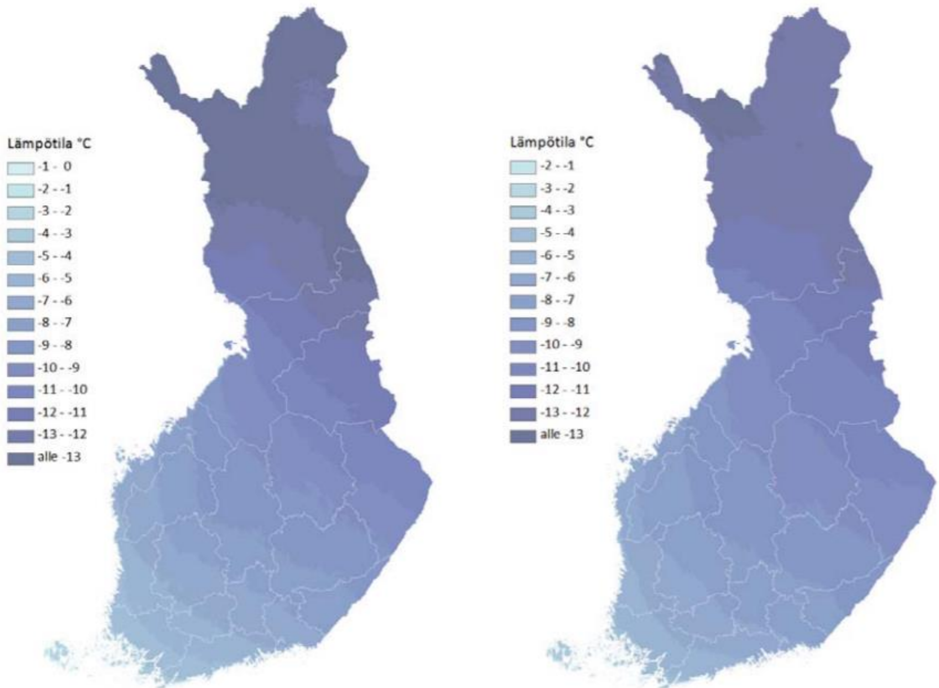

Figure 6-8. Monthly mean temperatures in the 1981–2010 period in Finland in January (left) and in February (right) (Pirinen et al., 2012).

6.6.2 Annual precipitation

The mean annual precipitation in Finland during the period 1981–2010 is shown in Figure 6-9. The mean annual precipitation at the site must be taken into account when defining how much of the precipitation is allowed to infiltrate trough the engineered barriers into the deposited waste (either as X% of the annual precipitation or X L/m^2/a). For example if the mean annual precipitation is 600 mm, then the total amount of precipitation over an area of 10 m x 10 m, would be 60 m^3. As an example, if 1 or 5% of the annual precipitation is allowed to infiltrate to the waste, the corresponding amount of infiltrating water over an area of 10 x 10 m annually would be 0.6–3 m^3 (corresponding to 6–30 L/m^2/a).

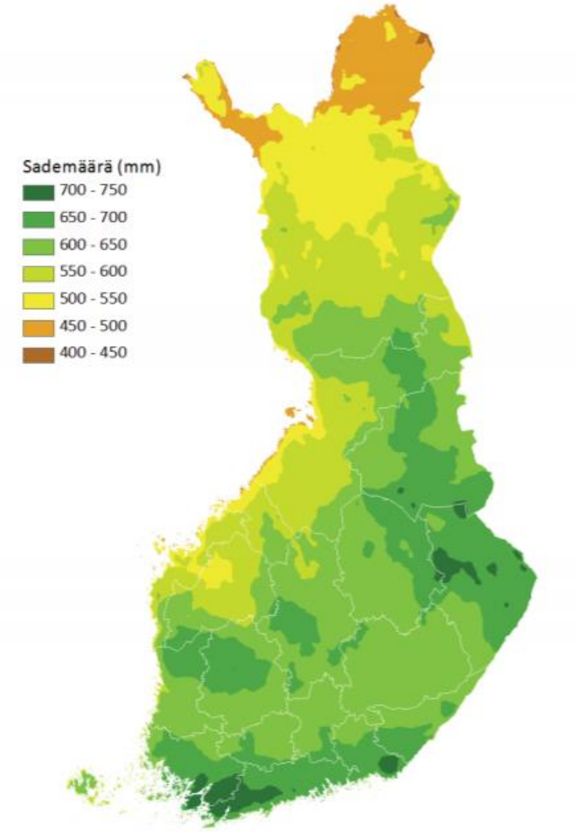

Figure 6-9. The mean annual precipitation in Finland during the period 1981–2010 (Pirinen et al. 2012).

6.7 Weather and climate related risks

The weather and climate related risks in Finland have been summarised in 2018 by Tuomenvirta et al. (2018). Considering risks to the built environment (Tuomenvirta et al., 2018, Parjanne & Huokuna, 2014), the primary risks to be taken into account include:

- **Risk of flooding in general.** Risk of flooding can be taken into account in the design of near surface repository by choosing a suitable elevation level for the repository and taking the control of surface waters into account in the design. How to define the lowest possible construction elevations in low elevation areas near shorelines in Finland are defined in the SYKE guide by Parjanne & Huokuna (2014). Prediction of areas of higher and lower risk of flooding for the next ~90 years in Finland are presented in Figure 6-10.

- **Sea level rise and flooding in coastal areas**. Considering a potential repository site near coastal areas, site-specific predictions for both sea level rise and geologic uplift (Parjanne & Huokuna, 2014) need to be considered during the selection of the site. As an example, the sea-level rise in Helsinki and Vaasa are presented in Figure 6-11. Recommendations for the minimum elevations (cm from the current sea level) for constructions located in the coastal areas considering the sea level rise and geologic uplift in the next 100 years are presented in Figure 6-12.
- **Stormwater flooding**. The capacity of the surrounding environment to store, uptake/infiltrate and cycle/evapotranspire stormwater is a critical factor in managing the surface runoff of stormwaters from a given area. A suitable repository site will have ditches, drains and appropriate structural inclinations to mitigate the potential impacts of stormwater on the repository.
- **Increased precipitation**. An increase in precipitation can be expected to increase overall infiltration through the waste deposited in a near surface repository and should also be taken into account when considering the amount of water allowed to infiltrate through the uppermost sealing layers (X% of the annual precipitation or X $L/m^2/a$). Increased rainfall intensity may also lead to erosion of the uppermost layers of the repository structures. Establishment of grassy vegetation on erodible surfaces is recommended to mitigate the potential impacts of heavy/intense rainfall.
- **Higher water content** of soil and sediments will have an effect on the material properties (e.g., friction angle and bearing capacity). This must be taken into account in the design and inclination of the engineered barriers to avoid destabilisation via, e.g., sliding of mineral-based layers on top of synthetic layers.
- **Drying of the structures**. Drying of the structures may also lead to changes in the bearing capacity of the layers. In addition, formation of shrinkage cracks on the uppermost layers can lead to erosion (e.g., as a result of heavy rainfalls following a drought). This risk can be decreased by the establishment of grassy vegetation on top of the repository.
- **Increased cloudiness and air humidity.** This will decrease evaporation of moisture during autumn and winter months and increase microbial activity. However, increased relative humidity should have limited effect on the microbial processes in the waste when the repository has been closed.
- **Increased risks for forest and ground fires** during dry seasons. Considering closed facilities, the risk of a ground fire to the safety of the repository is limited. However, this risk shall be taken into account in the operational phase of the repository.

Monitoring and warning systems for extreme weather conditions are discussed in more detail in the chapter discussing monitoring (see section 9, Monitoring).

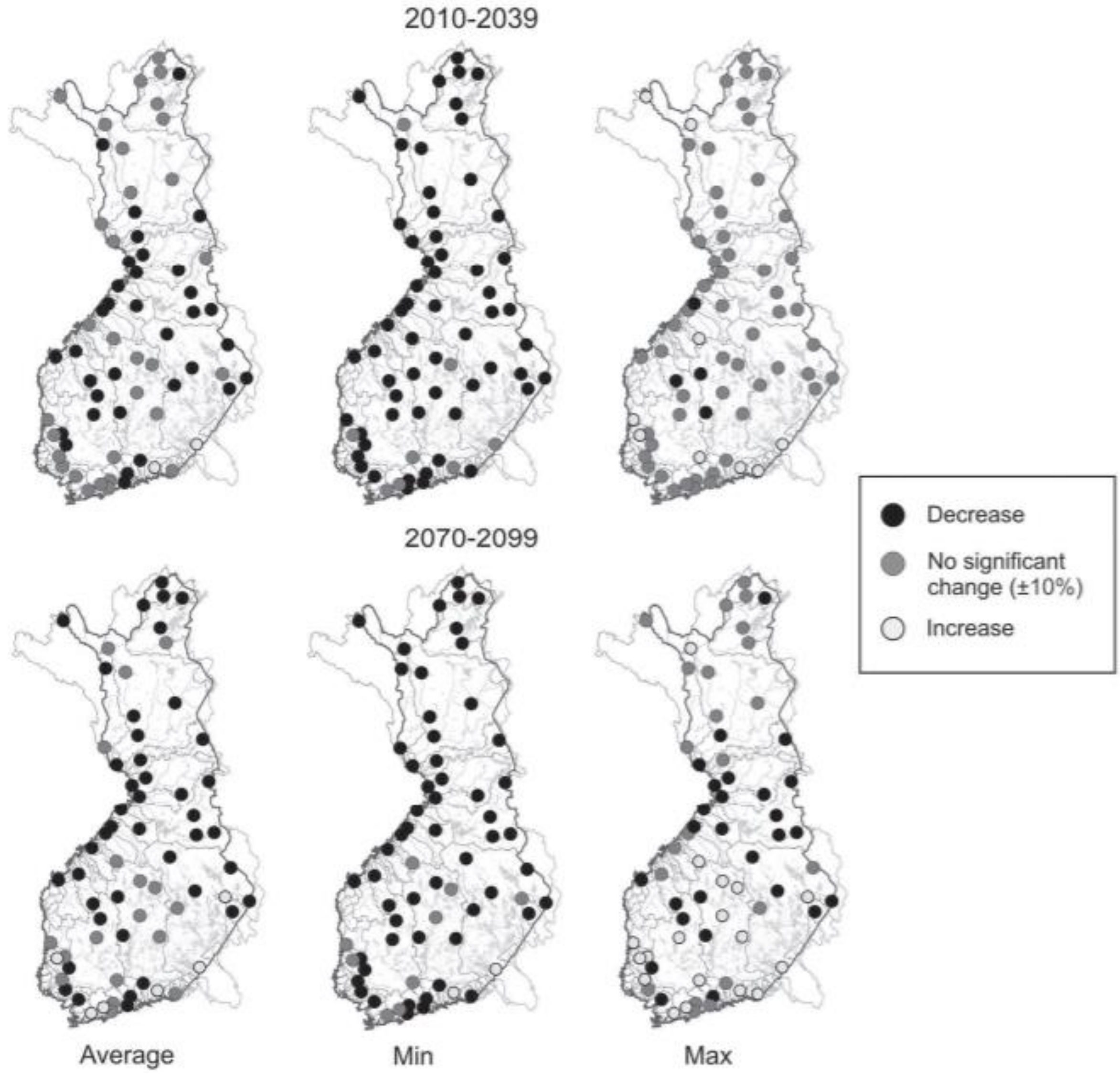

Figure 6-10. Veijalainen (2014) has predicted average (left), minimum (middle) and maximum (right) change in 100-year floods in frequency from 20 different climate scenarios in 2010–2039 (above) and 2070–2099 (below) in comparison to a control period of 1971–2000.

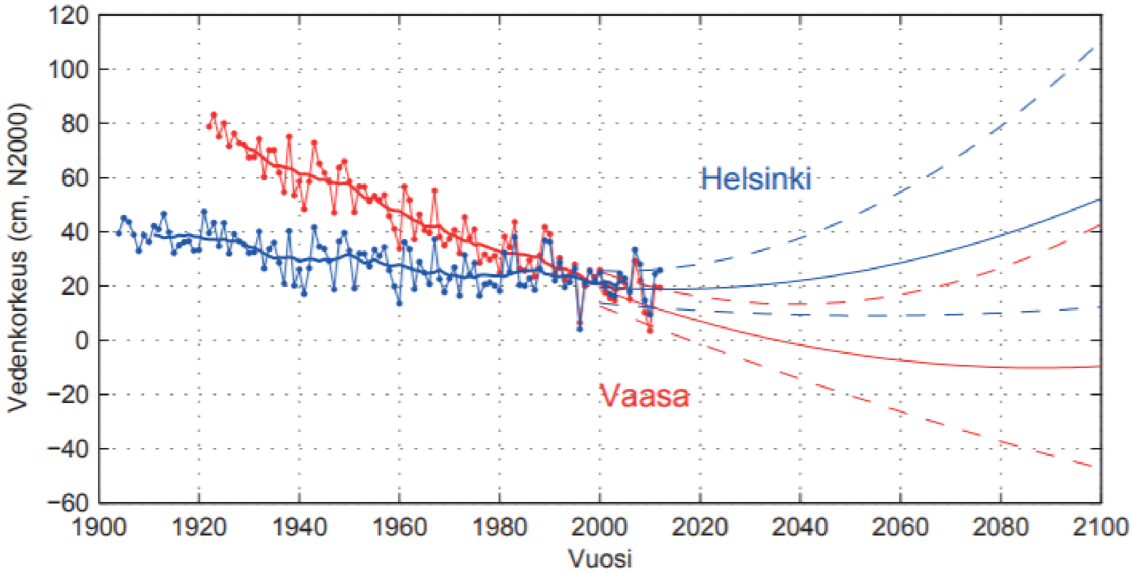

Figure 6-11. Predicted development of the sea level in Vaasa and Helsinki during the 100 years from 2000 to 2100 (Parjanne & Huokuna, 2014). The effect of ongoing geologic uplift in Finland is more pronounced in Vaasa in comparison to Helsinki.

Mareografi	Alin suositeltava rakentamiskorkeus ilman aaltoiluvaraa (cm)
Kemi	260
Oulu	250
Raahe	230
Pietarsaari	200
Vaasa	200
Kaskinen	200
Mäntyluoto	200
Rauma	210
Turku	240
Föglö	220
Hanko	250
Helsinki	280
Hamina	320

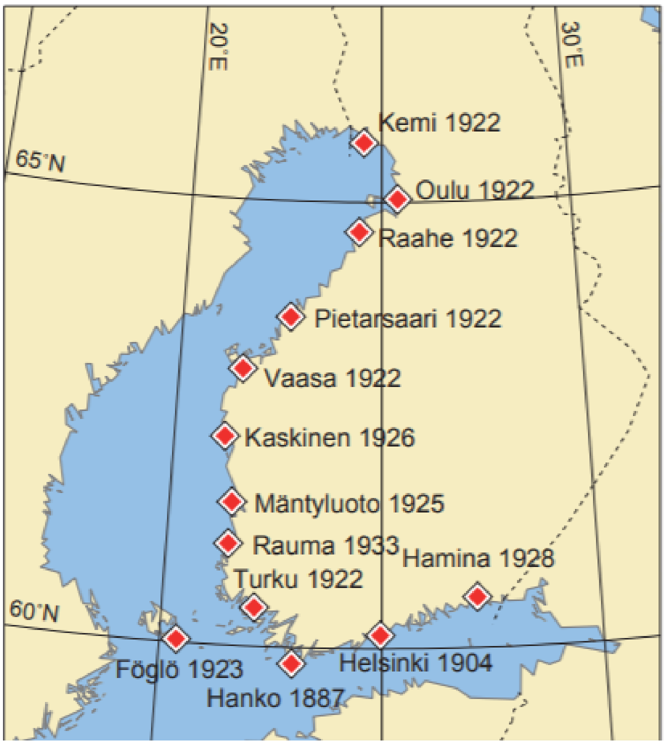

Figure 6-12. Recommendations for the minimum elevations (cm from the current sea level) for constructions located in the coastal areas considering anticipated sea level rise and geologic uplift in the 100 years from 2000–2100 (Parjanne & Huokuna, 2014).

7. Repository concepts

7.1 Location with respect to ground surface

Options for NSDF can be distinguished with respect to their location relative to the ground surface (IAEA, 2002, 2003, 2014, 2014b)

a. Above the original ground surface, e.g., mound/hill type facility

b. Below the original ground surface at shallow depth, e.g., trenches, vaults and pits

c. At intermediate depth (up to a few tens of meters underground), e.g., rock caverns, silos and tunnels. Borehole type repositories may also fall within this category.

The choice of the disposal option depends on various factors including the waste characteristics and site conditions, thus the variety of possible engineered structures is manifold (IAEA, 2014). In many countries, near surface disposal facilities are designed and constructed so that the disposed waste remains permanently above the groundwater table, i.e,. in the vadose zone (IAEA, 2002). In Finland, the groundwater surface is generally relatively close to the ground surface depending on the sediment types abundant in the area and local topography (see Section 6). This needs to be considered in the facility design in order to limit the inflow of water, avoiding that disposal units such as shallow trenches or pits become flooded. This type of disposal unit flooding is sometimes referred to as the “bath tubbing effect”, which is a possible failure mode (IAEA, 2003).

7.2 Repository design options and general principles for the design

Repository design options

In general, the repository design for a near surface repository depends on the type of the waste considered (VLLW, LLW, ILW) and the local site conditions.

IAEA (2009c) outlines the following near-surface disposal options:

1. Landfill disposal
    - Suitable for very low level waste (VLLW) with limited long-lived activity
    - No complex engineered barriers or elaborate sealing
    - Less stringent requirements for waste treatment and packaging
2. Trench disposal
    - Can be divided into individual compartments to increase radionuclide containment and flexibility of operation
    - After filling, a waterproof top cover is installed
    - Surveillance and monitoring are required after closure during the period of institutional control
3. Engineered surface repositories
    - For the disposal of short-lived waste with activity of long lived isotopes limited to 400–4000 $kBq \cdot kg^{-1}$
    - More elaborate engineered barriers to reduce water contact with waste
    - Equipped with surface barriers (caps), vertical barriers (cut-off vaults) and sub-horizontal barriers (floors)
    - Other containment technologies may be applied, including chemical barriers to retard migration of radionuclides without impeding water movement
    - May include drainage collectors to channel infiltrating water

- Underground galleries may be installed to facilitate maintenance checks of barriers
- After the waste is disposed, the void spaces in vaults are usually filled with grout or similar backfill material.

Examples of different repository disposal options currently operating in Europe are shown in Table 7-1 and example cases are described further in sections 7.3 (examples of landfill type of repositories), 7.4 (example of trench type of a repository) and 7.5 (example of a vault type of a repository).

Table 7-1. Examples of different disposal options currently operating in Europe.

Disposal option	Location	Waste type and characteristics of the repository	Reference
Landfill disposal	Ringhals, Sweden	VLLW. Landfill type of a repository built above groundwater table, see description in section 7.3.	Aronsson (2019)
	Forsmark, Sweden	VLLW. Landfill type of a repository built above groundwater table, see description in section 7.3.	Vattenfall (2018)
Trench disposal	Morvillies, France	VLLW. Trench type of structure located in ground on a clay formation. Isolated from groundwater with geomembranes. See description in section 7.4.	Andra (2017)
Engineered surface repository	El Cabril, Córdoba, Spain	VLLW + LILW. Concrete containers and vaults. Concrete slab. See description is section 7.5.	Bergström et al. (2011)
	Aube, France	LILW, short-lived. Waste packed in drums, metallic boxes and concrete containers and placed in large concrete vaults with temporary roof structures. Temporary roof is replaced by a final cap.	Espivent (2019)
	Dounreay, Scotland & UK National Low Level Radioactive Disposal Facility.	LLW. Waste packed in metallic containers and deposited in large concrete vaults. Temporary roof is keeping the facility dry during operations and in closure this roof is replaced by an engineered cap providing low permeability and prevention from human intrusion. See description is section 7.5.	Usher & Rossiter (2019)

7.2.1 General design principles for design

There are some general principles that are common for different repository options and should be considered in the design. The following principles are summarised from the presentations given in the SURFACE seminar held 26.9.2019 (presentations available at: http://kyt2022.vtt.fi/kyt2022_seminar_sept_2019.htm):

- Apply a waste hierarchy to limit the amount of waste deposited in a near surface repository.
- Prevent/limit surface infiltration of rain and surface runoff water from entering the repository and wetting the waste. Temporary structures are used during operations in

some cases. At time of closure, these structures are replaced by low-permeability final cap structures.

- Isolate the repository from groundwater. Isolate either by locating the repository above the groundwater table or by means of impermeable isolating structures (geomembranes or concrete structures) where the repository siting indicates potential intersection with the groundwater table.
- Control and monitor gases generated in the waste (if needed).
- Control and monitor leachate waters discharged from the waste.
- Prevent/limit the migration of radionuclides from the waste to the surrounding environment using both engineered and natural barrier materials (sediments and bedrock on the site).
- Ensure sufficient bearing capacity of the structures/layers below the waste.
- Ensure on-going mechanical stability of the structures/layers (sufficient stability to avoid, e.g., landslide or land slippage).
- Consolidation of the structures/layers. Some consolidation is to be expected after installation. Avoid uneven consolidation of layers by implementation of appropriate geotechnical design to prevent formation of defects in the layers/structures.
- Stabilise the uppermost layer using suitable vegetation in order to withstand erosion by heavy/intense precipitation events, etc.
- Remove deep-rooted vegetation as needed to minimise potential damage to repository structures/layers.
- Monitor, control and treat effluents/leachates from the waste as needed to mitigate discharge of potentially harmful or hazardous substances to the surrounding environment.
- Locate the repository site within a controlled area. Employ land use planning to restricted future land uses to ensure that future generations do not use the land for example for food production.

These general principles are applicable to a Finnish repository with the addition that ground frost during the winter months should also be considered in repository design and construction according to the Finnish standards (RIL, 2013).

7.3 Landfill type of repositories for VLLW

Landfill type of repositories are always constructed above the prevailing groundwater table. The foundation for the landfill depends on the local geological conditions and the repository can be built over local sediments (e.g., glacial till) or a concrete slab. The overall structure of the repository and the engineered barriers are similar to those of landfills for normal or hazardous waste. Examples of two different sites located in Sweden are presented below.

A schematic of the landfill type repository located in Ringhals, Sweden is presented in Figure 7-1 (Aronsson, 2019). The deposited waste at the Ringhals repository consists of: 1) combustible operational waste such as trash, cloth and plastic that is compacted into pallets; 2) non-combustible waste (mainly metallic waste); and, 3) resins (Aronsson, 2019). The waste is packed within plastic covered pallets and placed inside metallic containers (half the size of a sea container) with lids. All free space in the containers is filled with rock flour to prevent settlement of the repository with time. For this same reason, the spaces between the containers and volume surrounding the packages have been filled with rock flour. The waste packages with highest activity have been placed in the middle with less active waste placed in the margins. Waste deposition has been performed in different campaigns (1993, 1998, 2008 and 2016) with own licences for each phases allowing deposition of a total 100 GBq of gamma radiation emitting nuclides, maximum concentration of 300 kBq/kg (or nuclide specific) and maximum surface dose rate of 0.5 mSv/h (Aronsson, 2019). The repository was built on exposed bedrock surface and the cracks in the bedrock were evened out with concrete. Overlying the concrete is a drainage layer made of crushed rock which bears the weight of the deposited waste. Above the waste, the layers preventing infiltration of precipitation to the repository consist of a bentonite mat and moraine (glacial till). The inclination of the top layers is 1:3. This inclination was considered sufficient for controlling the surface waters, but at the same time reduces the risk of erosion and landslide. Experiences from this site (Aronsson, 2019) to date show that no leakage of leachate water to the environment has been detected through monitoring of the environment with groundwater pipes, and the deposited waste has remained dry during the ongoing operation period.

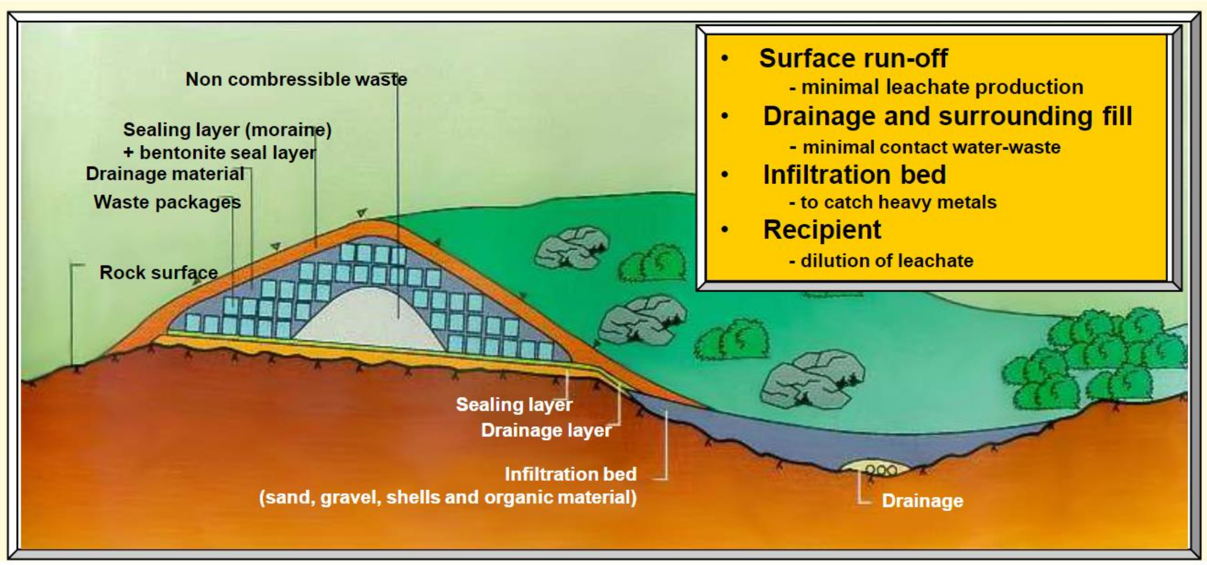

Figure 7-1. Conceptual illustration of a landfill type repository located in Ringhals, Sweden (Aronsson, 2019).

Another example of a landfill type repository is the repository site in Forsmark, where the main difference in comparison to Ringhals site is that the repository is built over a layer of sediments - fill material and glacial till as an engineered barrier material - rather than bedrock (Vattenfall, 2008). The repository was built in different phases since 1989 and has a licence from SSM (SSM, 2013) for deposition/storage of waste with a maximum 200 GBq total activity (including a maximum 0.2 GBq of alpha radiation emitting nuclides), and a maximum volume of 17 000 m^3. In addition, SSM defined nuclide specific limits (see Table 7-2) and further specified that the surface dose rate of a waste package shall not exceed 0.5 mSv/h (SSM, 2013). According to Vattenfall (2008), the design of the repository is based on Swedish regulations and guidelines for a landfill for normal and hazardous waste (Naturvårdverket, 2004). Based upon these guidelines, the geological barrier underlying the waste shall have hydraulic conductivity < $1x10^{-9}$ m/s and minimum thickness of 1 m for normal waste and 5 m for hazardous waste. In addition, the infiltration of the overlying and underlying hydraulic sealing layers shall not exceed 50 L/m^2/a for normal waste and 5 L/m^2/a for hazardous waste. According to Vattenfall (2008), the infiltration of the precipitation/surface runoff waters to the waste have been limited to ~1 L/m^2/a (hydraulic conductivity ~$1x10^{-11}$ m/s) with a combination structure overlying the waste consisting of a bentonite mat and a layer consisting of mixture of bentonite (5 wt.%) and rock flour. The sealing layer below the waste has a thickness of 0.4 m and is comprised of bentonite (1 wt.%) and rock flour (Vattenfall, 2008). The glacial till layer beneath the sealing layer has a thickness of ~2 meters and hydraulic conductivity of ~$1x10^{-8}$ m/s (Vattenfall 2008). The voids between the waste packages are filled with rock flour (Vattenfall, 2008). The drainage layers consist of coarse crushed rock.

Table 7-2 Nuclide specific limits (activity, Bq/g) for the landfill type near surface repository located in Formark (Svalören), Sweden (SMM, 2013). In case of multiple nuclides, the sum of the ratios between nuclide specific activities and the respective activity levels shall be less than one. Nuklide means nuclide and aktivitets kriterier means activity concentration limit for that specific nuclide.

Nuklid	**Aktivitets-kriterier** (Bq/g)
H-3	100
C-14	10
Co-60	0,3
Ni-59	100
Ni-63	100
Sr-90	1
Nb-94	0,1
Tc-99	1
I-129	1
Cs-137	3
Eu-152	1
U-238	1
Pu-238	0,1
Pu-239	0,1
Pu-240	0,1
Pu-241	10
Am-241	0,1
Cm-244	1

7.3.1 Applying this design option in Finland

There are currently no specific guidelines in Finland for the structures of a landfill type near surface repository, nor are there any numerical performance targets for, e.g., annual infiltration through the waste. However, regulations (Government decree on landfills 331/2013) and guidelines for landfills for hazardous waste in Finland (SYKE, 2002, 2008) could be applied to some extent in the design of the repository. An example based on SYKE (2008) guidelines for hazardous waste is presented in Figure 7-2. The basic idea of the structure is to: 1) limit infiltration of water into the waste and prevent/limit formation of contaminated leaching water (overlying layers); 2) control and collect gases generated in the waste; and, 3) prevent/limit discharge of contaminated leaching water to the groundwater and to the environment (underlying layers).

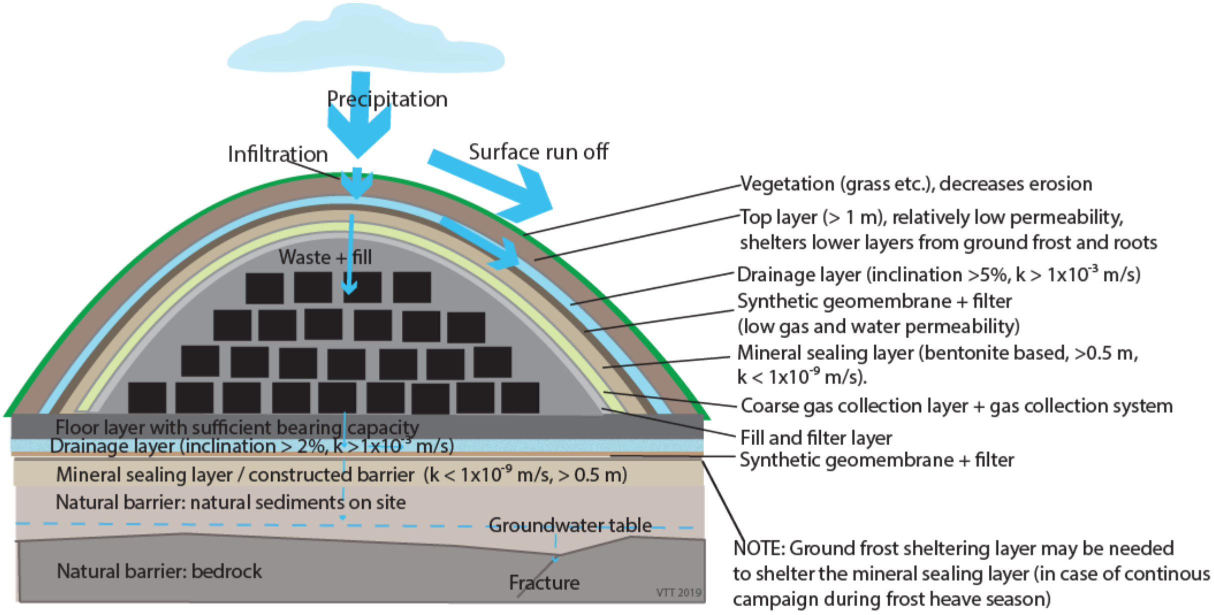

Figure 7-2. Example of a landfill type near surface repository based on Finnish guidelines for hazardous waste landfills (SYKE, 2002, 2008).

Considering the anticipated long service life of the repository and principles of multi-barrier containment, it is recommended that a mineral sealing layer be combined with a synthetic impermeable geomembrane or bentonite mat. This structure is analogous to that required for landfills for hazardous waste (SYKE, 2008) and the intent is to ensure long-term performance despite the potential for the development of defects in the geomembrane with time. The composition and thickness of the uppermost mineral sealing layer should be dimensioned based on the desired quantity of water allowed to infiltrate through the layer into the waste. According to SYKE (2008), if the hydraulic conductivity (k-value) of an uppermost sealing layer is $1x10^{-9}$ m/s, then 5% of the annual precipitation will be infiltrated through the layer. The underlying mineral sealing layer beneath the waste will act as a long-term barrier against the transport of radionuclides into the groundwater/surrounding environment. The dimensioning of this underlying layer should take into account local site conditions (underlying sediment properties and thickness), but considering guidelines for hazardous waste (SYKE, 2008) the thickness of the layer should be ≥0.5 m and possess low permeability ($k < 1x10^{-9}$ m/s).

The compatibility of the adjacent layers with one another is another matter to be aligned with the multibarrier principle. For example, there should be a mechanically protective filter layers of fine grained material on top of a geomembrane in order to avoid perforating the structure with coarse and sharp crushed rock grains, etc. In addition, the bearing capacity of the underlying layers and consolidation of the materials/layers should be accounted for to ensure that the layers remain intact (SYKE, 2008).

The vegetated uppermost layer protects lower layers from ground frost and invasion by plant roots (SYKE, 2008). Dimensioning of layers for protection against ground frost should be performed according to Finnish standards presented in RIL 261-2013 (Routasuojaus-rakennukset ja infrarakenteet (RIL, 2013).

The purpose of the gas collection layer is to avoid the deterioration of the uppermost layer due to uncontrolled gas outbursts (SYKE, 2008). Where the VLLW may produce significant quantities of gases, a plan should be in place to collect and monitor the gaseous discharges as this can be one route for escape of volatile radionuclides from the repository (e.g., ^{14}C).

There should be also a plan in place for the monitoring and possible collection and treatment of discharge waters from the drainage system, e.g., where leachate waters contain trace metals or radionuclides.

A range of engineered barrier options are discussed further in Section 8 (Multi barrier system).

7.4 Trench type of repositories for VLLW

The Morvilliers repository in France, described in detail below, provides an example of a trench type repository for VLLW. A common characteristic of all trench type repositories is their construction mostly or partly below the ground surface, usually with a temporary roof structure that will eventually be replaced by permanent structures made of geo-materials. In addition, the repository is isolated from the surrounding sediments/rock and groundwater by impermeable structures, either with geomembranes (as in the Morvilliers case) or with concrete.

The Centre Industiel de Regroupement d'Entreposage et de Stockage’s (CIRES) very low level waste repository (called CSTFA), is located in Morvilliers, France, 2 km from the Aube repository. Operation of the facility began in 2003. The overall capacity of the CIRES repository is 650,000 m^3. Approximately 50% of the VLLW at the Morvilliers repository consists of “industrial waste” (metal scrap and plastics), with the balance comprised of 40% “inert waste” (concrete, bricks, earths, etc.) and 10% “special waste”, which includes various substances such as sludges and, in some cases, powdered materials like ash (Andra, 2017). The VLLW has an average radioactivity level of 10 kBq kg^{-1}. Approximately 30% of all waste received at the CSTFA undergoes treatment prior to disposal. Low-density residues (plastics, thermal insulation materials, etc.) are first compacted by a baling press, then strapped and vinyl-coated. A bundle press is used to reduce the volume of metal scrap. Some waste, such as the polluted waters generated on site or the sludges received from producers are processed in the solidification and stabilisation unit. The containment envelope has no role in confining radioactivity, but rather in facilitating handling and disposal operations whilst protecting operators.

The CIRES waste repository is built on a homogeneous clay formation varying between 15 and 25 m in thickness (Andra, 2017) (see Figure 7-3 and Figure 7-4). Disposal cells with capacity of 10,000 to 25,000 m^3 are excavated in the clay formation to a depth of 8 m above the mean level of the water table. A geomembrane (high-density polyethylene, HDPE, 2 mm thickness) is first fitted at the bottom and on the sides of each cell before waste emplacement. Cells are filled in successive layers (ca. 10 on average) while void spaces between waste packages are gradually backfilled with sand. Once a cell is full, an identical HDPE membrane is placed over the top and thermo-welded to the existing geomembrane in order to surround the waste with a continuous water-tight barrier. The geomembrane is fully waterproof and is designed to prevent any dispersion of radioactivity and any seepage of external waters (rain, infiltrating waters) for several decades. The lower part of the envelope corresponds to the first 5 m of the clay layer located immediately beneath the geomembrane and for which a very low permeability (10^{-9} m/s or less) is guaranteed (Andra, 2017). The upper part of the waste envelope is made of clay-based materials removed during cell excavation and consists of a layer measuring 1 to 5 m thick, shaped and compacted mechanically in order to re-establish its initial low permeability (10^{-9} m/s or less). Finally, a permanent 30 cm-thick layer of grass-covered topsoil is laid over the entire structure. An opening roof is erected over disposal cells during filling to prevent the introduction of moisture via rainfall.

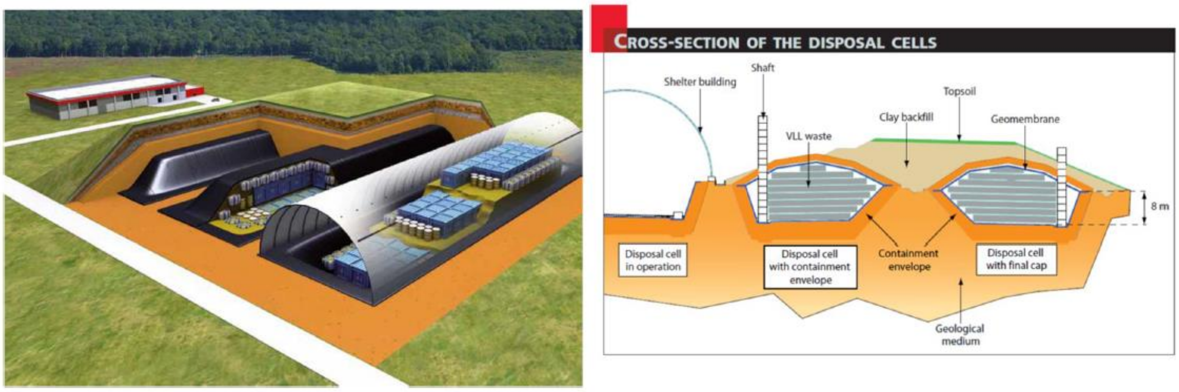

Figure 7-3. The very low level waste (VLLW) repository in Morvilliers, France (Andra, 2017; Andra solutions, 2017).

Figure 7-4. Morvilliers disposal site for VLLW in France (Tison, 2009).

The operating lifetime of the VLLW repository is about 30 years followed by a post-closure monitoring phase of at least 30 years. ANDRA carries out analyses of atmospheric releases resulting from the activities of the treatment building. As a complementary measure, the radioactive dose originating from the operation of the CSTFA waste disposal facility is estimated on a continuous basis thanks to the six dosimetric films installed at different points on the site fence. The collected data are compared with the reference film located in the woods of the nearby commune of *La Chaise*, located outside the influence area of the CSTFA waste disposal facility. Water quality is monitored while the site is filled and will continue during the subsequent monitoring period, pending the return to natural background radioactivity levels. Because water it the major potential pathway for radioactivity dispersion and the failure mechanisms, is monitored very carefully through:

- Radiological and physico-chemical follow-up of all surface waters (including stream waters) located in the vicinity of the site;
- Seven piezometers, consisting in small boreholes deep enough to reach the water table, for controlling groundwater characteristics;
- Automatic sampling operations aiming at monitoring the water contained in the storm basin before release in the *Forgeot* brook which is located downstream from the site.

7.5 Engineered concrete vault repositories

As an example of more engineered repository types for VLLW and LILW, the case study from El Cabril, Córdoba, Spain is presented briefly below. The Spanish radioactive waste management agency Empresa Nacional de Residuos Radiactivos S.A. (ENRESA) is responsible for the long term management, storage and disposal of all categories of radioactive waste in Spain. Since October 1992, LILW is managed through the El Cabril central disposal facility in Córdoba.

The El Cabril disposal system is based primarily on the interposition of engineered barriers and natural barriers which confine the deposited wastes to ensure the protection of people and the environment (Figure 7-5). There are three types of barriers interposed between the environment and the waste:

- The first barrier, consisting of the conditioned waste and the container.
- The second barrier, made up of the engineered structures that house the waste.
- 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.

Waste packages, mainly 200 L steel drums and 1.3 m^3 metal boxes, are placed within concrete containers (2.2 m x 2.2 m x 2.2 m) to form an 11 m^3 final package or disposal unit, which constitutes the first repository barrier (Zuloga, 2009). The internal volume of the concrete container may be back-filled with mortar grout, or may be used to condition institutional liquid waste or contaminated ash. These packages are placed inside 24 m x 20 m x 10 m concrete vaults. Once a vault is filled with 320 sealed 11-m^3 concrete containers, the vault is backfilled with gravel and a concrete closing slab emplaced then coated with impervious paint. When at least 50% of a given disposal zone is full, an engineered multi-layer cap will be constructed. Beneath each row of disposal vaults there is an inspection drift where two drainage systems are installed, one for rainwater collection from the vaults not yet in operation and one for the vaults containing waste packages (Bergström et al., 2016). A metallic shelter on wheels is used to protect each row of vaults from weather and to support the overhead lifting crane during the construction. The site’s monitoring and surveillance phase begins at the time of closure and lasts approximately 300 years.

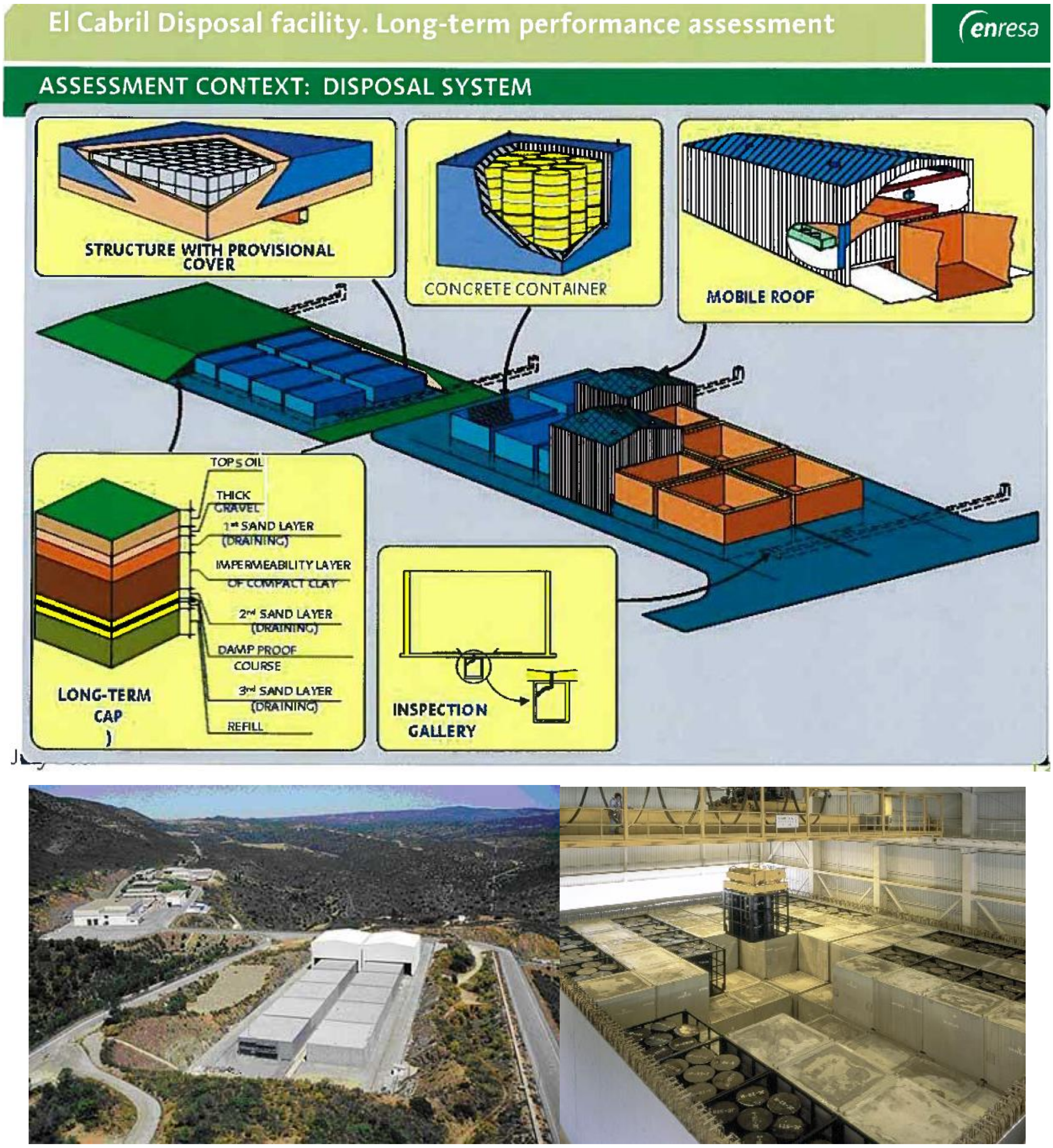

Figure 7-5. El Cabril disposal site in Spain (Zuloaga, 2009; http://www.enresa.es/eng/index/activities-and-projects/el-cabril).

Environmental Monitoring Programmes consist of measures for carrying out in-situ checks to verify that the environmental impact of activities is within established limits. This involves measuring air quality, the quality of discharges, groundwater and surface water, and the level of noise pollution, etc. These programmes set out the parameters that must be monitored, the location of the control points, the procedures for sampling, methods of analysis to be employed and the frequency with which checks must be carried out.

The main cement-based materials used in engineered barriers at El Cabril repository are the vaults, the containers and the mortar filling the gaps between the drums introduced in the containers. Vaults and containers are made of very similar concrete compositions, while the mortar was specifically designed to be pumpable as well as highly impermeable. The cement selected was with low C3A and alkali content in order to prevent sulphate attack or harmful alkali-aggregate reaction. (Andrade et al., 2006).

Andrade et al. (2006) have identified the long-term stability of concrete structures, particularly carbonation, water permeation and reinforcement corrosion, as the main disposal system durability issue in El Cabril. They have implemented the following measures to assess and monitor the concrete characteristics and the development of reinforcement corrosion parameters:

- Measurement of carbonation velocity in indoor and outdoor atmospheric conditions, using a number of cylindrical control specimens that can be measured (destructively) with respect to the depth of the carbonated front.
- Monitoring of concrete air permeability by creating a vacuum inside a device placed on the concrete surface and measuring the air flow after a certain time.
- Monitoring of corrosion parameters: corrosion potential and corrosion rate, of steel reinforcement in buried and atmospheric conditions.
- Characterisation of resistance to chloride transport of concrete and mortar. Chlorides are not in the environment but they are inside the drums as part of analytical wastes.

As another good example of an engineered vault type of a repository, the design of the Dounreay in Scotland is presented in Figure 7-6 to Figure 7-8. During operations, the function of structures in Figure 7-7 have the following functions:

- Waste packages: Allow simple waste handling, transportation and emplacement in vaults.
- Concrete vault: Structure within which to place waste packages. Keeps the waste packages dry during the operations.
- Temporary roof: keeps facilities dry during operations.
- Upper drainage systems: drain near surface groundwater away from the facility with gravity.
- Lower drainage systems: deeper groundwater pumped to upper drainage system.

After closure, the structures shown in Figure 7-7 have the following functions:

- Cement grout conditioning in waste packages: Reduces groundwater flow. Chemical conditions limit radionuclide migration.
- Cement grout backfill between packages and vault walls: Reduces groundwater flow. Chemical conditions limit radionuclide migration.
- Engineered gap: reduce upward groundwater flow and transport of radionuclides. Reduces risk of future human intrusion.
- Concrete vault: limit groundwater flow into waste.
- Bedrock: attenuates radionuclide migration.
- Permeable backfill: encourages groundwater to flow around the vault.
- Depth of facilities: Reduces risk of future intrusion. Repository is located deeper in the zone with less groundwater flow.

The site is owned by the UK Nuclear Decomissioning Authority (NDA) and the operation and closure is handled by a Cavendish Dounreay partnership (owned by AECOM, Cavendish Nuclear & Jacobs) through a company called Dounreay Site Licence Company (Usher & Rossiter, 2019). In Dounreay, low level waste is placed in metallic containers and deposited in large concrete vaults, located mostly below the ground surface (Usher & Rossiter, 2019). During the operations these vaults have temporary roofs for preventing water from entering the repository. During the closure phase, the repository is isolated from the surface with low-permeability engineered cap structures. The control of waters in the area is handled by the design (inclinations) and engineered drainage systems. The main barriers preventing the radionuclide dispersion into the environment consist of the metallic waste packages, cement grout between the packages, the concrete vault and the natural barriers at the site (Usher & Rossiter, 2019).

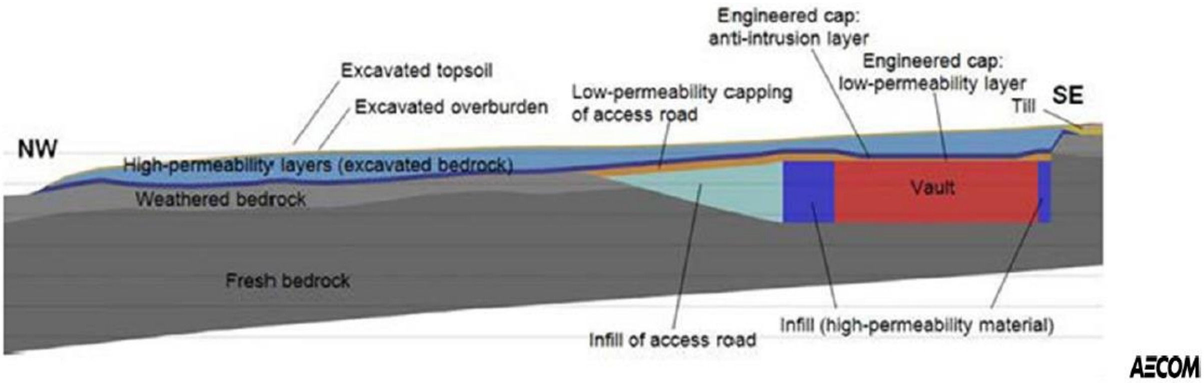

Figure 7-6. Overall principles of the design of the Dounreay repository (Usher & Rossiter, 2019).

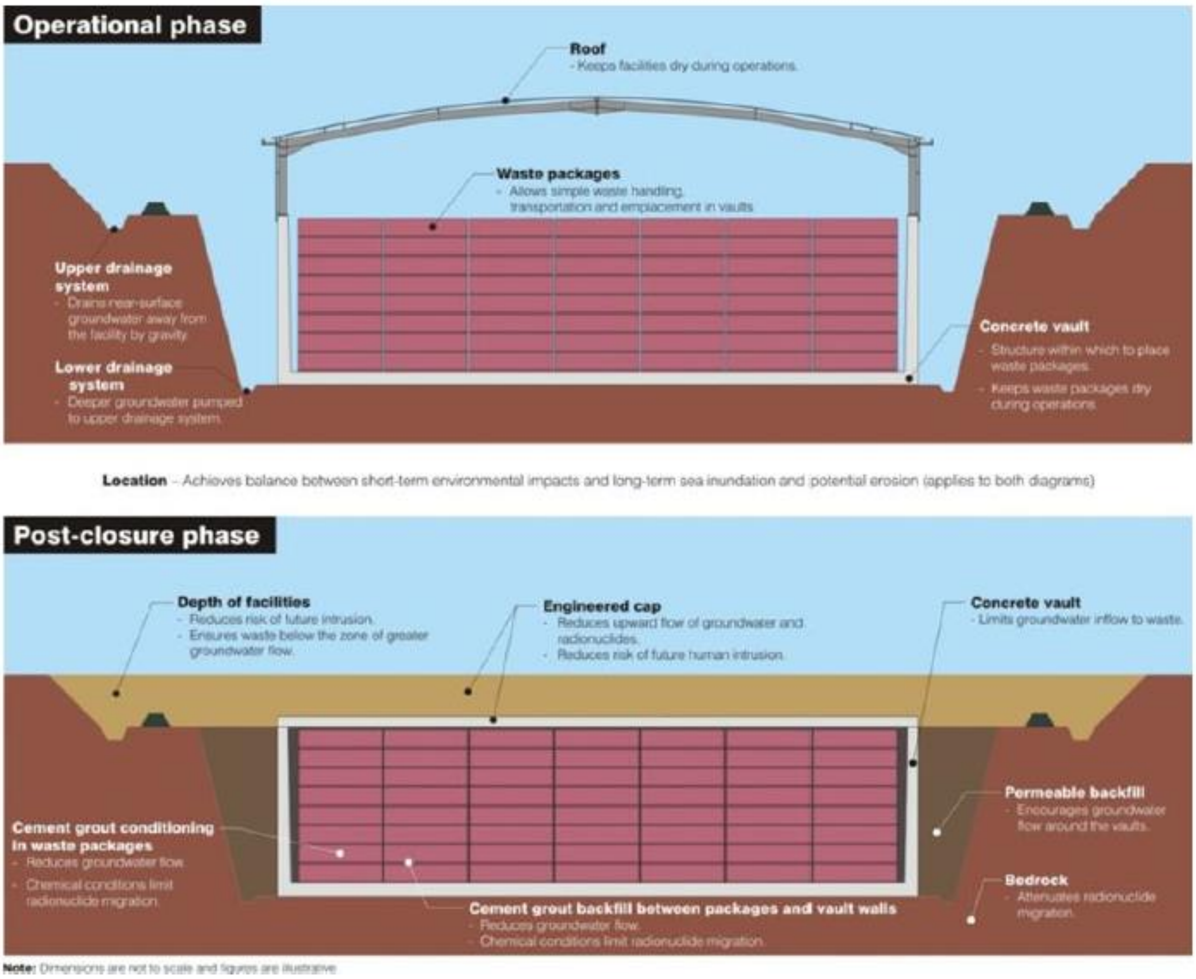

Figure 7-7. Engineered barriers in the Dounreay repository in Scotland (Usher & Rossiter 2019).

Figure 7-8. Overview of Dounreay vaults for LLW (Usher & Rossiter, 2019).

8. Multi barrier system

According to the IAEA (2011) "*In order to ensure that a disposal system is robust, a multiple barrier concept — which utilizes the properties of the waste form, engineered barriers and the site's natural barriers to prevent or restrict the release of the radionuclides from the facility is generally selected. The relative contributions of various barriers to the overall safety of a disposal facility will depend upon the characteristics of the waste, site conditions and the disposal concept, and will be time dependent*".

In practice this means that the type of the engineered barriers used in a near surface repository depend on the type of the waste (VLLW, LLW, ILW), the repository type selected (see section 7) and the characteristics of the site (see section 6). These barriers are described briefly in the following subchapters.

Considering Finnish near surface repository to be designed for VLLW a **graded approach** can be applied in selection of the engineered barriers used in the design. According to the IAEA (2012), the graded approach means the "*ability of a chosen disposal system to contain the waste and isolate it from humans and the accessible biosphere should be commensurate with the hazard potential of the waste*". In addition, the disposal system is expected to provide containment and isolation from the biosphere only for a limited time tied to the decline of radioactivity of the deposited waste (IAEA, 2012).

8.1 Natural barrier

The natural barrier system consists of the geological media hosting the repository and any other geological formations contributing to waste isolation. In safety assessments, the natural barrier system is often referred to as the far field or the geosphere. The geosphere comprises the vadose and the saturated zones (generally, above and below the water table). The geosphere protects the disposal facility, and retards and dilutes any radionuclides released from the near field. The natural barrier system is normally long lasting, although it may be affected by erosion, climate change, seismic events, and other processes and events (IAEA, 2002). **For the expected Finnish repository site conditions, see section 6**.

Considering the wider range of geological conditions around the world, a near surface repository can be built: a) on top of the ground/or partially within the vadose zone; b) within the vadose zone; or, c) in the saturated zone (below the groundwater table). Considering Finnish geological conditions, i.e., where the vadose zone is typically shallow and the groundwater table is usually located only few meters from the ground surface, it is more likely that the repository will be built entirely above the ground or only partly in the vadose zone (see section 6). This would also be the case in sites where there are essentially no sediments above the crystalline bedrock.

In other type of geological conditions (not typical in Finland), the **vadose zone** may show favourable features for the location of near surface repositories, such as allowing disposal unit designs that are intrinsically capable of minimizing contact between infiltrating water and deposited waste (IAEA, 2002). Ideally, for a shallow facility in the vadose zone the preferred host rock is one that has a low unsaturated moisture content and that provides effective drainage for water percolating around the facility, for example a sandy host medium. The dynamics of water movement through the vadose zone will depend on the permeability of soil layers, precipitation rate, extent of runoff and amount of evapotranspiration. When perched water layers are present, careful consideration needs to be given to protecting the disposal units from water inflow, not only from above but also laterally. This can be achieved by the construction of vertical capillary barriers, consisting of coarse grained walls surrounding the disposal units and underlain by a high permeability layer. This would prevent perched water from reaching the waste (IAEA, 2002).

For disposal in the **saturated zone**, candidate host media are generally low permeability materials in which radionuclides can be sorbed, resulting in limited radionuclide transport. Some examples include relatively unfractured clay, clay-rich till and mudstone (IAEA, 2002).

For disposal below the water table, groundwater moving towards the disposal units is likely to carry chemical species from the adjacent hydrogeological system. Mobilization of certain chemical species and their transport in natural waters may be enhanced by the presence of complexing agents or colloids, and by microbial activity (IAEA, 2002).

Repositories located in coastal regions may be subject to ingress of saline sea water into the disposal units (IAEA, 2002); however, this is typically not the case in Finland.

Considering various geological conditions the **geological features** of a candidate site on the development of an acceptable safety case and on the likelihood of successful licensing, information on the following is typically required (IAEA, 2002):

- Geological history.
- Stratigraphic, lithological, mineralogical and structural geological conditions of the region and the site, including the geometry and distribution of geological features.
- Recent evidence of active faulting, evidence of active tectonic processes, the occurrence of quaternary faults at the site and the age of recent movements, historical earthquakes, and an estimate of the maximum potential earthquake within the geological setting.
- Evidence of volcanism, history of volcanic activity near the site.
- Topography of the site, including actual drainage features, the location of existing and planned surface water bodies, and definition of areas containing poorly drained materials; data on the flood history of the region, upstream drainage areas, precipitation and the potential for extreme weather phenomena, such as hurricanes, tornadoes and severe winter storms.
- History of subsidence; records of past and present drilling and mining operations in the vicinity of the site, including groundwater extraction and use.

- Occurrences of energy and mineral resources, including groundwater, and estimates of their present and projected quality and value, and of their potential for use.

In general, groundwater flow is controlled by such features as geological structure, sediment texture, pore space, fractures and climate. Groundwater flow in any near surface geological environment is part of the hydrological cycle and is determined by the hydraulic conductivity, flow porosity and hydraulic gradients between higher head recharge areas and lower head discharge areas (IAEA, 2002).

Definition of groundwater flow in a **hydrogeological system** requires information on (IAEA, 2002):

- Distribution of hydraulic parameters, including hydraulic conductivity, porosity and storativity (storage coefficient);
- Spatial and temporal variation of the hydraulic head;
- Geometry of the flow domain;
- Recharge and discharge areas;
- Recharge and discharge rates, including infiltration, evapotranspiration, water balance and extraction volumes;
- Groundwater system boundaries, including rivers;
- Relationship between the different hydrogeological units;
- Flow velocity and residence time of groundwater in the system.

In summary, the **hydrogeochemical** characterization of a site represents most of the required geochemical information and may include the determination of the following parameters (IAEA, 2002):

- ”Master” variables: pH, Eh.
- Main components: Na, K, Ca, Mg, HCO_3, SO_4, Cl, Si and total dissolved solids (TDS), or the sum of the main components.
- Trace substances: Fe, Mn, U, Th, Ra, Al, Li, Cs, Sr, Ba, HS, I, Br, F, NO_3, NO_2, NH_4, HPO_4, rare earth elements, Cu, Zr.
- Dissolved gases: O_2, N_2, CO_2, CH_4, C_xH_x, H_2, Ar, He.
- Stable isotopes: ^{2}H in H_2O, ^{18}O in H_2O and SO_4, ^{13}C in dissolved inorganic carbon (DIC) and dissolved organic carbon (DOC), ^{34}S in SO_4 and HS, $^{86}Sr/^{87}Sr$, ^{3}He, ^{4}He, Xe isotopes, Kr isotopes.
- Radioactive isotopes: ^{3}H, ^{14}C in DIC and DOC, ^{36}Cl, $^{234}U/^{238}U$, ^{226}Ra, ^{222}Rn.
- Others: DOC, humic acids, fulvic acids, colloids, bacteria.
- Pore and fracture filling minerals: ^{18}O, ^{13}C, $^{86}Sr/^{87}Sr$, $^{235}U/^{238}U$, mineralogy, texture and sorption properties of deposited authigenic minerals.

In general, the information needed for assessing **radionuclide migration** in the far field, as discussed in the preceding sections, can be summarized as follows (IAEA, 2002):

- Groundwater flow, including advection and dispersion processes: groundwater flow rate and patterns, codes to model groundwater flow, groundwater flux and velocity, flow patterns and pathways, migration behaviour of radionuclides, geochemical codes.
- Fracture flow (preferential and faster pathways for radionuclide transport): nature, characteristics and distributions of fractures, water flow rates.

- Diffusion: characteristics of the host geological medium, radionuclide–host rock interactions, effective diffusion coefficients.
- Solubility: groundwater chemistry, Eh and pH, geochemical codes to calculate concentrations controlled by solubility limits.
- Complexation (both organic and inorganic): nature and type of complexes, groundwater chemistry, geochemical codes to model complex formation.
- Colloid formation: colloid characteristics, colloid–radionuclide interactions, groundwater flow rate and pattern (advective transport of radionuclides as colloids, enhanced migration).
- Chemical reactions in groundwater: Eh and pH of groundwater, chemical species in groundwater, chemical stability of precipitates that may result from interaction of these chemical species with radionuclides, rates of reactions.
- Sorption: sorption coefficients in site specific groundwater, mineralogy of sorbing phases, groundwater chemistry, geochemical codes to model sorption.
- Gas phase transport in the vadose zone: groundwater chemistry, degree of saturation in the vadose zone, partitioning of contaminant between gas and aqueous phase, geochemical codes to model partitioning and transport, groundwater flux.

8.2 Waste packages

Waste packages are referred to as the combination of the waste form and the container including eventual over-package and coatings (IAEA, 2001). The waste form and container provide for immobilization of radionuclides and mechanical and structural integrity and therefore, can be considered as part of the multiple barrier system contributing to the containment and isolation of the waste (IAEA, 2002, 2003). Mechanical and structural integrity are important for operational safety (Section 3.2) with regard to temporal storage, transport and handling of waste packages during normal operation and anticipated operational occurrences as well as under incident and accident conditions (IAEA, 2002, 2003). In terms of post-closure safety, the mechanical and structural integrity of waste packages contribute to the stability of disposal units and covers reducing the extent of cracks, subsidence and other mechanical loads (IAEA, 2002). In addition, waste package durability is essential for defining the source term and the importance of the waste package in relation to the other barriers of the disposal system and their assigned safety functions (IAEA, 2002). Besides standard testing of the waste package's properties related to mechanical and structural integrity (e.g., compressive strength), degrading chemical and physicochemical mechanisms need to be taken into consideration depending on the materials present in the waste form (concrete, bitumen or polymers) and container (e.g., steel, concrete or plastics) (IAEA, 2002). These processes include (IAEA, 2002):

- Concrete dissolution and development of cracking (cementitious waste forms, concrete containers);
- Corrosion (metal containers);
- Chemical attack by waste constituents and water transported species;
- Ageing and, in some cases, radiation effects (plastic containers).

Functions of different waste forms and containers, materials used and processes affecting their durability are further described in the next sub-sections.

8.2.1 Waste form

As pointed out previously, a typical conditioning step after volume reduction, for example by incineration of combustibles, is the confinement of radioactive wastes in a matrix to produce a waste form, which provides for immobilization of radionuclides and mechanical and structural integrity of the waste package by reducing the void space inside (IAEA, 2003). For some wastes, such as paper, wood, rubber or animal carcasses, there may be no conditioning but only a volume reduction by, for example, compaction (IAEA, 2002, 2017b). Conditioning materials comprise bitumen and polymers but the most commonly used is cement (IAEA, 2002, 2003). A cementitious waste form is favoured due its capacity to buffer the pH and thus, controlling the solubility limits and mobility of radionuclides, which is a determining factor for the source term used in safety assessments (Section 10). With respect to the temporal evolution of the source term, the degradation of cement controlled by the chemical composition of the infiltrating water and the effect of corroding agents needs to be taken into account. These coupled physicochemical processes lead to a progressive increase in porosity and permeability and developments of cracks and determine the mechanism and rate of radionuclide release over time (IAEA, 2002). For example, alterations in the diffusion controlled release mechanism typically found for many radionuclides immobilized in a cementitious matrix may occur due to the degradation of the waste matrix. The stability of bitumen, on the other hand, does not depend that much on the interaction with water but is rather affected by oxidation (IAEA, 2002). Another example is the dissolution controlled release of radionuclides present as contamination on metal or other surfaces (IAEA, 2002). The release and migration of radionuclides is further affected by sorption processes in the near field and the presence of competing complexing agents and the possibility for colloid facilitated transport. Regarding the latter, the scientific understanding is limited and the correct representation of colloid related processes in the modelling prepared for the safety assessments remains challenging (IAEA, 2002). Likewise, the role of microbes is not completely understood at present and their influence can be beneficial or adverse depending on the local conditions (IAEA, 2002).

In summary, the release and migration of radionuclides is conditioned by the characteristics of the waste form and the hydrogeochemical environment (e.g., pH, presence of gaseous oxygen) and a thorough understanding of the underlying processes is necessary for modelling the source term evolution. This also requires testing the waste forms under repository conditions during both the operational and the post-closure phase including, for example, immersion and freeze-thaw cycles (IAEA, 2002). In addition, the waste form cannot be considered separately but must be understood as part of the overall disposal system including the container/packaging and the surrounding engineered and natural barriers with mutual influences and interdependencies.

8.2.2 Containers

Waste containers provide mechanical and structural stability of the waste packages and, by containing the waste as long as their integrity is maintained, delay the release of radionuclides, which is particularly beneficial considering the decay of short-lived radionuclides to acceptable levels before leaving the near field (IAEA, 2002). Containers are usually fabricated from carbon steel or concrete. High integrity containers (HICs) are made from high density polyethylene (HDPE), stainless steel, metallic fibre reinforced or polymer impregnated concrete. In contrast to metallic containers, the advantage of plastic containers (e.g., HDPE) is that they are not susceptible to corrosion. However, radiation effects need to be taken into consideration in the degradation behaviour. Concrete over-packs are often used in the case of plastic containers to increase the mechanical stability of the waste package and provide for another containment barrier (IAEA, 2002). Metallic and concrete containers are further described in the following sub-sections.

8.2.2.1 Metallic containers

Introduction

Common method to manage metallic Low and Intermediate Level Waste (LLW/ILW) is to initially deposit it within metallic container made of steel and consequently encapsulate it into grout. Metallic container will provide a simple practical way to prevent contamination of the surrounding environment by waste and it will be a radiation shield in emplacement process (Larsson 1989). The advantages of the metallic barrels include also good mechanical properties and stability but the main disadvantage is the risk of corrosion. Corrosion forms may be general corrosion or localized corrosion like crevice corrosion, pitting or galvanic corrosion.

Corrosion on metal barriers is affected by several factors. Aggressive components to induce corrosion can be originated either from the waste or from the environment. Also pH, redox and temperature of the environment are important factors affecting corrosion. Besides microbes can enhance corrosion and change environment to more aggressive direction, e.g., sulfate reducing bacteria is known to accelerate corrosion of carbon steel in ground water in bedrock.

In Finland the low level waste is packed into standard 200-L drums. The compressible waste is compressed to half of its volume using a hydraulic press before to be packed in concrete containers. (Posiva Oy 2016) The type of the container for VLLW to be deposited in a near surface repository has not been defined yet by any Finnish operators. As an example, in the UK case (see section 7.5 Engineered concrete vault repositories), the container resembles a half-size sea container.

The near-surface repositories are subjected to short-term variations in environment, such as temperature change (particularly freezing time in case of Finland), and wet-dry periods with episodic flooding.

Design basics

In selecting materials for the containers and determining details of the container design, the extent and acceptability of both internal and external corrosion throughout the interim storage and disposal periods should be considered. The chosen materials should be corrosion resistant or have acceptable level of degradation or corrosion in the environment. The degradation of containers is a complex phenomenon which is correlated to mixed matrixes; the degree of degradation is related to combination of waste container design, waste form and backfill.

Carbon steel is commonly used as a construction material for radioactive waste containers with volume of 200 and 400 L drums in similarly light gauge metal, generally welded or folded shell construction. To improve the mechanical strength of a container, ring bands or ring flanges can be added to the structures. Many materials that may be present either in the waste or in the repository environment can cause corrosion to carbon steel and therefore some additional internal or external protection of metal containers is needed. The corrosion resistance of carbon steel containers can be improved by coating, for example epoxy resin, zinc phosphate, zinc chromate, enamel glaze, silicone resin, bitumen and coal tar would provide protection inner or outer surfaces of the container (Hauser and Koster 1989).

Stainless steels have generally high resistance to corrosion, but they can suffer from localized corrosion such as pitting, in particularly chloride containing environments. Corrosion of weld seams can be avoided by adequate welding and after-treatment procedures. In addition, hydrogen generation in the vicinity of the container walls should be considered to avoid scenario of gas diffusion into the metal with consequent embrittlement.

A big challenge is, that the environment during the decommissioning is not known. The conditions of the repository site should be addressed as a most challenging issues (elevated temperatures, high humidity, high water salinity, low pH). The experimental condition to simulate the real condition would provide valuable information and data for long term performance of waste packages at the similar manner.

At each step of the waste management from the interim storage of waste to transportation, the initial operational phase of the repository and final deposit, the container must maintain its reliability for several decades. To verify this, detailed experimental studies of metallic container material are required to determine corrosion resistance and corrosion rate. The study should take into account all possible failure modes including general corrosion, stress corrosion cracking, localized corrosion as well as reactions that can change material properties and lead to early damages.

Corrosion of metallic containers

Corrosion resistance is the main concern regarding to metallic containers and durability of the container. Both external and internal corrosion must be considered and all potential corrosion elements such as composition and amount of organic substances, other biodegradable materials and metals shall be identified with experimental study.

Generally, both the internal environment (waste and their impurities) and the external environment (repository condition including soil, microbial activity and climate condition) will influence the corrosion of the containers. In addition, metal character must be taken into account. Different metals may react differently under similar environmental conditions.

Metallic container is an internal container or primary waste package and that is why it is required to use fill material/backfill (unreinforced concrete or mineral materials) to fill the space between waste package and surrounding ground. In some cases, also the free voids inside the package could be filled with a fill material/backfill material. The interaction between metallic container and any fill material has to be taken into account when assessing the evolution of the repository. If grout is used, any possible failure of cementitious materials may affect corrosion of the metallic barrier.

Container design

Corrosion of metallic containers can be influenced by container design and by the details of the manufacturing process. Factors to be taken into account include:

- Crevices can lead to crevice corrosion. The risk of crevices can be induced between the container components, between the waste form and the container wall, or between the containers when stacked, and also in the area of the container lid.
- Surface finish and degree of cleanliness can affect the corrosion performance of the container. For example, a higher degree of smoothness on a surface will reduce the risk of corrosion.
- Drum manufacturing procedures reduce the material thickness resulting a higher risk of corrosion in particular areas. Especially welding and cold working may affect the corrosion performance of the steel.
- Internal voids, ullage or space between the waste matrix and the container wall can induce the risk of corrosion, for example access to higher amount of oxygen.

Effect of Waste type on containers

It is important to identify composition of the waste and its influence on the corrosion of the metallic container. Aggressive ions, particularly chlorides and sulfates can cause localized corrosion to steel. The waste can induce changes in pH and consequently, there is higher risk of general corrosion attack. If there are organic components in the waste there is a risk of microbial activity that may accelerate the corrosion of steel.

The phenomena of microbial degradation of cellulose and hemicellulose will generate gas under final repository conditions. Microbially mediated LLW degradation and gas generation processes can impact the performance of multi-barrier systems via rising corrosion and consequently transferring of radionuclides from the repository (Vikman, et al. 2019).

The weathering of concrete, changes in oxygen content and the flow of water into the repository and reaching to waste packages will expose the waste materials to different corrosion mechanisms, i. e. general corrosion, pitting, crevice corrosion and stress corrosion cracking.

Service environment

Container material should be compatible with the operating environment, both in terms of the corrosion performance and the possible mechanical loads. The main elements affecting to corrosion are the presence of water and the water chemistry, including pH, oxidation potential Eh and the concentration of dissolved salts. In addition, the diffusion of chloride ions from water may damage the passivity. Consequently, the diminished passivity will significantly increase the corrosion rate. This means that both degradation of the repository structures and the rate of release of radionuclides will increase (Eckerberg and Olsson 2013).

Researchers in the literature suggested that various factors including soil resistivity, level of dissolved salts, moisture content, pH, oxygen concentration and the presence of microbes corresponded to corrosion of the ferrous metals (like carbon steel and stainless steel) in soils. Relevantly, soil moisture content is the only factor consistently found to control the corrosion of the ferrous metals (Cole and Marney 2012).

The influence of the operating environment can be summarised as the interacting effects of selected factors:

a. pH is the key factor for corrosion resistance, since a high pH brings iron and steel in a passive state, resulting in a low corrosion rate. The pH around the steel in contact with cementitious materials such as concrete and embedment grout is about 12.5, resulting a passive state. High pH environment is considered to be desirable for inhibiting corrosion. Corrosion rate of steels and the migration of compounds are typically very low and insignificant in concrete environment due to the high pH. On the other hand, high pH can be detrimental for bentonite based barrier materials (Balmer et al., 2017) and this should be taken into account in the repository design.

b. A high chloride or sulfate content in ambient groundwater can lead to conditions that accelerate corrosion of carbon steel. Presence of chlorides enhance pitting and stress corrosion (the latter mainly at temperatures above 60 °C) for stainless steel. The chloride content should be kept at minimum and if it exceeds 100 mg/L possibilities of all corrosion mechanisms should be considered carefully. Also, consideration should be given to the producing of chloride ions, for example, in case of the chlorine-containing plastics radiolysis.

c. Oxygen concentration and presence of moisture. Water and oxygen are the primary elements which induce corrosion of metallic barrier. In the soil above the groundwater, water is stored by capillaries and pores. The finer soil particles and pore size will hold more water (e.g., clay). Sodium sulfate and sodium chloride concentrations are significantly changed by evaporation of water. Lower redox potentials and higher concentrations of carbon dioxide are result of lower percolation rate and higher accumulation of water. The water content, also the oxygen and carbon dioxide concentration are major corrosion-determining factors. An increase in water content of soil is the main factor in the corrosion of metallic barriers (Ismail and El-Shamy 2009). Typically, diffusion of the dissolved oxygen in the soil surrounding water is a driving force to control corrosion of the metals. Nonetheless, in the absence of water or low concentration of oxygen, other factors may become important for corrosion of metallic barrier, like micro-organism.

d. Microorganisms. Soil as well as surface water besides ground water contains diversity of microbes. Harmful microbes and/or their metabolic by-products are able to induce changes in chemical or physical conditions, giving rise to an aggressive environment. Microbiological activity can induce corrosion of metals as well as contribute to the integrity of concrete. Microbes can attach to the surfaces and can together with exopolysaccharides and other organic material form biofilms. Under the biofilm the amount of dissolved oxygen, inorganic and organic compounds as well as the pH can be totally different than in the surrounding liquid. In anaerobic conditions especially sulfate reducing bacteria (SRB) are known to cause microbially induced corrosion (MIC). Most often MIC appears as localized corrosion, but in some cases it can proceed more generally, as general biological corrosion. The metabolites produced by microbes can significantly reduce the pH of the environment which in turn affects the durability of concrete. Presence of concrete inhibits or diminishes the formation of a biofilm on the carbon steel surface and thereby lowers the corrosion of carbon steel. In biotic environment without concrete corrosion rates of carbon steel can be almost 500 times higher than that of concrete environment. However, in presence of concrete the corrosion tends to be more localised (Carpen, et al. 2015).

e. Temperature: Another considerable objective of locating the repository at near surface particularly in Finland is freezing during winter time. In waste packages holding large amount of waste water or water bound in grout, freezing may lead to expansion and thus high stresses in the containers. The effect of freezing is correlated to duration of freezing time and the depth of the located repository and it is indeed depended on the other barriers around the waste. For example, metals in the waste are not probable to be directly affected by freezing, while grout and concrete waste containers could burst due to freezing in the same way as the concrete barriers. Metallic container are not at the risk of degradation as long as the temperature is higher than the freezing degree for the concrete pore water. The effect of the temperature on the properties of the concrete barriers is negligible at such a condition. However, metal containers could break due to freezing if the contents expand to a sufficient degree (Eckerberg and Olsson 2013). However, with proper design of the cover layer of the repository, the waste packages or backfill surrounding the waste packages should not be exposed to minus degrees. The risk for freezing is however, to be taken into account during the operations before the cover structures are not yet at place.

Corrosion testing

The most important factors to be considered when planning an experimental program to study container corrosion:

a. Corrosive media. Storage and disposal site condition, the chemical composition of any solution in interaction with the container, the temperature and the humidity, must be simulated to obtain data on the corrosion of the outside of the container. Furthermore, waste forms in contact with the container need to be simulated to achieve information in term of the inside corrosion of the container.

b. Canister material and constructional details. In any experiments the typical container materials with possible coatings and constructional details, e.g., welds, crevices and roundings, must be applied. This can be done by using special samples that simulate such details or by doing experiments on the big scale container. It is noteworthy to take into account any possible damage to the container during handling, for example scratches in the coatings.

c. Exposure time. Effect of time on corrosion of the material can be evaluated by the immersion tests at different periods of time (e.g., 3, 6, or 12 months or longer). Obtained data can be extrapolated for longer periods but only within certain boundary conditions.

d. Methods of examination. Common methods for to examine test samples to discover the nature of the corrosion processes are gravimetric determination of mass loss, detection of surface profiles, metallographic micrographs and scanning electron microscopy (SEM).

e. Types of corrosion. Tests should be accomplished to reveal the common corrosion types i.e. general corrosion, pitting corrosion, crevice corrosion, galvanic corrosion, selective corrosion and for stainless steels also stress corrosion cracking.

The operational phase of the repository is also another factor that needs to be thought over. The container will need to endure the high pressure during backfilling process.

Internal protection would be a potential alternative to prevent galvanic corrosion or being insurance against the harsh products of radiolysis or chemical reactions from the waste composition. The demand of the internal protection is more vital in term of long-term storage.

In the case of a near surface waste repository issues of chemical reaction e.g. gas generation from aerobic and anaerobic corrosion of metals and metallic compounds, radiolysis and microbial activities has to be considered.

Low level waste packages could be placed outdoors during interim storage. In this case the containers should have an appropriate corrosion resistance to the site, and the degradation of the container should not cause problems for later disposal. This condition applies equally during the early stages of operation of shallow land burial sites before the structure is covered.

8.3 Engineered barriers

Engineered barriers, in particular the final cover, are emplaced to ensure the integrity of the repository, to minimize the ingress of infiltrating water to the waste, thereby limiting radionuclide releases, and to reduce the likelihood of disturbance by human activities (IAEA, 2002).

The engineered barrier system may consist of a number of separate components, including structural walls, buffer or backfill materials placed around the waste packages, chemical additives, liners and covers (IAEA, 2002).

- Below the water table, the disposal units can be lined with clay, concrete, bitumen or other materials to improve the isolation of the waste; above the water table, the same materials could be used to produce impermeable covers to prevent or minimize the ingress of percolating water into the disposal units.
- The space between the waste packages may be backfilled (for example with cementitious grout) to provide structural support for the waste packages and to reduce space for infiltrating water (bath tubbing).
- In some designs, wastes are combined with protective materials in monolithic blocks in special overpacks to facilitate their retrieval (for example at El Cabril in Spain).
- Capillary barriers (consisting of a coarse grained material, for example gravel, that has a higher permeability than the surrounding finer grained materials) may be used to limit the ingress of water into the disposal units.

- Weatherproof buildings (for example those at Centre de l'Aube in France), and water diversion and drainage systems can be constructed to direct water away from the disposal units.
- The disposal units can be protected from intrusion by a layer of rock (in rock cavity type repositories) or by capping (as in the case of the reinforced concrete roofs in the IRUS facility design in Canada), and from erosion by the planting of vegetation or the use of a rock rubble cover.

The disposal unit comprising the engineered structures/isolation layers (concrete, porous medium for drainage, bitumen, polymers, clay), and lining and backfilling materials (concrete, fly ash, clay mixtures) (IAEA, 2002).

8.3.1 Concrete structures and backfills

Concrete is an extremely versatile material, with widely varying characteristics that make it possible to be used as a structural material, and as a backfill.

8.3.1.1 Generic consideration for cementious backfill design

A cementitious backfill can offer long-term containment by provision of a highly alkaline buffer capacity and good sorption capacity. The versatility of cement mixes is another important benefit, as they can be designed to satisfy a range of engineering and strength requirements.

Concrete provides a high pH environment, which limits the mobilisation and transport of certain long-lived radionuclides as a result of reduced solubility under alkaline conditions. The important property of the cementitious backfill material is typically the chemical buffering property rather than the physical integrity of the cementitious engineered barrier (waste form, structural components, etc.). Conversely, high pH is not optimal in repositories relying on bentonite-based barrier materials (Balmer et al., 2017), as high pH in the porewater can lead to dissolution of smectite minerals. This is a matter to be taken into account in selecting barrier materials for the design.

Depending on the type of a near surface repository concrete can be used in:

- Foundation slab for a landfill type of a repository. The advantage of this would be good bearing capacity and no risk of frost heave in comparison to mineral foundation materials.
- Fill material between the waste packages (in a landfill type of a repository). The advantage of this would be in small settlements in time, favourable chemical conditions to shelter metallic waste packages from corrosion and easy installation using a shotcreting method.
- Structural material in a vault type of a repository.

An example of an outline specification to meet the required physicochemical and engineering properties of a practicable backfill system is given (Francis et al., 1997).

The construction of repository concrete (structural) elements should take into account the duration of the construction phase the mechanical and environmental loading that these elements are subject to prior to entering operational phase. For example, a foundation slab might be exposed to the environment for many months and therefore might be subject to freeze-thaw loading. Design consideration should take into account all the phases (construction, pre-operational, operational, and final/closure) and the particular loading conditions that might affect the performance of the concrete. When the repository has all cover layers at place (with adequate ground frost protection), the concrete will not be exposed to minus temperatures.

For such a case, to produce freeze-thaw resistant concrete, it should be air entrained (introduce many small air bubbles, sizes less than 0.3 mm, approximately 4–5% volume of air). This is common practice in countries that are subject to harsh winter environments.

Essential requirements for vault backfill:

- In the long term the pore water must be maintained at a high pH (e.g., pH 10.5 or greater) to chemical retention of radionuclides in the repository near field.
- Appropriate compressive strengths are required to provide adequate support for the placement of successive waste package and backfilling layers.
- A cube strength limit at any age up to 50 years to assist grout removal should there be a future need to retrieve backfilled waste.
- The workability must be suitable for flow without: vibration into a horizontal space 5m x 3m x75 mm high, which may be a typical under-package space.
- The mix must be suitable for pumping along a horizontal pipeline at most 250 m in length.
- Bleeding/settlement must not exceed 2% to reduce the possibility of under-package void formation and surfaces of weakness in the backfill.

Desirable qualities for vault backfill:

- Should act as a chemical barrier to migration of long-lived radionuclides by providing high pH to inhibit solubility, and good sorption capacity.
- Sufficiently permeable to promote homogeneous aqueous chemistry and to enable gas transport.
- Should inhibit corrosion of steel packages.
- Relatively low heat of hydration.
- The use of cement additives, which might compromise the cement performance as a chemical barrier or the engineering properties, should be avoided.
- Mineral composition should be sufficiently durable to provide long-term chemical conditioning of repository pore water.
- Should possess well-understood mineral characteristics, which can be predicted during the repository evolution.
- Uses materials, which can be reasonably assured in terms of quality and quantity, during repository operational period.
- Suitable for placement by remote methods in waste vaults.
- Self -levelling, compacting, and able to provide a firm level base for placement of further packages.
- Suitable for easy excavation to allow retrieval of waste packages if that was required.
- Relatively inexpensive to produce.

Other limitations for mix development were imposed:

- *Organic additives*: Frequently used in grouting work; to improve fluidity or cohesiveness; to ensure void filling and to enhance working time by retarding setting. Some preliminary research (Hakanen & Ervanne 2006) had shown however that it would be preferable to avoid some additives of this type as they could have deleterious effects on the solubilities and sorption behaviour of some radionuclides.

- *Minimum cement content:* To achieve an acceptable level of alkalinity, the backfill should contain a minimum of 400 kg/m^3 Portland cement (PC), or an equivalently alkaline material.
- *Supplementary cementing materials:* Since these could have an influence on the effectiveness of the PC, in particular its ability to maintain high pH, a limit should be placed on maximum content. This limit was set at 25% by weight of total cementitious content for pulverised fuel ash (PFA). No limit was set for ground granulated blast furnace slag (GGBS).

8.3.1.2 Concrete structures and ageing management

The deterioration of concrete structures is a major concern, both structurally and from a maintenance management perspective. Damage induced by deterioration processes can dramatically reduce the designed service life of the structure. Failure to manage the maintenance of reinforced concrete may result in structural failure, and structural replacement is not a possibility.

The design of a new repository structures should be in accordance with the design codes for reinforced concrete structures. These typically do not cover the service life duration required in repository conditions. The structural designer must make the necessary design assumptions (changes to the ULS and SLS) so that the service life can be achieved. The structural designer must be able to prove through calculations and testing that the service life requirements can be fulfilled with the selected materials and structural measures for the environmental conditions where the repository will be during its planned operating life. The design repository structures should include comprehensive quality control of the concrete used during the construction phase needed for reassessment of the design service life, and monitoring of structures during the use of the repository.

Design must provide the basis for an ageing management system. Such a system is developed for securing the safety, performance and uninterrupted use of the concrete structures during a prolonged service life. The management of service life is extended with a range of analysis tools and services related to monitoring, simulation, condition assessment and structural analyses.

Currently there are no guidance for the ageing management of repositories, contrasting with what has been prepared for nuclear power plants concrete infrastructures in compliance with relevant IAEA safety standards and that draws on lessons learned from ageing management practices worldwide (IAEA, 2017c). This guidance provides an overview of the topic and guidance on proactive ageing management within NPPs. It collates information on ageing mechanisms, effects on structures, systems and components, the regulatory framework as well as some details on innovative techniques and research and development in the area.

Such guidance is needed to support repository staff, maintenance managers, vendors, personnel at research organizations and regulators in their work related to the ageing of concrete structures.

The following are examples of the use of concrete in structural elements in repository designs:

- A typical vault consists of a reinforced concrete basemat/foundation with a thickness of several tens of centimetres, reinforced concrete walls and a roof. The roof can be directly concreted over the waste packages when the structure is filled, or alternatively can consist of reinforced concrete slabs put in place and jointed with cement and/or bitumen.

The vault can also be designed as a large concrete box where the bottom and the walls are sufficiently linked to result in a monolithic structure (IAEA, 2002).

- At El Cabril, concrete vaults are used for the disposal of waste. The bottom plate is the main element of the vault. It is 0.6 m thick at the edges and 0.5 m thick in the centre, and is covered with a waterproof layer of polyurethane and a 10–20 cm layer of porous concrete (IAEA, 2002).
- Irus, Canada, after the unit is filled with waste and backfill (sand and clinoptilolite), the one metre thick, reinforced-concrete roof slab will be poured in place and will be given a waterproof coating (IAEA, 2001).

8.3.2 Mineral sealing layers and barrier materials

Mineral sealing/barrier materials can be used in a near surface repository design:

- In cover structures for limiting infiltration of water into repository (mineral sealing layers). These type of layers are usually in situ compacted to gain higher density and better sealing capacity.
- As a filling/backfill materials in side or around the waste packages to provide stability, sorption capacity and depending on the grain size distribution work as drainage material or as a sealing material limiting advection through the material. It should be noted that the installation method for the fill material does not necessarily allow use of any compaction technique.
- Below the waste (in subsoil) as a mineral sealing/barrier material limiting the advection of leachate water and radionuclide transportation into the environment. These layers are also typically compacted in situ.

If the desired function is to limit the hydraulic conductivity either in the cover structure or below the waste and transport of radionuclides, the hydraulic conductivity of the layer shall be low and the material shall have sufficient surface area to promote sorption of radionuclides. In some cases, the mineral sealing/barrier material placed below the waste can consist of the natural sediment layer prevailing on the site (e.g., glacial till). However, in most of the cases the material can be manufactured e.g. by mixing of bentonite and some suitable aggregate material (sand, crushed rock or rock flour). The specific design for the mineral sealing layers (thickness, composition, installed density) depend on the requirements set for the repository, for example the maximum allowed infiltration (X L/m^2/a or X% of the annual precipitation). The mineral sealing layer can be combined with a bentonite mat or geotextile to provide sufficiently low hydraulic conductivity and low infiltration through the layer.

There are currently no specific requirements for the mineral sealing layers/barriers for a near surface repository (e.g., the maximum infiltration of rain water into the repository) and these should be checked when the STUK guidelines specific for near surface repository will be available in future. As an example, the following section discusses first the Finnish guidelines and design basis for mineral sealing layers in landfills for hazardous waste. It should be noted that these guidelines may be different from what will be required for a near surface repository in Finland. After the following section, bentonite-based and alternative mineral sealing/barrier materials are discussed in detail.

Design basis for mineral sealing layers in Finnish hazardous waste landfills

In the Finnish legislation on landfills, the subsoil of the landfill for hazardous waste must fill the hydraulic conductivity requirement of $k \leq 1.0 \times 10^{-9}$ m/s with the thickness of ≥5 m. If these requirements are not met, a sealing layer must be constructed below the waste to meet the targets. The thickness of the mineral sealing layer in the landfill of hazardous waste should be at least 1.0 m. To collect the water, a synthetic membrane and a drainage layer with a thickness of 0.5 m must be installed or constructed on top of the sealing layer. In the cover of the landfill for hazardous waste, the following layers are required: surface layer ≥1 m, drainage layer ≥0.5 m, sealing layer ≥0.5 m, and a synthetic membrane. If needed, also a gas collection layer will be constructed (VNp 331/2013). For the sealing layer in the cover of the landfill, no numerical requirements for hydraulic conductivity have been set but these requirements should correspond to the requirements set for the sealing layer in the bottom liner. Thus, it is recommended that the minimum thickness of the mineral sealing layer is 0.5 m and the recommended permeability is $k \leq 1*10^{-9}$ m/s (SYKE, 2002). The gas collection layer is made from coarse well-sorted soil or from geosynthetic material and it is connected to the gas collection network. The recommended thickness of the gas collection layer made from soil is 0.3 m (SYKE, 2008). For the drainage layer, the minimum thickness is 0.5 m (VnP 331/2013). The recommended water infiltration capacity is $k > 1*10^{-3}$ m/s and the minimum inclination is 5% (SYKE, 2008).

Considering normal or hazardous waste landfills, functions of the mineral sealing layers are to minimize filtration of detrimental elements and to minimize diffusion. Several properties affect the retention of detrimental elements. These properties include the clay mineral types, ion exchange properties, specific surface, and prevailing conditions (pH, minerals, composition and concentration of leachate water) (SYKE, 2002.).

The following issues need to be taken into account when designing the mineral sealing layer for landfills (SYKE, 2002):

- Meeting the permeability targets with proper material selection, thickness of the compacted layer, and compaction methods
- Constructability
- Effect of the compactness on the water permeability
- Risks for cracking, tolerance for settlements depending on the plasticity properties of the material
- Swelling and shrinking due to moisture conditions
- Chemical durability and behaviour in relation to the filtration water
- Erosion resistance
- Shear strength and its effect on slope stability and designed loading conditions
- Frost resistance and needs for frost protection taking into account the construction schedule, protective impacts of layers above, and phasing of the construction
- Suitability to act as a supporting and protective layer of the synthetic membrane.

Materials that are used in the sealing layer in landfills are mostly mineral soils. If there is need to decrease the permeability of the layer, additional materials can be added. These additional materials are usually bentonites. Suitability of the used materials should be tested by laboratory and field tests (SYKE, 2002).

If clay is used as the sealing layer material in a landfill, uneven settlements might cause bending and tension in the sealing layer which can lead to cracking. Also drying can lead to cracking in clays. Higher water content decreases water permeability but it can also increase shrinkage if clay dries. In all natural soil materials, freezing leads to increase in porosity, drying and cracking, which all result in increased water permeability. During melting of the frozen structures, risks for lower bearing capacity and slides increase. One more factor to take into account with the natural soils is internal erosion in which flowing water transports fines away from the material (SYKE, 2008).

Soil-bentonite mixtures are manufactured by mixing powdery bentonite with soil material. The mixture is compacted as the natural soil sealing layers. The needed amount of bentonite depends on the granularity of the aggregate and the compaction conditions. The sealing layer manufactured using the mixture should be compacted to sufficient dry density. An adequate degree of compaction for dimensioning is 90–92%. The mixture should be manufactured so that with this degree of compaction, the hydraulic conductivity is $k \leq 10^{-10}$ m/s. Soil-bentonite mixtures are quite resistant to freezing and thawing, and they are also more resistant to drying than the natural soils. Ion exchange in the bentonite and internal erosion are risks for the long-term performance of soil-bentonite mixtures. With the ion exchange from sodium to calcium or magnesium, the swelling capacity of the bentonite can be only one-third of the original swelling capacity (SYKE, 2008).

The mineral sealing layer is constructed in multiple different layers (normally approximately 250 mm) that are compacted. The requirement for the compactness of the mineral sealing layers is determined based on laboratory tests, usually using the maximum dry density determined by the standard Proctor test. The requirement for compactness is at least 95% of the maximum dry density achieved using the standard Proctor test. The compacted layer should be protected immediately due to effects of drying, erosion, and freezing (SYKE, 2002).

The material type, layer thickness, and compactness affect the bearing capacity of the mineral sealing layer. Requirements for the bearing capacity of mineral sealing layer depend on the used material, experiences, laboratory tests, and compaction tests. Uniformity of the compaction is important. The bearing capacity of the sealing layer is normally tested by measuring the degree of compaction (SYKE, 2002).

Bentonite-aggregate mixtures as mineral sealing/barrier materials

The material properties of mixtures consisting of bentonite and aggregate depend on: a) the proportion of the bentonite in the mixture; b) the properties and type of the aggregate (sand, glacial till, crushed rock and/or rock flour); c) the dry density where the material is installed; and, to some extent, d) the mineralogy (smectite content, main cation Na^+ or Ca^{2+}) of the bentonite used in the mixture.

For example, the hydraulic conductivity of different bentonite-aggregate mixtures with varying amount of bentonite (0–30%) are given in Table 8-1. The hydraulic conductivity of the material decreases with increasing amount of bentonite and for a material with 0% bentonite is ~1×10^{-8} m/s and for a material with 30% bentonite ~1×10^{-11}…10^{-12} m/s. The hydraulic conductivity is affected by the type of the water used in the test and can be affected by the substances in the leachate water. The Proct (%) in Table 8-1 refers to the density state of the sample that is X% from the maximum Proctor dry density (kg/m^3) gained for the material in standard or modified Proctor compaction test (Craig, 2005). This test is used for studying the optimum water content for compaction of the material *in situ* and for studying what is the expected dry density of the compacted layer at the site. In general, the higher the amount of aggregate is in the mixture, the higher is the achieved dry density (>90% Proctor maximum). For materials with higher proportion of clay fraction, the achieved dry density is lower (e.g., 70% from Proctor maximum). The amount of clay may also affect the type of the compactor to be used at installation.

Table 8-1. Summary of the hydraulic conductivity tests for soil-bentonite mixtures (Johannesson et al., 1999).

				Final properties						
Test No.	Clay cont. (%)	Water	w_{ini} (%)	Proct (%)	w (%)	ρ_d (t/m^3)	e	Sr (%)	K (m/s)	K with Back Pr. (m/s)
1	0	Dist.	5.6	96	8.6	2.21	0.24	97	1.40E-08	1.96E-08
2	0	Dist.	5.6	91	10.4	2.11	0.31	95	6.19E-08	1.33E-07
3	0	Dist.	5.6	101	5.7	2.33	0.19	90	1.66E-09	4.55E-09
4*	0	Dist.	5.6	94	8.3	2.18	0.26	88	8.50E-08	-
5*	0	Dist.	5.6	92	7.8	2.12	0.30	73	1.00E-07	-
6	10	Dist.	7.0	94	12.4	2.02	0.37	96	2.87E-10	3.40E-10
7	10	Äspö	7.0	94	12.1	2.04	0.36	96	5.12E-09	1.13E-08
8	10	Dist.	7.0	94	13.4	2.02	0.36	102	2.79E-10	5.62E-10
9	10	Äspö	7.0	94	12.4	2.04	0.35	99	2.67E-09	5.59E-09
10	10	Dist.	7.0	88	16.0	1.90	0.45	98	2.02E-10	1.34E-10
11	10	Äspö	7.0	90	14.1	1.95	0.42	95	4.26E-09	2.70E-08
12*	10	Äspö	7.0	98	10.7	2.11	0.30	97	1.77E-09	-
13	20	Dist.	8.0	93	15.5	1.91	0.45	97	4.62E-11	5.22E-11
14*	20	Äspö	8.0	94	15.2	1.93	0.43	98	4.14E-10	9.46E-10
15	20	Dist.	8.0	87	18.1	1.79	0.57	93	5.53E-11	6.25E-11
16*	20	Äspö	8.0	90	17.1	1.85	0.50	96	3.27E-09	5.51E-09
17*	20	Äspö	8.0	87	18.8	1.78	0.54	95	1.46E-09	2.17E-09
18	20	Äspö	8.0	79	23.1	1.62	0.69	91	5.36E-08	-
19	30	Äspö	13.0	91	19.7	1.77	0.56	98	4.71E-11	7.84E-11
20	30	Äspö	13.0	89	21.5	1.72	0.61	99	4.25E-10	4.49E-10
21*	30	Äspö	13.0	89	19.6	1.72	0.60	91	9.73E-11	2.01E-10
22	30	Äspö	13.0	88	21.5	1.71	0.61	96	6.32E-10	9.09E-10
23	30	Äspö	13.0	78	27.3	1.52	0.81	93	7.22E-09	-
24*	30	Äspö	4.6	96	15.9	1.86	0.47	95	6.00E-11	-
25*	30	Dist.	13.7	96	17.3	1.87	0.48	101	4.09E-12	-
26*	30	Dist.	13.7	95	17.1	1.85	0.48	99	1.10E-12	-
27	30	Äspö	6.5	89	20.8	1.73	0.59	97	2.86E-09	2.98E-09

Considering the ability of the bentonite-aggregate (e.g., crushed rock, rock flour or sand) mixtures to adsorb radionuclides, the ability is likely to be affected by the mineralogical and chemical composition of the bentonite (e.g., smectite content, exchangeable cations), the proportion of the bentonite clay in the mixture, the characteristics of the aggregate (mineralogy, grain size distribution) and the radionuclide considered and should be studied for the selected material composition. In the 1990'ies, bentonite-crushed rock mixtures were considered for backfilling deposition tunnels for KBS-3V repositories for spent nuclear fuel (HLW) (SKB 2006, Posiva 1999). The distribution coefficient (K_d) values given for mixture of bentonite and crushed rock are given in Table 8-2 for a mixture with 10% bentonite and in

Table 8-3 for a mixture with 30% bentonite. Based on these data, the quantity of bentonite in the mixture has an effect on the sorption of certain radionuclides.

Table 8-2. Distribution coefficients (K_d) in the buffer (100% bentonite) and in backfill (mixture of 10% bentonite and 90% crushed rock) (m^3/kg) (Posiva, 1999).

	Bentonite buffer					Backfill				
	cons. non-sal. red.	**cons. saline red.**	cons. non-sal. oxid.	real. non-sal. red.	real. saline red.	**cons. non-sal. red.**	**cons. saline red.**	cons. non-sal. oxid.	real. non-sal. red.	real. saline red.
C	**0**	**s**	s	s	s	**0**	**sss**	sss	sss	sss
Cl	**0**	**s**	s	s	s	**0**	**sss**	sss	sss	sss
Ni	**0.05**	**0.001**	s	0.5	0.01	**0.05**	**0.001**	sss	0.5	0.01
Se	**0**	**s**	s	0.005	ss	**0**	**sss**	sss	0.001	ssss
Sr	**0.05**	**0.001**	s	0.2	0.004	**0.005**	**0**	sss	0.02	0.0004
Zr	**0.2**	**s**	s	1	ss	**0.1**	**sss**	sss	0.5	ssss
Nb	**0.02**	**s**	s	1	ss	**0.01**	**sss**	sss	0.5	ssss
Tc	**0.01**	**s**	0	0.1	ss	**0.01**	**sss**	0	0.1	ssss
Pd	**0.001**	**s**	s	0.1	ss	**0.001**	**sss**	sss	0.1	ssss
Sn	**0.001**	**s**	s	0.2	ss	**0.001**	**sss**	sss	0.2	ssss
I	**0**	**s**	s	0.001	ss	**0**	**sss**	sss	0.001	ssss
Cs	**0.2**	**0.04**	s	1	0.2	**0.02**	**0.004**	sss	0.1	0.02
Sm	**0.2**	**s**	s	1	ss	**0.02**	**sss**	sss	0.1	ssss
Ra	**0.1**	**0.002**	s	0.5	0.01	**0.01**	**0**	sss	0.05	0.001
Th	**0.3**	**s**	s	3	ss	**0.03**	**sss**	sss	0.3	ssss
Pa	**0.05**	**s**	s	0.2	ss	**0.05**	**sss**	sss	0.2	ssss
U	**0.05**	**s**	0.005	0.5	ss	**0.05**	**sss**	0.0005	0.5	ssss
Pu	**0.3**	**s**	s	3	ss	**0.03**	**sss**	sss	0.3	ssss
Np	**0.1**	**s**	0.005	1	ss	**0.05**	**sss**	0.0005	0.5	ssss
Am	**0.3**	**s**	s	3	ss	**0.03**	**sss**	sss	0.3	ssss
Cm	**0.3**	**s**	s	3	ss	**0.03**	**sss**	sss	0.3	ssss

s same as the cons./non-sal./red. K_d in the bentonite buffer
ss same as the real./non-sal./red. K_d in the bentonite buffer
sss same as the cons./non-sal./red. K_d in the backfill
ssss same as the real./non-sal./red. K_d in the backfill

K_d(cons./sal./oxid.) = min [K_d(cons./sal./red.); K_d(cons./non-sal./oxid.)]

Table 8-3. Recommended distribution coefficients for a mixture of bentonite (30%) and crushed rock (70%) (SKB, 2006).

Species (Redox State)	**K_d (m^3/kg)**	**Lower K_d limit (m^3/kg)**	**Upper K_d limit (m^3/kg)**	**Correlation group**
Ag(I)	0.0035	$7.0 \cdot 10^{-4}$	4.5	–
Am(III)	19	3.2	$1.1 \cdot 10^{2}$	3
C, carbonate species[2]	$7.0 \cdot 10^{-5}$	$3.5 \cdot 10^{-5}$	$1.4 \cdot 10^{-4}$	–
C, methane[1]	0	0	0	–
C, organic acids[1]	0	0	0	–
Ce(III)[1]	2.4	0.24	28	3
Cl(–I)	0	0	0	7
Cm(III)	19	3.1	$1.2 \cdot 10^{2}$	3
Cs(I)	0.036	0.0061	0.19	1
Eu(III)	2.5	0.31	28	3
Ho(III)	2.5	0.31	28	3
I(–I)	0	0	0	7
Nb(V)	0.97	0.095	14	–
Ni(II)	0.091	0.0096	0.99	2
Np(IV)	19	1.2	$3.3 \cdot 10^{2}$	4
Np(V)[1]	0.0073	0.0020	0.063	5
Pa(IV)	0.97	0.095	14	–
Pa(V)	0.97	0.095	14	–
Pb(II)[1]	22	3.6	$1.4 \cdot 10^{2}$	2
Pd(II)	1.5	0.090	23	–
Pu(III)	30	3.1	$3.0 \cdot 10^{2}$	3
Pu(IV)	19	1.3	$3.3 \cdot 10^{2}$	4
Pu(V)[1]	0.0060	$6.0 \cdot 10^{-4}$	0.060	5
Pu(VI)[1]	0.90	0.090	8.4	6
Ra(II)	0.15	0.007	0.17	1
Rn(–)[1]	0	0	0	–
Se(–II)	$7.0 \cdot 10^{-5}$	$3.5 \cdot 10^{-5}$	$3.5 \cdot 10^{-4}$	
Se(IV)	0.012	$9.4 \cdot 10^{-4}$	0.12	–
Se(VI)	$7.0 \cdot 10^{-5}$	$3.5 \cdot 10^{-5}$	$3.5 \cdot 10^{-4}$	7
Sm(III)	2.5	0.31	28	3
Sn(IV)	19	0.69	$5.3 \cdot 10^{2}$	4
Sr(II)	0.0015	$2.8 \cdot 10^{-4}$	0.0097	1
Tc(IV)	19	0.73	$5.3 \cdot 10^{2}$	4
Tc(VII)	0	0	0	7
Th(IV)	19	1.9	$2.1 \cdot 10^{2}$	4
U(IV)	19	1.2	$3.3 \cdot 10^{2}$	4
U(VI)	0.90	0.15	5.4	6
Zr(IV)	1.3	0.065	31	4

Correlation groups according to subsection 5.5.7.

[1] Assumed non-sorbing on rock.

[2] Assumed non-sorbing on bentonite.

It should be noted that when in contact with high pH leachates from concrete, the performance (e.g., swelling properties) of the bentonite based barriers can be affected. According to Balmer et al. (2017) cement, and especially traditional Portland cement, in contact with bentonite will increase the pH of the groundwater/porewater and in worst case can lead to mineralogical alterations such as dissolution of smectite minerals, formation of non-swelling minerals (e.g., zeolites) and dissolution of some unstable accessory minerals (e.g., christobalite) (Balmer et al., 2017). According to site studies conducted by SKB in Äspö HRL in the Prototype test where mixture of bentonite and crushed rock was in contact with a concrete plug (Svemar et al., 2016), the effect of cement on bentonite mineralogy was very local (a few centimeters distance from the plug). No evidence of smectite dissolution was observed in the test (Svemar et al., 2016). The reactions observed in the bentonite were exchange of Ca^{2+} and Mg^{2+} to Na^{+}, increase of CEC (due to cation exchange), precipitation of calcite and an unidentified Mg-phase and increase in chloride concentration (Svemar et al., 2016). Considering the barriers used in a near surface repository, the effect of the cement on bentonite-based barriers is one thing to take into account, e.g. in the selection of the bentonite used (Na^{+} or Ca^{2+}/Mg^{2+}), location of the sealing layers with respect to concrete structures (cover structures are likely not to be as much affected by high pH), type of the cement used (typically Portland) and other materials placed between the concrete and the bentonite-based barriers that may be able to chemically buffer the chemical reactions between the cement and bentonite. In addition, the local site conditions are likely to have effect on these processes.

Alternative non-bentonite based mineral sealing/barrier materials

Sediments with wide grain size distribution (e.g., glacial till) and high content of fine/clay fraction and natural clays have low hydraulic conductivity, high particle surface-area and adsorption capacity and can thus be considered as natural barrier materials or engineered barrier materials either as they are or mixed with bentonite fraction. However, in using these materials, the homogeneity and bearing capacity of these materials (clays) need to be considered in the design.

The majority of Finnish clays have been formed during the deglaciation process approximately 7,000–10,000 years ago (Eronen & Haila, 1981) and consist mainly of illite (main component in Finnish clays), chlorite, vermiculite and mixed layer minerals (illite-chlorite-vermiculite) (Soveri, 1956). The non-clay minerals present in Finnish clays consist of quartz, feldspar and amphiboles (Soveri, 1956). The clays have organic content of 2–6% and clay fraction (particles with mean diameter <2 µm) of 15–30% in Litorinal clays, 40–70% in Anculys clays and 70–90% in Yoldia stage clays (Gardemeister, 1975). Fine-rich glacial tills have at least 30% fine fraction (particles with diameter <0.063 mm) at least 5% clay fraction (<2 µm) (Lintinen, 1995) and have good potential as barrier material.

Distribution coefficients (Kd) for illite and kaolinite have been reported in Poteri et al. (2014), see Table 8-4 showing relatively good sorption capacity for certain radionuclides. Thus Finnish illitic clay and clay fraction in glacial tills can be considered as potential barrier materials at least for some specific radionuclides.

The ability of Finnish natural sediments/soils is also decribed in a case study from Olkiluoto (Söderlund et al., 2013). In this study, the distribution coefficient (Kd) of sediment and soil samples from Olkiluoto were studied for radionuclides including caesium, chloride, iodine, niobium, selenium and technetium. The sediments prevailing in Olkiluoto varied from sandy till, to clayey till and clay (Söderlund et al., 2013). Organic humus was present in samples taken close to the surface (soil layer overlying the inorganic sediments). For example the results showed that Caesium was sorbed efficiently on inorganic mineral material, but less efficiently on organic humus (Söderlund et al., 2013). In addition, sorption decreased with decreasing cation exchange capacity and clay fraction content (Söderlund et al., 2013). Thus the migration and sorption of the radionuclides in a near surface repository conditions is not only affected by the mineral materials, but also the organic materials and plants prevailing close to the ground surface.

Table 8-4. Distribution coefficients (Kd) for illite and kaolinite (Poteri et al., 2014).

Parameter	Units	Element	Illite and Kaolinite
Layer thickness	m	All	2E-4
Grain density	kg m^{-3}		2700
Porosity	-		6E-2
Effective diffusion coefficient	m^2 s^{-1}		1E-12
Distribution coefficient	m^{3}kg^{-1}	Ag	0
		Am	2.5E+02
		Be	1.8E+02
		C	0
		Cl	0
		Cm	2.5E+02
		Cs	2.0E-03
		I	0
		Mo	1.5E-02
		Nb	4.5E+00
		Ni	6.5E-01
		Np	2.0E+01
		Pa	6.0E+01
		Pd	6.5E-01
		Pu	2.5E+02
		Ra	3.4E-03
		Se	0
		Sm	2.5E+02
		Sn	1.5E+02
		Sr	3.4E-03
		Tc	2.0E+01
		Th	2.0E+01
		U	1.3E+01
		Zr	2.0E+01

8.3.3 Other alternative barrier materials (chemical barrier materials)

There are many chemical processes that can affect repository performance. These include oxidation, corrosion, dissolution, solubility limitation, diffusion and sorption. These processes can in turn be affected by the near field temperature and pressure, pH, redox potential, ionic strength (total dissolved solids), buffer capacity, chemical composition, speciation and complexation. Processes of importance for near field performance depend on waste characteristics, repository design, and the location of and materials used in the engineered barriers and in waste packages. Consideration of the near field chemical environment is important specifically in defining the retardation properties of the materials within the disposal units (IAEA, 2002). For example, in a high pH environment provided by cementitious barrier materials, both the mobilization and the transport of certain long lived radionuclides, specifically ^{14}C and actinides, could be limited because of **solubility** considerations. The important point here is that it is the chemical buffering property, not the physical integrity, of the **cementitious** engineered barrier (waste form, structural components, etc.) that provides constraints on the solubility of certain radionuclides, thereby reducing the potential for mobilization and transport. This is particularly important in consideration of the fact that a high pH environment can be maintained in the near field for a long time (IAEA, 2002).

There are some chemical substances/materials that could be considered as alternative materials for bentonite-based barrier materials in a near surface repository, for example zeolite (mineral or synthetic), magnesium oxide (MgO) and natural limestone or calcite $CaCO_3$. Abdel Rahman et al. (2009) have been studying the feasibility of synthetic zeolite Na (A-X blend) manufactured from fly ash as backfill material for a near surface repository. The barrier function of zeolite is based on high cation exchange capacity and mechanical stability (Abdel Rahman et al., 2009). Based on the results, sorption of Cs was better in zeolite in comparison to crushed rock bentonite mixture.

The use of magnesium oxide has been used in a WIPP (Waste isolation pilot plant) located in the U.S (Monastra & Grandstaff, 1999). The barrier function of the MgO is based on reactions with carbon dioxide produced in the waste and the ability of the material to uptake water (delaying the movement of water towards environment. However, the WIPP is a facility located in very different geological conditions in comparison to Finland and MgO is not recommended as barrier material for a near surface repository located in Finland.

Natural limestone is also known to have good sorption capacity for certain specific radionuclides (see table 5.7 in Baeyens et al., 2014), as well as calcite ($CaCO_3$) especially in the case of Americium (see table 67 in Vilks & Yang, 2018).

8.3.4 Membranes & geotextiles

As presented in Figure 6-2, the multi-barrier system of the near surface repository can contain geotextiles and geomembranes both in the cover and the bottom liner. These experiences come from the landfill construction. One form of a geotextile is a bentonite mat that can be used to replace or complement the mineral sealing layer, which is usually constructed from clay, silt, silty till or soil-bentonite mixture. Also other geosynthetic materials or by-products of industry can be used if they fulfil the requirements of the sealing layer (SYKE, 2008).

The sealing layer acts as a supporting layer for the synthetic membrane. Other functions of the sealing layer in the landfill bottom liner are to minimize filtration and diffusion of detrimental elements from the waste fill, and also to adsorb detrimental elements from the waste fill. In the cover, functions of the sealing layer are to reduce infiltration of rain water to the waste fill, and lead gas to the gas collection layer (SYKE, 2008).

Stresses in the landfill sealing layer include the following (Tammirinne et al., 2004):

- Amount of the filtration water
- Chemical composition of the filtration water (risk for the change of granularity/water permeability)
- Biological composition (detrimental increase in the permeability)
- Hydraulic gradient (hydraulic pressure)
- Temperature of the waste fill
- Bioturbation (plants, animals), if there is no synthetic membrane
- Weight of the waste fill
- Deformation of the waste fill below the sealing layer (settlements, uneven settlements)
- Displacements of the waste fill (slope stability)

A geotextile is a synthetic fabric that is permeable for gases and fluids and it is used in different types of soil structures to separate, filter, or protect. Bentonite mats consist of different nonwoven and woven fabrics where there is bentonite powder added between two fabrics. The amount of bentonite is commonly 3.5–6 kg/m^2. Geotextiles used in the bentonite mats can be chosen according to its planned application. The other fabric can be a reinforcement mat. If the bentonite mat is used between the mineral sealing layer and the geomembrane, the fabric should not conduct water and thus it should be filled with bentonite powder (SYKE, 2002).

Bentonite swells when it is saturated with water and this results to very low water permeability. In high water content bentonite is also very impermeable to gas. Bentonite can dry and crack but it swells and repairs the cracks when water is introduced. Thus, bentonite mats are self-healing and they repair the holes in them. If there is an ion change from sodium to calcium during time, it decreases the swelling capacity of bentonite, which increases the water infiltration capacity. However, according to a study by Egloffstein (2001), after the increase in water infiltration capacity to 10–15 -fold due to the ion change, the bentonite mat still infiltrates only 1–2% of the yearly precipitation (SYKE, 2008).

Bentonite mats have a high resistance to deformations of the layers or the subsoil below them. This resistance is much better than with natural soil materials. However, if the bentonite mat tears, its water infiltration rate increases. Bentonite mats also have a high resistance to stresses from freezing and thawing. The internal shear strength of bentonite mats is an important parameter when the long-term durability of bentonite mats in the slopes is evaluated. The strength is weakened by creep and oxidation (SYKE, 2008).

The quality of the base for the bentonite mat is critical regarding the non-permeability of the bentonite mat. The base should not contain stones, wooden sticks etc. If the drainage layer above the bentonite mat contains crushed stones, the bentonite mat should be protected (SYKE, 2008).

The minimum quality requirements for the bentonite mats in the landfills can be presented as (SYKE, 2008):

- Bentonite should be natural sodium bentonite, activation is not allowed.
- Quality of bentonite, i.e. the montmorillonite content, is measured either by XRD method (requirement is 90%) or by methylene blue method (requirement is ≥300 mg/g). Quality of bentonite is tested also using a swelling test. Minimum requirement with the ASTM D5890 method is 24 mL/2 g and with the DIN 18132 test it is ≥600%.
- The minimum amount of bentonite in the mat is 4000 g/m^2 in the 0% water content. The average is then 4500 g/m^2.
- The upper fabric should be non-woven and it should weigh at least 200 g/m^2. The lower fabric should be a combination of woven (the minimum weight 100 g/m^2) and non-woven (the minimum weight 100 g/m^2).
- The tensile strength of the bentonite mat should be at least 7 kN/m.
- The peel strength of the bentonite mat measured using the Peel test should be (ISO 10319) ≥60 N/10 cm.
- Elongation of break of the bentonite mat should be at least 25%.
- Infiltration rate of the mat should be either <5x10^{-11} m/s (permeability) or <5x10^{-9} m^3/m^2/s (permittivity).

Above the sealing layer both in the bottom liner and the cover of the waste fill in landfills, a synthetic membrane is used. Usually the synthetic membrane is geomembrane which has a minimum thickness of 2 mm. The geomembrane is manufactured from HDPE or other material that has a sufficient chemical durability (SYKE, 2008).

The main materials used in the production of the geomembranes are polyethylene (PE) and polypropylene (PP). Especially polyethylene liners are commonly used in the landfill cover and bottom liner solutions. Advantages of polyethylene include its resistance to different chemicals, excellent seaming possibilities, and tolerance for low temperatures. A disadvantage of the polyethylene is its stiffness which can make its installation more difficult. Water does not have any major effects on the properties of the polyethylene due to its low water absorbancy. However, over time water can weaken polyethylene (SYKE, 2002).

Based on the density of the product, different groups of polyethenes can be specified: LDPE (low density), VLDPE (very low density), LLDPE (linear low density), and HDPE (high density). HDPE has better mechanical properties and chemical durability than LDPE. For the production of other geosynthetic products (for example, the fabrics used in bentonite mats), commonly used polymer types are polypropylene (PP), polyester (PETP), polyethene (PE), and polyamide (PA) (SYKE, 2002).

Properties that affect the selection of geomembranes are chemical durability, mechanical resistance, environmental stress crack resistance (ESCR), installability and weldability, friction properties (in slopes), duration of UV radiation (during installation), and frost resistance (SYKE, 2002).

According to SYKE (2002), the function of the geomembranes used in the landfills as part of the bottom liner and cover is to prevent rainwater infiltratation to the sealing layer and then to the waste fill and/or subgrade. In the cover, the geomembrane is also used for enhancing collection of gas and preventing the vegetation to penetrate to the sealing layer. Stresses on the geomembranes include the following (Tammirinne et al., 2004):

- amount of the infiltrated water (detrimental substances in the water)
- temperature (temperature of the waste fill and temperature of the outside air)
- chemical composition of the infiltrated water
- biological composition of the infiltrated water
- water pressure (especially in the bottom liner)
- gas composition and pressure
- mechanical stresses from above during the construction work
- weight of the waste fill
- deformation of the layers below (settlements, uneven settlements)
- displacement of the waste fill (in the slopes)
- UV stress (during the construction work and during storing)
- temperature during the construction work before covering (thermal expansion of the geomembrane, elongation/shrinkage).

Expected/recommended properties of the geomembranes include (Tammirinne et al., 2004):

- resistance for the long-term deformation including settlements
- long-term chemical durability
- long-term biological durability
- UV radiation durability (during the construction)
- frost resistance
- compatibility with the sealing layer (e.g., chemical compatibility, friction of the interface > stability, tight contact)
- Usually a HDPE membrane is used, thickness ≥2 mm.

8.3.5 Uppermost layer (overlying soil & vegetation)

The uppermost layer of the disposal unit is the outer barrier for near surface disposal facilities. Its main functions are to:

- Limit the quantity of rainwater infiltrating into the disposal units by surface runoff. The surface runoff is enabled using relatively impervious material (e.g., glacial till as in the case of near surface repository in Ringhals) and sufficient inclination of the layer. Ideally, the surface runoff waters will be directed to a separate system (e.g., with ditches) from leachate waters infiltrating through the waste.

- Protect the underlying barrier layers from freezing and thawing (thickness to be dimensioned according to RIL, 2013)
- Protect animals from intruding the repository.
- Protects the underlying mineral sealing layers/other layers from erosion by having suitable vegetation planted on the surface (e.g., grass). Tree roots should be avoided, since roots can puncture the geotextiles or bentonite mats used in the repository.

8.4 Other engineering components

Depending on the disposal system concept, the engineered barrier system may be supplemented by other engineered components, including leachate collection and drainage systems, cut-off walls, gas vents and monitoring wells (IAEA, 2002). Some of these components are briefly described below.

8.4.1 Drainage systems

Drainage systems in a near surface repository have two main functions:

- Limit the amount of precipitation infiltrated into the repository.
- Collect and monitor potentially contaminated leachate waters infiltrated through the waste.

The first function can be achieved with sufficient layer inclinations, water drainage systems for the surface-runoff water (e.g. ditches) and with a drainage layer placed above a sealing layer as in the case of hazardous waste landfills (described later in this chapter). Considering a vault type of repository, the drainage systems can also be built around the repository. Drainage systems can be implemented with various techniques, e.g., using coarse mineral layers and drain pipes.

The second function is achieved by drainage systems placed below the waste. For a near surface repository it is important to consider where the leachate water is led, how it can be monitored and how this drainage option affects the safety of the system. Ideally the leachate water should be kept in a separate system from other waters collected at the site. In the Ringhals example case, the water is lead into an infiltration bed and the water table and quality is monitored from groundwater pipes. In more engineered near surface repositories, the drainage system can be based on concrete structures as in example cases from Centre de l'Aube and El Cabril (described briefly below).

In Centre de l'Aube (IAEA, 2001), the drainage system is implemented with a sloping concrete floor (slope about 1%) located on the slab and the slope converging towards the orifice of the water collection system. The floor is covered with a polyurethane coating (IAEA, 2001). A gravity based water collection system is operated to collect any water that may seep into the cells during operating. The collection system starts at the base of each vault, runs through underground drains, and discharges into two 250 m^3 basins (IAEA, 2002). Another example is from El Cabril site, where any seepage water is collected at the base of each vault and is channelled to a network of pipes installed in inspection drifts located below the disposal vaults. Each vault is linked to this network, called the infiltration control network, via a holding tank, so that if water is collected in the control network, it is possible to know which vault it has come from in order to repair the protective covering, and to take samples of the water collected (IAEA, 2001).

Example of drainage systems in hazardous waste landfills in Finland

Considering the drainage systems from a hazardous waste landfill the function of the drainage layer in the bottom liner layer is to collect and remove the filtration water from the waste fill and to decreases the hydrostatic pressure on the sealing layer. In a hazardous waste landfill, the filtration water is pumped from the bottom of the waste fill and it is transported to a treatment plant. Basis for the design of the layer thickness, material selection, and pipe system is that the hydrostatic pressure should not be more than 1 m above the sealing layer (SYKE, 2002).

In the Finnish legislation for landfills (VNp 331/2013), the minimum thickness of the drainage layer above the sealing layer is h ≥0.5 m. The drainage layer is constructed using globular rock material that has a hydraulic conductivity of at least 10^{-3} m/s. Regardless of the used material, the long-term chemical and mechanical durability as well as stability of the hydraulic properties and stability of the structure should be checked (SYKE, 2002).

In a hazardous waste landfill, water is usually collected in the drainage layer by subsurface drains (pipes) from where the water is further led to pipes or ditches. Uncontrollable discharge of water may cause erosion damage (SYKE, 2002).

The function of the drainage layer in a landfill should be secured by designing the bottom so that the inclinations will not be disturbed despite of the expected settlements. In the landfills, usually a pipe collection system for the filtration water is installed to the bottom of the landfill. The pipe material should withstand the chemical loading of filtration water and mechanical loading of the waste fill. The recommended inclination of the drainage layer towards the pipe system is 3% and in direction of the pipe system it is 1% (SYKE, 2002).

In the drainage layer placed above the topmost sealing layer (cover overlying the waste), the function of the drainage layer is to reduce the hydrostatic pressure on the sealing layer and also to direct the filtration water out from the structure. The following factors should be taken into account (SYKE 2002, 2008):

- Minimizing the hydrostatic pressure by taking into account water infiltration capacity (material and layer thickness), hydraulic gradient, and amounts of filtration water
- Erosion durability
- Prevention of sliding on the slopes
- Prevention of clogging of the drainage layer.

Minimum thickness of the drainage layer in the Finnish landfills is 0.5 m (VNp 331/2013). The recommended hydraulic conductivity is k $\geq 10^{-3}$ m/s and the minimum inclination is 5%. This type of structure gives a very high safety regarding the water permeability (SYKE 2002, 2008).

Outside of the waste embankment in landfills, diches are constructed. In the ditches, waters coming from outside are collected and this prevents the water to access the waste fill. Polluted surface water and filtration water are collected separately to ditches and possibly to subsurface drain for their special treatment (SYKE, 2008).

The necessity of treatment of the collected water depends on its quality and quantity. The quantity of the water can be controlled by evaporation, which can be increased by leading the water to the surface of the landfill or by turfing or planting. The quality of the water has an effect on the required treatment methods. For the landfill waters, several physical, chemical, and biological treatment methods are available (SYKE, 2008).

8.4.2 Gas pressure release systems

Need for the gas collection layer in the near surface repository depends on the type and amount of gases generated in the waste. If the amount of gases generated is small, a separate gas-collection system may not be needed. If needed, then the systems used in landfills are suitable at least if the repository is of landfill type.

Gas collection systems for landfills in Finland

In the landfills, the most important requirements for the gas collection layer include (SYKE 2002, 2008):

- Gas collection capabilities; dimensioning based on the layer thickness and gas permeability
- Resistance against aggressive gas components
- Durability for the filtration water
- Safety against crusting.

The recommended minimum thickness of the gas collection layer in the landfills is 0.3 m when it is made from soil material. The layer should be made from well-sorted soil or from geosynthetic material and it is attached to the gas collection system (SYKE 2002, 2008).

In the landfills, the gas collection system is designed and installed taking into account, e.g., structure of the waste fill as well as quantity and quality of the gas. By examination, the zones generating the most gases can be detected. Gas escapes the embankment via the easiest routes like cracks and coarse layers. When designing the gas collection system, settlements of the embankment need to be taken into account. The gas collection system can be based either on vertical suction well system or horizontal subsurface drain type of a system. The gas collection system is attached to a gas pumping station and further to treatment. The landfill gases can be used for energy production (SYKE, 2008).

8.4.2.1 Prevention of human intrusion

In principle, human intrusion to the repository shall be prevented by limited access to the repository site during the operational period and during the institutional control period. In addition, the future land-use shall be limited by town and country planning.

9. Monitoring

Monitoring means the continuous or periodic observation and measurement of radiological, environmental, engineering and other relevant parameters. Monitoring helps in the evaluation of the behaviour of the different components of the disposal system, and of the impacts of the repository on the environment (IAEA, 2002). Basic principles for the design of the repository is that it operates safely with or without monitoring, which is only used to confirm the operation of the repository.

The monitoring programme should be defined prior to construction and in coordination with development of the safety case. As such, the monitoring programme should be made subject to audit and independent verification by the regulatory body (IAEA, 2014). Technical and scientific data obtained from the results of monitoring and measurement may also be used to improve the assumptions and models for safety assessments (IAEA, 2014).

The monitoring programme should be revised periodically to reflect new information gained during construction, operation and closure (IAEA, 2014).

Monitoring needs and requirements start already in the early phase of planning of the repository and continue to the site post-closure. These requirements can be divided to follow the site development phases, which are depicted in the Figure 9-1.

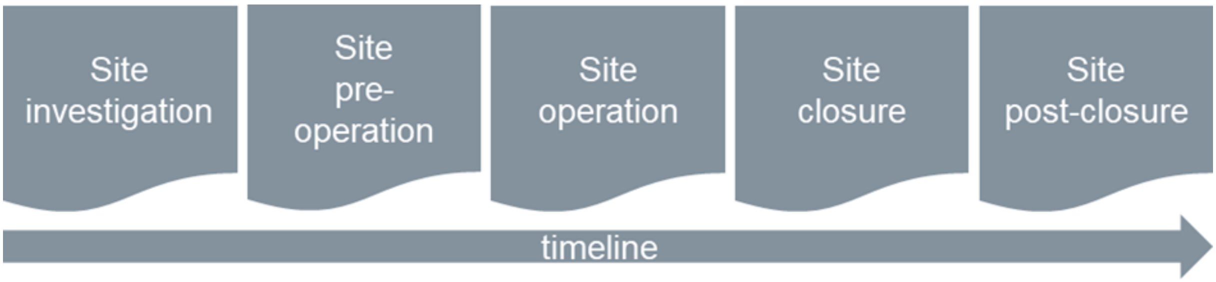

Figure 9-1. Monitoring phases following the site life cycle.

Although monitoring of the site in different life cycle phases is very important, it is still a supporting action in the NSDF. Because of this, economic factors are often the most limiting restrictions, when the planning and implementation of the monitoring is performed.

The timescale of the NSDF is much shorter than the expected timescale of the geological disposal facilities, but it is still considered to be in total few hundred years. Most of the installation costs of the monitoring systems are cumulated during the site pre-operation and site operation phases, but the maintenance, data collection and monitoring as well as data analytics of the collected data will be spread over the whole service time line of the site generating substantial operating costs. Often these long term operating costs are partly bypassed in the planning phase.

Lessons learned and best practices from other industry domains should be followed, when defining the monitoring plans. Monitoring system should be designed to be modular and expandable allowing adoption of novel techniques, as new technologies will emerge during the operation of the site. Critical measurements should be duplicated or followed with various techniques to ensure reliable data collection. Monitoring system should adopt Condition Based Monitoring principles, if possible. These techniques would give early warnings, before actual events occur and mitigating maintenance operations could be performed in time. Although monitoring systems will collect only measurement information, automated or human initiated actuators and controls can also be installed in the repositories to prevent or mitigate events happening because of special conditions.

A fully automated, reliable 24/7 on-line monitoring system would be an ideal target, but budgetary and other issues induce that the repository monitoring system will most likely be a hybrid solution with automated measurements, laboratory sampling and human interventions (visual inspections, etc.).

When planning and implementing the monitoring system, it has to be taken into account that the site environment with radiation, demanding chemical substances and long time period form a hostile and demanding environment for the monitoring system. Because of this, the monitoring system should be planned together with the site, not as a later add-on system.

9.1 Pre-operational monitoring

During the site investigation phase, prior environmental data from earlier collected information sources should be collected. Additionally early monitoring systems should be installed and data collected to find a suitable location for the NSDF. Long term historical data is desirable to support the design and operation of the site and prepare for extreme events, e.g., related to weather conditions, but this is not always possible, if the environment hasn't been monitored earlier.

After finding a candidate location, a baseline survey of the site, including characteristics of the host environment, should be conducted before commencing construction activities (IAEA, 2014).

Environmental monitoring covers a broad range of media, including air, surface waters, soils, and flora and fauna that may be part of the food chain. Groundwater and vadose zone (i.e., unsaturated zone) monitoring, on the other hand, can provide an early warning of the release of contaminants, especially the ones that are highly mobile. The broad nature of environmental monitoring provides reassurance that significant exposure pathways have not been overlooked or underestimated (IAEA, 2002).

The following are examples of repository features, processes, parameters and characteristics that can be monitored on an ongoing basis (IAEA, 2002).

- Meteorological conditions: precipitation, temperature, wind, evaporation.
- Geomorphological aspects: erosion mechanisms and their rates.
- Hydrological conditions: runoff, flow characteristics of existing water streams, lakes and wetlands.
- Hydrogeological conditions: infiltration and evapotranspiration, permanent and temporary springs, depth and oscillation of the water table, preferential flow pathways, direction and rate of groundwater flow in both vadose and saturated zones, travel times to existing and potential outflow and extraction points.
- Geochemical conditions and environmental quality: water quality, concentrations of naturally occurring radionuclides in a variety of environmental media, retention of radionuclides by soil and geological materials.
- Geotechnical conditions: rock stress, response of the geological media to excavation and load of support structures.
- Radiation background level.

For disposal in the saturated zone, the disposal units will be excavated, drained and exposed to atmospheric air. In addition to mechanical disturbances caused by the construction and operation activities, exposed rock surfaces may dry out or be oxidized. Some unlined excavations may crack and require support. After closure, the groundwater that has been drained during construction and operation is expected to re-enter the disposal zone and gradually fill the disposal units. These processes may need to be monitored and their effects accounted for (IAEA, 2002).

Monitoring systems should be installed, tested and ready to be started before the actual operation starts.

9.2 Operational monitoring

During the operational period, the monitoring programme should be used to demonstrate compliance with the regulatory requirements and licence conditions for operation, including compliance with safety requirements for environmental and radiological protection (IAEA, 2014).

Operational monitoring and surveillance data, besides being important in achieving radiation protection and physical security objectives, may indicate deviations from predicted conditions. Therefore many of the monitoring activities initiated in the pre-operational phase are likely to continue during the operational phase (IAEA, 2002).

The potential exists for changes to the local environment, induced by the construction and operation of the repository, which can affect the performance of the system. For example, increased water infiltration can be caused by the disturbance of the ground surface, the loss of native vegetation over the disposal units, the drilling of boreholes or the channelling of runoff water. Another example is the potential generation of preferential pathways for the migration of groundwater, and any released radionuclides, that may result from the construction of rock cavity repositories. Any such induced changes are likely to require specific modifications to the monitoring programme to determine their potential impact on the future performance of the system (IAEA, 2002).

Subjects to be monitored during the operation and closure phases of the NSDF include:

- Radiation safety (perimeter based radiation levels and radiation doses)
- Radiation background level
- Ground water (Characteristics, water accumulation, discharge areas, flow directions)
- Surface water run-off (amount and direction)
- Leakage waters from drains
    - Flow rate and location of the leakage
    - Chemical composition of the leakage water, including radioactive isotopes
- Leakage of gasses
- Condition of the waste packages
    - Surface contamination, dose rate
- Intactness of the engineered (and/or natural) barriers (this can be challenging)
- Weather conditions (including historical)
    - Precipitation, temperatures, air temperature, wind speed
    - Weather forecasts for future weather related event mitigation/prevention
        - E.g. extreme weather conditions
            - Protective measures to prevent flooding waters to enter the repository
            - Stormwater handling systems (active or passive)
- Biosphere (Flora and fauna)

Perimeter protection and access control

9.3 Post closure monitoring

For the post-closure period, the near surface disposal facility should not require or rely on a post-closure monitoring programme to provide assurance of safety. Post-closure monitoring may be performed to provide public assurance, if required, by the government or the regulatory body, but should not compromise the safety functions of the facility. Monitoring for non-radiological contaminants, which may be of concern, may also be necessary (IAEA, 2014, 2014b).

In general, post-closure monitoring of a disposal facility serves several functions. It can provide an early warning of system malfunctions that might lead to unacceptable impacts on individuals and the environment. It can help to confirm the predicted performance of the disposal system. It can also play a role in providing confidence for stakeholders that the system is functioning appropriately (IAEA, 2003).

Technical requirements for the monitoring programme during the post-closure phase are not expected to differ significantly from those relevant to the operational phase, with specific monitoring being required to ensure the performance of additional barriers installed at closure (IAEA, 2002).

For example, infiltration through engineered covers may be monitored and compared with predicted values. Other examples of measurements to comply with the post-closure monitoring objectives would be the collection and analysis of water samples taken from a leachate collection system, measurements of moisture distribution in low permeability covers and in unsaturated materials underlying the disposal units, and the collection and analysis of water samples taken below or immediately down-gradient from disposal units (IAEA, 2002).

The level of monitoring is anticipated to decline over time, in accordance with the monitoring strategy developed before closure (IAEA, 2003). If the repository is not performing as expected, then corrective actions might need to be taken (IAEA, 2003). Satisfactory monitoring results over an extended period of time are generally considered an essential precursor to the discontinuation of institutional controls, depending on acceptable results (IAEA, 2003).

Although the declination of the monitoring activities is foreseen, several of the operation phase monitoring should continue also in the site post-closure phase. Critical system monitoring must continue till the end of the post-closure phase.

Data management of the measured and analysed data should be also considered carefully. During the pre-operational and operational phases, this will be most likely happen automatically, but during the long post-closure period data management can become an issue. During all site phases, measured historical data and documents need to be stored reliably and securely.

9.4 Current monitoring technologies

As previously noted, a fully automated, reliable 24/7 on-line monitoring system would be ideal; however, budgetary and other issues mean that the repository monitoring system will most likely be a hybrid solution with a combination of automated measurements, laboratory sampling and human interventions (visual inspections, etc.).

During the timeline of the site, monitoring can be performed by several actors. These actors can be, e.g., the site owner, constructors, site operator, authorities or even actors performing institutional control measures.

Examples of current measuring methods and quantities include:

- Geophysical methods
  - Boreholes
  - Seismic monitoring
  - Pressure cells
- Material measurements
  - Displacement sensors
  - Moisture sensors
  - Strain gages
- Ground water

  - Groundwater sampling
  - Groundwater wells
- Leakage waters
  - Flow-meters
  - Pressure gauges
  - Sampling
- Weather
  - Weather stations (rain, temperature, wind, …)
- Radiation
  - Dosimeters (personal + automatic on-line measurements)
  - On-line radiation level measurements
- Visual inspections
  - Camera based
  - Human based
- Robots with carry on sensors and cameras
  - Drones
  - Rovers
  - Stationary robots.

9.5 New smart monitoring options

The nuclear industry has been traditionally conservative and careful when considering the adoption of novel technologies. This is also the case for wireless technologies. With the wireless communication the concerns have been reliability, security, electromagnetic compatibility (EMC) / - interference (EMI), spectrum management, heavy structures and radiation. As the NSDF is not an actual nuclear power plant (NPP) with heavy restrictions, combination of wired and wired monitoring systems could be utilized, if the radiation authorities would grant a permission. IAEA and many nuclear operators have been experimenting various wireless applications and technologies over the years and received promising results to adopt them also in the nuclear sector (Laikari et al., 2018).

As an example, novel sensors and radio technologies can nowadays provide the opportunity to create extensive wireless sensor networks to monitor and control complex systems without wires. Wireless communication enables the mobility of personnel and applications creating new ways to rationalize the operations in many business sectors. Freedom from the wires opens also opportunities to develop systems into processes, where wired systems would not be possible to be implemented.

Although use of wireless technologies bring out novel threats and disadvantages, there are countermeasures and solutions making them attractive to be used in NSDF monitoring solutions.

During the lifespan of the repository, lot of data will be created and generated. These include also the safety case and design documentation of the repository. The lifespan of the repository can be greater than the lifespan of the repository operator. This creates the requirement that all the data need to be stored securely and reliably. It has to be accessible for the needed stakeholders, including authorities even after the site post-closure phase taking into account the information security. As a future need, some information or data might be needed to be open at least to some external users. This implies that a data management plan need to be created already in the design phase and maintained and updated during the whole timeline of the site.

10. Safety case

At each key decision point in the development of a disposal facility (see Section 4), a safety case is prepared with the purpose of demonstrating safety, to serve the licensing process and to be used as the basis for the dialogue with different stakeholders and interested parties (IAEA, 2012). In addition, the safety case provides guidance on the allocation of resources and on the site investigation and research and development programs (IAEA, 2011a). A graded approach is applied in the development of the safety case, that is, the hazard associated with the radioactive waste in question and the stage of the development of the disposal program is reflected in the formality and level of (technical) detail of the safety case (IAEA, 2006b). Based on the application of the graded approach and the safety strategy described in Section 3, the safety case should be developed in a stepwise and iterative manner. Besides requirements arising from national legislation, the safety case also takes into account the decision at hand, i.e., not only the transition from one stage to another in the development of a repository but, for example, choosing between different design alternatives, the refinement of waste acceptance criteria or the definition of operational procedures (IAEA, 2002, 2011a). Regarding the safety case's role as a communication tool with different interested parties, the presentation of the safety case, its level of detail and form (technical-scientific versus qualitative-descriptive language) may vary depending on the intended audience (IAEA, 2011a). In any case, the presentation of the safety case should be clear, comprehensive, traceable and transparent. This is particularly important for the safety case review for scientific-technical aspects, regulatory aspects and non-technical aspects by the relevant decision-makers and interested stakeholders (NEA, 2013, IAEA, 2014).

In accordance with IAEA (2012), the safety case can be understood as the compilation of scientific, technical, administrative and managerial arguments and evidence in support of the safety of the disposal facility. The safety case addresses the suitability of the site and the facility design, its construction and operation. The core of the safety case form the safety assessments (Section 10.1), which primarily aim at the systematic assessment of radiation safety and at understanding the disposal system's behaviour considering different potential evolutions (scenarios) for the time frames over which the radioactive waste remains hazardous (IAEA, 2011a, 2012). The safety case further provides the necessary assurance of the adequacy and quality of all of the safety related work by using appropriate management systems (Section 3.3) (IAEA, 2012). This is particularly crucial for the information exchange between the researchers (i.e., experts) and safety assessors (i.e., generalists) (IAEA, 2003; NEA, 2013).

As described for the safety strategies for the operational and post-closure periods (Section 4.3), account is taken of active and passive safety measures, with greater emphasis on passive measures in particular for the post-closure safety. In this regard, reliance should be placed on institutional control (e.g. preventing human intrusion by access controls to the site) for a limited period only (no more than a few centuries) (IAEA, 2014).

Several safety cases for different types of near surface disposal facilities have been prepared in different countries (see, for example, Watts, 2002; Wacquier & Cool, 2014; Soulet & Griffault, 2016). In recent years, substantial effort has been made by international organizations (e.g., IAEA, OECD-NEA, WENRA) in the harmonization of the safety case methodology. However, small differences remain, in particular with regard to the terminology used. In the following, OECD-NEA's version (NEA, 2013) is used to present briefly the main components of safety case (Figure 10-1).

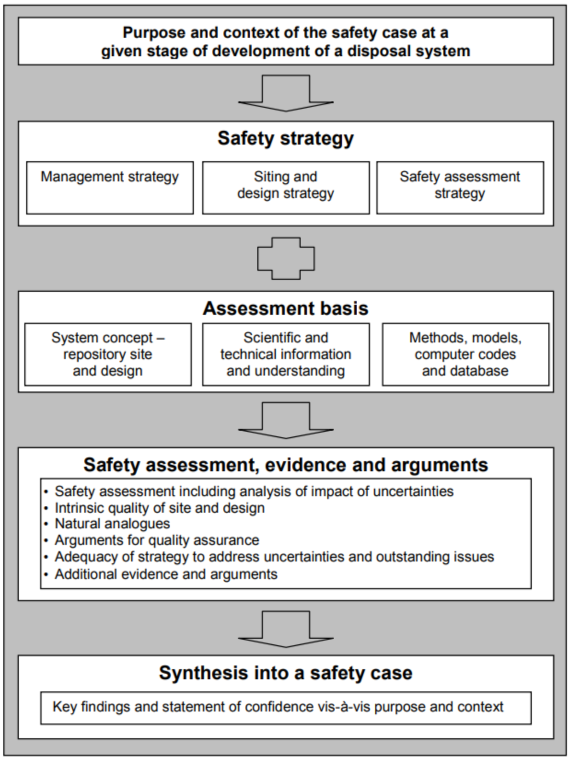

Figure 10-1. Safety case methodology according to NEA (2013).

The *purpose and context* of the safety case have been addressed above. The safety case will take into account the stage of development of a disposal facility, the radioactive waste in question, the decision at hand as well as national requirements. The main aspects relevant to the *safety strategy* with regard to management systems and the site features and repository design have been covered in Section 3. The safety assessment strategy is discussed together with *safety assessments, evidence and arguments* in the next sub-chapter (Section 10.1). In order to be able to perform reliable and well-founded safety assessments an adequate *assessment basis* needs to be established. This includes the information about the site and the facility design (Sections 6 and 7) and the scientific and technical basis, which has also been addressed in Section 7 and is further complemented by the discussions about monitoring and data management in Section 9 and Section 3.3, respectively. The methods and database have already been addressed partly in Section 3, while the main aspects regarding the required modelling capability are discussed in Section 10.1. The *synthesis into a safety case* including the potential outcomes and findings has been outlined together with the purpose and context of the safety case above and is further complemented by the discussions hereinafter.

10.1 Safety assessment, evidence and arguments

Safety assessments are primarily used to assess quantitatively the radiological impact on humans and the environment resulting from the potential future release of radionuclides from the disposal facility and their migration into the biosphere. The results of safety assessments can further be used to inform the R&D program and facility design, e.g. by indicating the need for solubility constraints on the concentrations of some radionuclides in certain compartments of the disposal facility and thus, the need for materials that help establishing and maintaining favourable hydrochemical conditions (IAEA, 2012). It should be noted that on an international level, the scope of safety assessments has been broadened and may involve the assessment of the non-radiological environmental impact, operational safety assessment, assessment of the adequacy of the site and the engineered systems and assessment of the appropriateness of the management system (Figure 10-2). The assessment of the post-closure radiological impact, however, constitutes the main element of safety assessments and is generically discussed in the following (IAEA, 2012). Nair & Krishnamoorthy (1999), Cool et al. (2013), Lee & Kim (2017) and Cota et al. (2018) present practical examples of safety assessments of near surface disposal facilities.

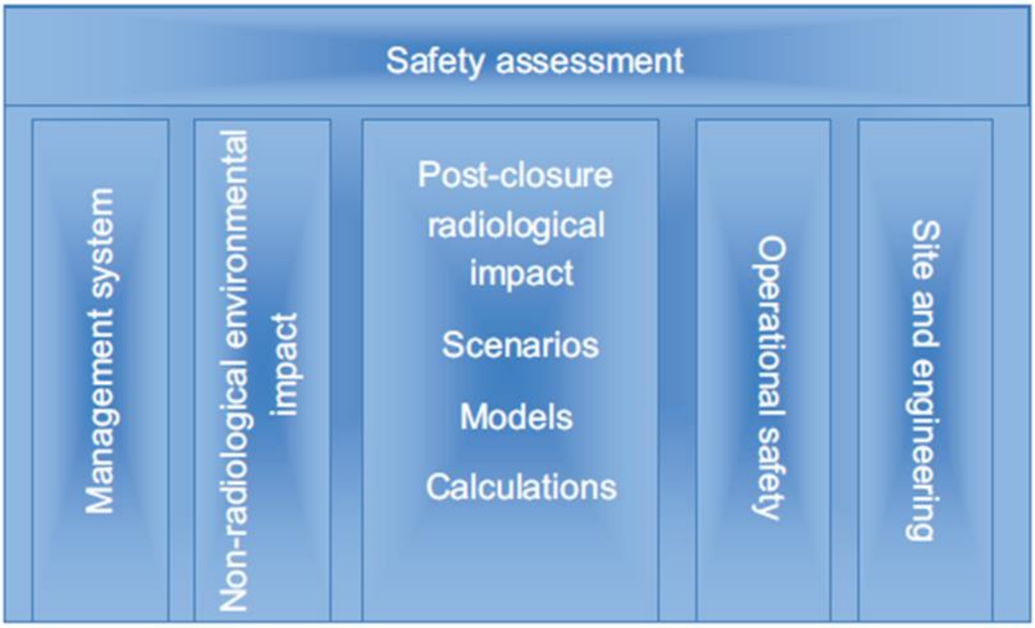

Figure 10-2. Aspects included in safety assessments (IAEA, 2012).

10.1.1 Scenarios

The uncertainties inherent in the evolution of the disposal system and its environment over the assessment period lead to the definition of hypothetical scenarios illustrating the expected evolution under normal conditions and likely and less likely disruptive events. Scenarios should be developed and selected for subsequent assessment in a formal process allowing for traceability and transparency and subjected to quality control. Different approaches can be employed and are typically used in a complementary manner. In a bottom-up approach, scenarios are constructed by combining features events and processes (FEPs) relevant to the disposal system provided by a solid assessment basis with information about the natural and engineered features and the relevant scientific and technical understanding. In addition, international and national compilations of FEPs exist covering factors connected to the different compartments of the disposal system (i.e., near field, geosphere and biosphere) and external factors (e.g., earthquakes, high rainfall, floods, landslides, erosion and human intrusion) (see, for example, NEA, 2019). Starting from a top-down approach, the safety functions (e.g., waste containment or restricting water infiltration into the facility) assigned to the different components of the disposal system (e.g., waste containers or the facility cap) are analysed for potential impairment due to different internal and external factors (IAEA, 2014). Note that near surface disposal facilities are generally more susceptible to human intrusion than geological repositories (IAEA, 2014). In this context, the role of the institutional control period (Section 4.3) and its implications on safety assessment needs to be considered. The inventory, i.e., predominantly short-lived radionuclides in case of a near surface disposal facility, determines both the assessment period, which is naturally shorter compared to geological disposal of waste containing mainly long-lived radionuclides, and the duration of the institutional control period. Again, it is important to stress that the use of active measures during institutional control period (e.g., access controls) do not justify any undermining of the passive safety concept. After the institutional control period, the safety of the disposal facility relies fully on passive measures (i.e., laws of nature) and human intrusion can no longer be excluded, which needs to be reflected in the definition of scenarios (IAEA, 2002). Regarding formal procedures for scenario development, the interested reader is referred to STUK's YVL D.5 guide (STUK, 2018) and the KYT2022 project SYSMET.

Additional 'what if' scenarios should also be described to allow assessment of, for example, different waste inventories or alternative design options and disposal methods (IAEA, 2014).

10.1.2 Modelling

The evolution of the disposal facility illustrated by the chosen set of scenarios is modelled numerically. For this purpose, *conceptual models* are developed describing the behaviour and interaction of the different components and materials taking into account the influence of the biosphere environment. In this regard, a close interaction between the modellers, site characterisation and repository design (assessment basis) and scenario development is required (NEA, 2013). Next, the processes identified in the conceptual model need to be described by a set of often complexly coupled mathematical equations to build a *mathematical model*, which then needs to be incorporated into a suitable computer code ('bits and bytes') to obtain the *numerical model*.

For the model development, a quality controlled qualification processes is indispensable (see Section 3.3). This includes (IAEA, 2002):

- Model verification to ensure that the mathematical equations are solved correctly, e.g., by comparison with analytical solutions;
- Model validation to examine the model's ability to reproduce measurements from a wide range of experiments of different scales and field measurements. Naturally, a full model validation over the relevant timescale is impossible;
- Model calibration, similar to model validation, aims at the comparison between numerical and experimental results or field data. The difference is that model calibration is based on a set of input data from a specific experiment or specific site conditions and is used to reduce the uncertainties related to certain conditions or processes of interest.

Models can be focused on single or coupled processes to gain better understanding of certain features of the disposal system. These process-level models are integrated to system-level models aiming at the interaction between the different system components and at representing the disposal system as a whole. System-level models are used for the assessment of potential radiological consequences (e.g., doses or risks) for humans and the environment resulting from the waste disposal and for demonstrating compliance with regulatory criteria and requirements (NEA, 2013). However, it should be kept in mind that models only provide illustrations of possible evolutions of the disposal system and the calculation results need to be understood as estimates and be interpreted in the light of the underlying assumptions and uncertainties (Section 10.1.3). In other words, modelling essentially represent extrapolation of assumptions made (if then logic). Therefore, multiple lines of evidence should be used in a safety case to support the confidence in the safety statement (Section 10.1.4) (NEA, 2013).

10.1.3 Uncertainty management

Uncertainties in safety assessments are unavoidable. They do not only stem from the impossibility to make an accurate prediction of the future evolution of the disposal facility, particularly in terms of the biosphere and the involved dynamic processes, but also originate from the approximations of the physical reality at each abstraction level of the model development process (i.e., conceptual model, mathematical model, numerical model). Thus, a systematic uncertainty management is an essential part of safety assessment and safety case (IAEA, 2011a, 2014a). The management begins with the identification of uncertainties and, where possible, their quantification. It is useful to assign uncertainties to groups or categories but, due to the complex interrelations, the grouping might not always be unambiguous and straightforward. However, it is essential that all relevant uncertainties have been considered. It follows an example of uncertainty groups by NEA (2013):

- Scenario uncertainties are associated with significant changes that may occur within the engineered and natural systems over time, and the uncertainties concerning physical and chemical processes accompanying those changes.
- Model uncertainties arise from an incomplete knowledge or lack of understanding of the behaviour of natural and engineered systems, physical processes, site characteristics and their representation using simplified models and computer codes.
- Data and parameter uncertainties are associated with the parameter values used in the implemented assessment models, since data may be incomplete, cannot be measured accurately or are not available.

Uncertainties can further be classified as epistemic uncertainties, which principally can be reduced (e.g. compressive strength of concrete used as a liner in the disposal facility), and aleatoric/stochastic uncertainties, which are linked to random spatial and or temporal variability of a quantity that cannot be determined further with existing methods (e.g., the exact groundwater flow field and it's evolution over time).

Different ways to reduce uncertainties exist. Obviously, additional and more accurate data from R&D and site investigation programs lead to a reduction of uncertainties (NEA, 2013). In addition, changes in the repository design, e.g., simplifying the design of certain barriers or components, or more stringent waste acceptance criteria (WAC) resulting in a more narrow range of possible forms or conditions of the waste, may help in this regard. The development and use of alternative models, in particular on a conceptual level, enable one to study the importance of the underlying assumptions of different models (NEA, 2013). The consideration and application of alternative models requires a wide range of expertise and therefore, should be subjected to peer review (IAEA, 2002). It also helps increasing the fundamental understanding of important processes and features, which forms a link to the FEPS used in scenario development. Thus, safety assessments are in general closely interlinked and iteratively developed with R&D programs (NEA, 2013).

Persisting uncertainties need to be treated appropriately. If possible, the irrelevance of an uncertainty may be demonstrated by taking into consideration their very low probability or consequences for the safety assessment. Regarding uncertainties connected to the biosphere and the future behaviour of human beings, a stylised approach can be used to bound the hypothetical future habits to the present ones. For example, similar food consumption patterns as common today may be assumed for a 'reference man', which hypothetically is going to be exposed in some way to radionuclides originating from the waste disposal facility. Another often-used assumption is that today's radioprotection levels meet also the future ones and thus, can be applied as reference or compliance criteria. A bounding of uncertainties can be achieved by incorporating a certain level of conservatism into the models and data used in the assessments. A safety assessment being carried out under conservative assumption and whose safety indicators meet regulatory compliance criteria can be considered as robust (IAEA, 2002). It is important to bear in mind that the applied conservative assumptions need to be justified and meaningful in the studied context. In addition, the formulation of what-if scenarios makes it possible to study the system's evolution and the significance of its different components under conditions lying outside the expected range (either still physical possible but or not at all physical possible) in order to demonstrate the system's robustness (NEA, 2013). Especially data and parameter uncertainties may necessitate a large number of calculation or assessment cases. An assessment case can be run deterministically, that is, using single-valued parameters (e.g., when applying a conservative approach). Deterministic calculation are therefore useful when presenting safety assessment results to various stakeholders. Deterministic calculation are often performed in combination with probabilistic calculations, for which parameter values are randomly sampled from established probability functions to carry out an array of assessment cases for studying the relative importance of parameter choices and of parameter value combinations on the assessment output (sensitivity analysis). Probabilistic sensitivity analyses can further be useful to illustrate the influence of different design or material alternatives or of how certain aspects of the system are represented in the model (e.g. numerical abstractions like using a 1D or 2D axisymmetric geometry to represent transport phenomena that act three-dimensionally in nature). The results of probabilistic calculations can also provide input to deterministic in terms of parameter value choices (NEA, 2013).

Uncertainties of models used and model input have also been discussed in SGI (2017) considering landfill type of repositories in Sweden.

10.1.4 Complementary evidence and arguments

Besides the safety assessments and the sensitivity and uncertainty analyses discussed above, the robustness and reliability of the safety case can be enhanced by assessing the performance of certain (sub-)systems incorporating themselves sensitivity and uncertainty analyses and comparing the results to appropriate standards or criteria, which not necessarily stem from radiation safety considerations in the first place. In this way, the roles, performance and significance of different components of the disposal system can be evaluated throughout the assessment period (IAEA, 2002). However, the decisions related to the development of the disposal facility should not be solely based on comparisons of calculation results with compliance criteria but wider judgement and complementary quantitative and qualitative assessments should be employed (IAEA, 2014). In line with defence in depth principle (Section 3.1), multiple lines of reasoning and evidence should be used to build confidence in the assessments and the safety statement. These can be based on, for example (IAEA, 2002, 2003, 2012, 2014a; NEA, 2013):

- the evaluation of alternative conceptual models (Section 10.1.3);
- alternative assumptions concerning the performance of barriers and alternative parameter values (Section 10.1.3);
- scoping and bounding calculations (Section 10.1.3);
- simplified compliance calculations (especially at an early stage of the repository development);
- demonstration and feasibility studies of the engineered features of the facility;
- appropriate and effective of controls such as waste acceptance criteria (WAC);
- studies of appropriate natural and archaeological analogues;
- use of sound scientific and engineering reasoning;
- QA procedures ensuring the quality of safety assessments and the uncertainty management (e.g., peer review);
- An overall healthy safety culture of the implementer organization and contractors.

11. Discussion and conclusions

The near surface repository concept is commonly used around the world for disposal of short-lived VLLW, LLW and in some cases ILW. The name is used for repositories located close to ground surface, typically partly or completely embedded in the sediment layer between bedrock and ground surface or built totally over the ground surface. In some cases, repositories excavated in bedrock, relatively close to ground surface are also considered near surface repositories, but this type of repository was not within the scope of the present study. The type of the waste deposited (VLLW, LLW or ILW), design basis (e.g., local regulations) and site conditions (natural barriers at the site, location of the repository with respect to groundwater surface) define the repository type used, for example:

- Waste (VLLW) buried in a rift in the ground with some engineered barriers isolating the waste from the surrounding environment.
- Vault type repository (also used for LLW and ILW), typically located completely below the ground surface with concrete or other barriers isolating the vault from the surrounding groundwater, or
- Landfill type repository for VLLW located above the ground surface and groundwater table.

Finnish legislation defines that when the waste is generated during production of nuclear energy, the management of the waste is regulated based on the Nuclear Energy Act (990/1987). This includes operational waste from nuclear power plants, decommissioning waste and waste produced in mining of uranium and thorium. Other radioactive waste produced in Finland (e.g., NORM waste produced in industry and mining of other minerals than uranium and thorium) is regulated by the Radiation Act (859/2018).

The Radiation and Nuclear Safety Authority in Finland (STUK) defines in YVL Guide D.5 that only VLLW can be disposed in repositories constructed near surface and the LLW and ILW are to be deposited at intermediate depth in the bedrock. The principles given in STUK/Y/2018 including defence in depth and using consecutive and mutually complementary barriers are also applied for near surface repositories, but with a graded approach considering the importance of the design and barriers for safety. At the time of writing this report, guidelines for near surface repositories are being prepared by STUK. Based on preliminary information, the disposal into a near surface repository shall not be large scale disposal, the average concentration of significant radionuclides should not exceed 100 kBq/kg and the total activity of significant radionuclides should not exceed the values mentioned in Nuclear Energy Decree (6§). In general, applying a waste acceptance criteria is important in order to reduce the amount of waste deposited. The exact type of the waste deposited and its location shall also be recorded (traceability).

For radioactive (NORM) waste produced in mining of uranium and thorium, regulation is specified in STUK Y/5/2016. Considering other radioactive (NORM) waste regulated under the radiation act, notification to STUK and dose assessment is required and if certain reference levels are exceeded licencing by STUK is required. Thus the concept of near surface repository is not applied as such to radioactive (NORM) waste produced in these two cases, but some principles can be taken into account as guidelines when planning the deposition of these materials.

The Finnish site conditions to be taken into account in the design of a near surface repository are the relatively large variations in temperature, baseline radiation, annual precipitation and prevailing geological and hydraulic site conditions. During the winter, the formation of ground frost needs to be taken into account based on Finnish regulations and guidelines.

The annual precipitation at the site affects the amount of water infiltrated into the repository. As a consequence of climate change, the increase of annual precipitation (during winter months) and on the other hand extended dry seasons during the summer need to be considered in the design. There are currently no precise guidelines regarding how much water is allowed to infiltrate annually into the waste (for example at maximum 1–5% of the annual infiltration OR 1-50 L/m^2/a) and this limit will have an effect on the selection of barrier materials placed in the cover structure of the facility. Considering the principles of a multi-barrier system, a combination of a mineral sealing layer and a synthetic liner (e.g., bentonite mat or geotextile) is recommended as in the design for hazardous waste landfills in Finland. In further studies, simulating/calculating the infiltration amounts and times through different barrier material combinations is recommended, also considering different precipitation scenarios. This is needed to have a more realistic idea of how much water can infiltrate into the waste annually. This information can then be used as starting data in assessing the evolution of the waste (bio-degradation) and waste packages (e.g., corrosion) and in radionuclide transportation analysis for the safety case.

Dry conditions increase the risk of drying and cracking of mineral materials and can consequently lead to higher risk of erosion in the cracks when the drought ceases with a heavy rain fall. In addition, the grassy vegetation at the topmost layer with the function of decreasing surface erosion may dry out. In addition, the risk for ground fires increases. Therefore, fire safety and alarm systems during the operations and in the design is a matter to be considered.

Local geological conditions need to be considered in the repository design and in the radionuclide transportation analysis. Considering typical geological conditions in Finland, there is only some meters thick layer of glacial till over the bedrock and therefore also the ground water table is also close to the ground surface. At coastal regions, the rock surface may be exposed and there is no or little formation of groundwater in the area (precipitation is transferred mostly by surface runoff). The areas with higher overburden of sediments are typically groundwater areas (sandy sediments) or clay formations with poor bearing capacity. Both of these latter mentioned sediment types are recommended to be avoided as foundations for a near surface repository.

Considering that the waste to be deposited in Finland (VLLW), a landfill type of a repository can be considered as basis of the design. If the site has no or very little sediment overburden, then the importance of barrier material placed below the waste increases in the design. In general, the barriers used in the design can be similar to the ones used for hazardous waste landfills in Finland, but the optimal thickness and performance of specific layers (e.g., mineral sealing layers) is recommended to be studied further considering different annual precipitation scenarios. Alternatively, the design could be based on a vault type of a repository (concrete vault) with vertical cut-off barriers isolating the repository from the groundwater.

The engineered barriers with high importance to long-term safety include metallic waste packages, fill material placed around the waste/waste packages and barrier mineral, possible concrete structures, and synthetic or mineral based barrier materials with the main functions to: a) establish stable foundation for the deposited waste; b) limit the amount of precipitation infiltrating to the waste; and, c) limit dispersion of radionuclides into the environment.

Metallic containers have good mechanical properties and stability but they have a high risk of corrosion induced by different factors. Aggressive components originating either from the waste or from the surrounding environment, as well as pH, redox and temperature of the environment are important factors affecting corrosion of steel. However, microbes can also enhance corrosion and change the local environment to relatively more aggressive conditions.

Cementitious composites (concrete, mortars, grouts, etc.) are an extremely versatile material for barriers, with widely varying characteristics that make is possible to be used as a backfill and a structural material. As a cementitious backfill it can offer long-term containment by provision of a highly alkaline buffer capacity and good sorption capacity. The versatility of cement mixes is another important benefit, as they can be designed to satisfy a range of engineering and strength requirements.

As a structural material, concrete deterioration over time is a major concern and raises ageing management issues. Damage induced by deterioration processes can substantially reduce the designed service life of the structure. Failure to manage the maintenance of reinforced concrete may result in structural failure, and structural replacement is often not a possibility.

The mineral sealing/barrier materials can consist for example of mixture of bentonite and crushed rock. The optimal amount of the bentonite in the mixture depends on the targeted hydraulic conductivity for the layer (e.g., <$1x10^{-9}$…$1x10^{-12}$ m/s) and on the smectite content/type of the bentonite (Na^+ dominant or Ca^{2+}/Mg^{2+}). The prevailing site conditions, e.g., presence of high-pH concrete structures and chemical components in the water may lead to some deterioration of the swelling properties of the bentonite through cation exchange and, in extreme conditions, through mineral dissolution. It should be noted that for specific radionuclides, some other mineral barrier materials may have better sorption abilities than bentonite has (e.g., sorption of Caesium to illite). Therefore it can be stated that the optimisation of the mineral barrier materials can be done based on the targeted k-value, site specific conditions (natural barrier materials present) and the radionuclides present in the waste. Bentonite mats and synthetic liners can be used in near surface repositories, but combined with mineral sealing materials. This is due to long service life of the repository and the risk of local defect (e.g., puncture with a root) in the thin sealing layer.

The fill material placed inside/around the waste packages could be based on a mineral material (e.g., rock flour) or cement based material (e.g., shotcrete). The purpose of the fill material is to: a) provide stability and reduce uneven settlements in the repository so that the structures of the repository remain intact; and, b) function as barrier material with at least some sorption capacity for radionuclides. In practice, the installation method shall selected so that the voids between the waste packages can be filled effectively in a practical manner.

The uppermost layer of the repository has the following functions: a) shelter the underlying layers for freezing and thawing; b) control of surface runoff waters; c) limit infiltration of water into the repository (together with drainage materials); and, d) shelter the uppermost layer from erosion (with suitable vegetation).

The drainage systems can be based on coarse mineral materials and in some cases concrete material. The control and monitoring of especially leachate waters is important from the long-term safety point of view. The drainage system can consist of drainage layers above and below the waste. With a pipe collection system, the drainage water is collected and directed to treatment.

If significant amounts of gases are generated during the life-cycle of the repository, the discharge of the gases should be controlled so that the eruption of gases does not lead to deterioration of the barriers. If needed, a gas collection layer can be constructed above the waste and with a collection system the gas can be collected and directed for treatment.

Monitoring needs and requirements start in the early phase of planning of the repository and continue throughout the life cycle of the repository to the post-closure phase. Monitoring system planning and development should not be considered as a separate entity, but should be performed together with the whole repository planning and construction. Lessons learned in other industry domains should be also followed and best practices should be adopted. Subjects to be monitored in the NSDF include various environmental and site structure related quantities. During the long time cycle of the repository data management has to be maintained with care and during all site phases, measured historical data and documents need to be stored reliably and securely, as the operator of the site may change.

Safety case is a well-established concept for the demonstration of safety and communication with different stakeholders and serves as the basis for the licensing process. The generic methodology has been developed to serve all types of disposal facilities for radioactive wastes. However, particularities of near surface disposal facilities, such as the proximity to the biosphere, limitations on the inventory in terms of long-lived radionuclides and possible measures during the institutional control period (e.g., post-closure monitoring), require specific consideration. Internationally there is a profound experience with safety case and safety assessment of near surface disposal facilities of various designs adapted for different waste types and site conditions. The long experience in Finland concerning safety case and safety assessment gained from the LILW disposal facilities at intermediate depth in Loviisa and Olkiluoto as well as the development of the deep geological repository for spent nuclear fuel, provides a solid basis for the envisaged disposal facility for VLLW and/or LLW located above the ground surface or at shallow depths.

Uncertainties concerning the near surface repository concept

The main uncertainties linked to near surface repositories is the evolution of the repository, waste and engineered barriers during the expected life-time of the repository extending form tens of years up to hundreds of years and analysis linked to radionuclide migration in a near surface repository conditions. This conclusion is also supported by a study performed by SGI (2017) on near surface repositories in Sweden aiming to build knowledge about processes and factors controlling the release and dispersion of radioactive substances into environment from landfilled waste. Therefore, further studies are recommended both for studying the radionuclide transportation in critical barrier materials and evolution of the waste and waste packages, e.g., the effect of corrosion on the integrity of the waste packages.

Considering a landfill type of a design for near surface repository in Finland the main principles are the same as for the landfill for hazardous waste. The detailed design basis for the repository and engineered barriers are however recommended to be studied further considering conditions relevant in the life-time of the repository.

References

Abdel Rahman, R., Ibrahim, H. & Abdel Monem, N. (2009). Long-term performance of zeolite Na A-X blend as backfill material in near surface disposal vault. Chemical Engineering Journal. Volume 149, Issues 1–3, 1 July 2009, Pages 143–152.

Andersson, K., Torstenbelt, B. & Allard, B. (1983). Sorption of radionuclides in geologic systems. Department of Nuclear Chemistry, Chalmers University of Technology. Göteborg, Sweden. SKBF Technical Report, KBS 83-63.

Andra (2017). A new disposal facility for very-low-level radioactive waste, Brochure. www.andra.fr

Andra solutions (2017). Andra the French National Radioactive Waste Management Agency. Brochure.

Andrade, C., Martinez, I., Castellote, M. & Zuloaga, P. (2006), Some principles of service life calculation of reinforcements and in situ corrosion monitoring by sensors in the radioactive waste containers of El Cabril disposal (Spain). Journal of Nuclear Materials 358, 82–95.

Aronsson, D. (2019). Shallow land repository for very low level waste at Ringhals. KYT Near surface repositories in Finland seminar. 26.9.2019. Seminar presentation by Dan Aronsson, Vattenfall available at: http://kyt2022.vtt.fi/kyt2022_seminar_sept_2019.htm

Baeyens, B., Thoenen, T., Bradbury, M.H. & Marques Fernandes, M. (2014). Sorption Data Bases or Argillaceous Rocks and Bentonite for the Provisional Safety Analyses for SGT-E2. National Cooperative for the Disposal of Radioactive Waste, Wettingen, Switzerland. NAGRA Technical Report 2012-04.

Balmer, S., Kaufhold, S. & Dohrmann, R. (2017). Cement-bentonite-iron interactions on small scale tests for testing performance of bentonites as a barrier in hihg-level radioactive waste repository concepts. Applied Clay Science. 135 (2017), p. 427–436.

Bergström U., Pers K. & Almén Y. (2011). International perspective on repositories for low level waste, Report SKB R-11-16.

Britschgi, R., Rintala, J. & Puharinen, S-T. (2018). Pohjavesialueet – opas määrittämiseen, luokitukseen ja suojelusuunnitelmien laadintaan. Ympäristöministeriö, 2018, Helsinki. Ympäristöhallinnon ohjeita 3/2018. ISBN: 978-952-11-4818-7.

Christensen, T., Cossu, R. & Stegmann, R. (ed.) (1989). Sanitary Landfilling: Process, Technology and Environmental Impact. Academic Press Limited, London. ISBN 0-12-174-255-5. 592 pp.

Cole, I.S. & D. Marney. (2012). "The science of pipe corrosion: A review of the literature on the corrosion of ferrous metals in soils." *Corrosion Science 56* 5–16.

Cool, W., Vermariën, E., Wacquier, W. & Perko, J. (2013). The long-term safety and performance analyses of the surface disposal facility for the Belgian category A waste at Dessel. *In ASME 2013 15th International Conference on Environmental Remediation and Radioactive Waste Management*. American Society of Mechanical Engineers Digital Collection.

Cota, S., Hiromoto, G., Gharbieh, H., & Silva, A. (2018). Preliminary post-closure safety assessment for a borehole-type repository for disused sealed radioactive sources in Brazil. *Progress in Nuclear Energy, 103*, 74–80. https://doi.org/10.1016/j.pnucene.2017.11.005

Coughtrey, P. & Thorne, M. (1983). Radionuclide distribution and transport in terrestrial and aquatic ecosystems: a critical review of data. Volume 1. 6 volumes. A.A. Balkema; Rotterdam (Netherlands); ISBN 90-6191-278-4, 496 pp.

Craig, R. (2005). Craig's soil mechanics. 7th edition. Spon press, New York, USA. ISBN 0-415-32702-4. 447 pp.

Eckerberg, C. & O. Olsson. (2013). RD&D Programme 2013: Programme for research, development and demonstration of methods for the management and disposal of nuclear waste. Technical Report, TR-13-18, ISSN 1404-0344, Stockholm: Svensk Kärnbränslehantering AB.

Egloffstein, T.A. (2001). Natural bentonites – influence of the ion exchange and partial desiccation on permeability and self-healing capacity of bentonites used in GCLs. Geotextiles and geomembranes 19(7): 427−444.

Eronen, M. & Haila, H. (1981). The Highest Shoreline of the Baltic Sea in Finland. Florilegium Florinis Dedicatum Striae, pages 157–258.

Espivent, C. (2019). Safety revies of French Surface Disposal. KYT SURFACE seminar 26.9.2019. Presentation available publicly at: http://kyt2022.vtt.fi/kyt2022_seminar_sept_2019.htm

Farouki, O. (1992). European Foundation Designs for Seasonally Frozend Ground. U.S. Army Corps of Engineers. Cold Regions Research & Engineering Laboratory (CRREL). Monograph 92-1. 128 pages. Unclassified publication.

Francis, A.J., Cather*, R. & Crossland, I.G. (1997). Development of the Nirex Reference Vault Backfill. Report n: S/97/014. December 1997.65 p.

Gardemeister, R. (1975). On engineering geological properties of fine-grained sediments in Finland. Buiding Technology and Community Development, Publication 9. Technical Research Centre of Finland. 91 pages.

Haavisto-Hyvärinen, M. & Kutvonen, H. (2007). Maaperäkartan käyttöopas. Geologian tutkimuskeskus 2007. http://uusi.gtk.fi/export/sites/fi/tietopalvelut/kartat/GTK_maaperakartan_kayttoopas.pdf

Hakanen, M. & Ervanne, H., The Influence of Organic Cement Additives on Radionuclide Mobility - A Literature Survey. Working Report 2006-06. POSIVA. 2006. 42 p.

Han, K.W., Heinonen, J. & Bonne, A. (1997). Radioactive Waste Disposal: Global Experience and Challenges, IAEA Bull. 39 1 (1997) 33–41.

Hauser, W. & Koster, R. (1989). "Evaluation of long-term durability of low and intermediate level waste packagings." Proc. Symp. Waste Management '89, Vol. 2 (POST, R.G., Ed.),. Tucson, AZ: University of Arizona. 795–799.

Hölttä, P., Poteri, A., Hakanen, M. & Hautojärvi, A. (2004). Fracture flow and radionuclide transport in block-scale laboratory experiments.

IAEA (2001). Technical Considerations in the Design of Near Surface Disposal Facilities for Radioactive Waste, IAEA-TECDOC-1256, INTERNATIONAL ATOMIC ENERGY AGENCY, Vienna (2001).

IAEA (2002). Scientific and Technical Basis for the Near Surface Disposal of Low and Intermediate Level Waste, Technical Reports Series No. 412, INTERNATIONAL ATOMIC ENERGY AGENCY, Vienna (2002).

IAEA (2003). Considerations in the Development of Near Surface Repositories for Radioactive Waste, Technical Reports Series No. 417, INTERNATIONAL ATOMIC ENERGY AGENCY, Vienna (2003).

IAEA (2004). Derivation of Activity Limits for the Disposal of Radioactive Waste in Near Surface Disposal Facilities, IAEA-TECDOC-1380, INTERNATIONAL ATOMIC ENERGY AGENCY, Vienna (2004).

IAEA (2006a). Fundamental Safety Principles, IAEA Safety Fundamentals No. SF-1, INTERNATIONAL ATOMIC ENERGY AGENCY, Vienna (2006).

IAEA (2006b). Application of the Management System for Facilities and Activities, IAEA Standards Series No. GS-G-3.1, INTERNATIONAL ATOMIC ENERGY AGENCY, Vienna (2006).

IAEA (2009a). Classification of Radioactive Waste, IAEA Safety Standards Series No. GSG-1, INTERNATIONAL ATOMIC ENERGY AGENCY, Vienna (2009).

IAEA (2009b). Predisposal Management of Radioactive Waste, IAEA Safety Standards Series No. GSR Part 5, INTERNATIONAL ATOMIC ENERGY AGENCY, Vienna (2009).

IAEA (2009c). Disposal approaches for long lived low and intermediate level radioactive waste IAEA NUCLEAR ENERGY SERIES No. NW-T-1.20.

IAEA (2011a). Disposal of Radioactive Waste, IAEA Safety Standards Series No. SSR-5, INTERNATIONAL ATOMIC ENERGY AGENCY, Vienna (2011).

IAEA (2011b). Nuclear Security Recommendations on Physical Protection of Nuclear Material and Nuclear Facilities (INFCIRC/225/Revision 5), IAEA Nuclear Security Series No. 13, INTERNATIONAL ATOMIC ENERGY AGENCY, Vienna (2011).

IAEA (2012). The Safety Case and Safety Assessment for the Disposal of Radioactive Waste. IAEA Safety Guide SSG-23. IAEA, Vienna (September 2012). ISBN 978–92–0–128310–8. 120 pages.

IAEA (2014). Near Surface Disposal Facilities for Radioactive Waste, IAEA Specific Safety Guide No. SSG-29, INTERNATIONAL ATOMIC ENERGY AGENCY, Vienna (2014).

IAEA (2016). Leadership and Management for Safety, IAEA General Safety Requirements No. GSR Part 2, INTERNATIONAL ATOMIC ENERGY AGENCY, Vienna (2016).

IAEA (2017a). Contents and Sample Arguments of a Safety Case for Near Surface Disposal of Radioactive waste. IAEA TECDOC SERIES, IAEA-TECDOC-1814. INTERNATIONAL ATOMIC ENERGY AGENCY, Vienna (2017).

IAEA (2017b). Selection of Technical Solutions for the Management of Radioactive Waste. IAEA TECDOC SERIES, IAEA-TECDOC-1817. INTERNATIONAL ATOMIC ENERGY AGENCY, Vienna (2017).

IAEA (2017c). Handbook on Ageing Management for Nuclear Power Plants, Nuclear Energy Series No. NP-T-3.24, IAEA, Vienna.

Ismail, A.I.M. & El-Shamy, A.M. (2009). "Engineering behaviour of soil materials on the corrosion of mild steel." *Applied Clay Science 42* 356–362.

Johannesson, L-E., Börgesson, L. & Sandén, T. (1999). Äspö Hard Rock Laboratory. Backfill material based on crushed rock (part 2). Geotechnical properties determined in Laboratory. Svensk Kärnbränslehantering AB. International Progress Report: SKB IPR-99-23.

Kallio, A. (2019). Implementation of the European Basic Safety Standards - what does it mean for radon and NORM. CORES symposium, 3-4.9.2019. STUK. Available at: https://www.stuk.fi/web/en/about-us/cooperation/cores-finnish-consortium-for-radiation-safety-research/events/cores-symposium-on-radon-and-norm-3-4-sep-2019

Laikari, A., Flak, J., Koskinen, A. & Häkli, J. (2018). Wireless in Nuclear: Feasibility study. Liekki 2: Poltto- ja kaasutustekniikan tutkimusohjelma, no. 2018:513).

Larsson, A. (1989). "State-of-the-art report on radioactive waste disposal. Different types of radioactive waste." IAEA Bulletin 4 18–25.

Lee, S. & Kim, J. (2017). Post-closure safety assessment of near surface disposal facilities for disused sealed radioactive sources. *Nuclear Engineering and Design, 313*, 425–436.

Leppäranta, M., Virta, J. & Huttula T. (2017). Hydrologian perusteet. Helsingin Yliopisto, Fyysikan laitos (2017). ISBN 978-952-68754-0-8. 223 pages. Available at: http://hdl.handle.net/10138/241220

Liikennevirasto, 2018. Ratatekniset ohjeet (RATO) osa 3. Radan rakenne. Liikenneviraston ohjeita, 13/2018. Liikennevirasto, Helsinki 2018. ISBN 978-952-317-534-1.Lintinen, P. 1995. Origin and physical characteristics of till fines in Finland. Geological Survey of Finland, Bulletin 379. Espoo. 83 pages.

Lintinen, P. (1995). Origin and physical characteristics of till fines in Finland. Geological Survey of Finland, Bulletin 379. Espoo. 83 pages.

Monastra, V. & Grandstaff. E. (1999). Kinetics of MgO dissolution and buffering of fluids in the Waste Isolation Pilot Plant (WIPP) repository. Materials Research Society Symposium - Proceedings, volume 556, pages 625–632.

Muikku, M., Bly, R., Kurttio, P., Lahtinen, J., Lehtinen, M., Siiskonen, T., Turtiainen, T., Valmari, T. & Vesterbacka, K. (2014). Suomalaisten keskimääräinen efektiivinen annos. Annoskakku 2012.Säteilyturvakeskus. STUK-A259 / MAALISKUU 2014. ISBN 978-952-478-981-3 (pdf). 57 pp.

Nair, R.N. & Krishnamoorthy, T.M. (1999). Probabilistic safety assessment model for near surface radioactive waste disposal facilities. *Environmental Modelling & Software, 14*(5), 447–-460.

Naturvårdverket (2004). Deponering av afvall. Handbok 2004:02 av allmänna råd till förordningen (2001:512) om deponering av avfall och till 15. kap. 34 § Miljöbalken (1998/808). Available at: https://www.naturvardsverket.se/Om-Naturvardsverket/Publikationer/ISBN/0100/91-620-0134-5/

NEA (2013). The Nature and Purpose of the Post-Closure safety Cases for Geological Repositories, “Safety Case Brochure 2012”, NEA/RWM/R(2013)1, OECD/NEA, Paris. https://www.oecd-nea.org/rwm/reports/2013/78121-rwn-sc-brochure.pdf

NEA (2016). The Safety Culture of an Effective Nuclear Regulatory Body, Regulatory Guidance Report 18, OECD/NEA, Paris. doi.org/10.1787/19900791

Niemi J. (ed.), (2009). Ympäristön seuranta Suomessa 2009-2012. Suomen Ympäristökeskus. Ympäristö 11/2009. ISBN 978-952-11-3413-5 (PDF). 155 pp.

Nironen, M. (2017). Guide to the Geological Map of Finland – Bedrock 1:1 000 000. Geological Survey of Finland, Special Paper 60, 41-76. 2017.

Parjanne, A. & Huokuna, M. (2014). Tulviin varautuminen rakentamisessa. Opas alimien rakentamiskorkeuksien määrittämiseksi ranta-alueilla. Ympäristöopas 2014. Suomen ympäristökeskus, Ilmatieteen laitos, Ympäristöministeriö, Maa- ja metsätalousministeriö. Helsinki, 2015. https://helda.helsinki.fi/bitstream/handle/10138/135189/YO_2014.pdf?sequence=1&isAllowed=y

Pirinen, P., Simola, H., Aalto, J., Kaukoranta, J-P., Karllson, P. & Ruuhela, R. (2012). Tilastoja Suomen Ilmastosta 1981-2010. Ilmatieteenlaitos 2012. Raportteja 2012:1. 551.506.3 (408), Helsinki 2012. ISBN 978-951-697-766-2 (pdf). 96 pp.

Posiva Oy (2016). Nuclear Waste Management of the Olkiluoto and Loviisa Power Plants, Summary of Operation in 2012. http://www.posiva.fi/files/3197/YJH_2012_Engl_LOW.pdf, Posiva Oy.

Poteri, A., Nordman, H., Pulkkanen, V-M. & Smith, P. (2014). Radionuclide Transport in the Repository Near-Field and Far-Field. Posiva Oy, Olkiluoto. Posiva 2014-02.

RIL (2013). Routasuojaus - rakennukset ja infrarakenteet. RIL 261 - 2013. ISBN: 978-951-758-547-7. Suomen Rakennusinsinöörien Liitto RIL ry, Helsinki. 257 pp.

Routamo, T. (2019). Ydinturvallisuusvaatimukset ja ydinjätehuollon vaatimukset Euroopan Unionissa ja IAEA:ssa. Ydinturvallisuuskurssi YJK17, luento 1.4.

SGI (2017). Spridning av radioaktiva ämnen från olika typer av avfall och förvar – modeller, ämnesdata och utvecklingsbehov. SGI - Swedish geotechnical Institute, September 2017. SSM2016-2218.

SKB (2006). Data report for the safety assessment SR-Can. Svensk Kärnbränslehantering AB, Stockholmn. Technical Report, TR-06-25.

Soulet, S., & Griffault, L. (2016). The Safety Case of Andra’s Low and Intermediate Level, Short Lived Radioactive Waste Disposal Facility in the Aube District (CSA) (IAEA-CN--242). International Atomic Energy Agency (IAEA)

Soveri, U. (1956). The mineralogical composition of argillaceous sediments in Finland. Annales Academiae Scientiarum Fennicae, Series A III, No. 48. 32 pages.

Soveri, J. & Varjo M. (1977). Roudan muodostumisesta ja esiintymisestä Suomessa vuosina 1955-1975. Vesientutkimuslaitoksen julkaisuja 20. 60 pp.

SSM (2013). Omprövning av tillstånd till markförvar för lågaktivt avfall vid Forsmarksanläggningen. Strålsäkerhetsmyndigheten. Besult: datum 2013-11-14, reference EXT-2010-0814, diarenr: SSM 2010/721, Dokumentnr: SSM 2010/721-54. Handläggaren: Anders Wiebert.

STUK (2016). Radiation and Nuclear Safety Authority Regulation on the Safety of Mining and Milling Operations Aimed at Producing Uranium or Thorium. Regulation STUK Y/5/2016. Säteilyturvakeskus, Helsinki.

STUK (2018). Radiation and Nuclear Safety Authority Regulation on the Safety of Disposal of Nuclear Waste. Regulation STUK Y/4/2018. Säteilyturvakeskus, Helsinki.

STUK (2019). STUK's Perspective on Near Surface Repositories. KYT Near Surface Repositories in Finland seminar. 26.9.2019. Presentation available at: http://kyt2022.vtt.fi/kyt2022_seminar_sept_2019.htm

Svemar, C., Johannesson, L,-E., Grahm, P., Svensson, D., Kristensson, O., Lönnqvist, M. & Nilsson, U. (2016). Prototype repository. Opening and retrieval of outer section of Prototype Repository at Äspö Hard Rock Laboratory. Summary Report. SKB Technical Report, TR 13-22. Stockholm, Sweden: Svensk Kärnbränslehantering AB.

SYKE (2002). Kaatopaikan tiivistysrakenteet. Ympäristöopas 36. ISBN: 952-11-0232-2. Suomen Ympäristökeskus, Helsinki. 145 pp.

SYKE (2008). Kaatopaikkojen käytöstä poistaminen ja jälkihoito. Ympäristöhallinnon ohjeita 1/2008. ISBN 978-952-11-3151-6. Suomen ympäristökeskus, Helsinki. 158 pp.

Söderlund, M., Lusa, M., Virtanen, S., Välimaa, I., Hakanen, M., Lehto, J. & Lahdenperä, M. (2013). Distribution Coefficients of Caesium, Chloride, Iodine, Niobium, Selenium, Technetium on Olkiluoto Soils. Posiva Oy, Olkiluotoo. Posiva Working Report WR 2013-68.

Tammirinne, M., Juvankoski, M., Laaksonen, R. & Rathmayer, H. (2004). Pohjavedensuojausrakenteiden käyttöikämitoitus ja tuotehyväksyntä. VTT Technical Research Centre of Finland. 256 pp. ISBN 952-5004-50-3.

Tison, J. (2009). 40 years of operation of near surface repositories. Andra experiences. CEG Workshop, 24-26.2. 2009 Bommersvik.

Tuomenvirta H., Haavisto R., Hildén M., Lanki T., Luhtala S., Meriläinen P., Mäkinen K., Parjanne A., Peltonen-Sainio P., Pilli-Sihvola K., Pöyry J., Sorvali J. & Veijalainen N. (2018). Weather and Climate Risks in Finland - National Assessment. Publications of the Government´s analysis, assessment and research activities 43/2018. September 2018.
http://julkaisut.valtioneuvosto.fi/bitstream/handle/10024/161015/43-2018-Saa%20ja%20ilmastoriskit%20Suomessa.pdf?sequence=1&isAllowed=y

Usher, S. & Rossiter, D. (2019). UK Lessons Learned in Managing LLW

Vattenfall (2008). Miljöprövning av markförvar is Svalören. Huvudförhandling, 10.11.2018.

Vieno, T., & Nordman, H. (1999). Safety assessment of spent fuel disposal in Hästholmen, Kivetty, Olkiluoto and Romuvaara. TILA-99. Posiva Oy, Helsinki. Posiva report 99-07. ISBN 951-652-062-6.

Vikman, M., Marjamaa, K., Nykyri, M., Small, J.S., Miettinen, H., Heikinheimo, L., Haavisto, T. & Itävaara, M. (2019). "The biogeochemistry of gas generation from low-level nuclear waste: Microbiological characterization during 18 years study under in situ conditions." *Applied Geochemistry, Volume 105* 55-67.

Vilks, P. & Yang, T. (2018). Sorption of Selected Radionuclides on Sedimentary Rocks in Saline Conditions – Updated Sorption Values. Nuclear Waste Management Organisation, Canada. NWMO-TR-2018-03.

VNp 331/2013. Valtioneuvoston asetus kaatopaikoista. Ministry of the Environment (Finland). 2.5.2013. Retrieved from https://www.finlex.fi/fi/laki/ajantasa/2013/20130331.

Watts, L. (2002). Development of the Post-Closure Safety Case for the Low Level Waste Disposal Site at Drigg, United Kingdom. MRS Proceedings, 713, JJ7.2. doi:10.1557/PROC-713-JJ7.2

Wacquier, W., & Cool, W. (2014). The safety case in support of the license application of the surface repository of low-level waste in Dessel, Belgium (NEA-RWM-R--2013-9). In *The Safety Case for Deep Geological Disposal of Radioactive Waste: 2013 State of the Art. Symposium Proceedings, 7-9 October 2013, Paris, France*. Nuclear Energy Agency of the OECD (NEA).

WENRA (2014). Radioactive Waste Disposal Facilities Safety Reference Levels. WENRA, (December, 2014). 81 pages. http://www.wenra.org/media/filer_public/2015/03/18/srl_disposal_final_version_2014_12_22.pdf

Won-Seok, K, Sangsoo, H., Ahn, J. & Wooyoung, U. (2019). Investigation of 3 H, 99Tc, and 90Sr transport in fractured rock and the effects of fracture-filling/coating material at LILW disposal facility. Environ Geochem Health (2019) 41:411–425 https://doi.org/10.1007/s10653-018-0123-y

Zuloaga, P. (2009). El Cabril low and intermediate level radioactive waste disposal facility, presentation, Salt lake city.

APPENDIX 1 Levels for the general clearance of unlimited amounts of material (Annex A in YVL D.4)

Radionuclide	Activity concentration (Bq/g)
H-3	100
Be-7	10
C-14	1
F-18 *	10
Na-22	0.1
Na-24*	1
Si-31	1000
P-32	1000
P-33	1000
S-35	100
Cl-36	1
Cl-38*	10
K-42	100
K-43 *	10
Ca-45	100
Ca-47	10
Sc-46	0.1
Sc-47	100
Sc-48	1
V-48	1
Cr-51	100
Mn-51*	10
Mn-52	1
Mn-52m *	10
Mn-53	100
Mn-54	0.1
Mn-56*	10
Fe-52 *	10
Fe-55	1000
Fe-59	1
Co-55 *	10
Co-56	0.1
Co-57	1
Co-58	1
Co-58m*	10000
Co-60	0.1
Co-60m*	1000
Co-61*	100
Co-62m*	10
Ni-59	100
Ni-63	100
Ni-65*	10
Cu-64*	100
Zn-65	0.1
Zn-69*	1000
Zn-69m*	10
Ga-72*	10
Ge-71	10000
As-73	1000
As-74*	10
As-76*	10
As-77	1000
Mo-101*	10
Tc-96	1
Tc-96m*	1000
Tc-97	10
Tc-97m	100
Tc-99	1
Tc-99m*	100
Ru-97	10
Ru-103	1
Ru-105*	10
Ru-106	0.1
Rh-103m*	10000
Rh-105	100
Pd-103	1000
Pd-109	100
Ag-105	1
Ag-110m	0.1
Ag-111	100
Cd-109	1
Cd-115	10
Cd-115m	100
In-111	10
In-113m*	100
In-114m	10
In-115m*	100
Sn-113	1
Se-75	1
Br-82	1
Rb-86	100
Sr-85	1
Sr-85m*	100
Sr-87m*	100
Sr-89	1000
Sr-90	1
Sr-91*	10
Sr-92*	10
Y-90	1000
Y-91	100
Y-91m*	100
Y-92*	100
Y-93*	100
Zr-93*	10
Zr-95	1
Zr-97*	10
Nb-93m	10
Nb-94	0.1
Nb-95	1
Nb-97*	10
Nb-98*	10
Mo-90*	10
Mo-93	10
Mo-99	10
Sn-125	10
Sb-122	10
Sb-124	1
Sb-125	0.1
Te-123m	1
Te-125m	1000
Te-127	1000
Te-127m	10
Te-129*	100
Te-129m	10
Te-131*	100
Te-131m	10
Te-132	1
Te-133*	10
Te-133m*	10
Te-134*	10
I-123	100
I-125	100
I-126	10

* indicates nuclides with half-lives less than one day

Radionuclide	Activity concentration (Bq/g)
I-129	0.01
I-130*	10
I-131	10
I-132*	10
I-133*	10
I-134*	10
I-135*	10
Cs-129	10
Cs-131	1000
Cs-132	10
Cs-134	0.1
Cs-134m*	1000
Cs-135	100
Cs-136	1
Cs-137	0.1
Cs-138*	10
Ba-131	10
Ba-140	1
La-140	1
Ce-139	1
Ce-141	100
Ce-143	10
Ce-144	10
Pr-142*	100
Pr-143	1000
Nd-147	100
Nd-149*	100
Pm-147	1000
Pm-149	1000
Sm-151	1000
Sm-153	100
Eu-152	0.1
Eu-152m*	100
Eu-154	0.1
Eu-155	1
Gd-153	10
Gd-159*	100
Tb-160	1
Dy-165*	1000
Dy-166	100
Ho-166	100
Er-169	1000
Er-171*	100
Tm-170	100
Tm-171	1000

Radionuclide	Activity concentration (Bq/g)
Yb-175	100
Lu-177	100
Hf-181	1
Ta-182	0.1
W-181	10
W-185	1000
W-187	10
Re-186	1000
Re-188*	100
Os-185	1
Os-191	100
Os-191m*	1000
Os-193	100
Ir-190	1
Ir-192	1
Ir-194*	100
Pt-191	10
Pt-193m	1000
Pt-197*	1000
Pt-197m*	100
Au-198	10
Au-199	100
Hg-197	100
Hg-197m	100
Hg-203	10
Tl-200	10
Tl-201	100
Tl-202	10
Tl-204	1
Pb-203	10
Bi-206	1
Bi-207	0.1
Po-203*	10
Po-205*	10
Po-207*	10
At-211	1000
Ra-225	10
Ra-227	100
Th-226	1000
Th-229	0.1
Pa-230	10
Pa-233	10
U-230	10
U-231	100
U-232	0.1

Radionuclide	Activity concentration (Bq/g)
U-233	1
U-236	10
U-237	100
U-239*	100
U-240*	100
Np-237	1
Np-239	100
Np-240*	10
Pu-234*	100
Pu-235*	100
Pu-236	1
Pu-237	100
Pu-238	0.1
Pu-239	0.1
Pu-240	0.1
Pu-241	10
Pu-242	0.1
Pu-243*	1000
Pu-244	0.1
Am-241	0.1
Am-242*	1000
Am-242m	0.1
Am-243	0.1
Cm-242	10
Cm-243	1
Cm-244	1
Cm-245	0.1
Cm-246	0.1
Cm-247	0.1
Cm-248	0.1
Bk-249	100
Cf-246	1000
Cf-248	1
Cf-249	0.1
Cf-250	1
Cf-251	0.1
Cf-252	1
Cf-253	100
Cf-254	1
Es-253	100
Es-254	0.1
Es-254m	10
Fm-254*	10000
Fm-255*	100

APPENDIX 2 Levels for the general clearance of limited amounts of material (Annex B YVL D.4)

B01. The clearance of waste for disposal in a public landfill is subject to the activity concentration levels specified in the table below that may not be exceeded in respect of the activity concentration of any nuclide averaged over a maximum of 500 kg of waste. Additionally, the activity of any nuclide in any single item or waste package with a mass of less than 30 kg may not exceed the value obtained by multiplying the respective activity concentration limits by 30,000 g.

B02. The clearance of bulky metal objects for recycling is subject to the surface activity contamination levels specified in the table below that may not be exceeded in respect of the surface activity contamination of any nuclide averaged over a maximum of 0.1 m^2 of accessible surface area.

B03. When the levels specified in the table are applied to several nuclide groups, due account shall be taken of the fact that the sum of the ratios between nuclide group specific activities and the respective maximum values shall be less than one. An equivalent rule also applies to surface activity contaminations and the levels imposed on them.

Table. The nuclide group specific activity concentration and surface activity contamination levels applicable to the clearance of waste (a maximum for 100 tonnes for a single nuclear facility per year).

Nuclide group	Activity concentration	Surface activity contamination
Alpha emitters	0.1 Bq/g	0.4 Bq/cm^2
Strong gamma and beta emitters*	1 Bq/g	4 Bq/cm^2
Weak gamma and beta emitters**	10 Bq/g	40 Bq/cm^2

* For example ^{54}Mn, ^{58}Co, ^{60}Co, ^{65}Zn, ^{90}Sr, ^{106}Ru, ^{110m}Ag, ^{124}Sb, ^{125}Sb, ^{134}Cs, ^{137}Cs, ^{144}Ce and nuclides having similar radiation emission energy

** For example ^{3}H, ^{14}C, ^{51}Cr, ^{55}Fe, ^{63}Ni and nuclides having similar radiation emission energy