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DESIGN PRINCIPLES AND APPROACHES FOR RADIOACTIVE WASTE REPOSITORIES

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The Agency's Statute was approved on 23 October 1956 by the Conference on the Statute of the IAEA held at United Nations Headquarters, New York; it entered into force on 29 July 1957. The Headquarters of the Agency are situated in Vienna. Its principal objective is "to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world''.

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DESIGN PRINCIPLES AND APPROACHES FOR RADIOACTIVE WASTE REPOSITORIES

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FOREWORD

The IAEA's statutory role is to "seek to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world". Among other functions, the Agency is authorized to "foster the exchange of scientific and technical information on peaceful uses of atomic energy". One way this is achieved is through a range of technical publications including the IAEA Nuclear Energy Series.

The IAEA Nuclear Energy Series comprises publications designed to further the use of nuclear technologies in support of sustainable development, to advance nuclear science and technology, catalyse innovation and build capacity to support the existing and expanded use of nuclear power and nuclear science applications. The publications include information covering all policy, technological and management aspects of the definition and implementation of activities involving the peaceful use of nuclear technology.

The IAEA safety standards establish fundamental principles, requirements and recommendations to ensure nuclear safety and serve as a global reference for protecting people and the environment from harmful effects of ionizing radiation.

When IAEA Nuclear Energy Series publications address safety, it is ensured that the IAEA safety standards are referred to as the current boundary conditions for the application of nuclear technology.

This publication provides Member States with an overview of repository design principles and approaches that may be used to address their radioactive waste disposal needs. Furthermore, it describes a range of well studied disposal concepts that have either been successfully implemented or developed to an advanced stage of design. Examples of potential design solutions are provided for both near surface disposal facilities and geological repositories situated at various depths. Near surface facilities, suitable for the disposal of very low and low level waste, include trenches, vaults, shafts and direct access silos, as well as natural and engineered subsurface structures such as caverns, drifts and tunnels. Geological repository concepts, suitable for the disposal of intermediate level and high level waste (including spent nuclear fuel when declared as waste), mainly comprise mined disposal facilities situated at various depths and in a range of rock formations. They typically comprise access tunnels, shafts or both, as well as waste deposition tunnels, chambers and vaults. They may also include shallow boreholes and silos constructed within such engineered features. Alternate disposal options are also discussed describing solutions that rely on the conversion of existing facilities such as mines or other underground openings. The potential for radioactive waste disposal in boreholes, including the use of a very deep borehole concept, is also considered.

This IAEA publication was developed specifically to assist Member States in meeting their radioactive waste disposal obligations. Radioactive waste requiring disposal is generated from many different activities in Member States, including from the production of energy, research, health care and various industrial activities.

The IAEA officers responsible for this publication were G.H. Nieder-Westermann and J. Faltejsek of the Division of Nuclear Fuel Cycle and Waste Technology.

CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

Radioactive waste in various forms is generated in industrial, research and medical facilities worldwide. For those countries with nuclear power programmes, these wastes are predominantly derived from energy production in nuclear power plants. Almost all countries produce smaller quantities of radioactive waste from research activities and medical and industrial applications. Each IAEA Member State thus possesses different types of radioactive waste in varying quantities that will eventually require disposal, in conditioned, solid form, in one or more purpose designed disposal facilities.

The path to disposal for each type of waste will depend upon its inventory of radionuclides, its physical and chemical form, its quantity, and other waste specific characteristics. Depending on the category of waste, quantities within a national inventory can range from a few cubic metres to hundreds of thousands. The radiological hazard will also vary, depending on the nature and amounts of radionuclides associated with each type of waste. Radioactive waste can remain hazardous to human health and the environment for a period lasting from a few decades to many thousands of years, depending on the radionuclides of concern and their concentrations. Unless the quantity is considered below defined clearance levels or exempted by national law based on very low radiation levels, all radioactive waste is considered hazardous, no matter the actual quantity, and is therefore subject to controlled and regulated disposal. The Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management [1] requires that each Contracting Party to the Convention provide assurance that "society and the environment are protected from harmful effects of ionizing radiation, now and in the future…" The recognized means by which assurance is provided is the safe management of radioactive waste through disposal in a purpose built repository. Member States are ultimately responsible for providing adequate resources and ensuring their efficient use to manage radioactive waste from initial generation to eventual disposal.

Significant experience in the disposal of radioactive waste has been gained over the last several decades. Numerous facilities have been constructed and operated in many Member States and disposal solutions for many waste forms and classes are available. Based on this and other experience, a Member State beginning the development of a new disposal facility will need to go through a design process that produces an optimized solution for the inventory under consideration. This publication is intended to provide such Member States with an overview of possible repository design principles and approaches that can be implemented. In addition, this publication describes concepts and existing facilities to provide illustrations of possible disposal solutions for consideration.

The number of Member States with plans to construct a radioactive waste repository is increasing. Decisions regarding the selection of an appropriate repository for design and implementation are important for the entire national radioactive waste management system. In particular, these can affect upstream consideration of how the waste will be managed from the point of its generation. Although each national repository project has unique features, it is possible to identify a range of basic design solutions suitable for disposal of all classes of radioactive waste.

This publication describes the range of waste types that can be disposed of in a variety of facilities, based on the intrinsic characteristics of the wastes. For example, near surface disposal of low level waste has progressed in several countries for over 50 years [2]. Several such facilities that exist worldwide have already applied experiences and lessons learned from antecedent facilities. Waste with higher levels of radioactivity has been routed for geological disposal in a number of different geological settings. Member States are encouraged to continue to take advantage of previous international work and build upon designs that have been demonstrated to function, within the constraints of their national policies. Examples of more recently established disposal facilities illustrate the application of previous experience and lessons learned from older facilities. These practices enrich the knowledge base for the disposal of nuclear waste and will serve as examples in this publication.

1.2. SCOPE

This publication provides an overview of design principles and approaches that have either already been fully implemented, or are in the implementation phase, in several Member States. Examples of mature designs are provided for a wide range of radioactive waste types and geological settings. Potential repository solutions are based on the characteristics of the waste, such as volume and radiotoxicity, and available conceptual disposal options. The approach presented is based on fundamental safety principles [3, 4] and uses a systems engineering, requirements driven design approach that can be considered a primer for the design of radioactive waste disposal facilities.

The design approach is staged. From the earliest conceptual designs to the final as-built state, each design stage increases the technical basis and refines the repository configuration and disposal concept. Each stage can be iterative, and the output from each stage provides input for the next. The stages of design comprise an evolving programme. If guiding design principles are applied along the way, the implementer will increase the likelihood of accomplishing a successful repository project.

As well as the guidance on good practices relevant to radioactive waste disposal presented in this publication, the IAEA is considering providing further guidance with a focus on the technical, scientific and programmatic aspects of implementing a disposal solution. Future publications with a technical and scientific focus could reflect international experience with the management of site investigations for a disposal facility; provide an overview of past experiments conducted in underground research facilities around the world, as a scientific and technical basis to developing a geological disposal system; discuss the engineering and technical specifications of a borehole disposal concept for disused sealed radioactive sources; and explore disposal concepts showing a potential for the safe and effective disposal of small waste inventories. Other future publications could discuss some of the wider programmatic considerations needed for successful disposal implementation, reflecting international experience with cost estimation methods and funding schemes for a disposal programme; with communication and stakeholder involvement in radioactive waste disposal; with the experiences of local stakeholders with radioactive waste management programmes; with setting up a waste management organization (WMO) with responsibility for implementing a disposal programme; and with a roadmap for implementing a geological disposal programme.

It is likely that future guidance will, like this publication, refer to or be based on a range of examples and practices that have contributed to the progress of the successful implementation of disposal programmes in Member States. As such, they will be intended to transfer good practices and lessons learned, to incite further developments and implementation as needed by national programmes. It is hoped in general that the transfer of good practices will be used to inform national developments within the framework of the corresponding national legal and regulatory framework, and within relevant international conventions.

In addition to the examples and good practices of other States, to guide the development of safe solutions and to align with the internationally agreed high level of safety, the relevant safety standards should be consulted when developing and implementing a disposal facility. Among the broader range of relevant safety standards, particular attention should be paid to IAEA Safety Standards Series Nos SSR-5, Disposal of Radioactive Waste [5], SSG-1, Borehole Disposal Facilities for Radioactive Waste [6], SSG-29, Near Surface Disposal Facilities for Radioactive Waste [7], SSG-14, Geological Disposal Facilities for Radioactive Waste [8], SSG-23, The Safety Case and Safety Assessment for the Disposal of Radioactive Waste [9], as well as GSG-1, Classification of Radioactive Waste [10].

1.3. OBJECTIVE

The objective of this report is to present guiding principles for repository design and to provide a general description of a staged approach to design, using widely accepted terminology. This includes a timeline that describes stages of design and correlated events and processes, such as siting, licensing, construction, operation and closure. The report provides examples of disposal concepts that have been designed, and in many cases implemented, for a wide range of existing radioactive waste inventories. Selected examples demonstrate how combinations of waste inventories, geological settings and concepts of operations have been developed. Guidance provided here, describing good practices, represents expert opinion but does not constitute recommendations made on the basis of a consensus of Member States.

1.4. STRUCTURE

Section 2 of this report provides background information on the design process and how it is based upon definitions of the waste inventory and waste characteristics, consideration of available options and the definition of a repository programme life cycle.

Section 3 presents the guiding principles and framework for an iterative, requirements driven, systems engineering design process, which moves from a conceptual stage, through optioneering to a final design and eventual siting, licensing, construction and, ultimately, closure.

Section 4 presents examples of designs that have been implemented or evaluated in detail over the full range of radioactive waste classes (see GSG-1 [10]) and in all recognized siting environments and host formations. These examples provide viable starting points for new disposal programmes.

2. THE REPOSITORY DESIGN PROCESS

2.1. BASIC CONSIDERATIONS

Implementation of a repository programme is an integral part of radioactive waste management, which is guided by a defined national waste policy and carried out within a national legal and regulatory framework. The IAEA has developed guidelines in Policies and Strategies for Radioactive Waste Management [11] to assist Member States in establishing a suitable national radioactive waste management programme.

A regulatory framework sets safety goals and defines requirements for demonstrating compliance. Because the timescale of implementation can range from a few decades to more than a century, depending on the waste type and the complexity of the disposal solution, the framework has to clearly define the roles and responsibilities required for implementation and regulation [12]. These include specification of requirements placed on waste producers, definition of the waste inventory, establishment of independent regulatory oversight and foundation or designation of an implementing organization. The implementer is the organization that is assigned overall responsibility for the repository programme and might be a government agency or other devolved authorized entity. The implementer manages and executes the programme in compliance with regulatory requirements stipulated by the governing authority, under the independent oversight of a regulator. Adequate funding mechanisms over the full life cycle of the programme are essential. The IAEA is considering developing guidance on costing methods and funding schemes for radioactive waste disposal programmes and presenting a methodology for estimating the costs associated with developing a repository.

The repository programme will consist of a series of steps. At the beginning of the programme, the waste inventory for disposal is defined, potential generic (i.e. not site specific) repository solutions are identified and development of a generic safety case is initiated. A safety case is a “collection of arguments and evidence in support of the *safety* of a *facility or activity*” [13]. The iterative process, scope and content of a safety case are described in SSG-23 [9]. Following guidelines set out in national laws and regulations, one or more potential sites can then be considered, in conjunction with the identification and development of the most appropriate repository concept or concepts. The repository programme (see Section 3) will continue to site investigation and suitability determination, design development for the chosen repository concept and licensing, construction, operation and, finally, decommissioning and closure, followed by a period of institutional control or monitoring, as required.

Isolation and containment for as long as waste presents a potential hazard are the main safety functions provided by a repository. Isolation and containment involve using a series of passive barriers that work in concert, together capable of preventing or limiting the release of radionuclides (see Section 3.1.2). These include both the natural barriers afforded by the repository site and purpose designed, engineered barriers. The repository design integrates engineered barriers with natural barriers into an overall safety concept.

Natural barriers are provided by the characteristics of the environmental and geological setting selected for the disposal facility, while engineered barriers are components of the repository system specifically designed and constructed to enable safe waste emplacement and facility operations, including closure, and to inhibit movement of radionuclides after closure. Engineered barriers can include waste packages, backfill materials, multilayer covers, liners or sealing systems and other constructed design elements. In addition to waste isolation, the repository design has to account for worker, public and environmental protection during operations. Demonstration that defined safety goals for operations and long term, post-closure waste isolation performance are met is achieved using safety assessment methodology.

A safety assessment is a multidisciplinary analytical exercise that evaluates repository performance during construction and disposal operations and after closure. Safety is evaluated against specific criteria, typically including radiological dose limits or risks for workers and members of the public. Safety assessment is repeated at each major phase of repository development using the growing information base available at each stage, including site selection, the stages of design development, licensing and, finally, closure. Safety assessments play an important role in the iterative process of design optimization.

2.2. REPOSITORY DESIGN AND EVALUATION CYCLE

As with any design undertaking for a major civil project, the initial step is the definition of the problem and the specification of functional and operational requirements. The specific disposal solution(s) selected for implementation by a Member State will depend on the inventory of wastes requiring disposal and national policy governing disposal. National policy might, for example, dictate whether one or more sites are to be used for disposal.

International experience has shown that successful repository programmes evolve through a series of distinct phases. Each phase is initiated by a major programmatic decision with inputs from a range of stakeholders, some of whom may have mandatory legal involvement at specific stages (e.g. regulatory and planning authorities), others being essential or defined consultees, including elected officials, waste producers and members of the public. The scope and objective for each phase is to develop the design further, to meet regulatory or legal milestones and progressively refine and optimize the design to meet user requirements. The phasing allows review by stakeholders at each major decision point.

As previously noted, the design phases can be linked to iterative updates to the safety case. Decision points demarcating design phases are defined by safety assessment against regulatory requirements. Confidence in the safety assessment is gained through iteration, coupled with confirmatory data collection activities. As more data become available, the design and safety assessment are updated. When sufficient confidence has been achieved, the implementing agency will initiate regulatory review (e.g. submit a licence application). The outcome of the review, in conjunction with stakeholder input, forms the basis of the major decision to proceed, which may also involve governmental approval.

2.3. WASTE INVENTORY

Knowledge of inventory is essential for planning waste management and disposal activities. Consistent with reporting requirements under Article 32 of the Joint Convention [1], the waste inventory addresses radioactive waste management policies and practices and criteria used to define and categorize waste, including spent fuel (SF). The inventory also provides a description of the facility of origin of the waste, current status and location, and material information, including the waste volume or mass, activity, and radionuclide content. Additional information for inclusion in the inventory is provided in the IAEA publication Status and Trends in Spent Fuel and Radioactive Waste Management [14] and includes other relevant material aspects needed for planning and designing disposal facilities, such as waste form, thermal and chemical characteristics, and waste treatment and conditioning practices, as well as projections of future waste generation.

The first major task of the implementing agency (here assumed to be a WMO) is to compile the national waste inventory. Information to be included in the waste inventory is specified by the WMO and provided by the waste generators. Based on assessment of the inventory, the WMO will be positioned within the national legal and regulatory framework to identify requirements and to formulate reliable disposal strategies.

The waste inventory and characteristics are used along with national or location specific information on potential environmental and geological settings for disposal to identify appropriate repository concepts. Alternatives can range from near surface systems, intended to provide isolation and containment for decades to hundreds of years, to mined geological systems designed to provide passive isolation and containment for hundreds of thousands of years.

2.3.1. Waste classification

The development of classification schemes based on waste characteristics is an essential aspect of planning waste management activities. Different waste classification schemes are in use by Member States. To address this issue, GSG-1 [10] was issued in 2009 to assist Member States in planning waste management activities directed at disposal. The classification laid out in GSG-1 [10] addresses the principles laid out in IAEA Safety Standards Series No. SF-1, Fundamental Safety Principles [3], specifically Principles 1, 5, and 7. It recommends a comprehensive range of waste classes and provides general definitions for them based on increasing isolation and containment needs. The conceptual relationship of different classes is shown in Fig. 1. The class of waste and its specific characteristics help to determine minimum safety related requirements for the disposal concept. More specifically, related safety requirements are presented in SSG-29 [7], SSG-14 [8], and SSG-1 [6]. The IAEA waste classification scheme is used as the basis to present disposal concepts in this report.

The IAEA system identifies six separate waste classes corresponding to increasingly stringent requirements for isolation and containment [10]. The lowest classes have no special disposal requirements because their radiological hazard is minimal. However, the remaining four classes have increasingly more complex disposal requirements. Paragraph 2.2 of GSG-1 [10] defines six classes of waste:

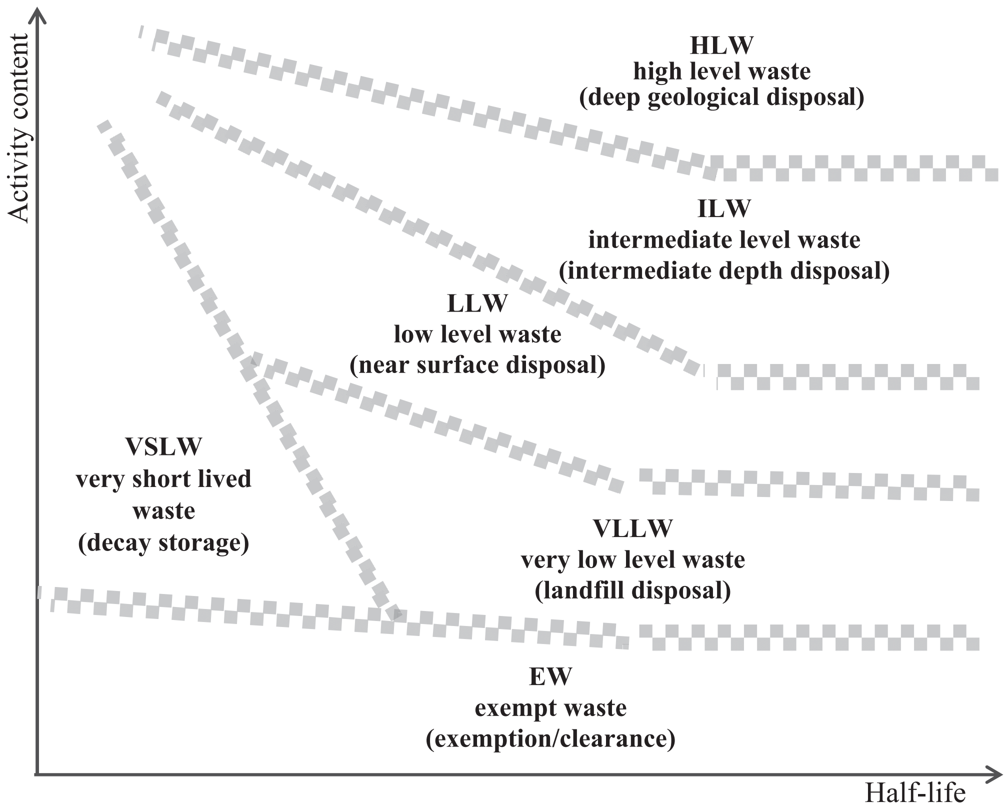

FIG. 1. The IAEA waste classification scheme [10].

(1) Exempt waste (EW): Waste that meets the criteria for clearance, exemption or exclusion from regulatory control for radiation protection purposes.
(2) Very short lived waste (VSLW): Waste that can be stored for decay over a limited period of up to a few years and subsequently cleared from regulatory control according to arrangements approved by the regulatory body for uncontrolled disposal, use or discharge. This class includes waste containing primarily radionuclides with very short half lives often used for research and medical purposes.[1]
(3) Very low level waste (VLLW): Waste that does not necessarily meet the criteria of EW, but that does not need a high level of containment and isolation and, therefore, is suitable for disposal in near surface landfill type facilities with limited regulatory control. Such landfill type facilities may also contain other hazardous waste. Typical waste in this class includes soil and rubble with low levels of activity concentration. Concentrations of longer lived radionuclides in VLLW are generally very limited.
(4) Low level waste (LLW): Waste that is above clearance levels, but with limited amounts of long lived radionuclides. Such waste requires robust isolation and containment for periods of up to a few hundred years and is suitable for disposal in engineered near surface facilities. This class covers a very broad range of waste. LLW may include short lived radionuclides at higher levels of activity concentration, and also long lived radionuclides, but only at relatively low levels of activity concentration.
(5) Intermediate level waste (ILW): Waste that, because of its content, particularly long lived radionuclides, requires a greater degree of containment and isolation than that provided by near surface disposal. However, ILW needs no provision, or only limited provision, for heat dissipation during its storage and disposal. ILW may contain long lived radionuclides, in particular, alpha emitting radionuclides, that will not decay to a level of activity concentration acceptable for near surface disposal during the time for which institutional controls can be relied upon. Therefore, waste in this class requires disposal at greater depths, of the order of tens of metres to a few hundred metres.
(6) High level waste (HLW): Waste with levels of activity concentration high enough to generate significant quantities of heat by the radioactive decay process or waste with large amounts of long lived radionuclides that need to be considered in the design of a disposal facility for such waste. Disposal in deep, stable geological formations usually several hundred metres or more below the surface is the generally recognized option for disposal of HLW.

This revised waste classification system is disposal oriented and can be used to help establish appropriate disposal concepts, including preliminary waste acceptance criteria (WAC) and other requirements. Additional information related to specific waste characteristics, including physical form, activity level and radionuclide content, chemical or biological composition, and the originating process, will further inform the preliminary WAC. Preliminary WAC are used as a first step in narrowing the definition of a waste stream for disposal and for developing a disposal concept.

2.3.2. Predisposal waste processing and packaging

Preliminary WAC guide predisposal waste management and are developed by the WMO to communicate requirements to the waste producer to ensure the suitability of waste for disposal in a future repository. The producer can then implement appropriate strategies for processing the waste into the required form. As the repository programme progresses, the WAC may evolve to reflect the actual conditions of the selected site and any requirements derived from the design, but it is important that these refinements do not entail radical changes to wastes already being processed and conditioned for disposal. The preliminary WAC thus need to be sufficiently broad that they do not preclude possible design changes within an overall disposal concept. It is not advisable to immobilize the waste in any type of matrix prior to finalization of the disposal concept and development of a detailed understanding of the WAC.

Some VLLW can be directly disposed in bulk form. However, most other classes require processing to varying extents after their initial generation. Waste processing options, for example to reduce volume, can be an essential consideration in formulating and selecting a repository design concept. The goal of processing is to create waste

[1] The waste classes, EW and VSLW, are included for completeness; however, as disposal is not required based on their radiological characteristics they are not further discussed in this publication.

in a form that is suitable for storage and/or disposal and that would meet the preliminary WAC for the repository concept(s) being considered.

A major function of predisposal management is to sort, characterize, treat, condition and finally package the processed materials into containers suitable for disposal. Following an initial sorting stage, generally based on physical characteristics, characterization is conducted to bin waste for subsequent treatment and conditioning. Treatment processes are implemented to enhance the safety and/or economy of their handling by volume reduction or otherwise changing its characteristics into a form compatible with storage, transportation and disposal. If needed, the treated waste can subsequently be conditioned into a stable solid form to immobilize radionuclides, often with deposition directly into a disposal container. Multiple containers can be packaged together. Characteristics of the waste and the final package play an important role in repository design.

For VLLW produced in association with decommissioning of nuclear installations, simplified pretreatment steps can generally be used. VLLW often consists of concrete, scrap metal, refuse and contaminated soil. Pretreatment is often limited to simple size reduction to allow for easier handling and transport, while preventing dust dispersion and containing loose materials. VLLW can often be contained in large bags, standard transportation drums or other simple containers.

Radiation levels for LLW can be high enough to require a minimum level of shielding for handling and storage prior to disposal. The largest volumes of LLW usually consist of contaminated personal protective equipment, tools, rags, spent ion exchange resins, construction debris and refuse, and scrap metal. The waste can undergo treatment by volume reduction (e.g. incineration or compaction), immobilization using an appropriate matrix (generally cement, polymer or bitumen based materials) or a combination of these. LLW in either treated or raw form is usually placed into a drum or other appropriate container. Waste contained in this manner can be placed in an overpack, capable of holding one or more containers, to provide shielding. Remaining void space inside the container or overpack can be filled if necessary, using either an active or inactive mortar. The resulting containerized LLW forms the waste package. Additional shielding can be incorporated into the designs for storage facilities and handling equipment.

ILW exhibits a higher activity content that can be attributable to short and long lived radioisotopes. The activity levels and concentrations of long lived radioisotopes associated with ILW are such that geological disposal is needed. ILW from nuclear power plants consists of reactor core components, spent ion exchange resins and filters used in reactor systems to purify coolant water. Waste treatment and conditioning are typically required. Volume reduction, which concentrates radioactivity, needs to be controlled to avoid increasing surface dose rates beyond safety limits. ILW is generally conditioned into a passively safe form by either cementation or vitrification and placed into appropriately shielded containers. One or more containers can be grouped together and overpacked to form a waste package. Storage of ILW in shielded facilities is needed, pending disposal.

HLW, and SF when classified by national policy as waste, necessitates the highest level of isolation and containment in a geological disposal facility. The fission products in SF or in HLW generated for reprocessing SF create high levels of both heat and radioactivity. The presence of long lived radionuclides in HLW from reprocessing requires immobilization into an insoluble solid waste form that can remain stable for very long time periods, with vitrified or ceramic waste forms being considered suitable for storage and disposal. Most geological disposal concepts for HLW and SF require packaging into specifically designed disposal canisters. Some less developed concepts propose direct disposal of storage containers (e.g. casks).

2.4. REPOSITORY DESIGN INPUTS AND OUTPUTS

A similar design approach is broadly applicable to all types of repository that might be considered. Construction, operation and closure concepts are essential parts of the design. The repository has to be constructible and operable within an acceptable timeframe, under realistic technical and financial constraints. Attributes of the repository design are incorporated in the evolution of the safety case, which may impose additional constraints. The overall repository layout, the scope of waste handling facilities and disposal throughput can depend on both existing and projected waste volumes.

As a project progresses, some level of research, development and demonstration (RD&D) will be necessary to support engineering analyses, which in turn support design decisions. Design elements that contribute to a safety function are subject to additional confirmation by investigations that may occur throughout the project, up to the time of closure. Investigations to confirm safety functions are generally performed under prescribed quality requirements.

The use of available technologies and well tried materials, components and systems can minimize project risk and reduce the need for RD&D. To a large extent, technological systems or components for radioactive waste disposal are available and have been thoroughly tested and one of the principles of repository design is the utilization of existing technology (Section 3.1). Where necessary, the technological readiness of a component may require confirmation as part of design review or operational readiness review. However, in the case that a clear need for a new technology is identified over the course of the programme, it will be necessary to ensure that it is developed to the point of readiness. Technological innovations might range from adapting existing equipment for use in radiological environments to the development of new equipment for specific tasks unique to a disposal concept (e.g. waste package deposition). New design elements that have not previously been used in the manner planned require thorough testing and confirmation, preferably at full scale and especially if the element is assigned a safety function.

With an initial disposal concept selected for a potential site, the design will be adjusted to the specific site conditions as determined by the ongoing site characterization programme. For near surface repositories such adjustments might accommodate geological variability and geomorphic features, such as topography and drainage paths. For repositories at depth, the design might need to be adjusted for thermal, mechanical, hydrogeological, chemical and seismic hazard conditions. As the design progresses, the WMO will use iterative, operational and post-closure safety assessments to confirm the performance of the facility. These safety assessments require information on:

- Waste characteristics and inventory (radiological, physical and chemical);
- Characteristics of the waste packaging (material properties and behaviours under projected repository conditions);
- Characteristics of the geosphere (natural processes in the soils, rocks and groundwaters around the repository, including responses to environmental changes and natural events, such as evolving climate and seismicity);
- Design characteristics of the repository (layout and emplacement strategy and use of engineered barriers, e.g. buffers, liners, caps, plugs, borehole and shaft seals);
- Characteristics of the biosphere (e.g. plant rooting depths, radionuclide uptake processes, burrowing animals, groundwater usage);
- Socioeconomic characteristics of the potentially affected population (land use, agriculture, population density, etc.).

Typically, the safety assessment will be iterated several times with increasing levels of detail over the life cycle of the repository programme as more information becomes available on the site and as the design is refined, generally to meet predefined regulatory milestones. With each iteration, the design will be better integrated into the natural setting associated with the site. This approach helps to ensure balance between reliance on natural and engineered barriers, which can improve defence in depth and limit the cost of constructing engineered barriers. The overall aim of this iterative design process is the development of a robust design and fully defensible safety case.

With the use of safety assessments, the disposal concept will evolve into a design suitable for licensing. A licence submittal will consist of the design, WAC and a safety assessment. Regulators might issue separate construction and operating licences, each requiring a submission from the WMO. A detailed compliance review by the regulator will be conducted for each submission. To facilitate the review, it is essential that decisions made during development are well documented and traceable and records maintained with appropriate application of quality management systems. The compliance review will include an evaluation of the technical quality, completeness and accuracy of the application, to ensure that the regulatory requirements have been met. Independent reviews might also be conducted. The licence application review can include an opportunity for litigation by potentially affected stakeholders, further justifying a fully transparent process. The regulatory review process might also result in the definition and approval of the final WAC, as well as the specification of any conditions that must be met by the WMO before or during construction or operation.

2.5. REPOSITORY FUNCTIONAL SYSTEMS AND LAYOUT

All repositories have certain elements in common, such as waste handling equipment and facilities, temporary waste storage, excavated material storage, shop areas for equipment maintenance and storage, administration areas, utilities, sanitary facilities, emergency facilities, visitor reception facilities and so on.

The repository layout will arrange these common elements with respect to topography and other site conditions, access routes and potential environmental elements (e.g. presence of natural drainages or wetland areas). Layout will also reflect legal requirements, regulations, codes and standards, and any programmatic constraints. Certain activities and facilities will need to be grouped into different zones to meet physical security, radiation protection and possibly nuclear safeguards requirements.

For physical security it is common to arrange the entire facility behind a secure perimeter, generally consisting of a perimeter security fence, access control and monitoring systems. Applicable IAEA recommendations specific to physical security are provided in Nuclear Security Recommendations on Physical Protection of Nuclear Material and Nuclear Facilities (INFCIRC/225/Revision 5) [15]. The repository grounds will include conventional facilities and the radiation protection area. Radiation protection requirements for a repository facility will essentially be the same as for other nuclear facilities or activities involving similar radioactive materials. Specific safety requirements for disposal facilities are given in SSR-5 [5], while general safety requirements are given in IAEA Safety Standards Series No. GSR Part 3, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards [12].

The key feature of a repository is the disposal area itself. Depending on the waste being disposed, this feature can range from near surface trenches or concrete vaults, subsurface vaults, chambers, silos or boreholes, to deep geological disposal systems that include excavated chambers or drifts. Deep borehole disposal systems can also be considered. Figure 2 presents a stylized overview of the different disposal options available for various waste classes. The presentation in Fig. 2 and the use of depth to discriminate between waste classes is consistent with the IAEA classification system presented in Section 2.3. It might be desirable for a repository facility to include more than one type of disposal area to accommodate different waste streams, in a modular approach, if safety and feasibility can be demonstrated.

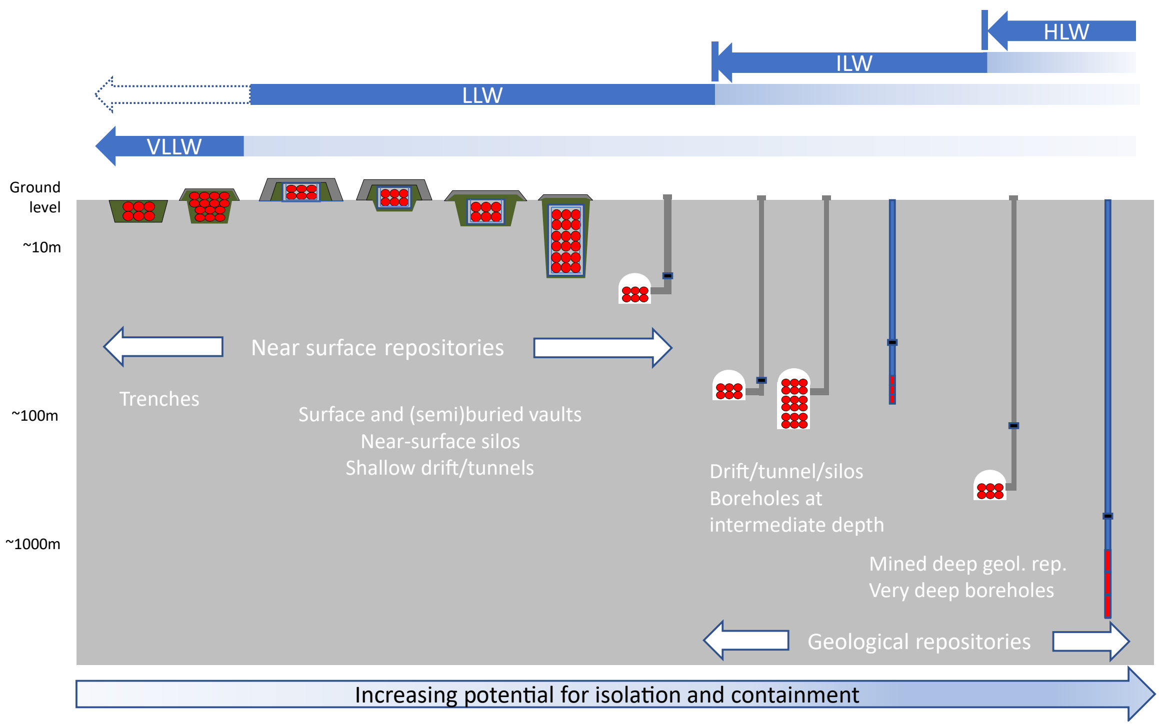

FIG. 2. Schematic illustration of the range of disposal options, from surface to deep, currently considered or implemented for different classes of radioactive waste.

Repository development and operation could involve concurrent construction and waste disposal operations in different parts of the facility, with construction activities limited to clearly defined areas to minimize interference. Best practice demonstrates that the full scope of disposal activities to be conducted at any site be addressed from the outset of project planning, to avoid having to revisit regulatory licensing later, for example, with a new or different waste stream.

Wastes are transported to a repository in suitable containment systems, generally being packaged for disposal before transport. For some wastes, packaging into disposal containers might be conducted at or near the repository site — for example, SF encapsulation. In addition, waste generated on-site at the repository, such as contaminated protective clothing, tools, wipes and equipment, might require treatment and packaging. These will either be transported to another disposal facility or processed on-site, depending upon the capabilities included in the design.

Figure 3 shows the layout of a hypothetical repository site. Actual facilities can vary from those presented. Functions can be divided into those directly related to waste disposal and those required for operational support:

— Waste disposal:
  - Waste reception control: Waste shipments arriving on-site must be confirmed for acceptance. A control area allows inspection of the arriving shipment to confirm contents and allow visual inspection and radiological surveys. WAC should be confirmed for each waste shipment designated for disposal (item 1 in Fig. 3).
  - Buffer storage: Inclusion of a buffer store provides flexibility in the rate of waste package receipt and disposal. This feature may not be required if the waste producer is nearby and can manage waste shipments in a manner that supports disposal operations. However, for other situations a designated buffer storage area can facilitate throughput of waste to the disposal area for efficient operations (item 2 in Fig. 3).
  - Waste processing and transport on-site: After receipt and acceptance the waste might need additional processing prior to movement to its final disposal position (item 3 in Fig. 3). As previously noted, most processing would be expected to be carried out prior to shipment, but in some cases a specialized facility (e.g. encapsulation plant for HLW and SF) might be located at or near the repository site. Waste packaged for disposal will require loading onto a dedicated transporter at the repository, delivery to the disposal area and unloading and emplacement in the final disposal position. Specialized equipment might be used for each of these activities. Selection of the most appropriate equipment will depend upon factors that include radiation protection, characteristics of the waste or waste package and the designated final disposal position in the repository, among others. The final position and other data for each waste package are recorded and entered into the records management system.
  - Disposal area: The disposal area is the only permanent feature of the repository system. Construction can be conducted either in a single campaign or following a phased, modular approach, which can be beneficial in controlling costs, as disposal can be better matched to demand. In addition, the phased approach can allow opportunities for improvements to be implemented that further enhance the overall efficiency of the system (item 4 in Fig. 3).

— Operations support:
  - Physical security: Engineered security features ensure control of access to the site, including personnel, vehicle control, control of supplies and materials and management of waste shipments (items 5(a)–(c) in Fig. 3).
  - Radiation protection and monitoring: Standard methods for radiological protection of workers and the public are used for repositories, commensurate with the type of radiological hazard. Design measures include compartmentalized facilities, ventilation systems and other measures to ensure protection during normal operations, accidents and other off-normal events (e.g. fire, seismic ground motion). Operational measures include dosimetry, alarms and administrative controls. Monitoring for potential radiation releases at the perimeter of the layout, or other boundary designated to protect the public, is also needed. Radiation monitoring is used at access and egress points associated with radiation protection areas.
  - Administrative support facilities: As with any large scale industrial activity administrative support will be needed. Facilities for management, technical personnel and records control are essential to the operation of the facility (item 6 in Fig. 3).

- Laboratories: Limited on-site analytical capabilities are required at all disposal facilities (e.g. analysis of wipe surveys). The extent of on-site laboratory capabilities and capacities is a function of WAC and disposal system requirements. An evaluation of analytical needs and potential local capabilities should be conducted early in the planning phase. Analytical capabilities might include provisions to ensure sample integrity and preserve archival samples (item 7 in Fig. 3).
- Auxiliary functions: Other auxiliary functions could include:
  - Equipment repair and maintenance facilities;
  - Garages and parking facilities for equipment, road vehicles and railway wagons;
  - Grounds maintenance.
- Infrastructure and utilities: Including internal roadways; electricity, water and sanitary facilities; heating, ventilation and air conditioning (separate from radiological controlled areas); data transmission and communication facilities; data management and computing services.
- Visitor reception and visitor centre (item 8 in Fig. 3): Such facilities are proven to be effective for public outreach and education, particularly in the surrounding communities.

Areas might also be reserved for future expansion (item 9 in Fig. 3) or storage depots for interim material (e.g. excavated spoil).

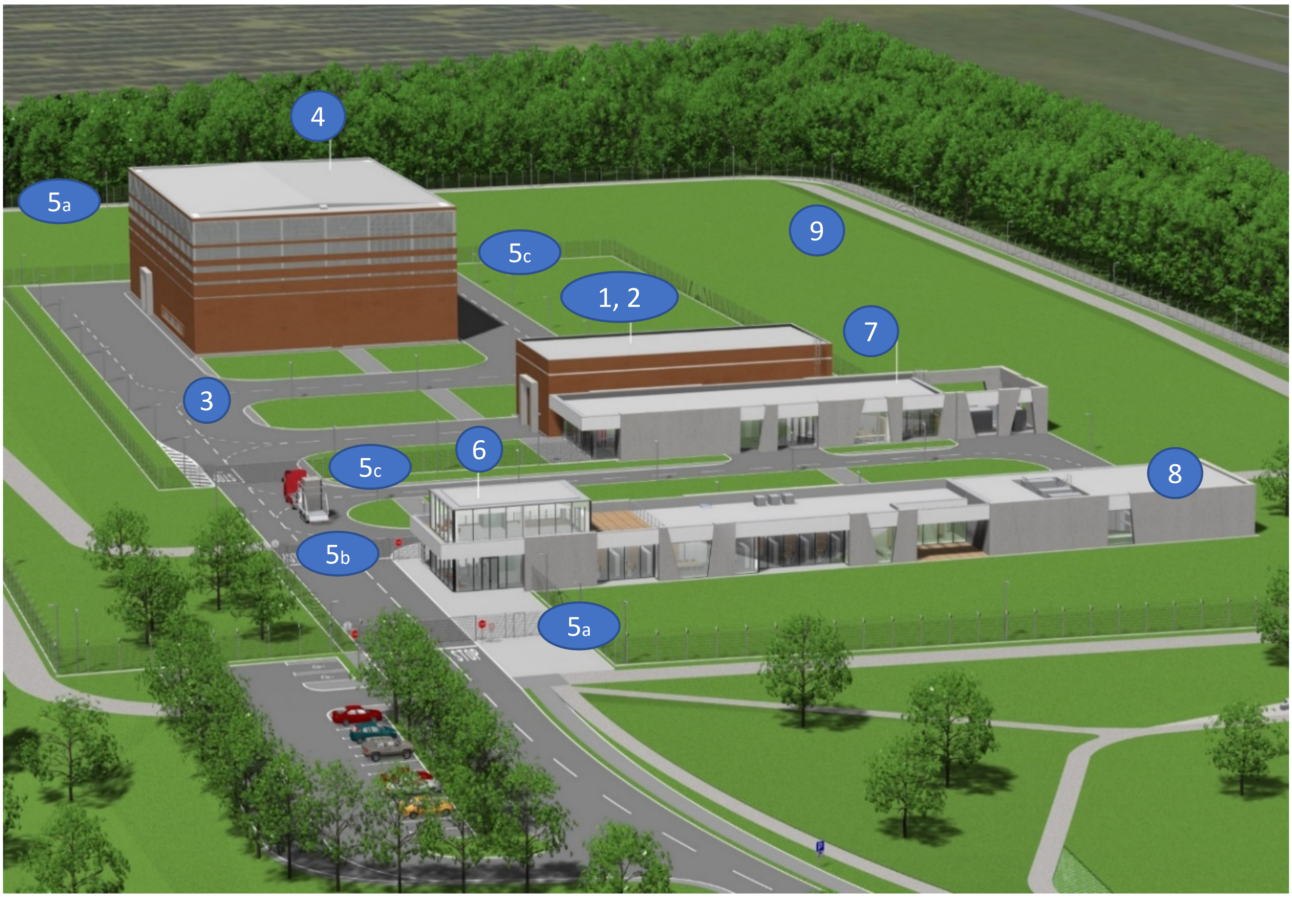

FIG. 3. General layout of a repository (prepared with the permission of the Agency for Radioactive Waste, Slovenia). Legend: 1 — waste reception control; 2 — buffer storage; 3 — waste transportation route; 4 — disposal area; 5(a)–(c) — physical security features; 6 —administrative support facilities; 7 —laboratories; 8 — visitor reception and visitor centre; 9 — area reserved for future expansion.

3. DESIGN PRINCIPLES AND STAGES

The timeline shown in Fig. 4 illustrates a generic repository programme life cycle and is used as a framework throughout this section, which focuses on design principles. The timeline identifies the main phases and stages of a repository programme, together with typical licensing points. Also shown are the five stages in the design process. Milestones, such as the identification of potential candidate sites and construction licence authorization, represent key decision points and are supported by specific design stages with associated design outputs used to help inform these decisions.

3.1. DESIGN PRINCIPLES

This section introduces principles that are intended to guide the development of repository design throughout a programme life cycle. Section 3.2 then describes activities specific to each design stage. The following guiding principles are used:

(a) Use of a requirements driven design basis;
(b) Design based on the multiple barrier safety concept;
(c) Use of safe, reliable, available and maintainable technology;
(d) Iterative development and optimization of the design;
(e) Maintenance of design integrity;
(f) Production of a transparent and auditable design;
(g) Incorporation of nuclear safeguards and security integrated design.

These principles are elaborated upon in Sections 3.1.1–3.1.7.

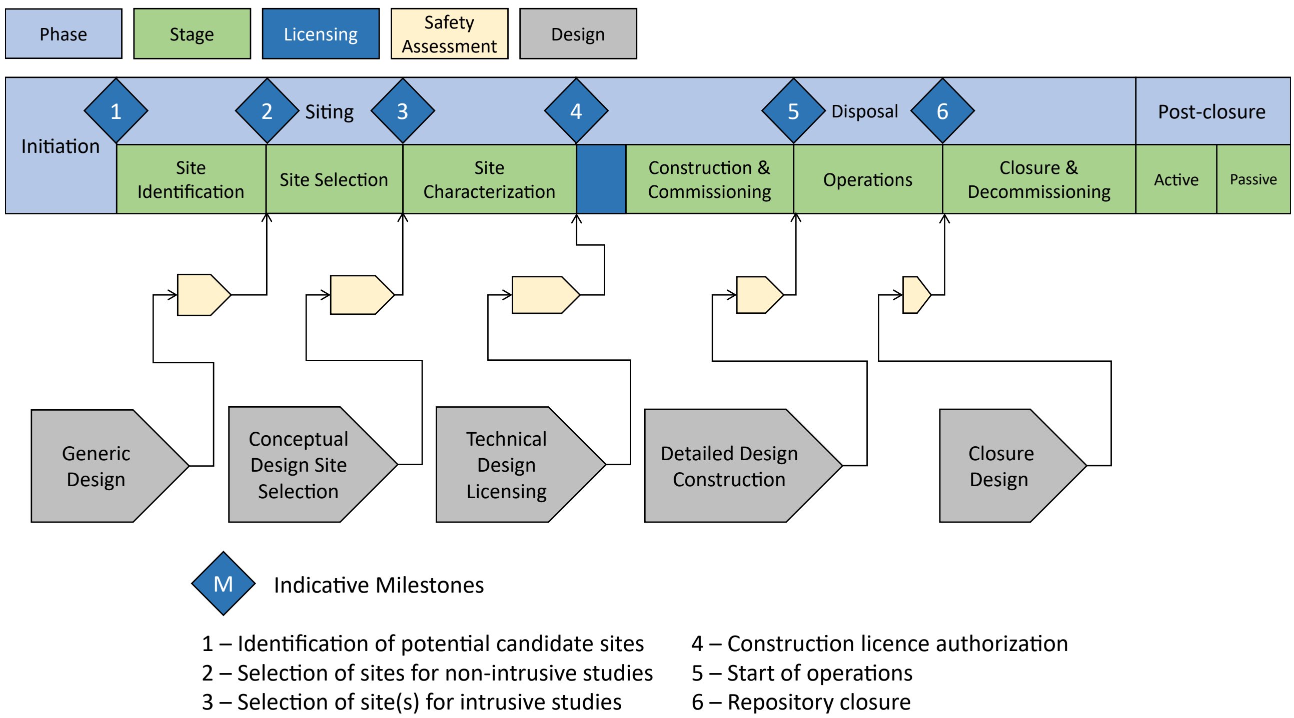

FIG. 4. Generic repository programme life cycle and associated design stages aligned with indicative project milestones.

3.1.1. Requirements driven design basis

Based on IAEA safety principles, the management system for a repository (SF-1 [3], para. 3.12):

> "has to integrate all elements of management so that requirements for safety are established and applied coherently with other requirements, including those for human performance, quality and security, and so that safety is not compromised by other requirements or demands."

International experience has shown that this principle is best addressed by the early development of a requirements driven design basis.

Technical specifications comprising the design basis for a repository are commonly articulated as a set of requirements, assumptions and constraints that are developed and managed throughout the design process. The process of 'requirements management'[2] is used by many WMOs to:

- Clearly define the requirements and assumptions pertaining to the disposal system and its individual components (e.g. engineered barriers);
- Make linkages and interdependencies explicit;
- Identify conflicting requirements and potential resulting trade-offs;
- Record formally the justification for decisions in support of design substantiation;
- Support design change control, by enabling tracking and recording of changes to either the requirements or the knowledge base and identifying how these are to be reflected in design changes.

This approach enables the articulation of an unambiguous design basis that can be used to communicate and define inputs and outputs across multiple disciplines, both within the WMO and in outreach to external stakeholders, including facilitating interaction with regulatory authorities.

A requirements driven approach helps ensure that the design basis that the WMO establishes is "justified by safety assessment, to ensure that the disposal facility is developed in accordance with the safety case. This has to include waste acceptance criteria…and other controls and limits to be applied during construction, operation and closure" (SSR-5 [5], para. 3.14). The highest level requirements determined by involved stakeholders define the framework under which further requirements can be elucidated in ever increasing detail following a requirements hierarchy.

Figure 5 illustrates a typical hierarchical structure where the design basis, as implemented within a defining framework of stakeholder requirements, is elucidated with increasing specificity as knowledge of the design intent is gained. Following this approach, requirements are increasingly detailed, first at a repository system scale, followed by more detailed subsystem requirements, which in turn translate to requirements for specific components and so forth. The required verification of the repository design should be specified at each level alongside the specific requirements, i.e. how the designed or constructed component fulfils the requirements placed upon it. By defining both the design intent and the design assurance in this progressive manner, repository wide validation against the iteratively developed safety case can be demonstrated. Table 1 describes the scope covered at each level of the design hierarchy.

A requirements driven approach enables the holistic integration of requirements, constraints and assumptions from a relatively early stage, thus ensuring that mandatory drivers for safety, physical protection, environmental protection and nuclear safeguards are integrated into the design basis.

The requirements are usually managed using software to facilitate both change and configuration control (see Section 3.1.5) and to create a traceable (i.e. auditable) record. Figure 6 illustrates an example of the hierarchy introduced in Fig. 5 for part of a near surface repository design basis.

A functional description of the repository as a holistic system helps develop system requirements. This functional description is used to develop operational and safety requirements. Satisfaction or achievement of each requirement can be measured by a prespecified verification activity. For example, accomplishments can relate to

[2] Requirements management is often referred to as the backbone of systems engineering, which is widely used across the software, aviation and civil manufacturing industries.

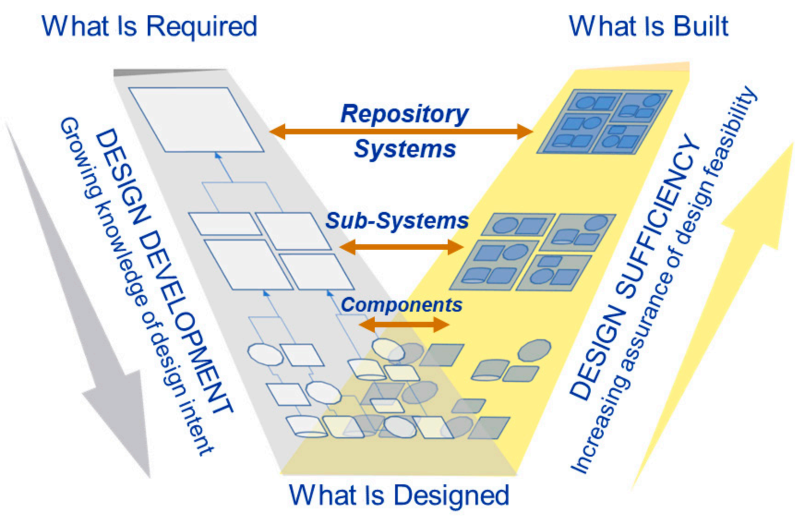

FIG. 5. Design basis hierarchy of derived requirement articulation and demonstration.

TABLE 1. DESIGN BASIS HIERARCHY AND SCOPE

Level (document)	Description	Application
High level requirements	These can also be termed ‘stakeholder’ requirements. High level requirements can be mandatory (e.g. imposed by legislation, regulations and local and national authorities responsible for licensing the repository) or by agreement, (e.g. with local and regional communities and with agencies responsible for funding the repository). They also include requirements from waste producers responsible for packaging. These entities will vary according to country regulatory regimes and the extant stage of repository implementation.	Generic and conceptual, but must be met by any proposed design.
Repository system requirements	These are functional (i.e. the function of a system) and non-functional (e.g. safety functional) requirements that define the total repository system and its management. These can include site specific constraints and characteristics, waste inventory, waste package types and numbers, the mode of transport for waste and construction materials to the repository, etc. Controlled assumptions (verifiable or not) are also often included.	
Subsystem requirements	At this level the safety concept (see Section 3.1.2) is specified as requirements for each of the major components, engineered (and geological) barriers and activities of the repository, where appropriate, expressed as ‘safety functions’.	
Component specification requirements	Detailed requirements for each component, barrier and associated safety function, which cover the design, construction and manufacturing.	Component specifications derive from the design.

compliance control and monitoring programmes, to provide confidence that the repository is being constructed and operated in accordance with the prerequisites of the original design basis.

The design basis is progressively updated, refined and extended (originating as a functional specification that is further detailed into technical specifications and specific component specifications) as site specific and inventory information becomes available. It provides the platform for design optimization when a specific site has been selected.

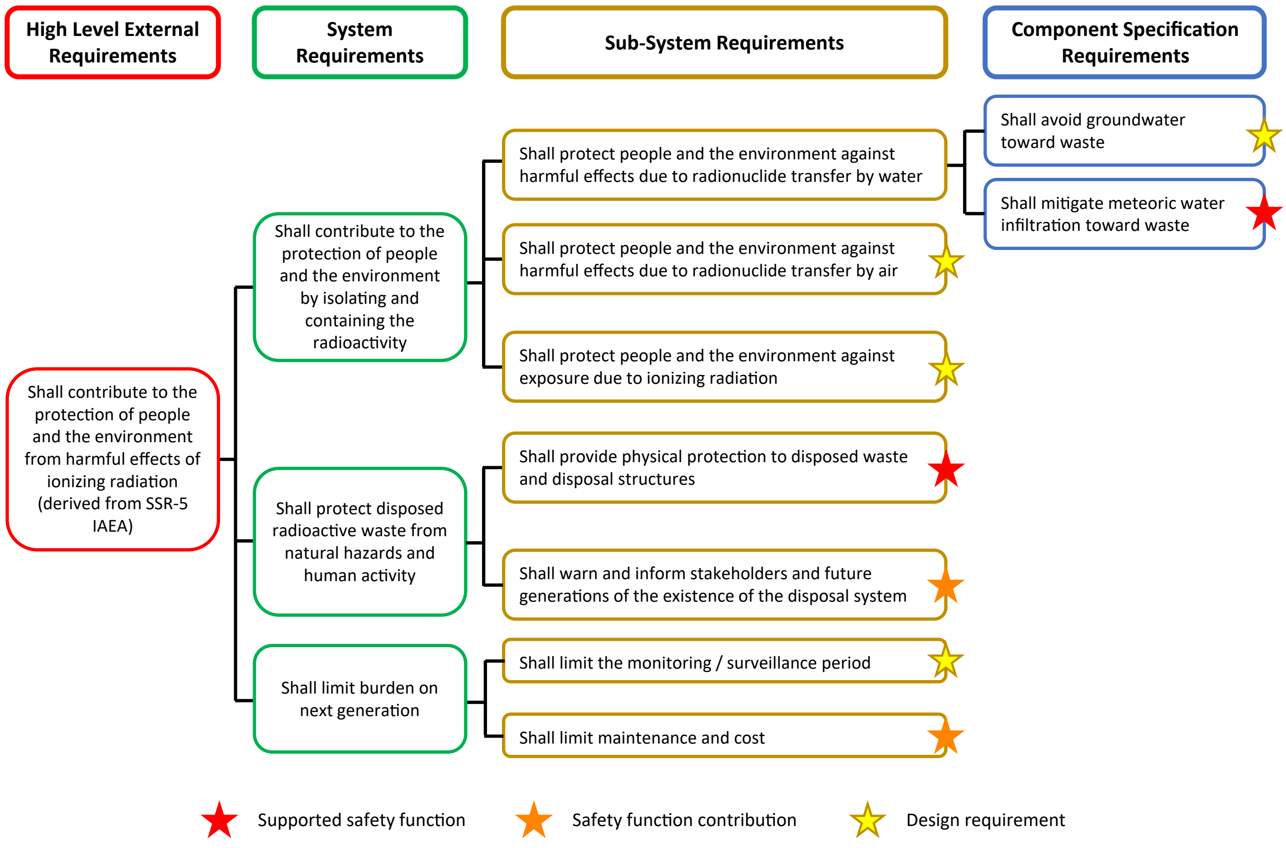

FIG. 6: Example of a repository design basis for a near surface repository final cap (adapted with permission from Agence nationale pour la gestion des déchets radioactifs (Andra)).

Further details on the implementation of a requirements driven process can be found in deliverables from the IAEA's project on the demonstration of the operational and long term safety of geological disposal facilities for radioactive waste (GEOSAF Part III) [16].

3.1.2. Multiple barrier safety concept

As previously discussed, international best practice is to design repositories with multiple barriers to prevent or control release and subsequent migration of radionuclides from the waste to the biosphere, so as to ensure (a) isolation of waste from the environment and (b) containment of radionuclides inside the disposal system [3, 5, 17]. A repository safety concept is commonly based on a multiple barrier system (as illustrated in Fig. 7) and is consistent with Principle 8 of SF-1 [3]. The barriers work in different ways, contributing to long term safety by providing one or more of the following functions:

— Physical containment (e.g. within the waste container);
— Chemical containment (e.g. on or within the materials of the waste form and buffer);
— Engineered containment (e.g. within major engineered components of the repository, such as liners used in excavated trenches or tunnels, bulk backfills and high integrity sealing systems);
— Geological isolation and containment (the geological environment provides physical and chemical stability for the engineered structures, low groundwater flow and retention of radionuclides) [3, 5, 18].

The individual safety functions of the components of a multibarrier system operate over different timescales [5]. Barrier performance requirements are linked to these specific safety functions, sometimes with quantitative performance targets set to ensure that the safety function is achieved. Typical safety functions include:

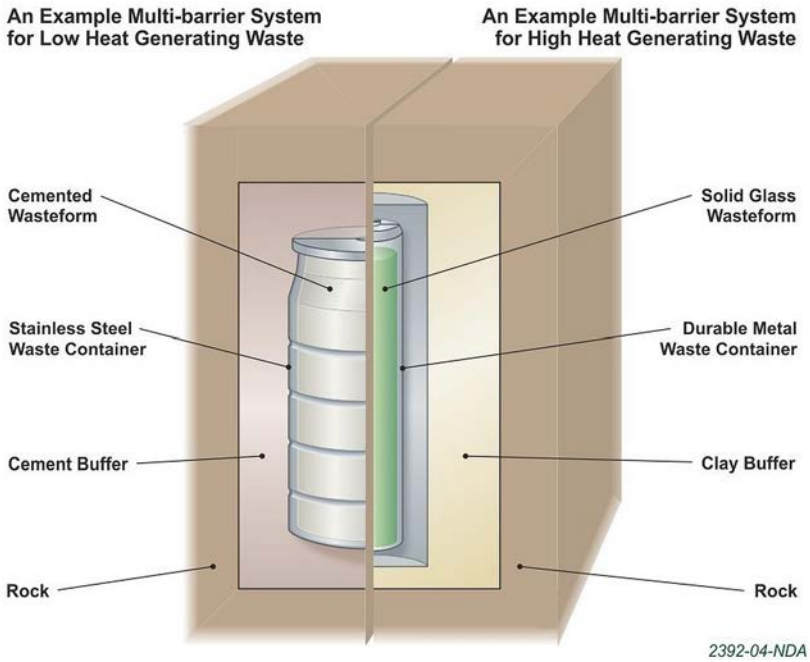

FIG. 7. Schematic illustration of a repository multibarrier safety concept (courtesy of UK Nuclear Decommissioning Authority, NDA).

— The physical protection of waste packages from the impacts of external events;
— Controlling the flux of water through and around the waste;
— Controlling the chemical environment around the waste;
— Limiting radionuclide release by immobilization or a period of physical containment in waste containers;
— Retardation or retention of radionuclides during migration through barriers or along air or groundwater pathways to the biosphere;
— Reducing the likelihood of inadvertent human intrusion into the wastes.

Each repository component thus contributes to the safety of the overall disposal system. This approach is illustrated using the flow diagram in Fig. 8 to show how a specific technical solution for a repository, in this case shaft/ramp seals, can act to meet disposal system safety functions and performance requirements.

Engineered barriers are designed to function passively, without human intervention, to act contiguously and to perform predictably and quantifiably on an individual basis, until such time that any radionuclide release to the environment is acceptably low. Post-closure performance assessments use these assumptions to model and validate barrier containment properties. Natural, archaeological and older industrial systems can also be used to provide insights and information on the long term processes and physical and chemical behaviour of engineered or geological barriers, over the very long timescales relevant to post-closure safety [19].

3.1.3. Safe, reliable, available and maintainable technology

Many engineered structures, systems and components are required in a repository:

- For construction, to excavate soil or rock and transport spoils from the construction zone, as well as transport engineered barrier materials to the repository, etc.;
- For operation, to transport waste containers and emplace them in assigned disposal locations;
- For closure, to backfill or cap disposal zones and to seal access ways in an underground repository.

The technology used may have to withstand relatively harsh environmental conditions, such as adverse climatic conditions at the surface or increased temperatures and humidity conditions in the subsurface, perhaps over long time periods. Safety during the construction, operation and closure of a repository is dependent on this

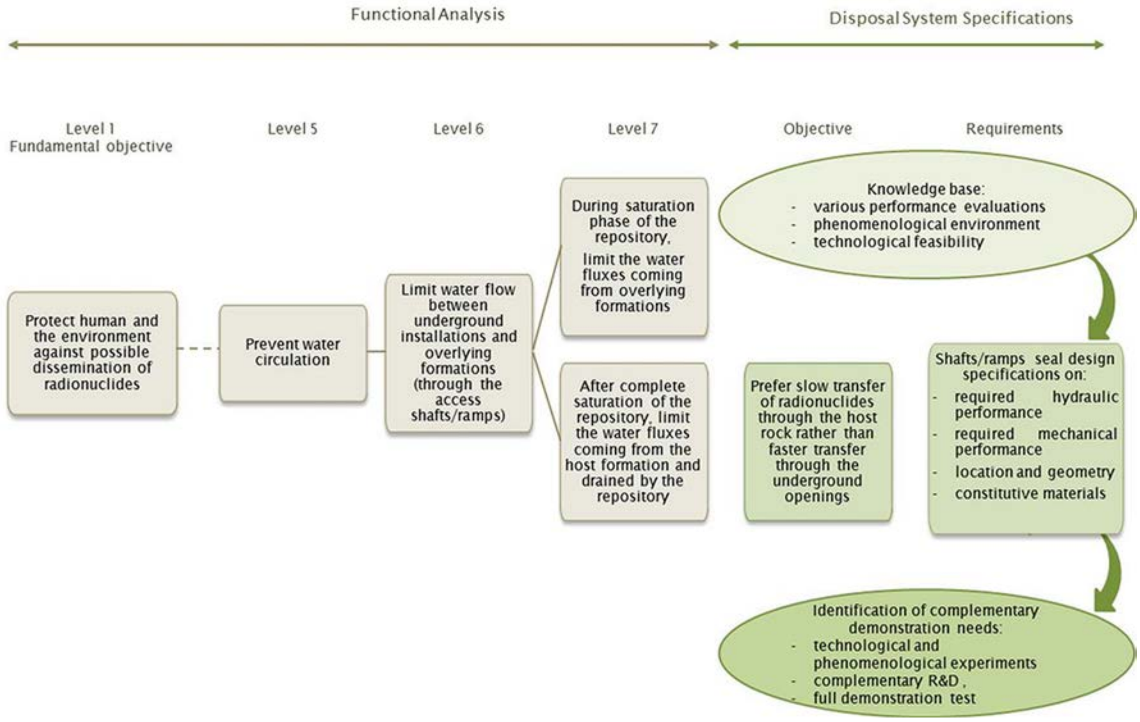

FIG. 8. Centre industriel de stockage géologique (Cigéo) project requirements flow diagram for shaft/ramp seal design (courtesy of Agence nationale pour la gestion des déchets radioactifs (Andra)).

technology being reliable and easy to maintain and replace. Allied to this is the use of technology that has been demonstrated to be robust enough to meet foreseen demands prior to its use in a repository.

Defence in depth is provided for a repository by an appropriate combination of "Design, technology and materials of high quality and reliability", as stated in para. 3.32 of SF-1 [3]. An understanding of the features of a repository and how they perform over time is necessary to be able to demonstrate dependability and reliability. This demonstration is assisted if such design features are robust (i.e. their performance is of low sensitivity to possible events and processes causing disturbances). Sufficient evidence of their feasibility and effectiveness needs to be obtained before construction activities are commenced [5].

The use of readily available technologies that have been proven at full scale and in similar environmental conditions to a repository provides design confidence. This reduces uncertainty and is fundamental to establishing a robust operational safety case to obtain a licence for construction and operation. The late identification of impractical or hard to implement technical solutions can lead to inefficient use of resources, high modification costs and longer implementation timescales, and can impact stakeholder confidence.

For example, during the 1980s and 1990s, the development of first generation deep geological disposal concepts was dominated by the need to demonstrate that post-closure safety could be achieved, based on scientific principles and performance assessment modelling. Establishing a sound basis of technical practicality for the constraints of actual repository implementation came later. This led to the development of repository concepts incorporating engineered components that could be difficult to implement remotely at full scale with the required quality assurance. Figure 9 presents examples from underground tests of various bentonite emplacement methods, illustrating the difficulties that can be encountered when transposing a concept into practical implementation. Figure 9(a) shows the example from the Grimsel Test Site in Switzerland. Figure 9(b) is a full scale tunnel mock-up in Toledo, Spain, showing the instability of bentonite blocks in the FEBEX tests when the emplacement operation was interrupted under humid conditions [20]. Figure 9(c) shows the disintegration of support blocks at Mont Terri, Switzerland, part of the LUCOEX experiment [21].

The Engineering Studies and Demonstration of Repository Designs (ESDRED) project (2004–2009) [22] is an example of subsequent work to demonstrate the technical feasibility of activities to construct, operate and close a deep geological repository.

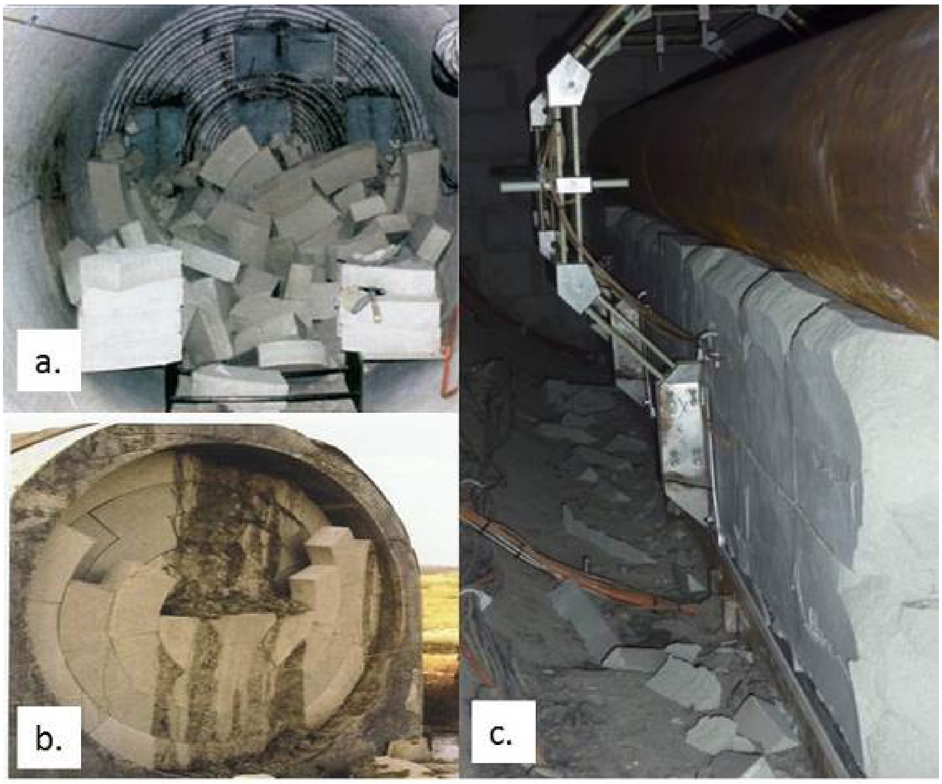

FIG. 9. Examples showing the difficulties in the handling and emplacement of highly compacted bentonite blocks underground.

Consequently, in most existing repository programmes, early design decisions are oriented towards the selection and use of proven technology as a fundamental principle, rather than the application of new technologies. When incorporating appropriate technology into the repository design basis, account needs to be taken of factors such as:

— When and how decisions on the selection of proven and available technologies need to be finalized, given the long project timescales usually involved;
— Whether some technology will need to be adapted or designed specifically for purpose, if it is not available, for example, waste container transfer and emplacement equipment might have to be tailored to country specific disposal concepts;
— How to ensure that an apparently practical and suitable technology does not degrade the ability of the engineered and natural barriers to provide safety functions;
— How to develop availability, reliability and maintainability requirements and incorporate them in the requirements management system;
— The ability to model the performance and impacts of different technology options;
— Balancing cost against other requirements such as safety and environmental considerations, which might lead to a relatively inexpensive technology being selected from a range of cost options if it can be shown to be robust, reliable and appropriate.

3.1.4. Iteratively developed and optimized designs

Major engineering projects, especially civil works requiring large capital outlays, usually follow conventional processes characterized by predefined stages. Figure 4 illustrates a generic timeline for repository implementation that includes the following design stages:

— Generic design;
— Conceptual design for site selection;

— Technical design for licensing;
— Detailed design for construction;
— Continued detailed design for repository expansion (represented by ‘Operations’ in Fig. 4);
— Design for closure.

Each design stage (see Fig. 4) is intended to allow iterative assessments of the repository site, re-analysis of safety and iterative changes to repository design options (see Fig. 10 and Table 2). Conventional staged design is described here because it mitigates management risk, conforms to existing practices, promotes quality products, allows the design team to develop and acquire needed competencies, and allows the development and improvement of management processes in conjunction with design. However, it is also recognized that iteration may be used within any step, following a somewhat less structured approach. Following an iterative and optimized design addresses Principle 5 of SF-1 [3] and is identified in SSG-23 [9].

This iterative approach provides a degree of flexibility to enhance safety and reduce uncertainties in analysed system performance. The approach allows incremental and controlled progress in the scope of safety analysis within each design stage, which is desirable for transparency. Periodic re-evaluations permit the insertion of new information (e.g. from ongoing site characterization and other RD&D activities) into the design basis and allow for technical maturation as design activities proceed, thus allowing requirements to be reassessed and waste inventory to be updated, prior to the next stage.

A stepwise approach to design generates new or revised design products at discrete points that apply to all design activities. Discrete intervals or stages increase in complexity as the design progresses towards construction. As described by the Organisation for Economic Co-operation and Development’s Nuclear Energy Agency in 2000 [23], these

> “steps facilitate the traceability of decisions, allow feedback from the public and/or their representatives, promote the strengthening of public and political confidence in the safety of a facility along with trust in the competence of the regulators and implementers of disposal projects.”

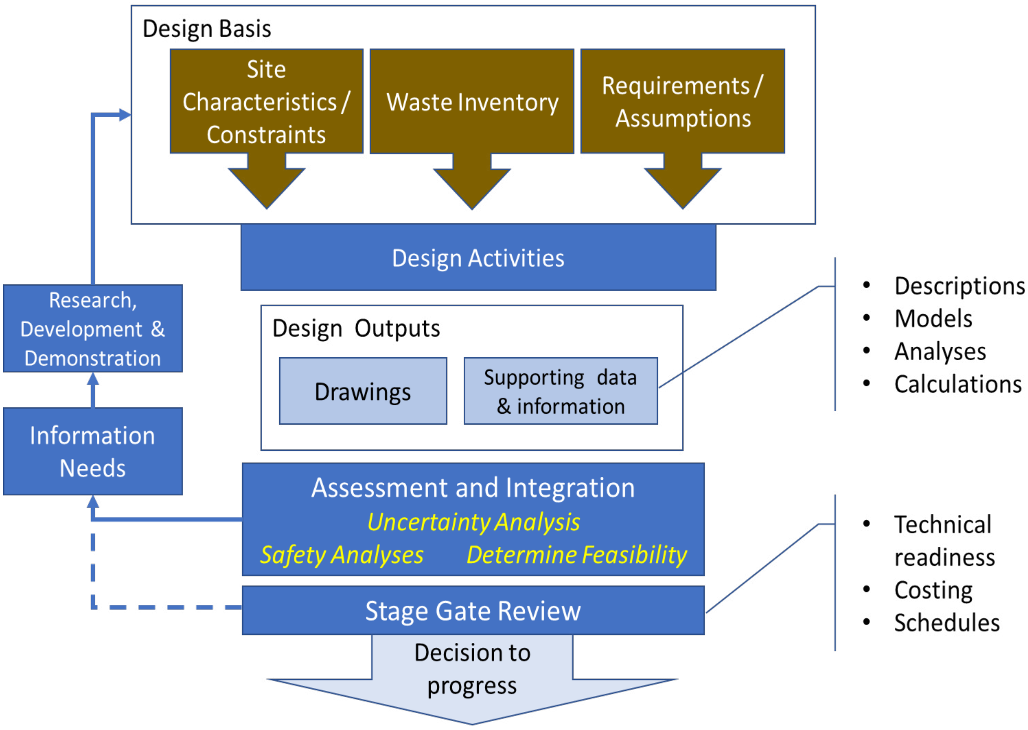

FIG. 10. The iterative design development process that is used during each design stage.

TABLE 2. DESCRIPTION OF THE ITERATIVE STEPS TAKEN DURING DESIGN DEVELOPMENT

Process step	Description
Design basis	See Section 3.1.1.
Design activities	Depending on the design stage, either feasible concept design options would be developed in response to the design basis, or a decision on the preferred design option would be taken and this would be used as this basis for a technical specification. A design would be developed that reflects the technical specification and could include: • An operational process description; • Drawings and calculations; • Digital design outputs (e.g. 3-D models and animations); • Cost and schedule analyses.
Assessment and integration	These include the following activities: • Operational safety analysis and post-closure safety assessment, with attendant uncertainty analysis; • Comparison of outcomes with regulatory requirements; • Assessment of interfaces between engineering design and safety analysis and environmental studies; • Assessment of changes to site, waste inventory, external requirements.
Stage gate review	At the end of each stage there is a reconciliation at a technical meeting, which uses formal, independent, interdisciplinary, expert scrutiny of the design basis, design outputs and associated safety analyses to support a management decision whether to proceed to the next stage (which would probably be taken with stakeholder input). Conditions and provisions for conducting the next phase might be provided. Technical readiness, cost and schedule will be assessed during this stage review.
Decision to progress	Depending on the outcome of stage gate review, a decision to proceed to the next implementation phase (e.g. a decision on the site).
Information needs	Identification of additional design work (enhancements or rework) necessary and the information and/or analysis required.
Research, development & demonstration	A programme of activities specified to address information needs. This could include, for example, further site investigation, waste characterization and technological development.

Finally, the idea of iteration within design stages may be extended to the overall process, as suggested by McCombie [24]. ‘Adaptive staging’ is a flexible process characterized by design and decision points that are progressively redefined, which may be more responsive to dynamically changing design basis information or stakeholder input.

3.1.5. Maintenance of design integrity

As stipulated under Principle 1 of SF-1 [3], the licensee for a nuclear waste disposal facility is responsible for safety, and for “Establishing procedures and arrangements to maintain safety under all conditions” (para. 3.6). This principle flows directly to the maintenance of design integrity, to ensure the safety of designed systems, structures and components throughout their lifetimes. As the repository licensee and the ‘design authority’ for the repository project, the WMO has a responsibility to preserve design integrity.

Further information on the design authority concept and other guidance can be found in INSAG-19 [25], Maintaining the Design Integrity of Nuclear Installations throughout their Operating Life.

3.1.5.1. Design control

Safety integrated design activities are implemented using a set of tools and processes collectively referred to as 'design control'. Design control requires the WMO to assume responsibility for the design as the design authority and the future repository licensee [25]. Design control tools and processes include the checking and review of design products and the use of systematic requirements management (see Section 3.1.1), design configuration management, change control, interface management, data management and control of software.

Configuration management (e.g. document control) controls design products so that they can be changed only through a 'change control' process. The design authority for the repository presides over the change control process and has ultimate responsibility for it. Interfaces occur between subsystems or different facilities, and between design and other technical activities, such as the operational and post-closure safety analyses. The accurate representation of design features is a necessary input to the safety analyses. A design basis requirements led approach (see Section 3.1.1) is commonly used to identify and specify interfaces, often well in advance of related design activities.

Design control typically requires an information management system for the retention and control of site data, engineering data, specifications and other information, such as codes and standards. Today, design products are predominantly digital — developed and integrated using software. The software used to develop and manage design products is subject to verification controls (for correct function) and validation controls (for proper application), following documented procedures.

Design control processes contribute to maintaining the integrity of the design and the overall basis for the safety of a repository throughout its full operational life cycle, including any periods of institutional control, which might be tens to hundreds of years. Controlled information and tools are necessary to understand the totality of the design and nuclear safety case over the long term, at each stage of the disposal system life cycle. The intergenerational nature of long term waste disposal programmes requires careful attention to the proper management of the design basis and design products.

3.1.5.2. Safety culture

Waste management programmes should develop and maintain a nuclear safety culture equivalent to that of nuclear power system operations [26]. Safety culture embodies tenets of good management and is applicable to any project. Projects that have to bear stakeholder scrutiny benefit especially from the effectiveness and transparency that can be achieved. Safety culture requires the investment of resources in management processes and staff training that can take years to develop.

An effective design team is organized with well defined roles, responsibilities and accountability for product quality. A well functioning design organization is supported by management with a condition reporting system, which can be used by employees to express quality or safety concerns without retribution, for tracking and resolution. Only in this way is it possible to reliably identify the limits or any degradation of operating conditions (including for design activities) and to maintain the authority and accountability for system modifications or any suspension of activities.

The management processes needed for effective waste disposal project management include issues management and commitment management. A database is used to track issues or commitments for resolution and the actions needed to resolve them. The long duration and multiple stakeholders involved in radioactive waste management and disposal necessitate a formal approach.

3.1.5.3. Knowledge management

A related need is referred to as 'knowledge management', whereby key records and data, and the bases for critical decisions, are preserved for use later, perhaps decades in the future. Knowledge management is also used to identify and implement state of the art practices and lessons learned from international experience. Such management practices include information and data management, and the management of key assets.

3.1.5.4. Competence management

Another key management function is broadly described as ‘competence management’, which includes training, staff development, performance evaluation and succession planning, to cover the long term needs of the project. Effective management will have processes and controls to implement remedial actions when specialized knowledge is found to be deficient. On the business side, effective management requires strategies for procurement, especially for relatively high cost activities in the implementation phase. Design analysis is used to identify items or processes important to safety, and special controls are placed on supply chain management and contractor operations. In addition, methods of risk and opportunity management are often used to underpin management decisions, particularly where long project duration and key uncertainties (e.g. site conditions, funding, licensing) can impact project performance.

3.1.6. Transparent and traceable design

Maintaining a transparent and traceable design supports the design development process and the defensibility of the design and the decisions made that lead to the design solution. The importance of stakeholder concerns warrants a concerted effort to make project decisions transparent, traceable and based on reliable information. As an example, an essential component of the licensing strategy for the Yucca Mountain repository project in the United States of America was an extensive database (the Licensing Support Network), which was established to address this basic principle. The express purpose was to demonstrate that all aspects and all decisions taken during the development of the design and the licence application could be traceable and were transparent.

Transparency helps to develop and maintain mutual understanding and trust with all stakeholders including local communities, regulators, the public, and government agencies. Communication with stakeholders has been shown to be crucial for public acceptance. This includes well supported outreach, education and dissemination activities. Any public relations strategy needs to preserve the integrity and independence of experts who represent the design organization and might be called to provide testimonials for to the quality of design products. The IAEA is considering developing further guidance on communication and stakeholder involvement in radioactive waste disposal.

Audits can be performed to verify the quality of products such as design analyses, drawings and specifications. They can also be used to verify the bases for critical programme decisions and compliance with requirements, such as those related to quality assurance and effective use of management systems. Maintaining a transparent and traceable design addresses Principle 2 of Nuclear Energy Basic Principles [4].

3.1.7. Nuclear safeguards and security integrated design

From the outset, the design of a repository must incorporate arrangements for nuclear safeguards (i.e. activity and equipment used to verify that nuclear materials are not being diverted to military use) and security (physical protection, personnel security and information technology security).

Experience demonstrates that appropriate requirements and constraints for both can be integrated effectively within the repository design basis (see Section 3.1.1) in order to comply with international and national legislation, standards and good practice.

The early integration of technical considerations for safeguards and security will enable the identification and resolution of any potentially conflicting requirements: for example, where safeguards or security systems or barriers would impede emergency response or routine maintenance. The use of passive systems, intended to avoid human errors, can also make it more difficult for potential aggressors to tamper with safeguards and security systems [27].

3.1.7.1. Nuclear safeguards

The aim of nuclear safeguards is to detect and deter the diversion of nuclear materials (e.g. plutonium, uranium and thorium) from peaceful uses to nuclear weapons. Verification that materials from nuclear activities established for peaceful, civilian purposes are not being diverted to military use is one of the means by which the international community limits the spread of nuclear weapons. It is a fundamental aspect of the Treaty on the Non-Proliferation of Nuclear Weapons [28]. Non-proliferation is Principle 5 in Nuclear Energy Basic Principles [4].

Means of achieving Principle 5, particularly as related to design projects, are outlined in Nuclear Power Objectives: Achieving the Nuclear Energy Basic Principles [29].

The possible technological implications of IAEA safeguards on the various phases of a repository life cycle containing SF and other nuclear material subject to IAEA safeguards are considered in the IAEA Nuclear Energy Series No. NW-T-1.21, Technological Implications of International Safeguards for Geological Disposal of Spent Fuel and Radioactive Waste [30]. For example, the requirements of the safeguards regime during repository siting include submission of a design information questionnaire to the IAEA. The design information questionnaire contains information about the characteristics and operational procedures of the proposed repository, thus facilitating IAEA verification activities.

Verification activity relies on the principle of 'continuity of knowledge' to account for the nature, quantity and location of nuclear material. A Member State will need to provide a nuclear materials accountancy system to support this, which must cover all national nuclear related activities, including decommissioning, storage, transport and radioactive waste disposal.

3.1.7.2. Nuclear security

Provisions for nuclear security are intended to prevent or detect the theft, sabotage, unauthorized access, illegal transfer or other malicious acts involving nuclear materials and other radioactive substances. As noted by the United Kingdom Office for Nuclear Regulation [31], a repository is intended to integrate "robust protection against the modern threat environment" across all security disciplines: physical, personnel, transport, cyber security and information assurance.

3.2. DESIGN STAGES

The principles described in Section 3.1 are typically applied during the staged repository life cycle introduced in Section 3.1.4 and illustrated in Fig. 4. The timeline in Fig. 4 identifies the main phases and the major milestones common to repository programmes. It also identifies the five design stages introduced at the beginning of Section 3 and considered in more detail in this section:

— Generic designs (Section 3.2.1);
— Conceptual designs for siting (Section 3.2.2);
— Technical design for construction licensing (Section 3.2.3);
— Detailed design for construction and operation (Section 3.2.4);
— Continued detailed design for repository operation and expansion (Section 3.2.5);
— Design for closure (Section 3.2.6).

In Fig. 4, phases represent the expected evolution in the focus of a programme and not necessarily the conclusion of specific activities within a phase. Milestones represent the products of activities such as siting, licensing, construction authorization and operations, as well as decision points within major programme phases and associated activities. The timeline might require adaptation by a Member State to fit its specific circumstances, such as geological, financial and societal factors.

Activities within each phase could require several iterations (see Section 3.1.4) to achieve the required information to support a decision; for example, when developing the initial concept and adapting it to a potential site location. Some activities within one phase could overlap with activities in another phase. For example, in a modular disposal system, construction activities might be taking place in one area, with emplacement activities taking place in another. Similarly, closure activities in one area could overlap with construction and operation activities in another. Certain activities will continue at varying levels of intensity over the whole repository life cycle. These include aspects of monitoring and RD&D, which could be ongoing after repository closure.

The following subsections provide specific details of activities within each design stage, explaining inputs and outputs as the repository project moves forward, adapted from international experience.

3.2.1. Generic designs

The first design stage is carried out during the initiation phase (Fig. 4) of the programme, when generic designs will be developed to support the repository siting process. At this stage, the designs are likely to be conceptual in nature, as only the broad nature of potential siting options might be known. The flow of information into and out of a generic design process is illustrated in Fig. 11. Generic design concepts broadly address the requirements, assumptions and constraints detailed in an initial functional specification (see Table 1). Work is undertaken to develop the waste inventories and properties, and to identify potential disposal solutions. These form the basis for the initial generic design studies, perhaps for a range of disposal concepts and options for different categories of waste. The premise is to develop a safe and feasible system within existing constraints, such as any requirements already set by regulatory authorities. Usually, this early stage also includes developing an R&D roadmap.

The generic designs allow definition of associated generic safety cases (how the concepts being considered would provide safe disposal for the waste inventory) and provide a starting point for programme planning and estimation of duration, costs and risks.

3.2.2. Conceptual designs for siting

The siting process consists of both technical and societal components. The technical dimension considers geological and environmental conditions of potential sites and the interactions of those conditions with potential design options, which will directly influence the disposal concept and subsequently the design of components. The societal aspect considers a wide range of stakeholder interests and concerns and is ultimately critical to public acceptance of a repository. Stakeholder concerns need to be adequately addressed by the design. International experience has shown that successful siting processes reflect a balance between these two elements.

A generally accepted approach to siting could involve an initial site screening process followed by more intensive site characterization at one or more candidate sites. Site screening could include a review of existing geological, geographical and environmental information, together with socioeconomic factors, and might initially be based on exclusion criteria. As screening progresses, potential sites might be investigated at increasing levels

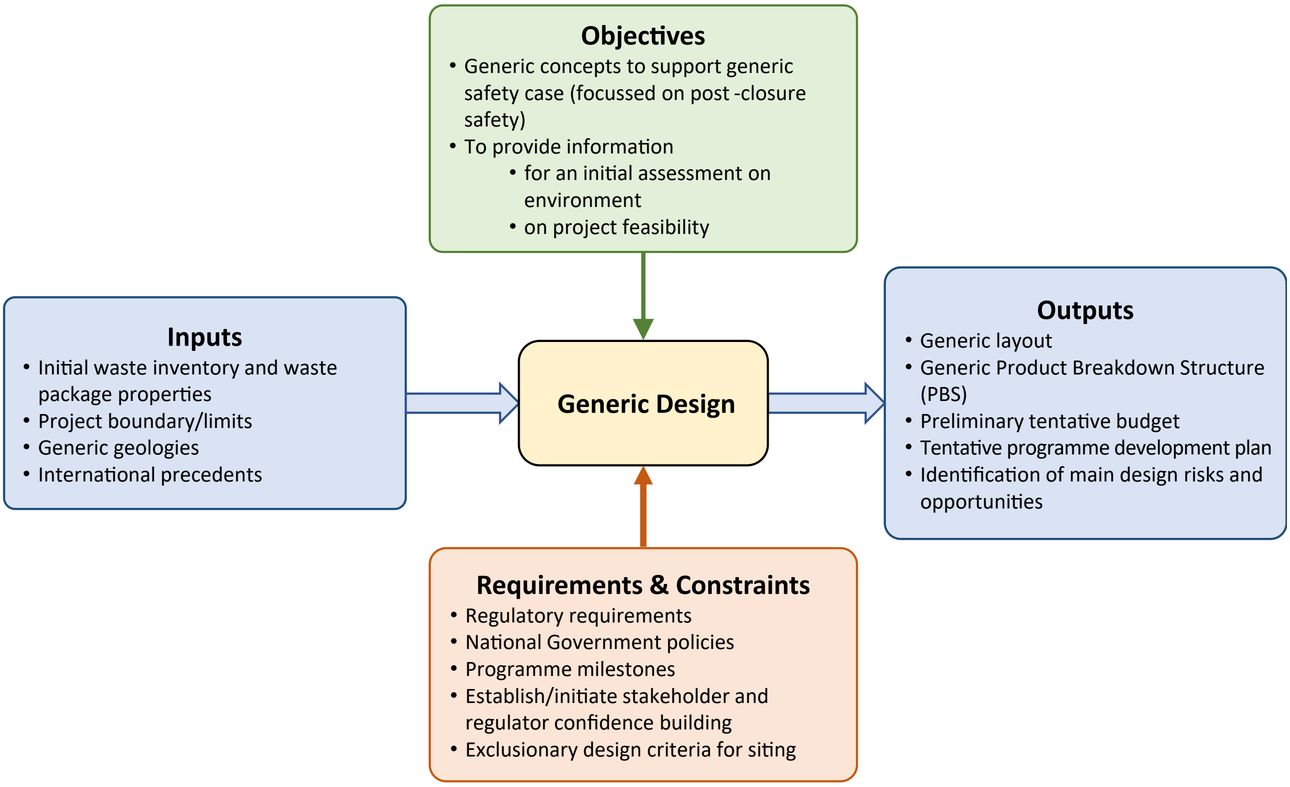

FIG. 11. Generic design: objectives, inputs, constraints, requirements and outputs.

of detail and assessed against favourable conditions for a repository. Ultimately the goal of screening is to identify a preferred site that meets a range of requirements, warranting more in-depth site characterization. Site characterization encompasses scientific and technical investigations aimed at providing an in-depth understanding of the geological and physical characteristics of the site and how the disposal system can effectively be integrated.

Throughout the siting process, the WMO will consider and develop design concepts specific to the waste inventory and the siting options emerging, augmented with a preliminary safety case for the design concepts that might be deployed at the sites (see Fig. 12). Existing disposal solutions for similar waste in other countries can be reviewed; compiling such solutions constitutes a basic purpose of this publication. If siting options are all in similar environments, then a single conceptual design might be appropriate for all of them. The conceptual design stage includes preliminary considerations of a basic layout that can accommodate the waste inventory at the sites under consideration.

The WMO will need to compare combinations of repository concepts, designs and siting options in order to decide which solution is the most appropriate for meeting the higher level programme requirements. Different repository solutions might be feasible, even if there is only one siting option available. Alternative solutions are likely to meet the different requirements to varying extents, although any option being considered must meet all of the critical requirements (e.g. regulatory or legal requirements) adequately. The process of considering and weighing the extent to which other requirements can be met is often termed ‘optioneering’ and the WMO can approach this in a flexible manner, involving other stakeholders as appropriate. A number of formal and less formal methods are available for weighing and ranking options so as to achieve a transparent decision on a preferred conceptual design/site combination. These include multi-attribute analysis and benefits/constraints comparison techniques.

The siting phase produces scientific documents such as geotechnical investigation reports and a preliminary environmental impact assessment, and, typically, a preliminary safety analysis report. These documents provide the basis for the selection of a preferred site (and an associated preferred repository concept), which will generally require the approval of regulatory authorities and/or government. At this stage, detailed coordination with local and regional stakeholders occurs, and schedule estimates and quality levels can be better established. The design programme should include a process and schedule for consultation and open discussion with all stakeholder groups.

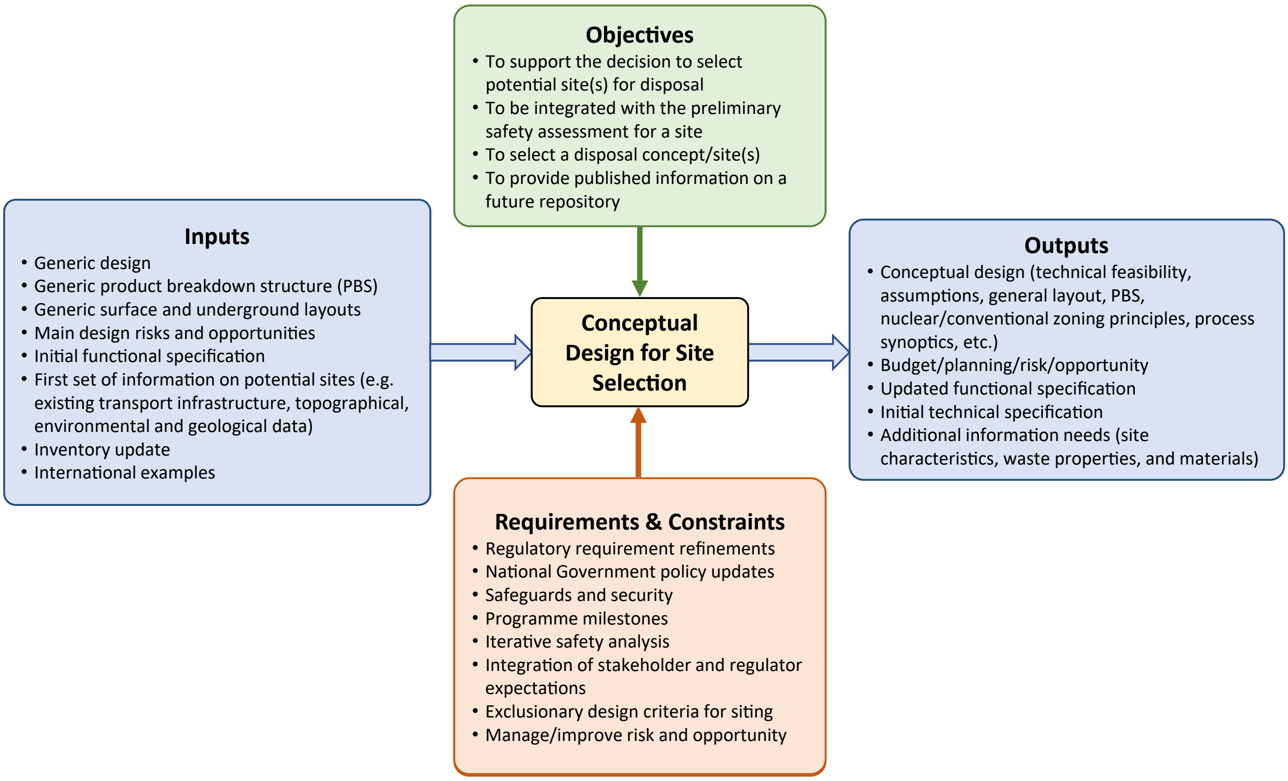

FIG. 12. Conceptual design for siting: objectives, inputs, constraints, requirements and outputs.

3.2.3. Technical design for construction licensing

The technical design for initial construction licensing at the chosen site (see Fig. 13) follows from the conceptual design. This stage develops the chosen design, based on the specific characteristics of the selected site, sufficiently for construction licensing application; i.e. the design must be sufficiently detailed to comprehensively demonstrate that the system can be constructed at the site and fulfil all licensing requirements imposed by regulation.

During this phase, models and analyses are developed and drawings, technical reports are completed with design control and quality management protocols in place. Technical bases are confirmed. Timing, scheduling, requisite resources and external interfaces are defined. The technical design expands on the conceptual design and incorporates additional site specific information obtained during characterization activities. The technical design would also be used to consult with key stakeholders before moving on to the detailed or final design.

Provisions to address closure requirements are identified at this design stage. Data will be provided for safety assessments and interactions with the regulator will take place. A description of active and passive institutional controls may be required. Provisional proposals can be made in the design for future extension of the site (both surface and subsurface) to accommodate the possible construction of new disposal structures and capabilities. Monitoring provisions for performance confirmation will take shape and define how future information will be evaluated against the existing technical basis. Completion of this phase might be marked by the issuance of an intermediate safety analysis report.

3.2.4. Detailed design for construction and operation

The detailed design for construction and operation (Fig. 14) entails complete drawings and reports containing the final technical design, considering detailed information on-site, environment and waste package throughput and capacity requirements. Workshop drawings, equipment specifications and detailed instructions must be appropriate for equipment procurement, commissioning of the facility and construction. Information communication, system integration, schedules and management systems are operationally ready.

The main objective of the detailed design stage is to prepare for the construction and operation phases and to provide information to support the safety assessment undertaken for licensing purposes. The safety case and detailed design confirm that the disposal facility can be operated and closed safely and efficiently. Satisfactory

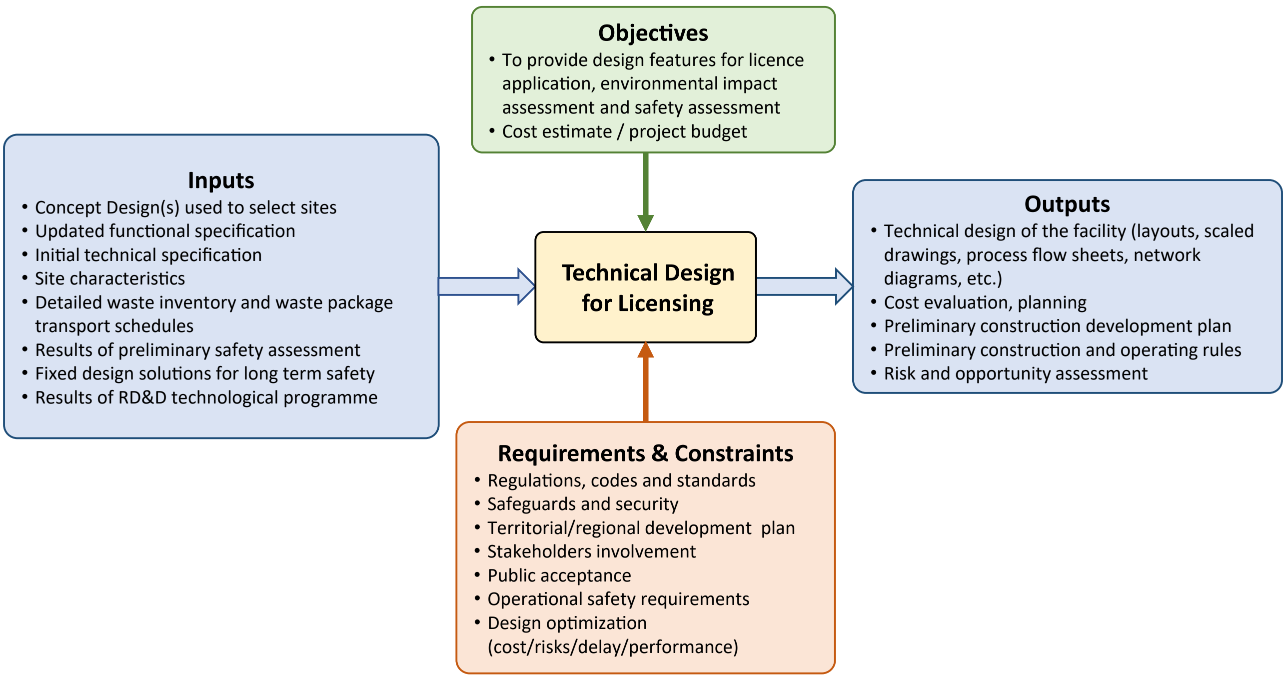

FIG. 13. Technical design for initial construction licensing: objectives, inputs, constraints, requirements and outputs.

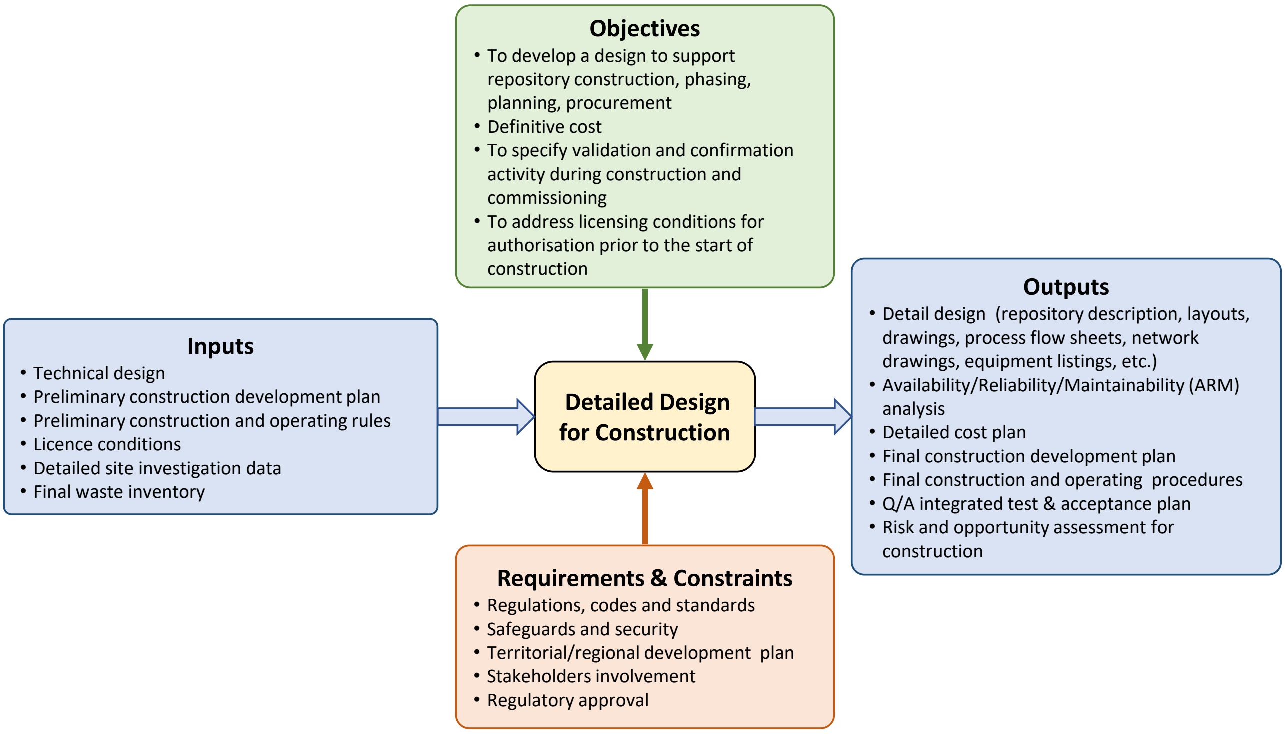

FIG. 14. Detailed design for construction and operation: objectives, inputs, constraints, requirements and outputs.

interaction with the regulator provides the basis for commencing construction and, subsequently, operations. The regulatory authorities might introduce conditions for certification.

During this stage, detailed cost estimates for facility construction, operation and closure are prepared. Environmental surveillance and radiological monitoring programmes to be conducted during operations and after closure of the disposal facility are identified and concerned parties are informed regarding requirements for final closure of the facility. Specified information requirements from construction activities and planned interface arrangements with construction and operations and schedules are finalized, to facilitate requests for bids, particularly in relation to design change management and 'as-built' records.

Once sufficient infrastructure and the initial disposal areas of the repository have been constructed and inspected to the regulators' satisfaction, a licence for operations can be issued. Construction documentation to support this process is divided into two groups. The first group shows that the technical requirements from the requirements management system and the design basis have been met by the final design of the constructed system. The second group contains as-built drawings, certificates of materials, equipment, declarations of conformity, changes, protocols, variance from the certified licence and detailed design. Completion of this phase might be marked by the issuance of a final safety analysis report, a licence for construction and contract(s) for work.

3.2.5. Continued detailed design for repository operation and expansion

The disposal phase (see Fig. 4) is considered to include construction, operation, and closure. It is initiated by the authorization to construct and generally has the greatest level of activity and, depending on its duration, the most significant element of total disposal costs. During construction, additional information will likely have been gained on the geological environment, particularly in deep geological repositories. This additional information will be integrated into the design and confirmed with respect to the safety case to ensure the mitigation of potential negative impacts on performance. As disposal progresses, areas of the repository will be filled and at least partially closed. The regulatory authorities will require confirmation that their requirements continue to be met as work progresses into new disposal areas of the repository.

Depending upon the volume of wastes and the rate at which they are delivered to the repository, a staged, modular approach can be an efficient means to manage disposal. It also allows designs for later phases of disposal to be updated, taking into account continuous learning and improvement. This approach can provide improved information on conditions at closure, relevant to long term safety, while at the same time aiding in control of capital investments versus routine expenditures.

As discussed in Section 3.2.4, a separate authorization is often required to begin disposal operations after commissioning activities have certified operational readiness. Depending on the regulatory system, an operating licence might thus be required. Even after operations have been initiated, the design process is likely to continue as optimization needs are identified and implemented. Owing to the multidecade operational period of many repositories, periodic updates to the safety case are likely to be required by regulators, perhaps every five or ten years, to account for new information gathered from operational experience, general technical advances and any changes in disposal procedures or engineered system designs. At the end of this period, the WMO will seek a licence to close the repository, accompanied by a closure safety report.

3.2.6. Design for closure

Sections of a repository might have been closed and sealed during its operational lifetime, as operations move to new disposal areas. Final closure occurs after all the waste has been emplaced. The basis to ensure safe closure capability is demonstrated prior to the start of emplacement, possibly using full scale tests. The final closure design (Fig. 15) might continue to be refined and tested throughout the disposal phase, as it might not be implemented until many decades into the future. The disposal phase is considered complete following approval of the closure safety report by the appropriate authorities, which will confirm that the long term, post-closure behaviour of the repository will continue to conform to regulatory requirements. It will be accompanied by agreed arrangements for post-closure institutional control and monitoring and the allocation of long term responsibilities for the site up to the end of institutional control and beyond.

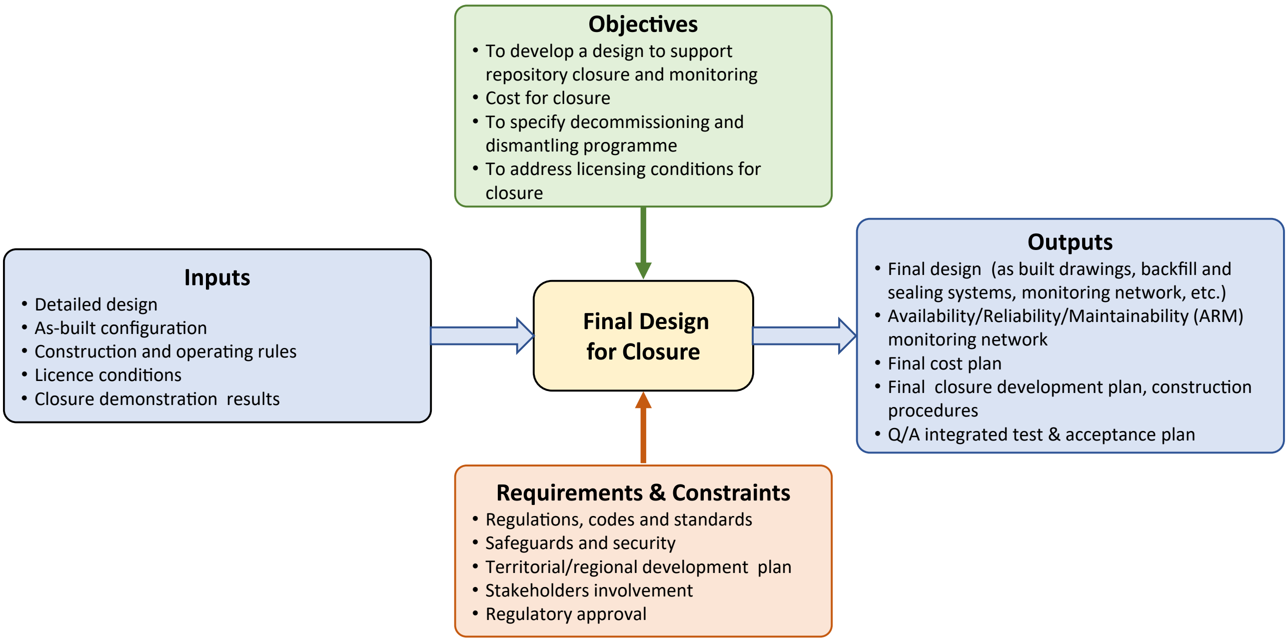

FIG. 15. Final design for closure: objectives, inputs, constraints, requirements and outputs.

4. EXAMPLES OF OPERATIONAL OR ADVANCED REPOSITORY DESIGNS

This section provides a series of examples of repository designs that are either licensed and in operation or are at an advanced stage of development and ready for operational licensing. The examples cover all of the main types of repository design concept illustrated in Fig. 2.

4.1. EARTHEN TRENCHES

A considerable level of international experience has been gained over the last several decades in the design, construction and operation of earthen trench disposal systems for radioactive waste. These systems are similar in many aspects to conventional municipal or industrial waste disposal facilities. It is noted that while referred to as ‘earthen trenches’, these types of facilities can also be constructed above ground as mounds. Both trench and mound systems are considered suitable for the disposal of waste with limited isolation requirements, generally for several decades. However, with additional design consideration, longer isolation periods can be achieved. The facilities are best suited for the disposal of VLLW and, given favourable site properties and provision of adequate lining and capping systems, they can also be suitable for LLW. Examples of simple trench or mound disposal systems are shown in Fig. 16.

4.1.1. General considerations

The main factor that makes this type of simple repository appropriate for disposal of these wastes is the limited radiological hazard they pose. Safety is provided by favourable site characteristics, the use of engineered barriers, careful management and operation and the implementation of institutional controls. Sites with properties that significantly limit the accessibility of water to the waste can be favoured in the selection process. Ideally, site characteristics will act to limit potential releases of radionuclides until radioactivity in the waste has decayed to safe levels. Trench and mound disposal facilities are well suited for arid climates, especially when located at sites with deep water tables.

When site conditions alone do not provide adequate isolation (e.g. for sites located in wetter climates, in areas with shallow groundwater or that are susceptible to events capable of causing significant releases), additional engineering measures can be designed into the facility to enhance site performance. In locations with shallow groundwater, barriers such as high density polyethylene sheeting, clay, bentonite or other appropriate materials can be used to line the walls and base of the trenches. The inclusion of lining systems is intended to limit the movement

FIG. 16. VLLW disposal: trench at El Cabril, Spain (courtesy of Enresa) (left) and mound at Oskarshamn Nuclear Power Plant, Sweden (courtesy of Svensk Kärnbränslehantering) (right).

of water from the repository and thus the release of radionuclides to the environment. In wetter climates and sites vulnerable to disruption by rare events (e.g. millennial floods) additional protective measures might be required, to ensure maintenance of the waste containment and isolation functions.

Additional protection from the impacts of precipitation and other adverse conditions can be provided during emplacement, to further minimize the introduction of water into the system. Temporary covers, including mobile roofs or removable structures installed over the emplacement site, can provide adequate protection. In arid areas such measures might not be necessary, although dust control can become an issue.

When disposal operations are complete, stable, long term covers (capping systems) are installed, with sufficient impermeability to protect the waste. The final cover serves to limit adverse climate effects by reducing water infiltration and limiting the effects of temperature variations. More complex lining or capping systems can be designed to ensure added isolation and containment. For instance, when LLW is disposed or when repositories are constructed in areas with less than favourable climatic conditions, multilayer engineered capping systems can be designed to divert precipitation water and prevent its infiltration into the disposed waste volume. Monitoring systems can detect breaches in isolation during the operational and active supervision periods, allowing restorative or remedial measures to be taken, if needed. Specifically, sumps have been employed in conjunction with access shafts or other drainage collection systems to intercept potentially contaminated water before it is released into the environment.

Engineered structural support systems are normally not required for these types of facilities because the waste itself is structurally competent. Package requirements for VLLW are limited; waste can be disposed of in bulk form or in simple waste containers used for transport. LLW will generally require a more complex waste container that also assumes some of the safety function for the repository. Void spaces in the repository are filled using an approved backfill material. VLLW and LLW can generally be disposed of using conventional package and container handling equipment, without radiation shielding. Waste can be transported directly to the trench or mound by truck, forklift or other standard haulage means. Attention to the stability of trench boundaries and sidewalls is an important consideration to avoid slope failures that could trigger accidents.

Following final closure, a period of active supervision and institutional control is necessary for these types of disposal systems. During the period of institutional control, repairs can be implemented if needed to ensure that the waste remains isolated. The length of institutional control depends on the radionuclide content of the waste, as well as site conditions. The timeframe can range from several decades to over a hundred years for VLLW or a few hundred years for LLW (300 years is often applied).

Internationally, two basic approaches to managing these types of facilities have been followed; waste is either disposed of at one or more centralized repositories or at decentralized, facility specific repositories. For example, the French VLLW repository, Centre industriel de regroupement, d'entreposage et de stockage (CIRES), located in the Aube Department, serves as a centralized disposal facility supporting all national VLLW disposal needs. In contrast, Sweden utilizes decentralized disposal facilities located adjacent to each of their nuclear power plants.

4.1.2. Disposal concept

A cross-section of a typical trench style repository is shown in Fig. 17, specifically the VLLW disposal trench as implemented at CIRES in France, which is described in more detail in Section 4.1.2. Figure 17 illustrates some of the components in this type of facility and is not intended to represent a specific design preference.

The trenches at the Morvilliers site are located in a thick, low permeability alluvial layer consisting predominantly of clay. The sides and base of the trench are lined with an impermeable geomembrane as an additional engineered barrier to prevent water from exiting the trench. A geotextile can be placed over the membrane to protect it from damage during waste emplacement and backfilling. The performance of geomembranes in excess of several decades or at most one hundred years cannot be guaranteed by current technologies. Therefore, if longer term safety functions are placed on the liners, alternative barrier systems such as bentonite or other clay layers can be considered to enhance performance.

The trench is designed and constructed with an overall slope that provides drainage to a single collection point, i.e. a sump. The base of the trench is completed with a permeable aggregate layer allowing drainage of water to the collection sump. The aggregate material is specifically selected to minimize clogging of pores, thereby maintaining the permeability of the drainage layer. A maintenance shaft provides access to the sump. The sump is installed in a concrete footing that provides adequate stability and support for the access shaft. Assuming

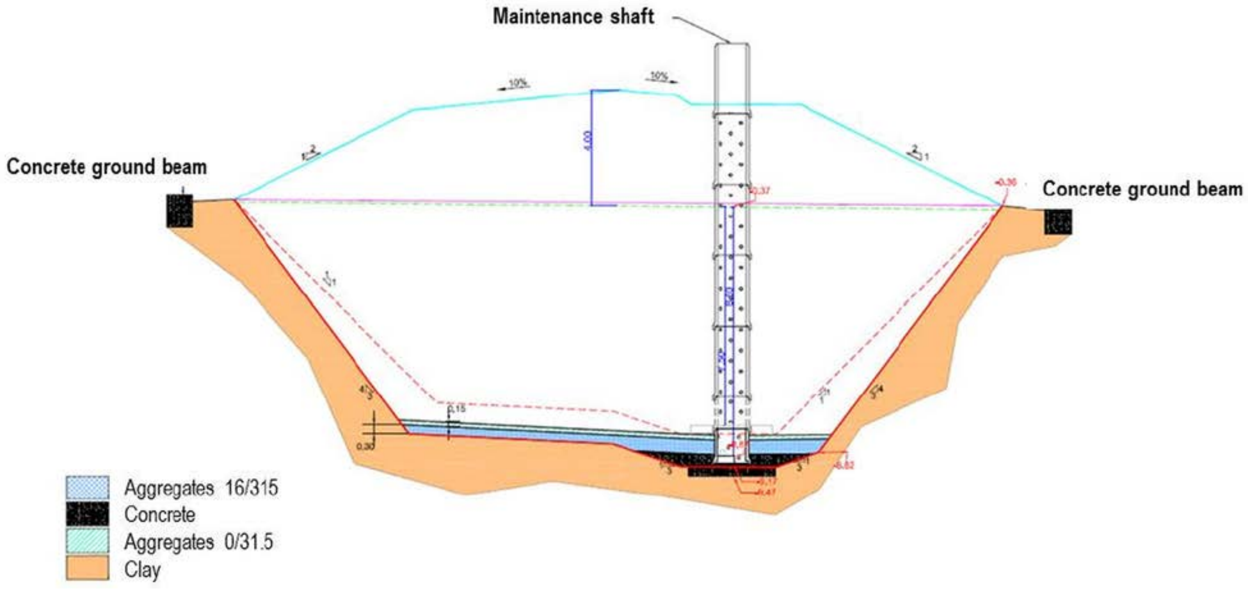

FIG. 17. Cross-section of a typical VLLW disposal trench (courtesy of Agence nationale pour la gestion des déchets radioactifs (Andra)).

appropriate performance of the disposal system, water will not be detected in the sump. However, if water is detected, a sample will be collected and analysed for the presence of radionuclides. Depending upon the results of the analysis, remedial actions could be required.

The trench is protected against precipitation during waste emplacement operations by erection of a temporary cover. An adequate foundation to support the structure is provided by concrete beams. After all the waste has been emplaced, a final impermeable cover of clay or other low permeability earthen material is installed to limit infiltration water. More complex capping systems with alternating drainage and low permeability layers might be required, along with additional protective layers, to limit plant rooting or damage from burrowing rodents, for example. A long term stabilization layer, which can be provided by a vegetative cover, is the final component installed over the facility.

4.1.3. Representative examples

Representative examples of earthen trench type repositories are the VLLW repositories at Morvilliers in France and the LLW repository at Vaalputs in South Africa. These examples are described in more detail in the following sections.

4.1.3.1. CIRES Center for VLLW Disposal, Morvilliers, France

The French National Radioactive Waste Management Agency (Agence nationale pour la gestion des déchets radioactifs (Andra)) operates CIRES in the Aube district of France near Morvilliers. CIRES began accepting VLLW in 2003 and has a design capacity of 650 000 m³. It is located on 45 ha of land, of which 28.5 ha are exclusively dedicated to disposal purposes. The average annual volume of waste accepted for disposal each year is 24 000 m³. The waste originates from French nuclear facilities currently in operation or undergoing decommissioning.

About 30% of the waste received at CIRES undergoes some pretreatment prior to disposal. Solid forms such as low density residues and metal scrap are compacted to reduce the overall volume. Liquid wastes in the form of contaminated water and sludge are processed in a solidification and stabilization unit.

A temporary Premorail shelter, a modular metal frame and tarpaulin structure (Fig. 18), is placed over the trench to provide protection against adverse weather conditions during emplacement. The shelters are designed for reuse and can be disassembled after waste emplacement and prior to final closure of the trench.

The host formation for the CIRES facility is a homogenous clay, which varies in thickness from 15 m to 25 m. A cross-section of the trench style disposal cells used at the CIRES facility is shown in Fig. 19. The repository

FIG. 18. Premorail shelter over the filled trench of CIRES facility (courtesy of Andra).

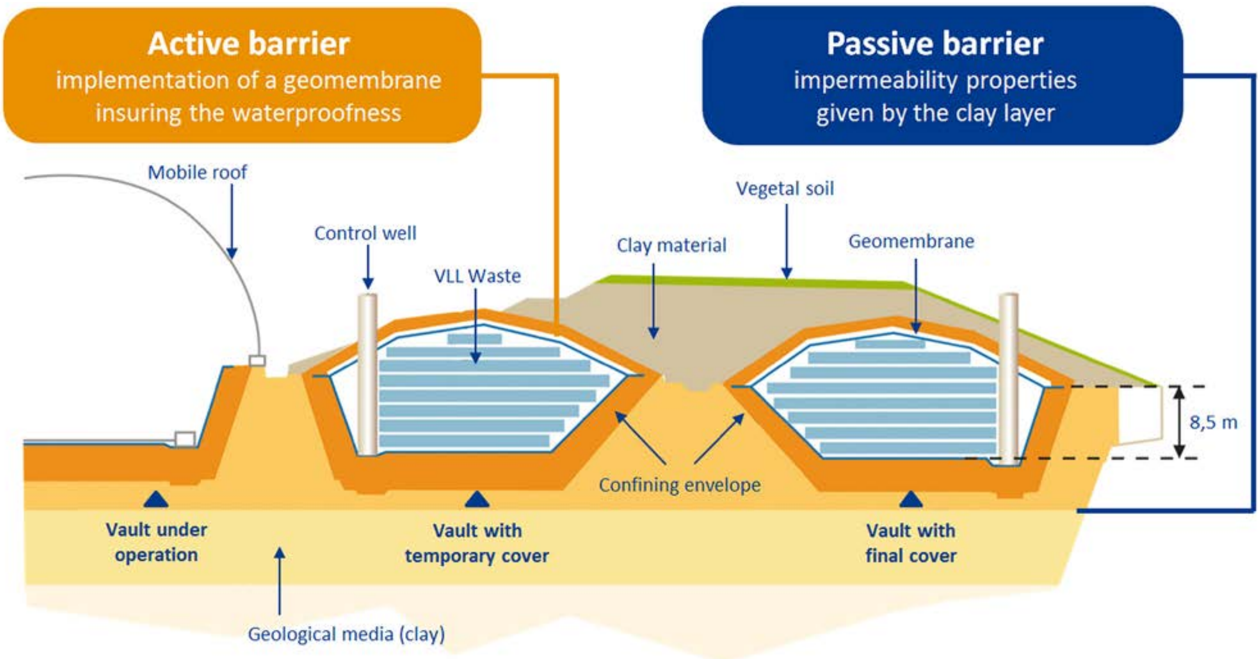

FIG. 19. Cross-section of the trench disposal cells implemented at CIRES (courtesy of Andra).

safety concept is a multibarrier approach that integrates the natural and engineered barrier systems. Several barriers serve to isolate and contain the waste:

— A 2 mm thick high density polyethylene geomembrane is used to line the floor and sides of the trench, forming a continuous watertight barrier to impede radionuclide migration for several decades. An identical membrane is used to cover the emplaced waste after the trench has been filled. The membranes are thermally welded to one another to fully encapsulate the waste. After completion of the encapsulation the temporary shelter can be moved to the next trench.
— A containment envelope is formed by the natural clay surrounding the excavation and by emplacement of a compacted clay layer with a minimum thickness of 1 m over the filled trenches, using previously excavated material. The clay cap is compacted to achieve a permeability similar to that of the in situ clay.
— Clay backfill with a minimum thickness of 2.5 m protects the sealed trenches from potential external damage from weathering, burrowing animals or plant roots, and from erosion. All backfill materials originate from previous excavation at the site.
— A final 30 cm thick vegetative cover is installed over the closed facility to act as a stabilizing layer and minimize potential negative erosional effects.

Air quality and groundwater monitoring, including monitoring for potential radionuclide release, are conducted throughout the operational life of the CIRES facility. Following the planned 30 year operation period, the CIRES facility will enter a subsequent 30 year post-closure monitoring phase, after which the further status of the site will be reassessed.

4.1.3.2. Vaalputs, South Africa

The Vaalputs repository is the only licensed facility for the disposal of LLW both in South Africa and in the African continent itself. LLW, as defined in the IAEA classification scheme discussed in Section 2.3.1, is disposed of in a near surface repository located in the Northern Cape Province. The Vaalputs facility covers an area of approximately 100 km². Disposal operations are currently concentrated in an approximately 1 km² area located in the western half of the site. Only 11% of the current disposal area has been used to date. The repository design makes this area suitable for disposing of 142 390 m³. The total capacity of Vaalputs over the entire 10 km² disposal area is therefore 1 423 900 m³. The site is shown in Fig. 20.

Vaalputs is owned and operated by the National Radioactive Waste Disposal Institute and is used primarily for the disposal of LLW generated at the Koeberg nuclear power plant and the South African Nuclear Energy Corporation. The Koeberg nuclear power plant waste consists essentially of compactable and non-compactable waste such as redundant equipment, filters, spent ion exchange resins, evaporator concentrate and contaminated paper gloves, plastic and coveralls. The South African Nuclear Energy Corporation waste currently disposed of at Vaalputs consists of solidified medium active concentrates, but other types may also be considered in the future.

The current disposal concept at Vaalputs uses trenches that are a few metres deep, situated above the groundwater table. The repository safety concept is a multibarrier approach that integrates the natural and the engineered barrier systems. The natural barrier system consists of low permeability clayey soils and a deep groundwater level. Safety assessments demonstrate that the concept is effective in isolating the waste from the environment. The effectiveness is enhanced by the containment properties of the engineered barrier system, which comprises the waste package, consisting of the immobilized waste after conditioning and the waste container.

Waste is disposed in specially designed trenches excavated to a total depth of about 8 m in the unsaturated clayey soils. The soil deposits comprise an upper red clay, which includes a large percentage of kaolinite and montmorillonite and has a thickness of between 10 and 15 m, and the lower white clay, with a 15 m to 20 m thickness, consisting primarily of fluvial deposits with clays. The soil deposits are underlain by granitic bedrock. The unsaturated zone is between 50 and 55 m thick. The main features supporting the design of the repository are the dry climate (average annual rainfall of about 125 mm), the depth to groundwater and the composition of the clayey soils. Even after heavy rainfalls, water has not been observed to penetrate more than a few metres below the surface. The majority of penetrating water is returned to the atmosphere by evaporation, which has resulted in the formation of a near surface calcrete layer that is overlain by a reddish surficial sandy layer.

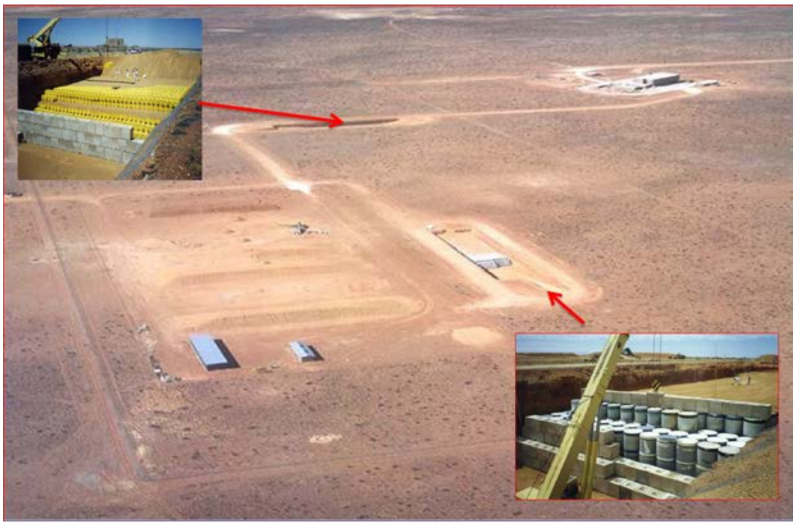

FIG. 20. Aerial view of the Vaalputs repository [32] (courtesy of the National Radioactive Waste Disposal Institute, South Africa).

Waste packages are stacked inside the trenches in such a manner as to optimize the available space in each trench in the most economical way and to reduce the possibility of slumping, settlement or collapsing within the trenches. The position of each package is accurately recorded and entered into the waste tracking database. Where possible, voids between packages are filled with dry, screened, natural material previously excavated from the same trench. A layer of hard cobbles is used to cover the backfilled trenches in order to deter burrowing animals, after which a cap is constructed using moistened and compacted, screened, natural material, previously excavated during trench construction. A thin layer of tracer material is included (fine copper slag) in the cap, to monitor termite activity, and a final layer of topsoil is emplaced, to facilitate plant growth over the cap. Trenches are constructed on an as needed basis.

Currently the facility consists of two operational disposal trench types with one of each trench type in operation: the A Trenches are utilized for metal waste packages and the B Trenches are utilized for concrete packages. A cross-section of a typical A Trench is shown in Fig. 21. The trenches are about 400 m apart. Each trench is about 100 m long, 7.7 m deep and 20 m wide at the base, with upward sloping sidewalls. Waste disposal is optimized following a compartmental, stepwise approach, where each compartment is filled and capped within 18 months of the first waste emplacement.

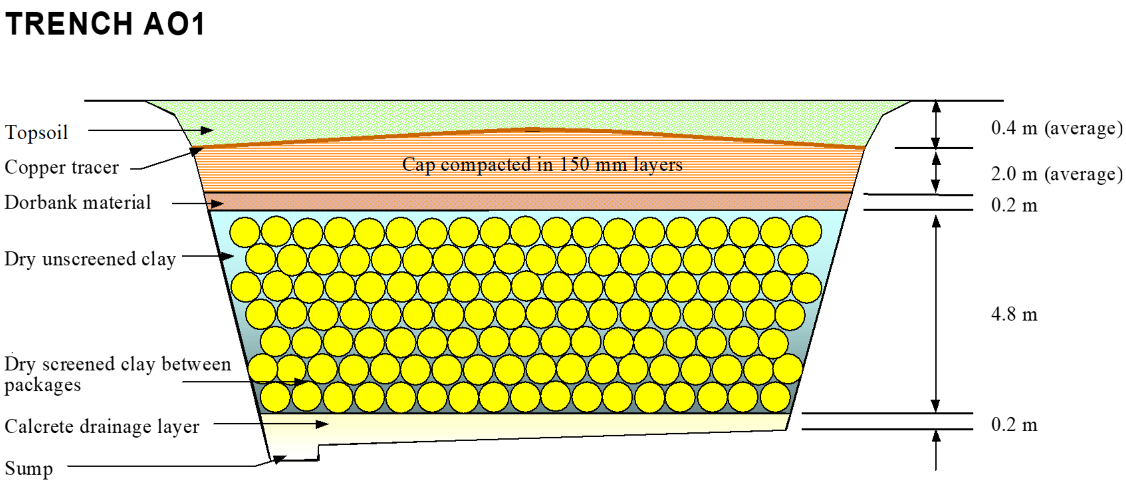

FIG. 21. Cross-section of a typical disposal trench at the Vaalputs Disposal Facility (courtesy of the National Radioactive Waste Disposal Institute, South Africa).

After all waste has been disposed of, the Vaalputs Disposal Facility will be closed. The trench area will be restored to its original topography and native vegetation will be re-established. Post-closure monitoring will be carried out. Unrestricted access to the Vaalputs site is presumed to be possible following a period of institutional control of 300 years.

4.2. NEAR SURFACE ENGINEERED STRUCTURES

Near surface engineered structures are constructed with more engineered barriers than earthen trenches, and thus provide a higher level of containment and isolation of waste. They can provide a robust solution with a wider range of application than simple earthen trenches. A typical solution that has been implemented in a number of Member States is the concrete vault. These systems are generally constructed at the surface and subsequently covered with an engineered cap. Advances have been made in several Member States in the licensing, construction and operation of near surface concrete structures for the disposal of LLW.

4.2.1. General considerations

The decision between a trench or a more complex concrete structure for disposal will depend on many factors, including climate, site conditions, the hazard posed by the waste, public acceptance and national policy, as well as funding considerations. These factors will each need to be evaluated for their relevance to the disposal need and appropriateness for application in each Member State.

Several disposal facilities for LLW have been licensed, constructed, commissioned and taken into operation, demonstrating the effectiveness of near surface engineered structures. These systems have proven particularly appropriate for use in non-arid climates, or in areas where the performance of the natural system, or national policy considerations, require the application of additional barriers.

A common approach is to combine engineered barriers such as reinforced concrete structures with specially designed water management and drainage systems. The concept relies on minimizing the access of water to waste by a combination of measures designed first to divert water away from the waste and then to prevent or significantly limit the direct access of water to it. Various monitoring concepts have been developed to confirm the performance of the barriers. These monitoring concepts are an essential element in demonstrating the adequacy of the safety case and enhancing regulatory and public confidence in the disposal facility.

Although bulk waste can be disposed in appropriately designed and constructed structures if allowed by WAC, most waste will generally be conditioned and packaged prior to disposal. A wide variety of packages are in use, ranging from simple steel drums to large, specially designed, reinforced concrete containers. International Organization for Standardization (ISO) containers are also used for disposing of LLW at some facilities. The waste packages contain and protect the waste during handling and transport. The performance of the package with respect to long term safety post-emplacement can vary as a function of the safety case for the specific repository.

4.2.2. Disposal concept

The disposal concept can be illustrated by consideration of the National Disposal Facility implemented by the Bulgarian State Enterprise Radioactive Waste, near the Kozloduy nuclear power plant (Fig. 22), which began construction in August 2017. The concept is derived from the design of the El Cabril repository in Spain, which in turn is based on the French design implemented at the Centres de stockage de l'Aube. The concrete vault concept has proven to be a flexible design solution and several variations have been adapted for implementation. Similar concepts have been built or are also under development in various Member States, including Belgium, China, the Czech Republic, India, Japan, Poland, Slovakia and the United Kingdom. The example presented in Fig. 22 is intended to illustrate some of the major considerations in the design of these types of facilities and is not intended to represent a specific design preference. Alternative designs providing similar capabilities include the surface access silo concept under development in Slovenia and described in Section 4.2.3.3.

The engineered structures are constructed from reinforced concrete to create chambers into which the waste can be emplaced. After emplacement, a concrete roof is installed to seal the chambers and prevent access of water to the waste. Concrete vaults constructed at the surface, as shown in Fig. 22, are typically waterproofed to further

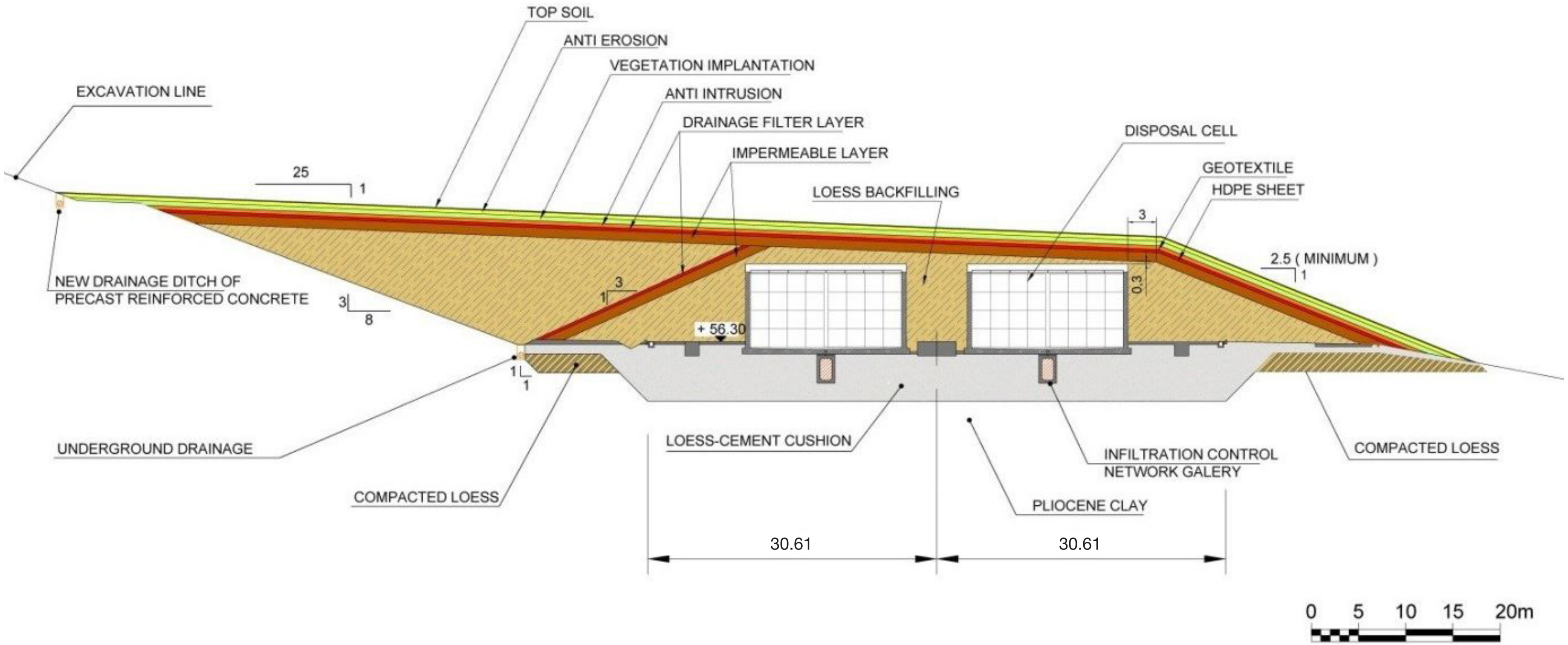

FIG. 22. Design concept developed for the Bulgarian National Disposal Facility . The figure is a cross-section of the cover, showing the shed cover and one side embankment with a multilayer engineered barrier and additional clay barrier inside the backfilling. All measurements are in meters. (Courtesy of Bulgarian State Enterprise Radioactive Waste.)

protect the waste and minimize the potential for water entering the system. After filling and approval for closure, a final engineered cover is installed, which is intended to serve to divert water away from the waste and to act as a barrier to seepage, greatly reducing the potential for percolating water to reach the waste. Some designs also assign a barrier function to the waste packages, although typically this safety function is limited to the operational period.

To provide confidence and confirm the performance of the disposal system, a monitoring system intended to capture any potentially contaminated water is typically included in the design. Such systems can include floor drains combined with inspection galleries installed beneath the disposal system, such as that shown in Fig. 22, or other configurations intended to capture any water that could potentially have encountered the waste. In a properly functioning disposal system, water should not be detected.

In the example shown, the floor of the vault is designed to drain towards a central drainage structure within each chamber. Each drain is fitted with a sample trap, installed in an inspection gallery beneath the disposal vaults. The gallery allows access to the sample traps throughout the period of institutional control. Thus, any potentially contaminated water will be captured by the drainage system to allow analysis for radionuclide content.

The specifications for the concrete used in construction of the vaults are rigorously controlled to ensure its required performance. The concrete can also provide radiation shielding throughout the operational life of the disposal system, until the vaults are sealed and the final engineered cover is installed. Radiation protection functions assigned to the concrete during the operational period are optimized based on a combination of the selected concrete formulation and thickness. In addition, the concrete must account for the long term requirements placed on the disposal system, as determined in the safety assessment. A key function is the maintenance of vault integrity, with criteria being applied with respect to crack size under the given and expected environmental conditions, including the expected chemical environment.

The long term performance requirements for these types of disposal system are generally predicated on a period of institutional control after closure. Requirements placed on the isolation and containment function of the disposal system will be determined by the radionuclides in the inventory. As with earthen trenches for LLW, a commonly used duration of about 300 years (equivalent to approximately ten half-lives of ^{90}Sr and ^{137}Cs) is often applied for the period of institutional control.

The primary barrier function in this multibarrier system — specifically the prevention of water from contacting the waste — is assigned to the engineered components. The performance of the natural system is also important to the overall safety case and site geological conditions therefore play an important part during the site selection process. Generally, sites located above local flood plains with seismically acceptable profiles and with low groundwater flow conditions provide preferential conditions.

4.2.3. Representative examples

Representative examples of near surface engineered structure repositories can be divided into two basic design concepts: concrete disposal vaults, of which numerous different examples have been developed, and direct access, near surface concrete silos, such as the concept recently adopted in Slovenia. In the following subsections, two different examples of concrete vault design solutions are described, based on the solution implemented at El Cabril, Spain, and the solution implemented at Mochovce, Slovakia. The Slovakian design is a further development of the Dukovany repository design operated in the Czech Republic and is presented in this context. In addition, the Slovenian silo concept, which is currently being licensed for construction, is also described. As previously noted, examples are provided for reference only and alternative design solutions can also prove suitable.

4.2.3.1. El Cabril Disposal Facility, Córdoba, Spain

The Spanish WMO Enresa (Empresa Nacional de Residuos Radiactivos) operates the El Cabril repository, located near Córdoba on a site previously developed for uranium mining. The disposal facility occupies about 20 ha of land on a property of about 1100 ha [33] and comprises both a trench style repository for VLLW and a concrete vault style repository for the disposal of LLW.

The safety approach combines engineered barriers in the form of reinforced concrete structures with specially designed water management and drainage systems. The concept is based on initially diverting water away from the waste during an operational control period and subsequently precluding or significantly limiting direct access of water to the waste.

The site is located on pre-Cambrian metamorphic rocks consisting mainly of migmatized biotitic gneisses and micaceous schists. It is drained by a stream, the Arroyo de la Montesina, which flows to the river Bembézar. The climate of the area is characterized by mild rainy winters and hot summers. The average annual precipitation in the Bembézar drainage basin can vary from 400–450 mm in dry years, to 600–700 mm in average years, and up to 850–950 mm in wet years.

The vault disposal system for LLW comprises two disposal platforms: a northern platform consisting of 16 disposal vaults and a southern platform with 12 vaults. The northern platform has been filled and waste disposal operations have moved to the southern platform. The facility has an LLW disposal capacity of 100 000 m^3, corresponding to approximately 35 000–50 000 m^3 of primary waste packages delivered from producers, depending on the waste types.

El Cabril accepts waste generated from any location within Spain. Most of the waste arriving at the facility originates from nuclear power generation and arrives in a conditioned form. Some waste from medical or industrial usage is treated at a conditioning facility on-site. Conditioned waste, generally contained in steel drums, is placed into large concrete waste packages for disposal. These waste packages have external dimensions of 2.25 m × 2.25 m × 2.20 m and can hold up to 18 drums of 220 L. After the drums are emplaced inside the waste packages, grout is injected to fill void spaces and the packages are sealed with concrete lids. The grouted and sealed waste packages can weigh up to 24 t and are placed inside the concrete disposal vaults (Fig. 23), with each vault capable of receiving 320 waste packages. The waste packages are placed in contact with one another in such a manner as to leave a central cross-shaped void to allow for minor tolerance difference in placement of the packages or in their manufacture. The external dimensions of the vaults are 24 m × 19 m horizontally and 10 m in height. The wall thicknesses are 0.5 m. The base of the vaults is provided with a slight slope to ensure drainage to a centrally installed drain. The thickness of the base slab ranges from 0.5 m to 0.6 m. A layer of permeable concrete is installed to provide a flat surface for waste package emplacement. After filling, the vaults are closed by installation of a reinforced concrete roof slab.

On both the southern and northern platforms, the disposal vaults are arranged in two parallel rows. The southern platform houses two rows of six vaults and the northern two rows of eight vaults. Each of the 28 disposal vaults is equipped with a floor drain that is connected to a seepage control and monitoring network, installed in galleries that run beneath each row of vaults. The seepage control system allows the early detection of potentially contaminated water leaking from the vaults. Detection of water, although unlikely, would be indicative of a potential fault in the disposal system and could result in the implementation of remedial actions to repair the facility, if found necessary.

FIG. 23. Filling a vault at the El Cabril LLW repository (courtesy of Enresa).

Two mobile roofs equipped with nuclear rated cranes are used to emplace waste on the operational platforms. The mobile roof is parked over a vault throughout emplacement operations and acts to protect activities from unfavourable weather conditions. The roofs remain in position over the vault until it has been filled and the concrete roof slab has been installed. The operation of two mobile roofs allows continuous disposal; while one vault is being sealed, disposal operations can continue in the other. The same strategy has been adopted at the Bulgarian National Disposal Facility, which uses El Cabril as a reference model. The total capacity of the LLW repository is 50 000 m^3.

The waste packages and the disposal vaults are designed to withstand seismic events with a ground acceleration of 0.24*g*. The concrete used in the disposal cells and containers was specifically developed as part of a RD&D programme conducted by the Instituto Eduardo Torroja and was formulated to optimize the durability of the concrete barriers. It exhibits a high degree of compressive strength and compactness, as well as resistance to sulphate and chloride attack. Groundwater at the site is low in both sulphates and chlorides, which is beneficial to the longevity of the concrete.

After all vaults have been filled, each platform will be covered with an engineered multilayer cover as a final capping system (Fig. 24). The multilayer cover will function to prevent precipitation water from coming into contact with the buried and sealed vaults and the waste contained within them. To this end, a series of drainage and sealing layers will be installed, along with anti-intrusion layers and a vegetative cover to enhance long term stability. Once the capping is complete, a 300 year period of surveillance and institutional control will commence.

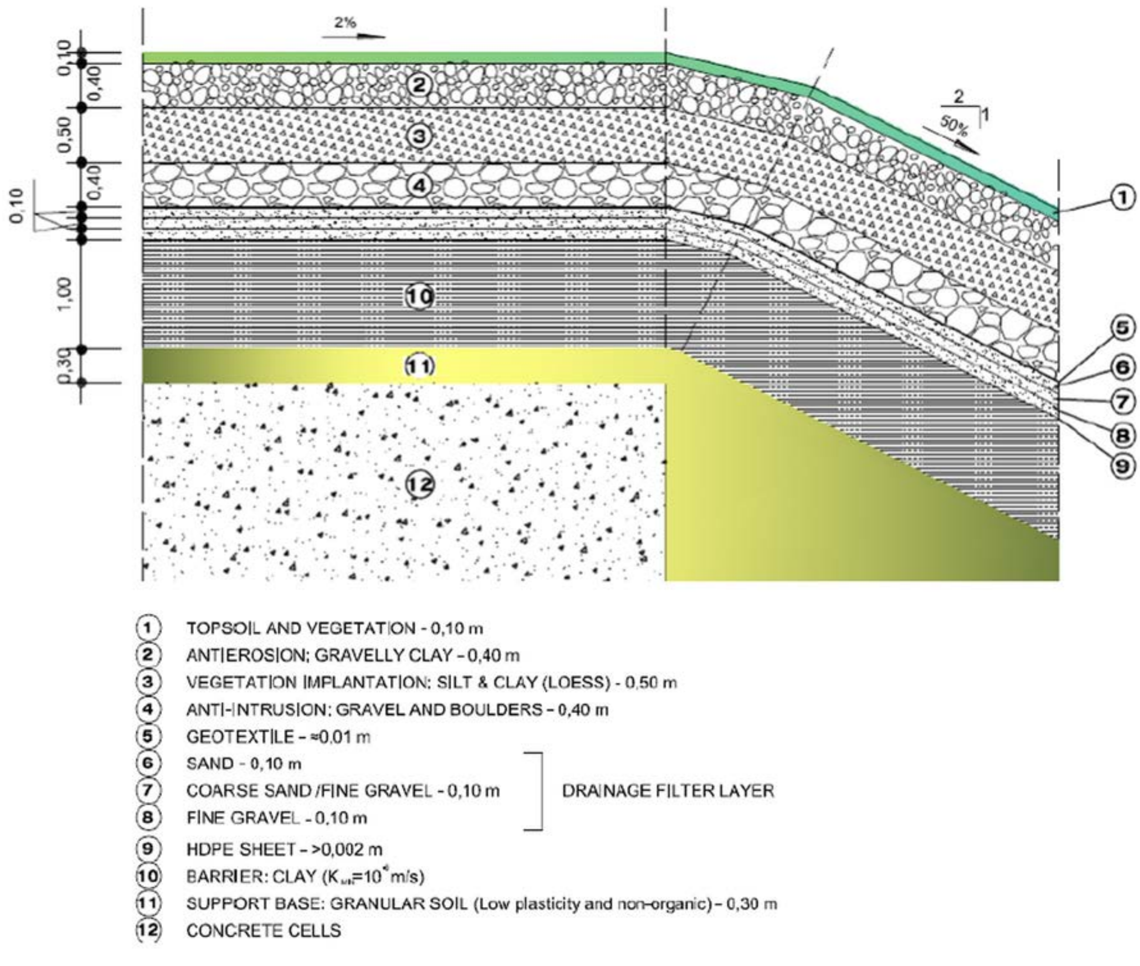

FIG. 24. Conceptual design of the final multilayer engineered capping system (courtesy of Enresa). HDPE — high density polyethylene.

4.2.3.2. Mochovce (Slovakia) and Dukovany (Czech Republic) repositories

The Mochovce repository in Slovakia is a further development of the Dukovany repository in the Czech Republic. A comparison of the two disposal facilities illustrates how different geological conditions and changing requirements between the licensing and construction of Dukovany and the subsequent (more than 10 years later) licensing and construction of Mochovce can affect the design and operation of a repository.

The Radioactive Waste Repository Authority (Správa úložišť radioaktivních odpadů (SÚRAO)) was established by the Ministry of Industry and Trade of the Czech Republic in June 1997 and is responsible for the design, development and construction of radioactive waste repositories within the country. Construction of the Dukovany repository (now under the responsibility of SÚRAO) began under the government of the former Czechoslovakia in 1978. The repository was put into operation in 1994 by the Czech Republic government.

Dukovany is located on Quaternary sediments overlying a basement composed of gneiss and migmatites. The safety approach consists of a combination of engineered barriers in the form of reinforced concrete structures with specially designed water management and drainage systems. The concept is based on initially diverting water away from the waste during an operational control period and subsequently precluding or significantly limiting direct access of water to the waste.

The repository consists of two disposal units constructed adjacently to one another. Each unit is a double row of seven vaults with each vault divided into four disposal chambers (dimensions 17.3 m × 5.3 m × 5.4 m). The vaults are separated by construction joints to allow for differential movement related to settling, so as to minimize the potential for cracking. The chambers are not equipped with a drainage system, but the floor of each chamber is sloped towards a corner sump from which water can be removed prior to waste emplacement, if needed. Prior to being put into operation for disposal, the interiors of the chambers are protected by removable concrete panels (Fig. 25(a)). During emplacement of the waste drums (Fig. 25(b)), the active chamber is opened. A mobile roof is used to protect the opened chamber when waste drums are not being emplaced.

The vault walls comprise 60 cm of reinforced concrete. Each disposal unit is completed with a 10 cm thick hydroinsulation layer of asphaltopropylene concrete that surrounds the exterior of the disposal unit. The asphaltopropylene concrete layer is held in place by a structural supporting wall in exposed areas of the disposal unit. Rainwater is drained away from disposal units. A dedicated drainage collection system is installed below the base and to either side of the disposal unit. The function of the drainage is to collect water that may potentially have encountered waste from the overlying disposal units. The drainage leads to a single sampling point at the end of each row. In the unlikely event of a detected release, targeted surveillance can be conducted to localize the source to a specific chamber, thereby facilitating remedial actions, if they are found necessary.

Various types of waste are disposed of at the facility, including evaporator concentrates, miscellaneous trash, contaminated solid waste, large bulk waste from maintenance and nuclear power plant upgrades, and spent ion exchange resins. Except for bulk items, waste is placed in 200 L steel drums, which are stacked in six layers inside each disposal chamber. The remaining void space between waste drums is backfilled with concrete. After each

(a) (b)

FIG. 25. Dukovany Repository, Czech Republic. Temporarily covered disposal chambers (a) prior to waste emplacement and (b) during waste emplacement activities (courtesy of SÚRAO).

chamber has been filled, the concrete panels are again placed over the chamber and covered with a hydroinsulation layer and a temporary cap to ensure that rainwater does not enter the chamber prior to closure of the facility.

The Slovakian WMO, the Nuclear and Decommissioning Company (Jadrová a vyraďovacia spoločnosť, a.s. (JAVYS)), was established in 2005. The construction of the Slovakian National Radioactive Waste Repository at Mochovce started in 1986 under the government of the former Czechoslovakia.

The safety concept (Fig. 26) is based on a combination of natural and engineered barriers (an existing clay formation at the site and reinforced concrete structures with specially designed water management and drainage systems). The concept is based on initially diverting water away from the waste during an operational control period and subsequently precluding or significantly limiting the direct access of water to the waste. The clay formation beneath the facility acts as an important barrier between the waste and the accessible environment and the first step in constructing the facility was to form the naturally occurring clay into a suitable foundation for construction. The engineered structures at both the Mochovce and Dukovany repositories originally followed almost identical designs. However, because of the timing of the construction start, work at the Mochovce repository was impacted by the political changes that occurred in the early 1990s. These changes included the establishment of a new regulatory structure in Slovakia, which delayed the commissioning of the facility. In addition, several design changes were implemented based on recommendations from the IAEA, which had been provided at the request of the new Slovakian government. More stringent controls were also placed on the construction of the repository to reflect its classification as a nuclear facility. Owing to both the increased construction controls and the decision to incorporate IAEA design recommendations, final commissioning of the facility was delayed. Disposal operations began in 2000.

The general design of the disposal units remained the same as at Dukovany, with the exception that at Mochovce each disposal unit consists of a double row of five vaults (rather than seven) that are each subdivided into four chambers with identical dimensions. However, two significant design differences were implemented at Mochovce. First, instead of relying on a mobile roof, the entire operational disposal unit is covered by a temporary shelter to protect waste emplacement operations from adverse weather conditions. This modification allows vaults to remain open, protected by removable concrete panels (Fig. 27(a)), until after the waste has been emplaced and the chambers sealed. It also allows a waste emplacement strategy to reduce potential differential settlement resulting from the plastic nature of the clay upon which the repository is constructed. Second, rather than having a single monitoring point for water potentially leaching from the repository, an underground gallery system was constructed at the end of each row to provide direct monitoring for each chamber.

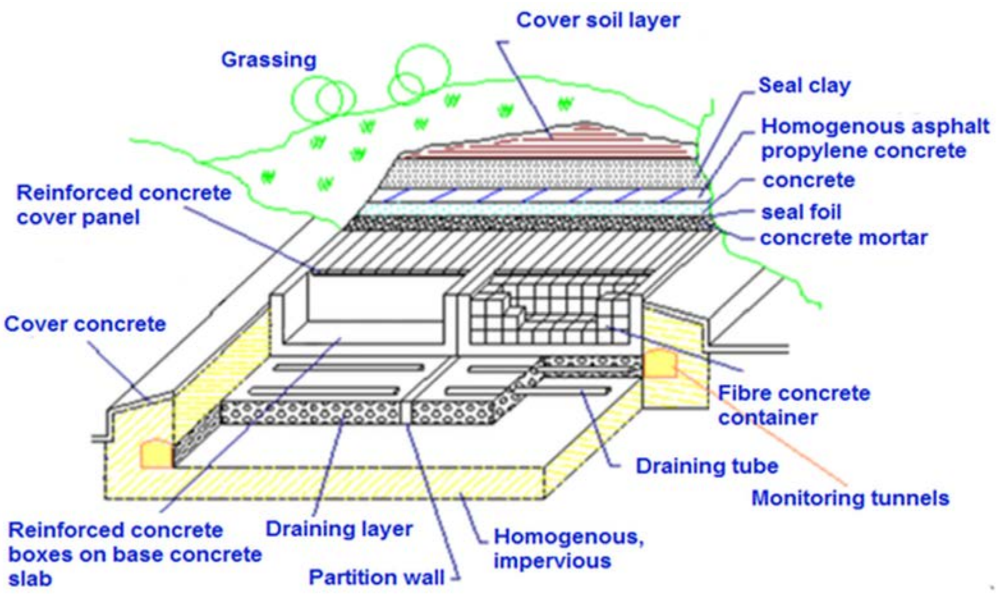

FIG. 26. Design concept for the Mochovce repository (courtesy of JAVYS, Slovakia).

(a) (b)

FIG. 27. Mochovce Repository, Slovakia. (a) Covered disposal system with chambers protected by concrete panels and (b) waste package being moved for disposal (courtesy of JAVYS, Slovakia).

Finally, changes to the waste packages were also implemented. Instead of the disposal of drummed waste, at Mochovce a specially formulated fibre reinforced concrete container is used, produced under licence from the French company SGN/Sogefibre. The containers measure 1.7 m on one side, with an internal disposal volume of about 3 m^3. Waste is disposed of after treatment directly into the cubic waste packages and immobilized in cement at the Bohunice Treatment Center and at the Mochovce Liquid Radioactive Waste Final Treatment Facility. After confirmation of the WAC and inspection for package integrity, the containers are unloaded directly from the transport vehicles into a disposal vault in a preassigned position (Fig. 27(b)).

4.2.3.3. Vrbina repository, Slovenia

The Vrbina repository is intended for the disposal of short lived low and intermediate level waste (LILW), as defined by the Slovenian waste classification, corresponding to LLW according to Refs [10, 34]. The Agency for Radioactive Waste was established in 1991 to manage radioactive waste arising in Slovenia. The repository is primarily intended for the disposal of operational and decommissioning waste associated with the Krško nuclear power plant and for various institutional wastes. It is being designed and licensed following the requirements of the Slovenian National Programme of Radioactive Waste and Spent Fuel Management [35].

The Krško nuclear power plant is owned and operated jointly with Croatia. In its first phase, the repository is intended to provide disposal for Slovenia's half of the waste generated by the plant and other radioactive waste arising in Slovenia. The projected waste volume includes a possible future extension of the nuclear power plant's operating life. The design concept allows for the possibility of expansion if a future agreement is reached with the Croatian government to dispose of all waste from the plant.

The safety approach consists of a combination of natural and engineered barriers (existing geological formations and reinforced concrete structures with specially designed water management and drainage systems). The disposal concept relies on minimizing the access of water to the waste by a combination of measures. To this end, the design combines existing surface disposal concepts implemented in France, Spain and other countries, with the underground silo disposal concept used in Finland, the Republic of Korea and Sweden. The safety case and licensing strategy considers two silos.

The silos in the Slovenian concept act as very large disposal vaults with direct access to the surface during the operational phase. The silos will have an interior diameter of 27.3 m and a height of 51.5 m. They will be constructed completely below ground. Standard construction methods will be used to prevent groundwater from entering the excavation during construction, for example, using a diaphragm wall. After excavation is completed, an initial primary lining system will be installed to provide stability. The liner will consist of the 1.2 m thick diaphragm wall and a secondary 1 m thick liner of reinforced concrete. The second liner provides the isolation and containment functions for the waste. The silo concept is shown in cross-section in Fig. 28.

Initially, one silo will be constructed, which will provide adequate disposal volume for all existing and forecasted waste and support Slovenian disposal needs. Additional silos can be added to account for future

FIG. 28. Vrbina repository, Slovenia: vertical cross-section of the silo during the operational period and an illustration of a waste container (courtesy of the Agency for Radioactive Waste, Slovenia).

operating extensions and decommissioning waste, as well as other potential waste sources. The planned silo, including any future extensions, will be constructed below the water table and in formations that lie beneath a thin upper aquifer. This solution is considered to contribute to a significant reduction in the probability of human intrusion: at this depth, inadvertent intrusion by drilling that could pass through the repository is considered to be very unlikely. Prior to disposal the waste will be packaged in specially designed concrete containers, as shown in the right of Fig. 28, with dimensions of 1.95 m × 1.95 m × 3.25 m and a maximum weight of 40 t. These waste packages will then be disposed directly into the silo. Spaces between the waste packages will be filled with sealing materials. Containers will be arranged in 10 levels with 99 containers per level. Bulk disposal of some dismantled components will be considered as an alternative to cutting and packaging.

The silos are to be constructed in a layer of Miocene silts, with permeabilities of between 10^{-9} and 10^{-7} m/s. The silt layer is located under a 3–15 m thick, sandy carbonate gravel deposit of the River Sava. Groundwater is encountered at a depth of approximately 4 m under the site. In flood conditions, water can rise to the level of the site and appropriate precautions against this are included in the design.

To account for possible changes in waste arisings, the repository construction concept allows for a modular approach, with each silo representing one module. The licensing strategy envisions a progressive construction approach. This progressive strategy can be roughly divided into two construction stages: in the first stage, platforms and the required earthworks for a single silo will be completed, as well as connections to commercial public infrastructure. This stage of construction will encompass all operational support facilities and the first silo. The potential extension of the facility by a second silo would be completed as a second stage of construction.

The operational facilities will be constructed on a raised platform to protect against probable maximal floods. Structures classified as important to nuclear safety and with a potential 100 year operational lifetime have a seismic design requirement for earthquakes with a return period of 1000 years. For a closed silo, including an institutional control period of 300 years, a design basis earthquake with a return period of 15 000 years is evaluated. All remaining structures are designed consistent with Eurocodes, considering an operational lifetime of 50 years.

4.3. SUBSURFACE DISPOSAL SYSTEMS AT INTERMEDIATE DEPTHS

The long lived nature of ILW compared with LLW means that a disposal system for ILW will require a significantly longer period of containment and isolation of up to several thousands of years. Unlike the surface disposal of VLLW and LLW described in the previous sections, reliance on institutional controls for such time periods is not feasible and safety can only be guaranteed by passive measures. Subsurface disposal in suitably engineered facilities is the internationally recognized disposal option considered suitable for waste classified by the

IAEA as ILW [10]. This section considers this type of facility, constructed at intermediate depths (typically, tens of metres) below the surface.

4.3.1. General considerations

Waste classified as ILW exhibits higher levels of radioactivity levels than LLW and requires shielding when handling. The commonest types of ILW generated by the nuclear power industry arise as a result of nuclear power plant operations and consist of spent ion exchange resins used to clean cooling water circulated through the reactor as well as some contaminated trash and scrap metal components. Upon decommissioning, some reactor components are also classified as ILW. ILW is also generated in the reprocessing of SF. Most countries generate relatively smaller volumes of ILW from research activities and from medical and industrial uses, including some longer lived disused sealed radioactive sources (DSRSs).

Operational ILW from nuclear power plants is generally treated and conditioned by solidification into cement or bitumen and subsequently placed in shielded containers. Typically, volume reduction is not practised with ILW to avoid potential heat generation when the radionuclides are concentrated. Experience has shown that operation of a 1000 MW(e) reactor produces about 20 m^3 of ILW annually. The variety of materials in ILW waste streams can require considerations of long term corrosion rates, gas generation, flammability and so on.

Some countries have opted for deep geological disposal for ILW in dedicated mined repositories, converted underground facilities or in facilities co-located with HLW repositories. The Waste Isolation Pilot Plant (WIPP) is an example of ILW disposed in a dedicated deep geological repository (Section 4.4). Canada is also considering a mined deep geological disposal facility for disposal of much of its LILW.

4.3.2. Disposal concept

As well as providing isolation and containment for thousands of years, an engineered disposal facility for ILW located at intermediate depth needs to consider and minimize as far as possible the likelihood of inadvertent human intrusion, by activities that encroach into the subsurface (e.g. building foundations, installation of utilities and transportation systems). A typical depth for this type of facility would be several tens of metres.

The time required for ILW to decay to safe levels means that the natural barrier properties of the host formation can become an important consideration in limiting potential releases to the accessible environment. The barrier capabilities of the natural system include the ability of the host formation to control groundwater movement around the facility and to retain radionuclides or significantly delay their release to the accessible environment. The engineered barriers are selected and designed to work in concert with the natural system over the longer time frames required for ILW to decay to safe levels. Engineered barriers can include the waste form and waste package, containment structures (e.g. concrete enclosures) and sealing systems. Additionally, the higher operational dose rate potential associated with ILW needs to be accounted for in the design concept.

Two types of subsurface ILW repositories at intermediate depths have been developed: underground silos and drifts. At Wolsong in the Republic of Korea, silos are used exclusively for ILW disposal, while in Hungary underground drifts are used at the Batapaati repository for the disposal of all nuclear power plant operational waste. In Finland and Sweden, both systems are employed. In Finland, the management and disposal of operating waste is the responsibility of the nuclear power generators: at the Olkiluoto nuclear power plant the silo concept has been employed for the disposal of LLW and ILW, while at the Loviisa nuclear power plant, drift disposal is used for both classes. The two corresponding design concepts are illustrated in Fig. 29.

Various factors need to be balanced when selecting between the two disposal concepts. Experience has shown that underground silos can be more difficult and costly to construct than drifts, but they also provide a greater disposal volume relative to excavation volume. Silos can offer advantages in terms of reducing radiological exposures to workers from emplaced wastes, while emplacement in a drift might require additional shielding measures to achieve a similar level of worker safety. The final selection is dependent on the specific circumstances and waste characteristics planned for the repository and its operational requirements. Examples of both concepts are provided in the following subsections.

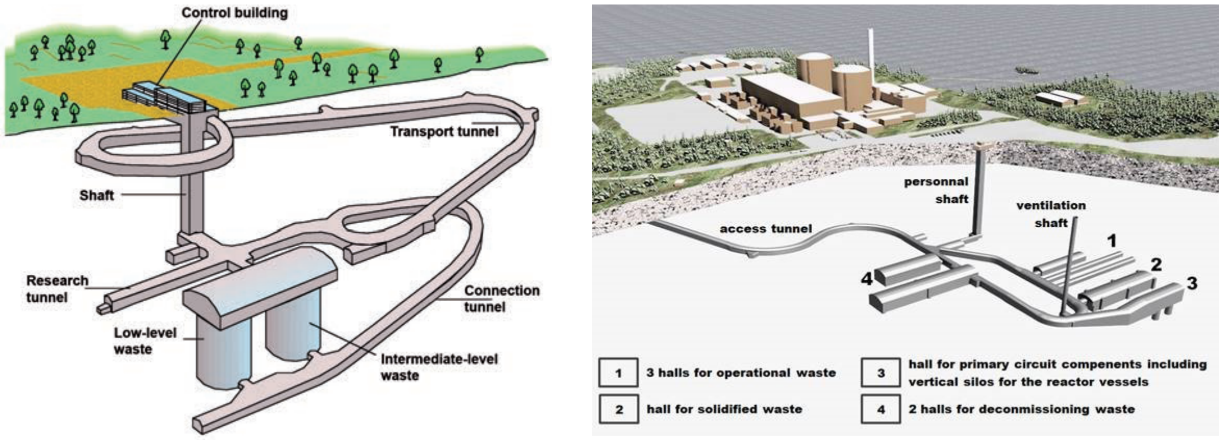

FIG. 29. Design concepts for intermediate depth subsurface disposal of ILW at sites in Finland: Silo concept (left) (courtesy of Teollisuuden Voima Oyj) and drift concept (right) (courtesy of Loviisa nuclear power plant).

4.3.3. Representative examples

4.3.3.1. Wolsong LILW Disposal Centre, Republic of Korea

The Korea Radioactive Waste Agency (KORAD) was created in 2009 and is responsible for the management of all types of radioactive waste, including LLW, ILW, HLW and DSRSs. In 2007, KORAD began construction of Korea's first repository for LLW and ILW, the Wolsong Low- and Intermediate-Level Radioactive Waste Disposal Centre (WLDC). When completed, the Wolsong repository complex will provide disposal capacity for ILW in underground silos, which are currently in operation, and LLW in a near surface concrete, engineered vault type system. WLDC is the first radioactive waste disposal facility that has been specifically developed to host two different types of repository. Because some LLW might be disposed in the silos along with ILW, KORAD refers to the waste content in the silos as LILW.

The WLDC will dispose of LLW and ILW generated from the operation and decommissioning of nuclear power plants, research facilities, nuclear fuel processing facilities and other facilities where wastes are generated. Radioactive waste is packaged in either 200 L or 320 L drums. The final planned disposal capacity at WLDC is 800 000 waste packages, based on a 200 L drum size. Waste drums destined for disposal in the silos are placed inside either 16 pack (4 × 4) or 9 pack (3 × 3) concrete disposal containers. The maximum weights of the 16 pack and 9 pack disposal containers are 18.34 t and 10.81 t, respectively. In total, 100 000 packages (assuming the smaller drum size) will be disposed in the 6 existing silos. Each silo can hold approximately 1000 concrete containers. The design concept for the silos at closure is shown in Fig. 30.

The first phase of construction at WLDC began in 2007 and involved the construction of 6 underground silos, which were completed in 2014 and subsequently licensed to receive waste. The facility is in operation and emplacement activities are ongoing. The second phase of the repository complex, the near surface repository, is undergoing licensing for construction.

The six ILW disposal silos are located at a depth of 80–130 m with a total height of about 50 m and a diameter of 25 m. The silos have been excavated in a granite host formation. They are connected to the surface through parallel tunnels for construction and operations. The construction tunnel provides access to the base of each silo, while the upper operations tunnel is used for waste emplacement. A vertical shaft is also provided for personnel and visitor access and ventilation. The supporting facilities are located at the surface and include a waste reception, inspection and storage building, a radioactive waste treatment building, the main control centre, an equipment maintenance shop and a visitor centre.

Radioactive waste is transported in disposal containers along the upper tunnel and lowered by crane into its final, preassigned position. The precise location and content of each container are recorded. After all the waste has been emplaced, the remaining void space in each silo will be backfilled with crushed rock. Concrete plugs will be installed in the silo access areas. Figure 31 shows waste being emplaced in the silo.

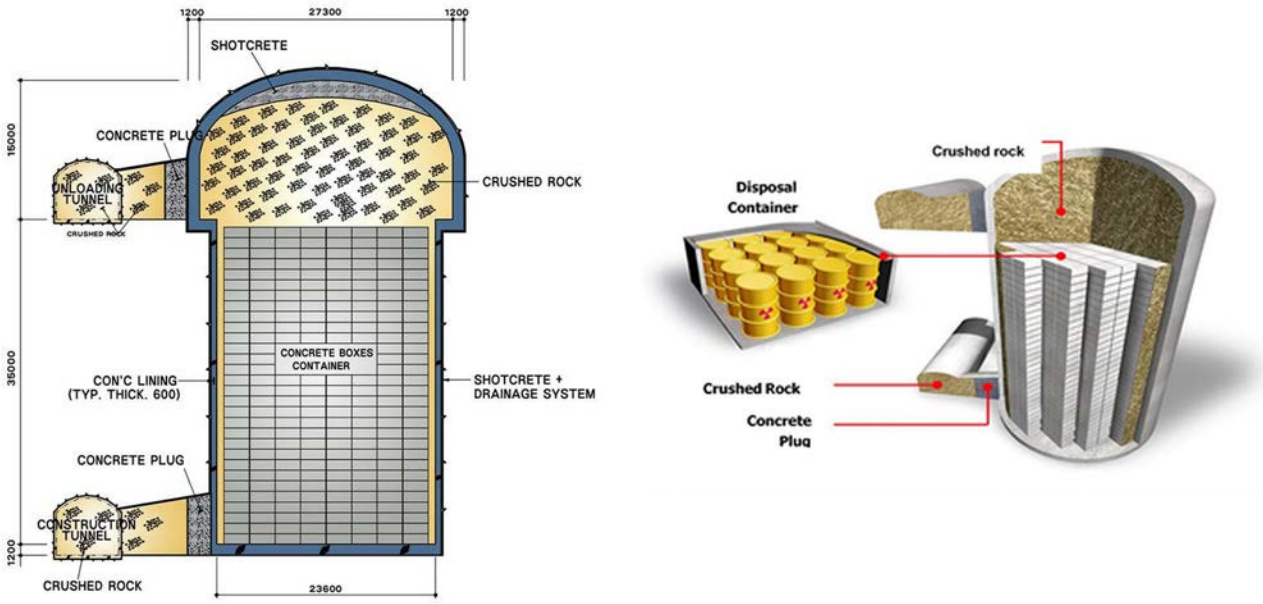

FIG. 30. Waste emplacement concept for silos at the Wolsong Low- and Intermediate-Level Radioactive Waste Disposal Centre, Republic of Korea (courtesy of KORAD).

FIG. 31. Waste emplacement operations in a silo at the Wolsong Low- and Intermediate-Level Radioactive Waste Disposal Centre, Republic of Korea (courtesy of KORAD).

The main barriers to radionuclide release are the engineered systems designed specifically to function in the natural environment at the site, as confirmed by safety assessment. The engineered barriers include the waste package, the disposal container, backfill material and the concrete silo. The concrete walls in each silo have a design thickness of at least 1.2 m and play an important role in limiting the infiltration of groundwater into the silo and significantly reducing the migration of radionuclides dissolved from the disposed waste to the environment. The permeability of the concrete silo will gradually increase with time owing to natural degradation processes and the concrete silos will eventually lose their effectiveness as a barrier against groundwater infiltration and radionuclide retention. At this point, the natural barrier capabilities of the site will assume the primary role in waste isolation and containment. The mineralogical composition of the granite functions to sorb radionuclides and measured groundwater flow rates are low, significantly reducing any radionuclide concentrations that might reach the environment.

4.3.3.2. Final Repository for Short-lived Radioactive Waste, Sweden

The final repository for short-lived radioactive waste (SFR) is located near the site of the Forsmark nuclear power plant. It is owned and operated by Svensk Kärnbränslehantering (SKB), the Swedish Nuclear Fuel and Waste Management Company. The SFR is located at a depth of about 50 m beneath the Baltic Sea, as illustrated in Fig. 32. The host formation consists of crystalline bedrock comprising gneiss and granite. The facility currently accepts LLW and ILW, most of which originate from operations at Sweden's nuclear power plants, although the facility also accepts waste generated from medical, veterinary, research and industry use.

Operation of the SFR began in 1988. The facility has a disposal throughput of about 600 m^3 of waste per year, with a total capacity of 63 000 m^3. Depending upon requirements, waste is disposed of either in one of four excavated rock vaults or in a silo. These facilities are shown in Fig. 32 in white and include an underground control room. Plans are also underway to extend the disposal capacity to accept waste from the future decommissioning of Swedish nuclear power plants. The planned extension will have five rock vaults each with a length of 275 m and a sixth vault with a length of 240 m. A new access tunnel is also planned to allow the transportation of intact reactor vessels. The existing facility and planned extension are also shown in Fig. 32, highlighted in blue.

The current SFR has three rock vaults with two different configurations dedicated to disposal of LLW and a rock vault and a silo for ILW. Waste with the highest level of radioactivity is disposed in the silo [36].

LLW is disposed in Vault 1BLA and consists primarily of contaminated trash and scrap metal stored in ISO containers. The vault is 160 m long and completed with a concrete floor. Some of the waste in the containers is additionally drummed or otherwise contained. The ISO containers are stacked side by side in two rows, either with three full height, or six half height containers on top of one another. A temporary corrugated steel roof helps limit the access of moisture to the waste (see Fig. 33). Vaults 1BTF and 2BTF are dedicated to the disposal of higher activity LLW, consisting largely of dewatered spent ion exchange resins in 10 m^3 concrete containers (referred to by SKB as tanks) as well as ash in steel drums and concrete boxes (referred to by SKB as moulds), with cement

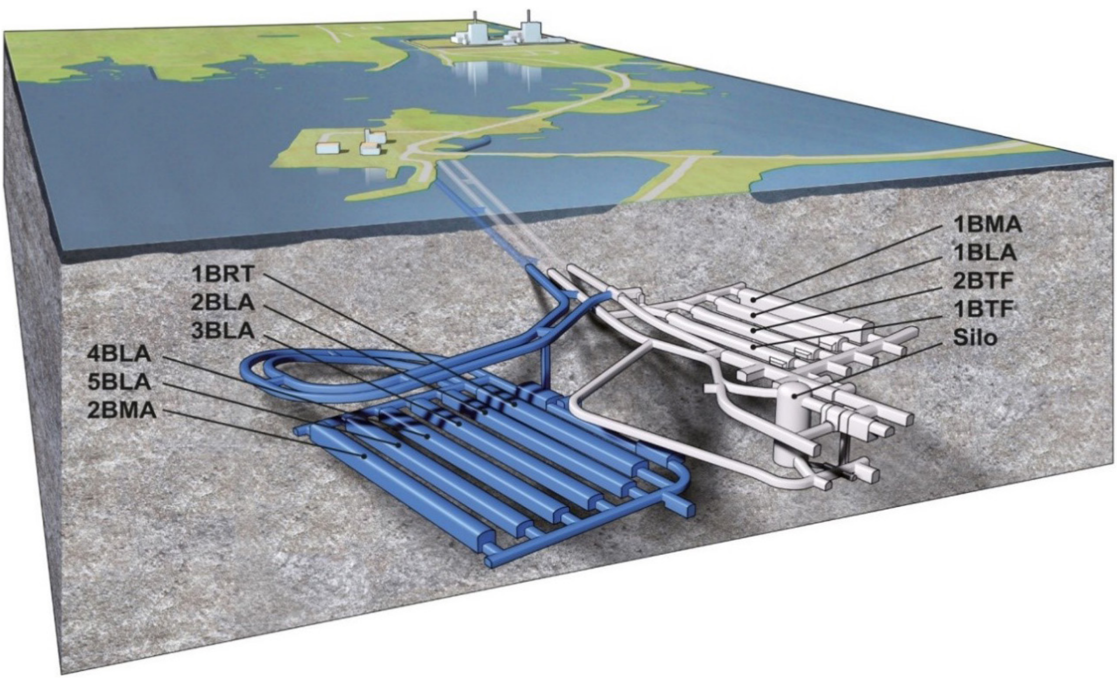

FIG. 32. Scheme of existing (white) and planned enlargement (blue) of SFR, Forsmark, Sweden (figure courtesy of SKB). Note: BMA, BLA, BTF, BRT are designators used by SKB for different disposal vault designs for different waste types.

FIG. 33. Disposal vaults for LLW at the SFR, Sweden (courtesy of SKB).

solidified spent ion exchange resins. 2BTF is dedicated to the disposal of 10 m^3 concrete waste containers. Vault 1BMA is used for the disposal of LLW with higher levels of short lived activity. Vault 1BMA is also 160 m in length. At closure, the vaults will be backfilled with concrete.

ILW is disposed in the concrete silo excavated in the crystalline host rock. The silo is 70 m in height with 50 m available for disposal. It has a diameter of about 30 m. The outer wall of the silo is constructed from reinforced concrete. The gap is backfilled with bentonite. The base of the silo is constructed from 1 m thick reinforced concrete over a bed of sand-bentonite, separating the silo from the host rock. The silo itself is subdivided into several shaft-like rectangular compartments into which the waste is emplaced. The waste is primarily composed of spent ion exchange resins solidified in cement or bitumen with some cemented trash and scrap metal components. It is contained in concrete or steel waste boxes (moulds) or steel drums grouped onto trays for disposal. After emplacement in the shafts at predefined intervals, concrete is used to seal void spaces, allowing disposal operations to continue. At closure, the remaining void space in the silo will be filled with concrete.

About 90 percent of the total radioactivity currently emplaced at the SFR is contained in the silo. In the safety case, emphasis was placed on the engineered barriers to delay potential releases of radionuclides after closure; the bentonite buffer and concrete walls of the silo are considered significant barriers to radionuclide migration. The safety assessment for the repository extends to 10 000 years, although most of the radioactivity will have decayed to background levels after only 500 years [37].

The extension to the SFR will provide more capacity for the disposal of LLW and lower activity ILW by providing additional vault space, following the BLA and BMA designs.

4.4. DISPOSAL IN DEEP STABLE GEOLOGICAL FORMATIONS

HLW and SF, if designated as waste, remain hazardous over geological time scales. The waste generates intense levels of radioactivity and heat that must be accounted for in the disposal concept. Significant heat generation can last several hundreds to some thousands of years, while radioactivity levels can remain hazardous to human health and the environment for hundreds of thousands of years, many tens of times longer than recorded human history. The disposal concept for these wastes therefore needs to be based on long term passive safety considerations and an international consensus has long existed that this type of waste can only be disposed in appropriate deep stable geological formations, often referred to as deep geological disposal. The concept of deep geological disposal of HLW was first explored more than 60 years ago by the National Academy of Sciences in the United States of America [38]. Since then, every Member State contemplating disposal considers some form of deep geological disposal as the most appropriate means for isolating HLW and SF permanently from the environment.

4.4.1. General considerations

HLW generally refers to waste resulting from the chemical treatment (reprocessing) of spent nuclear fuel. The resulting waste includes highly concentrated liquid solutions containing nuclear fission products. These liquids are normally solidified into either a glass or ceramic waste form suitable for storage and subsequent disposal. Many Member States also classify unreprocessed SF as waste that will require direct geological disposal.

Although no Member States have to date constructed and operated a deep geological repository for HLW or SF, considerable progress has been made in demonstrating the feasibility of several design concepts. In particular, Finland has made significant progress and has begun repository construction activities, after receiving a construction licence authorization in 2015. The WMO Posiva is expected to submit an application for an operating licence to dispose of SF around 2020. Sweden has also submitted a construction licence application for a SF repository, which has been reviewed by the regulatory authorities and is currently being considered by the Swedish environmental court. The United States of America completed a licence application for a proposed repository at Yucca Mountain in Nevada, which was submitted for regulatory review in 2008. However, political developments resulted in the suspension of almost all work related to the review of the licence. In France, a major effort is currently underway to prepare a licence application and technical design for a deep geological repository for HLW. Germany, which had made considerable progress, particularly in developing pilot scale surface facilities as well as in conducting subsurface exploration in support of a preliminary safety assessment, has decided to restart its siting programme based on stakeholder considerations. Most other nations actively involved in HLW and/or SF management are either in the siting or pre-siting phases for a geological repository.

The first purpose built deep geological repository, the WIPP in Carlsbad, New Mexico, was completed and taken into operation in 2000. Although originally designed to include HLW, subsequent decisions limited the inventory for disposal at WIPP to non-heat generating LILW. A compendium of worldwide radioactive waste disposal methodologies has been compiled by the Lawrence Berkeley National Laboratory, United States of America [39], which concentrates on deep geological repositories and describes potential issues that can arise as repository programmes mature, as well as identifying techniques for sharing concepts for design and safety cases.

4.4.2. Disposal concept

The conceptual basis of geological disposal is also based upon the multibarrier system, whereby a series of engineered and natural barriers act passively and in concert to isolate the wastes and contain the radionuclides associated with the wastes. The relative strengths of the various barriers at different times after the closure of a geological disposal facility and the way that they interact with one another depend upon the geological environment in which the facility is to be constructed. Consequently, the components of the multibarrier system can work in different ways at different times in different geological disposal concepts to fulfil the high level safety objectives of isolation and containment. The practice in many national geological disposal programmes is to define safety functions for each component, which set out what each specific barrier component contributes towards post-closure safety. These functions will vary specific to the defined disposal concept, the geological environment and the timeframes involved for their effectiveness. The overall safety of a disposal system does not depend upon any one of these functions alone, but upon how the functions interact with one another as a function of time as the closed disposal facility slowly evolves.

To date, four basic rock types have been studied in detail as host formations for geological disposal: argillaceous sediments (clays, mudstones, marls), hard crystalline basement rocks (gneiss, granite), evaporite formations (principally bedded or dome rock salt) and welded volcanic tuff. The characteristics of the selected host formation and the actual conditions at the selected site will dominate design considerations.

Safety functions for all rock types can include the leach resistant properties of the waste form, corrosion resistant waste containers, the sorptive and low flow properties of the backfill and the properties of the host geological formation, which provides a mechanically and chemically stable, low groundwater flow environment for the engineered barriers, as well as retarding the migration of radionuclides and providing isolation from the accessible environment.

The following generic guidelines can be used to help identify potentially suitable host rock environments for consideration for geological disposal facilities:

— Depth: Geological isolation is attained by ensuring sufficient separation between repository and biosphere, including zones for engineered sealing systems. As well as site specific factors such as deep groundwater flow and chemistry, rock mechanical and strength properties can determine a practical and functional repository depth.

- — Thickness: An adequate thickness and lateral extent of the host formation is required to host the engineered openings and separate the facility from surrounding formations that might have less adequate containment properties. Thickness is a particularly important factor in layered sedimentary formations.
- — Uniformity and structure: A reasonably homogeneous host formation is desirable, as this allows more straightforward rock characterization, reduces uncertainties in performance assessment, and facilitates repository construction planning and operations.
- — Tectonic stability: Regions with low seismicity, uplift/erosion and volcanicity favour repository design and long term performance.
- — Hydrogeology: Low hydraulic conductivity of the host formation and low groundwater flow through it and surrounding formations favours long term containment. Very low flow and geologically stable conditions can also contribute to a diffusion (rather than flow) dominated transport system when coupled with specifically designed engineered barriers.
- — Geochemistry: Chemically reducing conditions minimize the corrosion of engineered barriers and waste forms and can reduce radionuclide solubility and improve sorption.

Many programmes currently developing HLW or SF disposal facilities plan on co-locating the disposal of ILW in the same host formation. The programme plans typically consider emplacement in different regions of the underground facilities, perhaps at different depths, to accommodate the different engineered barrier requirements applicable to each waste class. Special consideration is given to ensure the proper separation of the volumes of rock where different categories of waste are emplaced, so as to avoid any potential adverse (e.g. chemical) interactions between the wastes or their barrier materials.

4.4.3. Representative examples

The following sections provide examples of geological disposal facilities for HLW and/or SF under development in each of the major rock types previously outlined.

4.4.3.1. Argillaceous formations

Argillaceous sedimentary formations contain large percentages of clay minerals, which contribute to their low permeability and high retention capacity. They can display various degrees of compaction and mechanical strength, from plastic clays to indurated mudrocks and shales. Several countries are considering argillaceous formations for the disposal of HLW/SF, including Belgium (plastic clay), France (argillite) and Switzerland (claystone). Each of these countries has advanced repository programmes with extensive scientific support from years of in situ testing.

(a) Centre industriel de stockage géologique — the French HLW and ILW disposal concept

In France, the feasibility of the safe and reversible disposal of HLW and ILW in deep clay formations has been studied by Andra since 1993. R&D studies and experiments performed in the underground laboratory at Bure have been used to consolidate and demonstrate iteratively the safety performance of the Centre industriel de stockage géologique (Cigéo), the deep geological disposal facility designed by Andra. If permitted by the responsible authorities preparation for the construction of Cigéo could begin in 2022/2023 in order to start construction of the repository itself soon after a licence has been granted. Current plans call for the operation of the pilot phase to start before the end of the decade. The full operational period for Cigéo is expected to last for over a century. The concept incorporates provisions for reversibility to allow future generations the freedom to make their own decisions on the best way to manage HLW. The concept of reversibility includes an approach to management that allows decision processes to continue well into the future. It also implies maintaining a certain level of flexibility in the design of Cigéo. Furthermore, reversibility requires monitoring of Cigéo's activities and development, and supports continuous R&D work to acquire data and information that may be needed to support future decisions.

Cigéo is designed to contain both HLW and long lived ILW, as illustrated in Fig. 34. The French waste classification distinguishes between long lived ILW and short lived ILW; only long lived ILW as well as HLW will be disposed at Cigéo.

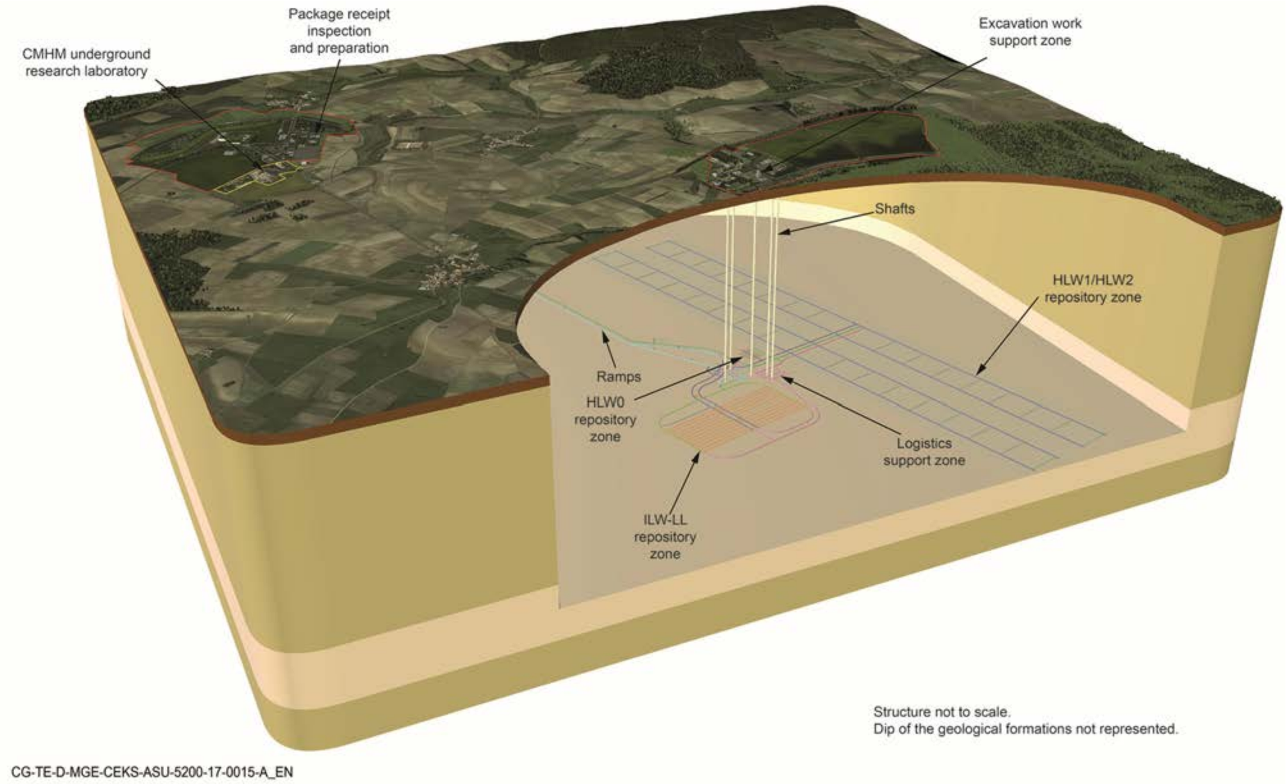

FIG. 34. Diagram showing the surface and underground facilities for Cigéo at the final operational stage (courtesy of Andra).

The wastes will be disposed at a depth of about 500 m near the centre of an impermeable argillaceous rock formation (Callovo–Oxfordian argillite) with a thickness of about 140 m. The underground facility comprises three main areas: a central logistic support zone connected to above ground facilities through shafts and an access ramp for transferring the waste packages, and two separate disposal areas for HLW and ILW. Because construction will continue in phases during the operational period, waste transfer and construction activities are physically separated in different access galleries. Andra decided early in their programme to dispose of waste in packages that would require the use of shielded casks during all transfer operations. Disposal tunnels, oriented according to the direction of the principal major rock stress, comprise two sub-units, disposal cells and a docking/interface zone.

ILW disposal cells, with a tunnel length of between 400 m and 500 m, are connected at one end to an access drift. Ventilation for each cell is provided by a second drift at the far end of the cell, equipped with a dedicated air filtration room that allows a return flow of air. The spacing between the disposal packages reflects a compromise between operational requirements and long term stability considerations, to meet the retrievability requirement. The exact diameter of each ILW disposal cell is driven by disposal package geometries, the height of package stacks and co-disposal criteria for different types of long lived ILW. The design concept for the ILW disposal cells is illustrated in Fig. 35.

HLW disposal tunnels (also referred to as microtunnels or disposal cells) will have lengths ranging from 80 m to 100 m. The disposal package will be emplaced in a metal sleeve with a minimum thickness of 25 mm lining each tunnel. The cross-section of the microtunnels was determined as a compromise between waste handling clearance requirements for retrievability and limitations imposed by the excavation disturbed zone of the host rock. Depending on the geometries of the various disposal canisters, the excavated diameters of the microtunnel ranges from about 800 mm to 900 mm, with sleeve diameters between 800 mm and 700 mm. The microtunnels are designed as blind tunnels without ventilation. The spacing of the tunnels is driven by post-closure thermal loading considerations of the host rock. A schematic representation of an HLW disposal cell is shown in Fig. 36.

The disposal cells are lined with a metal sleeve with a minimum thickness of 25 mm and a diameter of 700 mm. HLW waste packages, referred to as CSD-Vs (colis standard de déchets vitrifies, standard packages for vitrified waste) are disposed in metal disposal containers. Spacers are used to separate disposal packages as part of the thermal management strategy. During operations, a 10 m cell head separates the access drift and the emplaced

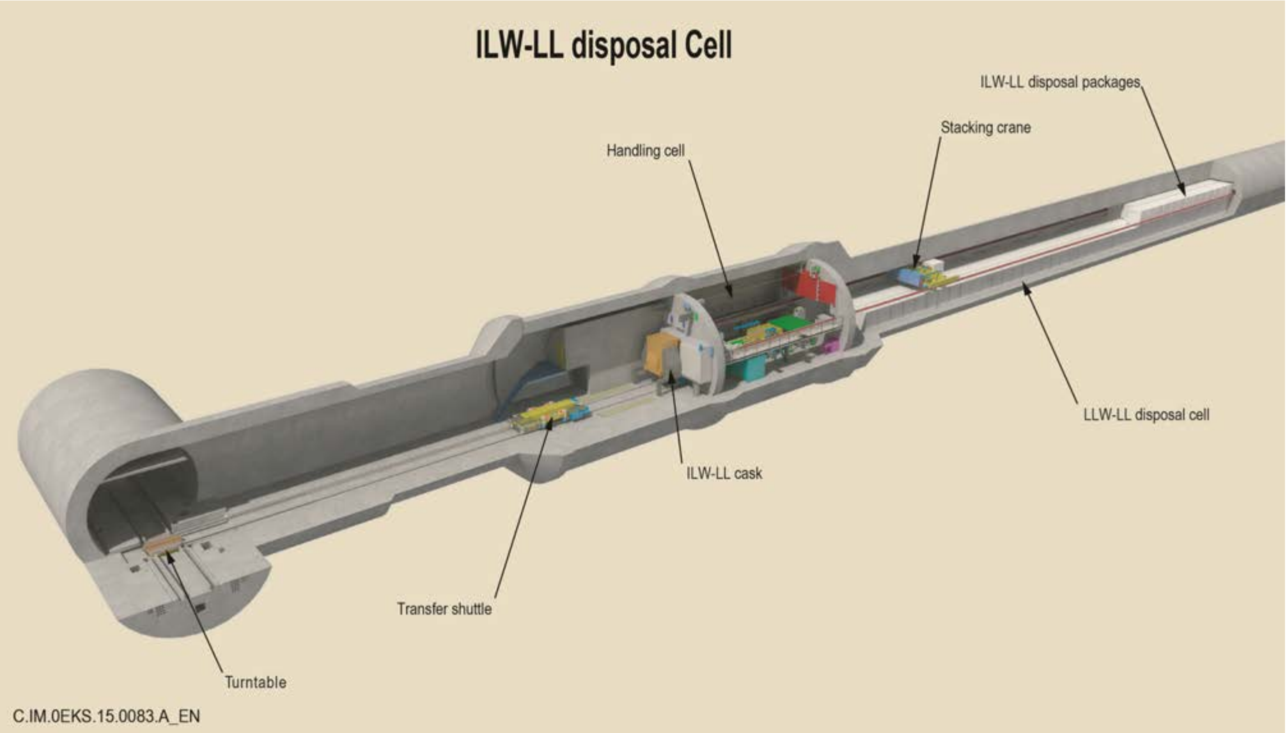

FIG. 35. Long lived ILW disposal cell design (courtesy of Andra).

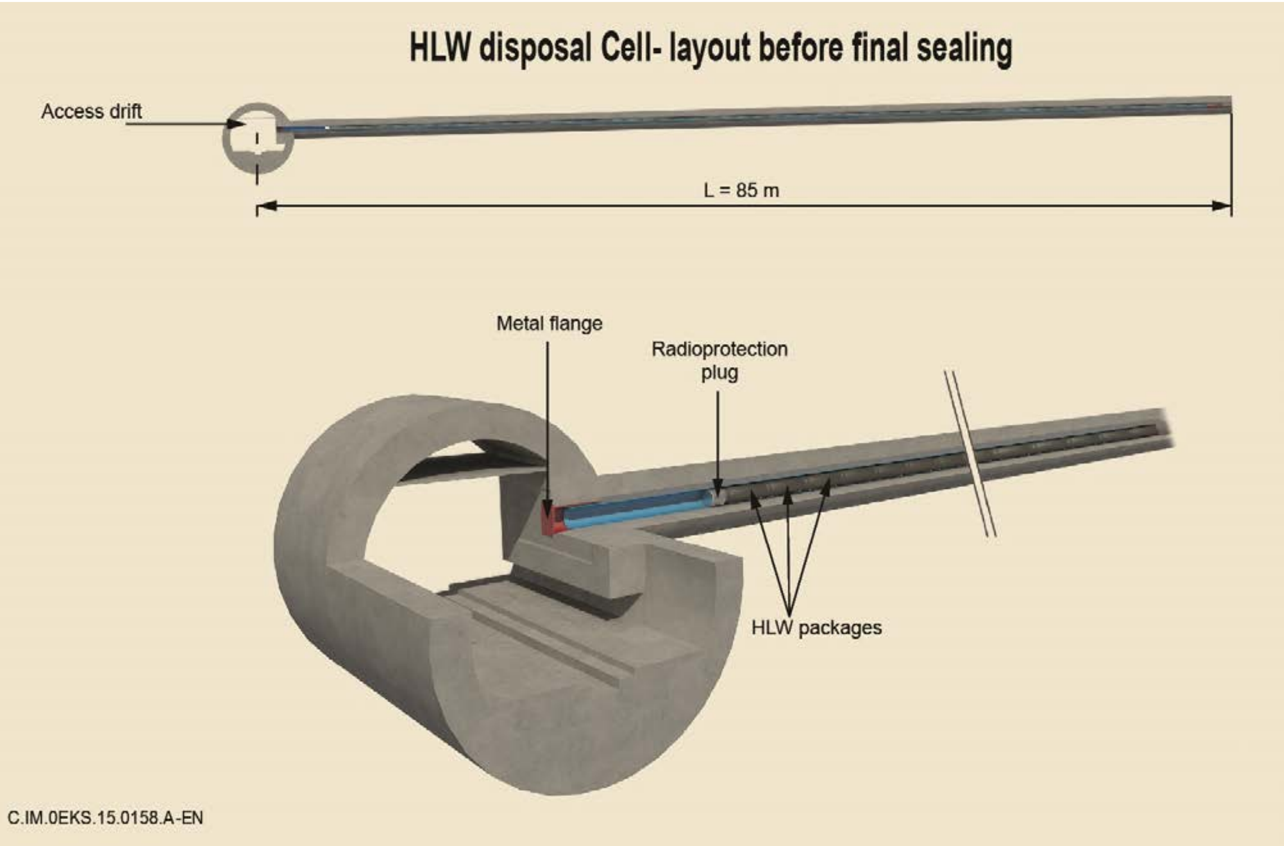

FIG. 36. Cross-section through an HLW disposal cell (courtesy of Andra).

waste. The cell head is equipped with a metal flange and cask docking devices. After a cell has been filled, a radiation protection plug is installed, and the devices are removed. The cells are not backfilled.

Closure of the underground facility will follow a step by step approach. After the disposal cells have been plugged, the access drifts will be backfilled and sealed, followed later by the main drifts, then the shafts, and finally the ramp. The surface facilities will be dismantled at the same time as the underground facility is closed.

(b) Belgian deep geological repository concept

The Belgian National Agency for Radioactive Waste (ONDRAF/NIRAS) is considering two potential host rock formations for its deep geological repository programme: the Boom and the Ypresian clays. Both formations consist of poorly indurated clays with low permeability and high plasticity. The low permeability and low hydraulic conductivity of the clays are key elements in the safety case.

The reference design for the repository includes separate disposal areas for ILW and HLW/SF. The repository concept considered in the current RD&D programme envisions emplacement at a depth of between 200 m and 600 m. The layout of the repository is shown in Fig. 37. Dead-end disposal galleries are excavated perpendicular to the main access galleries in the host formation. The project is conducted in two phases, without simultaneous construction and waste disposal activities. ILW will be emplaced in the first phase of operations, with construction and emplacement activities for HLW following in the second phase. Access to the underground area will be provided by two shafts located between the two emplacement areas. The waste shaft will allow the transfer of both ILW and HLW packages to the disposal areas. The access shaft will allow the transfer of personnel and material. Both shafts are used to provide adequate ventilation.

Before transport and emplacement in the repository, the primary waste packages will be placed into specifically designed disposal packages. ILW will be disposed of in concrete 'monolith B' packages, while HLW/SF will be disposed of in a steel overpack and then in 'supercontainers' constructed from concrete with an outer steel envelope. The cement in the concrete creates a highly alkaline environment capable of retarding radionuclide migration. The two types of waste package are shown in Figs 38 and 39.

The monolith B packages and supercontainers provide radiological protection to workers (maximum 25 μSv/h at 1 m), enabling the safe handling of the disposal waste packages without additional protection. Apart from radiological shielding, the supercontainer also has to ensure complete waste containment during the thermal

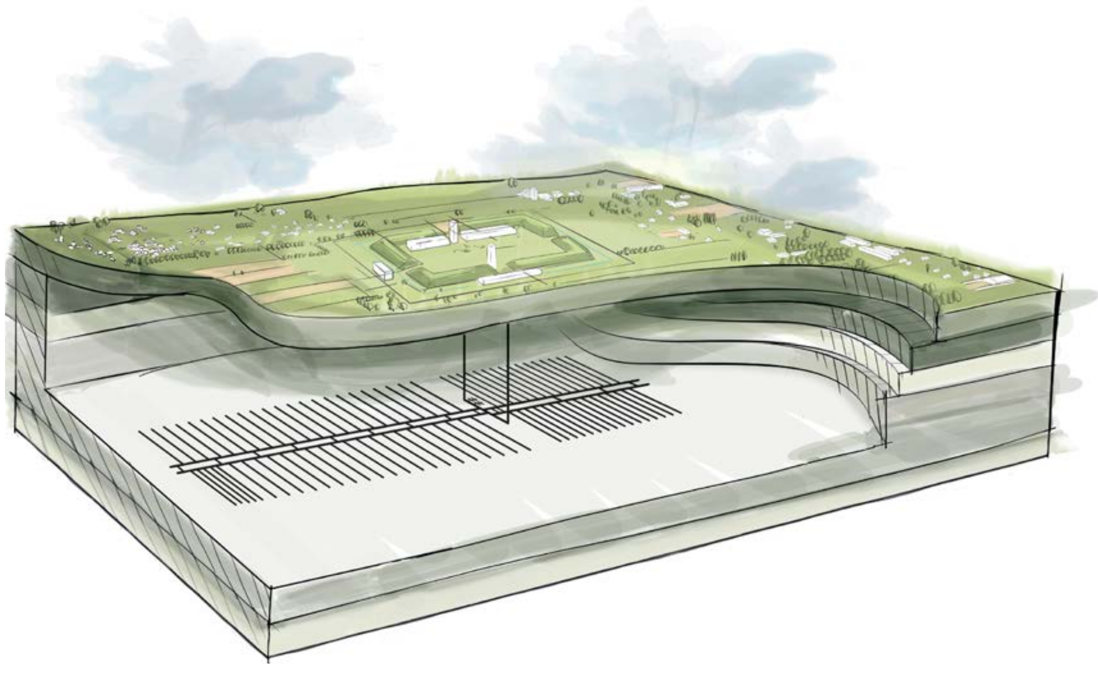

FIG. 37. General layout of the Belgian deep geological repository concept (courtesy of ONDRAF/NIRAS).

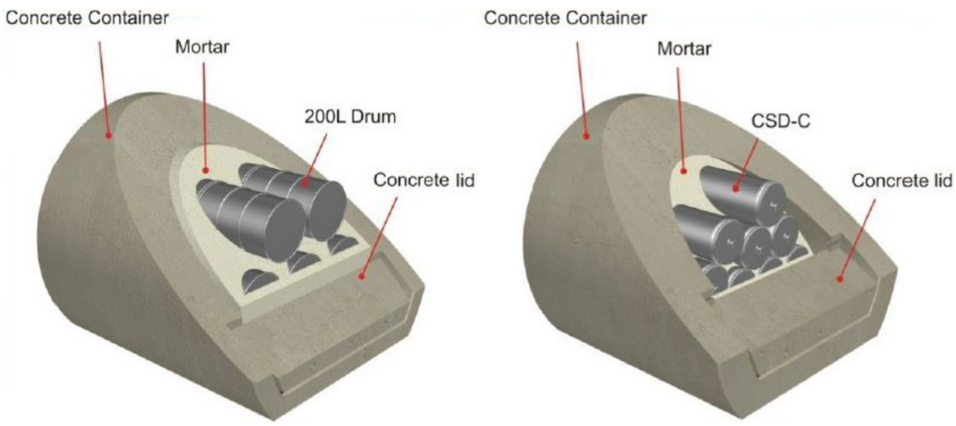

FIG. 38. Belgian waste container concepts: ILW concrete monolith B [41]. (Courtesy of ONDRAF/NIRAS.)

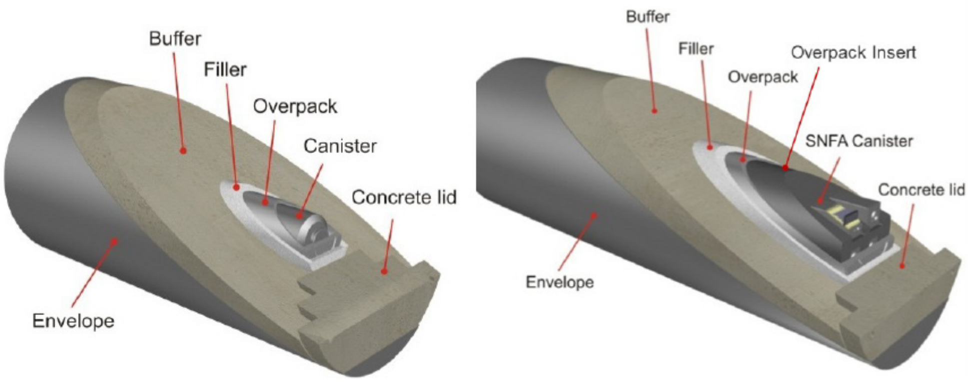

FIG. 39. Belgian waste container concepts: HLW/SF supercontainer (right) [40]. (Courtesy of ONDRAF/NIRAS.)

phase, defined as the period of time where the temperature at the drift wall remains above, which lasts from several hundred to a few thousand years, depending on the type of waste disposed.

A reversibility requirement is included in the design, to address public concerns, and the design concept considers waste retrievability up to the end of the operational phase. To support retrieval, low compressive strength cementitious backfill that can be removed with relatively little effort is being evaluated [41]. The disposal galleries will be progressively backfilled and plugged as disposal progresses and, when all disposal galleries are closed, the access galleries will be backfilled and sealed. After closure of the two emplacement areas, the support zone and the shafts will also be backfilled. Installation of plugs and seals at several locations in the repository is still an open issue and part of ongoing RD&D.

(c) Swiss HLW/SF disposal concept

In Switzerland, the National Cooperative for the Disposal of Radioactive Waste (Nagra) has evaluated six potential sites with favourable properties for construction, operation, closure and long term safety for a deep geological repository as part of the second stage in their site selection process [42]. Following a safety based comparison of the potential sites in 2014, Nagra is continuing site investigation activities on three potential sites. Opalinus Clay is the planned host formation at all sites. It is a homogeneous, moderately plastic, indurated claystone with little to no fluid advection and with geochemically reducing conditions. The current Swiss design concept for HLW/SF disposal includes backfilling tunnels after emplacement and foresees an option for retrievability without undue effort before closure. The planned underground facilities will comprise a pilot facility and a main disposal area for HLW/SF, as well as a research facility. A separate ILW disposal area is under consideration but can also be co-located with the HLW repository to be constructed in the Opalinus Clay. (It should be noted that in Switzerland ILW includes IAEA waste classes LLW and ILW.)

As a first step, a representative waste inventory will be emplaced in the pilot facility to observe the behaviour and interactions between the different waste types, backfill material and host formation for a specified monitoring period. During this time, data will be collected to confirm post-closure safety with a view to closure. A facility for underground geological investigations, where site specific data for safety relevant properties and processes can be acquired and underground activities can be demonstrated prior to disposal, can continue studies during at least part of the monitoring phase. Monitoring activities will be carried out from surface and subsurface installations, particularly from the pilot facility, throughout the monitoring period.

The underground facilities as conceived (Fig. 40) will be constructed at a depth of several hundred metres below ground surface, roughly in the middle of the Opalinus Clay formation. To provide for optimum use of the available space at the repository level, the concept allows for several spatially separated disposal areas, each with multiple emplacement rooms, dead-ending in the host formation. The length of each emplacement room may vary but will be restricted to about 1000 m. Minimum offset distances are considered between individual underground structures, as

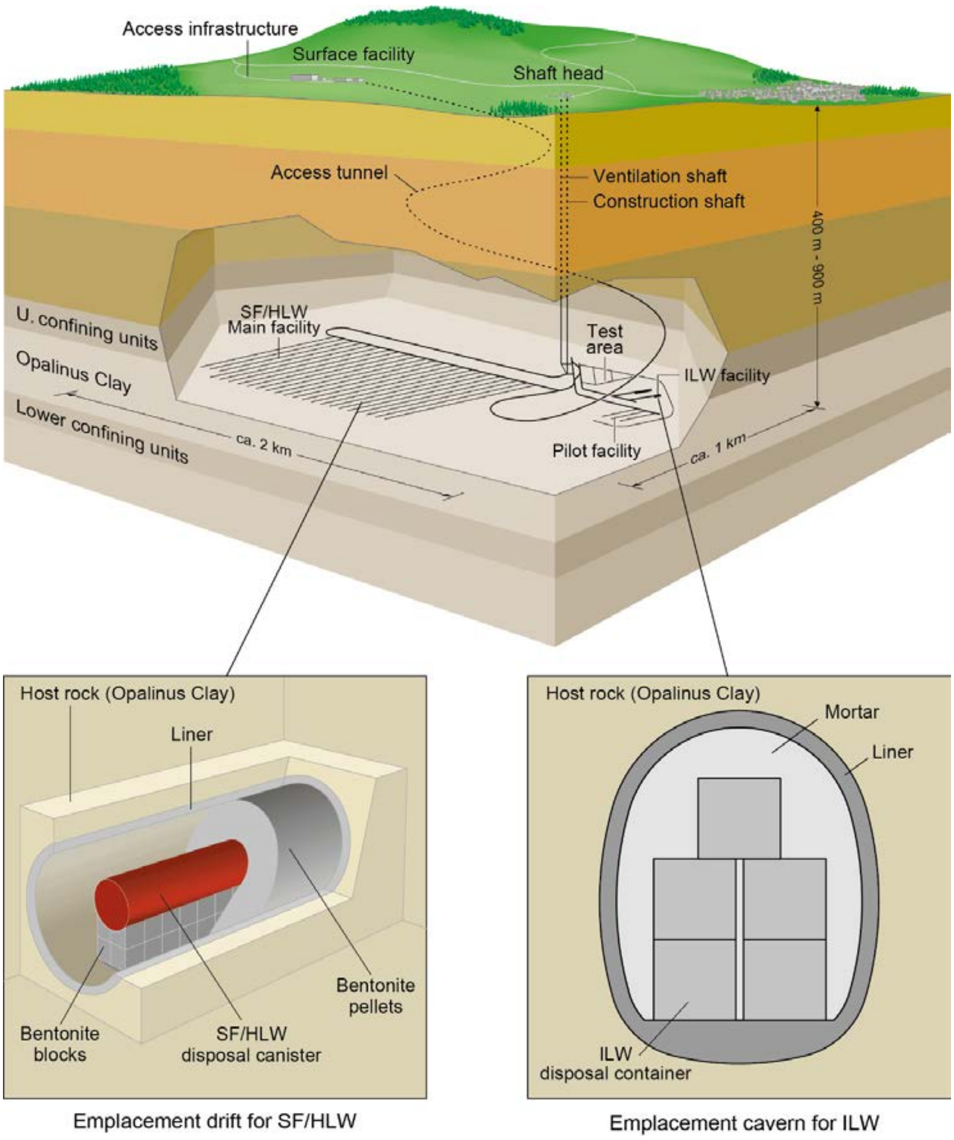

FIG. 40. Example layout of the Swiss ILW and HLW repository, and its main features (not to scale). (Courtesy of Nagra and reproduced with permission from fig. 3-3 of Ref. [43].)

well as potential geological features. The emplacement rooms are conceived to provide sufficient mechanical stability and suitable conditions for safe and reliable construction, operation and backfilling and sealing. Emplacement rooms for HLW/SF will be excavated with an internal diameter of about 3 m and completed with concrete lining. The spacing between individual emplacement rooms is foreseen as about 40 m, to meet thermal constraints.

SF assemblies and HLW casks will be loaded into specially designed disposal canisters. The current concept utilizes canisters fabricated from carbon steel. The disposal canisters can hold between four and nine SF assemblies, but alternative configurations are possible. The heat output per canister is restricted in the concept to 1500 W per canister at emplacement for optimal usage of the available underground volume. The disposal canisters will be emplaced co-axially at intervals along the rooms and supported on pedestals constructed from compacted bentonite blocks. Immediately after emplacement of each canister, the respective section of emplacement drift will be backfilled with highly compacted granular bentonite. The bentonite blocks and granules will together form a protective mechanical, hydrogeological and chemical buffer around the disposal canisters. Other backfill materials remain possible. The final decision will be taken in anticipation of the construction licence application.

Intermediate bentonite seals can be installed to provide direct physical contact between the bentonite in the seal and the clay of the host formation. These seals may be installed at frequent intervals along the emplacement rooms. In the current concept, after each room has been filled, a final seal consisting of bentonite will be installed.

At this stage of the programme Switzerland is also considering a ‘combined repository’, in which both the ILW and HLW/SF repositories are co-located, as well as separate facilities for the different waste streams. In the combined repository concept, the waste disposal areas will be spatially separated from one another but serviced by the same accesses. The decision to build two entirely separate repositories (one for ILW and one for HLW) at different locations or a combined repository at one location will be made in anticipation of the general licence application(s).

4.4.3.2. Crystalline rock formations

Crystalline basement rocks are the chosen formations for SF disposal in Finland and Sweden and granites or gneisses are also being considered in several other countries, including the Czech Republic, Japan, the Republic of Korea, the Russian Federation and the United Kingdom. The Finnish and Swedish programmes have developed their conceptual design together, have both selected and characterized their repository sites (Forsmark in Sweden and Olkiluoto in Finland) and are both at similar licensing stages, with a construction licence already having been granted in Finland. The Finnish programme is led by Posiva and the Swedish programme is led by SKB. Both WMOs were established by the respective nuclear power industries in each country, specifically to manage the disposal of radioactive waste. Both Finland and Sweden employ the KBS-3 design concept, which is highly developed and provides a basis for the reference cases being evaluated in several other countries.

The geotechnical properties of the basement rocks of the sub-Cambrian peneplain of the Fennoscandian shield combine strong mechanical behaviour and low permeability with flow restricted to networks of fractures, varying degrees of heterogeneity and adequate thermal conductivity. The fracture networks in crystalline rock give rise to heterogeneous groundwater movement that can be relatively rapid in the upper hundreds of metres of the bedrock, though much reduced at typical repository depths. Advective transport of any radionuclides released from the engineered barriers into the fracture network is estimated to provide limited retention, thus placing increased performance requirements on the waste package and engineered barrier system in order to provide the required containment.

The selected repository depths are between 400 m and 500 m. The designs of both repositories are relatively similar, with the only significant difference being in the mode of waste transportation to the subsurface for disposal; while the Posiva design utilizes a shaft hoisting system, the SKB designs employs a ramp. Both designs utilize a rectilinear set of disposal tunnels that will be excavated along the direction of maximum horizontal stress. The conceptual layout for both repositories is situated on a single horizon; however, in some countries (e.g. Japan, United Kingdom), multiple level repository designs are being considered, to minimize the areal footprint. The repository layout for the Finnish design, currently under construction, is illustrated in Fig. 41.

The KBS-3 design used in both Finland and Sweden, and originally developed by SKB, uses corrosion resistant long lived copper canisters to encapsulate the SF. Each canister will be placed in a single vertical deposition hole, constructed from the floor of a disposal tunnel, and surrounded by a bentonite buffer in prefabricated rings, as illustrated in Fig. 42 [44]. The bentonite provides physical protection of the canister in case of any movements in the bedrock. It also acts as a hydraulic barrier that limits the movement of groundwater and corrodants to the canister

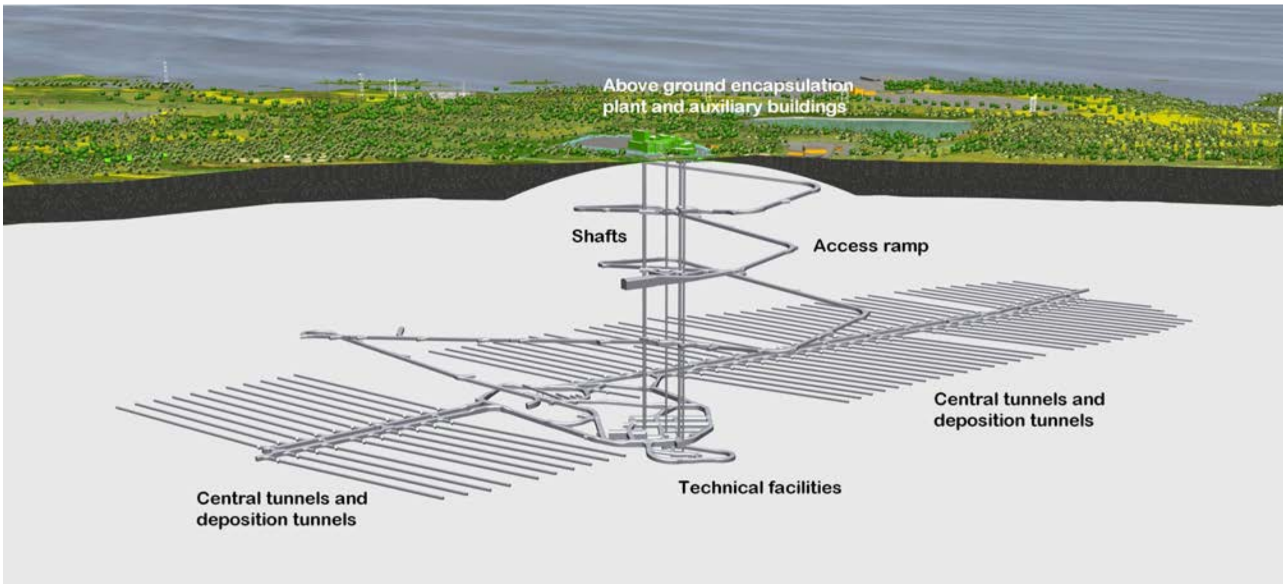

FIG. 41. Layout of Posiva's SF disposal facility at Olkiluoto, Finland (courtesy of Posiva Oy).

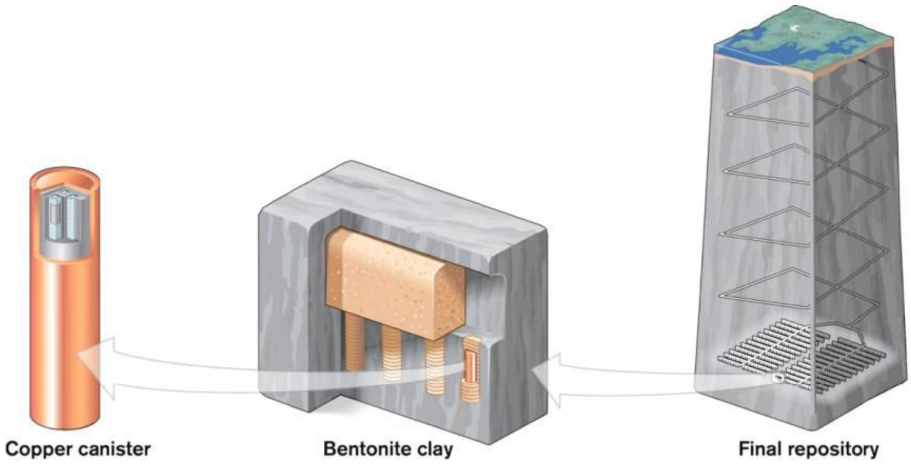

FIG. 42. KBS-3 disposal concept [44].

and any radionuclides that might be released from the waste in solution or in colloidal form. Very slow diffusion is the dominant transport process for solutes in the bentonite. The copper canister is expected to resist corrosion for at least 100 000 years [45]. The construction of deposition holes and the installation of a bentonite buffer, together with tunnel backfill and sealing systems, have been demonstrated in underground research and demonstration facilities (at Äspö in Sweden and ONKALO in Finland, at the site of the repository [46]).

A thermal requirement for the buffer material is to remain below 100°C, so the heat output of SF packages needs to be limited and the spacing between canisters adapted, according to the thermal properties of the rock. Spacing is site specific and typically around 6–10 m. Predisposal SF storage times and fuel packaging strategies need to be adapted to satisfy thermal requirements. A combination of minimum deposition hole spacing and the maximum heat content of the package ensures the buffer thermal requirement is met.

At closure the entire repository, including all ramps, shafts and tunnels, will be backfilled and sealed. The backfill and seal material is generally bentonite. The backfill may be in a combination of block and pellets. In some areas, excavated rock could be mixed with the bentonite to reduce the volume of spoil left at the surface and the required volume of bentonite; however, the low permeability requirements of the engineered barrier system must be maintained. Posiva and SKB have developed prototype machinery for drilling deposition holes and for installing bentonite buffer, disposal canister and tunnel backfill, and have tested them in their underground facilities.

4.4.3.3. Evaporite formations (such as rock salt)

Rock salt (halite), which can occur in both bedded and dome structures, possesses many favourable characteristics for the disposal of radioactive waste. It is essentially impermeable, easily mined and, owing to the plasticity of the rock, induced fractures are rapidly healed and openings naturally closed. Waste disposed in a salt repository will be fully encapsulated and sealed off from the environment within a relatively short period after closure. In addition, salt possesses high thermal conductivity, making it an ideal medium for the disposal of HLW and SF. Extensive salt mining experience has existed in several countries for more than 150 years and the disposal of HLW and SF in salt domes has been investigated in detail since the 1970s in Germany [47] and the Netherlands. Both bedded and dome salt has been investigated in the United States of America, with the earliest studies for disposal of heat generating waste having started in the 1960s.

The WIPP is a deep geological repository for long lived radioactive waste, authorized by the United States Congress in 1979. Construction took place in the 1980s, along with many experiments and demonstrations associated with site characterization and performance confirmation. The initial concepts for WIPP included disposal of both HLW and ILW, but WIPP was restricted to the disposal of defence generated transuranic waste in 1992, with limits placed on both the total volume and activity of the wastes allowed to be disposed [48]. In 1998, the US Environmental Protection Agency certified WIPP compliant with safety regulations and operations began in 2000.

The WIPP facility permanently isolates ILW 655 m below surface in a thick, bedded salt formation of Permian age. Contact handled waste is transported in robust reusable containers, which accommodate standard waste boxes, seven packs of 208 L drums, or ten-drum over-packs. This type of waste is stacked in open rooms. Remote handled waste is transported to WIPP in shielded containers by road. After reaching WIPP, the contents are removed robotically, conveyed underground in a separate shielded package, placed in a horizontal borehole and capped with a concrete plug. All waste bound for WIPP is packaged at the generator site. Disposal operations are shown in Fig. 43.

Disposal rooms are 4 m high, 14 m wide and 93 m long, with a 30 m pillar separating rooms. There are eight panels, each of which has seven rooms. Underground access is by vertical shafts for personnel transport, material handling, waste conveyance and ventilation, as shown schematically in Fig. 44. Panels are excavated immediately before they are needed because the surrounding salt deforms into the open void space. Salt creep plays an important role in operations and closure. Over time, the salt formation will completely encapsulate the waste, ensuring its permanent isolation. Worldwide experience in mining salt and other evaporite minerals such as potash provides an exceptional knowledge base for safe mining, ground control and operations.

After closure, permanent isolation is assured by a robust shaft seal system that uses common construction materials, including bitumen, salt based concrete, bentonite and reconsolidated crushed salt. Panel closures also utilize excavated salt. An additional engineered barrier of magnesium oxide (MgO) was required by the regulator. It is intended to decrease the solubility of actinides potentially present in transuranic waste. Large sacks of MgO can be seen on top of the waste stacks in Fig. 43.

In Germany, the Gorleben salt dome was selected as a potential site for HLW disposal in the 1970s. In 2013, work was completed on a preliminary safety assessment [47], which confirmed the feasibility of disposing heat generating waste at this site. However, two years later, to address public and political concerns regarding the fairness of the original site selection process for Gorleben, a new law restarted the search for a suitable site. As part of Germany's re-evaluation of available disposal options, the KOSINA project was undertaken [49] to investigate the technical feasibility and safety of generic repository concepts using two geological models in bedded salt. Geomechanical considerations determine the room and pillar geometries needed to dispose of the existing and future waste inventory, with the design of the emplacement area predicated on meeting thermal requirements.

Salt disposal concepts often include crushed salt as a natural backfill material. Construction demonstrations and natural analogues have shown that reconsolidated granular salt evolves to achieve thermal, mechanical and hydraulic properties approaching those of undisturbed salt formations. Reuse of mined salt provides operational efficiency, reduces hoisting and optimizes material transport logistics.

FIG. 43. Disposal operations at the WIPP, USA, showing both remote and contact handled waste packages (courtesy of the United States Department of Energy).

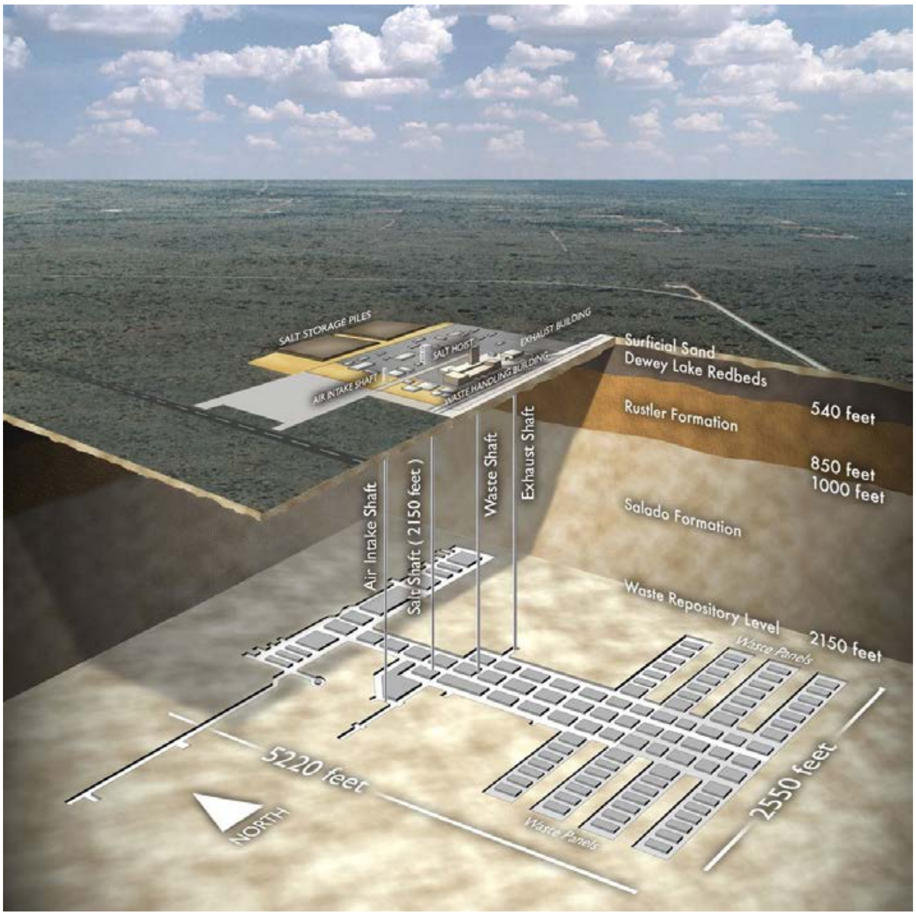

FIG. 44. Layout of the WIPP, USA (courtesy of the United States Department of Energy).

4.4.3.4. Volcanic tuff formations

Hard, welded volcanic tuff has been considered for HLW/SF disposal at the Yucca Mountain site located approximately 160 km northwest of Las Vegas, Nevada. The site, as seen in Fig. 45, lies in a remote desert area. It is characterized by a very deep groundwater table such that the repository host rock remains unsaturated. Such hydrogeological conditions make the site unique among those being considered worldwide for geological repositories. The final site selection was made in 1987 by amendment of the Nuclear Waste Policy Act designating Yucca Mountain as the only site for investigation as a repository for HLW and SF. The potential repository would be located in volcanic terrane approximately 300 m below ground level within a sequence of 13 million year old, highly fractured welded tuffs. A design with heavy reliance on engineered barriers was taken to licence application for construction in 2008. The principal characteristics that governed the design are the long term unsaturated conditions, hard rock with elastic-brittle deformation, low to medium matrix porosity and permeability, high fracture porosity and permeability, high heterogeneity and a mid-range thermal conductivity. Advective transport in the geosphere was assumed to be likely upon release of radionuclides from the engineered barriers, so little credit was taken for potential delays in transportation due to the unsaturated conditions. Emphasis was therefore placed on the performance of the waste package and engineered barrier system acting in concert with the natural system's unsaturated host unit to limit or reduce the amount of water contacting waste, thus providing the required containment.

The design features in-drift disposal of corrosion resistant waste packages, tightly spaced with no backfill around them, making this a unique 'open mode' emplacement concept. Prevention of rapid corrosion, vapour flux and condensation during the thermal phase and the associated cool down period are primary concerns. Waste package spacing specification ensures even heating along the emplacement drift, reducing the possibility of a cool area and associated liquid condensation. Drift spacing was adjusted to minimize pooling of liquid vapour above drifts during the thermal phase, providing a large enough spacing to allow vapour to condense and drain in the pillar between the drifts. A titanium drip shield system was included over the waste packages throughout the entire disposal area to provide a moisture and corrosion barrier. The drip shield also provides protection of the waste packages against damage by rock fall. The engineered barrier system and package layout are illustrated in Fig. 46 [50].

Over at least 30 years, the proposed Yucca Mountain repository design has undergone a long evolution from generic analyses to the eventual licence application. An in-depth discussion of the design characteristics and evolution over the project lifetime as site characterization progressed can be found in Ref. [51]. A construction

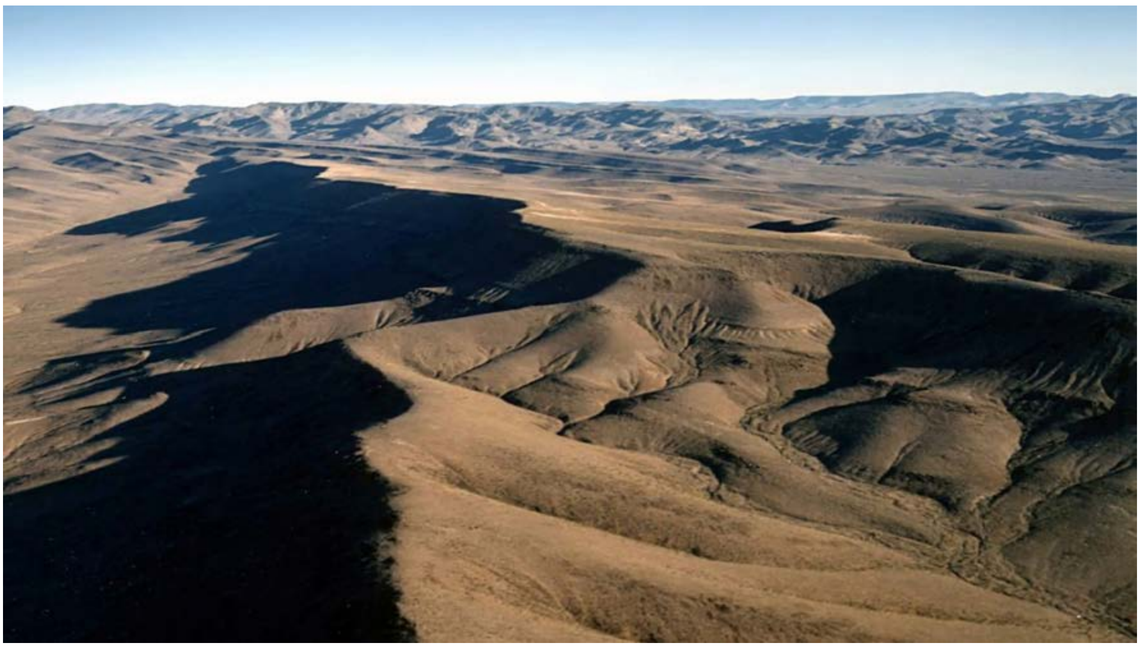

FIG. 45. Aerial view of the crest of Yucca Mountain, Nevada, USA (courtesy of the United States Department of Energy).

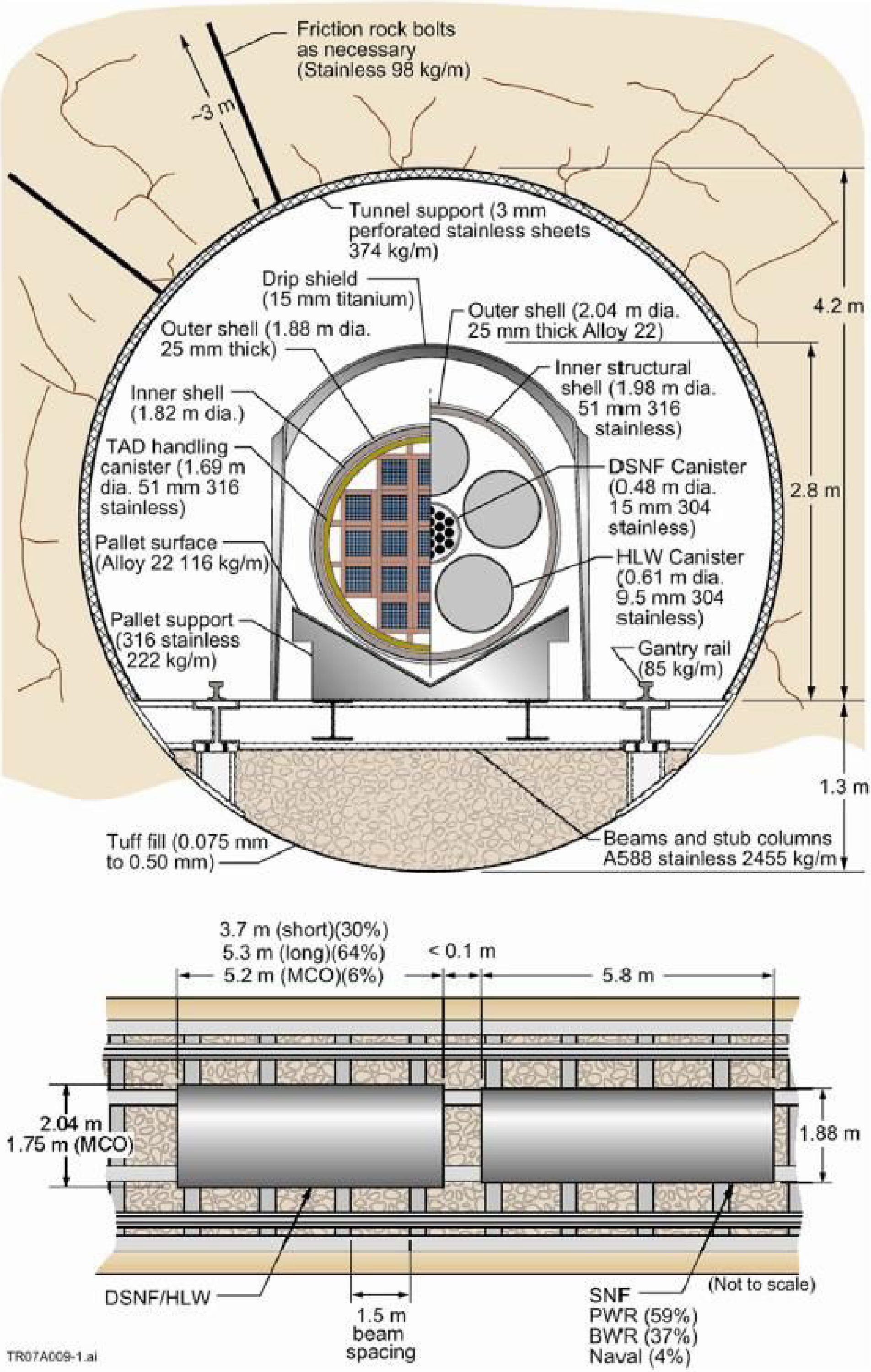

FIG. 46. Engineered barrier and waste package layout for the Yucca Mountain repository (courtesy of the United States Department of Energy).

application was submitted in 2008, 10 years after the United States Department of Energy was to have assumed ownership of civilian spent nuclear fuel. In 2014, the Nuclear Regulatory Commission found that the Yucca Mountain site met long term safety requirements after permanent closure [52]. At present, the project is not being carried forward and its future awaits direction and funding from Congress to complete the regulatory process.

The repository design situates the disposal horizon in a sequence of welded tuff units in an area characterized by a long term arid climate and deep unsaturated zone, with disposal taking place in an oxidizing environment, which is fundamentally different to the previously discussed geological disposal concepts. The fluid flux rate is low and the fractured nature of the rock results in a high degree of spatial and temporal variability across the repository. The geological setting and unsaturated nature of the host horizon have required the evaluation of water flux into the repository, corrosion processes and rates, and subsequent transport of radionuclides from the repository through the unsaturated zone to the saturated zone before potentially entering the accessible environment.

Several fundamental requirements controlled the Yucca Mountain design. The first is the requirement to dispose of an extremely large and heterogeneous waste inventory (70 000 tHM) largely composed of civilian spent nuclear fuel, but also including defence and civilian HLW. Important thermal requirements were 200°C at the drift wall and a 375°C limit on the spent nuclear fuel cladding. The design must also allow retrievability for 50 years [50]. An emplacement rate of 3000 tHM/year, 50% higher than the production of SF in the United States of America, was required to successfully draw down the stored nuclear fuel stockpile [50]. Construction and operation costs are to be borne by a nuclear waste fund accumulated from a levy on energy production, in accordance with the Nuclear Waste Policy Act of 1982 (collection of the fee was suspended at the time of writing).

The final design at Yucca Mountain was based on ramp access and direct disposal in drifts with no backfill. Waste packages were to be placed in approximately 99 parallel linear drifts of an average length of 605 m. The repository would be opened and operated in four stages. A modular, staged construction was anticipated to meet financial requirements, while at the same time providing flexibility in adapting the layout to accommodate potential for technological advances throughout the operational lifetime.

4.5. OTHER DISPOSAL SOLUTIONS

Other solutions have also been evaluated for the underground disposal of radioactive waste. These include boreholes of various depths and the conversion of existing underground facilities, such as disused mines, for the disposal of ILW. Examples of both are presented in this section.

4.5.1. Adapting existing underground facilities

Several existing LILW repositories have been developed by repurposing existing underground facilities such as mines or tunnels that are no longer in use. Repositories developed in this way are unique to the national programme under which they are developed and the circumstances and conditions of the specific facility selected for repurposing. Therefore, decisions leading to selection for their reconstruction as a repository will be specific to each Member State's national policy. In Germany, for example, a policy decision was made in the 1960s that all radioactive waste would be disposed in a deep geological environment. Because of this early decision, Germany has made exclusive use of former mines for the disposal of LILW, referred to in Germany as "waste with negligible heat generation". However, for HLW disposal, German law requires the development of a purpose built facility. Other nations have also chosen to use former mines, if not exclusively, for disposal of part of their LILW inventory, including Romania (Baita-Bihor repository) and the Czech Republic (Richard repository).

When developing the design concept for repurposing a facility, a detailed understanding of extant surface and underground facilities and accommodation, equipment and fittings is a prerequisite to ensuring that the requirements for safe disposal radioactive waste can be met, including operational safety aspects. It is also important to recognize and appreciate that the facility was built for a completely different purpose. The original purpose and the time elapsed since the facility was in operation can have important impacts on design considerations needed to achieve safe isolation and containment.

A potential benefit of utilizing existing facilities is that at least a limited understanding of geological conditions is available, although it is unlikely that this will include the level of detail needed for developing a reliable safety case, so additional site characterization work will be required. It might also be possible to derive direct economic benefits. For example, access to the subsurface will be available, although access routes are likely to require modification to meet waste handling requirements. Experienced personnel could be available locally and some of the installed equipment might be suitable for reuse. At least with respect to initial operations, cost savings in access to the underground might be realized.

However, regions of the facility not planned to be converted, as well as peripheral installations and equipment, will need to be stabilized and managed throughout the lifetime of the repository and after closure, which can represent a major commitment of resources. This work could include dismantling installed systems and the backfilling and sealing of unused openings. The existing underground workings can also impact design flexibility and can impact the barrier performance of the natural system. Excavation disturbed zones may be more extensive than would be expected in a purpose built repository, requiring the application of specialized design solutions. Preferential pathways, such as existing shafts or drifts, not required for disposal related purposes will require sealing. Large unused open volumes, common in former mines, can impact the overall stability of the facility, also requiring specialized design solutions. Consequently, determining the suitability of an existing facility for conversion to a repository can be more complex than doing the same for a purpose built repository constructed in an undisturbed environment.

Germany has extensive experience in the conversion, operation and closure of repositories developed from repurposed mines. The Morsleben repository (Endlager für radioaktive Abfälle Morsleben (ERAM)) was adapted in the early 1970s from a former potash and rock salt mine for use as a LILW disposal facility. At ERAM a total of 37 000 m^3 of LILW were disposed at a depth of about 480 m, until operations were suspended in 1998. The facility is currently being licensed for closure. To ensure the stability of the underground works, between 2003 and 2011 almost one million cubic metres of ‘saltcrete’ (concrete made with rock salt) were used to backfill 27 chambers located in the central portion of the former mine. The closure concept envisions that most of the remaining underground chambers will be filled in a similar fashion. As an additional precaution, drift plugs will be installed to isolate the repository areas prior to final backfilling. A waste chamber as it currently exists at ERAM is shown in Fig. 47.

A second disposal facility in Germany, the Konrad repository, is being converted from a former iron ore mine. Konrad is the first radioactive waste repository that will have been approved and constructed fully in accordance with the national Atomic Law. Following a licensing process of nearly 20 years, the repository was granted approval in 2002 for the disposal of up to 303 000 m^3 of LILW, with the licence being confirmed in 2007 by the German Federal Administrative court. Construction activities include the excavation of new emplacement drifts (Fig. 48) and retrofitting existing facilities for waste handling and disposal operations.

The former mine operated for a little over a decade, extracting 6.7 million tonnes of iron ore, before being closed because it was no longer economically viable. The host formation for the repository is a low permeability oolitic limestone that is hydrogeologically isolated by surrounding low permeability formations. Thick beds of low

FIG. 47. LILW disposed in the ERAM repository at Morsleben, Germany (courtesy of Bundesgesellschaft für Endlagerung mbH).

FIG. 48. Construction of a transport drift at the Konrad repository, Germany (courtesy of Bundesgesellschaft für Endlagerung mbH).

permeability clay, marls and mudstone occur both above and below the ore body, providing substantial isolation and containment. Waste will be emplaced in newly excavated emplacement chambers at a depth of about 800 m to 1100 m. Eleven chambers will be constructed and will be about 7 m wide, 6 m high and up to about 1000 m in length. The former mine workings and the repository are illustrated in Fig. 49.

The WAC for Konrad allow for six cubic and five cylindrical types of containers made of steel, cast iron or concrete. Waste container volumes range from 0.7 m to 10.9 m^3, with a maximum weight of 20 tonnes. Two classes of waste containers will be accepted, based on radiological requirements. Typical waste containers are shown in Fig. 50. Both cubic and cylindrical waste packages will be transported underground by truck to their disposal location. Emplaced waste will be stacked in 50 m long disposal sections. Each section will be sealed by a shotcrete wall applied directly against the last row of waste packages. Pipes carrying a water–cement–aggregate mixture will be used to fill the void space in each section. A final concrete plug will be installed to seal each emplacement chamber after filling. Once all waste emplacement activities have been completed, all remaining underground openings from both the repository and the mine will be backfilled, shaft seals will be emplaced and the repository closed.

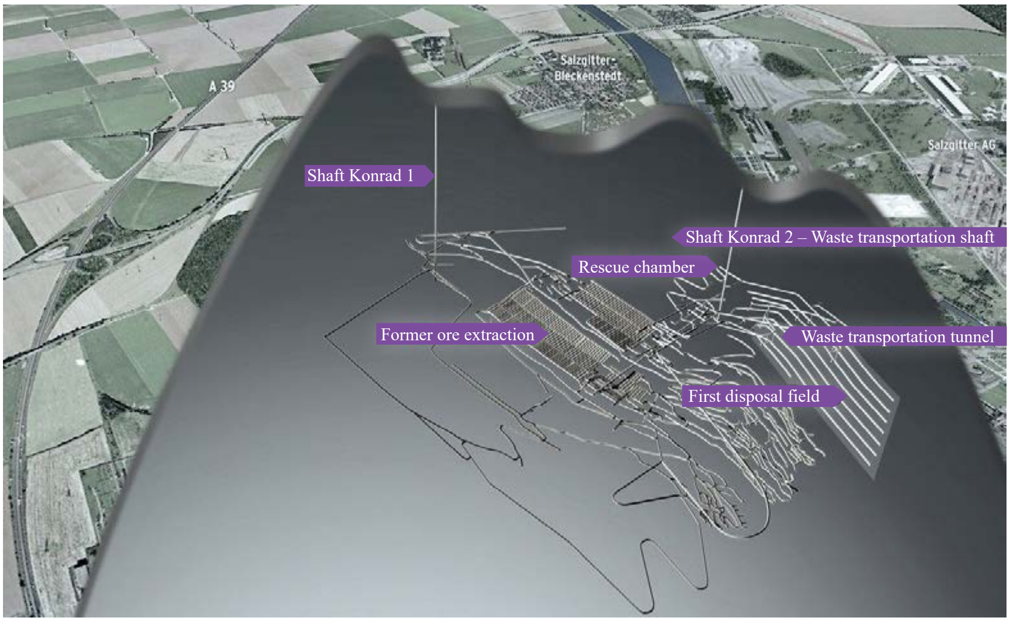

FIG. 49. 3-D model of the Konrad repository and former mine workings, Germany (courtesy of Bundesgesellschaft für Endlagerung mbH).

FIG. 50. Waste containers (cylindrical and cubic) meeting the WAC for disposal at Konrad, Germany (courtesy of Gesellschaft für Nuklear Service (left) and Bundesgesellschaft für Endlagerung mbH (right)).

4.5.2. Borehole disposal

Both shallow boreholes (tens to hundreds of metres) and very deep boreholes (several kilometres) have been evaluated for the disposal of specific categories of radioactive waste that are generally of relatively low volume.

4.5.2.1. DSRS borehole disposal concept

The concept of the disposal of DSRS — mainly arising from medical and industrial applications in boreholes was first proposed in 1995 during an African Regional Cooperative Agreement for Research, Development and Training related to Nuclear Science and Technology course hosted by the South African Nuclear Energy Corporation. Many Member States do not have the possibility to co-dispose DSRS inventories with other radioactive waste and have expressed an interest in economically feasible disposal solutions. Disposing DSRS inventories in boreholes was proposed as a possible cost effective solution. Since 1995, the idea has evolved into a well defined concept offering an internationally accepted solution for a wide spectrum of DSRSs for implementation in different geological environments and climatic conditions.

For disposal purposes, a DSRS is first conditioned into a waste package. The concept relies on the waste package to provide the primary engineered barrier to ensure radionuclide containment for the period accounted for in the safety assessment. Additionally, to facilitate handling and disposal, the concept uses standardized waste packages with a diameter of 115 mm, consisting of a 3 mm thick stainless steel capsule placed in a cementitious buffer that is, in turn, encased in a 6 mm thick stainless steel container (Fig. 51). The containers range in length from 250 mm to 600 mm.

FIG. 51. DSRS standardized waste package design. Stainless steel capsule (left) with a 7 cm diameter; interior cement buffer (centre) in the various sizes available (right).

The use of a corrosion resistant material (e.g. 316L stainless steel), combined with emplacement in the highly alkaline environment provided by the cement buffer, is expected to result in low corrosion rates and correspondingly long container lifetimes. Depending upon the geochemical environment of the site, waste packages designed in this manner can retain their safety function for several thousand years after disposal.

The waste packages will be emplaced in a 26 cm, high density polyethylene or steel cased borehole at a depth defined by a site specific safety assessment — typically several tens to hundreds of metres. After emplacement, the borehole will be backfilled with a cementitious material. The borehole design concept is illustrated in Fig. 52.

During closure of the borehole, a 10 mm thick stainless steel plate is placed at an angle of 60° above the disposal zone, to deflect any future inadvertent drilling away from the waste. To reduce the potential of the casing to act as a preferential pathway for radionuclide migration to the surface, the upper portion of the borehole will be removed to ensure that the borehole seal will be in direct contact with the surrounding rock formation. The uppermost portion of the borehole will then be plugged with bentonite.

Based on the requirements for disposal at intermediate depths given in SSG-1 [6], the minimum depth of disposal is 30 m below ground surface, corresponding to the depth used in SSG-1 [6] to differentiate between near surface disposal and disposal at intermediate and greater depths. A threshold of 30 m is considered appropriate to significantly reduce the likelihood of inadvertent intrusion related to common construction activities.

Site specific repository design details, such as the number of boreholes, depth of disposal and (where appropriate) borehole spacing, will depend on the number of sources requiring disposal, the type of sources and any limiting factors related to the site, as determined by the post-closure safety assessment.

Sources containing radionuclides with half-lives up to 30 years (e.g. ^{60}Co, ^{90}Sr or ^{137}Cs) will decay to insignificant levels of radioactivity after several hundred years, up to about 1000 years (e.g. 1200 years represents about 40 decay half-lives of ^{90}Sr or ^{137}Cs). Considering the intended lifetime of the waste packages and the specific conditions of the site, sources containing these radionuclides can be expected to decay below exemption levels of activity well before the package fails and the radionuclides could be released to the geosphere. However, radionuclides with longer half-lives, such as ^{226}Ra or ^{241}Am, are likely to enter the geosphere as the canisters eventually lose their integrity. For disposal of sources with longer lived radionuclides, the containment characteristics of the rock formations, such as permeability and the geochemical environment, will thus play an important role in the overall safety of the disposal system.

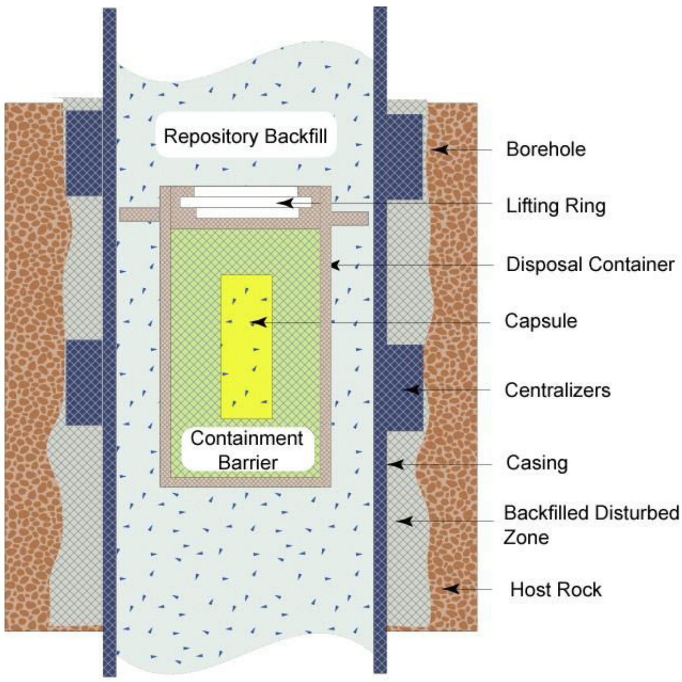

FIG. 52. Diagrammatic cross-section of the disposal zone in a borehole for the disposal of DSRSs.

A generic safety assessment [53] has been conducted assuming a reference source inventory consistent with DSRS inventories found in many African countries. The generic safety assessment considered three basic climate types (humid, seasonally humid and arid to semi-arid) in twenty different environment scenarios. The generic safety assessment concluded that a wide range of different geosphere situations and biosphere conditions could be suitable for implementation of the DSRS borehole disposal concept.

4.5.2.2. Very deep borehole disposal concept

Concepts have also been considered for disposal of long lived, high activity wastes in much deeper boreholes, at depths of 5 km or more, as illustrated in Fig. 53 [54]. The most advanced studies to date have been in the United States of America, where the United States Department of Energy has studied the feasibility and safety of the deep borehole disposal concept. Although not fully developed, the concept has attracted interest from several countries. A RD&D roadmap, which included plans for scientific investigations and engineering demonstrations, and a plan for a deep borehole field test, was drafted in 2014, although at the time of writing there was no work taking place on this project.

The deep borehole disposal concept could be suitable for Member States with small amounts of high activity radioactive waste that require or lend themselves to separate management. For example, the concept was considered for the disposal of very high specific activity separated caesium waste in the United States of America and has been

FIG. 53. Diagrammatic sketch of the deep borehole concept [55] (courtesy of Sandia National Laboratories).

suggested for disposal of plutonium waste. Although retrievability is a requirement in several Member States, this concept is intentionally designed to provide a very high level of isolation of the wastes, with retrieval after closure practically impossible, making it potentially attractive for the disposal of separated fissile materials. It has also been considered for the disposal of SF in both Sweden and the United States of America. However, significant RD&D will be required, as will the development of a comprehensive safety case before this concept could be considered suitable for implementation. The safety of deep borehole disposal is based on the very long term immobility of deep fluids, which is associated with low rock permeability and strong salinity gradients. Reducing conditions will sharply limit the solubility of dose critical radionuclides at depth, and high ionic strength of deep fluids can inhibit colloidal transport [55]. Currently available drilling technology allows completed boreholes to be constructed with diameters up to about 440 mm at 5000 m depth, with consequent constraints on waste package diameter [56]. There has been no practical experience of placing packages in relatively large diameter boreholes at these depths. A preliminary assessment of the safety of the deep borehole disposal concept was prepared by a programme in the United States of America [57].

5. CONCLUSION

Member States establishing a radioactive waste disposal programme will ideally make an initial consideration of the national waste inventory, followed by an evaluation of disposal concepts that are appropriate to the wastes and to possible siting options for disposal facilities. This publication provides information on the range of options currently available and on how to initiate and carry out a staged, requirements led, systems engineering repository design process. Guiding principles for design should be applied throughout sequential phases and milestones on an established path through siting, licensing, construction, operation and closure. Repository programmes entail commitments that often stretch over many decades. A clearly defined, transparent and stepwise design and option selection process, with intermediate milestones and deliverables, and formal exchange with stakeholders is essential.

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ABREVIATIONS

DSRS	disused sealed radioactive source
HLW	high level waste
ILW	intermediate level waste
LILW	low and intermediate level waste
LLW	low level waste
RD&D	research, development and demonstration
SF	spent fuel
SFR	final repository for short-lived radioactive waste
VLLW	very low level waste
WAC	waste acceptance criteria
WIPP	waste isolation pilot plant
WMO	waste management organization

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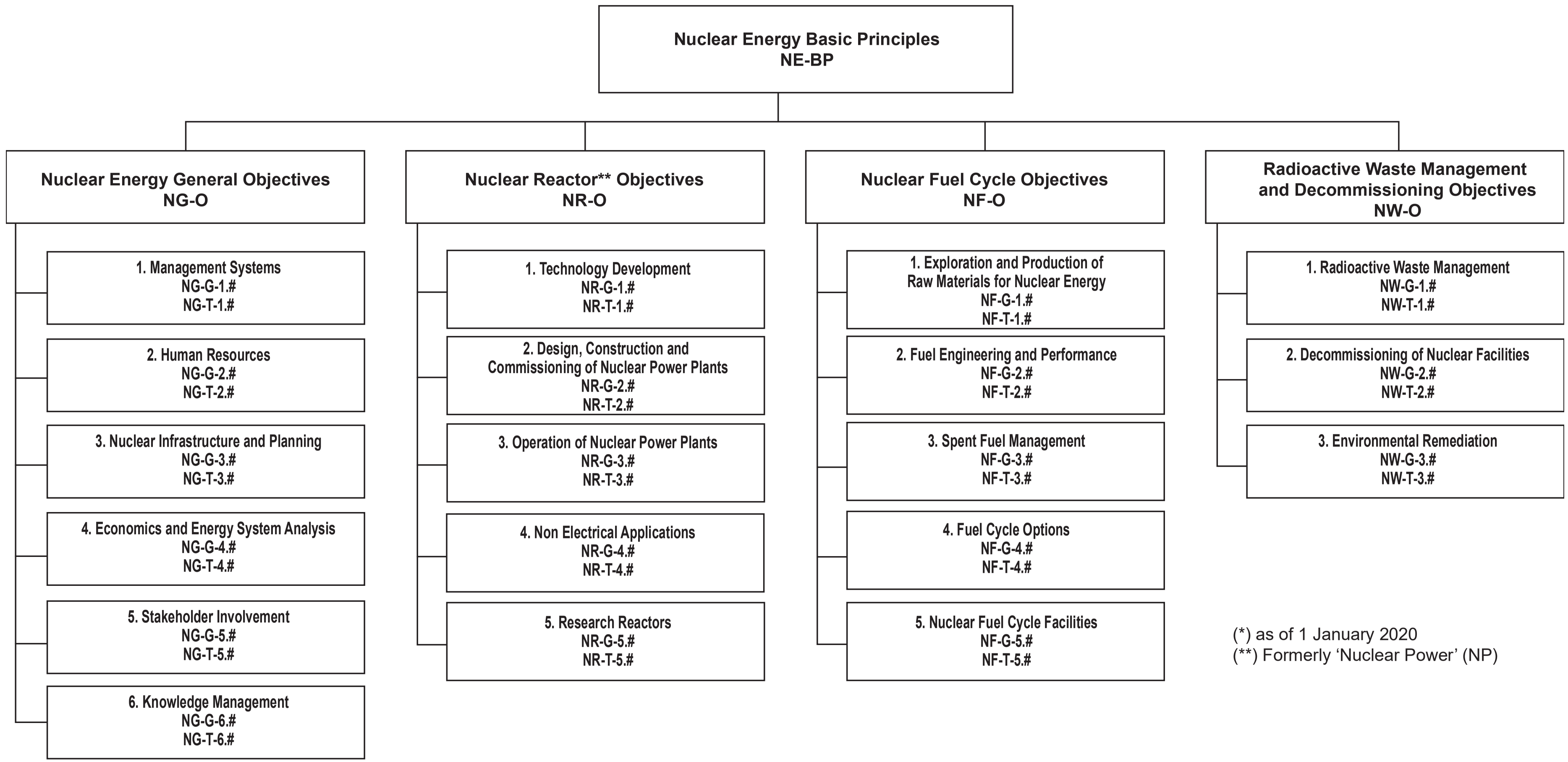

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