Deliverable 7.11: HITEC – Safety Case Guidance

Work Package 7

The project leading to this application has received funding from the European Union's Horizon 2020 research and innovation programme under grant agreement No 847593.

Document information

Project Acronym	**EURAD**
Project Title	**European Joint Programme on Radioactive Waste Management**
Project Type	**European Joint Programme (EJP)**
EC grant agreement No.	**847593**
Project starting / end date	**1st June 2019 – 30 May 2024**
Work Package No.	**7**
Work Package Title	**Influence of Temperature on Clay-based Material Behaviour**
Work Package Acronym	**HITEC**
Deliverable No.	**7.11**
Deliverable Title	**HITEC Safety Case Guidance**
Lead Beneficiary	**Nagra**
Contractual Delivery Date	**March 2024**
Actual Delivery Date	**May 2024**
Type	**Final report of WP7 Task 4.1**
Dissemination level	**Public**
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To be cited as:

Leupin O., Papafotiou A., Villar M. V., Levasseur S., Niskanen M., Hausmannová L., Mayor Zurdo J.C., Kirby M., De Lesquen C., Simo E., Sellin P.., Olin M. (2021): HITEC Safety Case Guidance. Interim version as of 21/05/2024 of deliverable D7.11 of the HORIZON 2020 project EURAD. EC Grant agreement no: 847593.

Disclaimer

Acknowledgement

This document is a deliverable of the European Joint Programme on Radioactive Waste Management (EURAD). EURAD has received funding from the European Union's Horizon 2020 research and innovation programme under grant agreement No 847593.

Status of deliverable		
	By	Date
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Reviewed (Reviewers)	Patrik Sellin [SKB]	27/02/2024
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Executive Summary

This report provides guidance for safety case development and repository optimisation based on the outcomes of EURAD WP7 HITEC (Influence of temperature on clay-based material behavior). Eleven teams have contributed to this synthesis of HITEC outcomes and writing the safety case guidance. Nine of those are Waste Management Organizations (WMOs), namely Nagra, SURAO, ANDRA, ONDRAF, BGE, NWS, POSIVA, SKB, and ENRESA. In addition, contributions and review has been provided by CIEMAT (Lead SotA Deliverables 7.1 & 7.2) and VTT (Lead WP 7).

Section 1 provides an introduction that includes the relevant background and the purpose of the safety case and repository optimization guidance, linked to the activities undertaken in HITEC.

Section 2 gives an overview of the safety cases currently implemented by the WMOs contributing to this guidance report. This includes an overview of the current disposal facility concept in each country and an outline of the corresponding safety case methodology. Links between safety requirements and safety functions are discussed, focusing on repository induced effects and particularly the thermal impact associated with heat generated by the radioactive waste.

Section 3 describes how HITEC project results are utilized by the WMOs and improve the respective safety cases. The discussion focuses on the one hand on HITEC impact on performance and safety assessments and on the other hand impact on repository optimization efforts.

Finally, Section 4 concludes with a summary and discussion based on the outcomes.

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Glossary

CAE	Claims, Arguments and Evidence
DB	Design Basis
DGR	Deep Geological Repository
DNLEU	Depleted Natural and Low Enriched Uranium
DSSC	Disposal System Safety Case
EBS	Engineered Barrier System
ESC	Environmental Safety Case
FEPs	Features, Events and Processes
HEU	Highly Enriched Uranium
HHGW	High-Heat Generating Waste
HLW	High-Level Waste
HSR	Higher Strength Rock
LHGW	Low-Heat Generating Waste
L(I)LW	Low (and Intermediate) Level Waste
LSSR	Lower Strength Sedimentary Rock
S(N)F	Spent (Nuclear) Fuel
SotA	State-of-the-Art
WDP	Waste Disposal Package
WP	Work Package

1. Introduction

1.1 Background / The HITEC Project

The early period in the evolution of a deep geological repository (DGR) after emplacement of heat-generating radioactive waste is characterized by elevated temperatures and associated thermal, hydraulic, and mechanical gradients in the materials and rock formation surrounding the waste. Clay-based engineered materials and rock formations are considered favourable for deep geological disposal thanks to their significant isolating and retaining properties.

HITEC is the Research, Development and Demonstration (RD&D) Work Package (WP) of EURAD aiming at improving the thermo-hydraulic-mechanical (THM) description of clay-based materials at elevated temperatures. HITEC aims to develop and document improved THM understanding of both host rock and buffer clay-based materials exposed to temperatures above 100 °C for extended durations. The WP allows to evaluate whether or not elevated temperature limits of 100 – 150 °C are feasible and safe for a variety of geological disposal concepts for high heat generating wastes (HHGW). The targeted analyses are determined on the basis of a state-of-the-art synthesis of safety case approaches and methods followed by WMOs currently planning or implementing DGRs (Task 1). In particular, the analyses on clay host rock formations (Task 2) evaluate the possible extent of heat-induced damage (e.g., from pressure increase associated with thermal expansion) and also the consequences of any such damage. The analyses on buffer bentonite (Task 3) determine if temperature has an impact on buffer swelling pressure, hydraulic conductivity, erosion or transport properties. The results feed into an updated state-of-the-art synthesis for safety case development.

1.2 Purpose

Purpose of Task 4 is the development of a guidance for safety case development and repository optimisation utilizing the outcomes of HITEC (Tasks 1 to 3). The Task 4 deliverable is documented in the present report. The target audience are organizations in the planning or implementation phase of DGR programmes. The guidance document is based on the compilation of state-of-the-art safety case approaches adopted by WMOs participating in HITEC, namely:

- ANDRA, France
- ENRESA, Spain
- Nagra, Switzerland
- NWS, United Kingdom
- ONDRAF/NIRAS, Belgium
- Posiva, Finland
- SKB, Sweden
- SÚRAO, Czech Republic

These are updated with relevant scientific information gained from HITEC and utilized to improve the safety case and optimization efforts in each case. In particular, the WMOs evaluate the use of bentonite and how knowledge from Task 3 allows reduce uncertainties and increase margins by which safety functions can be fulfilled. In terms of repository optimization, the WMOs assess consequences of higher temperatures associated with increased canister loading and to what extent the performance of the bentonite is still intact. With regards to clay host rock, the WMOs (re-)assess threshold criteria used to ensure integrity of the geological barrier in their safety case. In terms of repository optimization, it is likewise evaluated how higher thermal loads (i.e., associated with changes in canister pitch and/or tunnel spacing) can affect the likely performance of the geological barriers. Overall, the gained process understanding at higher temperatures in both the bentonite barrier and the clay host rock contributes to define (or re-assess) performance criteria and their thresholds in higher temperature ranges.

2. Overview of WMO safety cases

2.1 Introduction

The following sections provide an overview of the safety cases implemented by the WMOs on a case-by-case basis. These include Nagra, Switzerland (Section 2.2), ONDRAF/NIRAS, Belgium (Section 2.1), SURAO, Czech Republic (Section 2.3), Posiva, Finland (Section 2.3), ENRESA, Spain (Section 2.3), NWS, United Kingdom (Section 2.1), ANDRA, France (Section 2.2), and SKB, Sweden (Section 2.4).

2.2 ANDRA

2.2.1 Current Cigeo geological disposal project and safety concept

2.2.1.1 General presentation of Cigéo

The purpose of the Cigéo (centre de stockage géologique profonde) geological disposal facility for HLW (High-Level Waste) and ILW-LL (Intermediate Level Waste, Long Life) is to allow the safe disposal of radioactive waste in order to eliminate or reduce the burden to be borne by future generations.

Following final closure, Cigéo is designed to isolate the waste from humans and the biosphere and to confine it within a deep geological formation to prevent dissemination of the radionuclides and chemical toxic elements contained in the waste. These functions are performed over very long-time scales, passively, i.e. without the need for maintenance or monitoring, as explained by the safety guide for the final disposal of radioactive waste in a deep geological formation, published by the French Nuclear Safety Authority (ASN) in 2008 (ASN, 2008). This relies on the chosen geological medium, and specifically the host rock, and on the design of the disposal facility, and specifically its architecture and its engineered components.

In accordance with the safety guide number 1 for the final disposal of radioactive waste in a deep geological formation (ASN, 2008), the components of the disposal system that play a role in the post-closure safety are grouped in three categories:

- the waste disposal packages;
- the engineered components ensuring the backfilling and sealing of the disposal cells and galleries, as well as the access shafts and ramps;
- the host rock in which the Cigéo underground facility is situated: the Callovo-Oxfordian clay formation.

Only the last two items are shortly described here. Information about the waste packages can be found in Andra, 2016 and 2022a (in French).

2.2.1.2 Underground facility and closure structures

The underground facility has been designed with separated zones for the disposal HLW (and possibly also nuclear fuel that will not be reprocessed) and ILW-LL in order to limit phenomenological interactions between these different waste categories (Figure 2-1).

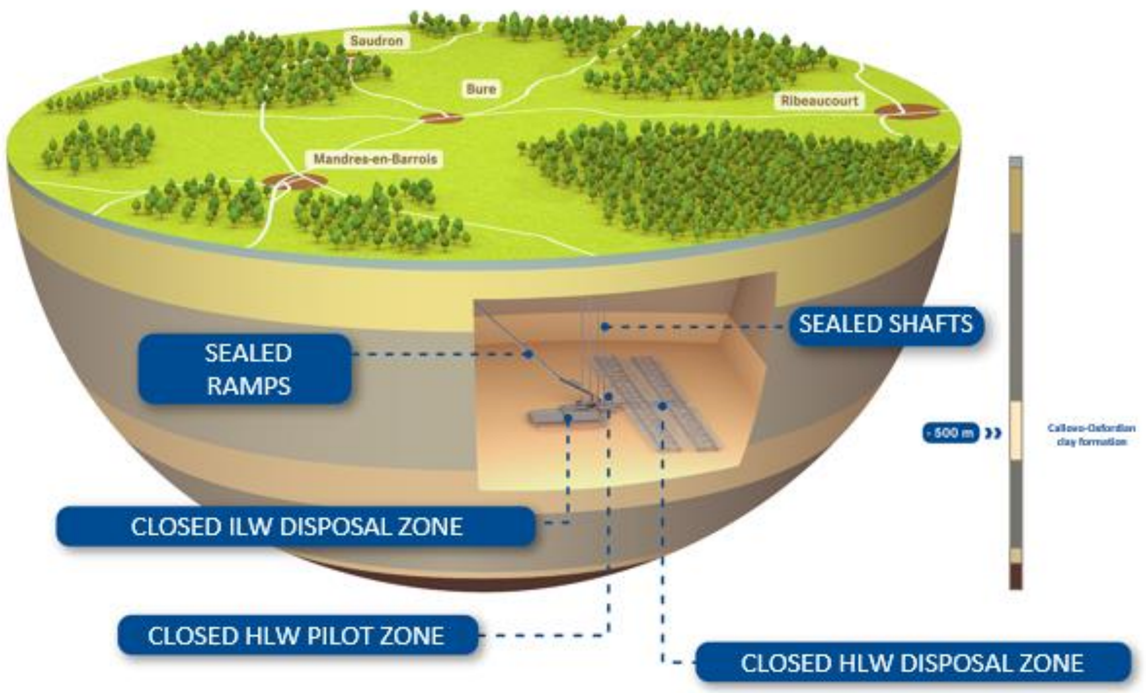


Figure 2-1 Illustration of the Cigeo underground facility after its final closure (subject to a law giving permission being passed)

The HLW from the reprocessing of spent fuel consists of fission products and minor actinides separated from the uranium and plutonium, calcined and incorporated into a glass matrix, which is poured into a stainless-steel canister.

The ILW-LL consists mainly of the structural elements of spent fuel and waste associated with the operation, maintenance and dismantling of nuclear facilities. The containers used by the waste producers for conditioning the ILW-LL are of different types, shapes and sizes. They can be made from non-alloy steel, stainless steel, standard reinforced or fibre-reinforced concrete.

The waste is disposed of in "disposal packages", which are emplaced in "disposal cells" in the underground facility.

There are two categories of disposal cells (Figure 2-2):

- HLW disposal cells containing only one disposal package per cell section; each disposal cell has a "sleeve" (steel casing) designed to provide mechanical support to the cell at least during its operation;
- ILW-LL disposal cells containing several disposal packages per cell section. The mechanical stability of these during operation is guaranteed by a concrete liner (left in place on closure).

Figure 2-2 Schematic diagram of an HLW cell and arrangement of waste packages in an ILW-LL cell

Each disposal cell is served by one or more access drifts for its construction, the loading of the waste packages and management until closure. The lining of the access drifts ensures their mechanical stability during the operating phase (it is also left in place on closure).

The HLW disposal zone is a dead-end grouping of cells and drifts The HLW0 zone for moderately exothermic waste packages will be used from the industrial pilot phase, and the HLW1/HLW2 zone will be used for highly exothermic waste packages (Figure 2-1). The HLW disposal cells are arranged in each of the zones mentioned above into one or more separate sections.

The disposal zones are linked to the connecting structures between the surface and the underground facility by "connecting shaft and ramps". They consist of two ramps and five vertical shafts all grouped in the same zone.

After the final closure of Cigéo, all the drifts in the underground facility, the ramps and the shafts will be backfilled and sealed in places (Figure 2-3).

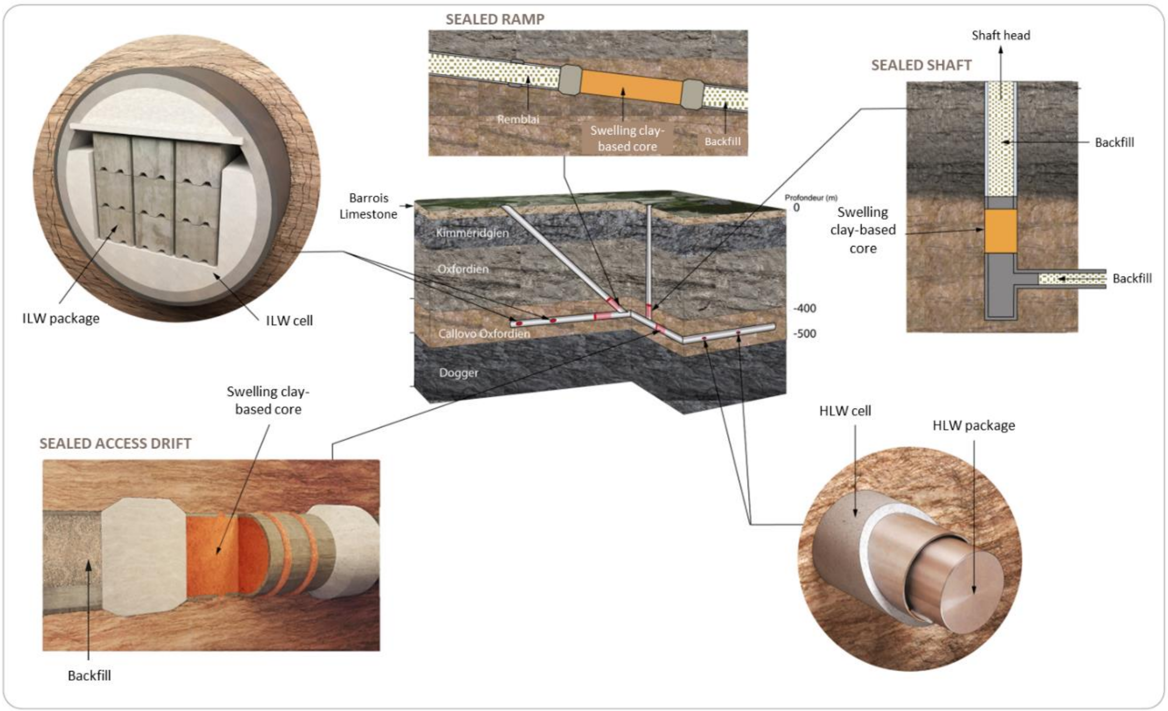


Figure 2-3 Illustration of the closure structures in the disposal facility

The backfill will be emplaced at the end of the operational phase in all the drifts. Its function is to limit the development of the fractured zone after the rupture of the lining strengthening the sides and roofs of the drifts in the underground facility. It helps therefore to preserve the favourable characteristics of the COx. The backfill will consist of excavated clay rock mixed with sand.

The seals are designed to prevent the water flows between the underground facility and the overlying formations and to limit the water circulation in the drifts. There are three categories of seals: vertical seals in the shafts, inclined seals in the ramps and horizontal seals at the main level of the repository (Figure 2-3). At this stage, they comprise a core based on swelling clay mixed with additives such as siliceous or calcareous sand occupying the entire cross-section area of the shaft/ramp/drift. The swelling clay is in direct contact on all its length with the clay rock in the shafts and ramps, the concrete linings being totally removed. For horizontal seals, only a few portions of the concrete lining are removed due to the nature of the clay rock, inducing partial contact between the swelling clay and the host rock. For the inclined and horizontal seals, two designs are currently studied with or without concrete containment walls.

The seals increase the 'hydraulic resistance' of the drifts. There are at least two seals on each route between a section and the surface-bottom connections; this arrangement enables hydraulic head losses to be distributed within the drifts. In the event of loss of hydraulic function of the vertical or inclined seals, the horizontal seals limit the water flows and the flows of radionuclides in solution along the drifts to the bottom of the surface-bottom connections.

The seals are also designed to let gasses, generated by the corrosion of the waste packages and other metallic components left in place after closure, flow through in order to limit the pressure build-up in the installation and thus preserve the properties of the Callovo-Oxfordian formation during and after the hydraulic-gas transient.

The seals are not in contact with exothermic waste packages and their behaviour when exposed to high temperature is not critical in the Cigéo project.

2.2.1.3 Callovo-Oxfordian clay formation

In the ZIOS (Zone de transition), the depth of the Callovo-Oxfordian (top between 340 m and 530 m, base between 500 m and 675 m, and thickness varying from 140 m to 160 m from south-west to north-east) and of the Cigéo underground facility, protect the disposal system from erosion processes, from the effect of an earthquake, or from a simple 'normal[1]' human intrusion, and ensures satisfaction of the function "isolate the waste".

The repository design aims to take advantage of all the favourable properties of the Callovo-Oxfordian while minimising the thermal, mechanical, hydraulic and chemical stresses likely to be generated by the repository. The aim is to keep the Callovo-Oxfordian in a state as close as possible to its current equilibrium state. The favourable physical and chemical characteristics of the Callovo-Oxfordian preserved in this way, along with its thickness, enable the radionuclides and toxic chemicals to be contained over long timescales while limiting their release and migration.

2.2.2 Safety assessment

2.2.2.1 Post-closure safety strategy

Based on national regulatory texts including Safety Guide No. 1 (ASN, 2008), on international standards related to post-closure safety and on experience feedback acquired by Andra, a post-closure safety baseline has been defined in order to present the post-closure safety assessment approach for the

[1] It is understood by 'normal', human activities such as road construction or residence construction.

Cigéo geological disposal. This approach is part of an iterative multidisciplinary design/scientific and technological knowledge/safety studies system updated at each milestone in the progressive deployment of the BNI (Basic Nuclear Installation).

A certain number of principles govern the post-closure safety assessment approach, including, in line with ASN guide no. 1 (ASN, 2008) (i) the post-closure safety principles and functions of the repository, (ii) the objectives of protecting human health, and (iii) the method for assessing safety after repository closure. The objectives of the post-closure safety assessment include three fundamental aspects:

- The verification of the favourable character, for safety, of the performance of the components participating in the performance of the safety functions;
- The evaluation of the disturbances brought to the disposal system by the interactions between the various components;
- The estimation of the impacts on human health for a set of scenarios representative of potential evolutions of the disposal system.

To meet those objectives, Andra's approach relies on:

- the acquisition of scientific and technological knowledge (characteristics of the geological medium, in particular the host rock, waste and waste packages, engineered components, radioactive substances) in order to understand the physical and chemical phenomena which govern their behaviour and their evolution, over very long periods;
- a repository design in line with the definition of safety function and the state of scientific and technological knowledge (disposal containers, underground structures and their organisation/location in the host rock, general architecture, package handling operations, seals in the ramps, shafts and galleries) in order to propose a repository architecture once it has been definitively closed that meets the long-term safety objectives;
- the description (e.g. understanding) of the (phenomenological) behaviour of the repository and its geological environment (in particular the interactions between the waste, the engineered components and the Callovo-Oxfordian host rock and the geodynamic evolutions) in order to understand the thermal, mechanical, chemical and hydraulic evolutions as well as the release of radioactive substances in time and space, using in particular modelling and digital simulation;
- the post-closure safety assessments carried out based on knowledge/design/safety iterations.

2.2.2.2 Post-closure safety functions

In order to meet the fundamental objective of protecting humans and the environment against the risks associated with dissemination of the radioactive substances and toxic elements in the waste, Andra has identified and organised the post-closure safety functions of the Cigéo disposal facility as follows.

The first fundamental safety function consists in isolating the waste from surface phenomena and human actions. The site of the disposal and the depth at which it is located protect it from surface phenomena, erosion and everyday human activities, which should only affect the ground down to a depth of less than 200 m on a scale of hundreds of thousands of years. In accordance with the safety guide for the geological repository, the protection of humans must be guaranteed "*without depending on institutional control, which cannot be relied on with any certainty for more than a limited period of time (...) (500 years)*". The memory of the repository will be maintained for as long as possible. The technical solution chosen will provide a reasonable level of confidence over a very long period, such that the possibility of inadvertent human intrusion does not have to be considered until after 500 years, in accordance with the safety guide N°1 from ASN (ASN, 2008).

The second fundamental safety function is to limit the transfer into the biosphere of the radioactive substances and toxic elements in the waste. This means controlling the physicochemical degradation of the waste, the packages and the engineered components, keeping the radioactive substances and toxic elements as close as possible to their sources, and controlling the transfer pathways that could, in the long term, lead these elements into the biosphere. They are:

- aqueous pathways, as the substances are liable to dissolve in water and migrate to the surface;

- gaseous pathways, as certain radionuclides can migrate in this form.

Water is the main factor in the alteration of waste packages and the main transfer vector of radioactive substances and toxic elements. Controlling the aqueous pathways is therefore a key objective of post-closure safety.

Limiting the transfer of radioactive substances and toxic elements by water is the purpose of the following three safety functions:

1. preventing the circulation of water;
2. restricting the release of radionuclides and toxic elements and immobilising them in the repository;
3. delaying and reducing the migration of radioactive substances and toxic elements released from the packages and then from the disposal cells.

While the Callovo-Oxfordian formation plays a central role in long-term safety, the packages and the repository's engineered components, specifically the underground facility's architecture on completion and closure structures, also contribute to containment of the radionuclides and toxic elements and to maintaining the conditions for flows of water through the facility to be very slow.

Therefore, a supplementary function has been defined "to preserve the favourable characteristics of the Callovo-Oxfordian clay formation and engineered components contributing to post-closure safety functions".

In order to limit the consequences of interaction phenomena, Andra has defined restrictive and constructive design provisions to preserve the Callovo-Oxfordian formation (definition of an undisturbed guard) and limit transfers through the structures (galleries, ramps and shafts). All "phenomena/processes of internal origin" which are linked to disturbances or transients of mechanical, thermal, hydraulic, gas, chemical, bacteriological and radiological origin, and their possible interactions, are taken into account. Phenomena of external origin such as the geodynamic evolution of the site are also taken into account.

Andra has adopted a set of general principles for the design of the disposal facility which incorporate the preservation of the favourable characteristics of the Callovo-Oxfordian layer selected for this purpose. Where applicable, the safety functions are broken down into high-level requirements which guide the overall disposal design (architecture, engineered components). Thus, for example, the design incorporates the thermal effect from the disposal waste in order to preserve the Callovo-Oxfordian.

2.2.2.3 Post-closure safety assessment

The post-closure safety assessment includes:

- a qualitative analysis of risks and uncertainties, that aims at identifying and defining the scenarios and their classification;
- a quantitative assessment of the chosen scenarios that is based mainly on numerical simulation.
- an analysis of the results and definition of lessons with regard to the objectives of protecting long-term interests and with regard to the performance of the disposal system and its robustness.

Based on the phenomenological understanding of the disposal system and its evolution and the models and ranges of parameter values available as well as the associated residual uncertainties, the risk and uncertainty analysis leads to:

- identify the potential causes of malfunction of the components contributing to the performance of the safety functions and to qualitatively assess the risk of non-performance of a safety function;
- justify certain choices of representation of the components taken individually or as a whole.

Through a qualitative but reasoned assessment, the analysis makes it possible to propose a management of the risks and uncertainties analysed:

- by design choices using provisions that make the system insensitive to these uncertainties;

- by taking them into account in appropriate scenarios when no design provision makes it possible to mitigate these risks and uncertainties. A reasoned assessment of the residual risks and uncertainties leads to the choice of assumptions (representation, models, data) to be used in the scenarios:
  - either in the normal development scenario if the residual risks or uncertainties do not lead to the loss of a safety function.
  - either in an altered evolution scenario or in a What-if type scenario in the event of loss of safety function. It is through a reasoned assessment of the causes of dysfunction identified that the classification in scenario of altered evolution or “What if” is proposed.

Four types of scenarios are defined:

- the normal evolution scenario (NES) This scenario aims to represent the repository system as envisaged by the designer, assuming that all post-closure safety functions are performed. This scenario is defined based on acquired knowledge to represent the expected (or normal) long-term evolution of the repository system by considering "certain or very probable" events and processes (which refers to the recommendations of safety guide No. 1 of the ASN (ASN 2008). The uncertainties associated to the thermal transient are covered in the NES;
- altered evolution scenarios (AES) representative of events or processes deemed unlikely based on the knowledge acquired but likely to lead to the loss of a safety function or a significant degradation in the performance of the components that contribute to its realization. This category includes the malfunction scenarios of the components contributing to the post-closure safety functions identified by the analysis of risks and uncertainties;
- scenarios qualified as "What-if" based on the highly unlikely nature of the events taken into account, or based on postulated events to consider, for example, the loss of one or more safety functions. These hypothetical scenarios make it possible to “push” certain malfunctions to the extreme and to show the robustness of the disposal system as a whole;
- scenarios of inadvertent human intrusion, called SIHI, in the disposal system, due to ignorance of its existence. They are due to drilling of boreholes.

The thermal transient and the related uncertainties are taken into account in a conservative way in the design of Cigéo and are therefore included in the Normal evolution scenario.

2.2.3 Current approach for consideration of the disposal-induced effects – focus on thermal impact

The primary packages of HLW waste are characterised by heat release, which is taken into account in the dimensioning: Andra has chosen to limit the temperature and the thermo-hydro-mechanical effects on the Callovo-Oxfordian in a range (i) avoiding irreversible degradation of the disposal system characteristics contributing to the safety functions, particularly the Callovo-Oxfordian, and (ii) within which the processes governing changes in the disposal system can be represented and modelled reliably. In particular, it implies a cell temperature always below 100°C (in practice, the criterion applied in the clay rock is 90°C). Compliance with this range is based on the relationship between the heat output of the packages, which is related to their radiological content and the duration of their prior interim storage, and the dimensioning of the underground facility.

2.2.3.1 Preservation of the favourable properties of the Callovo-Oxfordian

To preserve its favourable properties, the clay rock should not be exposed to overheating generated by the high-level waste. A temperature increase to above 100°C in the rocks can lead to complex coupled processes. Moreover, under such conditions, there are numerous experimental difficulties in acquiring knowledge about the processes. Following on from the 2005 and 2009 reports, it has been decided to define a maximum permitted temperature of 100°C for the clay rock, with a margin of 10°C, giving a design-basis maximum temperature of 90°C in the clay rock, allowing the uncertainties in the thermal properties and the thermal models of the repository to be taken into account.

2.2.3.2 Thermal pressurisation

A temperature increase also generates an increase in pore pressure in the clay rock: it results from its low permeability and the difference in heat expansion between the pore water and the solid phases of the clay rock. Depending on the interstitial pressure and stress levels reached, and the loading pathway followed (extension, shear, etc.), the combination of these two increases may lead to diffuse and/or localised damage to the clay rock (fracturing/break). Andra has therefore also decided to dimension the HLW disposal sections to avoid a risk of fracturing. The selected indicator is the maximum Terzaghi effective stress (σ_{eff}), linked with tensile behaviour with the following criteria:

- σ_{eff} (Terzaghi) < 0 as reference;
- σ_{eff} (Terzaghi) < tensile strength of the rock (1.5 MPa), variant for HLW1/HLW2 waste.

Lastly, Andra has decided to verify that the temperature changes, on the scale of the Callovo-Oxfordian and for up to a million years, do not lead to significant irreversible mineralogical transformations of the clay rock. A clay rock temperature below 70°C after when waste comes into contact with water is desired (after corrosion of the waste disposal container).

The planar tunnel architecture of the HLW cells is particularly favourable to dissipation of the heat from HLW waste in the Callovo-Oxfordian, taking the economic factors into account. Thermal and THM (thermo-hydraulic and mechanical coupled processes) dimensioning of the HLW1/HLW2 sections leads to spacing out the waste packages in the cells to reduce the thermal load density. This provision is not necessary for the HLW0 waste. The space between two HLW1/HLW2 disposal packages is occupied by voids or less exothermic but similar HLW wastes package to limit voids in the overall cells.

The spacing between the cells in a section is also determined to comply with the thermal and THM criteria above.

Reduced-scale and full-scale in-situ heating tests in the Meuse Haute-Marne Underground Research Laboratory (MHM URL) have demonstrated the understanding of heat dissipation in the Callovo-Oxfordian (heat transfer predominantly by diffusion) and the validity of the models and the simulation tools for the thermal dimensioning of Cigéo.

2.2.4 Impact of improved understanding on safety case

2.2.4.1 Impact of the thermal transient on the buffer performance

In the French concept, the engineered barrier systems are located in the connecting drifts and will not be in contact with exothermic waste packages. The swelling clay in the seal core will therefore not be exposed to high temperatures.

2.2.4.2 Impact of the thermal transient on the host-rock performance

The effects of temperature increase are well understood and can explicitly be considered: for instance, the influence of temperature on transport and retention properties is taken into account by applying corrective factors to the effective diffusion coefficients and the retention parameters of certain radionuclides. In addition, the associated design and safety options are aimed at preventing irreversible effects of temperature on the Callovo-Oxfordian properties and to prevent its influence on the migration of radionuclides and toxic elements. In particular, these operations include:

- limiting the exothermicity of disposal packages;
- spacing between packages, where necessary;
- spacing between the exothermic disposal cells;
- use of sealed disposal containers for vitrified waste;
- separation of the ILW-LL disposal section from the HLW disposal section.

Simulations that take into account these design measures indicate a short-term thermal transient, when compared with the period required to dissolve the glass and the time it takes for radionuclides to migrate into the Callovo-Oxfordian.

A short version of the post-closure evolution of Cigéo (“storyboard”), with emphasis on the thermal and pore-pressure transients, is given below.

Within an HLW repository zone, temperature increases very fast in the near-field around the HLW cells: at the cell wall (yellow curve on Figure 2-4), it reaches a maximum of 80°C between 10 and 15 years after loading the packages. This peak is limited by design (max 90°C at the container wall) and because of the low equivalent conductivity of air in the container / liner annulus. Temperature at the cell wall decreases then rapidly over a century, and more slowly afterwards.

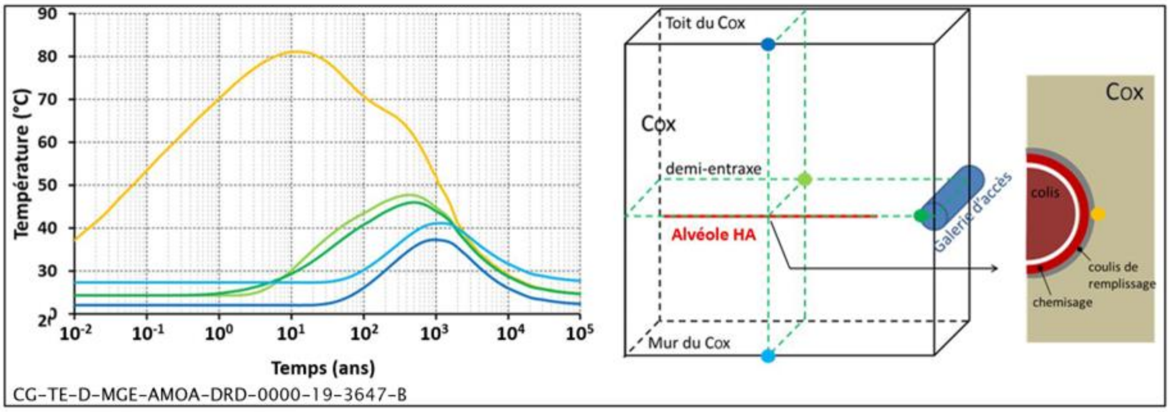


Figure 2-4 Graphical representation and evolution over time of the temperature at different locations around an HLW cell in the HLW1/HLW2 repository zones

In the far-field, temperature peaks are lower and happen later. At mid-distance between two cells, (corresponding to around 25 m from the HLW cells), the peak is reached between 400 and 500 years; it varies between 40 and 50°C, depending on the package type and their location within the repository. Around the HLW repository zone, the temperature peak at top and base COx is reached after 1000 years and reaches around 35 to 40°C (blue curves on Figure 2-4).

This temperature increase generates an increase in pore pressure due to the low permeability of the host formation and the difference in thermal expansion between the pore water and the solid phase of the clay rock. As mentioned above, the intercell spacing is designed to avoid any risk of thermal fracturing. Previous studies (Seyedi et al., 2020 and Bumbieler et al., 2020), showed that a good prediction of the evolution of both the temperature and the pore pressure can be achieved if an anisotropic poro-elastic behaviour is considered.

The corrosion under post-closure anoxic conditions of the steel in primary waste packages, disposal containers (solid steel of HLW containers and reinforcement of ILW-LL disposal containers) and underground facility structures (HLW cell metallic sleeves, reinforcement of the concrete liners of all the drifts and the ILW-LL cells, etc.) produces hydrogen. Gas generation by radiolysis, mainly of organic materials in ILW-LL waste, is added to this hydrogen source term. Production continues for several tens of thousands of years, and the production rate varies over time as a function of the available surface areas of steel in the case of corrosion and the decrease in dose rates in the case of radiolysis.

In the first thousand years, the hydraulic-gas transient is influenced by the thermal pressurisation around the HLW repository zone and by the desaturation of the host formation in the excavated-damaged zone (EDZ) around the galleries that were ventilated during the operating phase. After this period, the closure structures are resaturated and no gas flow can reach the Oxfordian. In a longer term, the hydrogen production exceeds the dissolution capacity of the pore water in the COx, and the gas pressure increases in the whole repository. The peak is reached a few tens of thousands of years after closure and lasts for around a hundred thousand years, much later than the thermal transient peak.

Andra checks with respect to the gases and the function “preserve the favourable characteristics of the Callovo-Oxfordian clay formation and components contributing to post-closure safety” that the containment performance of Cigéo is maintained despite this disturbance. The aim is to make sure that

(i) the gases do not affect the performance of the various components of the disposal system, in particular the undisturbed clay rock around the disposal sections (for example by fracturing as a consequence of high gas pressure), and the engineered structural components contributing to the safety functions, and (ii) that the gas transfer pathway does not put the disposal system in communication with the biosphere.

2.2.5 Conclusion

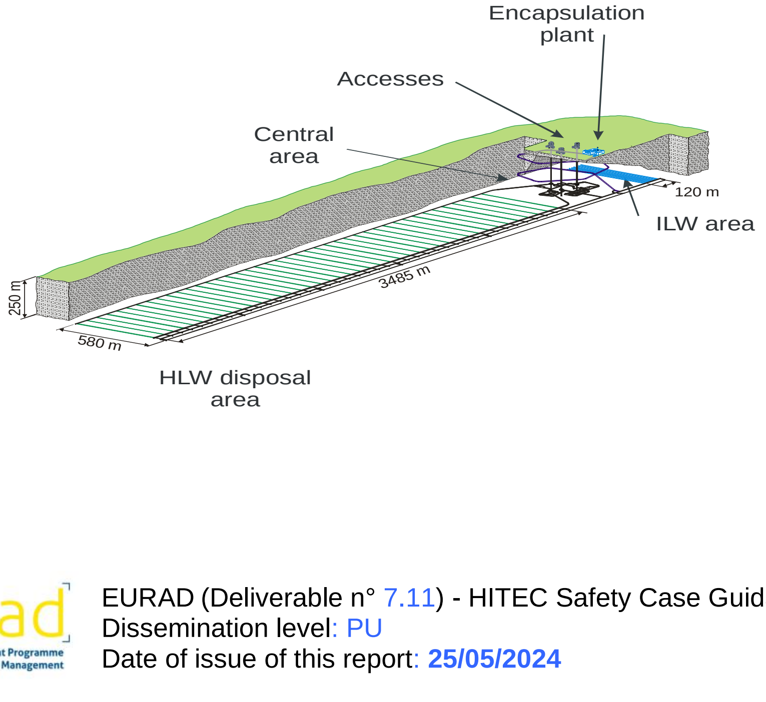
Andra submitted the Safety Options Report for Cigéo to the French Nuclear Safety Authority (ASN) in April 2016. In its opinion of 11 January 2018, ASN considered that "overall, the project has attained a satisfactory level of technological maturity at the Safety Options Report stage". Cigéo's safety relies to a large extent on the Callovo-Oxfordian clay formation within which the underground facilities will be built. The acquisition of scientific and technological knowledge over almost 30 years has given Andra a good understanding of the effect of temperature on its behaviour. The thermal load generated by the high-level waste is considered as a characteristic quantity for the post-closure safety of Cigéo and the thermal transient and the related uncertainties are taken into account in a conservative way in the design of Cigéo. They are included in the Normal evolution scenario in the safety assessment.

2.3 ENRESA

2.3.1 Current repository and safety concepts

The Spanish current repository concept in plastic clay rock is based on the disposal of spent fuel and HLW in carbon steel canisters in long horizontal disposal galleries (HLW disposal area). There is an independent area (ILW area) for the disposal of ILW. Canisters are disposed of in cylindrical disposal cells constructed with pre-compacted bentonite blocks of 1.700 kg/ m^3 dry density (to achieve a final dry density of 1.600 kg/ m^3). The blocks are initially non-saturated (degree of saturation of 66%). The disposal galleries of 580 m in length and 2.4 m in diameter are located at a depth of 250 m in the host formation. A 0,3 m thick concrete liner is required to deal with the plastic nature of the clay host rock.

The separation between canisters is determined mainly by thermal constraints. Separations of 1 m between canisters and 50 m between disposal galleries have been established, in order not to exceed a temperature of 100 ºC in the bentonite. Actual separation is a function of the properties of the host rock. Once a disposal gallery is completed, it is sealed with a 6 m long seal made of bentonite blocks and closed with a concrete plug at its entry. After completion of all the disposal galleries, the main drifts, ramp, shafts, and other remaining rock cavities will be backfilled with compacted clay from the excavation of the repository, and subsequent projection of clay pellets in the remaining openings. The concept is shown in Figure 2.5.



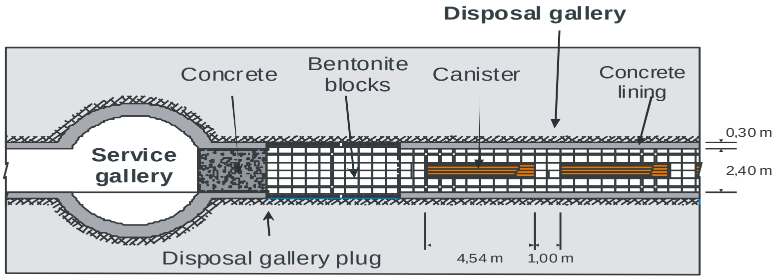


Figure 2.5: Schematic diagram of the geological repository in clay rock (Enresa 2004)

The clay host formation considered for this design is generic although it is considered compatible with the real characteristics that feature some Spanish clays. It is assumed that this reference formation, located at its top around 150 m deep, is composed of massive clays of high plasticity of lacustrine origin. Its longitudinal extension is greater than 8 km and exceeds 4 km in the transverse direction, the layer thickness is approximately 200 m.

Enresa´s safety concept considers the isolation of the radioactive waste from the biosphere in the long term, through its disposal in a system of multiple barriers housed in a deep geological formation, which ensures the containment safety function by minimizing its contact with water and delays and dilutes potential radionuclide releases into the biosphere (retardation safety function).

The barriers or components of this concept are artificial (or engineering) and natural. The steel canister acts as a containment barrier during an initial period in which disintegrates most of the fission products. The buffer and sealing material (bentonite) reduce the flow of water around the canister, protects it, provides a favourable chemical environment, and slows and attenuates the release of radionuclides into the geological barrier. The waste is very stable and hinders the dissolution process of radionuclides. The geological formation provides engineering barriers with a chemically, mechanically, thermally, and hydrogeologically stable environment in the long term and retards and limits the flow of released radionuclides into the biosphere. The safety of the concept does not rest on a single barrier, but on a combined action of different barriers with different safety functions over time.

2.3.2 Safety assessment methodology

The methodology developed considers the following steps (Enresa 2003):

- *Description of the disposal system*. It represents the initial state of the system. The quantities and characteristics of the radioactive waste are identified at this stage, the characteristics of the site and its biosphere, the waste canister and the materials of backfilling and sealing, and the geometric definition of the facilities that constitute the repository.

- *Analysis of possible future evolutions of the system (scenarios),* considering the factors (features, processes, and events) that could exert a significant effect on the evolution of the disposal system. A reference scenario, a climatic scenario and several alternative scenarios are considered that will later be the subject of quantitative analysis. The scenario development begins with the process of identifying the factors that, depending on the properties of the waste, the design of the repository and the clayey formation, can appear at any moment of time and affect the security of the disposal system. Next, the most significant factors, which once integrated will form the so-called Reference System and will allow the description of a predictable hypothetical evolution of the disposal system, which is called the Reference Scenario, as well as the rest of the scenarios.

- *Analysis of the performance of the barriers* of the disposal system in the Reference Scenario. The performance of the physical-chemical form of the waste, the container, the bentonite barrier, and the geological barrier are individually modelled. Consequences on the barriers of the hydrogeological and thermomechanical processes are considered, and the geochemical evolution of the system is analysed.

- *Analysis of consequences*, by a detailed analysis of the transport of radionuclides from the repository to the biosphere. Calculations are made for the potential radiological consequences that would result from the Reference and the other scenarios on the human being.

- *Analysis of results*, sensitivity and uncertainty analyses and comparison with the established safety criteria. Conclusions about the performance of the barriers and on the safety of the disposal system.

ENRESA 2003 used the probabilistic method for the analysis of consequences of the assessment, implemented in the GoldSim software (Golder Associates). The deterministic method attempts a detailed representation of the processes related to the release and transport of radionuclides. However, most of these are complex and are not the most suitable for the realization of repetitive global calculations that can statistically simulate the performance of the system. For this reason, other simplified models have been used in the probabilistic calculations of the exercise. The analysis and calculations extend at least up to a million years.

The fundamental criterion regarding safety has been stablished by the Spanish safety authority (CSN) to ensure that the annual risk of harmful effects after repository closure does not exceed 10^{-6} for a representative individual in the group exposed to the greatest risk. This risk criterion corresponds to an effective dose of 0,1 mSv/y that is much lower than those due to naturally occurring background radiation.

2.3.3 Repository induced thermal effects

Different repository induced thermal effects were considered in the disposal system and its barriers, namely the host rock and the bentonite buffer (Enresa 2003).

Since the clay host formation considered in Enresa 2003 is generic, no performance indicator/target was formulated regarding the maximum calculated temperature in the host rock and its maximum paleotemperature. Consideration was only given to the thermally induced increase of the pore pressure at the repository level, due to the differential thermal expansion of water and solid rock. To assess this effect, the assigned performance target stablishes that the pore pressure is below the lithostatic pressure at the repository level.

Further thermal restrictions on the repository performance were defined in the biosphere, namely the maximum permissible temperature increases in the upper shallow aquifer and on the land surface. Performance targets are $\Delta T < 5$ ºC in the upper aquifer and $\Delta T < 0{,}5$ ºC on the land surface.

Related to the bentonite buffer performance, consideration was given to avoid excessive temperatures that could result in mineralogical and chemical alteration of the bentonite and jeopardize its safety functions. Performance target is set at $T < 100$ ºC on the entire buffer thickness.

2.4 Nagra

2.4.1 Current repository project and safety concept

The current Nagra generic repository project envisages two disposal areas, one for Spent Fuel (SF) and High-Level Waste (HLW disposal area) and one for Low- and Intermediate-Level Waste (L/ILW disposal area), co-located at the same site in a combined repository [1].

In the combined repository, as illustrated in Figure 2.6, the emplacement drifts for SF and HLW would be spatially separated from the emplacement caverns for L/ILW. The focus of the description is on the HLW repository.

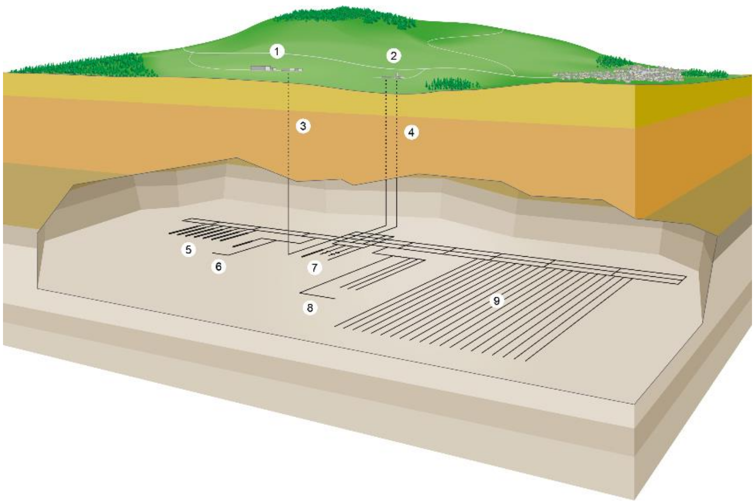


Figure 2.6: Illustration of a combined repository for SF/HLW and for L/ILW [1] Legend: 1) Surface facility. 2) Auxiliary access facility. 3) Access shaft (main access). 4) Operations and ventilation shafts (auxiliary accesses). 5) Main repository L/ILW. 6) Pilot repository L/ILW. 7) Underground geological investigations/test areas. 8) Pilot repository HLW. 9) Main repository HLW.

The underground part of the combined repository will be constructed in Opalinus Clay at a depth of a few hundred meters below the surface. It will include a series of dead-end emplacement drifts for SF and HLW. At the surface, spent fuel assemblies and fabrication flasks with HLW will be loaded into disposal canisters, which, according to the current Nagra repository project, are made of carbon steel. The disposal canisters will be transported underground and emplaced in the drifts, co-axially with respect to the drift direction, on pedestals of compacted bentonite blocks. Immediately after emplacement, the respective drift section will be backfilled with highly-compacted granular bentonite material.

In Nagra's safety concept, the natural barrier is considered to be of primary importance, due mainly to the excellent qualities and long-term stability of the host rock and its confining geological units of similar high quality (the containment-providing rock zone, or CRZ). The natural barrier is complemented by a mutually compatible set of engineered barriers. A key role of the engineered barriers is the minimisation and mitigation of disturbances to the CRZ inevitably caused by the waste and its emplacement, including, for example, the effects of heat generated by the waste and the disturbance to the rock caused by the excavation and ventilation of underground openings. The engineered barriers around the waste are expected to provide complete containment of radionuclides for a certain period. Although the eventual release of radionuclides and their migration through the multibarrier system cannot be excluded, most radionuclides will remain immobilised in the waste forms and waste containers until they have decayed. Furthermore, those that are released will migrate only slowly towards the surface environment, with substantial attenuation by radioactive decay first within the engineered barriers and then, for any that penetrate these barriers, within the surrounding natural barrier. A schematic overview of the post-closure disposal concept is given in Figure 2.7.

The components and environmental conditions for the current repository concepts that are judged to be most critical to providing the safety functions are termed "pillars of safety". The current pillars of safety, which will be reviewed and, if necessary, updated for the general licence applications, are:

- The deep underground location of the repository.
- The host rock and its confining geological units of similar high quality (the containment-providing rock zone).
- The backfill and seals of the disposal areas and repository access structures.
- The bentonite buffer (for SF and HLW).
- SF and HLW waste forms.
- SF and HLW disposal canisters.

For the L/ILW disposal area, the high-pH environment in the cementitious near field is also a pillar of safety.

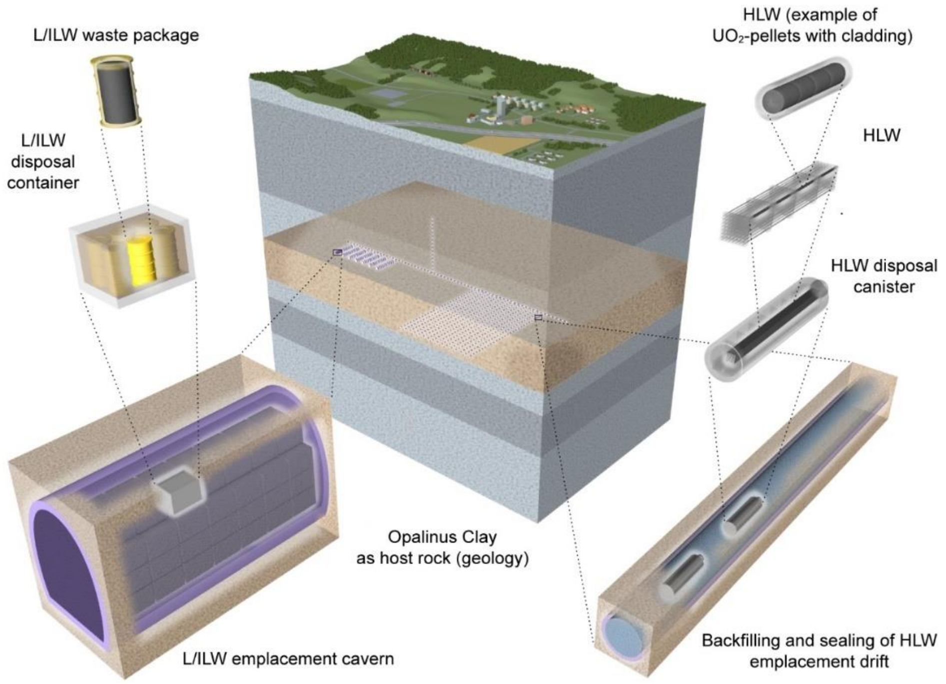


Figure 2.7: Post-closure disposal concept for the HLW and L/ILW repository according to the current disposal project (combined repository): For HLW, as an example, SF encapsulated in a carbon steel canister with a bentonite buffer is shown (right), while for the L/ILW, as an example, a 200 l drum in an LC container in an emplacement room backfilled with M1 mortar is shown (left).

2.4.2 Link between post-closure safety requirements and safety assessment

The process of safety assessment involves a systematic analysis of whether a given repository concept and design meet a set of pre-defined post-closure safety requirements. The post-closure safety requirements to be met by the repository concept and design are hierarchical in nature and are identified or developed in a top-down fashion, as illustrated on the left-hand side of Figure 2.8.

Having developed a repository concept and design that is judged to have good prospects of meeting the post-closure safety requirements, a safety assessment is carried out in a series of steps, which examine adherence to the requirements as well as any attendant uncertainty. The safety assessment steps proceed from the bottom up, as illustrated on the right-hand side of Figure 2.8, and include a performance assessment, which examines whether the requirements on expected performance are met, including adherence to a set of performance targets.

All safety assessment steps make use of the available scientific understanding and suitable assessment tools, collectively termed the assessment basis.

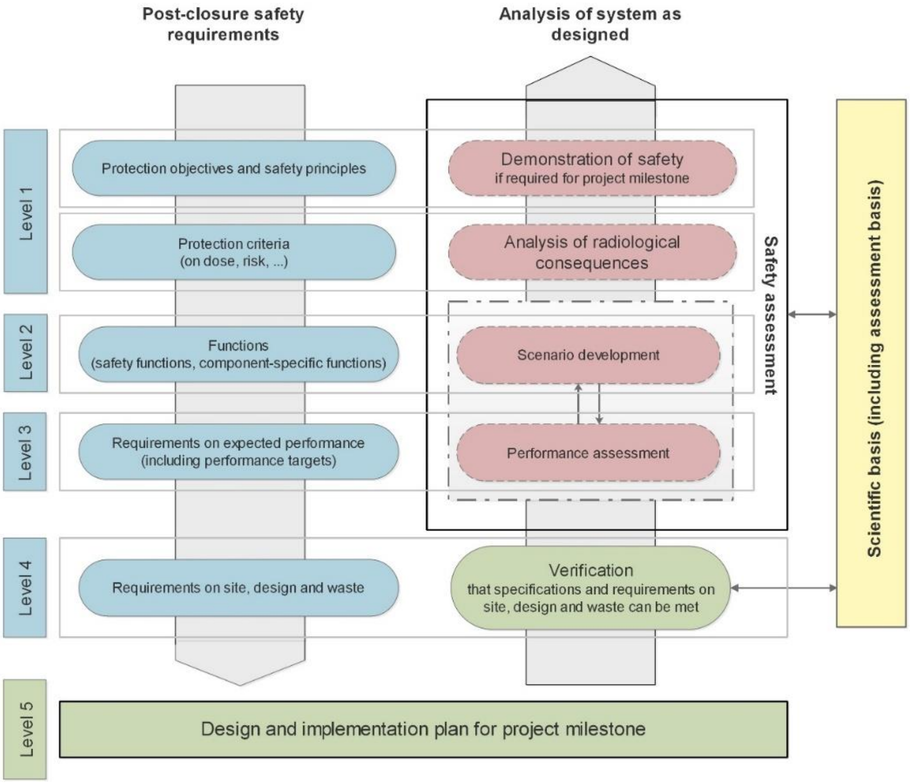


Figure 2.8: Overview of post-closure safety requirements and their assessment to develop input to the safety case.

2.4.3 Current approach to performance assessment of repository-induced effects: focus on thermal impact

The requirements on expected performance include, among others, the requirements on repository - induced effects, such as the requirement that thermal impacts should not compromise the performance of the disposal system and its key components (barriers). The assessment of whether detrimental thermal impacts could arise is carried out by comparing performance indicators evaluated using model analyses with pre-defined performance targets or evaluation scales [2].

For example, when considering the possibility of thermal alteration of the host rock, the performance indicator that is evaluated is temperature, and the performance target is that the calculated temperature should remain below the maximum paleotemperature experienced by the rock, i.e., the maximum temperature to which the rock has been subjected throughout its geological history (see Table 2.1). In the context of the safety assessment for SF, HLW and L/ILW in Opalinus Clay at the site Zürich Nordost (ZNO), the performance target derived in this way is that the maximum temperature in the host rock should be < 85 - 90°C [3]. If the temperature meets this performance target, significant thermally induced mineralogical alterations can be excluded.

Table 2.1: Performance indicator and target for the assessment of the possibility of thermal alteration of the host rock.

Issue	Definition and description
Relevance to safety functions	Precautionary criterion to avoid significant changes in the safety-relevant properties of the host rock [3]
Performance indicator	Temperature
Performance target	Maximum rock temperature < paleotemperature maximum (80 – 90 °C)
Domain of applicability	The performance target applies to the part of the host rock that acts as a barrier
Justification	Heating the host rock above its paleotemperature maximum engenders an inherent uncertainty regarding its chemical and mineralogical evolution [3]

A further thermal effect is that elevated temperatures will cause a transient increase of pore pressure within the host rock. The pore pressures attained will also potentially be influenced by the generation of gas in the repository. In the case of SF and HLW, little gas is generated while the thermal output of the canister is still high and, in the case of L/ILW, the thermal output of the waste is too low to greatly increase the pressure in the rock. The performance indicator is the thermally-induced increase of porewater pressure (Table 2.2) and the performance target is set to the lithostatic pressure at repository depth[2]. This performance target is used in the oil and gas industry in the assessment of borehole stability, as discussed in more detail in [4]. In the context of repository performance assessment, if the target can be shown to be met, the possibility that a rock fracture will be generated by pore pressures and propagate to a feature that could form a preferential release pathway (fractures, sedimentary architectural elements, faults / fault zones, and combinations of these features) can be excluded. As an additional, precautionary measure, the respective distance to such features will be set, based on site- and repository-specific considerations.

These performance indicators and targets are currently being revised in the context of the General Licence Application in Switzerland. The outcomes of HITEC will be taken into account for this revision. Further insights related to this are provided in Section 3.1.

Table 2.2: Performance indicator and target for the assessment of the possibility of damage to the rock due to thermally induced increase of porewater pressures.

Issue	Definition and description
Relevance to safety functions	Reactivation of existing, or creation of new, water-conducting pathways could lead to faster radionuclide transport through the rock, and hence less attenuation of radionuclide release by radioactive decay [5]
Performance indicator	Porewater pressure

[2] In this generic approach lithostatic pressure is used as a rough indicator for minimum stress at repository level. It is possible to constrain the magnitude of minimum stress when site specific information about the repository perimeter is available. The criterion will be revised in the context of the General Licence Application planned to be submitted in 2024.

Performance target	$p < \sigma$ i.e., the porewater pressure p is below the lithostatic pressure σ (The lithostatic pressure, and hence this criterion, are dependent on depth)
Domain of applicability	The criterion applies to the part of the host rock that acts as a barrier
Justification	Reactivation of existing water-conducting pathways, and the creation of new pathways, is avoided if shear stresses in the host rock remain below the yield limit. In the oil & gas industry (e.g., [6]), a positive effective stress is considered as a pragmatic indicator for the integrity of the host rock, when detailed knowledge of the local stress conditions at repository level is unavailable ("least principal stress approach")

2.1 NWS

2.1.1 NWS generic disposal concepts

The UK Government published its updated policy on geological disposal in December 2018, which marked the launch of the siting process in England. In Wales, the siting process was launched by the Welsh Government in January 2019. Both policies set out a consent-based approach of working in partnership with communities, where a geological disposal facility (GDF) will only be built where both a willing community and a suitable site exist. At the time of writing, NWS is working in partnership with several communities that may have potential to host a facility in their locality. However, no disposal site or geology has been selected, and therefore illustrative designs have been adopted for suitable generic rock types.

In the UK, wastes and nuclear materials destined for disposal in a UK GDF are grouped into “Low Heat Generating Wastes” (LHGW) and “High Heat Generating Wastes” (HHGW). LHGW encompasses long-lived low level waste, intermediate level waste and depleted natural and low enriched uranium (DNLEU). HHGW encompasses high level waste, spent nuclear fuel, highly enriched uranium (HEU) and plutonium. These waste groups are currently proposed to be disposed of within two separate areas within the GDF. Different disposal concepts for LHGW and HHGW have been developed which are based on the differences in the properties of the wastes. The disposal concepts utilise a multi-barrier system which incorporates a series of engineered barriers (wasteform, container, local and mass backfill, and accessway plugs and seals) and the natural barrier (the geological environment). These barriers work together to isolate and contain the waste so that it does not cause harm to life and the environment. The multi-barrier approach, including the barriers employed for LHGW and HHGW, is illustrated in Figure 2.9.

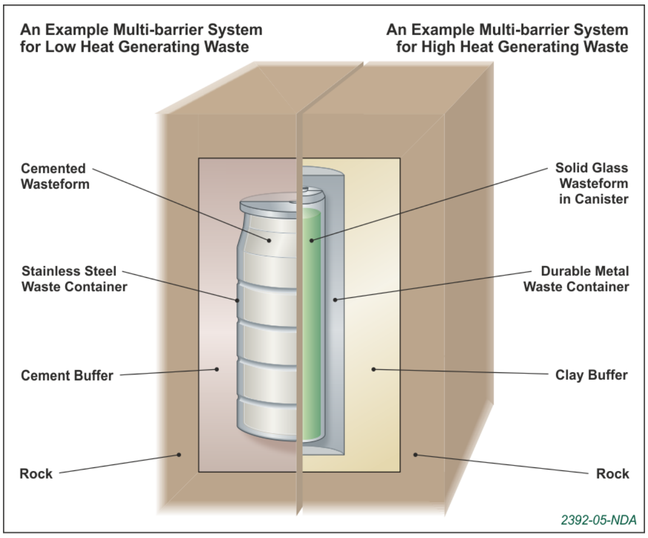
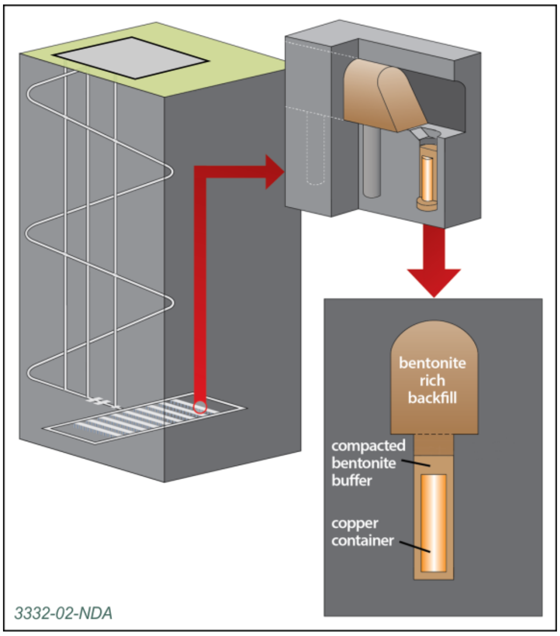


Figure 2.9: Illustration of the key engineered and natural barriers employed in the multi-barrier system, for LHGW (left) and HHGW (right)

For illustrative purposes before a GDF site is chosen, NWS has broadly adopted the NAGRA concept for a Lower Strength Sedimentary Rock (LSSR) type, and the KBS3V system developed by SKB for a Higher Strength Rock (HSR) type, for the disposal of high heat generating waste. Both disposal concepts assume the use of Na montmorillonite bentonite such as Wyoming MX-80 for the buffer/local backfill surrounding the waste container. These concepts are illustrated in Figure 2.10. An evaporite concept is also being considered in the UK, but the illustrative concepts do not involve any clay-based barriers.

NWS acknowledges that the UK inventory differs from that in Sweden, Finland and Switzerland, and therefore tailored concepts will be developed for a UK disposal facility. Selection of final disposal concept will be driven by waste constraints, site specific characteristics (once these become available) as well as practical and economic considerations.



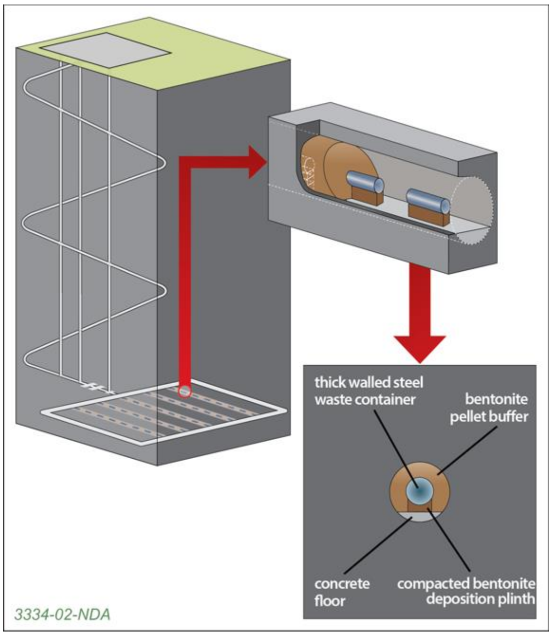


Figure 2.10: NWS illustrative designs for a Higher Strength Rock (based upon KBS3V, left) and Lower Strength Sedimentary Rock type (based upon NAGRA concept, right)

2.1.2 NWS generic disposal system safety case

In the absence of a site, the UK has developed a generic Disposal System Safety Case (gDSSC). The main purpose of the generic DSSC is to give confidence that the GDF can be implemented safely in the UK. It does this by describing and assessing the safety and environmental implications associated with all aspects of geological disposal of higher activity wastes. In addition, environmental and sustainability assessments consider the non-radiological socio-economic and health impacts of the GDF.

Development of a disposal system is iterative, as shown in Figure 2.11. The process starts with the identification of requirements from which a Disposal System Specification is developed. The requirements include external considerations such as regulatory and stakeholder requirements, as well as the nature, characteristics and quantities of the radioactive wastes requiring disposal.

A range of illustrative disposal concepts is selected for the implementation of geological disposal in a range of geological environments, typical of those found in the UK. From these, illustrative designs are produced to address the requirements. The designs are assessed for safety and environmental impacts and the outputs from these assessments inform subsequent development of the requirements and illustrative disposal concepts and identify knowledge gaps where further research and development is required. Note that ‘Safety Cases’ and other ‘Outputs’ in Figure 2.11 refer only to the finished documentation; the process of developing such documents is considered as part of the ‘Assessment’ box and feeds back into the iterative cycle. Assessment includes disposability assessments, site assessment and site suitability criteria.

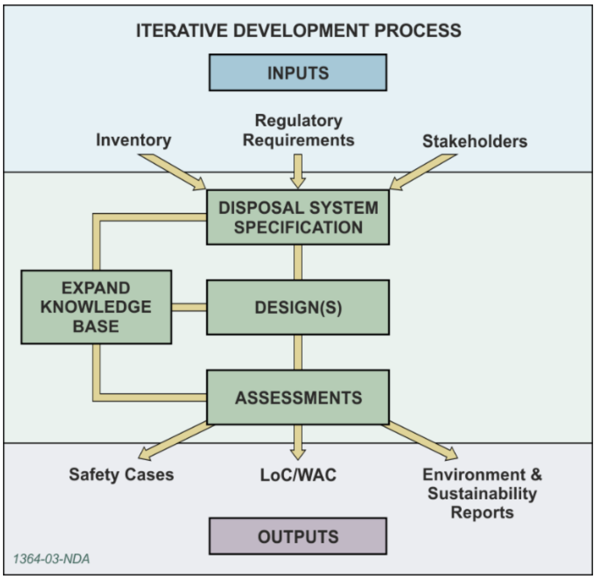


Figure 2.11: Workflow showing an iterative approach to developing a disposal system

Throughout this iterative process, NWS maintains an up to date knowledge base underpinned by needs-driven research and development such as that addressed through HITEC. Developing this knowledge base is a fundamental component of NWS’s business model, as it is the key means of meeting the needs of the disposal system design development.

The gDSSC contains a suite of documents organised into a hierarchical structure. At the top of the hierarchy is the Overview, which presents the main reasons why NWS is confident that the waste can be disposed of safely, and provides a summary of, and guide to, the document suite. The second tier comprises the safety cases themselves. This includes the generic ESC, which covers long-term safety following GDF closure. The assessments support these safety cases with more in-depth data and illustrative evaluations, and also address other environmental and sustainability considerations. In the fourth tier, the Disposal System Specification, design and knowledge base provide the basis for these assessments. The whole suite is underpinned by an extensive set of supporting references. A detailed breakdown of these documents for the generic Environmental Safety Case (ESC), which is of most relevance to HITEC, is shown in Figure 2.12.

The UK is progressing its GDF siting programme, working in partnership with communities to understand site suitability. NWS is therefore currently transitioning from a generic stage to a site-specific stage and as such is developing its site-specific safety case strategy, and approach to requirements to underpin the GDF design process. It will utilise an iterative approach, increasing in detail as further information and understanding on the potential sites become available.

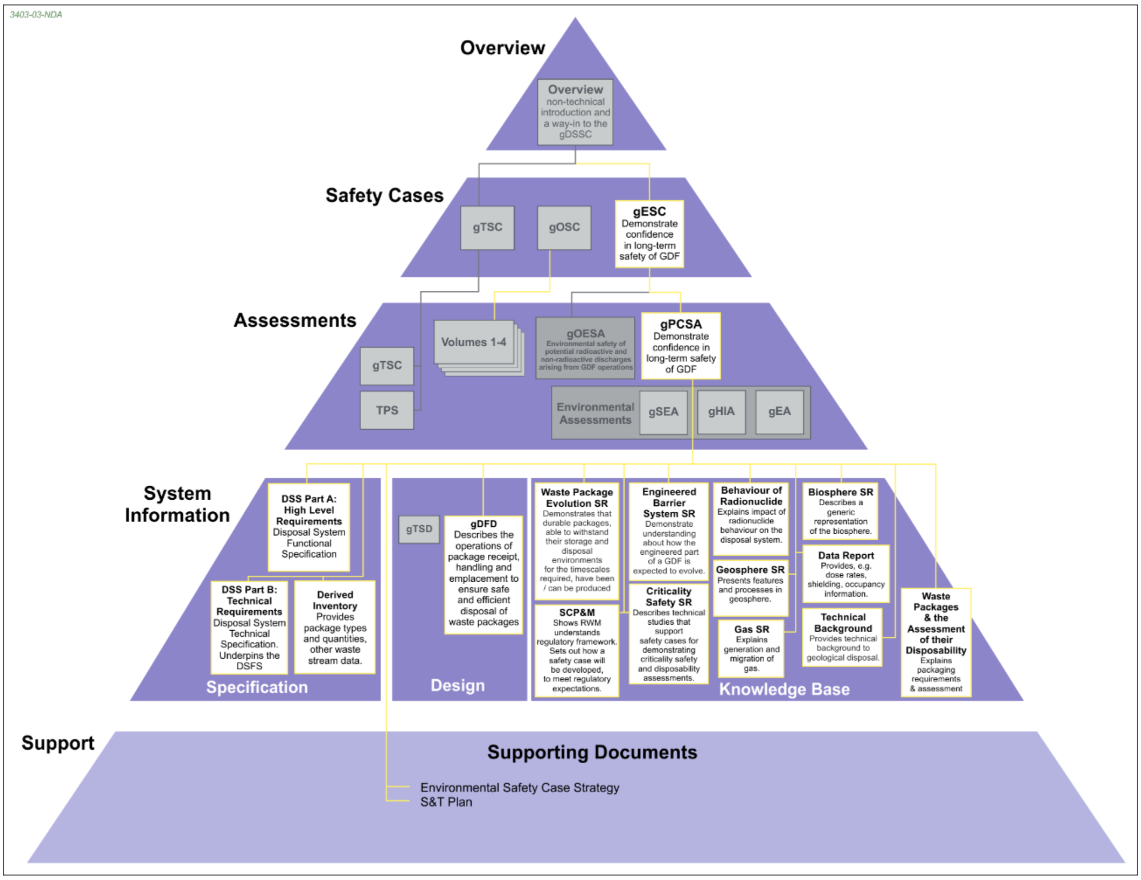


Figure 2.12: Map of relationships between ESC documents within the gDSSC

2.2 ONDRAF/NIRAS

2.2.1 Belgian generic geological disposal facility concept

Even if no site or host formation has yet been chosen in Belgium, the reference solution considered at time of writing is geological disposal in a poorly indurated clay layer at least 100 m thick, at a depth between 200 m and 600 m. This reference solution implies either the Boom Clay, or the Ypresian clays as reference host clayey formation.

The Belgian concept for long-term radioactive waste management currently being studied foresees to dispose of intermediate-level and/or, long-lived waste in disposal packages called “monoliths” and high-level waste in disposal packages called “supercontainers”; both are intended to be disposed of in a geological disposal facility (GDF).

Together, the waste, the Engineered Barrier System (EBS) and the host formation form a geological disposal system that will **contain** the radioactive waste for hundreds of thousands of years while the host formation and overlying layers will **isolate** the disposal facility from changes in the surface environment. The multi-barrier approach ensures that the disposal system is not dependent on any single barrier. Geological disposal systems are designed to be passively safe once closed, which means their safety does not depend on human maintenance.

In the current concept, the engineered barrier system is mostly made of metallic and cementitious components.

2.2.1.1 Supercontainer

In the supercontainer design, the primary waste packages of high-level waste are successively surrounded by a carbon steel overpack, a buffer made of concrete containing Portland Cement and a stainless steel envelope Figure 2.13**Erreur ! Source du renvoi introuvable.**, Figure 2.14). The length of the supercontainers varies from 4 metres up to 6.5 metres to accommodate the lengths of the different types of waste. The supercontainer is constructed at the surface before being transported underground for disposal.

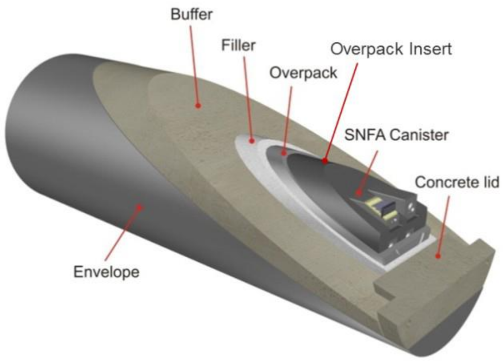


Figure 2.13: Schematic view of the supercontainer for the disposal of high-level vitrified waste [7].

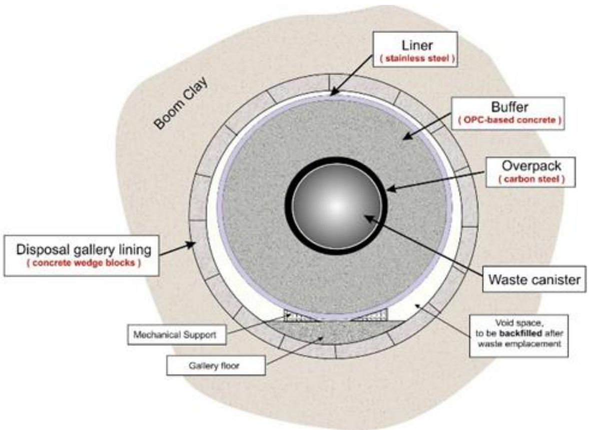


Figure 2.14: Supercontainer design in a geological repository, waste package, concrete lining and host clay formation [7].

2.2.1.2 Monolith B

The disposal packages for intermediate level and long-lived waste are called monoliths B. In the reference design, the primary waste packages of this type of waste are immobilised in mortar in concrete caissons made with Portland cement. Several monolith B designs exist to accommodate the large variety of primary waste packages (Figure 2.15). The outside diameter is always 2.8 metres, the length of the different monolith designs ranges between 1.9 metres and 2.9 metres, and their mass ranges from 32 to 39 tons [8]. The monolith B is constructed at the surface before being transported underground for disposal.

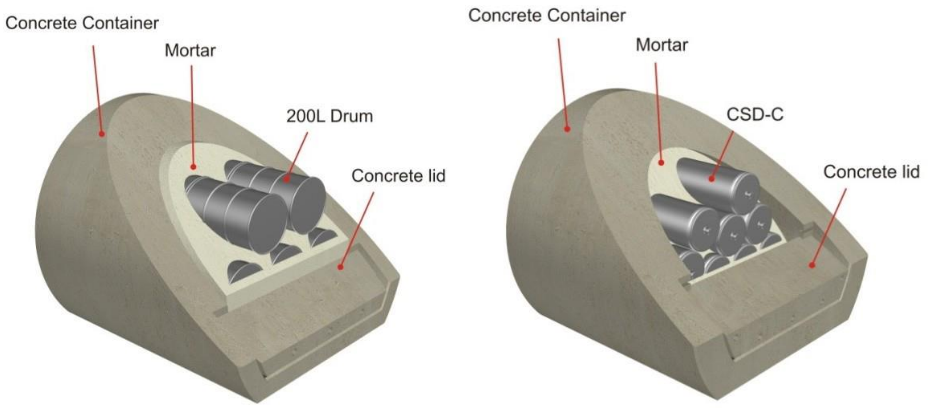


Figure 2.15: Schematic view of the monoliths B with two types of primary waste packages [8].

2.2.1.3 Layout of the geological disposal facility

The present architecture of a geological disposal facility comprises two access shafts and two main access galleries, lined with concrete. The disposal galleries are perpendicular to the access galleries. The length of the disposal galleries can be up to 1000 m. An artistic view of a potential geological disposal facility is seen in Figure 2.16.

In the disposal galleries, the spacing between each supercontainer in a gallery is 10 cm. The inter-axis distance between the disposal galleries in the zone for the disposal of high-level waste depends on the thermal output of the waste and is 50 m for the vitrified waste and 120 m for the spent fuel.

After disposal, all galleries are backfilled with a cementitious material.

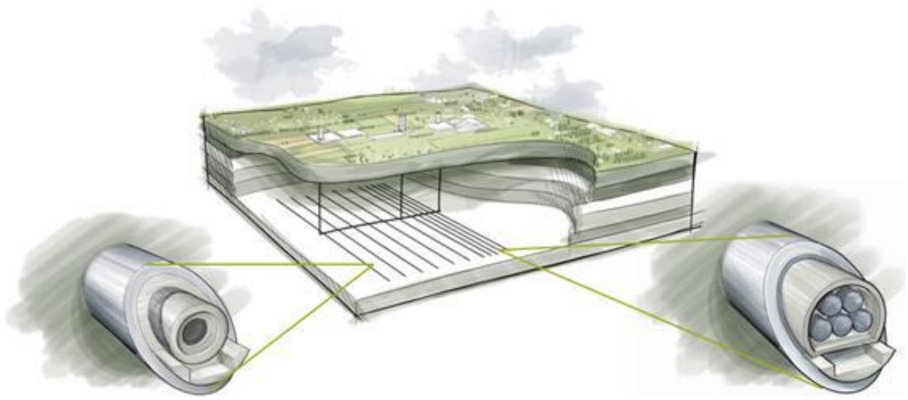


Figure 2.16: Artistic view of a geological disposal facility [8].

2.2.2 Safety functions

The disposal system has to provide three main safety functions [7] (Figure 2.17):

1. Full engineered containment (only for high-level waste), preventing any radionuclide release from the disposal packages during at least the thermal phase. The thermal phase is the time frame during which the temperature of the host formation is expected to lie above the range of temperatures within which nominal migration properties can be relied upon. The component contributing to this safety function are those that make up of the supercontainer. With this function, pore fluids are prevented from coming into contact with the waste matrix, and therefore, radionuclides are prevented from entering the poorly indurated clays when the waste is most active and temperatures are elevated by the decay of the most heat-emitting radionuclides.

2. Delay and attenuate the releases in order to retain the contaminants for as long as required within the disposal system and limit their release rates in the long term. The components contributing to this safety function are the waste forms, the engineered barrier system and the host formation. Three sub-functions are defined:
   - Limitation of contaminant releases from the waste forms: This function consists of limiting the release rates and spreading in time the releases of contaminants from the waste packages.
   - Limitation of the water flow through the disposal system: This function consists of limiting the flow of water through the disposal system as much as possible, thus preventing or limiting the advective transport to the environment of the contaminants released from the waste packages.
   - Retardation of contaminant migration: This function consists of retarding and spreading in time the migration to the environment of the contaminants released from the waste packages.
3. Isolation of the waste from man and the environment for as long as required, by preventing direct access to the waste and by protecting the repository from the potential detrimental processes and events occurring in its environment and on the surface. The host formation and its geological coverage provide this safety function. Two sub-functions are defined:
   - Reduction of the likelihood of inadvertent human intrusion and of its possible consequences: This function consists of limiting the likelihood of inadvertent human intrusion and, in case such intrusion does occur, of limiting its possible consequences in terms of radiological and chemical impact on humans and the environment.
   - Ensuring stable conditions for the disposed waste and the system components: This function consists of protecting the waste and the EBS from changes and perturbations occurring in the environment of the facility, such as climatic variations, erosion, uplifting, seismic events or relatively rapid changes in chemical and physical conditions.

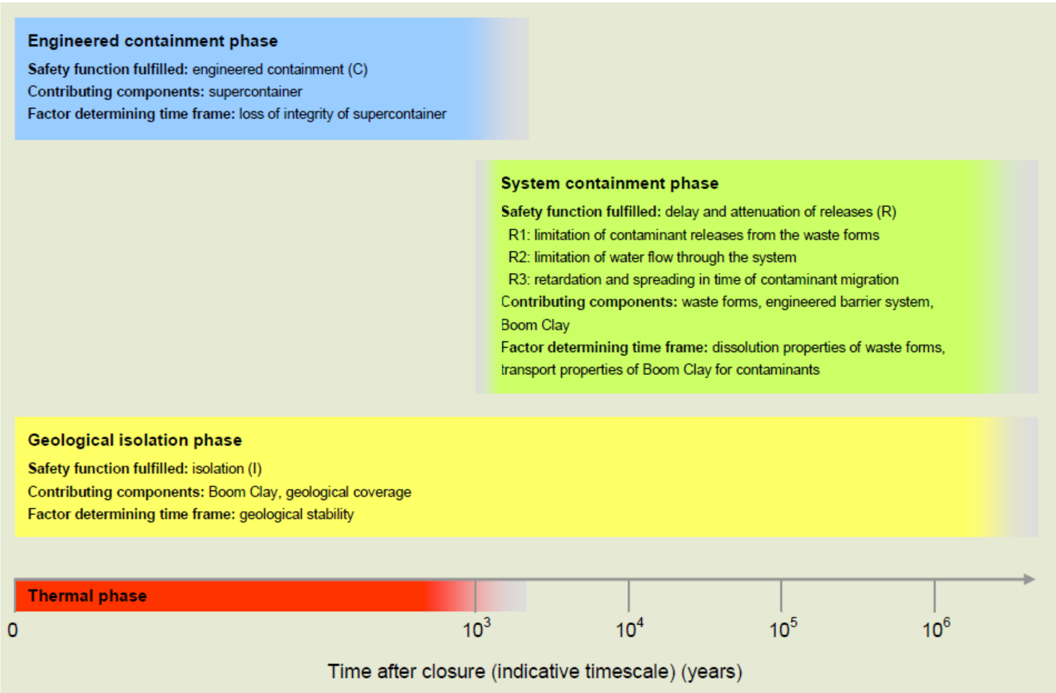


Figure 2.17: Safety functions provided by the main components of the disposal system in Boom Clay and its geological coverage and the time frames over which they are expected to be fulfilled. The engineered containment phase is specific to heat-generating waste (vitrified high-level waste and irradiated fuel) [7].

2.2.3 Requirements to constrain/limit impact of heat generation on poorly indurated clays

The requirements for the disposal system include that thermal impacts should not compromise the safety functions provided by the host rock. After repository closure, the heat produced within the geological disposal facility must be removed by passive means to avoid the occurrence of excessive temperatures or temperature gradients in the GDF or in its environment that could jeopardize the safety-relevant properties of poorly indurated clays, listed hereafter:

- low permeability. There is therefore practically no water movement in these clays. As a result, transport is essentially diffusive, which means species migrate under the influence of their concentration gradient, not under the influence of the pore water movement.
- strong retention capacity for many radionuclides and chemical contaminants (e.g. sorption capacity, favourable geochemical properties). Their migration through the clay is thus considerably delayed.
- capacity of self-sealing. Fractures such as those induced by excavation works seal within weeks in Boom Clay and Ypresian clays.

2.2.4 Current approach to include thermal impact in the safety case

The approach followed by ONDRAF/NIRAS to integrate the thermal impact in the safety case includes two steps:

1. Design a repository such that thermal constraints are met.

The geological disposal facility is designed in order to limit the increase of the temperature at the overpack, in the supercontainer, and at the interface between the clay formation and the upper aquifer. Several parameters may be modified to accommodate the heat output: the period of cooling of the waste prior to its disposal, the number of primary packages in the supercontainer (the number of vitrified waste canisters or the number of spent fuel assemblies), the spacing between supercontainers within a disposal gallery and the distance between the disposal galleries. The three first parameters essentially control the near-field temperature while the last one can be used in complement to control the far-field temperature. The temperature constraint at the overpack is 100°C while the temperature constraint at the top of the host clay formation is an increase of 10°C at most. The restriction at the overpack is driven by chemical reasons, in particular corrosion issues, while at the interface with the aquifer, the limitation is guided by protection of water resources.

The temperature evolution in the clay for a particular combination of the thermal design parameters can then been assessed for the disposal of the different types of waste. Figure 2.18 shows the results of a thermal calculation for the disposal of vitrified high-level waste after 80 years of cooling on surface, with 4.2 metre long supercontainers each containing two waste canisters and 50 m gallery spacing. The temperature increases quickly around the disposal gallery before rising slowly farther away into the clay massif, thanks of the heat dissipation. Results showed that the temperature increase at the interface Boom Clay – lining has a maximum value of 50°C, the peak of the temperature is reached within the first two decades. At just one metre distance from the gallery, the maximum temperature is reduced by about 7°C. After the peak, temperatures in the near field slowly decrease, following the decay of the source term. The change in slope during the cooling period shows the beginning of the influence of neighbouring galleries, which affects only the long-term cooling and has no consequence on the peak temperature.

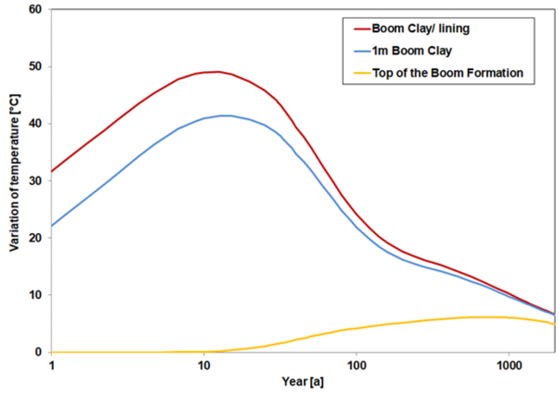


Figure 2.18: Calculated temperature evolution at the interface between the lining support and the clay, 1 m deep into the clay and at the interface between the Boom Formation and the upper aquifer. Note that the results shown are the temperature variations and the time axis is logarithmic.

2. Check whether the safety-relevant properties of Boom Clay are not damaged by the thermal transient.

Once the temperature evolution has been calculated, the consequences on all components of the system can be assessed. This is possible because while temperature may significantly affect the chemical and the mechanical evolutions of the system, these evolutions have in turn little effect on the temperatures. In general, the results from the multiple studies performed to assess the thermo-hydro-mechanical (THM) and chemical impacts of the thermal transient on Boom Clay indicate that the variations of mechanical and chemical conditions resulting from the temperature evolution will not jeopardize the safety-relevant properties of Boom Clay [9], [10], [11], [12], [13].

For instance, [10] have assessed the thermo-hydro-mechanical (THM) impact of the thermal transient of a disposal gallery containing different types of high-level waste on Boom Clay (repository at a depth of 400 meters). The most important THM impact is expected on the part of the host rock near the EBS, or excavation damaged zone (EDZ) that has already been perturbed during repository construction. At the interface Boom Clay – concrete lining, the pore water pressure increases (Figure 2.19, left) because the thermal expansion coefficient of the liquid phase is significantly larger than the one from the solid phase. After waste emplacement (here after 5 years), the pore water pressure rises quickly to a value close the undisturbed value of the pore pressure which is about 4 MPa in this case (repository at 400 m depth). The temperature at the wall of the gallery increases to value around 60°C within the first 20 years and then slowly decreases. The analysis of the variation of the state of mechanical stress in Figure 2.19 (right) shows that the excavation process generates plastic strain which is consistent with the EDZ observed during the excavation of the connecting gallery in the HADES URF. The variation of temperature affects the mechanical state of stress by accumulating minor additional plastic strain around the gallery (if any). This additional irreversible strain may slightly modify the EDZ. However, it has also be shown in e.g. TIMODAZ that self-sealing of cracks can be enhanced by elevated temperatures [9]. Hence, it is not expected that the thermal transient will significantly degrade the hydro-mechanical clay properties beyond the damage created during the excavation.

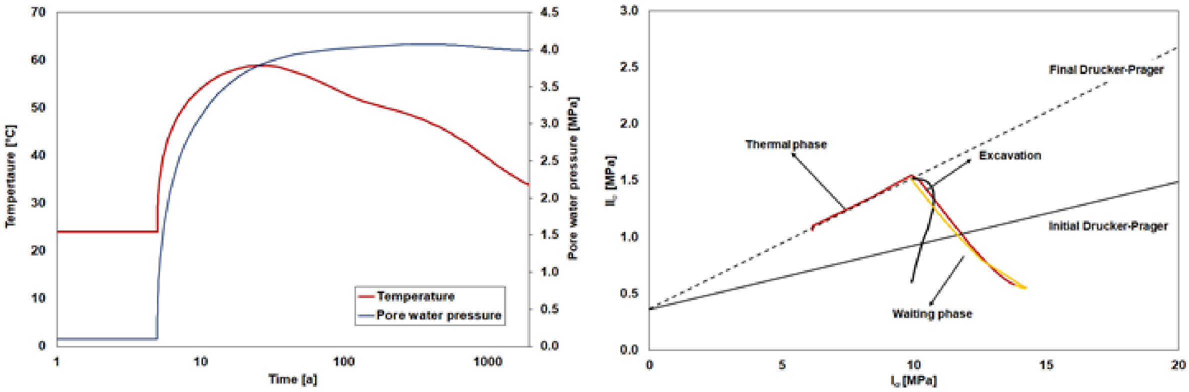


Figure 2.19: Pore water pressure and temperature evolution at the gallery wall obtained during a numerical prediction of geological disposal facility (left); Stress path in the plane of first invariant of the effective stress tensor and second invariant of the deviatoric effective stress tensor for appoint at the gallery's wall (right) [10].

2.3 POSIVA

In Finland, spent nuclear fuel from the nuclear power plants of Olkiluoto (owned by Teollisuuden Voima) and Loviisa (owned by Fortum) will be disposed in the Olkiluoto bedrock at a depth of 430 m metres. The repository is currently under construction and the first five deposition tunnels have been excavated (Figure 2.20). Posiva has submitted operational licence application for the use of ONKALO® repository for spent nuclear fuel at the end of 2021.

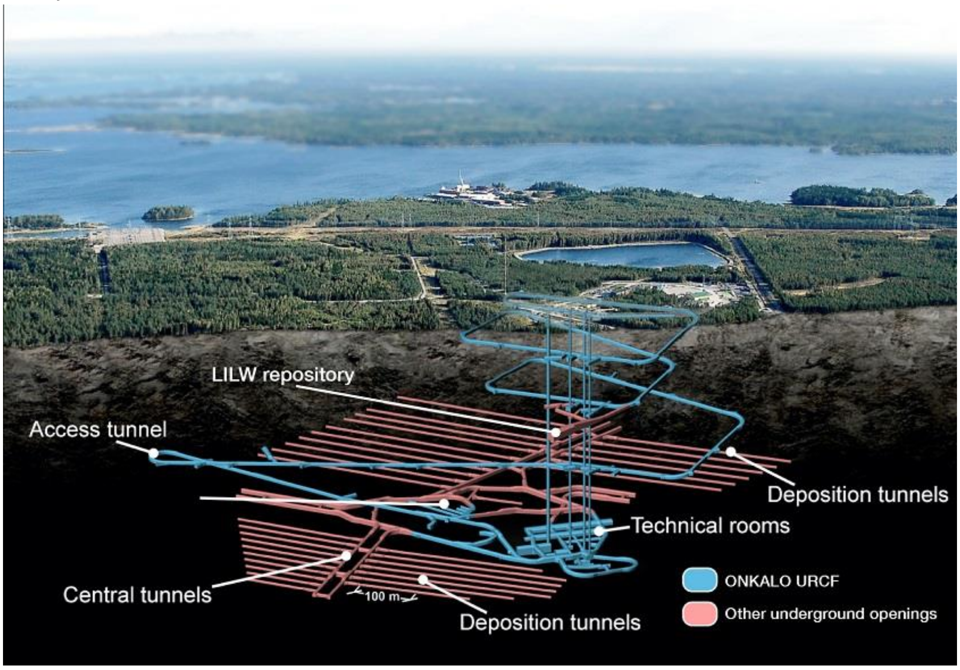


Figure 2.20: ONKALO® spent nuclear fuel repository in Eurajoki, Finland. Illustrative figure also showing the LILW repository, intended for waste from the operation and decommissioning of the encapsulation facility.

2.3.1 The safety concept and the multi-barrier system

Posiva's safety concept is illustrated in Figure 2.21. The concept is based on the high-level principles of long-term isolation and containment (primary safety pillars) and retention and retardation of radionuclides (secondary safety pillars):

- isolation, i.e. separation of the repository from the living environment, in order to minimise any interactions,
- containment, i.e. confinement of radionuclides within the canisters, and
- retention and retardation, i.e. limitation of radionuclide transport in case of canister failure.

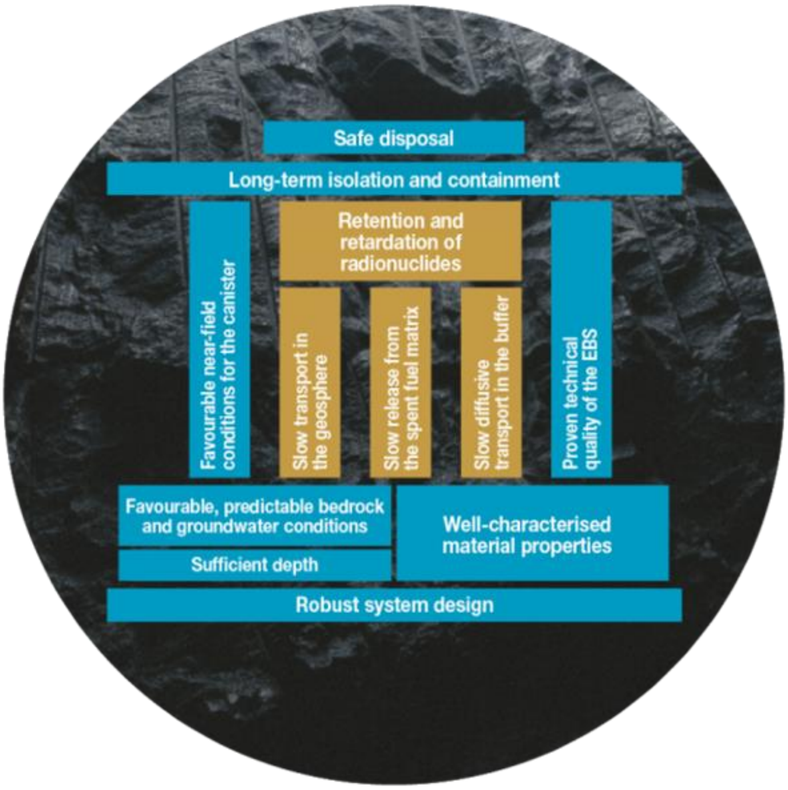


Figure 2.21: Outline of the safety concept for a KBS-3-type repository for spent nuclear fuel in crystalline bedrock ([19], Figure 3.3-1). Pillars and blocks shown in blue indicate the primary safety pillars and properties of the multi-barrier system. Brown pillars and blocks indicate the secondary safety pillars that become important in the event of radionuclide release from a canister.

According to the safety concept, safety depends first and foremost on the long-term containment of radionuclides within the copper-iron canisters and their long-term isolation in the deep bedrock. Long-term containment within the canisters, in turn, depends primarily on the proven technical quality of the EBS and favourable near-field conditions for the canisters. The technical quality of the EBS is favoured using components with well-characterised material properties and by the development of appropriate acceptance specifications and design criteria. For example, a clay buffer protects the canisters from rock movements and potential detrimental substances, limits groundwater flow around the canisters and limits and retards radionuclide releases in the event of canister failure. Favourable and predictable bedrock and groundwater conditions are requirements for the natural barrier, i.e. the host rock. Favourable conditions are defined with respect to either a natural site property (e.g. geological stability) or a given engineered barrier (e.g. the canister) or the spent nuclear fuel (e.g. anoxic conditions in the near field). The bedrock and groundwater conditions are predictable, i.e. the present conditions are characterised and understood as their long-term evolution can be modelled. Sufficient depth is the minimum depth required to isolate adequately the repository from the natural and human-induced processes near the surface.

Retention and retardation of radionuclides rely on a secondary set of safety pillars (shown in brown in Figure 2.21): slow release from the spent fuel matrix (and cladding), slow diffusive transport in the buffer, and slow transport in the geosphere.

Finally, the safety concept relies on a robust system design. A robust system design is a design that includes only components that behave in a well understood and predictable way. Furthermore, the performance of a robust system is relatively insensitive to possible imperfections in its implementation and to the unavoidable residual uncertainties in the understanding of its future evolution.

The safety concept is implemented via the KBS-3V design, developed in collaboration with its Swedish counterpart, SKB (Figure 2.22). The KBS-3V design is based on a multiple-barriers principle where radioactive substances are contained within several overlapping protective barriers so that no deficiency in one barrier and no predictable geological or other change will endanger the isolation. The KBS-3V design is suitable for the crystalline rock, such as the one at Olkiluoto. In the KBS-3V design, the multi-barrier system consists of engineered barriers and the natural barrier. The EBS consists of the disposal canister, the bentonite buffer, the backfilling of the tunnels and the natural barrier is the surrounding rock (Figure 2.22).

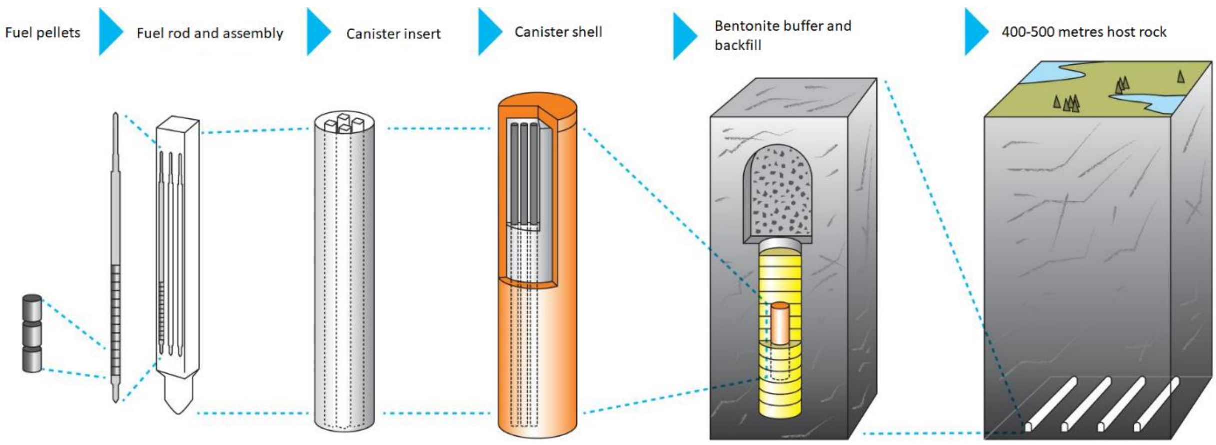


Figure 2.22: The spent nuclear fuel and the multiple barrier system in the KBS-3V design.

The spent nuclear fuel is not considered a barrier in Posiva's concept. Nonetheless, it has some favourable properties: the UO2 ceramic structure is stable and poorly water-soluble, which means that the release of radionuclides from the fuel matrix will be slow. Furthermore, the cladding has favourable mechanical and corrosion properties which will delay the contact between water and spent nuclear fuel in case of canister breach.

The inner structure of the spent fuel disposal canister comprises a spheroidal graphite cast iron component, which serves as the load-bearing component of the canister and as the fuel element placement rack. The inner structure is sealed with an iron lid. The inner structure of the canister is surrounded by a 5-cm thick copper shell and a copper lid welded to it. The sealed copper canister has been designed to act mainly as a corrosion barrier for the cast iron inner part.

Compacted bentonite is used as the buffer material, which is installed in the deposition hole to surround the canister. The bentonite is in form of compacted bentonite blocks (dry density 1650-1800 kg/m^3) and granules (dry density 1250-1350 kg/m^3) to fill gaps between canister and the compacted blocks as well as between the compacted blocks and host rock.

After installing the canister and buffer material, the deposition tunnels are filled with a granular bentonite mixture. A massive steel-reinforced concrete plug is constructed at the end of the deposition tunnel filled with bentonite clay. When the operations of the disposal facility end, the open central tunnels, access connections to ground level and other spaces will be closed with mixture of bentonite and crushed rock. At a depth of 430 m, the crystalline bedrock isolates the spent fuel repository from the surface environment and the biosphere.

2.3.2 The safety case methodology

The safety case methodology and the structure of the safety case report portfolio are depicted in Figure 2.23 [19].

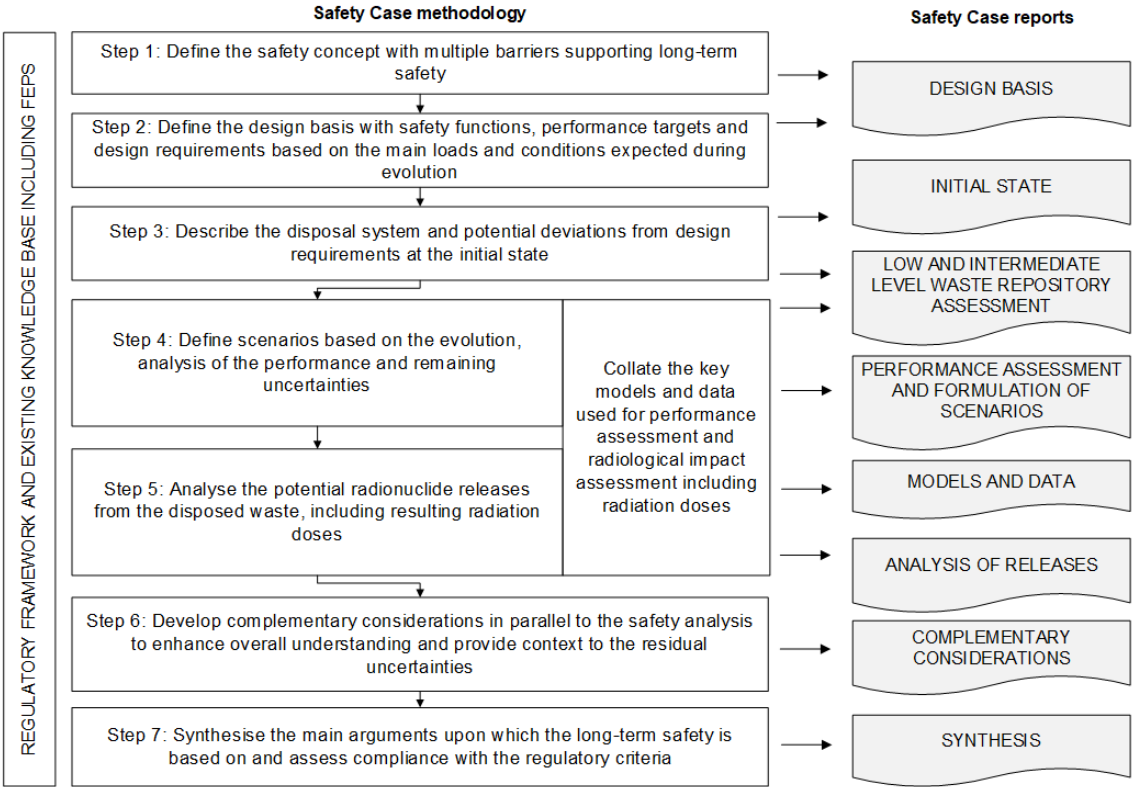


Figure 2.23: Safety case methodology and the connections to the main reports of the portfolio [19].

The methodology is based on the national regulatory guidance on the safety case contents, on the experience gathered from previous safety cases as well as on international guidance from the Nuclear Energy Agency [20] and the International Atomic Energy Agency [21].

In short, the methodology consists of describing the safety concept (see Figure 2.21) and safety functions of the various components of the system which play an important role in providing long-term safety. The design basis as well as the performance targets and design requirements (see the section below) is presented in a dedicated report [19].

The design solution fulfilling the requirements in the design bases is described in the Initial State report [22]. The report describes the conditions in the repository when direct control of the barriers is no longer possible. This point marks the start of the evolution of the disposal system.

Possible evolution paths for the repository are identified (formulation of scenarios) based on key epistemic uncertainties and the impact of such uncertainties on the EBS performance is analysed. These are described in the Performance Assessment and Formulation of Scenarios report [23]. Depending on whether or not the performance targets of various barriers are met, the containment function of the canister may be lost and radionuclide release and transport calculations are carried out to assess the radiological consequences of such scenarios. The radiological consequences of possible canister failure scenarios are presented in a dedicated report [24]. In addition to the modelling part of the safety case (usually referred to as ”safety assessment”), the safety case includes complementary considerations to enhance the overall understanding of the evolution of the disposal ssystem and to provide a more familiar context to frame the magnitude of residual uncertainties. This is described in the Complementary Considerations report [25]. The safety case portfolio also includes a safety assessment of the low and intermediate level repository for the waste streams issued from the operation and decommissioning of the encapsulation plant [26] and the ”Models and Data” report [27] which collates the main models, input data used in the safety assessment and their interfaces. Finally, the main arguments in support of long-term safety are compiled in the Synthesis report [28].

2.3.3 Link between post-closure safety requirements and safety case

The Design Basis (DB) report [19] presents the design basis for the geological disposal of SNF and of LILW arising from the operation and decommissioning of the encapsulation plant for that fuel. The report provides the long-term safety related requirements, their bases, justification and their hierarchical relationships. In Posiva’s requirements management system (VAHA), requirements are organised in a hierarchic system of five levels:

Level 1 consists of the stakeholder requirements. These are the requirements arising from laws, regulatory requirements, decisions-in-principle and from other stakeholders, such as Posiva’s owners.

Level 2 consists of the system requirements as defined by Posiva on the basis of requirements listed at Level 1. Level 2 requirements define the EBS components and the safety functions of the EBS and host rock.

Level 3 consists of the subsystem requirements, which are mainly specific requirements for the individual barriers. Level 3 includes the performance targets for the EBS and the host rock, applying to the long-term performance of the barriers. Fulfillment of performance targets is often possible only by modelling.

Level 4 presents the design requirements, which further clarify and provide more details of the requirements specified at Level 3, with the focus on those properties of the barriers that can be verified during the operational phase.

Level 5 presents the design specifications. These are the detailed specifications to be used in design, construction and manufacturing.

For the safety assessment, the most critical task consists of assessing the degree of fulfilment of the performance targets (Level 3) and in the assessment of impact of residual uncertainties on potential radionuclide releases and their radiological consequences. The performance targets are based on the features, events and processes (FEPs) governing the design of the barriers and of the repository.

In the context of the HITEC project, key processes concerning the clay buffer are:

- the radiogenic decay of radionuclides contained in the spent nuclear fuel, which impose a thermal load on the buffer
- heat transport (across the fuel, canister, buffer/backfill and host rock)
- water uptake and swelling of the clay components
- mineralogical alteration of the swelling clay (due to the initial thermal pulse from the spent nuclear fuel)
- advection and diffusion in the clay components

Relevant features of the system are the distribution of the SNF in the canister (thermal source term) and the thermal conductivities of the canister, the buffer and the host rock. Taking into account such FEPs, the design basis loads and conditions for the repository evolution are defined in order to define the performance targets on the various release barriers. For the buffer, the key performance targets related to the HITEC project are set on having a low hydraulic conductivity to ensure diffusive mass transport (these are based on having an adequate swelling pressure and self-sealing capability to reach the initial swelling pressure and maintain a diffusive regime even in cases of fracturing or piping erosion), as well as low stiffness to protect the canister from mechanical loads, such as those coming from a rock shear movement in the deposition hole.

A performance target (level 3) of maximum 100°C temperature for buffer has been set to avoid thermal alteration of the bentonite and consequent uncertainties in the fulfilment of its performance targets. This target is taken into account in the spent fuel canister loading plans and in the thermal dimensioning of the repository where the distance between disposal canisters and disposal tunnels is set and which is basis for the repository layout.

After the performance target is defined (i.e. maintaining a < 100°C temperature in the buffer), Level 4 design requirements give effective thermal conductivity for buffer as whole and Level 5 design specifications give thermal conductivity of individual buffer components.

2.3.4 Approach to thermal dimensioning and related performance assessment in the safety case

The modelling of the water saturation of the repository near field coupled with the decay heat evolution is an important modelling exercise for the performance assessment in order to determine the conditions in the buffer and the range of duration of the thermal pulse (depending on the evolution of the decay heat and of the thermal conductivities of the various components of the repository system).

Posiva's performance assessment [23] describes the thermal evolution of the buffer. Posiva's background report "*Buffer, Backfill and Closure Evolution*" [29] has both statistical analysis of likely temperatures reached in buffer and bounding high-temperature case. In the bounding case, the bentonite temperature may exceed 100 °C of a few degrees. In this case, the overall performance of the buffer is still largely intact because the peak temperature only takes place near the canister and large volumes of buffer still stay below 100 °C. Combined loads could potentially affect the buffer safety functions but, for example, a combination of events that cause erosion of buffer bentonite are not likely combined to high-temperature event, because high-temperature requires a dry deposition hole. On the other hand, loss of water (or vapour) in the vicinity of the canister leads to desiccation of the buffer and a decrease of thermal conductivity. Thus far, it is assumed that such process is reversible, and the buffer will recover the original properties.

2.4 SKB

2.4.1 Current repository project

Several decades of research and development has led SKB to put forward the KBS-3 method for the final stage of spent nuclear fuel management. In this method, copper canisters with a cast iron insert containing spent nuclear fuel are surrounded by bentonite clay and deposited at approximately 500 m depth in groundwater saturated, granitic rock, see Figure 1 The purpose of the KBS-3 repository is to isolate the nuclear waste from man and the environment for very long times. Around 12 000 tonnes of spent nuclear fuel is forecasted to arise from the currently approved Swedish nuclear power programme (where the last of the 6 operating reactors is planned to end operation in 2045), corresponding to roughly 6 000 canisters in a KBS-3 repository.

The Forsmark site in the municipality of Östhammar has been selected based on findings emerging from several years of surface-based investigations of the conditions at depth at the site.

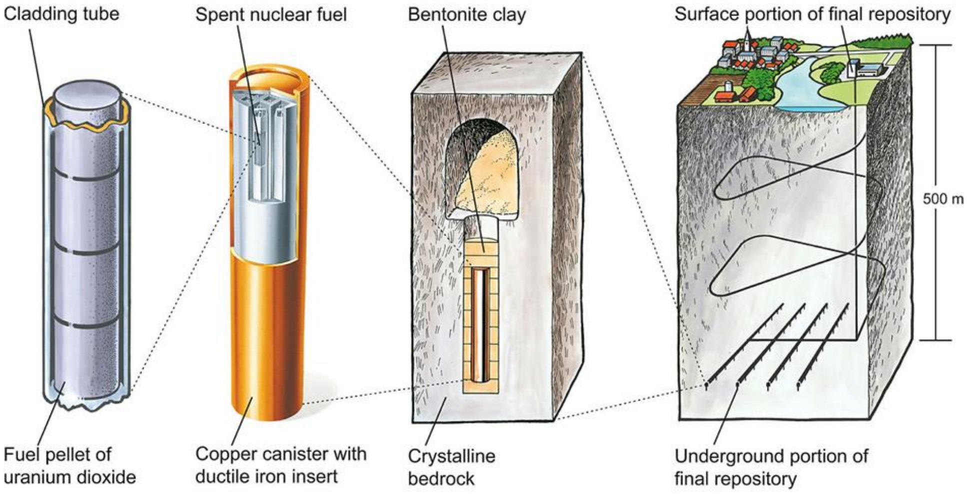


Figure 1 The KBS-3 concept for disposal of spent nuclear fuel.

2.4.2 Current safety case

An assessment of post-closure safety for the KBS-3 concept (SKB 2022) is a part of the Preliminary Safety Assessment Report (PSAR) for a spent fuel repository at the Forsmark site in the municipality of Östhammar. An approval of the PSAR by SSM is required for SKB to start construction of the repository. The main purposes of the PSAR post-closure safety assessment project were:

- To assess the safety, as defined in applicable Swedish regulations, of the proposed KBS-3 repository at Forsmark.
- To provide feedback to design development, to SKB's RD&D Programme, to detailed site investigations and to future safety assessment projects.

Society's requirements on post-closure safety of nuclear waste repositories are ultimately expressed in legal regulations. Two detailed regulations are issued by the Swedish Radiation Safety Authority (SSM) under the Nuclear Activities Act and the Radiation Protection Act.

Since work on the Swedish final repository project commenced at the end of the 1970s, SKB has established a number of principles for the design of a final repository. The principles can be said to constitute the safety philosophy behind the KBS-3 concept. They are summarised below.

- By placing the repository at depth in a long-term stable geological environment, the waste is isolated from the human and near-surface environment. This means that the repository is not strongly affected by either societal changes or the direct effects of long-term climate change at the ground surface.
- By locating the repository at a site where the host rock can be assumed to be of no economic interest to future generations, the risk of human intrusion is reduced.
- The spent fuel is surrounded by several engineered and natural safety barriers.
- The primary safety function of the barriers is to contain the fuel within a canister.
- Should containment be breached, the secondary safety function of the barriers is to retard a potential release from the repository.
- Engineered barriers shall be made of naturally occurring materials that are stable in the long term in the repository environment.
- The repository shall be designed and constructed so that temperatures that could have detrimental effects on the long-term properties of the barriers are avoided.
- The repository shall be designed and constructed so that radiation induced processes that could have detrimental effects on the long term behaviour of the engineered barriers or of the rock are avoided.
- The barriers should be passive, i.e. they should function without human intervention and without artificial supply of matter or energy.

Together with many other considerations, like the geological setting in Sweden and the requirement that the repository must be feasible to construct from a technical point of view, these principles have led to the development of the KBS-3 system for spent nuclear fuel. In practice, safety is achieved through the selection of a site with favourable properties for post-closure safety and through the design and construction of a repository that fulfils requirements related to post-closure safety. The site conditions today and the design and layout of the KBS-3 repository at Forsmark constitute the initial state of the safety assessment. These are also the aspects that are controlled by the implementer, through the choice of the site and through the design and site adaptation of the repository.

The Forsmark site is in the northern part of the county of Uppland within the municipality of Östhammar, about 120 km north of Stockholm. The Forsmark area consists of crystalline bedrock that belongs to the Fennoscandian Shield and formed 1.85 to 1.89 billion years ago. Tectonic lenses, in which the bedrock is less affected by ductile deformation, are enclosed in between ductile high-strain belts. The candidate area is in the north-westernmost part of one of these tectonic lenses. This lens extends from north-west of the Forsmark nuclear power plant south-eastwards to the area around Öregrund (Figure 2).

In summary, the main safety related features of the Forsmark site are:

- A low frequency of water conducting fractures at repository depth, which is beneficial for both bentonite erosion, corrosion processes and radionuclide retention.
- Favourable chemical conditions, in particular reducing conditions at repository depth, (which is generally found at depth in granitic rocks in Sweden) and salinity that would ensure stability of the bentonite clay buffer.
- The absence of potential for metallic and industrial mineral deposits within the candidate area at Forsmark.

In addition, the relatively high thermal conductivity at the site facilitates an efficient use of the rock volume and the rock mechanics and other properties of importance for a safe and efficient construction of the repository are also favourable.

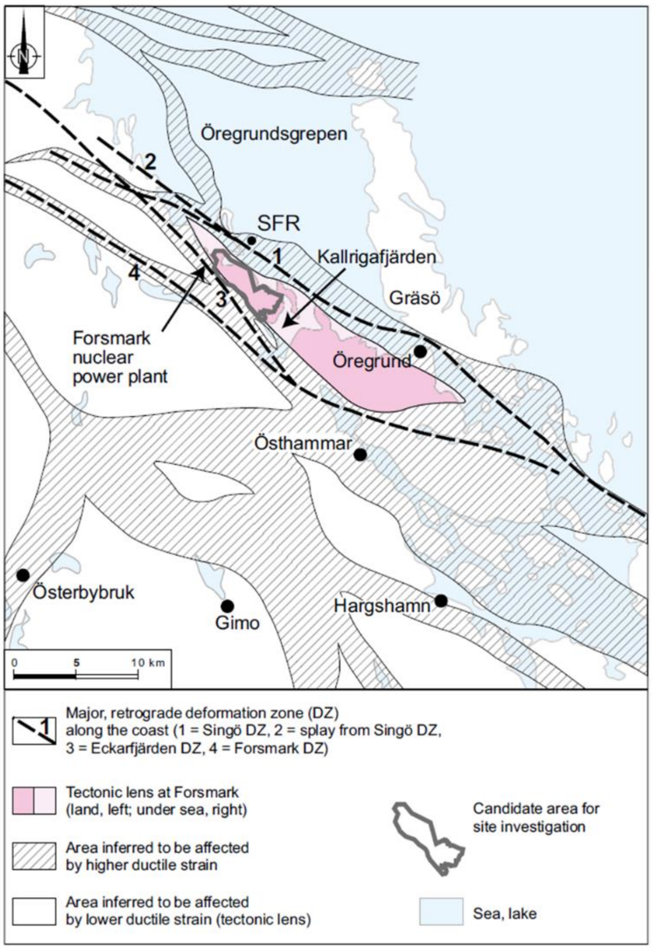


Figure 2 Tectonic lens at Forsmark and areas affected by strong ductile deformation in the area close to Forsmark.

A comprehensive description of the initial state of the repository system is one of the main bases for the safety assessment. The initial state in the PSAR is defined as the state at the time of deposition/ installation for the engineered barrier system and the natural, undisturbed state at the time of beginning of excavation of the repository for the geosphere and the biosphere. (Excavation induced impacts on the geosphere and the biosphere are analysed as part of the safety assessment.)

The KBS-3 repository concept has been continuously developed since it was first introduced. The current design is based on the design originally presented in the KBS-3 report in 1983. Feedback from assessments of post-closure safety is a key input to the refinement of the design. The PSAR technical design requirements have been established based primarily on feedback from safety assessment SR-Site, the Finnish safety assessment Turva-2012 and the technology development until 2016, as compiled in a joint Posiva SKB report from 2017. Design requirements typically concern specification on what mechanical loads the barriers must withstand, restrictions on the composition of barrier materials or acceptance criteria for the various underground excavations. A range of technical design requirements on the canister, buffer, deposition holes, deposition tunnels and backfill and on the main tunnels, transport tunnels, access tunnels, shafts, central area and closure are now established. The design requirements constitute design constraints, which, if all fulfilled, form a good basis for demonstrating repository safety. In summary, the following are among the most important safety related features of the initial state of the repository:

- The canisters' 5 cm copper shell providing a corrosion barrier.
- The canisters' ability to withstand isostatic loads, provided by the mechanical properties of the cast iron insert.
- The canisters' ability to withstand shear loads, also provided by the mechanical properties of
- the cast iron insert.

- The deposited buffer density, and the quality assured material composition of the buffer that ensures the development of the buffer into a diffusion barrier when water saturated.
- The deposited density and material composition of the deposition tunnel backfill.
- The general layout of the repository, with respect distances to fracture zones that can potentially host large earthquakes and with a distance between deposition holes that, together with the limitations on thermal output from the deposited canisters, ensure that the temperature of the repository is below 100 °C with a sufficient margin.
- Acceptance of deposition positions according to established criteria, which reduces the likelihood that deposition positions are intersected by large and/or highly water conducting fractures.

The repository system will evolve over time. Future states will depend on:

- the initial state,
- internal processes, i.e. a number of radiation related, thermal, hydraulic, mechanical, chemical and biological processes acting internally in the repository system over time, and
- external factors acting on the system.

Internal processes are e.g. the decay of radioactive material, leading to the release of heat and the subsequent warming of the fuel, the engineered barriers and the host rock. Groundwater movements and chemical processes affecting the engineered barriers and the composition of groundwater are other examples. External factors include effects of future climate and climate-related processes, such as glaciations and land uplift. The initial state, the internal processes and the external factors and the way they together determine repository evolution, can never be fully described, or understood. There are thus uncertainties of various types associated with all aspects of the repository evolution and hence with the evaluation of safety. A central theme in any safety assessment methodology must therefore be the management of all relevant types of uncertainty. This management amounts to identifying, classifying and describing uncertainties, as well as handling them in a consistent manner in the quantification of the repository evolution and of the radiological consequences to which it leads. A methodological approach also implies comparing the results of the assessment with regulatory criteria in such a way that appropriate allowance is made for the uncertainties associated with the assessment. The methodology for the analysis of post-closure safety applied in the PSAR consists of eleven main steps. Figure 3 is a graphical illustration of the steps.

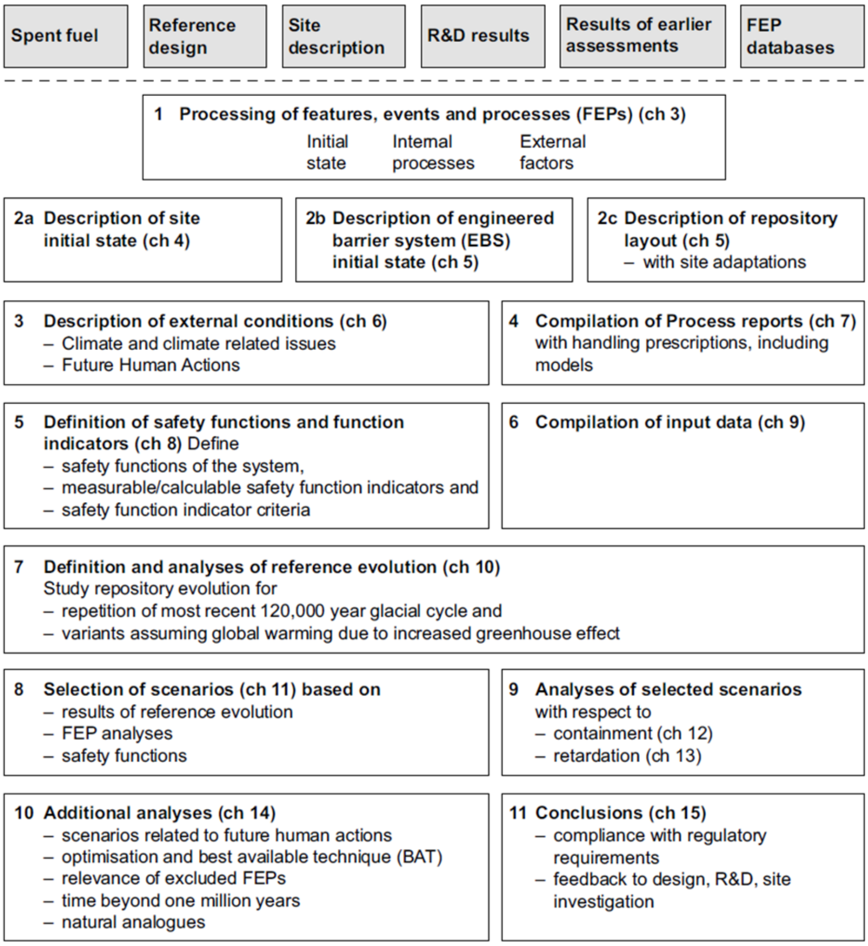


Figure 3 An outline of the eleven main steps of the PSAR safety assessment. The boxes at the top above the dashed line are inputs to the assessment. The chapters in the main report (SKB 2022) where the steps are further documented are also indicated

The central conclusion of the PSAR post-closure safety assessment is that a KBS-3 repository built at the Forsmark site according to the specifications in the PSAR can be expected to fulfil the requirements of post-closure safety expressed in SSM's relevant regulations. This conclusion is reached since the favourable properties of the Forsmark site ensure the required long-term durability of the barriers of the KBS-3 repository. In particular, the copper canisters with their cast iron insert have been demonstrated to provide a sufficient resistance to the mechanical and chemical loads to which they may be subjected in the repository environment. The detailed analyses, performed systematically according to a well-defined methodology, demonstrate that canister failures in a one-million-year perspective are rare. Even with a number of pessimistic assumptions regarding detrimental phenomena affecting the buffer and the canister, they would be sufficiently rare that their cautiously modelled radiological consequences are well below one percent of the natural background radiation, meaning that they are also well below the Swedish regulatory risk criterion.

2.4.3 Current approach to performance assessment of repository-induced effects: focus on thermal impact

2.4.3.1 Safety functions

The overall criterion for evaluating repository safety is the risk criterion issued by SSM, which states that "the annual risk of harmful effects after closure does not exceed 10−6 for a representative individual in the group exposed to the greatest risk". This is a "top level" criterion that requires input from numerous analyses on lower levels, and where the final risk calculation is the integrated result of various model evaluations using a large set of input data. A detailed and quantitative understanding and evaluation of repository safety requires a more elaborated description of how the main safety functions of containment and retardation are maintained by the components of the repository. Based on the understanding of the properties of the components and the long-term evolution of the system, a number of subordinate safety functions to containment and retardation can be identified. In this context, a safety function is defined qualitatively as a role through which a repository component contributes to safety. In order to quantitatively evaluate safety, it is desirable to relate the safety functions to measurable or calculable quantities, often in the form of barrier conditions. safety function indicator is thus a measurable or calculable quantity through which a safety function can be quantitatively evaluated. In order to determine whether a safety function is maintained or not, it is desirable to have quantitative criteria against which the safety function indicators can be evaluated over the time period covered by the safety assessment.

In the PSAR there are three safety functions regarding containment that involves the thermal impact. Two of those a related to the bentonite buffer and one is related to the geosphere.

Buff4. Resist transformations (requirement on temperature)

The buffer shall maintain its barrier functions and have long-term durability in the environment expected in the final repository. The buffer must resist transformation in order to maintain its safety functions in a long-term perspective. At elevated temperatures, chemical alterations of the swelling clay material acting to decrease the development of swelling pressure would occur. With respect to the temperature increase resulting from the disposal of the spent nuclear fuel, the buffer shall retain its favourable characteristics at temperatures up to 100 °C.

$T^{Buffer} < 100$ °C

An additional reason to requiring this maximum buffer temperature is that the extent of abiotic sulphate reduction increases with temperature and will become non-negligible at some temperature above100 °C, whereas it is negligible below 100 °C.

Buff6. Limit pressure on canister and rock

The key issue for the freezing process is when the pressure generated from the volume expansion of the ice can be harmful for the canister and the rock. The safety function indicator criterion for the canister is that it should withstand isostatic load ≤ 50 MPa. The mean crack initiation stress level for the main rock type in Forsmark is 116 MPa and the minimum is 60 MPa According the Clausius-Clapeyron equation 50 MPa is generated when ice is cooled by 3.7 °C below the freezing point of water. This pressure would be reached in a deposition hole at atmospheric pressure filled with pure water at −3.7 °C. The bentonite swelling pressure will lower the freezing point of water, thus lowering the temperature when the pressure from the ice will be 50 MPa. The temperature when ice starts to form can be predicted with an empirical approach. Using the lowest swelling pressure value for the technical design requirement of 3 MPa, the critical temperature is −2.5 °C. The most pessimistic value for when freezing causes a pressure above 50 MPa is therefore −2.5–3.7 °C = −6.2 °C. Based on this −6 °C is selected as the safety function indicator criterion for the safety function “Limit pressure on canister and rock – Buffer freezing”. In a situation with ice formation in the deposition hole, no hydrostatic pressure and no swelling pressure needs to be considered.

$T^{Buffer} > -6$ °C

R4. Provide favourable thermal conditions

The safety evaluation is simplified if water in the various components of the repository does not freeze. It is, however, not a global requirement that freezing does not occur. For example, freezing is part of the expected evolution for the groundwater down to typically 100–200 m depth for permafrost conditions, during which also the closure in the access shaft and ramp is expected to freeze. The rock, through its current background temperature at repository depth and its thermal conductivity, affects the peak temperature in the buffer. A low background temperature and a high thermal conductivity is favourable for keeping the buffer temperature below the required 100 °C.

Of these only Buff4 is relevant with respect to HiTEC. Freezing is another issue, and the thermal conductivity of the rock is a site-specific feature that will be constant with time.

2.4.3.2 Assessment of maximum temperature

The thermal evolution of the near field is of importance as general input information to the mechanical, chemical and hydrological processes. The direct safety relevant thermal criterion concerns the buffer peak temperature, safety function indicator Buff4 that requires that this temperature does not exceed 100 °C, chosen pessimistically in order to avoid, with a margin of safety, mineral transformations of the buffer. The thermal evolution of the repository depends on the thermal properties of the rock and the initial temperature at the site being considered, on the repository layout, i.e. canister spacing and tunnel spacing, and the canister power. For the thermal evolution in the interior of the deposition holes the properties of the bentonite buffer and of possible air-filled gaps are additional parameters. These properties depend strongly on the water supply, i.e. on the degree of saturation and may differ from one deposition hole to another depending on the local hydraulic conditions. The peak buffer temperature occurs some 5-15 years after deposition. At this time, approximately 50 % of the local temperature increase is caused by the heat from the canister itself and 50 % by the heat contribution from all the other canisters. This means that the local rock heat transport properties are particularly important to the peak temperature for the individual canisters and that, therefore, the low tail of the conductivity distribution, the spatial variability and the scale of variation are important for the dimensioning issue. An estimate of the distribution of peak buffer temperatures in both dry and wet deposition holes can be made by use of an analytical solution In dry deposition holes the maximum buffer temperature is found at the top of the canister where the bentonite is in direct contact with the copper surface, see Figure 4. Note that the hottest point on the canister surface is located at canister mid-height. In wet deposition holes, the air-filled gap between the canister and bentonite blocks will be closed at the time of the peak temperature, and the bentonite will also be in direct thermal contact with the copper shell at points on the vertical canister surface. In this case the maximum buffer temperature will coincide with the hottest point on the canister surface, i.e. at mid-height. Figure 5 shows the peak temperature distribution using the canister spacing in the layout. There are two cases: with and without the temperature correction above. Without the correction there are temperature over- and underestimates, for canisters associated with the low- and high conductivity parts of the distributions, respectively.

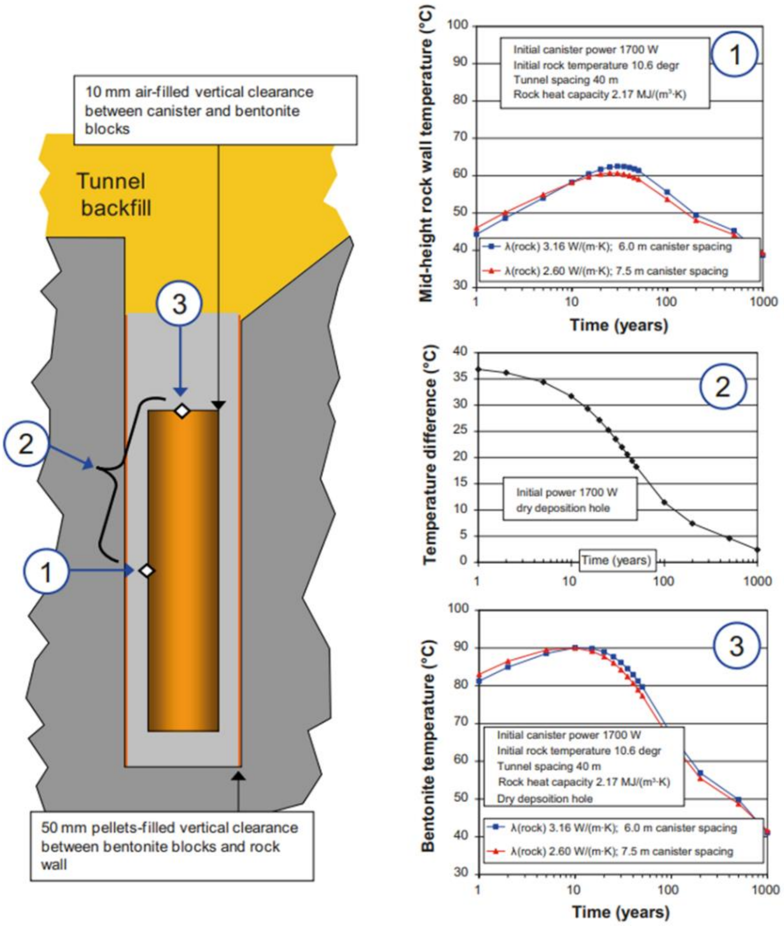


*Figure 4 Principle of peak buffer temperature calculation. The rock wall temperature at canister mid***[1]***height (1) is added to the temperature difference (2) between rock wall and the top of the canister to find the maximum bentonite temperature (3). The difference (2) is due to the heat resistance over buffer and gaps (local solution) whereas the rock wall temperature (1) depends on layout and rock thermal properties.*

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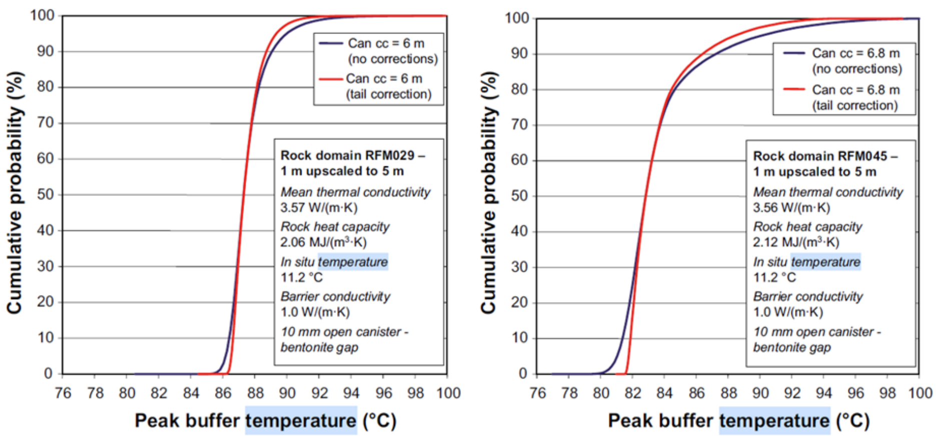

Figure 5 Distribution of buffer peak temperature in rock domains RFM029 (left) and RFM045 (right), with and without correction for spatial variability.

On average, less than one canister position, out of 6 000 canister positions, would have a peak buffer temperature larger than 95 °C meaning that the design requirement would be satisfied with a margin of

5 °C, based on this analysis. A very large majority of the canisters, about 98 %, will have a margin of 10 °C or more.

2.4.3.3 Mineral alteration

The advantageous physical properties of the buffer, e.g. swelling pressure and low hydraulic conductivity are determined by the ability for water uptake between the montmorillonite mineral layers (swelling) in the bentonite. However, montmorillonite can transform into other naturally occurring minerals of the same principal atomic structure but with less or no ability to swell in contact with groundwater. The transformation processes usually involve several basic mechanisms. At the expected physico-chemical conditions in a repository, the following possible mechanisms are identified:

1. congruent dissolution,
2. reduction/oxidation of iron in the mineral structure,
3. atomic substitutions in the mineral structure,
4. octahedral layer charge elimination by small cations,
5. replacement of charge compensating cations in the interlayer.

The montmorillonite transformation in a KBS-3 repository is assumed to be small based on the following observations and arguments:

1. The time scale for significant montmorillonite transformation at repository temperatures in natural sediments is orders of magnitude larger than the period of elevated temperature in a KBS-3 Repository.
2. The bentonite material is close to mineralogical equilibrium to start with.
3. Transformation is limited by transport restrictions.
4. All published kinetic models based both on natural analogues and laboratory experiments indicate that the transformation rate is very low at repository conditions.

Based on this reasoning two safety function indicator criteria have been defined. As long as the maximum temperature is below 100 °C and the pH of the water in the rock is below 11 the montmorillonite in the buffer is assumed to stable for the time scale for assessment of the repository (1 000 000 years). In the reference evolution, both the pH and the temperature in the buffer are assumed to be within the given limits and the alteration is not expected to proceed to a level where it will affect the properties of the buffer.

2.5 SURAO

2.5.1 Czech current repository project and safety concept

The Deep Geological Repository (DGR) concept is based on requirements for the disposal of Spent Nuclear Fuel (SNF) from the Czech Republic's two existing nuclear power plants - Dukovany (four units) and Temelín (two units), and three proposed new units. The disposal concept assumes the direct disposal of SNF (without reprocessing). A total of 7600 waste disposal packages is assumed and these waste disposal packages will be deposited either vertically or horizontally. The area for this part of the repository has been estimated for a depth of 500 m under the surface and represents roughly 3 km^2 with regard to site-specific rock thermal properties.

The technical design of the Czech DGR is shown in Figure 2.24. The facility consists of disposal areas for SNF - disposal wells, disposal areas for other radionuclide waste (e. g. ILW) - disposal chambers, a transfer node including a hotcell, loading corridors, transport and service areas, mining and ventilation shafts and an experimental and rock characterization workplace. The two disposal areas should not interact with each other over the time and will be placed at an adequate distance [14].

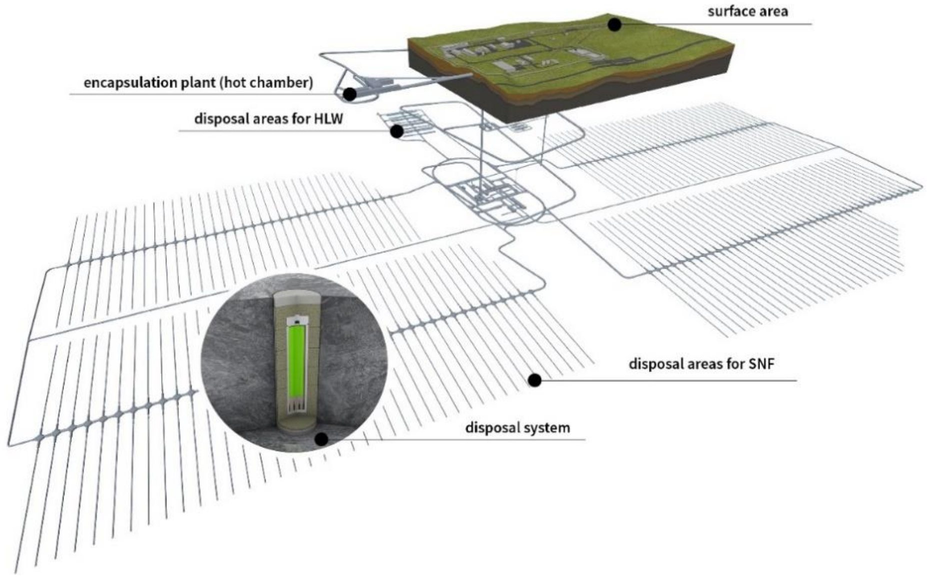

Figure 2.24: SÚRAO´s Deep Geological Repository concept

It is planned that the Czech DGR will be constructed in a crystalline rock environment at a depth of around 500 m below the surface. The safety of the disposal system will be ensured by a multi-barrier system consisting of a natural barrier (the rock mass) and a set of engineered barriers (Figure 2.25). The system has been designed in such a way that in the case of the accumulation of radioactive substances released from the DGR, the activity of these substances will be well below the dose constraint set for a representative person of 0.25 mSv per year (Act No. 263/2016 Coll., section 82). The achievement of this limit will be substantiated and verified via conservative safety calculations that consider a period of up to one million years.

As mentioned, it is assumed that the Czech DGR will be built in a crystalline rock environment. The isolation function of this rock type is affected by the presence of fractures and, therefore, engineered barrier system will play a key role. The waste disposal package (WDP) comprises the primary engineered barrier, which will be required to ensure the subcriticality of the disposed SNF and its integrity over 1 million years. Such a long WDP service life depends on ensuring both corrosion and mechanical resistance. SURAO selected a double-layer WDP as the reference disposal package with an outer casing consisting of several centimeters of carbon steel that provides a combination of shielding, a low and predictable corrosion rate and high mechanical strength and an inner casing consisting of several centimeters of stainless steel that ensures a very low corrosion rate and high mechanical strength [15].

A further important barrier comprises the buffer that fills the space around the WDP in the disposal well. Its main functions are:

- to protect the WDP, i.e. the retardation of the supply of groundwater and the development of corrosion and the "cushioning" of potential mechanical stress and
- to retard the potential migration of radionuclides.

This simply means that the buffer must have excellent sealing properties, which must be retained throughout its expected lifetime. Since the Czech Republic has relatively large deposits of bentonite, it was decided that bentonite of Czech origin will be used.

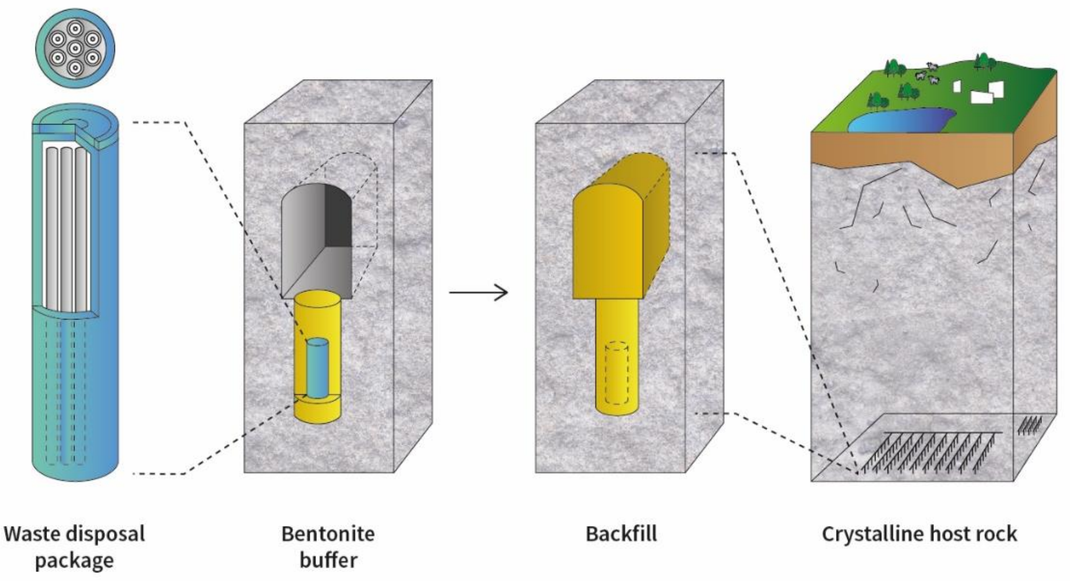

Figure 2.25: Diagram of the proposed Czech multi-barrier DGR system

2.5.2 Current approach to performance assessment of the repository-induced effects: focus on thermal impact

In the SURAO DGR concept is stated that heat from decay cannot negatively influence the functioning of the engineered barriers. Therefore, the temperature limit for the bentonite barrier was set at 95 °C [14] to prevent bentonite resp. montmorillonite alteration. To meet this requirement, the whole EBS system is designed to keep the temperature up to this limit.

The Czech concept is based on using Czech bentonite for buffer and backfill. As it is only one program using this type of bentonite lots of experimental data are needed to be able to accomplish the performance assessment of the EBS system. The experimental program is running for more than 20 years and is focused on the different treatments of bentonite [16]. Bentonite temperature-treated experiments (laboratory and in-situ) were in most cases up to 95 °C to test proposed DGR conditions, all the data up to now are summarized in [17]. Ongoing and planned laboratory programs [18] should fulfill the bentonite database which will serve as an input for mathematical models. A complex performance assessment of the Czech EBS system is planned to be completed in 2026.

3. Impact of HITEC project results on safety cases

3.1 Introduction

The WMO involved provide below the expected impact of the outcome of the WP7 results and understanding. The impact of temperature on the barrier performance is highly dependent on the source term, its evolution, the design of the barrier, the choice of material and ultimately the component specific safety functions that the material has to fulfil.

3.1 ANDRA

3.1.1 Contribution of EURAD HITEC to Andra's safety case

As mentioned above, Andra has gathered over nearly 30 years a vast knowledge on the properties of the Callovo-Oxfordian claystone and a good understanding of the effect of temperature on its behaviour. Several models have in addition been developed to study its THM behaviour, design the Cigéo disposal facility and test the scenarios defined in the safety assessment. The EURAD HITEC project gave an opportunity to improve Andra's understanding of the behaviour of the Callovo-Oxfordian at very high temperatures and test the robustness of the modelling approaches. The experimental part of the project studied how the exposure to high temperatures affects the mechanical and transport properties of the COx claystone. In the modelling task, after comparing how different codes model simple generic cases, both in near-field and in far-field scenarios, the simulation of some laboratory experiments and of the ALC1605 in-situ experiment helped improve the existing models.

3.1.1.1 Optimisation with regard to thermal transients

The triaxial compression tests performed by ULorraine in subtasks 2.1 and 2.2 (Gbewade et al., 2023) indicate a small decrease in the peak strength with increasing temperature.

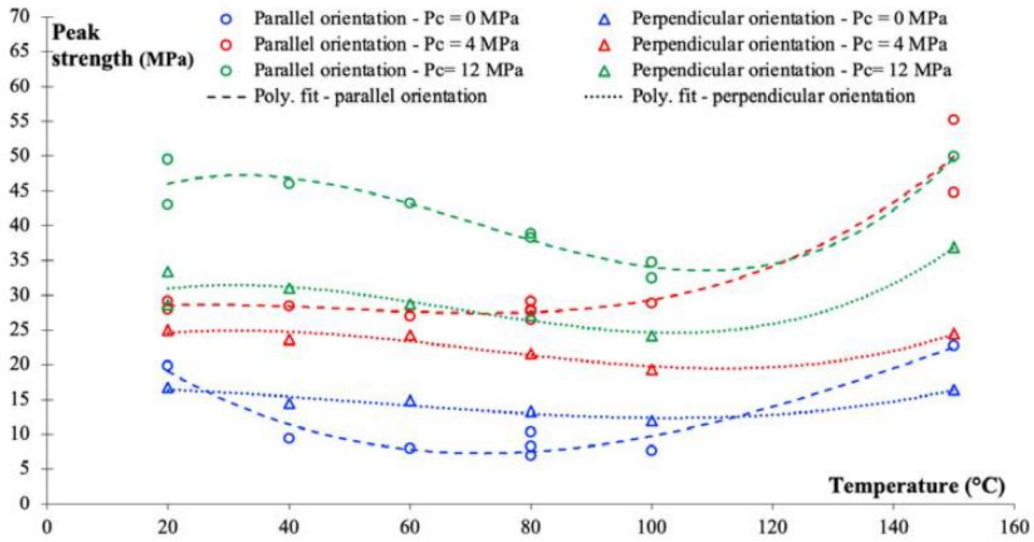

Figure 3-1 Evolution of the deviatoric stress at peak (peak strength) as a function of temperature for parallel and perpendicular orientations and for all confining pressures (0, 4 and 12 MPa), with polynomial (3rd order) fitting curves

This decrease in the peak strength is most significant for samples oriented parallel to bedding plane under uniaxial conditions for which volumetric dilatancy appears with the temperature increase. These samples may however have been damaged during the heating phase because of the high thermal loading rate applied on the samples (50°C/hr whereas in the repository conditions, the models indicate that it would take 3 weeks to reach 40°C at the borehole wall, and 12 years to reach 70°C) and of the thermal expansion of the pore water that may induce the formation of microcracks along the bedding. For all other tests, the decrease of the peak strength as the temperature increases is more moderate. The sample damage may then have been limited thanks to the application of the confining pressure when heating. In both orientations, the peak strength increases for the highest temperatures (100 and 150°C) as the pore water is vaporised and the samples get desaturated. Saturation appears to have a much greater effect on the strength than temperature. These compression tests did not show any clear effect of the temperature on the Young's modulus and Poisson's ratios.

BGS performed some thermal pressurisation tests on COx and Opalinus clay samples in subtask 2.2. The permeability was measured at various temperatures (up to 89°C). The tests performed on the Opalinus clay show a slight decreasing trend on the permeability as temperature increases, but the data indicates that the bulk permeability of the clay remains unchanged after heating (Figure 3-2).

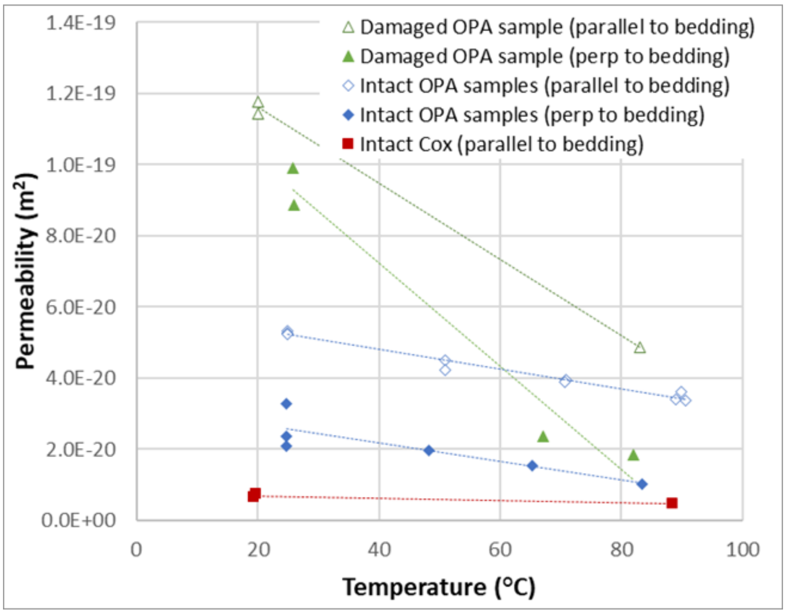

Figure 3-2 Pulse permeability measurements performed on Opalinus Clay and COx samples at various orientations and temperatures

As part of subtask 2.1, the influence of temperature on the self-sealing process was studied by ULorraine by performing self-sealing tests on artificially cracked samples at 20°C and 80 °C using the same protocol (Agboli et al., 2023). The results for samples with a crack parallel to bedding are compared on Figure 3-3.

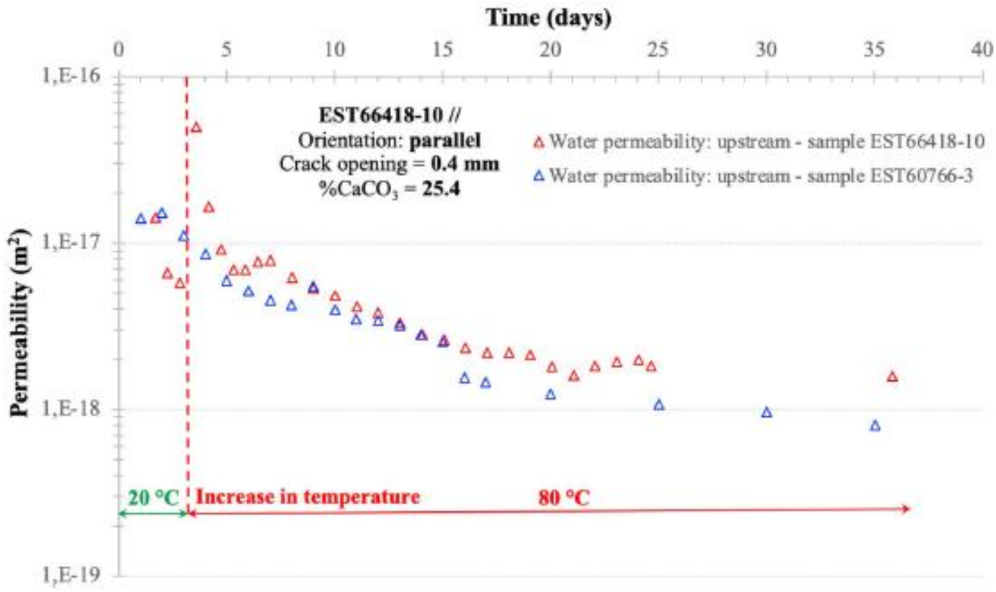

Figure 3-3 Evolution of water permeability of parallel sample EST66418-10 during the self-sealing test at 80 °C compared to the permeability evolution of sample EST60766-3 (performed at 20 °C)

The permeability of the intact COx claystone (in the order of 10^{-20} m^2) is not reached during the duration of these experiments (one month), but they confirm the ability of the COX claystone to close over time the fractures generated by the excavation around the galleries or HLW cells. Temperature seems to have a delaying effect on the self-sealing process, but the final permeability in the test at 80°C is similar to that reached at room temperature (Figure 3-3).

Overall, the results of these laboratory experiments are positive as they confirm that the Callovo-Oxfordian claystone keeps its good mechanical and retention properties, even when heated at high temperature (up to 100°C). Reference is made to the D7.3 (Final technical report on thermal effects on near field properties) and D7.5 (Final technical report on effect of temperature on far field properties) deliverables for more detailed information on these tests and their results.

The subtask 2.3 modelling subtask was divided in three steps. The benchmarking exercise on near-field generic cases showed that six different codes give matching results when considering an isotropic elastic behaviour, but some discrepancies were identified when anisotropy is introduced. The difference in THM formulations and assumptions made in the different codes may be the main reason for these disparities. The precise location of the integration point depending on the mesh and on the averaging method may also explain some discrepancies, especially at/near the contour of the tunnel where the gradients are the largest. The last step of the near-field benchmarking exercise let the teams improve their models by taking into account the EDZ and study the impact of heating on its extension. In the far-field benchmarking exercise, all teams considered an anisotropic poro-elastic behaviour, that is known to provide a good prediction of the evolution of both the temperature and the pore pressure (Seyedi et al., 2020). They got matching results for the pressure and stress evolution at mid-distance between two parallel high-level waste cells, which confirms the confidence one can have in these calculations and lends robustness to the modelling approach. Some of the triaxial compression tests performed by ULorraine were simulated and some models were updated to take into account the effect of the temperature on the mechanical properties. Five teams have finally been modelling the ALC1605 full-scale in-situ heating experiment. The work is being finalised, but the data will help calibrate and improve the models developed in the first steps. Reference is made to the D7.6 (Modelling report on effect of temperature on near field properties) deliverable for more detailed information on the benchmarking exercises.

3.1.2 Impact of HITEC on repository optimisation studies

Andra submitted in January 2023 the construction licence application (DAC) for the Cigéo disposal facility. During the review of this application by the French Nuclear Safety Authority (ASN), some technical and economic optimisation opportunities are studied in the framework of the technical and scientific activities. They focus mainly on the design of the waste packages or on the construction techniques, and some may be implemented during the gradual development of Cigéo.

Concerning the thermal transient, a reduction of the size of the underground facility through longer HLW cells or a shorter spacing between two adjacent cells would have a clear economic impact. Such optimisations should however always guarantee the achievement of the objectives for Cigéo and especially its safety. In addition, any optimisation should comply with the thermal (T<90°C) and THM (maximum Terzaghi effective stress) design criteria.

The HITEC project confirmed the robustness of the Cigéo project through its experimental and modelling activities and helped reducing some uncertainties by showing the small impact of heating on the properties of the Callovo-Oxfordian claystone.

3.1.3 ANDRA Conclusions on HITEC impact

The laboratory experiments performed in the framework of the EURAD HITEC project confirmed that the Callovo-Oxfordian claystone keeps its favourable mechanical and retention properties, even when heated at high temperature (up to 100°C). The near-field modelling benchmark showed that in an elastic case, very consistent results could be obtained with various codes and modelling teams. Some discrepancies were observed when anisotropy is introduced, that may be explained by differences in the mesh, the THM formulations and assumptions made in the different codes. An anisotropic poro-elastic behaviour was also considered in the far-field case. All the teams got then matching results for the pressure and stress evolution at mid-distance between two parallel high-level waste cells. These results confirm the robustness of the approach currently used by Andra to design the Cigéo disposal. The last step of the near-field benchmarking exercise let the teams improve their models by taking into account the EDZ around the HLW cells and study the impact of heating on its extension and behaviour. Some of the models were also enhanced after modelling triaxial compression tests performed in subtask

2.1 and 2.2. Finally, the modelling of the ongoing full-scale in-situ experiment (ALC1605) gave an opportunity to improve our understanding of the behaviour of the Callovo-Oxfordian claystone around a heated cell.

3.2 ENRESA

Some key HITEC experimental and modelling results have been formulated so far, both in the host rock formation and in the bentonite buffer, that could contribute to further refinements and considerations in the safety assessment of the repository.

Namely:

a) Host rock (Task 2)

- The effect of temperature on the self-sealing process of the host rock near field is considered not relevant.
- The peak strength of the host rock increases beyond 100 ºC due to desaturation effects induced by temperature.
- The increase of temperature results, as expected, in a noticeable increase of the pore pressure, but the related hydraulic conductivity decreases slightly, even though local mechanical failure is observed.
- The assessment of temperature effects on the long-term behaviour of the selected host rocks (Boom clay, Opalinus clay) could not be completed since it requires more data and longer duration creep tests. Concerning COx clay, only multi-step creep tests at ambient temperature were performed. The impact of temperature on the creep properties and long-term strength could not be analysed yet. The remaining multi-step creep tests at higher temperatures (40, 60 and 80 °C) are still in progress.

b) Bentonite buffer (Task 3)

- The temperature induced mineralogical alteration is minor up to 150 ºC.
- The swelling pressure decreases with increasing temperature, over time.
- Na based bentonite type is more affected by thermal effects than Ca based bentonite.
- Some temperature effects may reversible upon rehydration.

The experimental evidence from the HITEC project (Task 2 and Task 3), related to the thermal induced effects of both the clay host rock and the bentonite buffer for temperatures above 100°C, provides important input to the existing evidence basis and could contribute to further refinements and considerations in the safety assessment of the repository. Furthermore, evidence from the HITEC project will support the optimisation of the repository at a later stage.

3.3 Nagra

3.3.1 Impact of HITEC on performance assessment and safety case

Performance assessment is a key element of Nagra's safety assessment methodology, ensuring that safety-related claims are supported adequately by evidence.

Nagra's performance assessment is carried out at the component level for each element of the multibarrier systems, but also on the level of the entire system (total system performance) to account for any possible interactions between the individual components of the multibarrier system.

The assessment is carried out by formulating claims for the barrier performance at component / total system level, which are each supported by one or several arguments. Qualitative and quantifiable evidence are provided to confirm the arguments.

Claims are postulated to express the intended contributions of each barrier component and of the entire multibarrier system, respectively, to post closure repository safety. The claims essentially state that the requirements on the expected performance of the system as a whole and of the individual components are met. One or several safety functions are associated with each claim. Arguments are elaborated which support the claims on a qualitative or quantitative level. Convincing evidence must be provided to support the arguments in a robust manner. The evidence provided originates e.g., from experimental work and/or from model-based assessments. The assessment of this evidence is carried out qualitative or quantitatively, where quantitative evaluation involves the formulation of a traceable evaluation metrics, i.e., performance indicators, targets and evaluation scales, tailored to the claims and arguments to be addressed (an example is given in Table 3.1) .

Table 3.1: Example of the evaluation of a claim

Claim	(Dimensionless) performance indices	Performance target
The bentonite contributes to favorable thermal evolution of the nearfield.	Normalised temperature in the buffer (TT_{bb}/130 °C)	$TTbb < 1$

By producing a large body of evidence on the thermal response of clay-based material to temperatures above 100°C, the experimental evidence from the HiTec project provides important input to the evidence basis used to support the arguments made with regard to the thermal transient period and its impact on safety, e.g., it may support the establishment or re-evaluation of performance targets. Furthermore, evidence from HiTec will support the optimisation of the repository at a later stage.

Table 3.1: Contribution of EURAD HiTec to Nagra's safety case.

Relevant task	Topic	Impact
Task 2 - Clay host rock <120 °C / Sub-Task 2.1 and 2.2	Reduction of uncertainties related to the extent of the EDZ (propagation of fracture zone) and better characterisation of self-sealing processes in, and hydromechanical properties of, the EDZ after a thermal transient	Affects integrity of the hostrock
Sub-task 2.3	Development of a THM process model that will be benchmarked with large scale experiments, leading to increasing confidence in system analyses	Affects integrity of the hostrock
Task 3 Clay buffers > 100 °C (Subtask 3.1)	Reduce uncertainties in the impact of the clay buffer being subjected to high temperatures over a prolonged time period.	Affects the lifetime of the canister

3.3.1.1 Impact of thermal transient on buffer performance

In Nagra's concept, a layer of compacted bentonite, termed the bentonite buffer, fills the space between the canisters and the host rock. The primary function of the bentonite buffer is to provide a protective environment for the canister, ensuring that it completely contains the waste and the radionuclide inventory for the minimum required lifetime. During the non-isothermal re-saturation of the repository near field, coupled THMC (thermal, hydraulic, mechanical and chemical) processes are expected to occur, the impact of which on buffer performance has to be evaluated.

Despite the steady improvement in computational capacities and in the level of process understanding derived from the increasing body of experimental evidence, modelling coupled processes remains challenging in terms of conceptualisation and parameterisation. The thermal and thermo-hydraulic evolution of the bentonite buffer can be reasonably well captured by the current state-of-the-art models. However, the capability of these models to predict mechanical behaviour is less advanced. Difficulties in coupling the hydro-mechanical (HM) and chemical-mechanical (CM) behaviour of clay lie essentially in the conceptualisation of the clay's micro- and nanostructure evolution. Nevertheless, a dual-porosity framework, which has recently been developed in THM models, as well as in THC models, appears to offer a robust and consistent framework to represent coupled THMC processes. Within this framework, several different features and observations can be readily explained. To date, however, a fully coupled coherent THMC model is lacking. On the other hand, it is questionable whether HMC interactions are significant enough in terms of their impact on the safety-relevant properties of the buffer to justify the development of such a model framework.

3.3.1.2 Impact of thermal transient on host-rock performance

The thermal transient can potentially impact the transport and retention properties of the intact host rock, as well as its physical integrity. The effects of thermal transients on the transport properties of the intact host rock are mainly driven by changes in porosity (which affects diffusion) and mineralogy (which affects its sorption potential), which can be assessed at a small scale (with e.g., lab experiments), i.e., at a component level. The mechanical integrity of the host rock can also be evaluated at a small scale, but to develop an integrated understanding of their effects on disposal performance, far more complex assessment methods, including coupled THM models, are needed to provide robust evidence that supports arguments for adequate performance and safety.

Both performance indicators mentioned earlier, i.e., porewater pressures and paleo temperature, are suitable for assessing the impacts of the thermal transient on the transport properties of the host rock and its physical integrity, and hence on its performance and safety.

Once the phenomena related to the thermal transient have been assessed in terms of their magnitude, they are woven into a broad-brush description of the phenomenological evolution of the repository. In Nagra's case, this type of broad-brush description is written in the form of a storyboard, which allows for a consistency and completeness check regarding the handling of such phenomena throughout the safety case. The storyboard also provides the basis for the development of safety scenarios and analyses of their consequences in dose calculations.

A short version of the post-closure evolution of Nagra's repository, with emphasis on the thermal and pore-pressure transients, is given below:

After emplacement of spent fuel and high-level waste canisters, as well as buffer and seals of the emplacement drifts, the partially saturated excavation damage zone and rock support around the tunnels re-saturates with water, followed by saturation of the buffer and seals. The compacted clay structures saturate relatively quickly due to their high capillary pressure (suction), though the rate of saturation is limited by the low permeability of the rock and by decay heat generated by the waste. Full saturation is expected to occur some hundreds of years after emplacement. Saturation is sufficiently even to avoid potentially damaging stresses being exerted on the canisters. Any initial inhomogeneities in the buffer density are largely reduced over time. The buffer density around the canister is sufficiently high to prevent, when saturated, microbial activity that might otherwise increase the rate of canister corrosion.

Anaerobic and reducing conditions develop due to the consumption of O_2, e.g., by canister corrosion. Radiation shielding provided by the disposal containers is sufficient to protect the barriers from radiation-induced effects. Gas generation takes place due to the gradual anaerobic corrosion of the disposal canisters. Much of this gas dissolves in the porewater of the saturated bentonite buffer surrounding the canisters and diffuses through the rock support into the Opalinus clay porewater. However, some gas also migrates in the gas phase through the buffer and into the rock, where it dissolves. Gas migration causes no irreversible changes to either the buffer or rock.

In the first few thousands of years post-closure, repository generated heat, primarily decay heat from the waste forms, leads to a transient increase in temperature and pore pressure within and around the high-level waste disposal area, including within the confining rock zone, but these elevated temperatures and pore pressures have no long-term impact on the performance of the barriers. The tunnel support system around the emplacement drifts degrades and loses its mechanical strength, but the resulting stress redistribution has no impact on the safety-relevant properties of the barriers (any reactivation of excavation damage zone fractures will be temporary, as the fractures will reseal).

Porewater composition in the buffer gradually equilibrates with that in the rock. Chemical interactions, including interaction of canister corrosion products with the clay barriers and between the cementitious liner lining and the surrounding rock, are of limited spatial extent and do not affect the safety-relevant properties of the repository barriers.

In the period from a few thousands to more than ten thousand years post-closure, increasing numbers of disposal canisters become locally breached due to mechanical failure following weakening by corrosion or local flaws in the welding. The exposure of internal metal surfaces to water results in corrosion of these surfaces and increased gas production rates locally.

Based on the phenomenological description of the evolution of the repository (storyboard), the impact on safety can be assessed by formulating and analysing safety scenarios. These scenarios highlight the impact that uncertainty in the phenomenological evolution of the repository has on the demonstration of long-term safety. At Nagra, the following categories of scenarios exist: 1. The reference scenario, representing the most likely evolution of the repository, 2. alternative scenarios, representing paths of evolution that are less likely but nonetheless physically possible, and 3., "what-if?" cases, that are highly unlikely or even entirely hypothetical. "What-if?" cases and, to a certain extent, the alternative scenarios, help to define the resilience of the repository system towards disturbances. Consequence analysis is then used to evaluate the radiological consequences of the scenario, taking into account the geological boundary conditions, repository layout and the radionuclide inventory. According to current understanding, significantly degraded performance of the barriers due to the thermal transient does not form part of the reference scenario, but may be considered in alternative scenarios or "what-if?" cases.

3.3.2 Optimisation with regard to thermal transients

Robustness of the system with regard to the thermal transients is of particular interest; if porewater pressure increases to a degree that causes the host rock to lose its integrity due to the re-activation of faults, or even the creation of new fractures, then new discrete pathways for radionuclide transport can form. It is therefore essential to understand the processes that could occur during the thermal transient phase. This will allow appropriate mitigation methods to be developed, including increasing the size of the repository footprint, e.g., by increasing the distance between the drifts, or increasing the pitch between the canisters, or lowering the amount of spent fuel loaded in the canisters.

3.4 NWS

3.4.1 Scientific underpinning of the generic Environmental Safety Case

NWS' generic ESC employs a Claims, Arguments and Evidence (CAE) approach where:

- A top-level Claim is an overall statement about the safety of the system (e.g., that the disposal system is safe in the long term), and lower-level claims should support the top-level claim.
- An Argument is a description of what has been proved or established, and provides a link to the claim and to the supporting evidence.

- Evidence is then provided which supports the claims and arguments. This can be qualitative or quantitative, with multiple strands of evidence for one claim or argument. Evidence includes numerical modelling, laboratory experiments, field experiments and natural analogues.

This approach ensures that NWS can demonstrate in a transparent way that key safety functions of each component of the multi-barrier system are being met. The structured, interactive approach used makes it easier for the reader to find specific CAE they are interested in, and provides a golden thread from the top-level claim through to the underlying evidence. The HITEC work package feeds into this by providing the fundamental underpinning evidence for relevant safety claims and arguments.

3.4.2 Underpinning GDF optimisation and waste emplacement dates

Currently NWS employs thermal limits on bentonite and clay host rocks, to limit any alterations that will prevent the barriers from meeting their safety functions. Of key importance to NWS is to be able to optimise the GDF design to ensure the GDF footprint is minimised, while maintaining the key safety functions. The benefits of doing this are the significant financial and environmental savings that can be obtained. This can be achieved by increasing the thermal limits on the bentonite and clay host rock. Increasing the thermal limits also provides NWS with the opportunity to dispose of hotter waste earlier, significantly reducing interim storage times and the costs associated with it. Therefore the results from HITEC are critical to establish thermal limit requirements on clay-based barriers in a UK GDF once a site has been selected.

3.4.3 Development of a scientific community

Beyond the technical understanding gained, HITEC has also been very useful in establishing a Europe-wide community. Critically, it has further connected research institutions with waste management organisations, helping to ensure that the research being conducted is both timely and needs-driven in relation to GDF programmes. It has also brought together modellers and experimentalists, which ensures that both communities understand each other's needs better, allows for problem solving, and for critique. This ensures the scientific results are robust. NWS notes that this has been achieved during a very challenging time due to COVID-19, which is commendable.

3.4.4 Key findings and significance to NWS

3.4.4.1 Lower strength sedimentary rocks (LSSR)

The relative importance of thermal processes on the geological environment depends on the host rock properties. Participation in HITEC has ensured NWS is aware of key processes that need to be considered for one of the UK generic geologies (LSSR) which is based on Jurassic clay host rocks such as the Opalinus Clay and Callovo-Oxfordian Clay.

The laboratory experiments have helped to further elucidate the implications of temperature on pore pressures, and the subsequent risk of fracturing, in several clay host rocks. This is highly dependent on the rock properties, whereby significant increases in pore pressure are initially seen in lower permeability clay host rock, noting that these pore pressures will dissipate over time. This can lead to pore pressures greater than the confining stress of the system, subsequently causing fracturing. The likelihood of this appears to be significantly reduced in systems with higher confining pressures. This indicates that initial local stress conditions (e.g., initial confining pressure and pore pressure) need to be considered on a site-specific basis. In general, the results of the programme indicate that the ability of the clay host rocks to self-seal and its final permeability are not impacted negatively by increased temperatures up to 80-90°C under the conditions tested.

Coupled thermal-hydro-mechanical models have also been developed in HITEC to represent key processes occurring during in-situ full-scale field experiments in Jurassic clay by a number of teams, these models have benchmarked against each other. These models are extremely useful as a way of upscaling understanding gained from the laboratory scale to large, GDF scales, and also a way of extrapolating over long periods of time. These models play an integral part in safety case development through NWS' modelling strategy.

The results of HITEC provide fundamental underpinning of the UK safety case should a Jurassic clay host rock be selected, and provide an initial indication that a thermal limit on LSSR of 90°C is acceptable (depending on site-specific conditions). This allows for optimisation of the GDF footprint, should bentonite be shown to meet its safety functions at temperatures greater than 100°C (see bentonite section below).

3.4.4.2 Bentonite

HITEC has also pushed forward the understanding of bentonite performance at temperatures above 100°C across a range of scales, using numerical and experimental approaches. Tests on thermally treated bentonite indicate that thermal-chemical reactions that lead to unfavourable mineralogical and chemical properties of bentonite (e.g., illitisation) are limited in the temperature range and geochemical conditions studied. The results also help to understand the performance of bentonite under different saturation states (i.e., dry vs fully saturated) that are representative of those that bentonite is likely to experience as the HHGW disposal areas evolve over time. These results generally indicate limited impact on bentonite swelling pressure and hydraulic conductivity under the conditions tested (typically up to 150 °C).

The swelling pressure and hydraulic conductivity in the above tests have been conducted once the bentonite has cooled, representative of the long-term post-closure conditions. Equally as important is understanding bentonite in-situ performance while temperatures are elevated, representative of short-term post closure conditions. The indicative results from HITEC suggest a significant drop in swelling pressure which becomes more extensive as temperature increases (≥150 °C), although there is evidence that hydraulic conductivity remains relatively unaffected. The mechanism for this is currently unknown and occurs to different extents depending on the type of bentonite tested and the initial dry density. This needs to be investigated further.

The above results from HITEC suggest that bentonite is relatively unaffected at elevated temperatures in the long-term, which may allow for increased thermal limits on bentonite under the conditions tested. However, consideration needs to be given to the impact of reduced swelling pressures during short-term post-closure timescales. Further understanding needs to be developed regarding the mechanism leading to in-situ swelling pressure reduction and the permanency of this reduction observed through this type of testing. A comparison of the boundary conditions, bentonite material processing and experimental apparatus needs to be conducted between the in-situ tests and thermal treatment tests, to fully understand observed differences.

NWS recognises the importance of the boundary conditions employed within the numerical and experimental research undertaken in HITEC, which is being considered on a site-specific basis. This includes (but is not limited to) the groundwater composition, the mineralogical/chemical properties of the bentonite and host rock, and initial stress conditions in the host rock.

3.5 ONDRAF/NIRAS

3.5.1 Optimisation with regard to thermal transients

In the case of the Belgian programme, if the selected host rock turns out to be a poorly indurated clay, there is little margin for optimisation of the thermal design. Indeed, it has been explained above that the main constraint designs are not related to the host rock but to the corrosion overpack and to the protection of surrounding water resources. It has been shown that a design that meets these constraints leads to a thermal evolution of the system that will not significantly affect the favourable barrier properties of the host clay. Hence, no additional constraints directly linked to the mechanical resistance of the host rock had to be considered for this design.

3.5.2 Benefits from HITEC to the safety case

In consequence of the above, the main benefits from HITEC to the Belgian programme are related to Task 2. In particular, the modelling benchmark performed for the large scale in situ heater test PRACLAY in HADES allowed to confirm and strengthen confidence in our understanding of the THM behaviour of the Boom Clay. Sensitivity analyses performed within HITEC have allowed to further refine the values of the anisotropic mechanical properties and of the thermal conductivities of the Boom Clay.

Table 3.2: Contribution of EURAD HiTec to the safety case of ONDRAF/NIRAS.

Relevant task	**Topic**	**Impact**
Task 2 - Clay host rock <120°C / Sub-Task 2.1 and 2.2	Reduction of uncertainties related to the extent of the EDZ (propagation of fracture zone) and better characterisation of self-sealing processes in, and hydromechanical properties of, the EDZ after a thermal transient	Strengthen confidences in understanding of THM behaviour of Boom Clay; refine the values of THM parameters
Sub-task 2.3	Development of a THM process model that will be benchmarked with large scale experiments, leading to increasing confidence in system analyses	Strengthen confidences in understanding of THM behaviour of Boom Clay; refine the values of THM parameters
Task 3 Clay buffers > 100°C (Subtask 3.1)	Reduce uncertainties in the impact of the clay buffer being subjected to high temperatures over a prolonged time period.	Not applicable to the Belgian programme
(Subtask 3 .2)	Derivation of parameters concerning the hydro-mechanical properties of bentonite for temperatures higher than 100°C in support of modelling.	Not applicable to the Belgian programme
(Subtask 3.3)	Review of key processes at high temperature relevant to safety and, more specifically, to resaturation rate. Further calibration and development of suitable THM models for clay buffer at higher temperatures.	Not applicable to the Belgian programme

3.1 POSIVA

3.1.1 Contribution of HITEC to safety case

Posiva's Performance Assessment assesses the uncertainties related to the fulfilment of the performance targets, such as the effect of temperature on the buffer swelling pressure, self-sealing capability and mechanical properties. The HITEC project contributes to the understanding of the thermal effect on the bentonite properties. It has been identified in HITEC that even after relatively high-temperature treatment (~150 °C) bentonite maintains most (if not all) its properties. This highlights that in reality the buffer is more reliable than assumed in safety case and gives more certainty on buffer working as intended even if high temperatures are reached.

3.1.2 Impact of HITEC on repository optimisation studies

HITEC results may allow higher allowed temperature in the repository in the future. However, as temperature affects other barriers than bentonite (namely canister insert, canister shell, near-field host-rock, and far-field host-rock), the effects on all the other barriers still have to be analysed.

3.2 SKB

3.2.1 Optimisation with regard to thermal transients

The upper temperature limit of 100 °C was established a long time ago in the Swedish program. The application for a KBS-3 repository in Forsmark was based on this criterion. The favourable outcome of the regulatory review included the assumption that this criterion would be met.

The current canister design with room for 12 BWR or 4 PWR spent fuel elements together with the duration of the interim storage and the ambient temperature and the thermal conductivity of the rock at Forsmark makes it relatively easy to meet the temperature criterion of 100 °C. The practical and economical gain of a higher maximum temperature would be rather limited.

3.2.2 Contribution of HITEC to safety case

The rock in Forsmark is crystalline, which means that the studies of clay host rocks in HiTEC is of limited interest for SKB. SKB has currently no plans to increase the maximum temperature in the bentonite above 100 °C. The result from the bentonite tests in HiTEC at higher temperatures are still very valuable in the safety case since they can be used to interpolate observations to 100 °C.

Some important findings are:

- The lack of observations of significant transformation of montmorillonite strengthens the assumption in the safety case that the montmorillonite in the bentonite will be stable during the entire assessment period.
- The observations that the swelling pressure is unaffected by heating strengthens the assumption that the bentonite will behave as expected during the thermal phase.
- The lack of observations of “new” high temperature process indicates that our understanding of the bentonite behaviour under repository conditions is sound.
- Deviator stress seem to decrease after heating, which is helpful since it will decrease the possible shear load on the canister.
- Observations of precipitation/dissolution of accessory minerals are consistent with the general understanding of the processes.

There are however also observations that are not fully understood

- The CEC bentonite is affected by heating in some tests, despite the fact that there were no observations of montmorillonite transformation.
- There are observations of increased hydraulic conductivity after heating.
- The effect of heating of a unsaturated bentonite may be different from the effect of heating a saturated bentonite.
- The liquid limit and swell index of dry treated bentonite are lower. The decrease of both parameters is observed as a function of the heating time.
- Disintegration of installed bentonite blocks during heating in field tests was observed.

These would need further explanation, but are most likely not critical for the long-term performance of the barrier.

3.3 SURAO

A complex performance assessment of the Czech EBS system is planned to be completed in 2026. The current concept which will be assessed is with a temperature limit of up to 95°C for normal scenarios, alternative scenarios will count with locally occurred temperatures above 100°C.

The results of the HITEC project show that the temperature limit of 95°C for bentonite is possibly too conservative and no significant alteration of montmorillonite should not be the case with temperatures up to 150°C. If these findings will be verified it will be necessary to verify the impact of this elevated temperature on other DGR elements (canister, host rock) and processes (i.e. vapor generation, stresses, transport of fluids, colloid formation…).

4. Summary and conclusions

To ensure the long-term safety of deep geological disposal Waste Management Organizations (WMOs), formulate safety related claims pertaining to their technical and natural barriers. These high-level claims that explain why safety is given are typically substantiated by compelling arguments that explain why these barriers are performing as required. These arguments, in turn, draw strength from a plethora of evidence, including experimental studies, empirical knowledge, the study of natural analogues, and modelling evidence.

Claims associated with technical and natural barriers, especially those subjected to high temperature gradients, demand a repository specific understanding of thermal transients and their couplings. Consequently, building a specific body of arguments becomes fundamental for the safety case of every repository concept.

The needed evidence for supporting the claims is comprising an array of experimental and empirical research, insights derived from natural analogues, and modeling-based findings. This extensive evidential foundation often finds its origins in collaborative initiatives like the High-Temperature Experiments in Crystalline Host Rock (HiTec) and similar international projects.

The strength and reliability of the evidence provided hinge upon its capacity to accurately represent the anticipated conditions within a geological repository. Therefore, it is essential to establish a well-constrained evolutionary path ("storyboard") for thermal, hydraulic, and mechanical (THM) conditions before specifying the parameters for testing. By gaining a comprehensive understanding of the evolution of safety-related properties in both technical and natural barriers, we can reduce unnecessary conservatisms and allow for optimizations for various aspects of the repository, from individual components to the layout of the entire facility.

In conclusion, the robust substantiation of safety claims is an indispensable aspect for a convincing safety case. This process hinges on a diverse and comprehensive body of evidence, requiring an understanding of thermal transients, the evolutionary path of THM conditions, and the ability to reduce conservatism, ultimately enabling the optimization of key components and repository designs. This concerted effort contributes significantly to the safeguarding of our environment and future generations.

5. References

Agboli M., Grgic D., Moumni M., Giraud A., 2023. Study Under X-Ray Tomography of the Impact of Self-Sealing Process on the Permeability of the Callovo-Oxfordian Claystone. RMRE, https://doi.org/10.1007/s00603-023-03350-y

Andra, 2016. Safety Options Report - Post-Closure Part (DOS-AF). Document CG-TE-D-NTE-AMOA-SR2-0000- 15-0062

Andra, 2022a. Dossier d'autorisation de création de l'installation nucléaire de base (INB) Cigéo – Pièce 7 : Version préliminaire du rapport de sureté – Partie II : Description de l'INB, de son environnement et de son fonctionnement et évolution du système de stockage après fermeture – Volume 3 : Les colis de déchets. Document CG-TE-D-NTE-AMOA-SR0-0000-21-0007/A

Andra, 2022b. Dossier d'autorisation de création de l'installation nucléaire de base (INB) Cigéo – Pièce 7 : Version préliminaire du rapport de sureté – Partie III : Démonstration de sureté – Volume 8 : La démonstration de sureté après fermeture. Document CG-TE-D-NTE-AMOA-SR0-0000-21-0007/A

Autorité de sûreté nucléaire (ASN), 2008. Guide de sûreté relatif au stockage définitif des déchets radioactifs en formation géologique profonde. Guide de l'ASN. 32 p.

Bumbieler F, Plúa C, Tourchi S, et al., 2020. Feasibility of constructing a full-scale radioactive high-level waste disposal cell and characterization of its thermo-hydromechanical behavior. Int J Rock Mech Min Sci. 2021;137:104555. https://doi:10.1016/j.ijrmms.2020.104555

Gbewade C.A.F., Grgic D., Giraud A., Schoumacker L., 2023. Experimental study of the effect of temperature on the mechanical properties of the Callovo-Oxfordian Claystone. RMRE, under review

Seyedi D.M., Plúa C., Vitel M., Armand G., Rutqvist J., Birkholzer J., Xu H., Guo R., Thatcher K.E., Bond A.E., Wang W., Nagel T., Shao H., Kolditz O., 2021. Upscaling THM modeling from small-scale to full-scale in-situ experiments in the Callovo-Oxfordian claystone, Int J Rock Mech Min Sci., Volume 144, https://doi.org/10.1016/j.ijrmms.2020.104582

Enresa (2004). ENRESA 2003. Evaluación del comportamiento y de la seguridad de un almacén geológico profundo de residuos radiactivos en arcilla. 49-1PP-M-A1-01. February 2004.