



TECHNICAL REPORT 24-10 Rev. 1

Post-Closure Safety Report

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Disposal of Radioactive Waste**

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Post-Closure Safety Report

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Key outcomes

The Nuclear Energy Act (KEG) requires deep geological disposal for all of Switzerland's radioactive waste. Nagra foresees constructing a repository for high-level and low- and intermediate-level waste, including alpha-toxic waste, in the Opalinus Clay host rock with the surface facilities at the Haberstal site. The present post-closure safety case demonstrates that this repository will remain safe after closure.

Safety-based site selection, design and implementation process

The Sectoral Plan for Deep Geological Repositories defined the safety-based site selection process that began in 2008. This process led Nagra to select the Opalinus Clay as the preferred host rock and to propose Nördlich Lägern as the most suitable siting region. A provisional repository design, elaborated to the extent necessary for the present programme stage, has been developed with a focus on safety. It is based on a mature but still provisional safety and repository concept, which itself is the result of many years of RD&D carried out by Nagra and in line with international best practice. Post-closure safety is based on well-understood features of the system, termed pillars of safety, that provide safety functions. The procedures for constructing, operating and closing the repository are established, available and affordable.

Quality of the site, containment-providing rock zone and engineered barriers

The geological barrier is of primary importance due to the excellent qualities and long-term geological stability of the Opalinus Clay host rock and the wider containment-providing rock zone (CRZ). The site and the CRZ, which includes both the Opalinus Clay and adjacent rock layers, have several favourable safety-relevant properties, including, in the case of the CRZ, generally low permeability, excellent radionuclide retention properties and a high swelling capacity, indicating a high self-sealing potential. The sufficient spatial extent of the CRZ, the broader hydrogeological situation, the geologically quiet location and the lack of any significant underground resources specific to that region that could attract future human actions are also important characteristics of the site. The engineered barriers and the repository design are adapted to the site and CRZ, with a focus on safety. The barriers are compatible with each other and contribute passively to post-closure safety, particularly by minimising disturbance of the CRZ.

Quality and comprehensiveness of the safety assessment

The quality and comprehensiveness of the safety assessment build on the quality and comprehensiveness of the assessment basis. This includes a sound scientific understanding of relevant phenomena and processes, a well-defined safety assessment methodology that manages uncertainty and promotes comprehensiveness, and well-established models, codes, and databases. Nagra has developed these over many years within the framework of site characterisation, repository development activities, and RD&D programmes. The safety assessment is carried out in a systematic, structured, comprehensible and verifiable manner. It is also consistent with national and international requirements, guidance and developments and includes measures to ensure that safety-relevant phenomena have been represented appropriately in the assessment. A management system ensures that the safety assessment methodology is fully and correctly applied. In particular, measures of quality assurance and quality control are applied to all activities that use or produce models and data.

Favourable assessment results

Performance assessment shows that the repository system will perform its safety functions in all reasonably foreseeable circumstances. The analysis of radiological consequences, which considers all identified uncertainties, shows that the protection criteria are met with substantial safety margins. These include uncertainties in the reference safety scenario as well as those in alternative safety scenarios and those concerning future human actions. Unrealistic and purely hypothetical “what-if?” cases are also considered. These calculation cases assume the impairment of one or more pillars of safety. Nevertheless, the safety margins are generally large here too, further demonstrating the robustness of the system.

Complementary lines of argument

Complementary lines of argument have been developed in addition to those gained from the main safety assessment processes (performance assessment, scenario development and the analysis of radiological consequences). These include findings from studies of natural and archaeological analogues, providing insight into the durability of system components and into the effective isolation of radioactive materials in a repository. Furthermore, radiotoxicity is used, for example, to compare the hazard of the deep geological repository with naturally occurring radiotoxicity in our environment.

Synthesis of the post-closure safety case

Based on the above-mentioned arguments and the wealth of supporting evidence at hand, Nagra showcases large margins in the post-closure safety of a repository for high-level and low- and intermediate-level waste constructed in the Opalinus Clay host rock with surface facilities at the Haberstal site. Because of the robustness of the multi-barrier system, and especially of the Opalinus Clay, assessment results have demonstrated that the post-closure safety of the repository is ensured even under the most pessimistic assumptions.

Executive Summary

This report presents the post-closure safety case for a geological repository for low- and intermediate-level waste (L/ILW), alpha-toxic waste (ATW) and high-level waste (HLW) in Opalinus Clay with surface facilities at the Haberstal site.

Chapter 1 sets the scene. The repository will be located in the Opalinus Clay at the Haberstal site. The chapter contextualises the post-closure safety case within the framework of the general license application, presents the scope of the report, and outlines its structure.

Chapter 2 emphasises the international consensus on deep geological disposal as the appropriate long-term management option for high-level radioactive waste and details the international principles and guidance with respect to post-closure safety of repositories. It sets out the national laws and legislation that form the context and framework for the post-closure safety case.

Chapter 3 presents the current safety and repository concept and shows that it is aligned with legal requirements and both national and international principles. It defines and explains the passive safety functions provided by reliable and well-understood pillars of safety. The chapter also describes a provisional repository design suitable for the current post-closure safety case. It explains how post-closure safety, via the current safety and repository concept, results from the properties at the chosen site and the provisional repository design.

Chapter 4 demonstrates that the post-closure safety case is built on a traceable and transparent methodology, including performance assessment, development of safety scenarios, and radiological consequence analysis. It defines the time period for assessment, considering the decline in radiological toxicity over time, and shows how the methodology integrates uncertainty management and promotes comprehensiveness.

Chapter 5 describes the assessment basis, which has been developed over decades and provides the foundation for a comprehensive performance and safety evaluation. This includes the scientific understanding and safety-related properties of the containment-providing rock zone and the geological setting. The chapter also summarises the expected evolution of the repository system and the fate of radionuclides, as well as the models, codes and databases used in the safety assessment.

Chapter 6 focuses on the performance assessment (PA), evaluating the functionality of each barrier component and of the total system and considering the expected evolution and uncertainties. It develops arguments supporting the current safety and repository concept, addresses uncertainties, and screens PA outcomes to identify performance deviations relevant to safety, ensuring comprehensive coverage of features, events and processes (FEPs).

Chapter 7 discusses the safety scenarios that capture uncertainties in the performance of the pillars of safety and their functions. It connects performance assessment arguments with radiological consequence analysis to demonstrate the fulfilment of protection criteria. The chapter explains how safety scenarios link real-world phenomena and processes with highly abstracted radiological consequence analysis models, including scenario variants and calculation cases.

Chapter 8 presents the radiological consequence analysis. Uncertainties in radionuclide release, retention, and transport within each safety scenario are identified and analysed. Calculated safety indicators, namely dose rates and risks, are compared with protection criteria to evaluate safety margins. The results demonstrate that the protection criteria are met for all safety scenarios. The assessment of sensitivities and the impact of uncertainties on radionuclide behaviour illustrate ample safety margins.

Chapter 9 describes natural and archaeological analogues that enhance confidence in understanding phenomena and processes. It presents complementary safety indicators to provide additional insights into the hazard potential of the waste and radiotoxicity evolution. The chapter highlights the numerous conservative assumptions and simplifications incorporated into the models for radiological consequence analysis.

Chapter 10 integrates the safety arguments formulated throughout the report into a holistic demonstration of post-closure safety.

Chapter 11 explains the positioning of the current post-closure safety case within the legally required licensing steps for the deep geological repository in Switzerland.

Chapter 12 is the bibliography and in **Chapter 13**, a glossary is presented.

App. A highlights the main commonalities and differences of the present post-closure safety case compared with the safety case developed for Project *Entsorgungsnachweis*, while **App. B** summarises the key phenomena identified in performance assessment, their relevance to system evolution and performance, and their potential to lead to performance deviations. Finally, **App. C** considers non-radioactive but potentially hazardous materials that are chemically toxic or could pose a risk to water quality, which can also be safely contained in a deep geological repository.

Zusammenfassung der wichtigsten Resultate

Das Kernenergiegesetz (KEG) schreibt die geologische Tiefenlagerung für die Entsorgung aller radioaktiven Abfälle in der Schweiz vor. Die Nagra plant, ein geologisches Tiefenlager für hochaktive und schwach- und mittelaktive Abfälle, inklusive alpha-toxischer Abfälle, im Opalinuston am Standort Haberstal zu bauen. Der vorliegende Sicherheitsnachweis für die Nachverschlussphase legt überprüfbar dar, dass dieses Tiefenlager auch nach Verschluss langfristig sicher ist.

Sicherheitsgerichteter Standortauswahl-, Design- und Implementierungsprozess

Der Sachplan geologische Tiefenlager legt seit 2008 den sicherheitsgerichteten Standortauswahlprozess fest. Diesem zufolge hat die Nagra den Opalinuston als bevorzugtes Wirtgestein sowie Nördlich Lägern als Standortregion vorgeschlagen. Die Entwicklung einer beispielhaften Auslegung eines Tiefenlagers, das für den aktuellen Projektstand ausgearbeitet worden ist, war ebenfalls sicherheitsgerichtet. Die Auslegung basiert auf einem ausgereiften, wenn auch beispielhaften, Sicherheits- und Lagerkonzept, welches das Ergebnis jahrelanger Forschung und Entwicklung der Nagra für ein Tiefenlager im Opalinuston ist, und im Einklang mit internationalem Wissen, steht. Die Sicherheit nach dem Verschluss beruht auf sehr gut verstandenen Pfeilern der Sicherheit, welche jeweils wichtige Sicherheitsfunktionen erfüllen. Die Verfahren zum Bau, Betrieb und Verschluss des geologischen Tiefenlagers sind etabliert, verfügbar und umsetzbar.

Qualität des Standorts, des einschlusswirksamen Gebirgsbereichs und der technischen Barrieren

Die geologische Barriere ist aufgrund der hervorragenden Eigenschaften und der langfristigen geologischen Stabilität des Opalinustons und des einschlusswirksamen Gebirgsbereichs (EG) von grosser Bedeutung. Der Standort und der EG weisen eine Reihe sehr günstiger sicherheitsrelevanter Eigenschaften auf. Der EG, und im Besonderen das Wirtgestein Opalinuston, besitzen eine geringe Durchlässigkeit, gute Radionuklid-Rückhalteeigenschaften und eine hohe Quellfähigkeit, welche Risse im Gestein abdichtet. Die ausreichende Ausdehnung des EG, die übergeordnete hydrogeologische Situation, die geologisch ruhige Lagerung, sowie das Fehlen von geologischen Rohstoffen im besonderen Masse, die zukünftige menschliche Aktivitäten im Untergrund anziehen könnten, sind weitere wichtige Eigenschaften des Standorts. Die technischen Barrieren und die Lagerauslegung sind gemäss dem Standort und dem EG ausgelegt, mit einem Fokus auf die Sicherheit. Die Barrieren sind untereinander kompatibel und tragen passiv zur Sicherheit bei. Im Besonderen werden dadurch mögliche sicherheitsrelevante negative Einflüsse auf das Wirtgestein und den EG minimiert.

Qualität und Vollständigkeit der Prüfung der Langzeitsicherheit

Die Qualität sowie die Vollständigkeit des Sicherheitsnachweises bauen auf der Qualität und Vollständigkeit der Bewertungsgrundlage auf. Die Bewertungsgrundlage umfasst ein fundiertes Verständnis relevanter Phänomene und Prozesse, die Methodik zur Prüfung der Langzeitsicherheit sowie Modelle, Modellierungswerkzeuge und Datenbanken. Diese hat die Nagra im Rahmen der Standortcharakterisierung, der Aktivitäten zur Tiefenlagerentwicklung sowie RD&D Aktivitäten über viele Jahre erarbeitet. Die Prüfung der Langzeitsicherheit erfolgt systematisch, strukturiert, nachvollziehbar und überprüfbar. Zudem ist sie mit nationalen und internationalen Anforderungen, Leitlinien und Entwicklungen konsistent. Des Weiteren umfasst die Prüfung eine systematische Behandlung von Ungewissheiten, Massnahmen zur Sicherstellung der vollständigen Abdeckung potenziell sicherheitsrelevanter Phänomene sowie Massnahmen, die sicherstellen, dass diese in der Beurteilung angemessen berücksichtigt werden. Zudem stellt ein

Qualitätssicherungssystem sicher, dass die Prüfmethodik vollständig und korrekt angewendet wird. Als Teil der Qualitätssicherung werden im Besonderen Qualitätskontrollen auf alle Aktivitäten angewendet, die Modelle und Daten nutzen oder produzieren.

Günstige Beurteilung der Langzeitsicherheit

Die Prüfung der Barrierenwirksamkeit zeigt auf, dass das Tiefenlagersystem seine Sicherheitsfunktionen für alle realistischerweise möglichen Entwicklungen mit hohen Sicherheitsmargen erfüllt. Die Analyse der radiologischen Konsequenzen demonstriert das Einhalten der Schutzkriterien und zeigt, dass grosse Sicherheitsmargen auch dann bestehen, wenn alle identifizierten Ungewissheiten berücksichtigt werden. Darunter fallen die Ungewissheiten im Referenz-Sicherheitsszenario, diejenigen in alternativen Sicherheitsszenarien sowie diejenigen in Sicherheitsszenarien zukünftiger unbeabsichtigter menschlicher Aktivitäten. Ebenfalls betrachtet wurden unrealistische und rein hypothetische «what-if?»-Fälle. Bei diesen Spezialfällen wird angenommen, dass einer oder mehrere Sicherheitspfeiler signifikant beeinträchtigt werden. Trotzdem sind auch hier die Sicherheitsmargen generell gross, was die Robustheit des Systems zusätzlich unterstreicht.

Ergänzende Argumente

Zusätzlich zu der Methodik folgenden Prüfung der Langzeitsicherheit wurden weitere unterstützende Argumente hergeleitet. Dazu gehören beispielsweise die Erkenntnisse aus Studien natürlicher und archäologischer Analoga. Sie bieten Einblick in die Haltbarkeit von Systemkomponenten und die wirksame Isolation radioaktiver Materialien in einem Tiefenlager. Weiter wird zum Beispiel mit Hilfe der Radiotoxizität das Gefährdungspotenzial des geologischen Tiefenlagers in den Kontext natürlich vorkommender Radiotoxizität in unserer Umwelt gestellt.

Synthese des Sicherheitsnachweises für die Nachverschlussphase

Die oben genannten Argumente und die Fülle an Evidenzen, die sie stützen, demonstrieren mit grossen Sicherheitsmargen die Sicherheit für die Nachverschlussphase eines geologischen Tiefenlagers im Opalinuston am Standort Haberstal. Aufgrund der Robustheit des Mehrfachbarrierensystems, insbesondere des Opalinustons, haben Sicherheitsanalysen gezeigt, dass die Langzeitsicherheit des Endlagers selbst unter den pessimistischsten Annahmen gewährleistet ist.

Kurzfassung des Berichts

In diesem Bericht wird der Sicherheitsnachweis für ein geologisches Tiefenlager für schwach- und mittelaktive Abfälle (SMA), alphanukleare Abfälle (ANA) und hochaktive Abfälle (HAA) im Opalinuston mit der Oberflächenanlage am Standort Haberstal vorgestellt.

Kapitel 1 setzt den Rahmen für diesen Bericht. Das geologische Tiefenlager am Standort Haberstal wird im Opalinuston gebaut. Das Kapitel kontextualisiert den Sicherheitsnachweis für die Nachverschlussphase für das Rahmenbewilligungsgesuch. Darüber hinaus wird der Berichtsinhalt abgegrenzt und der Aufbau des Berichts dargelegt.

Kapitel 2 unterstreicht den internationalen Konsens der geologischen Tiefenlagerung zur langfristigen Entsorgung für hochradioaktive Abfälle und beschreibt die internationalen Grundsätze und Leitlinien zu Langzeitsicherheitsaspekten der geologischen Tiefenlagerung. Es führt die nationalen Gesetze und Vorschriften auf, welche den Kontext und den Rahmen für den Sicherheitsnachweis für die Nachverschlussphase bilden.

In **Kapitel 3** wird das aktuelle, beispielhafte Sicherheits- und Barrierenkonzept vorgestellt, das den gesetzlichen Anforderungen sowie den nationalen und internationalen Grundsätzen entspricht. Es definiert und erläutert die passiven Sicherheitsfunktionen, die von zuverlässigen und gut verstandenen sogenannten Pfeilern der Sicherheit erfüllt werden. Das Kapitel beschreibt zudem die beispielhafte Umsetzung, die für den aktuellen Sicherheitsnachweis geeignet ist. Weiter wird erläutert, wie die Sicherheit nach dem Verschluss gemäss dem Sicherheits- und Barrierenkonzept aus den Eigenschaften am gewählten Standort und der darauf abgestimmten beispielhaften Umsetzung gewährleistet ist.

Kapitel 4 zeigt, dass der Sicherheitsnachweis für die Nachverschlussphase auf einer nachvollziehbaren und transparenten Methodik beruht. Diese umfasst die Prüfung der Barrierenwirksamkeit, die Entwicklung von Sicherheitsszenarien und die Analyse möglicher radiologischer Konsequenzen. Im Kapitel wird der Nachweiszeitraum definiert, wobei die Abnahme der Radiotoxizität im Laufe der Zeit berücksichtigt wird. Zudem wird gezeigt, wie die Methodik den systematischen Umgang mit Ungewissheiten integriert.

Kapitel 5 beschreibt die über Jahrzehnte entwickelten technischen und wissenschaftlichen Grundlagen für eine umfassende Prüfung der Barrierenwirksamkeit und für die Beurteilung möglicher radiologischer Konsequenzen. Es liefert die aus den Eigenschaften des einschliessungswirksamen Gebirgsbereichs und der geologischen Standortregion abgeleiteten sicherheitsrelevanten Argumente. Das Kapitel fasst zudem die zu erwartende Entwicklung des Tiefenlagersystems und den Verbleib der Radionuklide zusammen.

Kapitel 6 präsentiert die Prüfung der Barrierenwirksamkeit (PA, aus dem Englischen «*performance assessment*»). Die Funktionalität jeder Barrierenkomponente sowie des gesamten Systems wird, ausgehend von der erwarteten Entwicklung und unter Berücksichtigung der Ungewissheiten, beurteilt. Dies liefert einerseits Argumente zur Bestätigung des Sicherheits- und Barrierenkonzepts, andererseits werden Ungewissheiten bewertet und aus den Ergebnissen der PA potenziell sicherheitsrelevante Abweichungen von der erwarteten Entwicklung identifiziert. Eine umfassende Abdeckung von Eigenschaften, Ereignissen und Prozessen (FEP, aus dem Englischen «*Features, Events and Processes*») wird so gewährleistet.

In **Kapitel 7** werden die Sicherheitsszenarien diskutiert, welche die Ungewissheiten bei der Wirksamkeit der Pfeiler der Sicherheit und deren Funktionen abdecken. Dies liefert die Verbindung zwischen den Argumenten der Barrierenwirksamkeit und der radiologischen Konsequenzenanalyse, welche zum formellen Nachweis des Einhaltens der Schutzkriterien durchgeführt werden. Zudem wird erläutert, wie die Sicherheitsszenarien zusammen mit ihren Varianten und Rechenfällen reale Phänomene und Prozesse mit den stark abstrahierten Modellen der Konsequenzenanalyse verknüpfen.

Kapitel 8 präsentiert die radiologische Konsequenzenanalyse. Ungewissheiten bei der Freisetzung, dem Rückhalt und dem Transport von Radionukliden werden identifiziert und beurteilt. Sicherheitsindikatoren, namentlich Dosisleistung und Risiko, werden berechnet und mit den Schutzkriterien verglichen, um Sicherheitsmargen auszuweisen. Die Resultate zeigen, dass die Schutzkriterien für alle Sicherheitsszenarien eingehalten werden. Die Analyse von Sensitivitäten und der Auswirkung von Ungewissheiten zeigt auf, dass grosse Sicherheitsmargen vorliegen.

Kapitel 9 beschreibt natürliche und archäologische Analoga, die das Verständnis von Phänomenen und Prozessen zusätzlich erhöhen. Es werden ergänzende Sicherheitsindikatoren präsentiert, die das Gefahrenpotenzial der Abfälle und die Entwicklung der Radiotoxizität einordnen. Das Kapitel fasst die zahlreichen konservativen Annahmen und Vereinfachungen zusammen, die in die Modelle der Konsequenzenanalyse einfließen.

In **Kapitel 10** werden die im Bericht formulierten Sicherheitsargumente in eine ganzheitliche Darstellung der Langzeitsicherheit integriert.

Kapitel 11 erläutert die Positionierung des aktuellen Langzeitsicherheitsnachweises für die Nachverschlussphase innerhalb der gesetzlich vorgeschriebenen Schritte im Bewilligungsprozess für das geologische Tiefenlager in der Schweiz.

Kapitel 12 beinhaltet das Literaturverzeichnis und **Kapitel 13** das Glossar.

App. A erläutert die wichtigsten Gemeinsamkeiten und Unterschiede des vorliegenden Langzeitsicherheitsnachweises und demjenigen, welcher im Kontext des Entsorgungsnachweises erstellt wurde, während in **App. B** die Schlüsselphänomene, welche über die Prüfung der Barrierenwirksamkeit identifiziert wurden, deren Relevanz für die Systementwicklung und das Systemverhalten sowie deren Potenzial, zu Abweichungen in der Wirkung zu führen, zusammengefasst sind. Schliesslich werden in **App. C** nicht-radioaktive jedoch potenziell gefährliche Substanzen, die chemotoxisch sind oder ein mögliches Risiko für die Wasserqualität darstellen, behandelt, die ebenfalls in einem geologischen Tiefenlager sicher eingeschlossen werden können.

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List of Acronyms

ATW	Alpha-toxic waste
BDCF	Biosphere Dose Conversion Factor
CDF	Cumulative Density Function
CRZ	Containment-providing rock zone
EDZ	Excavation-damaged zone
ENSI	Swiss Federal Nuclear Safety Inspectorate (<i>Eidgenössisches Nuklearsicherheitsinspektorat</i>)
FHAs	Future human actions
FEPs	Features, events and processes
FGR	Fission gas release
HLW	High-level waste (SF + RP-HLW)
IRF	Instant release fraction
JO	Jura Ost siting region
KEG	Nuclear Energy Act (<i>Kernenergiegesetz</i>)
KEV	Nuclear Energy Ordinance (<i>Kernenergieverordnung</i>)
L/ILW	Low- and intermediate-level waste
MOX	Mixed Oxide Fuel
NAB	Nagra Work Report (<i>Nagra Arbeitsbericht</i>)
NEA	Nuclear Energy Agency
NL	Nördlich Lägern siting region
NPP	Nuclear power plant
NTB	Nagra Technical Report (<i>Nagra Technischer Bericht</i>)
OECD	Organisation for Economic Cooperation and Development
OPA	Opalinus Clay
OPC	Ordinary Portland cement
PA	Performance Assessment
PDF	Probability Density Function
RD&D	Research, Development and Demonstration
RH	Relative humidity
RN	Radionuclides
RP-HLW	High-level waste from reprocessing
SF	Spent fuel (assemblies)

SFOE	Swiss Federal Office of Energy (<i>Bundesamt für Energie BFE</i>)
SGT	Sectoral Plan for Deep Geological Repositories (<i>Sachplan geologische Tiefenlager</i>)
THC	Thermo-hydro-chemical
THM	Thermo-hydro-mechanical
URL	Underground research laboratory
ZNO	Zürich Nordost siting region
ZWILAG	Zwischenlager Würenlingen AG

1 Introduction

Aims of the chapter:

- To set the scene: This report is the high-level documentation of the post-closure safety case for a deep geological repository accommodating both Switzerland's high-level waste and low- and intermediate level waste.
- To provide the context, i.e., explain the significance of the post-closure safety case in the context of the general licence application.
- To introduce the structure of this report.

1.1 Context

In Switzerland, legal regulations (the Nuclear Energy Act, KEG 2003 and the Nuclear Energy Ordinance, KEV 2004) and guidelines are in place for the management of radioactive waste. The Nuclear Energy Ordinance (KEV) classifies Switzerland's radioactive waste into low- and intermediate-level waste (L/ILW), alpha-toxic waste¹ (ATW) and high-level waste (HLW), the latter comprising spent fuel (SF) and vitrified HLW from reprocessing (RP-HLW). Currently, this waste is stored safely in interim storage facilities. The Nuclear Energy Act (KEG 2003, Articles 30 and 31) requires the disposal of all types of radioactive waste in deep geological repositories. The feasibility of this disposal solution has been demonstrated for L/ILW in Nagra (1985) and for HLW and long-lived intermediate-level waste (ILW) in Nagra (2002).

The Sectoral Plan for Deep Geological Repositories (SGT) regulates the search for suitable sites for deep geological repositories in Switzerland. This site selection process is conducted in three stages (BFE 2008). Stages 1 and 2, which had the goals of identifying six and then of proposing at least two potential siting regions per repository type, have already been completed and Nagra is currently in Stage 3 (SGT-E3). Clarifications on the safety-related specifications for this stage are provided in ENSI 33/649 (ENSI 2018), while ENSI Guideline G03 and the corresponding explanatory report (ENSI 2023, 2020) specify the protection objective, protection criteria and design principles for deep geological repositories.

In SGT-E3, three siting regions located in Northern Switzerland have been considered – Jura Ost (JO), Nördlich Lägern (NL) and Zürich Nordost (ZNO) – with the objective of selecting one siting region for an HLW repository and one for an L/ILW repository, or alternatively one for a repository for both HLW and L/ILW.

Following a comparison of the siting regions, Nagra concluded that NL is the most suitable and most flexible siting region. Therefore, it is applying for a general licence for a repository for both HLW and L/ILW, with surface facilities at the Haberstal site (municipality of Stadel, ZH).

¹ According to Nagra (2024a), ATW can coexist with L/ILW within the same emplacement caverns. As a result, for the sake of simplicity in the context of deep geological disposal, the term L/ILW encompasses ATW without explicit mention.

The justification for this selection and the safety demonstration for a repository at the site are submitted as part of the general licence application (RBG²). The reports relating to post-closure safety, shown in Fig. 1-1, are structured as follows:

- The overall safety-related argumentation, including post-closure and operational safety, is summarised in an overarching safety report (“Sicherheitsbericht”) NTB 24-01 (Nagra 2025b) shown at the top of Fig. 1-1.
- The upper left side of Fig. 1-1 shows the reports supporting Nagra’s site comparison. The available options are compared with respect to safety and technical feasibility and the results are synthesised in NTB 24-03 (Nagra 2025a, “Bericht zur Begründung der Standortwahl”).
- The upper right side of Fig. 1-1 shows the reports documenting the post-closure safety case for a geological repository for both HLW and L/ILW at the selected site, including its synthesis in the post-closure safety report NTB 24-10, which is the present report and highlighted with a red box.
- The lower middle part of Fig. 1-1 shows a report that presents the current safety and repository concept, including the provisional design.
- The bottom of Fig. 1-1 shows a selection of the many reports that document the assessment basis, which encompasses the methodologies, assessment tools, databases and all other evidence and knowledge developed or acquired in support of the safety assessment.

² From the German *Rahmenbewilligungsgesuch*

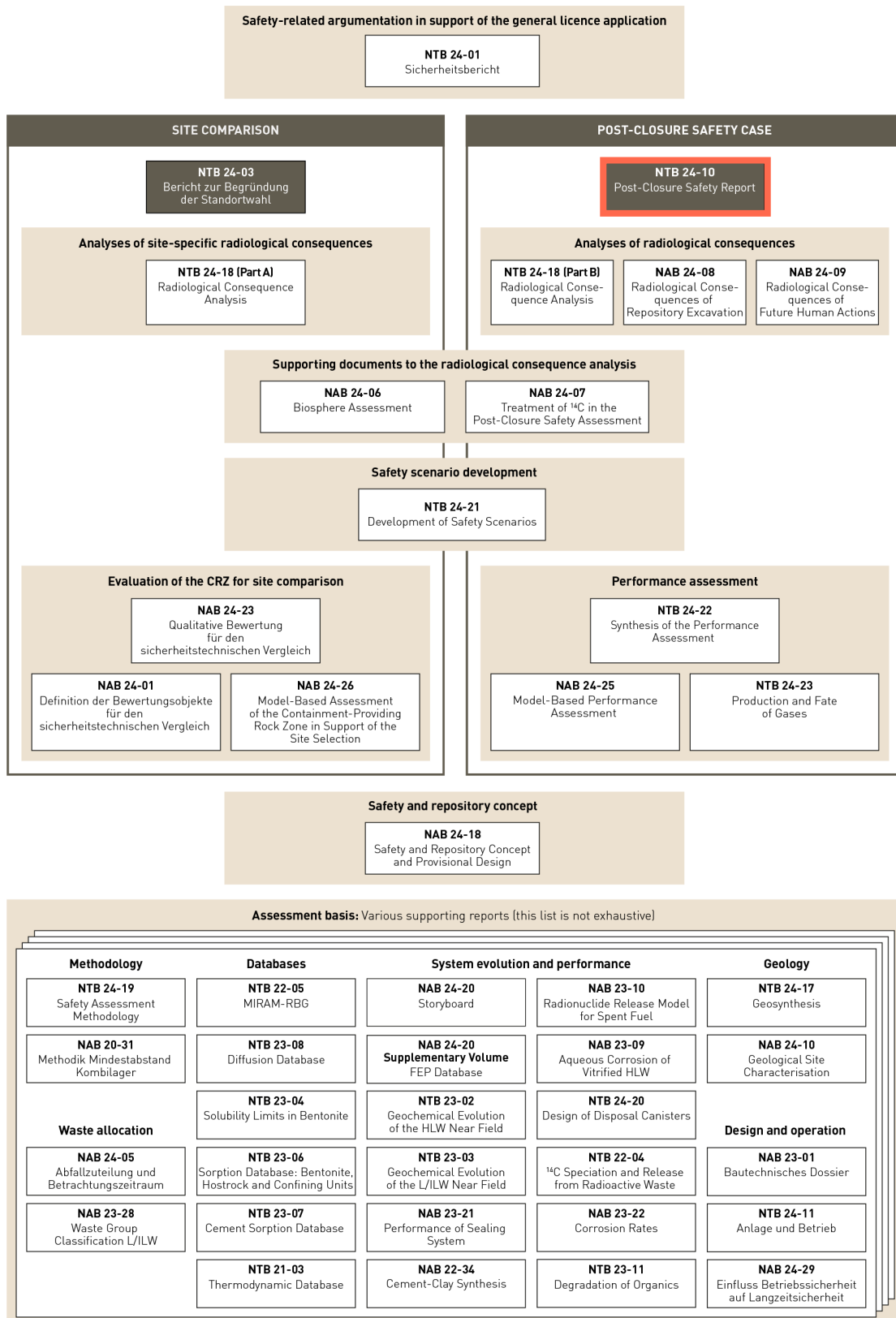


Fig. 1-1: Structure of reports covering post-closure safety aspects of the site comparison and the post-closure safety case

The present report is highlighted with a red box.

1.2 Scope of the present report

The overriding post-closure safety objective of a deep geological repository is “*the long-term protection of humans and the environment from the effects of ionising radiation, without imposing undue burdens and obligations on future generations*” (English version of ENSI Guideline G03, ENSI 2023). Before a disposal system can be implemented at a site, a post-closure safety case must be made that clearly demonstrates that this protection objective is met. This is in accordance with the International Atomic Energy Agency (IAEA) Joint Convention on the Safety of Spent Fuel and on the Safety of Radioactive Waste Management (IAEA 1997), to which Switzerland is a signatory.

Recently, the Swiss waste management programme has mainly been concerned with finding the most suitable site for the disposal of the nation’s radioactive waste. This followed the demonstration of the general feasibility of disposal of L/ILW (Nagra 1985) and of HLW and ILW (Nagra 2002) in Switzerland, its review by the regulatory body (HSK 1986, 2005), and approval by the government (Schweizerischer Bundesrat 1988, 2006). During the siting process, knowledge of the geology within the potential siting regions has significantly increased, design aspects of the repository have been refined, and process understanding has further improved. Safety analyses performed regularly by Nagra have continuously confirmed that the waste can safely be disposed of in Switzerland. The formal demonstration of post-closure safety of a repository in Nagra’s preferred siting region is set out in the present report as an up-to-date safety case.

A comparison of the present safety case with the most recent previous safety case in Switzerland (Nagra 2002), reveals the following similarities and differences (App. A gives a more detailed comparison):

- The same host rock, the Opalinus Clay, a clay rock formation found in large areas of Northern Switzerland, and a broadly similar safety and repository concept are considered.
- Significantly increased knowledge is now available for the siting region, due to an extensive geological characterisation programme (see, e.g., Chapter 7 of the Geosynthesis of Northern Switzerland, NTB 24-17, Nagra 2024i).
 - The favourable properties of the Opalinus Clay have been confirmed and uncertainty in many of the parameter values used in safety assessment (e.g., diffusivity, sorption capacity) have been reduced.
 - The favourable properties of the confining geological units and the hydrogeological situation in the siting region have been characterised in detail and can now be relied upon in the safety case.
 - The proposed depth of the repository has been confirmed as sufficient to efficiently protect the repository from long-term surface processes such as deep glacial erosion.
- Geomechanical properties of the host rock have been re-assessed (NAB 24-10 Rev. 1, Nagra 2024h) and the technical feasibility of constructing and operating the repository has been confirmed with ample robustness at the depth under consideration (NAB 23-01 Vol. 1 – 9, Nagra 2023a).

- Evolution of the repository design
 - Rather than two separate repositories for HLW and L/ILW, as considered in Nagra (2002) and earlier work, a provisional design has been developed for a repository for both HLW and L/ILW, with common access structures and separate repository sections for HLW and L/ILW. A repository for both HLW and L/ILW has been shown to meet the operational and post-closure safety requirements, as would two separate repositories, but has clear advantages in terms of construction-related risks, resource consumption and greenhouse gas emission, as fewer facilities have to be constructed and operated (Nagra 2021c).
 - The provisional design of the repository and its components considers recent technical engineering developments. This allows, e.g., the emplacement drifts and the rock support to be optimised.

The aim of the present post-closure safety report, which is the synthesis of the post-closure safety case, is to demonstrate concisely the long-term (post-closure) safety of the planned deep geological repository for both HLW and L/ILW. In other words, the report presents the outcome of a safety assessment, which is the means by which the safety case is developed. In the context of this report, safety assessment is defined by Nagra as the process of gathering all relevant evidence and arguments and carrying out analyses for the safety of the disposal system during the post-closure phase.

1.3 Structure of the present report

As a synthesis, the present report is underpinned by numerous supporting reports (see Fig. 1-1), as described in the context of the following report structure:

Following this introduction in **Chapter 1**, the contextual and regulatory framework guiding the development of the safety case is summarised in **Chapter 2**. **Chapter 3** provides information on the current **safety and repository concept** and **provisional repository design**, for which the safety case has been developed. These two chapters summarise report NAB 24-18 Rev. 1 (Nagra 2024s) on the current safety and repository concept and provisional design which describes in more detail the safety requirements related to design and how different engineered and geological barriers contribute to system safety by means of the safety functions they perform. The **methodology** adopted for the safety assessment, which provides the central elements of the safety case, is described in **Chapter 4**. It is based on the extensive description of the safety assessment methodology in NTB 24-19 (Nagra 2024t) and the methodological sections within a set of reports that describe the main processes of safety assessment, together with their outcomes, in more detail.

Chapter 5 then summarises the remainder of the **assessment basis**, where the assessment basis as a whole is defined as the evidence, knowledge, assessment tools, and methodologies developed or acquired by Nagra in support of the safety assessment. Key aspects of the assessment basis, besides the methodology, are the integrated understanding of the geology of Northern Switzerland as detailed in the Geosynthesis of Northern Switzerland in NTB 24-17 (Nagra 2024i), the phenomenological understanding of the expected evolution of the repository described in NAB 24-20 Rev. 1 (Nagra 2024m) and the understanding of the retention, release and migration of radionuclides (several reports; see Fig. 1-1). Finally, the assessment basis also contains a features, events and processes (FEPs) database in a supplementary volume of NAB 24-20 Rev. 1 (Nagra 2024l) and the many models, codes and databases used in safety assessment (grouped by topic and described in the corresponding reports; see Fig. 1-1).

Chapters 6, 7 and 8 are dedicated to three of the central processes within safety assessment, namely:

- The **performance assessment** comprising an analysis of the thermal, hydraulic, mechanical, and chemical evolution of the repository system. Details are given in the synthesis report NTB 24-22 Rev. 1 (Nagra 2024u), in NAB 24-25, which describes in more detail the modelling work carried out for the performance assessment (Nagra 2024k) and NTB 24-23 on the production and fate of gases within the repository (Nagra 2024o).
- **Safety scenario development**, which entails the identification and description of a set of safety scenarios that capture uncertainty in the broad ways in which the disposal system can evolve over time, considering all relevant sources of uncertainty, as well as the identification of additional “what-if?” cases to test system robustness, as described in detail in NTB 24-21 (Nagra 2024e).
- **Analysis of radiological consequences**, which consists of an evaluation of radionuclide release rates to the surface environment as a function of time for the various safety scenarios and “what-if?” cases. The main results for the present safety case are detailed in Part B of NTB 24-18 (Nagra 2024p), and complemented by analyses linked to scenarios of future human actions in NAB 24-09 (Nagra 2024r) and the excavation of the repository by erosion processes in NAB 24-08 Rev. 1 (Nagra 2024q). Furthermore, the treatment of ^{14}C , which differs in some aspects from that of other radionuclides, is set out in NAB 24-07 Rev. 1 (Nagra 2024w). The fate of radionuclides that reach the surface environment and can ultimately contribute to the dose rate is discussed in detail in the biosphere report NAB 24-06 (Nagra 2024n).

In **Chapter 9**, **complementary lines of argument** are made, before all the evidence, arguments and analyses from the previous chapters are brought together in **Chapter 10** to **demonstrate safety**.

Since the development of the safety case is a prolonged and iterative process, an **outlook for future developments** is given in **Chapter 11**.

Chapter 12 provides a list of references, and, in **Chapter 13**, a glossary is presented.

App. A highlights the main commonalities and differences of the present post-closure safety case compared with the safety case developed for Project *Entsorgungsnachweis*, while **App. B** summarises the key phenomena identified in performance assessment, their relevance to system evolution and performance, and their potential to lead to performance deviations. Finally, **App. C** considers non-radioactive but potentially hazardous materials that are chemically toxic or could pose a risk to water quality, which can also be safely contained in a deep geological repository.

2 Contextual and regulatory framework

Aims of the chapter:

- To recall that deep geological disposal is, by international consensus, a suitable long-term management option for (high-level) radioactive waste.
- To set out international principles and guidance, together with national laws and regulations, which form the context and framework for the post-closure safety case.

2.1 Basis for geological disposal as a waste management option

Radioactive waste needs to be managed (see, e.g., SSR-5, IAEA 2011a). It can, in principle, be safely stored in surface facilities. However, because these facilities need maintenance and care, their safety is inevitably reliant on continued societal stability. Societal stability may, for example, be linked to political or financial stability, uncertainties which are far greater than those associated with the evolution of conditions deep underground in geological formations that are suitable to host a repository.

As a long-term waste management option, deep geological disposal has the positive attribute that, if the site and design are chosen appropriately, neither the societal stability needed for governmental and regulatory control, nor any specific controls on activities occurring at the surface of the Earth are prerequisites for post-closure safety. In fact, geological disposal provides long-term passive safety. Passive safety means that safety does not require any active measure to be undertaken following closure of the repository. Thus, no burden is placed on future generations to maintain and control the repository once it has been closed, although such control is certainly possible if society decides in favour of it.

Geological disposal contains the waste for a prolonged period while its associated radioactivity decays naturally. It is designed to prevent radioactivity from reaching the surface environment in amounts that could cause harm to humans or the environment. Placing the waste in a deep rock formation also enhances security in that it reduces the possibility of sabotage or theft.

The proof of existence of suitable rock formations, together with the findings of safety assessments conducted for a wide range of sites and designs world-wide (see, e.g., OECD/NEA 2017), lend support to the feasibility of safe geological disposal.

2.2 International principles and guidance

Comprehensive international principles and guidelines for dealing with radioactive substances and for deep geological disposal establish a robust framework for the Swiss waste management programme and are presented in detail in NAB 24-18 Rev. 1 (Nagra 2024s). In the following, the most relevant aspects are summarised.

The International Commission on Radiological Protection (ICRP) has contributed significantly to the framework with its system of radiation protection, updated in ICRP Publication 103 (ICRP 2007b). This system is built on the principle of dose limitation. Furthermore, ICRP Publication 81 (ICRP 1998) emphasises that future generations should receive at least the same level of protection from radioactive waste as the current generation. This principle involves applying current quantitative dose and risk criteria to estimated future doses or risks for appropriately defined critical groups.

At a high level, safety principles and provisions are laid down in the International Atomic Energy Agency (IAEA) Joint Convention on the Safety of Spent Fuel and on the Safety of Radioactive Waste Management (IAEA 1997), to which Switzerland is a signatory. Safety standards and safety guides establish more detailed guidelines and requirements: IAEA Safety Standard SF-1 (IAEA 2006) underlines the fundamental safety objective to protect humans and the environment from harmful effects of ionising radiation. This standard articulates ten associated safety principles that provide the foundation for requirements and measures for the protection of humans and the environment against radiation risks and for the safety of facilities and activities generating those risks. More specific safety guides are concerned with the safe development of deep geological repositories, referred to here as geological disposal facilities (SSG-14, IAEA 2011b) and with the safety case and safety assessments (SSG-23, IAEA 2012). Furthermore, specific safety requirements are given in SSR-5 (IAEA 2011a).

Internationally recognised principles that underlie the disposal concepts and designs for deep geological disposal include passive safety and a stepwise and iterative development of the repository for the optimisation of protection. While many detrimental phenomena and uncertainties can be avoided, reduced, or their effects mitigated by optimisation, some uncertainty will inevitably remain. It is by following the principle of robustness that geological disposal systems are sited and designed in such a way that post-closure safety can be demonstrated irrespective of the remaining uncertainty. The Nuclear Energy Agency (NEA) of the Organisation for Economic Cooperation and Development (OECD) has stated that *“robust systems are characterised by a lack of complex, poorly understood or difficult to characterise features and phenomena, demonstrated quality control, and an absence of, or relative insensitivity to, detrimental phenomena arising either internally within the repository and host rock, or externally in the form of geological and climatic phenomena that introduce processes with the potential to compromise safety”* (OECD/NEA 2013).

Regarding the design of a disposal facility, IAEA SSR-5 (IAEA 2011a) states that *“the disposal facility and its engineered barriers shall be designed to contain the waste with its associated hazard, to be physically and chemically compatible with the host geological formation and/or surface environment, and to provide safety features after closure that complement those features afforded by the host environment”*. Safety is provided by means of multiple safety functions, which provide **defence-in-depth**, according to IAEA SSR-5 (IAEA 2011a):

- *“the host environment shall be selected, the engineered barriers of the disposal facility shall be designed and the facility shall be operated to ensure that safety is provided by means of multiple safety functions”*,
- *“a safety function may be provided by means of a physical or chemical property or process that contributes to containment and isolation, such as: impermeability to water; limited corrosion, dissolution, leach rate and solubility; retention of radionuclides; and retardation of radionuclide migration”* and
- *“adequate defence-in-depth has to be ensured by demonstrating that there are multiple safety functions, that the fulfilment of individual safety functions is robust and that the performance of the various physical components of the disposal system and the safety functions they fulfil can be relied upon, as assumed in the safety case and supporting safety assessment”*.

Thus, the safety case is built on multiple types of evidence and lines of argument for the performance of individual barriers, including their contributions to the safety functions, and for the performance and robustness of the disposal system as a whole.

2.3 International developments

Besides following international principles and guidance (see Section 2.2 above), Switzerland's waste management programme benefits from the experience gained from work for deep geological repositories abroad. Of particular interest are more advanced programmes that also work towards disposal in clay rock. France, for example, recently applied for the construction licence for a repository in Callovo-Oxfordian clay rock. Canada is currently investigating two siting regions and host rocks, one of which is an argillaceous limestone in southern Ontario and the other a crystalline batholith in northwestern Ontario. Finland and Sweden are the most advanced with respect to the implementation of deep geological repositories, which are hosted in crystalline rock. Relevant work relating to post-closure safety assessment carried out by these programmes is listed in Tab. 2-1.

Tab. 2-1: Some recent safety cases for deep geological disposal, their main objectives in terms of the project milestones they support and main aspects with relevance to the present safety case

Organisation, country	Name Main objectives and main aspects of relevance	Reference
Andra, France	DAC Preliminary safety report supporting the construction licence application (original title: Dossier d'autorisation de création de l'installation nucléaire de base (INB) CIGEO. Pièce 7: Version préliminaire du rapport de sûreté) Repository for high-level and long-lived waste in Callovo-Oxfordian clay host rock with a similar multi-barrier system	(Andra 2022)
Andra, France	DOS-AF Summary of knowledge related to post-closure safety in view of finalising the pre-project for the construction licence application (original title: Dossier d'option de sûreté. Partie après fermeture) Repository for high-level and long-lived waste in Callovo-Oxfordian clay host rock with a similar multi-barrier system	(Andra 2006)

Tab. 2-1: Cont.

Organisation, country	Name Main objectives and main aspects of relevance	Reference
NWMO, Canada	TR-2023-08 Confidence in Safety – South Bruce Site Demonstrating suitability of the site to host a repository for spent fuel Repository for high-level waste in argillaceous (shaley) limestone with a partially similar multi-barrier system	(NWMO 2023)
Posiva, Finland	SC-OLA Safety case for the operating licence application Safety concept and design development process with link to requirements with a partially similar multi-barrier system	(Posiva 2021)
SKB, Sweden	TR 21-01, PSAR Post-closure safety for the final repository for spent nuclear fuel at Forsmark Support for the construction licence application Safety concept and design development process with link to requirements, with a partially similar multi-barrier system	(SKB 2022)
SKB, Sweden	TR 23-01, PSAR Post-closure safety for SFR, the final repository for short-lived radioactive waste at Forsmark Support for the application for an extension of the repository Safety concept and design development process with link to requirements, with a partially similar multi-barrier system	(SKB 2023)

2.4 Regulatory requirements and guidance

Requirements for the documentation of the general licence application and for the final stage of the site selection process are set out in ENSI 33/649 (ENSI 2018). Chapter 5 of that document states that, for the safety case supporting the general licence application, the safety analysis carried out for the proposed site in the site comparison is to be supplemented by a comprehensive scenario analysis and analysis of radiological consequences. Excavation of the repository by erosive processes is specifically mentioned as a scenario to be evaluated.

The protection objective and the criteria by which the post-closure safety of a deep geological repository is to be evaluated are those set out in ENSI Guideline G03 (ENSI 2023). The objective of deep geological disposal is to ensure long-term protection of humans and the environment by means of a system of staged, passively functioning engineered and geological barriers, termed the multi-barrier system. It is acknowledged that absolute containment of all radioactive substances over very long periods of time is impossible, and the multi-barrier system has therefore to be designed in such a way that the release of radionuclides through the engineered and geological barriers to the biosphere remains so low that the protection of humans and the environment is ensured. Specific elements to be addressed in the safety assessment, according to ENSI Guideline G03, are presented in Tab. 2-2, along with the sections in the present report where each element is covered. Further requirements for safety assessment and the safety case are also set out in the guideline, and their implementation in the present safety assessment methodology is described in NTB 24-19 (Nagra 2024t).

Tab. 2-2: Elements to be addressed in safety assessment, according to ENSI (2018) and sections in the present report and key supporting reports where these are covered

	Element	Section in this report
A	Description of the deep geological repository system.	Repository concept and design in Sections 3.2 and 3.3; geological setting in Section 5.2.1
	<i>Beschreibung des geologischen Tiefenlagersystems.</i>	
B	Use of verified data on the geological conditions at the site.	Data on geological conditions summarised in Section 5.2.1 (see the Geosynthesis of Northern Switzerland, NTB 24-17, Nagra 2024i, for full description)
	<i>Verwendung verifizierter Daten zu den geologischen Gegebenheiten am Standort.</i>	
C	Demonstration of the function, retention capacity and robustness of the engineered and geological barriers.	Performance assessment in Chapter 6
	<i>Aufzeigen der Wirkungsweise, des Rückhaltevermögens und der Robustheit der technischen und natürlichen Barrieren.</i>	
D	Description and evaluation of the effects of coupled processes and of gas formation and dispersion on the engineered and geological barriers and on radionuclide transport.	Assessment of total system, as part of the performance assessment, in Section 6.2
	<i>Darlegung und Bewertung der Auswirkungen gekoppelter Prozesse und der Gasbildung und -ausbreitung auf die technischen und natürlichen Barrieren sowie auf den Radionuklidtransport.</i>	

Tab. 2-2: Cont.

	Element	Section in this report
E	Description of the expected long-term geological evolution.	Long-term geological evolution described in Section 5.2.2
	<i>Beschreibung der zu erwartenden geologischen Langzeitentwicklung.</i>	
F	Description of the expected development of the materials contained in the deep geological repository, including the radioactive waste and the engineered and natural barriers.	Expected evolution of the repository system described in Section 5.3; fate of radionuclides described in Section 5.4
	<i>Beschreibung der zu erwartende Entwicklung der im geologischen Tiefenlager befindlichen Materialien, einschliesslich der radioaktiven Abfälle und der technischen und natürlichen Barrieren.</i>	
G	Scenario analysis and definition of the calculation cases used to investigate the evolution of the deep geological repository.	Safety scenario development and definition of calculation cases described in Chapter 7
	<i>Szenarienanalyse und Festlegung der Rechenfälle, mit denen die zu betrachtenden Entwicklungen des Tiefenlagers untersucht werden.</i>	
H	Justification of why the assumptions and calculation models used are applicable to the situation under consideration.	Validation of codes and databases used in performance assessment described briefly in Section 6.3 (see NTB 24-22 Rev. 1, Nagra 2024u for details). See Appendix A of NTB 24-18 (Nagra 2024p) for justifications of assumptions and simplifications used in the analysis of radiological consequences
	<i>Begründung, warum die verwendeten Annahmen und Rechenmodelle auf die vorliegende Situation anwendbar sind.</i>	
I	Systematic sensitivity and uncertainty analysis to determine the influence of uncertainties in the data, processes, and models on the calculation results.	Described in the context of performance assessment in Sections 6.2.4 and 6.3. Described in the context of the analysis of radiological consequences of the reference safety scenario in Section 8.2
	<i>Systematische Sensitivitäts- und Unsicherheitsanalyse zur Ermittlung des Einflusses von Unsicherheiten in den Daten, Prozessen und Modellen auf die Berechnungsergebnisse.</i>	
J	Consideration of envelope scenarios ³ covering possible lines of evolution, particularly regarding surface morphology and climate, within the framework of biosphere modelling.	Safety scenario development described in Chapter 7. Noted in Section 7.2.1 that biosphere dose conversion factors are used to convert the radionuclide release rate to dose rates by assuming different enveloping biospheres with respect to climate and geomorphology
	<i>Betrachtung umhüllender Szenarien von möglichen Entwicklungen, insbesondere der Gebietsmorphologie und des Klimas im Rahmen der Biosphärenmodellierung.</i>	

³ Nagra uses the term “safety scenarios” in the present safety assessment and safety case. As these safety scenarios are broad descriptions of the initial state and subsequent evolution of the repository that each cover a range of different, though similar, lines of evolution, the safety scenarios can be regarded as “envelope scenarios”.

Quantitative protection criteria in the form of individual effective dose and risk limits are specified in the ENSI Guideline G03 (ENSI 2023). These are as follows:

- Any future evolution of a deep geological repository must not lead to the release of radionuclides causing an individual dose exceeding 0.1 mSv per year or not cause the risk value according to criterion (b) in paragraph 2.15 of IAEA Safety Standard SSR-5 (IAEA 2011a) to be exceeded⁴.

The IAEA Safety Standard SSR-5 gives a risk constraint of the order of 10^{-5} per year, where risk is understood as the probability of fatal cancer or serious hereditary effects.

- During the assessment period, the radiological consequences of inadvertent human intrusion into the deep geological repository have to be assessed on the basis of criteria (c) and (e) as set out in paragraph 2.15 of IAEA Safety Standard SSR-5⁵.

Here, SSR-5 states that, if intrusion leads to a calculated annual dose of less than 1 mSv to those living around the site, “... then efforts to reduce the probability of intrusion or to limit its consequences are not warranted” whereas, if the calculated annual dose is in the range 1 – 20 mSv, “... then reasonable efforts are warranted at the stage of development of the facility to reduce the probability of intrusion or to limit its consequences by means of optimisation of the facility’s design”.

- After the end of the time period for assessment (see Section 4.3), the effects on the surface must not be significantly higher than the average current radiation exposure of the Swiss population⁶.

Compliance with the protection criteria for the post-closure phase has to be shown in a safety case. Protection criteria for the operational phase are also set out in ENSI Guideline G03 (ENSI 2023). These are addressed in the separate report NAB 24-02 (Nagra 2024f).

⁴ Für keine zukünftige Entwicklung eines Tiefenlagers darf die Freisetzung von Radionukliden zu einer Individualdosis grösser als 0.1 mSv pro Jahr oder zu einer Überschreitung des Risikorichtwerts gemäss Kriterium (b) in Absatz 2.15 des IAEA Safety Standard SSR-5 führen.

⁵ Im Nachweiszeitraum sind die radiologischen Folgen eines unbeabsichtigten menschlichen Eindringens ins Tiefenlager anhand der Kriterien (c) und (e) gemäss Absatz 2.15 des IAEA Safety Standard SSR-5 zu beurteilen.

⁶ Nach Ende des Nachweiszeitraums dürfen die Auswirkungen an der Oberfläche nicht wesentlich höher sein als die durchschnittliche heutige Strahlenexposition der Schweizer Bevölkerung.

3 Current safety and repository concept and provisional repository design

Aims of the chapter:

- To set out a safety and repository concept that is consistent with legal requirements as well as national and international principles.
- To define safety functions and describe how they are provided by reliable and well understood pillars of safety.
- To describe a provisional design suitable for the present safety case.
- To explain how post-closure safety of a deep geological repository relies on a balanced interplay between the chosen site, the safety concept and design.

3.1 Concept and design development workflow

The workflow for the development of a safety and repository concept and a repository design, which then leads to a provisional design and implementation plan, is illustrated on the left-hand side of Fig. 3-1 and described briefly in the following sections. The figure also shows, on the right-hand side, the workflow for safety assessment as described in later chapters of this report.

The starting point is the overarching protection objective, as well as design principles and protection criteria prescribed by Swiss regulations and national and international principles. These underlie a high-level safety and repository concept for deep geological disposal, which in turn provides a basis for an iterative design development.

In Nagra's terminology, the safety concept explains how, through the safety barriers and safety functions they provide, the repository performs its overarching safety functions and ensures the protection of humans and the environment. The repository concept describes the individual safety barriers and their properties, including interactions. The safety concept and repository concept are thus closely intertwined and are referred to collectively as the safety and repository concept. Nagra's current safety and repository concept has been iteratively developed and is now considered well understood. The current concept, though mature, is still provisional and offers flexibility for future adaptations.

By contrast, the more detailed design and implementation plan is provisional and specifically developed for the purposes of the general licence application. As also shown in Fig. 3-1, it represents the current outcome of an ongoing iterative process entailing:

- the identification and development of a set of design requirements to be met, including, but not limited to, those related to post-closure safety,
- the design development itself, whereby solutions are proposed and refined to meet these design requirements in a manner that is compatible with site-specific conditions, feasible to implement with proven technology, and that avoids unnecessary complexity and excessive cost, and
- analyses of the system as a whole and its various subsystems to test whether the requirements are met and to guide the further modification of the design.

In summary: the protection objective, protection criteria, safety functions, component-specific functions, performance objectives and design requirements together constitute a hierarchy of safety requirements that guide design development.

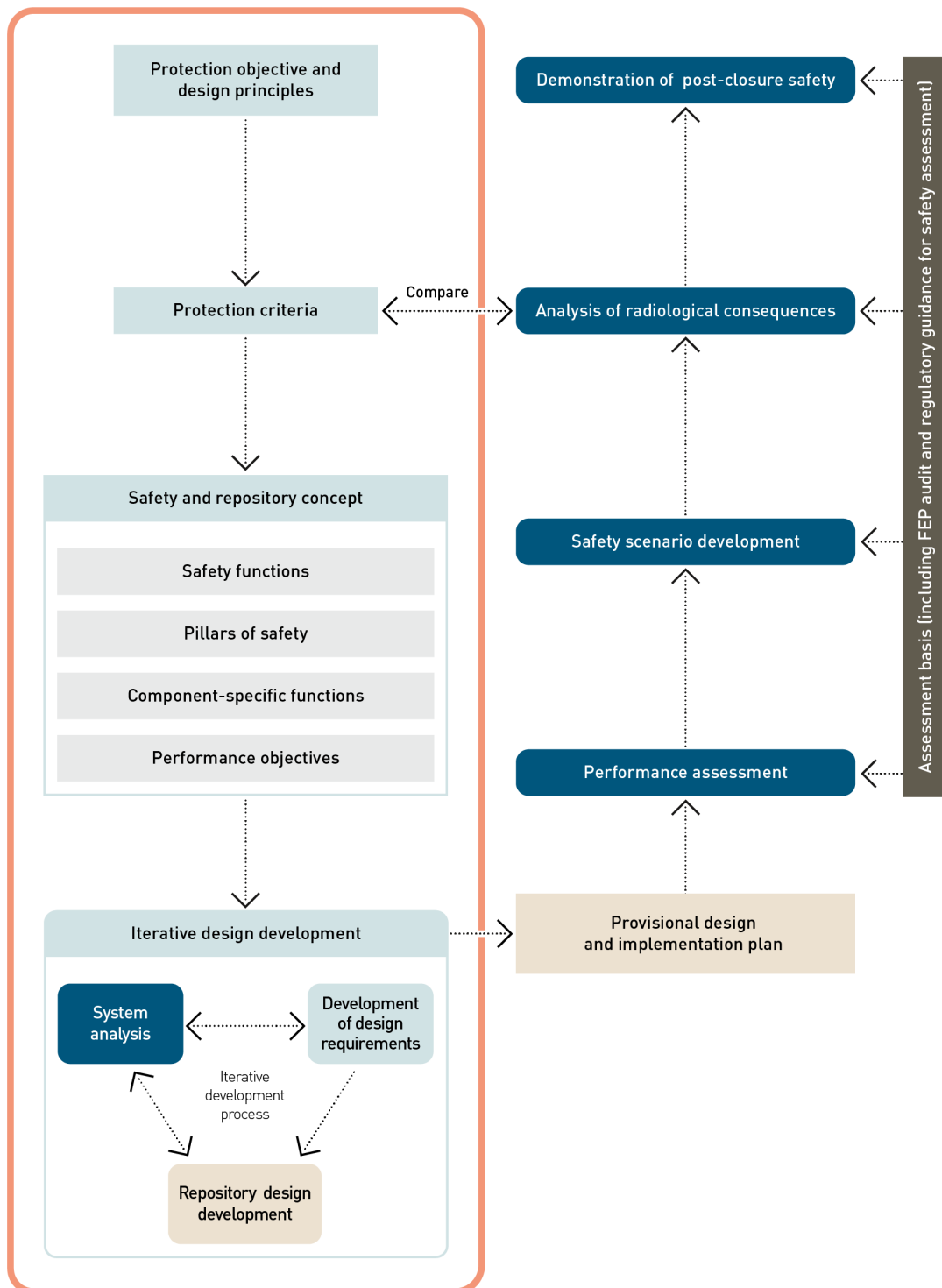


Fig. 3-1: Workflow for the post-closure safety case, highlighting, in the orange box, the process for developing the current safety and repository concept and a repository design, which then leads to a provisional design and implementation plan

3.2 Current safety and repository concept

The current safety and repository concept is discussed in detail in NAB 24-18 Rev. 1 (Nagra 2024s). In the following, we summarise the main aspects with relevance to post-closure safety.

3.2.1 Overview

In line with the Swiss legal and regulatory framework, as well as national and international principles and practice (see Chapter 2), the repository provides safety by means of multiple geological and engineered barriers, i.e., it is a multi-barrier system. These barriers operate passively, meaning that, after closure of the repository, no further measures, such as maintenance, control or monitoring, are required to ensure post-closure safety. This relieves future generations of the burden of having to manage or monitor the repository. The individual barriers contribute to so-called safety functions of the repository system, which together ensure a robust system, as discussed below in Sections 3.2.2 and 3.2.3.

Waste is disposed of in waste emplacement rooms, located deep underground within the Opalinus Clay, between its confining geological units (Fig. 3-2). The innermost engineered barriers for HLW disposal consist of spent fuel (SF) assemblies with fuel rods (UO₂ or mixed oxide fuel (MOX) pellets in cladding tubes) and vitrified HLW from reprocessing of SF (RP-HLW), packaged in steel canisters (NTB 24-20, Nagra 2024d). These HLW disposal canisters are emplaced in drifts, which are subsequently backfilled with a buffer material and sealed. For L/ILW disposal, waste packages for L/ILW are placed inside disposal containers, which are in turn emplaced in caverns that are backfilled with cement-based mortar. The inventory of the waste to dispose of is summarised in Section 5.3.1 and described in detail in NTB 22-05 (Nagra 2023b).

The geological barrier is of primary importance due to the excellent qualities and long-term geological stability of the containment-providing rock zone (CRZ) (Chapter 5 in NTB 24-17, Nagra 2024i). The CRZ includes the Opalinus Clay host rock, as well as the geological units above and below the host rock with similarly favourable properties. The CRZ is delimited in the course of the site selection process according to ENSI 33/649 (ENSI 2018) and documented in NAB 24-01 Rev. 1 (Nagra 2024c).

The Opalinus Clay formation is characterised by:

- A fine and homogeneous pore structure with a very low hydraulic conductivity. This ensures that water movement in this clay formation is limited, and that radionuclide migration is essentially diffusion-dominated.
- A high clay-mineral content. This results in a high sorption capacity for many radionuclides, thereby delaying their migration through the clay formation.
- A high self-sealing capacity. This means that any fractures that may form in the clay formation will rapidly close.

The geological units above and below the Opalinus Clay, albeit with a lower or more heterogeneous clay-mineral content, contribute further to the retardation and slow transport of radionuclides.

The depth of the geological barrier is sufficient to limit any potential impact of events and processes at the surface, including erosion, on the safety-related properties of the Opalinus Clay itself and its confining geological units.

The geological 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. Such disturbances are 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 a period of complete containment for at least as long as the heat generated by the waste has significant potential to affect transport processes in the surrounding rock. Eventual release of radionuclides and migration through the multi-barrier system cannot, however, be excluded, although most radionuclides will remain immobilised in the waste packages and waste containers until they decay (see Section 5.4).

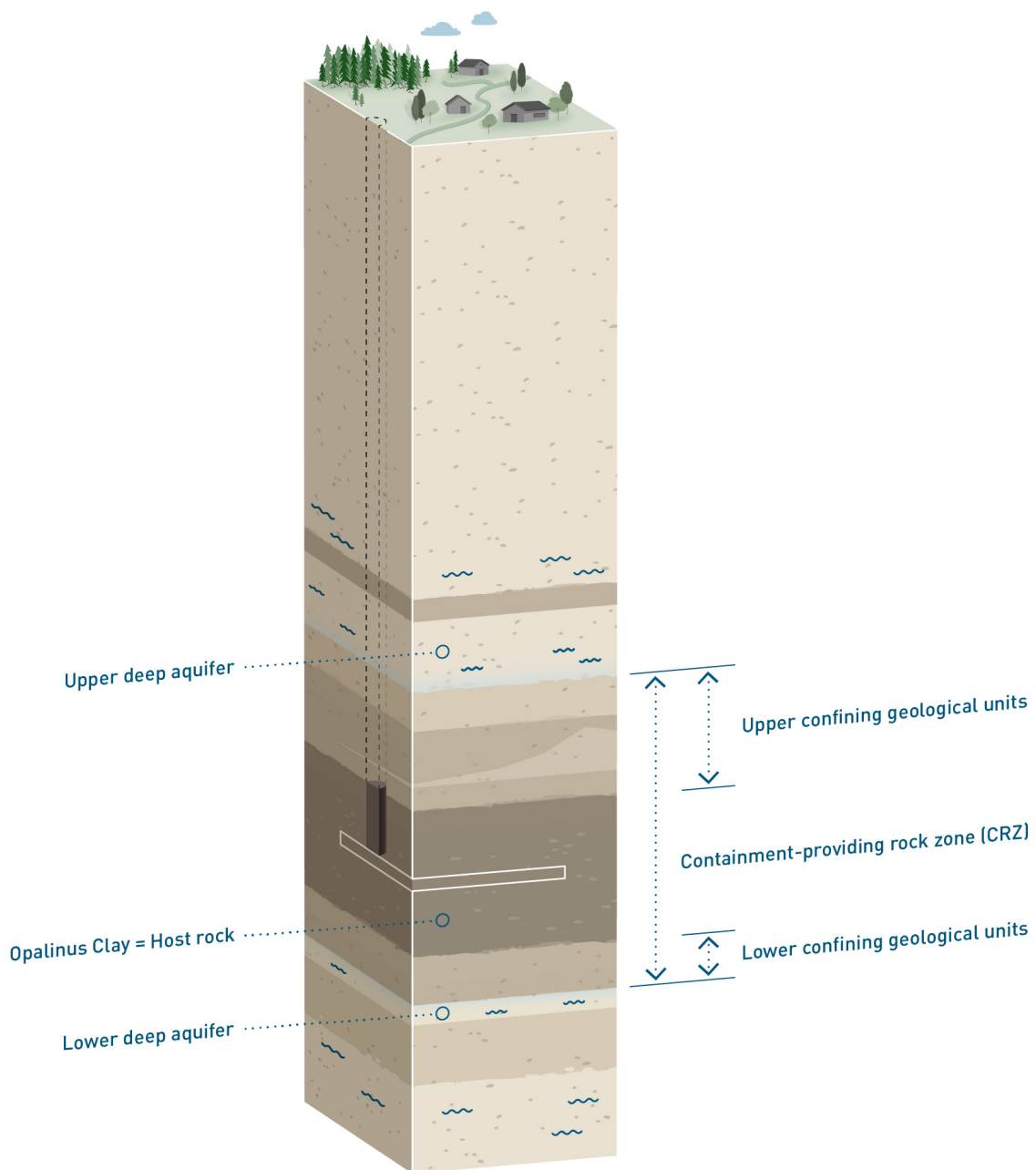


Fig. 3-2: Placement (white rectangle) of the repository deep underground within the Opalinus Clay, between its confining geological units

Furthermore, the radionuclides 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 migrate through these barriers, within the surrounding geological barrier.

The location of the deep geological repository will be such that the risk of future inadvertent human intrusion is small. Nonetheless, inadvertent human intrusion or disturbance of the repository in the distant future cannot be completely excluded, and thus repository design features that mitigate their potential impact also form part of the current safety concept.

3.2.2 Safety functions and related safety requirements

Collectively, the multi-barrier system provides the following safety functions:

- S1: Isolation** of radioactive waste from humans and the environment;
- S2: Complete containment** of radionuclides for a period of time;
- S3: Immobilisation, retention, and slow release** of radionuclides;
- S4: Compatibility** of the components of the repository system;
- S5: Long-term stability** of the multi-barrier system with regard to long-term geological and climatic evolution.

These safety functions are used to derive a set of safety requirements on the system components, including the “pillars of safety” described below. In the current repository design the safety functions are provided by the elements of the multi-barrier system. Future changes to the system remain possible as there is no single solution for how to provide these safety functions.

Performance assessment, carried out as part of the safety assessment, analyses whether the barriers perform their various functions and thus fulfil their assigned safety functions. Performance assessment is described further in Chapter 6.

3.2.3 Pillars of safety

The repository barriers and their properties that are the main contributors to these safety functions are termed “pillars of safety”. They are well understood and insensitive to perturbations. The seven pillars of safety are:

- The **deep underground location of the waste emplacement rooms in a stable geological environment** that lacks underground resources specific to the region that could attract future human exploitation. It ensures that the repository is not susceptible to geological events and processes relevant to post-closure safety and that inadvertent human intrusion remains unlikely.
- The **containment-providing rock zone (CRZ)** includes the host rock as well as upper and lower confining geological units. The host rock, Opalinus Clay, is characterised by a fine and homogeneous pore structure, extremely low hydraulic permeability, a homogeneous and high clay-mineral content and the associated high sorption capacity for many radionuclides. The CRZ is characterised by favourable properties (especially self-sealing ability) for long-term stability, forming a very effective and stable barrier against radionuclide transport and a compatible (physical and chemical) environment for the engineered barriers (NAB 24-18 Rev. 1, Nagra 2024s).
- Both the **SF and RP-HLW matrices** are chemically stable in the expected environment.

- The **SF and RP-HLW disposal canisters** are mechanically stable and corrosion-resistant in the expected environment. They ensure complete containment of radionuclides for at least as long as the heat generated by the waste has significant potential to affect transport processes in the surrounding barriers. They satisfy the regulatory requirement of complete containment for at least 1,000 years (ENSI 2023).
- The HLW emplacement drifts will be backfilled with a **buffer**. This buffer serves as a suitable physical and chemical environment that enhances the longevity of the HLW disposal canisters. Additionally, the buffer offers structural support to the surrounding rock after the degradation of the tunnel support. It compartmentalises the repository by isolating the disposal canisters from one another, thereby mitigating the impact of accidental human intrusion. The buffer reduces the potential effects of heat release on the safety-relevant properties of the CRZ and acts as a barrier against radionuclide transport.
- The **L/ILW cementitious near field** promotes sorption of radionuclides, decreases corrosion of metals and inhibits degradation of organic materials, which in turn reduces gas generation. The cement-based backfill can also store a significant volume of gas and provides mechanical stability to the host rock, even after the degradation of the tunnel support.
- The **closure system** (backfill and seals) provides retention of radionuclides released from the waste packages, and a transport and storage system for repository-generated gas, which retains gaseous radionuclide species within the underground structures while limiting the build-up of gas pressure. The sealing of both the individual waste emplacement rooms and the HLW and L/ILW repository sections reduces safety-relevant interaction between the repository sections and the radiological impact of inadvertent human intrusion. The closure system further provides mechanical stabilisation of the host rock after the rock support has degraded.

Each pillar of safety can contribute directly and/or indirectly to the safety functions. For example, the HLW disposal canisters contribute directly to the containment of radionuclides, while the surrounding buffer contributes indirectly as it helps to protect the canisters. Compatibility between the different pillars of safety is based upon a solid technical and scientific understanding of the geological and engineered components (see Chapter 5).

The contributions of these seven pillars of safety to the five safety functions are summarised schematically in Fig. 3-3 and Fig. 3-4 and are elaborated in the discussion of component-specific functions in Section 3.4 of NAB 24-18 Rev. 1 (Nagra 2024s).

SF and RP-HLW matrix

- contributes to the **immobilisation, retention, and slow release** of radionuclides.
- is **compatible** with the components of the repository system.

SF and RP-HLW disposal canisters

- contribute to the **complete containment** of radionuclides for a period of time.
- support the **immobilisation, retention, and slow release** of radionuclides.
- are **compatible** with the components of the repository system.

Buffer of the HLW emplacement drifts

- contributes to the **immobilisation, retention, and slow release** of radionuclides.
- is **compatible** with the components of the repository system.

Deep underground location in a stable geological environment

- contributes to the **isolation** of radioactive waste from humans and the environment.
- contributes to the **long-term stability** of the repository system with regard to long-term geological and climatic evolution.
- supports the **immobilisation, retention, and slow release** of radionuclides.

Containment-providing rock zone (CRZ)

- contributes to the **isolation** of radioactive waste from humans and the environment.
- contributes to the **immobilisation, retention, and slow release** of radionuclides.
- contributes to the **long-term stability** of the repository system with regard to long-term geological and climatic evolution.
- is **compatible** with the components of the repository system.

Closure system (backfill and seals)

- contributes to the **isolation** of radioactive waste from humans and the environment.
- contributes to the **immobilisation, retention, and slow release** of radionuclides.
- is **compatible** with the components of the repository system.

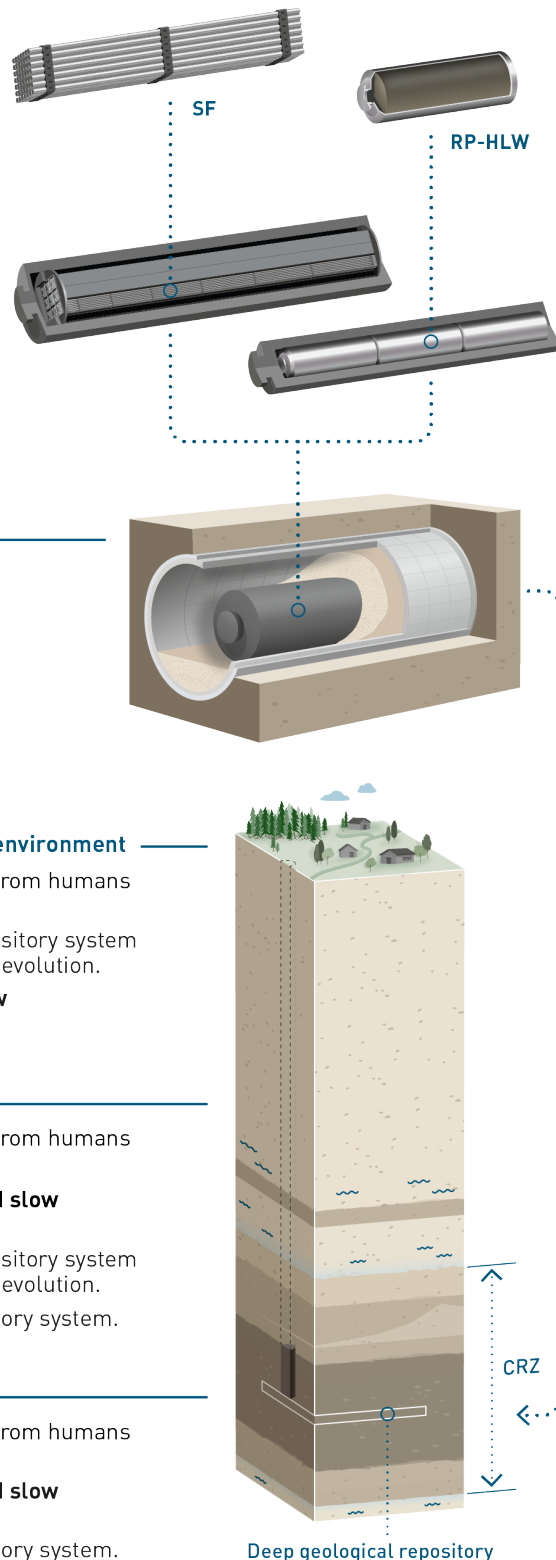
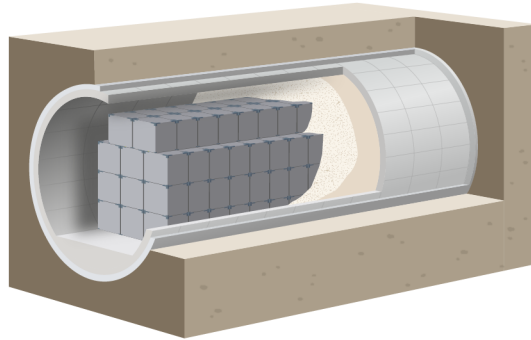


Fig. 3-3: Summary of the functions of the pillars of safety for HLW disposal in the current repository concept

Cementitious near field

- contributes to the **immobilisation, retention, and slow release** of radionuclides.
- is **compatible** with the components of the repository system.



Deep underground location in a stable geological environment

- contributes to the **isolation** of radioactive waste from humans and the environment.
- contributes to the **long-term stability** of the repository system with regard to long-term geological and climatic evolution.
- supports the **immobilisation, retention, and slow release** of radionuclides.

Containment-providing rock zone (CRZ)

- contributes to the **isolation** of radioactive waste from humans and the environment.
- contributes to the **immobilisation, retention, and slow release** of radionuclides.
- contributes to the **long-term stability** of the repository system with regard to long-term geological and climatic evolution.
- is **compatible** with the components of the repository system.

Closure system (backfill and seals)

- contributes to the **isolation** of radioactive waste from humans and the environment.
- contributes to the **immobilisation, retention, and slow release** of radionuclides.
- is **compatible** with the components of the repository system.

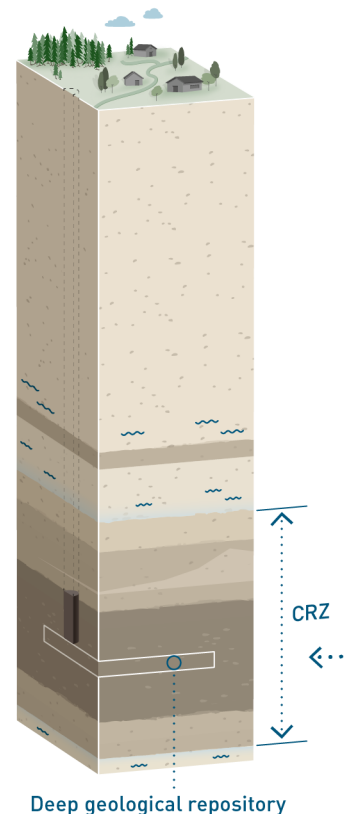


Fig. 3-4: Summary of the functions of the pillars of safety for L/ILW disposal in the current repository concept

3.3 Provisional repository design and implementation plan

The provisional design for the geological repository, i.e., the implementation of the current safety and repository concept, has been derived from the design principles⁷ set out in Article 11, Paragraph 2 of the Nuclear Energy Ordinance (KEV 2004) and developed for the purposes of the current licensing step. An overview is given in the following paragraphs; for more details, see Chapter 4 in NAB 24-18 Rev. 1 (Nagra 2024s) and NTB 24-11 (Nagra 2024b). Regarding implementation, it is sufficient to note that procedures for constructing, operating and closing the repository are established, available and affordable. Nevertheless, deviations between the “as-built” state and the “design” specifications must be anticipated, and measures to deal with such situations are foreseen. The safety assessment methodology, with its approach to uncertainty management, contributes significantly to developing such measures.⁷

As illustrated in Fig. 3-5, the provisional repository design comprises two separate sections: one for the disposal of L/ILW (including ATW), and one for HLW (including both RP-HLW and SF). It will be connected to the surface via three shafts: an access shaft, an operations shaft and a ventilation shaft.

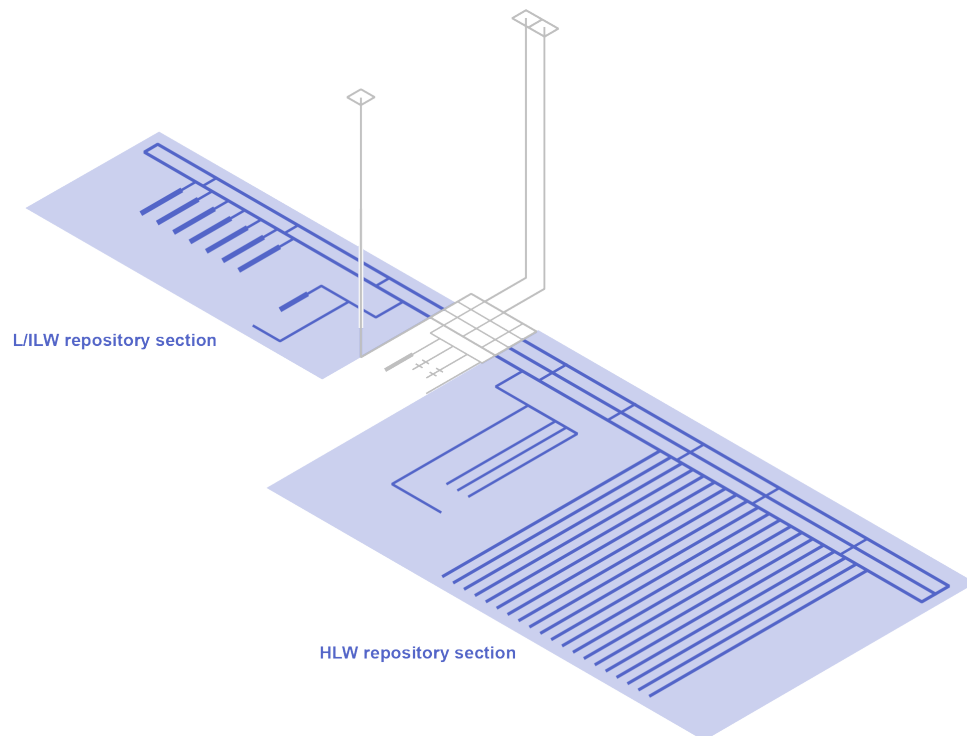


Fig. 3-5: Representation of two separate repository sections (both coloured in blue) in the provisional design for a deep geological repository for both HLW and L/ILW

⁷ The design principles for a deep geological repository are set out in Article 11, paragraph 2 of the Nuclear Energy Ordinance of 2004 (KEV 2004). A deep geological repository must be designed to ensure that: a. *It complies by analogy with the principles of Article 10, paragraph 1* [Basic principles for the design of nuclear power plants]; b. *It guarantees long-term safety through multiple passive safety barriers*; c. *Steps to ease surveillance and repairs of the repository, or for the recovery of the waste, in no way impair the effectiveness of the passive safety barriers after closure of the repository*; d. *It can be closed within a period of a few years*. Several additional principles and requirements relating to geological disposal defined in the Nuclear Energy Ordinance are listed in NAB 24-18 Rev. 1 (Nagra 2024s).

A central area will be constructed at the base of the shafts, which will serve as the starting point for the excavation of the L/ILW and HLW repository sections. Repository section accessways connect the shafts and central area to the emplacement rooms. These accessways include an operations tunnel for waste transfer and a ventilation tunnel in both repository sections. A third tunnel is included in the HLW repository section to enable the continuous construction of additional HLW emplacement drifts in parallel with the emplacement of HLW disposal canisters. Continuous construction in parallel with waste emplacement is advantageous for HLW in that it limits the time the emplacement drifts remain open. By contrast, construction of the L/ILW repository section is foreseen to be completed before emplacement operations start, and the tunnel used for the construction can thus also serve as operations tunnel.

Disposal takes place in parallel sets of emplacement rooms (HLW emplacement drifts and L/ILW emplacement caverns). The HLW emplacement drifts are supported by pre-cast concrete tubing elements (i.e., ring elements). The tunnel support for the L/ILW caverns and other underground openings consists of an outer layer of shotcrete and an inner concrete liner.

The engineered barriers installed within the emplacement rooms are illustrated in Fig. 3-6.

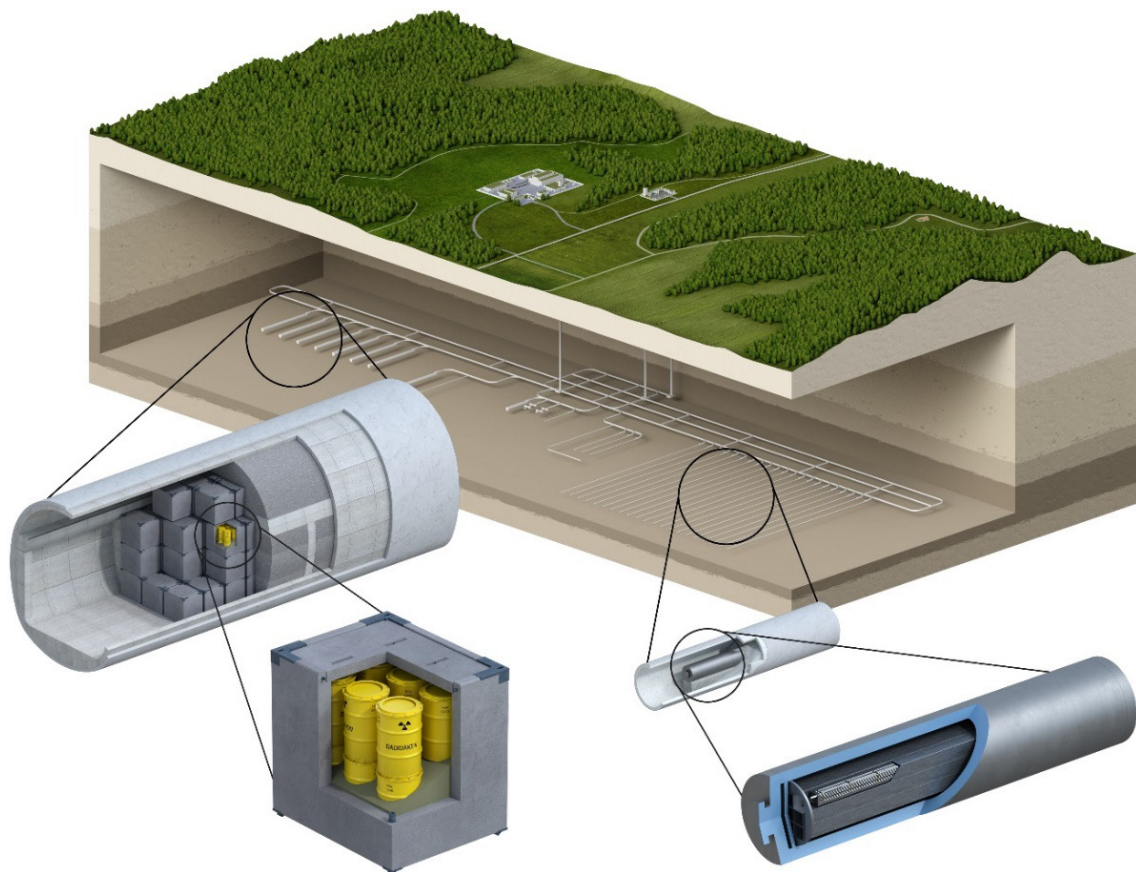


Fig. 3-6: Conceptual illustration of engineered barriers for the disposal of L/ILW (left) and HLW (right) based on the current repository concept

In addition to the main HLW and L/ILW repository areas, each repository section includes a pilot repository, in which the behaviour of the wastes, the backfill, the seals and the host rock is monitored until the end of a monitoring period. Access to the pilot repositories is via short tunnels that branch from the service tunnels for the main HLW and L/ILW disposal areas.

HLW is encapsulated in carbon steel disposal canisters, placed coaxially along the emplacement drifts, with the surrounding space filled with the buffer material, as illustrated in Fig. 3-7. The buffer design consists of:

- prefabricated bentonite blocks, which are placed on the floor of the emplacement drifts; the canisters are then placed on these blocks so that they are in the centre of the drift, and
- granular bentonite material used to backfill the voids around and between the canisters.

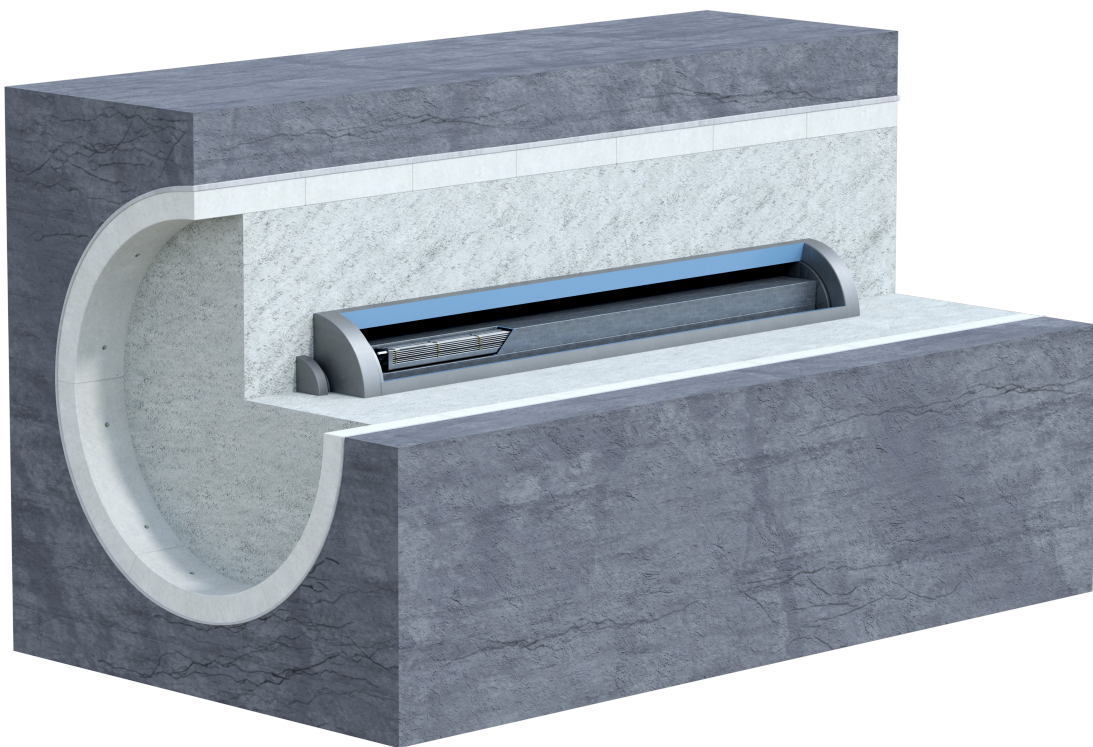


Fig. 3-7: Schematic representation of an HLW emplacement drift backfilled with bentonite

Most of the L/ILW is compacted, embedded in a cement/mortar matrix and placed in standardised 200 litre drums, MOSAIK-II-type casks made of cast iron, or steel casks. These waste packages are placed inside containers made of reinforced concrete and infilled with a cement-based mortar (Fig. 3-8).

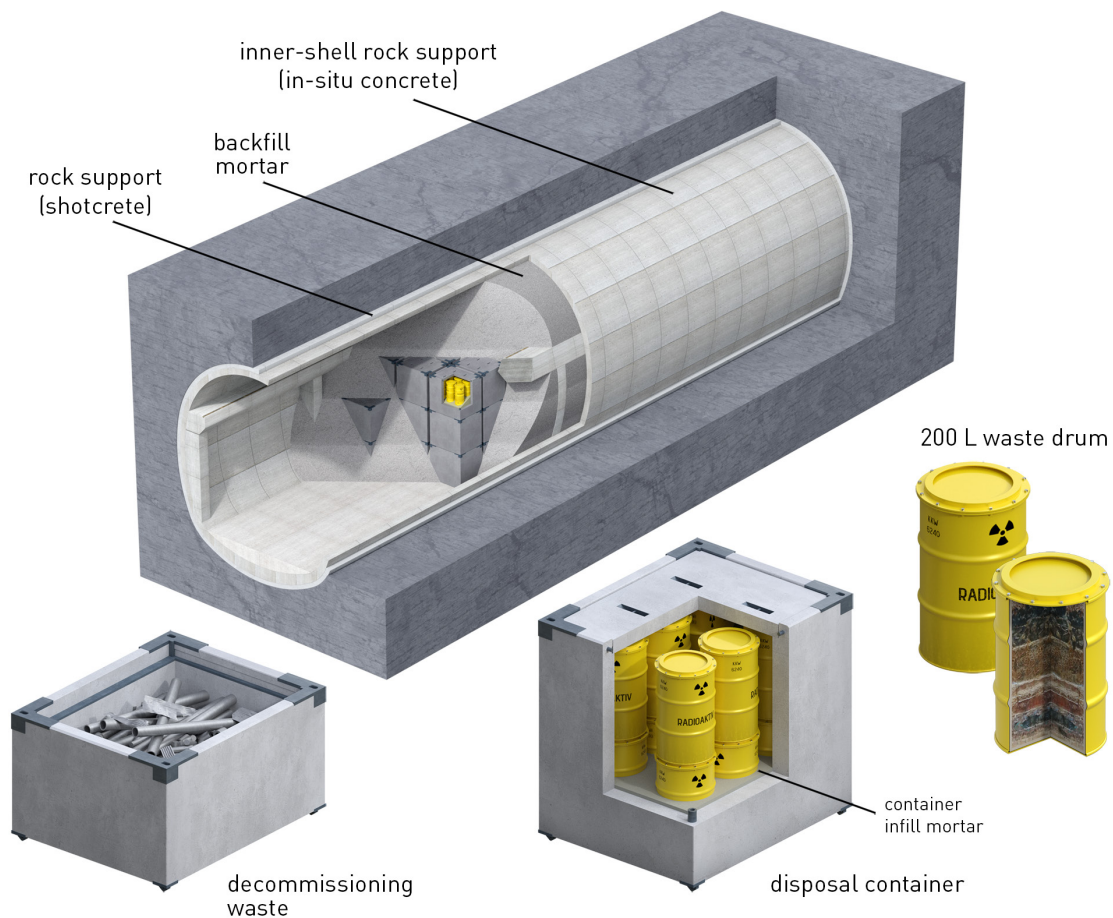


Fig. 3-8: Examples of L/ILW disposal containers, one filled with directly packaged waste (left), and one containing 200 litre drums (right)

Once the L/ILW containers are emplaced in the caverns, the caverns are also backfilled with a cement-based mortar.

Finally, the closure system is foreseen to consist of various clay-based seals and tunnel backfill. As illustrated in Fig. 3-9, the seals comprise:

- V1-HLW and V1-L/ILW seals (emplacement room seals) located at the entrances of the HLW emplacement drifts and L/ILW emplacement caverns, respectively. These seals are installed immediately after emplacement of the waste and the complete backfilling of the emplacement rooms.
- V2-HLW and V2-L/ILW seals (disposal area access seals) located at the entrances to the pilot repository accessways and the disposal area accessways of the HLW repository section (in the construction, operations and ventilation tunnels) and the L/ILW repository section (in the ventilation and operations tunnel), respectively.
- V3 seals (shaft seals) located in the shafts up to the top of the Opalinus Clay host rock. These are the outermost seals.

Two types of clay-based backfills, termed VF1 and VF2 (see Fig. 3-9), fill the repository structures up to the V3 seals. In addition, shaft backfill fills the part of the shafts above the V3 seals up to the surface level (NAB 21-12 Rev. 1, Nagra 2021b).

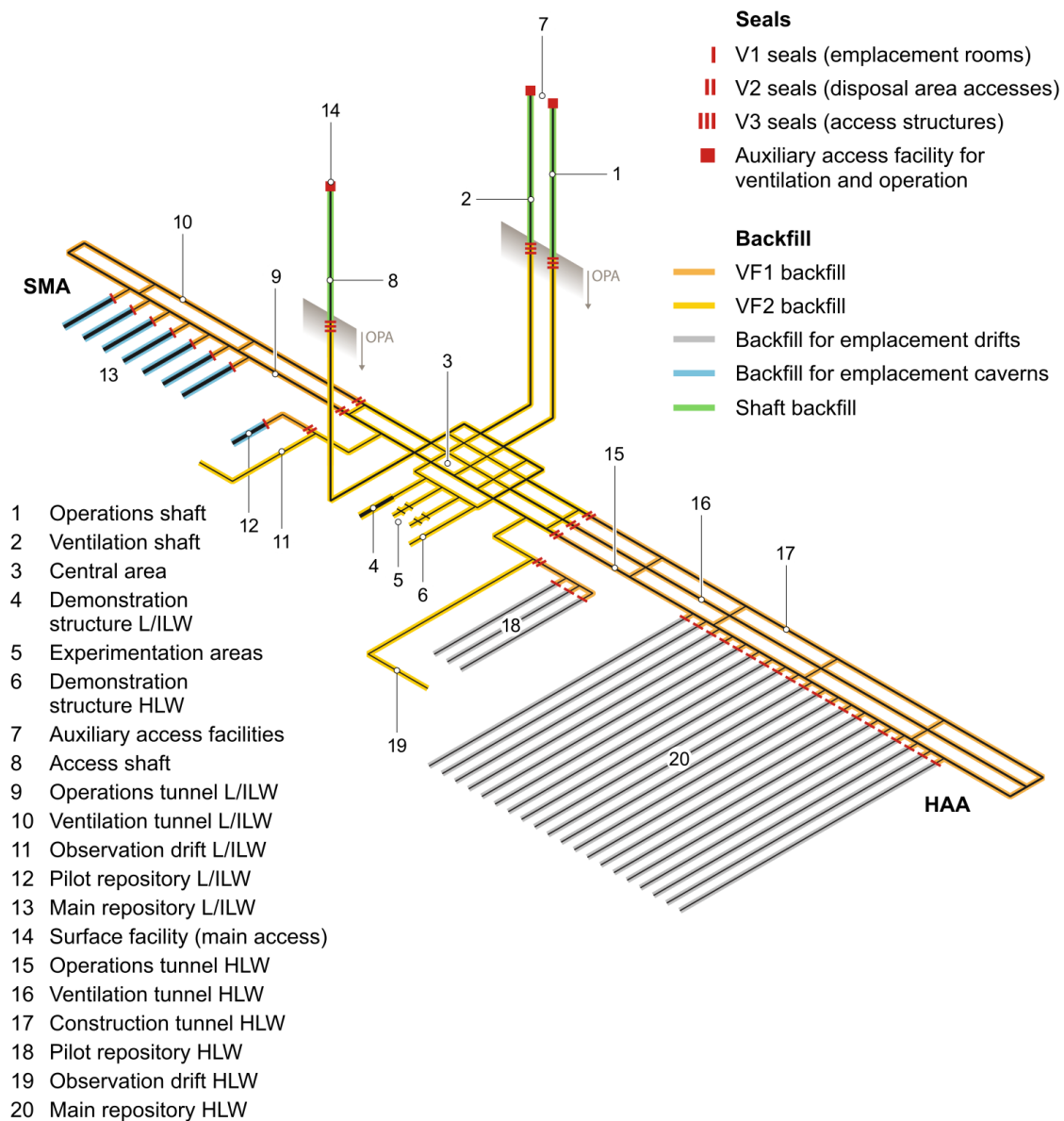


Fig. 3-9: Schematic view of the underground facilities in the provisional design, including seals and backfill of the repository structures

From NAB 21-12 Rev. 1 (Nagra 2021b).

The V1-HLW, V2-HLW and V3 seals are designed to be both gas- and water-tight. In contrast, the V1-L/ILW and V2-L/ILW seals and the backfill materials are designed with a higher gas permeability to prevent the build-up of potentially damaging gas pressures, due to gas generated by corrosion of metals and degradation of organic materials, predominantly in the L/ILW emplacement caverns.

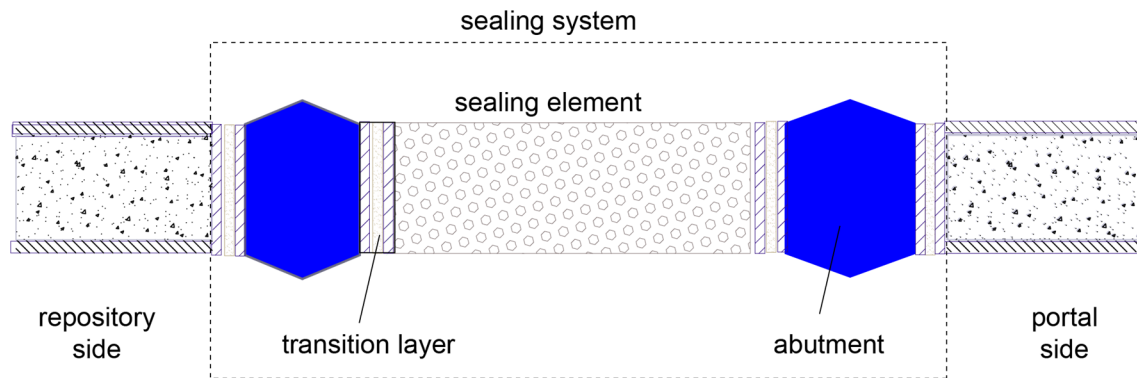


Fig. 3-10: Generic illustration of a seal, showing its principal components
(not to scale)

The basic design concept is broadly similar for all types of seals and is illustrated in Fig. 3-10. The principal components of a seal comprise:

- one or more sealing elements composed of compacted bentonite clay or bentonite/sand,
- one or more concrete abutments to hold the sealing elements in place, absorbing mechanical stresses and bentonite swelling pressure from the adjacent sealing elements, and
- one or more transition layers for the purpose of structural design and to provide the physical separation between clay and concrete components.

4 Safety assessment methodology

Aims of the chapter:

- To demonstrate that the post-closure safety case is built on a traceable and transparent methodology that encompasses performance assessment, development of safety scenarios and the analysis of radiological consequences.
- To demonstrate that the methodology is founded on sound principles.
- To define the time period for assessment, taking account of declining radiological toxicity over time.
- To show that the methodology integrates management of uncertainty and is comprehensive.

Safety assessment is, in the present context, defined by Nagra as the process of gathering the evidence and arguments and carrying out analyses regarding the safety of the disposal system during the post-closure phase and is the means by which the safety case is developed.

The methodology described here is a high-level summary of the workflow carried out for the safety assessment of a geological repository for both HLW and L/ILW in support of the general licence application, which is described in greater detail in NTB 24-19 (Nagra 2024t).

4.1 Safety assessment principles and objectives

Nagra's safety assessment methodology has been developed in line with international guidance, advances, and publications. Based on these and on Nagra's own experience, the following principles and objectives for the safety assessment have been identified:

- **A systematic approach to information gathering and integration** – The assessment relies on a systematic approach to information gathering and integration to assemble the assessment basis, the main elements of which are summarised in Chapter 5.
- **Rigorous consideration and treatment of uncertainty** – As far as possible, all potential sources of uncertainty and the various sources of bias must be considered and recognised when conducting and interpreting the analyses. Uncertainty management is described in general terms in Section 4.4 and is a feature of all the processes within the safety assessment methodology.
- **Assurance of completeness** – All reasonably foreseeable possibilities for the characteristics and evolution of the disposal system are considered in developing safety case arguments. The assurance of comprehensiveness, i.e., the minimisation of “completeness uncertainty” is discussed further in Section 4.5.
- **Validation and verification of models and databases** – Appropriate quality assurance and control measures are adopted to ensure that the models and databases developed and applied in the analyses are suitable for their intended purpose. Each computer code used to perform analyses must be verified. Model abstractions, including the use of simplifications and conservatism in modelling, must be justified.
- **Use of stylised approaches** – Stylised approaches⁸ are adopted for the modelling of the biosphere and the nature of future human behaviour and actions, on account of the largely irreducible, poorly quantifiable or unquantifiable uncertainties associated with predictions regarding these aspects, even over relatively short timescales.
- **Multiple lines of argument for safety** – Claims related to the performance and safety of the disposal system are supported, as far as possible, by multiple lines of argument and wide-ranging evidence. The development of arguments and evidence supporting a set of high-level claims is a key element of the performance assessment methodology (see Chapter 6). The multiple lines of argument underpinning the overall safety case are brought together in Chapter 10.
- **Internal and external reviews** – All relevant aspects within the safety assessment are subject to internal and/or external review.

⁸ A stylised approach involves imposed, rather than scientifically derived, assumptions for handling poorly quantifiable or unquantifiable uncertainties in scenario development (see also the glossary in Chapter 13).

4.2 Main processes in safety assessment

The safety assessment, which provides the safety case for the repository as defined by the provisional design and implementation plan, consists of four main processes and is illustrated in dark blue boxes on the right-hand side of Fig. 4-1.

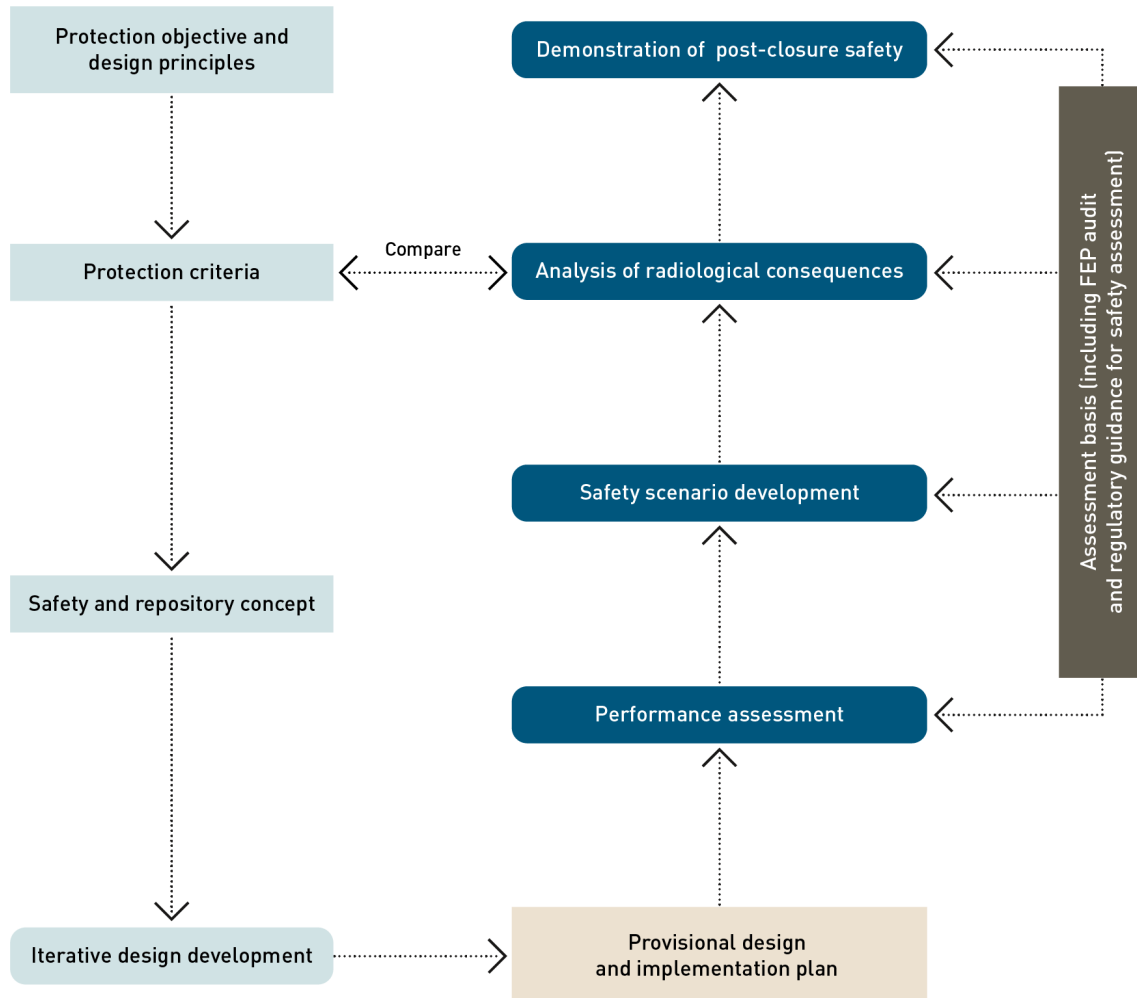


Fig. 4-1: Workflow for the post-closure safety case, with the four main processes of safety assessment shown in dark blue boxes on the right-hand side of the figure

Performance assessment (see Chapter 6) develops arguments and presents evidence to show that the safety functions of the repository are upheld and that the system will evolve as expected in most reasonably foreseeable situations, as described in detail in NAB 24-20 Rev. 1 (Nagra 2024m). Performance assessment also identifies uncertain and potentially detrimental phenomena and other uncertainties that could degrade the functionality of the components of the disposal system or of the overall disposal system, as input to safety scenario development.

Based on the knowledge provided by the assessment basis and the findings of performance assessment, **safety scenario development** (see Chapter 7) defines a set of safety scenarios that capture the broad ways in which the repository system could fulfil its safety functions over time, as well as additional hypothetical “what-if?” cases, which are aimed at further demonstrating the robustness of the disposal system. Safety scenarios and “what-if?” cases can contain several variants and most of them are, via a set of calculation cases, carried through to the analysis of radiological consequences. There, radionuclide release, retention and transport are modelled, and dose assessments are carried out. Some variants, however, can be handled by qualitative argumentation. These are also identified within the scope of scenario development.

The analysis of radiological consequences (see Chapter 8) uses quantitative models to evaluate annual individual dose rates or risks and complementary safety indicators. Suitable conceptualisations are formulated for the calculation cases that are propagated from scenario development. Both deterministic calculations and probabilistic uncertainty and sensitivity analyses are performed, and the impact of uncertainty is evaluated systematically (see Section 4.4). The results are compared with the regulatory protection criteria and the safety margin for each calculation case is determined. The output of the radiological consequence analysis also includes complementary safety indicators, such as radionuclide fluxes and concentrations, which are described in Chapter 9.

Demonstration of post-closure safety (see Chapter 10) is achieved by bringing the results of the three previous steps in safety assessment together with other information supporting the quality of the safety assessment, the safety and robustness of the repository system, and the strength of the stepwise repository siting, design and implementation processes, in the safety case. The calculated safety margins provide a key line of argument for the overall demonstration of safety in the safety case. As explained in Chapter 9, however, this line of argument is complemented and strengthened by additional lines of argument that address, e.g., the quality of the system.

4.3 Time period for assessment

Regulatory guidance in ENSI Guideline G03 (ENSI 2023) states that a period of up to one million years shall be considered by the safety case, and that the temporal development of the radiological hazard potential of the emplaced waste and the predictability of the long-term geological evolution must be taken into account⁹. In NTB 08-05 (Nagra 2008), by considering the decrease in radiotoxicity that occurs over time, the time period for assessment (referred to in the context of site comparison as the time period under consideration) was defined as extending up to 100,000 years for an L/ILW repository and up to one million years for an HLW repository. These time frames are also adopted for SGT-E3 when considering separate HLW and L/ILW

⁹ Für den Sicherheitsnachweis ist ein Nachweiszeitraum von bis zu einer Million Jahre festzulegen. Die zeitliche Entwicklung des radiologischen Gefährdungspotenzials der eingelagerten Abfälle und die Prognostizierbarkeit der geologischen Langzeitentwicklung sind zu berücksichtigen.

repositories (see NAB 24-05, Nagra 2024a for details). The shorter period considered for the L/ILW repository, based on considerations of radiotoxicity, is also consistent with guidance from ENSI in ENSI 33/649 (ENSI 2018)¹⁰.

For a geological repository for both HLW and L/ILW, a one-million-year time period for assessment is considered, although the time frame specific to the assessment of certain processes or phenomena (e.g., heat production from HLW) can be significantly shorter. In case of the calculations of radionuclide release, retention and transport carried out for the analysis of radiological consequences, ENSI Guideline G03 lists the additional requirement that the dose calculations shall be performed up to the time of the maximum radiological impact of the deep geological repository¹¹. To meet this requirement, dose calculations are carried out up to the maximum radiological impact time, extending to 10⁷ years for HLW and 10⁶ years for L/ILW releases. Analyses for volatile radionuclides, which focus on ¹⁴C with a 5,700-year half-life, are carried out for a 10⁵-year period. After this time period, ¹⁴C will have decayed to activities that are insignificant.

Beyond the time period for assessment, ENSI Guideline G03 requires determination of the range of potential impacts by considering uncertainties and scenarios involving geological processes. Notably, glacial and non-glacial (fluvial) erosion, impacting the repository barrier system, is addressed in the dedicated report NAB 24-08 Rev. 1 (Nagra 2024q), taking into account time frames well beyond the time period for assessment.

4.4 Uncertainty management

For a complex problem, such as analysing the future evolution, performance and safety of a deep geological repository and its environment, it is impossible to make exact predictions. Some level of uncertainty can, however, be accepted, provided the uncertainties do not compromise the demonstration of safety. As explained in Section 4.1, rigor in the consideration and treatment of uncertainty is one of the basic principles underlying the safety assessment methodology. Furthermore, ENSI Guideline G03 (ENSI 2023) states that, for the safety case, data, processes and model concepts shall be used that are in accordance with the state-of-the-art in science and technology, and that their uncertainties shall be identified¹².

Uncertainty management in safety assessment is discussed at length in Chapter 3 of NTB 24-19 (Nagra 2024t) and includes the following particular aspects:

- Types of uncertainty

In line with common practice, the safety assessment categorises uncertainty based on its source as data uncertainty (also termed parameter uncertainty), model uncertainty (also termed conceptual uncertainty), and uncertainty in the broad evolution of the safety functions (sometimes termed scenario uncertainty). In some instances, uncertainty may also be classified by its nature as epistemic (caused by deficiencies in knowledge) or aleatory (due to the apparent randomness of relevant phenomena). In principle, the former relates to uncer-

¹⁰ Falls gezeigt werden kann, dass durch das geologische Tiefenlager aufgrund des radiologischen Gefährdungspotenzials der Abfälle bereits nach weniger als einer Million Jahre nur noch vernachlässigbar kleine radiologische Auswirkungen für Mensch und Umwelt zu erwarten sind, kann der Nachweis für einen kürzeren Betrachtungszeitraum geführt werden.

¹¹ In der Sicherheitsanalyse sind Dosisberechnungen bis zum Zeitpunkt der maximalen radiologischen Auswirkungen des geologischen Tiefenlagers durchzuführen, mindestens jedoch bis zum Ende des Nachweiszeitraums.

¹² Für den Sicherheitsnachweis sind Daten, Prozesse und Modellkonzepte gemäss Stand von Wissenschaft und Technik zu verwenden und deren Unsicherheiten aufzuzeigen.

tainty that can be avoided, reduced, or eliminated and the latter relates to uncertainty that cannot be reduced or eliminated. In practice, it can be difficult to make a clear distinction between the aleatory and epistemic uncertainties.

- Management of uncertainty in the assessment basis

The assessment basis, i.e., the evidence, knowledge, assessment tools, and methodologies developed or acquired by Nagra in support of the safety assessment, is described in Chapter 5. The body of information that is contained within the assessment basis includes information gathered from a variety of sources, and the associated uncertainty represents a mixture of epistemic and aleatory types. Epistemic uncertainty is reduced, as far as reasonably possible, e.g., by site characterisation, research, and design considerations. Remaining uncertainties are then identified and, if possible, quantified, e.g., by specifying either likely values and ranges or probability density functions (PDFs) for associated parameters. Where different types or sources of information are available, an ensemble of information, possibly with different representations of uncertainty, is integrated to provide, as far as possible, a coherent, logical, realistic, and defensible description of the disposal system and its environment, including statements of uncertainty.

- Uncertainty in the broad evolution of the repository system and its components and how they contribute to the safety functions

Uncertainty in the broad evolution is handled in performance assessment and safety scenario development (Chapters 6 and 7), resulting in a set of safety scenarios, namely the reference safety scenario, a set of alternative safety scenarios and a set of future human action (FHA) safety scenarios. The latter consider potential future actions undertaken by human society that could impact the repository and are treated as a separate class of safety scenarios, due to their particularly speculative nature. Furthermore, “what-if?” cases, which involve extreme and hypothetical assumptions and primarily aim at demonstrating the robustness of the repository system, contribute to bounding the consequences of uncertainty in the broad evolution (see also Section 4.5).

- Management of uncertainty in modelling studies of performance assessment and analysis of radiological consequences

All models used in safety assessment incorporate a substantial degree of simplification. Simplification is needed because of the complexity of the disposal system, the impossibility of complete system characterisation, and the limited understanding that is available of some processes. Simplification can include, for example, the omission of some less well characterised phenomena. Moreover, where omission of a phenomenon cannot be justified, the approach is often to incorporate the complex or poorly understood phenomenon in a relatively simple form in the mathematical models and codes, while applying relatively pessimistic ranges to the parameter values to address the resulting uncertainty. Simplifications and assumptions that are subject to uncertainty can be justified on the grounds that they either have a negligible impact on the calculation endpoints (e.g., performance and safety indicators), or that they are conservative. Uncertainty in model output arises from the uncertainty in input data as well as from the model uncertainties handled as described above. Two complementary techniques are employed to quantify the uncertainty in model output due to the uncertainty in the input data. The first is called deterministic uncertainty analysis, in which uncertainty in model output is explored by defining and testing specific input parameter values. The second is probabilistic uncertainty analysis, where the selection of input data is carried out based on the probabilities assigned to the data (e.g., using the Monte-Carlo method), and the model output is analysed using standard graphical and statistical methods.

Finally, there are irreducible, poorly quantifiable or unquantifiable uncertainties associated with the evolution of the biosphere and future human lifestyles and actions, and stylised approaches are adopted to deal with these. For example, for the purpose of the assessment, possible future human actions that could affect the repository are constrained to those that are possible using present-day technology or moderate developments thereof.

- **Quality assurance and control**

The possibility of human error in applying the methodologies presented in this report represents an additional source of uncertainty. To minimise this type of uncertainty, measures of quality assurance and control must be applied to all activities that use or produce models and data. Specific measures to ensure that the models and databases used in safety assessment are fit for purpose, that the mathematical models are implemented correctly in computational models, that the computational models are reliable and that they are applied correctly and without error are presented in underlying reports (Nagra 2024s, 2024u, 2024k, 2024o, 2024p). These include a comparison of model outputs with the results of experiments covering a range of spatial and temporal scales and with observations of natural systems and the verification of numerical codes, e.g., by benchmarking against analytical solutions and against other codes that can address the same or similar problems.

The potential for human error during the construction, operation, and closure of the repository cannot be entirely ruled out. While implementing appropriate quality assurance and control measures is foreseen, uncertainties – such as those arising from failure to meet design requirements – remain a concern. These uncertainties are currently mainly covered by the definition and analysis of “what-if?” cases that assume a hypothetical degraded performance for each of the main repository barriers (see Section 7.4). To enhance safety and reliability, these aspects may be more thoroughly evaluated in future assessments, contributing to the ongoing development and revision of design requirements.

4.5 Comprehensiveness

Although it is impossible to prove beyond all possible doubt the completeness of a safety assessment, measures are taken to ensure the inclusion of a comprehensive set of potentially relevant phenomena and, of equal importance, to ensure that the safety-relevant phenomena have been represented appropriately in the safety scenarios and analyses carried out within safety assessment.

An assessment basis has been developed that includes all identified and relevant phenomena and processes, as outlined in Chapter 5. As part of this assessment basis, understanding of the initial state and post-closure evolution of the repository, which forms the basis for performance assessment, safety scenario development, and analysis of radiological consequences, is synthesised in NAB 24-20 Rev. 1 (Nagra 2024m), while the characteristics and evolution of the site are detailed in NTB 24-17 (Nagra 2024i). These descriptions have undergone iterative development over many years, with each version being subjected to peer review and taking into account Nagra’s extensive RD&D programme documented in NTB 21-02 (Nagra 2021a). The comprehensiveness of these descriptions is further enhanced by effective information exchange among safety assessors, technical experts within Nagra, and the broader scientific community.

The assessment basis also includes a comprehensive catalogue of features, events and processes (FEPs), which has also been developed over many years, benefiting from reviews by both internal and external experts, and comparisons with similar international databases. A FEP audit has been conducted to verify the inclusion of all relevant FEPs in the safety assessment and to justify the omission of others. The audit evaluates whether all FEPs in the catalogue are adequately addressed in the safety assessment, either through inclusion in safety scenarios or via explicit or

implicit consideration in performance assessment models and in the models used for the analysis of radiological consequences and dose assessment. It is ensured that the exclusion of any FEPs is justified by showing them to be either outside the assessment scope or irrelevant to safety. This process, documented alongside the phenomenological evolution of the repository in a supplementary volume of NAB 24-20 Rev. 1 (Nagra 2024l), guarantees a thorough assessment and justification for FEP inclusion or exclusion. Similar audits have been conducted in Project *Entsorgungsnachweis* (demonstration of disposal feasibility, NTB 02-23, Nagra 2003), ensuring consistency and reliability across safety assessments. More details are also provided in Section 5.5.

Finally, as noted in Section 4.2 and discussed further in Section 7.4, “what-if?” cases are defined and analysed as part of the safety assessment. These “what-if?” cases involve extreme and hypothetical assumptions, primarily aimed at demonstrating the robustness of the repository system. Such analyses may also pre-empt potential criticism that the selected ranges of parameter values are too narrow or that some detrimental FEPs are either unknown or have been overlooked.

5 Assessment basis

Aims of the chapter:

- To describe the assessment basis, which has been developed over decades and allows for a comprehensive assessment of performance and safety.
- To describe the geological setting and the containment-providing rock zone in which the repository is located.
- To describe the initial state of the repository system and provide a short summary of its expected evolution and the fate of radionuclides.

5.1 General scientific understanding

Confidence in the findings of safety assessment requires a sound scientific understanding of the repository system and its evolution and that uncertainties are identified and, where possible, quantified. Nagra's scientific understanding relevant to the post-closure safety of the proposed repository system and its associated uncertainties has been developed through an interdisciplinary and collaborative programme of RD&D and site characterisation. Understanding of the long-term behaviour of the repository system and its geological setting is supported by wide ranging information, including:

- field observations and monitoring activities,
- laboratory experiments, field testing and demonstration/verification activities,
- modelling studies,
- literature surveys, and
- expert elicitation.

The following paragraphs summarise the assessment basis relevant to the safety assessment of a geological repository for both HLW and L/ILW. A detailed description, including references to data sources, can be found in the referenced reports.

5.2 Site characterisation and understanding

The current integrated understanding of the geology of Northern Switzerland, encompassing the siting regions considered in Stage 3 of the Sectoral Plan, is provided in the Geosynthesis of Northern Switzerland in NTB 24-17 (Nagra 2024i), and includes information on:

- the overall geological setting,
- past geological evolution,
- the properties of relevant geological elements, including the CRZ, and
- the expected long-term geological evolution in the future.

The following summary is a short excerpt of the broader information presented in NTB 24-17 (Nagra 2024i) and supporting materials.

5.2.1 Geological setting

Key geological properties (datasets) that led to the selection of the site are summarised in Fig. 5-1. The site lies in a tectonically quiet and stable area with limited deformation in the external Northern Alpine Foreland basin (Section 1.5 of NTB 24-17, Nagra 2024i). It is located away from large-scale tectonic zones of increased geological complexity (Alps, Internal Jura, Upper Rhine Graben, Hegau – Bodensee Graben). The area is characterised by low seismic activity and low deformation and uplift rates. In the potential repository zone, the host rock (Opalinus Clay) occurs ~900 m below ground level in a flat-lying configuration with a thickness of around 105 m (Section 4.2.6 of NTB 24-17, Nagra 2024i). 3D seismic data indicate that the potential repository zone is in an area devoid of any seismically mappable faults (Section 4.3.4 of NTB 24-17, Nagra 2024i). Past deformation has essentially been accommodated in regional faults zones situated to the north and to the south. The same is also expected for tectonic deformation that could potentially occur during the time period for assessment (Section 6.6.2 in NTB 24-17, Nagra 2024i). The regional geological setting and the local host rock configuration thus provide stable geological conditions regarding long-term geological changes of the site. The size of the area without mappable signs of tectonic faults ensures flexibility with respect to the repository design.

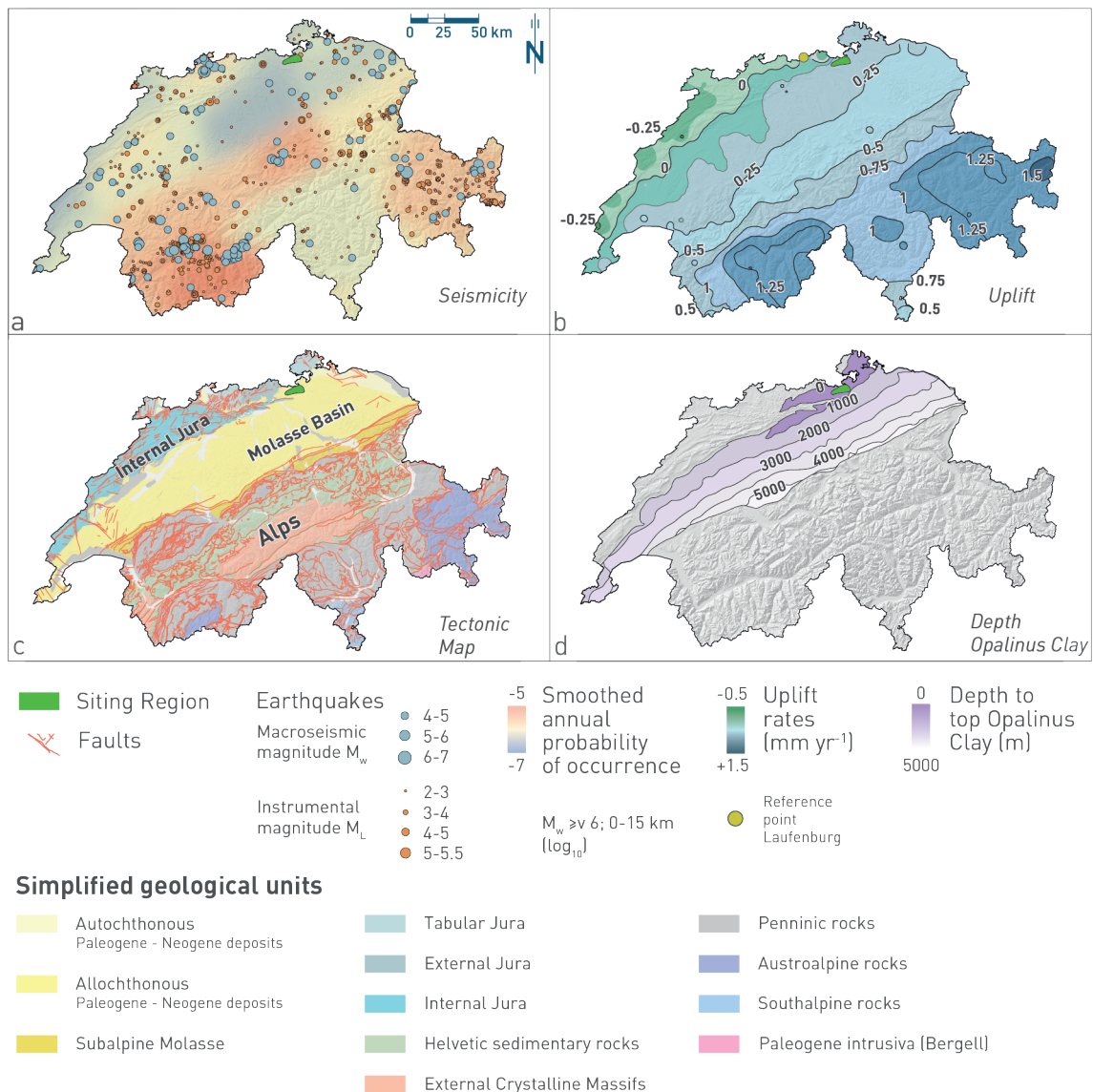


Fig. 5-1: Key datasets used for selecting the site in Northern Switzerland

The site is characterised by low (a) seismic activity and (b) uplift rates. 3D seismic data indicate that the repository will be located in an area (c) devoid of any seismically mappable faults and (d) where the Opalinus Clay occurs ~900 m below ground level in a flat-lying configuration with a thickness of around 105 m.

Simplified from Figure 1-3 in Chapter 1 of NTB 24-17 (Nagra 2024i).

The host rock, Opalinus Clay, consists of a ca. 174 Ma old, ca. 105 m thick and clay-mineral-rich unit with very low hydraulic conductivity (Fig. 5-2), embedded in low-permeability confining geological units (Section 4.2.6 and Chapter 5 in NTB 24-17, Nagra 2024i).

The aquitard sequence comprising the host rock and its low-permeability confining geological units, which is at least 300 m thick, is overlain by the regional Malm aquifer (Section 4.5.3 in NTB 24-17, Nagra 2024i). The ca. 40 m thick, clay-mineral-poor «Herrenwis Unit» is regarded as part of the low-permeability upper confining geological units, as the observed hydraulic conductivities are low, and the whole unit is embedded in clay-mineral-rich sequences. Observed low hydraulic heads indicate that the «Herrenwis Unit» is hydraulically isolated from the regional flow systems.

The Malm aquifer overlying the thick aquitard sequence is characterised by a fracture-dominated permeability, which mainly occurs in the upper part of the unit. Groundwater in the Malm aquifer contains a fossil marine component indicating rather stagnant flow conditions. This is confirmed, e.g., by compositions of stable isotopes of groundwater as illustrated in Fig. 5-2 (Section 4.5.5 in NTB 24-17, Nagra 2024i).

The first potential aquifer below the aquitard sequence is the Keuper aquifer (Section 4.5.3 in NTB 24-17, Nagra 2024i). Regionally, this is a complex aquifer with potential water-conducting zones in different subunits. Next to the site, zones with increased hydraulic conductivities have only been observed in conjunction with localised fluvial deposits with increased porosity in two boreholes in the western part of the region (in the BAC1-1 and STA2-1 boreholes). The sampled groundwater is a mix of meteoric and higher saline water components (Section 4.5.5 in NTB 24-17, Nagra 2024i). As the transmissive fluvial deposits occur only locally, the Keuper aquifer is not regarded as an effective flow system and the underlying units can contribute at least locally to the transport barrier (Section 4.9 in NTB 24-17, Nagra 2024i). Due to the heterogeneous nature of water-conducting units, the uncertainty regarding the significance of the Keuper aquifer for advective transport remains comparably large.

The next aquifer of regional hydrogeological importance below the host rock is the Muschelkalk aquifer, a fractured aquifer including zones with increased porosity (see Section 4.5.3 in NTB 24-17, Nagra 2024i). The Muschelkalk aquifer hosts the most dynamic deep aquifer at the site. The groundwater is of cold-time meteoric origin (Section 4.5.5 in NTB 24-17, Nagra 2024i).

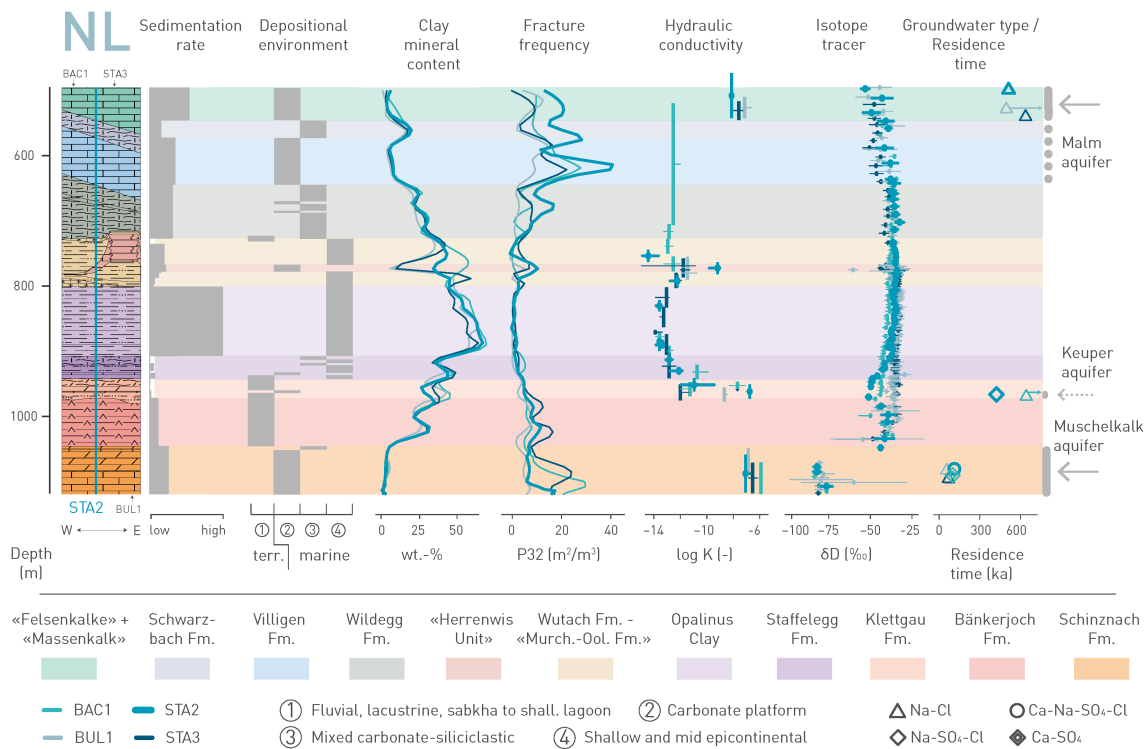


Fig. 5-2: Synthesis of key observations of the rock sequence between the Muschelkalk and Malm aquifers at the site

Data from three deep boreholes, labelled BAC-1, BUL-1 and STA-3, are projected to the data of the reference borehole labelled STA-2. For further explanation of this figure, consult Section 4.9 in NTB 24-17 (Nagra 2024i). The figure shows that the low-permeability zone around the host rock is delimited in the vertical direction by the occurrence of aquifers. The profiles of the natural tracers in between the aquifers can be explained by diffusive transport. The regional Malm and Muschelkalk aquifers define the maximum extent of the low-permeability zone. In between, the locally occurring Keuper aquifer must also be considered.

The hydraulic head distribution within the CRZ is controlled by the difference of hydraulic heads and distribution of conductivities between aquifers above and below the Opalinus Clay. Hydraulic head measurements in the regional aquifer systems were carried out in deep boreholes. The hydraulic head was regionally assessed using a hydrogeological model (Section 4.5.4 in NTB 24-17, Nagra 2024i). By considering the distance between aquifers and the vertical distribution of hydraulic conductivities in the aquitards, the magnitude and direction of the vertical hydraulic gradient could be determined. As stated above, the Malm aquifer is the closest above the Opalinus Clay, with the Muschelkalk aquifer below and the Keuper aquifer potentially reducing the distance between aquifers.

Hydraulic head differences are small within the site. While the combination of the Malm and Muschelkalk aquifers shows overall downward gradients, the northern part of the site indicates weak upward gradients when the Keuper aquifer is considered.

The absolute magnitude of the hydraulic head difference is mainly below ca. 100 m within the site.

The host rock is characterised by a fine and homogeneous pore structures, a very low hydraulic conductivity and, consequently, diffusion-dominated solute transport and a high self-sealing capacity (Chapter 5 in NTB 24-17, Nagra 2024i). It has a homogeneous and high clay-mineral content giving it a high sorption capacity for many radionuclides. Moreover, the mineralogy allows for efficient porewater buffering, providing stable and reducing geochemical conditions. Its exceptionally low lateral and horizontal variability is due to a relatively uniform marine depositional environment characterised by high clay-mineral input and relatively high sedimentation rates (Section 3.3.2 of NTB 24-17, Nagra 2024i).

Any so-far undetected subseismic faults are not expected to represent distinct flow paths. This is primarily due to deformation in the Opalinus Clay, which is typically accommodated in a distributed manner with soft-linked segmented fault planes, making continuous flow paths unlikely (Section 5.10 in NTB 24-17, Nagra 2024i). Secondly, the low stiffness and the high swelling capacity of the clay-mineral-rich host rock results in efficient self-sealing of fractures. A clear relationship between high clay-mineral content and low hydraulic conductivity was confirmed by the analyses of numerous hydraulic packer tests carried out during the most recent field investigations (Fig. 5-3).

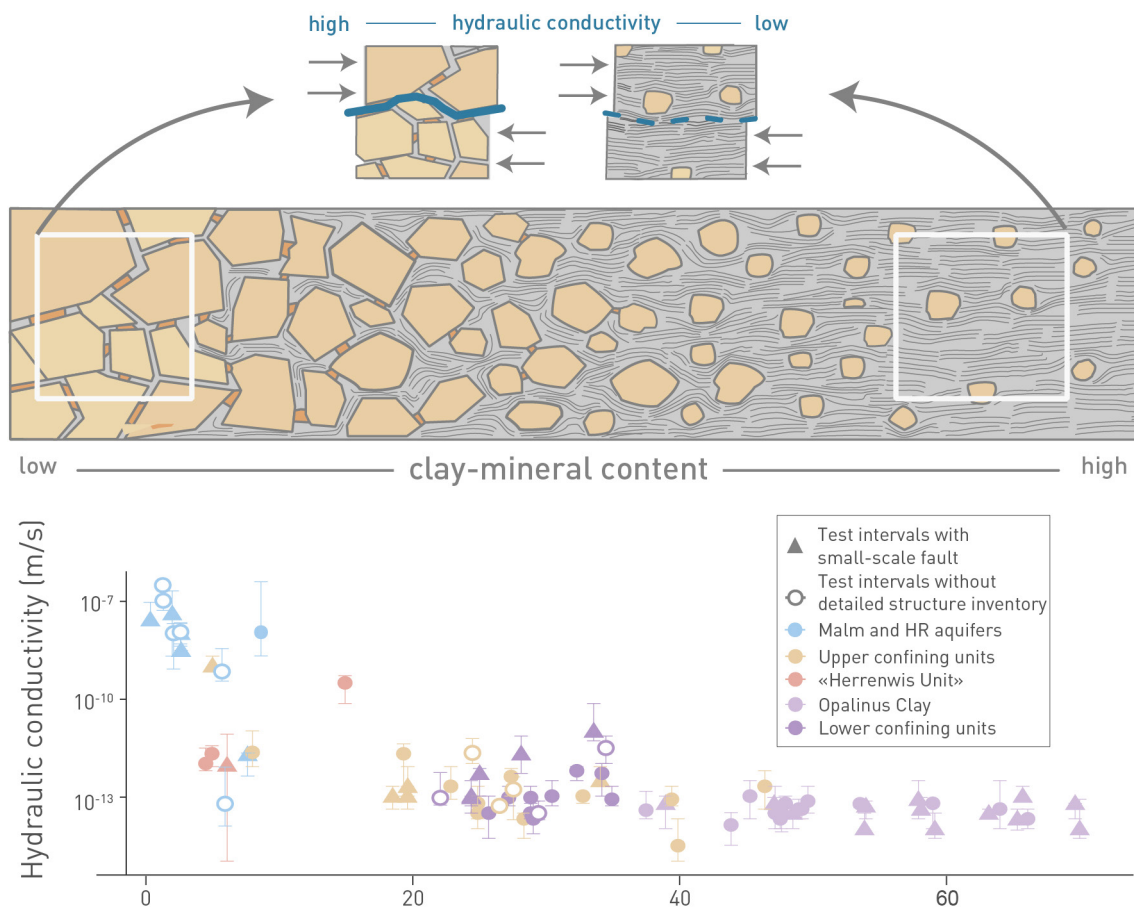


Fig. 5-3: Link between texture and clay-mineral content and the hydraulic properties in Mesozoic sediments of Northern Switzerland (lower confining geological units to upper aquifers)

Note Opalinus Clay values with consistently high clay contents and very low hydraulic conductivities.

The uppermost figure depicts a highly simplified transition from a grain-supported framework (clastics or carbonate skeleton, with low clay-mineral content partially filling the pores) on the left side, to a clay-matrix-supported framework on the right. Transition to the clay-matrix-supported framework at approximately 20 to 40 wt.-% is consistent with the low hydraulic conductivities. Note: figure shows data from Liassic to Malm. For further explanation of this figure, consult Section 5.10 in NTB 24-17 (Nagra 2024i).

Independent evidence for the highly effective barrier function of the host rock and confining geological units over geological timescales and for diffusion-dominated transport is provided by natural tracer profiles of the porewater composition (see Section 9.1.1).

5.2.2 Long-term geological evolution

Long-term geological evolution has been described in Chapter 6 of NTB 24-17 (Nagra 2024i) and in the references therein. A short excerpt is given in the following paragraphs to contextualise the results from the perspective of safety assessment.

Recent tectonic deformation rates in Northern Switzerland are small compared with those in the geological past. The study of past geological evolution shows that deformation is preferentially accommodated by larger existing faults, which represent zones of weakness. Future deformation will thus most likely result in fault reactivation of existing faults. These are avoided when placing the disposal areas. If new faults were to develop cutting through the host rock, they are expected to be segmented and small in length and offset.

Future erosive processes (glacial and non-glacial) are expected to be comparable with those that occurred in the Quaternary period, but the rates are likely to be lower (no major drainage reorganisation, smaller glaciers, delay in glacial inception, and harder-to-erode rocks). Due to the great emplacement depth, the repository system is protected against future erosion. Excavation of the repository by either glacial or non-glacial (fluvial) erosion processes within the next million years is not expected. Furthermore, a residual cover of at least 200 m, which ensures the self-sealing capacity of the host rock, is expected to remain present throughout the time period for assessment of one million years.

Climate change, glaciations, permafrost, and erosion can affect aquifer dynamics, including a shortening of the transport path along aquifers and the possibility of increased flow rates. Glacial loading may lead to overpressures in the host rock and to a temporarily increased hydraulic gradient and transport rates from the host rock towards the aquifers. However, mass transport is expected to remain diffusion dominated. Because of the buffering capacity of the host rock minerals, geochemical conditions will remain reducing.

5.3 Expected initial state and evolution of the repository system

A narrative of the initial state and expected evolution of the repository system is presented in the following sections. It is based on sound scientific understanding and accounts for uncertainty, as documented in the assessment basis reports (Fig. 1-1), and as brought together in more detail in NAB 24-20 Rev. 1 (Nagra 2024m). Arguments and evidence supporting key claims within this account of the expected evolution are validated by the performance assessment (PA), as summarised in Chapter 6.

5.3.1 Waste inventory and waste acceptance criteria

In Switzerland, the largest producers of radioactive waste are the nuclear power plants (NPPs). Other sources are applications in medicine, industry and research. All waste emplaced in the repository will be required to conform with waste acceptance criteria. The corresponding radionuclide inventory, the waste packages, masses, and volumes are included in the MIRAM (Model Inventory of Radioactive Materials) database for the general licence application (Nagra 2023b).

Five types of waste will be disposed of in the HLW repository section: two types of RP-HLW and three types of SF. RP-HLW contains most of the fission products from the spent fuel assemblies reprocessed at La Hague, France and at Sellafield, United Kingdom. During the production of this reprocessed waste, the liquid glass is poured into cylindrical steel moulds, which are then sealed. After cooling, the glass forms a homogeneous waste matrix. Spent fuel (SF) consists of fuel assemblies removed from Swiss NPPs and, to much lesser extent, from research reactors. The numbers of canisters for each waste type are presented in Tab. 5-1.

Tab. 5-1: HLW types and the numbers of canisters for each type

HLW type	Number of canisters
Boiling water reactor SF	891
Pressurised water reactor SF without MOX	694
SF from PSI Hot Laboratory and from the PSI-Diorit II Reactor*	5
Pressurised water reactor SF including MOX	345
RP-HLW from La Hague	145
RP-HLW from Sellafield	65

* Considered together with the pressurised water reactor SF without MOX in the analysis of radiological consequences.

Alpha-toxic waste (ATW) and low- and intermediate-level waste are disposed of in the L/ILW repository section. Both cover a broad spectrum of radioactive waste. ATW comprises (see Tab. 5-2):

- waste from the reprocessing of spent fuel assemblies (structural parts of reprocessed fuel assemblies and waste from the operation of reprocessing plants), and
- operational waste from research facilities, especially PSI and CERN, and various wastes from medicine and industry.

L/ILW (see Tab. 5-2) comprises:

- operational waste from the NPPs (resins, concentrates, sludges, mixed waste), the ZWILAG interim storage facility and the encapsulation plants for the deep geological repository,
- reactor waste from NPPs (mainly activated, replaceable reactor components),
- decommissioning waste from the NPPs (activated and contaminated reactor and building components and secondary waste arising during conditioning of the decommissioning waste), the interim storage facility and the encapsulation plants for the deep geological repository, and
- operational and decommissioning waste from research facilities, in particular PSI and CERN, and various wastes from medicine and industry.

ATW and L/ILW both consist of metallic, organic and inorganic materials, which are generally conditioned prior to interim storage.

Tab. 5-2: L/ILW types and the numbers of waste packages for each type

Origin of waste	KEV	Number of waste packages
Operational	L/ILW	51,020
Reactor	L/ILW	1,121
Decommissioning	L/ILW	12,475
Reprocessing	L/ILW	0
Operational	ATW	1,702
Reactor	ATW	0
Decommissioning	ATW	4
Reprocessing	ATW	552

5.3.2 Location, construction, operation, and closure of the underground facilities

The repository is constructed around 900 m below ground, in the middle of the Opalinus Clay host rock layer, with separate repository sections for the emplacement of SF and RP-HLW (the HLW repository section) and for L/ILW and ATW (the L/ILW repository section). Regional tectonic elements and seismically mappable faults are avoided with a sufficient respect distance, which is determined by performance assessment, see Section 3.7.2 in NTB 24-22 Rev. 1 (Nagra 2024u).

The following description refers to the current repository design. Future adaptations and optimisation to simplify and optimise the repository implementation will be carried out in full compliance with the regulatory requirements.

The construction of all underground structures is carried out in compliance with all relevant requirements and specifications. A respect distance to the exploratory boreholes is also observed (50 m from the disposal areas, Section 4.1.5 in NAB 24-18 Rev. 1, Nagra 2024s).

If necessary, measures are undertaken to ensure that any deviations from the planned implementation occurring in the pre-closure period have no impact on post-closure safety.

The HLW emplacement drifts are continuously backfilled with a bentonite buffer immediately after emplacement of the individual disposal canisters. As soon as an emplacement drift has been filled entirely, it is closed with a V1-HLW seal. The emplacement caverns for L/ILW and ATW are divided into compartments. Each compartment is backfilled with cementitious mortar within a few days after emplacement of the disposal containers. After all compartments in a cavern have been filled, the cavern is closed with a V1-L/ILW seal (Fig. 3-9).

The period of repository monitoring starts once all waste has been emplaced, as described in the Waste Management Programme 2021 (Nagra 2021c). Early in this phase, the accesses to the HLW and L/ILW repository sections are backfilled and sealed at the repository level with V2 seals. At the end of the monitoring period, test areas and all other underground structures that are still open are backfilled, and the access structures are sealed with V3 seals. The closure of the repository is again carried out with state-of-the-art technologies and best engineering practices and is assumed to comply with all relevant requirements and specifications.

5.3.3 Evolution of underground openings prior to backfilling

The excavation, cooling, and ventilation of the underground openings results in the gradual desaturation and depressurisation of the surrounding rock, which eventually resaturates when ventilation ceases. Transient processes associated with ventilation include:

- drying out of tunnel support materials,
- creation of desiccation cracks in the surrounding rock,
- precipitation of salts in these cracks due to evaporation of porewater,
- an increase in salinity of the porewater, and
- carbonation of cementitious materials, due to reaction with CO₂ in the air.

An Excavation Damaged Zone (EDZ) also forms around the underground openings. The EDZ, however, reseals in the post-closure period as it resaturates (see Chapters 3 and 4 in NAB 24-20 Rev. 1, Nagra 2024m).

5.3.4 Saturation of the access structures

After closure, the V3 seals that are placed in the access shafts to seal the repository from the surface environment saturate relatively rapidly, due to their high suction and the flow from more permeable rock formations above down the shafts and into the seals. Once saturated, the sealing elements have a low permeability to both water and gas (permeability to water of 10^{-19} to 10^{-18} m²). This largely prevents further water flow down the shafts into the as yet unsaturated underground structures of the repository. It also largely prevents the migration of repository-generated gas from the underground structures of the repository towards the surface environment (see the discussion of the fate of repository-generated gas in Section 6.2.3). This is relevant to radiological safety, as repository-generated gas can transport ¹⁴C, mainly in the form of methane (¹⁴CH₄).

5.3.5 Evolution of the L/ILW disposal area and connected underground structures after backfilling

After the emplacement of L/ILW and ATW and the backfilling and sealing of the emplacement caverns, the partially saturated EDZ and rock support around the L/ILW caverns largely resaturate with water and the EDZ fractures re-seal, although some gas-filled pores can remain as a result of repository-generated gas (see below). Some water can also enter the caverns and accumulate under gravity at the cavern floor. Anaerobic and reducing conditions develop due to the consumption of O₂ in trapped air and hyperalkaline conditions develop in the porewater within and around the caverns due to the presence of cementitious materials (see Chapter 6 in NAB 24-20 Rev. 1, Nagra 2024m).

Gas generation takes place due to corrosion of metals and degradation of organic materials in the caverns. Some of this gas dissolves in porewater within and around the caverns, with the remainder accumulating within the caverns, increasing the gas pressure and limiting further inflow of water. The V1-L/ILW seals that close the L/ILW caverns and the V2 seals that isolate the L/ILW repository section from the rest of the deep geological repository also start to saturate, but full saturation is prevented by the gas pressure in the caverns. The gas migrates through the seals and accumulates in the backfill of the adjoining underground structures, providing a storage volume for the gas in addition to that provided by the caverns themselves. Desaturated conditions persist in the upper parts of the L/ILW emplacement caverns and in the backfill of the connected underground structures for several tens of thousands to hundreds of thousands of years, until gas

generation ceases. Some gas may enter the host rock but will migrate through it without causing damage. Pore clogging due to cement – clay interactions within the V1-L/ILW and V2-L/ILW seals that might impact overall seal permeability is not expected, as explained in Section 6.5 of NAB 23-21 (Martin et al. 2023), especially given that it is likely that the seals remain only partially saturated for the entire period of gas production (Section 7.2.4 in NAB 23-21, Martin et al. 2023).

Transient heat initially produced by cement hydration and, in the longer term, radiogenic heat from the L/ILW, and especially from ATW, leads to a transient short-term increase in temperature within and around the L/ILW disposal area. This heat, in combination with gas penetration, also leads to elevated pore pressures. These elevated temperatures and pore pressures, however, are small and have no long-term impact on the performance of the barriers (Appendix A.5 in NAB 24-20 Rev. 1, Nagra 2024m).

The disposal containers provide an initial resistance to water ingress, but they degrade over time, and, after a few hundred years, water encounters the waste matrices in at least some of the containers, and the radionuclides embedded within them start to be dissolved. ^{14}C , mainly in the form of methane ($^{14}\text{CH}_4$), may also mix with the repository-generated gas in the emplacement caverns and closure system.

The tunnel support system around the emplacement caverns degrades and loses its mechanical strength over time, but the resulting stress redistribution has no impact on the safety-relevant properties of the barriers. In particular, the presence of backfill material limits cavern convergence to the extent that no previously sealed EDZ fractures are reactivated (Geomechanica Inc. 2025).

The cementitious near field evolves over time due to the mineralogical evolution of cement phases. A high pH is expected to be maintained for at least 100,000 years (Section 3.3.5 in NTB 23-03, Kosakowski et al. 2023). Interactions between the cementitious near field and the surrounding rock are of limited spatial extent and do not affect the safety-relevant properties of either barrier (Section 3.3.4 in NTB 23-03, Kosakowski et al. 2023). In particular, any pore clogging that occurs due to precipitation does not reduce the overall permeability to gas.

5.3.6 Evolution of the HLW disposal area and connected underground structures after backfilling

After emplacement of the HLW disposal canisters, as well as the buffer and V1-HLW seals in the emplacement drifts, the partially saturated EDZ and rock support around the tunnels resaturate with water, followed by saturation of the buffer and seals. EDZ fractures re-seal. The compacted clay structures (the buffer and the sealing elements of the V1-HLW seals) 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 impacted by decay heat generated by the waste. Full saturation is expected to occur some hundreds of years after emplacement (Chapter 7 in NAB 24-20 Rev. 1, Nagra 2024m). Saturation is sufficiently regular and any initial inhomogeneities in the buffer density are reduced over time, thus no potentially damaging stresses will be exerted on the canisters. The buffer density around the canister is sufficiently high to prevent, when saturated, microbial activity that might otherwise increase the rate of canister corrosion (Section 5.6 in NTB 19-03, Nagra 2019).

Anaerobic and reducing conditions develop due to the consumption of O_2 , e.g., by canister corrosion. Radiation shielding provided by the disposal canisters is sufficient to protect the barriers from radiation-induced effects. Slow 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 tunnel support system into the Opalinus Clay porewater. However, some gas also migrates in the gas phase

through the buffer and into the rock, where it dissolves (Chapter 6 in NAB 24-20 Rev. 1, Nagra 2024m). Gas migration causes no irreversible changes to either the buffer or the rock (Section 4.4.2 in NTB 24-22 Rev. 1, Nagra 2024u).

In the first few thousands of years following repository closure, repository-generated heat, primarily decay heat from the waste packages, leads to a transient increase in temperature and pore pressure within and around the HLW disposal area, including within the CRZ (Chapter 6 in NAB 24-20 Rev. 1, Nagra 2024m). However, 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 (Section 6.7.2 in NTB 23-02 Rev. 1, Nagra 2024v), but the resulting stress redistribution has no impact on the safety-relevant properties of the barriers (any reactivation of EDZ fractures will be temporary, as the fractures will re-seal, see Appendix A-2 in NTB 19-03, Nagra 2019).

Porewater composition in the buffer becomes anoxic and reducing, gradually equilibrating with that in the rock (Chapter 7 in NTB 23-02 Rev. 1, Nagra 2024v). Chemical interactions, including interactions between canister corrosion products (Section 8.2.3 in NTB 23-02 Rev. 1, Nagra 2024v) or the cementitious lining and the clay barriers (Section 6.7.2 in NTB 23-02 Rev. 1, Nagra 2024v), are of limited spatial extent and do not affect the safety-relevant properties of the barriers.

In the post-closure period, but after at least ten thousand years, disposal canisters become locally breached due to mechanical failure following weakening by corrosion (Section 4.3 in NTB 24-20, Nagra 2024d). The exposure of internal metal surfaces to water results in corrosion of these surfaces and locally increased gas production rates. The evolution of the waste packages following canister breaching is described in the context of radionuclide release, retention, and transport in Section 5.4.1.

5.3.7 Evolution driven by climatic and geological processes

Long-term geological and climatic evolution, including glacial and fluvial erosion and uplift, geomorphological effects and activation or re-activation of fault zones due to natural processes, are not expected to negatively impact the safety-relevant properties of the barrier system over the time period for assessment of one million years. While, over a period of hundreds of thousands of years, erosion processes, in combination with uplift, gradually result in a partial loss of overburden, an exhumation of any part of the disposal areas, or any decompaction of the CRZ that would affect safety-relevant properties (especially its porosity, hydraulic conductivity, and self-sealing capacity) can be ruled out over the time period for assessment.

5.4 Fate of radionuclides within the repository system and the biosphere

Understanding transport pathways for any radionuclides released from the repository is necessary to guide performance assessment, focusing on key aspects of barrier functionality and barrier evolution over time, and to evaluate potential radiological consequences. This section provides a broad-brush description of how radionuclides are expected to migrate through repository barriers. It is based on the qualitative description of the expected phenomenological evolution of the repository as presented in Chapter 9 in NAB 24-20 Rev. 1 (Nagra 2024m).

A monitoring period starts after the completion of waste emplacement operations. After an initial period of monitoring, the disposal area accessways will be backfilled and sealed. This is followed by a second monitoring period that is currently assumed to last several decades (as e.g., described in the Waste Management Programme, Nagra 2021c). After this monitoring period, the whole

repository is closed¹³ and the post-closure period begins. Evolution in the post-closure period with respect to the fate of radionuclides can be described in terms of five periods (which overlap to some extent):

1. The radionuclides are contained in the waste packages. In the case of HLW disposal canisters, this period is expected to last at least 1,000 years. No period of complete containment can be assumed for L/ILW disposal containers. However, the L/ILW containers, backfilled with cement mortar, contribute to the retention of radionuclides.
2. A small proportion of the radionuclides in the HLW are released after canister breaching. Some releases are relatively rapid (e.g., gaseous fission products), but most radionuclides (from activated SF assemblies or vitrified glass from reprocessing) are released very slowly, congruently with the degradation of the waste matrices.
3. The radionuclides released from the waste packages start to migrate through the surrounding bentonite buffer and cementitious backfill and into the host rock and/or the closure system. This migration is inhibited, reduced, and delayed by the physical and chemical environment created by the bentonite buffer and cementitious backfill.
4. The small fraction of radionuclides (corresponding to less than 1% of the initial activity) reaching the host rock continues to migrate through the CRZ away from the repository. This migration is diffusion-driven and very slow, allowing for further decay.
5. Only a very small proportion of the initial radionuclide inventory (corresponding to less than 0.1% of the initial activity) leaves the CRZ and enters the deep aquifers that bound it. These radionuclides may eventually reach a local aquifer and the biosphere, where they are further dispersed and diluted. Nevertheless, this can lead to radiological exposure of humans and the environment, albeit at extremely low levels.

In the following sections, additional details of these processes are presented.

5.4.1 Release and transport in the near field

When an HLW disposal canister is breached and sufficient porewater enters, the breached inner containment allows water to contact the waste matrix. This interaction initiates the release of radionuclides from the waste packages. Some of the radionuclides, located in cracks in the SF UO₂ matrix or in the annular gap between fuel pellets and rods, may be relatively rapidly released as fission gas or as dissolved nuclides (the instant release fraction, or IRF (Chapter 4 in NAB 23-10, Johnson et al. 2023)). However, most radionuclides are released much more slowly congruently with the dissolution of the UO₂ matrix and the corrosion of the cladding or congruently with the dissolution of the RP-HLW (Chapter 7 in NAB 23-09, Curti 2022 and Chapter 5 in NAB 23-10, Johnson et al. 2023).

In the case of L/ILW, water entering the waste containers and packages initiates the release of radionuclides from the waste packages.

¹³ This will involve backfilling the central area, the facilities for underground geological investigations and the observation drifts.

Fission gas release (FGR)

The quantification of FGR from SF rods is relevant to short-term as well as to post-closure safety and safety assessment. In the short term, it is relevant to the safe handling and encapsulation of damaged fuel assemblies in the final disposal canisters. In the context of post-closure safety, FGR from SF is known to correlate with the IRF for I-129 and Cs-137 as explained in Section 4.1 in NAB 23-10 (Johnson et al. 2023). Trapped fission gas also contributes to gas pressure build-up that can affect the integrity of the cladding. Thus, campaigns to calculate the FGR for all Swiss reactor fuel assemblies, except for Mühleberg, have been carried out to obtain reliable average FGR data. These data (Nagra 2010, AREVA 2012, Oldberg 2009), combined with measurements of radionuclide release from SF carried out by Nagra in collaboration with PSI, Switzerland, and SKB and Studsvik in Sweden, for fuel with burnups from 45 to 65 GWd/tHM, have provided the basis for a radionuclide release model. FGR has been further investigated within the framework of an international project (Evins et al. 2021) that includes modelling and analogue studies, as well as sampling campaigns carried out for fuel in hot cells. The derivation of the IRF relevant to the analysis of radiological consequences is documented in NAB 23-10 (Johnson et al. 2023).

Dissolution of radionuclides

Once sufficient water is present inside the SF canister and has reached first the cladding and, after cladding failure, the spent fuel pellets, radionuclides start slowly to be released and dissolve in the in-canister water (Section 9.2 in NTB 23-02 Rev. 1, Nagra 2024v). They then diffuse through the breached canister shell and into the bentonite buffer in the bentonite porewater (see Chapter 7 in NTB 23-02 Rev. 1, Nagra 2024v for details). Most of the radionuclides present in the Swiss waste inventory are expected to be released in solution, see, e.g., arguments in Section 3.5 in NTB 24-18 (Nagra 2024p), with release rates constrained by the solubility limits of the corresponding elements, which are, for most elements, in the nano- to micromolar concentration range (see Section 9.2 in NTB 23-02 Rev. 1, Nagra 2024v and Chapter 4 in NTB 23-04, Hummel et al. 2023).

Upon contact with porewater, RP-HLW also dissolves extremely slowly according to the rates provided in Chapter 7 in NAB 23-09 (Curti 2022) and the released radionuclides diffuse along microcracks in the glass, first through the breached inner stainless-steel coquille and then through the breached canister shell into the bentonite buffer (see Section 9.1 in NAB 24-20 Rev. 1, Nagra 2024m and NAB 23-09, Curti 2022), where solubility limits for most elements are in the nano- to micromolar range (Section 9.3 in NTB 23-02, Nagra 2024v and Chapter 4 in NTB 23-04, Hummel et al. 2023).

Resaturation of the L/ILW emplacement caverns is a slow process. When gas pressure increases, it limits the inflow of water from the host rock into the cavern. This, in turn, limits gas production. However, once the L/ILW caverns are sufficiently saturated, radionuclides may dissolve in the porewater and start to diffuse away from the waste (see Section 9.1 in NAB 24-20 Rev. 1, Nagra 2024m and NTB 23-03, Nagra 2024v for details). Within the L/ILW caverns, sorption processes are mainly controlled by secondary minerals formed from the degraded cementitious backfill in the emplacement caverns (see Appendix D in NTB 23-07, Tits & Wieland 2023 for details).

Most of the radionuclides released to the buffer decay before they reach the surrounding host rock and most of those that do reach the host rock decay there or in the wider CRZ, as is illustrated in Chapter 8 and in more detail in Section 7.1.1.1 of NTB 24-18 (Nagra 2024p). However, some negatively charged, non-sorbing and long-lived radionuclides (such as I-129 and Cl-36) may eventually reach the biosphere.

Radionuclides released as volatile species

The most relevant radionuclide released as volatile species is ^{14}C , with a half-life of about 5,700 years, predominantly in the form of methane ($^{14}\text{CH}_4$). Transport mechanisms for $^{14}\text{CH}_4$ within the repository structures vary with time, with the degree of saturation and between the disposal areas (i.e., HLW vs. L/ILW). While the HLW near field saturates with water within a few hundred years, the L/ILW near field is expected to remain only partially saturated for a much longer period. Moreover, the HLW bentonite buffer exhibits relatively homogenous and isotropic porous media properties relevant to transport, whereas the cementitious near field of the L/ILW repository section is more heterogeneous. Volatile radionuclides migrate along the repository structures through advection and diffusion in the gas phase and also eventually dissolve in pore-water, as described in more detail in NAB 24-07 Rev. 1 (Nagra 2024w).

5.4.2 Transport in the host rock and CRZ

When radionuclides move away from the engineered barriers and into the Opalinus Clay, the concentrations of most radionuclides sharply decrease with distance because of slow diffusion and strong retention and remain significantly below the corresponding elemental solubility limits. The Opalinus Clay environment provides a diffusion-dominated transport environment that, combined with retardation (sorption) and anion exclusion facilitated by the abundance of negatively charged surfaces of clay minerals, strongly limits the rate at which transport occurs.

The ability of the Opalinus Clay to immobilise radionuclides efficiently is further enhanced by the incorporation (absorption) of some radionuclides into minerals, as supported by the findings of Bradbury et al. (2017). This process reduces their mobility and contributes to long-term immobilisation.

Transport of dissolved radionuclides

Based on evidence from studies of the Opalinus Clay and of compacted and saturated bentonite described in NTB 23-08 (Glaus et al. 2024a), alongside observations from natural clay analogues, it is expected that the long-term transport of radionuclides in the host rock and clay-rich rocks in the over- and underlying geological units will primarily be dominated by diffusion. In above cited work, diffusion experiments have been carried out for many anionic, cationic and neutral species. In addition, studies on the diffusivity of tracers in the Opalinus Clay have made it possible to estimate the diffusion rates of radionuclides using performance assessment, see NTB 24-22 Rev. 1 (Nagra 2024u). Diffusion studies in the laboratory and field experiments give consistent values for different spatial and temporal scales (Leupin et al. 2017). In the Opalinus Clay and in clay-rich rocks in the over- and underlying geological units, the phenomenon of anion exclusion has been experimentally observed, and corresponding diffusivities have been determined and documented in NAB 23-26 (Van Loon et al. 2024) and NTB 23-08 (Glaus et al. 2024a).

Temperature also impacts diffusion processes. The relationship between temperature and diffusion coefficients follows an Arrhenius-type behaviour (Van Loon 2014). When extrapolating these data to the conditions encountered in the host rock (approximate $T \approx 45 - 50^\circ\text{C}$ at 900 m depth), the expected effective diffusion coefficients for neutral species are approximately $10^{-11} \text{ m}^2 \text{ s}^{-1}$, while for anions, they are approximately $10^{-12} \text{ m}^2 \text{ s}^{-1}$. The experimentally determined low values for this parameter indicate the presence of anion-exclusion effects. This phenomenon implies that certain anions experience restricted diffusion in the host rock compared with neutral species, further impacting the overall radionuclide transport behaviour in a positive way. The understanding of these diffusion characteristics allows for an accurate assessment and prediction of radionuclide migration to be made for the prevailing conditions.

Besides the diffusion-dominated transport process, sorption is a critical factor controlling radionuclide transport in both the near field and geosphere. Scientific understanding of radionuclide retardation by sorption has been demonstrated through numerous experimental studies in the laboratory and in field studies and sorption data have been compiled in NTB 23-06 (Marques Fernandes et al. 2024a).

The potential influence of complexing organic molecules naturally occurring in the Opalinus Clay on the sorption of radionuclides has been investigated by Glaus et al. (2001). Their study examined whether the presence of organic matter affects the sorption behaviour of radionuclides of specific elements, namely Ni, Eu, and Th, on the Opalinus Clay. The findings of the study revealed that the complexing organic molecules in the Opalinus Clay had no significant impact on the sorption of Ni, Eu, and Th. In other words, the sorption behaviour of these radionuclides on the Opalinus Clay remained largely unaffected by the naturally present organic matter.

Advection in the host rock and within the clay-rich rock zones is expected to play a minimal role in radionuclide transport. Occurrences of preferential flow through fractures in the Opalinus Clay at depths within the repository zone are not expected. The presence of fractures in the Opalinus Clay has been observed, but at the depth relevant for the repository system, i.e., with the prevailing contractional tectonic influence and lithostatic pressure, the natural self-sealing mechanism prevents enhancement of any fluid flow. Experimental studies conducted 300 m underground at the Mont Terri rock laboratory (confirmed that artificially created pathways formed in the Opalinus Clay did not lead to irreversible changes in permeability (Guglielmi et al. 2025)). Despite these findings, there are remaining uncertainties that need to be considered, and minor increases in fracture transmissivity cannot be completely ruled out. Hence, some advective transport with matrix diffusion perpendicular to the direction of flow is possible. Nevertheless, modified or enhanced transmissivities under such circumstances are expected to be limited, i.e., fall below $<10^{-10} \text{ m}^2 \text{ s}^{-1}$ as is described in NTB 24-22 Rev. 1 (Nagra 2024u).

Colloids can exist in Opalinus Clay porewater, but their mobility is expected to be extremely low due to the specific properties of the host rock. The small pore sizes, typically ranging from 2 – 15 nm with limits of 2 to 100 nm (see Section 5.3.4 in NTB 24-17, Nagra 2024i), and the presence of the charged surfaces of clay platelets, contribute to the immobilisation of the colloids. In addition, the moderate ionic strength of the Opalinus Clay porewater further limits colloid movement (Mäder & Wersin 2023). Considering these factors, colloid-facilitated transport through the Opalinus Clay can be ruled out as a relevant transport mechanism for radionuclides (Section 3.3.6 in NTB 23-03, Kosakowski et al. 2023).

Transport of volatile species in the CRZ

Repository-generated gases could penetrate the CRZ as a separate gas phase but, because of the slow gas production rate, they mostly dissolve in porewater. Gas (mainly H_2) is therefore transported through the host rock, predominantly via diffusion in the porewater and to some extent by water displaced due to the gas pressure build-up in the underground structures. Some dissolved gas can reach the over- and underlying confining geological units, depending on the amount of gas generated in the repository and the transport properties of the barriers. The existence of a continuous gas phase across the host rock and its confining geological units can, however, be excluded, as documented in NTB 24-23 (Nagra 2024o). Regarding the radionuclide ^{14}C , the generally slow migration of gases together with the relatively fast decay of ^{14}C ensures that no significant release into the biosphere occurs, as can be seen from the analysis of radiological consequences in Chapter 8, with more details in Section 7.2 of NTB 24-18 (Nagra 2024p).

5.4.3 Transport in the closure system

Transport of dissolved radionuclides in the closure system

Once the closure system is sufficiently saturated, radionuclides can be transported in dissolved form through it and further along the shafts. Transport rates, however, remain slow and diffusion-dominated due to the favourable and to some extent redundant properties of the geological and engineered barriers. In particular, (i), the low hydraulic conductivity of the closure system, and especially of its various seals, as well as the limited capacity of the surrounding low-permeability rock to supply water, limit the rate of water flow and advective transport of dissolved radionuclides through the closure system, and (ii), the high sorption capacity of the clay materials within the seals and backfill further retards the transport of many radionuclides (Section 5.2.2 in NAB 23-21, Martin et al. 2023 and Chapter 4 in NAB 21-12 Rev. 1, Nagra 2021b).

Transport of volatile species in the closure system

As noted above, saturation with water occurs within a few hundred years in the HLW near field. The V3 shaft seals also saturate relatively rapidly. In contrast, the L/ILW near field, the sealing elements and much of the closure system of this repository section are expected to remain only partially saturated for a much longer period, facilitating the transport of repository-generated gases including volatile radionuclide species, especially ^{14}C in the form of methane. As detailed in Section 4.3 in NAB 24-07 Rev. 1 (Nagra 2024w), these gases are, however, largely contained by the V3 seals, which have a low gas permeability once saturated. The gas within the closure system gradually dissolves in the porewater within the closure system itself and within the surrounding rock.

5.4.4 Transport to and in the biosphere

The few radionuclides that leave the CRZ are eventually transported via aquifers into a local aquifer of the surface environment, which is part of the biosphere. Note that, in the quantitative assessments in Chapter 8, this transport, which is expected to further delay, disperse and dilute the radionuclides, is not taken into account.

Consistent with ENSI Guideline G03 (ENSI 2023), radiological consequences of radionuclide releases to the biosphere are evaluated by assessing potential doses to humans. The purpose of biosphere modelling described in NAB 24-06 (Nagra 2024n) is to provide the means to carry out this assessment and to demonstrate that the disposal system meets the quantitative protection criteria, thus protecting humans and the environment. Specifically, biosphere modelling is used to provide sets of radionuclide-specific biosphere dose conversion factors (BDCFs), that are applied to calculated release rates to the biosphere in order to obtain effective dose rates that can be compared with regulatory criteria.

The biosphere model (i) represents the behaviour of the released radionuclides in the human habitat, and (ii) determines the potential radiation exposure for a representative individual from the most affected population group based on major exposure pathways.

The modelling aims to be realistic in terms of representing radionuclide migration in the biosphere, but pessimistic in evaluating potential doses. This includes the assumption that the aforementioned population group is making maximum use of local resources through a self-sufficient lifestyle. To help avoid undue pessimism within this assumption, the assumed behaviour of the population group aims to be realistic and credible, based on present-day behaviour relevant to the assumed local and climatic conditions.

5.5 Features, events and processes and their audit

A prime concern in safety analyses and in safety assessment is the issue of phenomenological “completeness”, that is, the need to demonstrate that:

- features, events and processes (FEPs) and their interactions that could affect post-closure safety have been thoroughly identified, and
- these FEPs have been adequately accounted for in the safety assessment.

Although it is impossible to prove beyond all reasonable doubt the completeness of a safety assessment, measures can be taken to ensure the inclusion of a comprehensive set of potentially relevant phenomena and, equally importantly, to ensure that the safety-relevant phenomena have been represented appropriately in the assessment.

In the safety assessment that supports the general licence application, an assurance of phenomenological completeness and appropriate FEP management is provided by:

- the development of a comprehensive database of FEPs that could be relevant to the safety assessment, and
- a careful audit of the phenomena included in the safety assessment against the FEP database.

Nagra’s FEP database has been developed iteratively over many years, largely independently of the main safety assessment process. Each version has been reviewed by internal and external experts and compared with other relevant FEP databases compiled by national organisations in other countries and by international bodies.

A FEP audit has been performed and documented in a supplementary volume of NAB 24-20 Rev. 1 (Nagra 2024l). It considers whether all the FEPs in the FEP database have been adequately treated in the safety assessment and whether omission of any FEPs is well justified. In particular, the report indicates which FEPs can be excluded a priori, based on the existing scientific understanding or their negligible likelihood and/or consequences, or on the grounds that they are outside the scope of the safety assessment, and which FEPs are considered either as part of the performance assessment, or directly in the analysis of radiological consequences. The audit report can thus be viewed as a “filter” as to which FEPs are to be considered in detail in the safety assessment and which can be excluded from the outset.

FEPs can be included in the safety assessment in different ways, for example:

- in the description of the safety scenarios, and/or
- in the models used in performance assessment and in the analysis of radiological consequences, either explicitly, or implicitly through the selection of parameter values for those models.

Where FEPs are excluded, it is shown that these are either outside the scope of, or irrelevant to, the safety assessment in support of the general licence application. This helps to ensure that no relevant FEPs have been overlooked and that there is a sound justification for the exclusion of any FEPs.

5.6 Models, codes and databases

The models, codes and databases that support the present and previous safety assessments are the result of considerable research and development work.

The models and codes used are discussed and their reliability described in corresponding specific reports, e.g., Nagra (2024o, 2024k, 2024n, 2024p). This implies, in accordance with the safety assessment principles (Section 4.1):

- Accuracy and verification:

In general, verification of the codes, which provides confidence in the accuracy of the results, is documented in the respective reports. Modifications made to the codes over the course of Nagra's programme have also been tested and documented.

- Applicability and validation:

Validation of codes and databases involves testing their consistency, as far as possible, with available information, including laboratory and field experiments, general scientific understanding, experience from other safety assessments, and observations from nature. Natural analogues and palaeohydrogeological models have been used to evaluate and, where possible, reduce uncertainties arising from the temporal scales of concern. Finally, use has been made of expert judgement and critical internal and external review. Where available, alternative conceptual models have been considered.

- Correct application:

Care has been taken that the codes and models used in support of the post-closure safety case have been applied correctly and in line with their intended use.

6 Performance assessment

Aims of the chapter:

- To describe performance assessment (PA), which provides an evaluation of the functionality of each barrier component and of the entire system, and that covers the expected evolution and uncertainties.
- To develop arguments that support the claims of the safety concept.
- To systematically assess uncertainties and manage uncertainties during PA.
- To screen the PA outcome in order to identify potentially safety-relevant deviations in performance as input to safety scenario development.
- To ensure that potential lines of evolution, scenarios, and variations in performance are comprehensively covered.

Performance Assessment (PA) represents the first stage in the general safety assessment workflow aimed at producing a robust post-closure safety case. Following international standards of transparent and traceable assessment workflows (OECD/NEA 2004), PA addresses the post-closure safety of the repository at a level of simplification that bridges the gap between the real-world system and its highly abstracted representation in the analysis of radiological consequences.

The auditability of the assessment workflow is ensured with a traceable method of proof. For this, claims are formulated, assigning one or more safety functions to each component of the multi-barrier system and to the repository system as a whole. The claims are substantiated by a suite of arguments. A claim can be said to be robust if it is well substantiated using sound arguments, which are themselves supported by convincing evidence, such as: (i) *empirical knowledge*, including experience from other waste management programmes, (ii) *dedicated scientific and technical databases* that have been elaborated specifically in the context of the repository project and (iii) a *safety-oriented repository design*.

Traceable handling of uncertainty is an indispensable element of PA and of overall safety assessment. As part of uncertainty management, PA provides evidence that the modelling tools and approaches applied in the quantitative assessment of barrier performance at component and total-system levels adequately represent the safety-relevant processes and phenomena.

It is the role of PA to screen the possible paths of repository performance and to identify safety-relevant deviations from expected repository performance as input to the development of safety scenarios for radiological consequence analysis. From the PA perspective, the possible paths of repository evolution are classified in terms of expected performance of the multi-barrier system and deviations thereof.

Performance assessment workflow

Fig. 6-1 shows how PA is positioned within the workflow for the post-closure safety case, highlighting, in the orange box, the main elements of the performance assessment process. These elements comprise:

- an assessment of barrier performance for each of the pillars of safety of the multi-barrier system,
- an assessment of the entire system (total system performance), to account for thermo-hydro-mechanical and chemical (THM-C) interactions between the individual barrier components,
- an assessment of uncertainties associated with the simplified representation of the real-world repository system in an abstracted PA framework (uncertainty propagation associated with the abstraction process), and
- an assessment of deviations from expected repository performance by barrier and by interaction (“screening of assessment scenarios”).

The key phenomena relevant to the evolution and performance of the repository system are considered in the PA, including those that could lead to deviations from expected repository performance.

These phenomena are identified in NAB 24-20 Rev. 1 (Nagra 2024m), which synthesises Nagra’s understanding of the initial state and expected post-closure evolution of the repository. As noted in Section 4.5, this understanding has developed iteratively over many years, with each iteration having been peer-reviewed and informed by Nagra’s extensive RD&D programme.

The key phenomena identified in NAB 24-20 Rev. 1 (Nagra 2024m), their relevance to system evolution and performance, and their potential to lead to performance deviations are listed in App. B (Tab. B-1). This appendix also indicates the sections of the PA synthesis in NTB 24-22 Rev. 1 (Nagra 2024u) where these phenomena are addressed.

Assessment of barrier performance and of the performance of the total system employs a combination of qualitative argumentation and quantitative analyses. The latter can involve the evaluation of performance metrics or indicators, considering all identified sources of uncertainty, and a comparison of these metrics or indicators with performance targets or evaluation scales. Performance targets, if met, may then indicate, for example, that uncertainties do not compromise the expected performance of a barrier or of the total system.

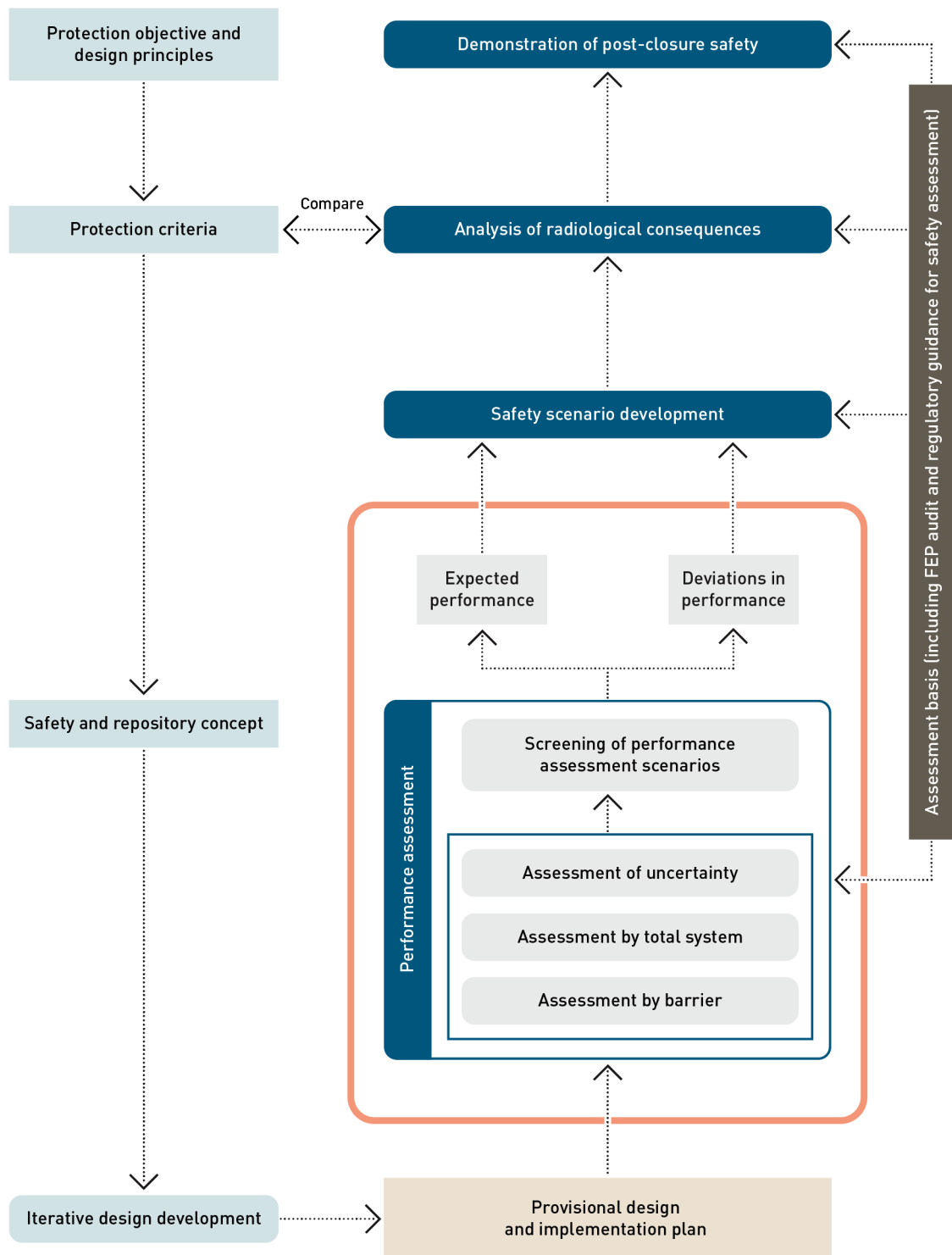


Fig. 6-1: Workflow for the post-closure safety case, highlighting, in the orange box, the main elements of the performance assessment process

6.1 Assessment of barrier performance

The current safety and repository concept is set out in Chapter 3. The main barriers and other features contributing to the safety functions are termed “pillars of safety”. Fig. 3-3 and Fig. 3-4 illustrate schematically how the pillars of safety contribute to the various safety functions. PA provides detailed descriptions of the pillars of safety and their intended functionalities under relevant conditions.

In PA, the expected performance of the multi-barrier systems is expressed in terms of a set of claims concerning the contribution of each component of the multi-barrier system and of the repository system as a whole to the repository safety functions, in line with the current safety concept. Each claim is formulated based on, and examined through, a comprehensive analysis of empirical data, experimental results, and modelling evidence. The most important claims, arguments, and evidence in the assessment of the individual pillars of safety (assessment by barrier) are presented in the following sections. A more detailed discussion of the corresponding claims, arguments, and evidence is given in Chapter 3 of NTB 24-22 Rev. 1 (Nagra 2024u).

6.1.1 Deep underground location of the waste emplacement rooms in a stable geological environment

The deep underground location of the repository in a setting that is not at risk of disruptive geological events and processes and with all access routes sealed ensures that the radioactive waste is safely enclosed for a very long time. Careful selection of a favourable repository site at a suitable depth contributes to the safety functions of isolation of radioactive waste from humans and the environment, immobilisation, retention and slow release of radionuclides and long-term stability of the repository system regarding the long-term geological and climatic evolution, encompassing erosion, neotectonics, and glaciations. The favourable *overall geological setting* of the repository guarantees stable conditions over geological time periods.

6.1.2 Containment-providing rock zone (CRZ)

The CRZ encompasses several complementary barrier elements, addressed in PA as separate assessment objects, each of them contributing to one or several safety functions of the geological barrier. The Opalinus Clay represents the diffusion-dominated primary geological barrier and provides long-term confinement of the waste, ensuring a stable geochemical and geomechanical environment. The upper and lower confining geological units serve as additional transport barriers.

The geological barrier, in particular the Opalinus Clay, has been subjected to extensive investigations for more than 20 years, comprising empirical studies and dedicated site investigations, with flow and transport experiments at a borehole scale and at a core scale and including experiments within the scope of Nagra’s recent deep drilling campaign as described in the Geosynthesis of Northern Switzerland in NTB 24-17 (Nagra 2024i). There is ample evidence that the intact Opalinus Clay represents a diffusion-dominated transport barrier with sufficient vertical and lateral extent and that the confining geological units also have favourable properties.

The diffusive properties of the Opalinus Clay and the confining geological units are well characterised as documented, e.g., in NTB 23-08 (Glaus et al. 2024a) and NAB 23-26 (Van Loon et al. 2024).

Excavation Damaged Zone (EDZ)

A large body of evidence exists, showing that the shape, extent, and characteristics of the EDZ in the Opalinus Clay around the HLW emplacement drifts and the L/ILW emplacement caverns impairs the barrier functions of neither the CRZ nor the engineered barriers. The following points summarise the main arguments (for details, see Section 3.3.3 in NTB 24-22 Rev. 1, Nagra 2024u):

- the choice of the excavation method and tunnel support limits the extent of the EDZ, such that it does not significantly impact the transport distance of radionuclides through the Opalinus Clay,
- simulations show that the resealed EDZ around the HLW drifts and L/ILW caverns does not represent a significant axial transport path for dissolved radionuclides, the possibility of which is, in any case, minimised by the dead-end architecture of the HLW drifts and L/ILW caverns,
- the hydromechanical evolution and self-sealing of the EDZ limits microbial activity,
- the higher porosity within the EDZ around the HLW drifts and L/ILW caverns provides an additional storage volume for repository-generated gas, further limiting gas pressure build-up,
- the existence of an EDZ around the HLW emplacement drifts reduces the loads on the drift lining (Geomechanica Inc. 2025), and
- the evolution of the EDZ around the backfilled emplacement rooms in response to eventual failure of the lining is limited, because the swelling pressure of the bentonite and the low compressibility of the L/ILW backfill compensate for the loss of strength of the lining.

6.1.3 SF and RP-HLW matrices

SF and its cladding

SF and its cladding, which are illustrated in Fig. 6-2, ensure the immobilisation, retention and slow release of radionuclides. Under repository conditions, SF, which mainly consists of UO_2 , is chemically stable. Due to its very slow dissolution rate, it does not impair the chemical integrity of the canister or of the bentonite buffer. The main arguments supporting this claim are:

- the SF pellets and the surrounding cladding are chemically stable and do not chemically interact with the canister and bentonite, or do so only to a very limited extent (Section 9.2 in NTB 23-02 Rev. 1, Nagra 2024v and Chapter 5 in NAB 23-10, Johnson et al. 2023),
- the majority of radionuclides are retained within the UO_2 and MOX mineral phases (Kleykamp 1985, Thomas et al. 1992, Walker et al. 1996, Johnson et al. 2023 and references therein),
- the most rapidly released fraction of the radionuclide inventory of SF (the instant release fraction, or IRF) is small compared with the total radionuclide inventory (Section 4.2.1 in NTB 23-02 Rev. 1, Nagra 2023b and Chapter 1 in NAB 23-10, Johnson et al. 2023), and
- following canister breaching, most of the radionuclides will be released only slowly due to limited solubility of the waste matrix and low degradation and dissolution rates after water has entered the canister (Cui et al. 2010, Ekeröth et al. 2020, Spahiu 2021, Nagra 2024v, Johnson et al. 2023 and references therein).

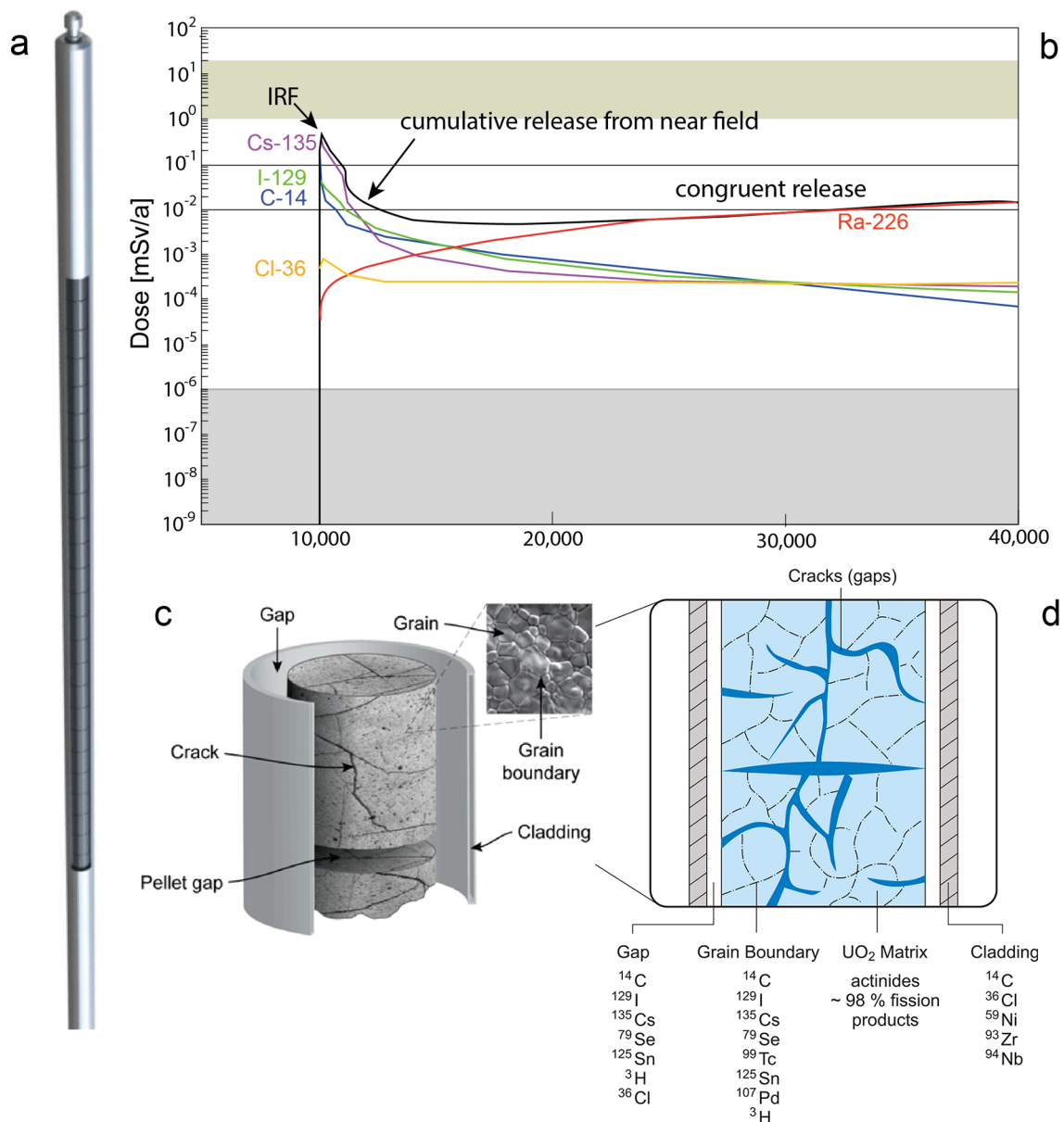


Fig. 6-2: Main elements of the spent fuel that are assessed

a) Schematic view of a fuel rod, illustrating fuel pellets in Zircaloy cladding. b) Illustration of radionuclide release rates from the spent fuel following canister breaching, converted to dose rates using biosphere dose conversion factors. c) The macro- and micro-structure of a fuel pellet after its irradiation. d) conceptual distribution of radionuclides in a spent fuel rod (from Johnson 2014, redrawn after Johnson & Tait 1997).

RP-HLW

RP-HLW (Fig. 6-3) contributes to the immobilisation, retention and slow release of radionuclides. Under repository conditions, RP-HLW is chemically stable and does not impair the chemical integrity of the canister or of the bentonite buffer. RP-HLW is the product of the treatment and reprocessing of SF, which is carried out to extract fissionable U and Pu for re-use in NPPs. Research on RP-HLW has been performed for decades and has been followed by Nagra, e.g., in

the scope of the NF-PRO projects (e.g., Grambow et al. 2008). Current understanding of the aqueous dissolution of RP-HLW and of the rates at which this process occurs have been synthesised in NAB 23-09 (Curti 2022). The borosilicate glass matrix of the RP-HLW provides a chemical environment in which most radionuclides are immobilised and are released very slowly. This is mainly due to the chosen glass composition and to the formation of a passivation layer, which provides a barrier that protects the glass from dissolution (Gin et al. 2016, 2020a, 2020b, Nagra 2024v, Curti 2022 and references therein).

The stainless-steel flasks containing the RP-HLW are compatible with the glass, the disposal canister and the bentonite, mainly due to limited chemical interactions (Section 9.3 in NTB 23-02 Rev. 1, Nagra 2024v).

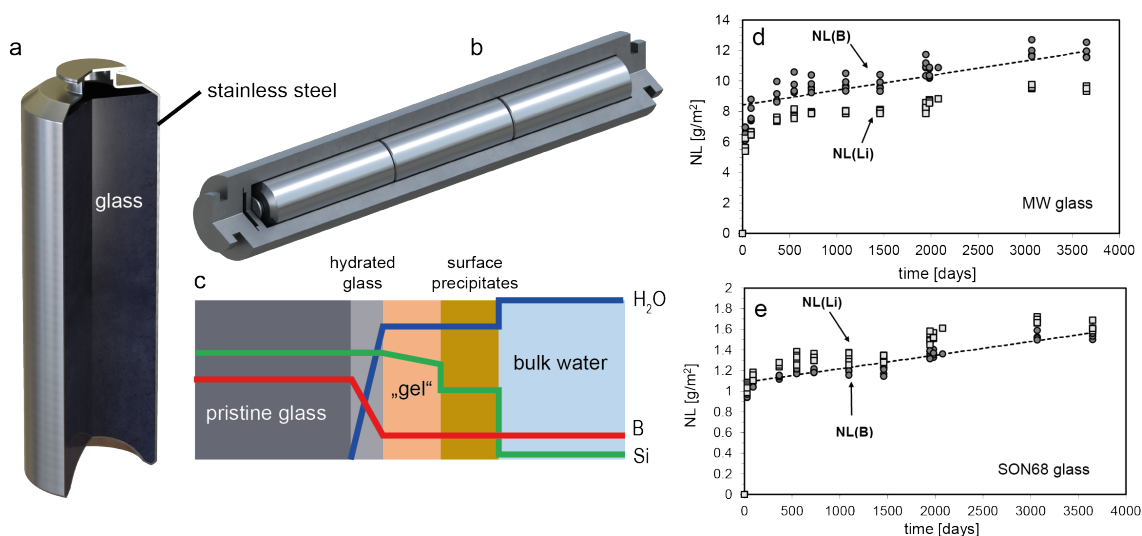


Fig. 6-3: Main elements of the vitrified waste from reprocessing that are assessed

a) Stainless steel flask for RP-HLW. b) Disposal canister for 3 RP-HLW flasks. c) Schematic cross-section through altered borosilicate glass in contact with the leaching aqueous solution, showing idealised profiles for Si, B and water concentrations. d) and e) Normalised mass loss (NL) for the non-active borosilicate glass “MW” and “SON68” glass determined from long-term experiments at PSI, Switzerland (Curti 2022).

6.1.4 SF and RP-HLW disposal canisters

The SF and RP-HLW disposal canisters (Fig. 6-4) provide complete containment of radionuclides for at least the early post-closure phase, when the HLW near field is exposed to distinct thermal and hydraulic disequilibrium conditions. The main arguments are that:

- the canister has been designed such that structural integrity is ensured for at least the minimum 1,000 years required by regulations, even under the assumption of unfavourable corrosion rates, material degradation and external state conditions (Section 4.3 in NTB 24-20, Nagra 2024d),
- manufacturing, welding, heat treatment, inspection and handling procedures can be tailored to minimise the occurrence of undetected defects and residual stresses (Chapter 3 in NTB 24-20, Nagra 2024d), and

- the gas production rate due to corrosion of the canisters is limited compared with the gas storage and transport capacity of the rock and repository structures, such that excessive pressure build-up in the repository, is avoided (Nagra 2024o).

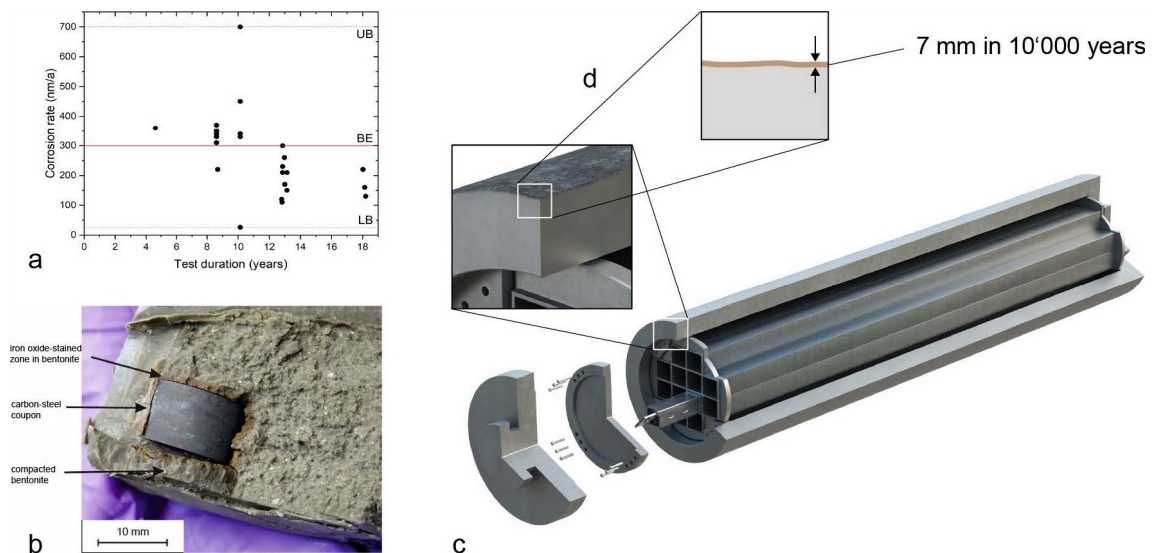


Fig. 6-4: Illustration of a disposal canister for SF and RP-HLW

A canister lifetime of at least 1,000 years is ensured by appropriate design even under the assumption of unfavourable corrosion rates.

(a) Laboratory measurements of steel corrosion rates in anaerobic bentonite saturated with artificial Opalinus Clay porewater as part of Nagra's RD&D programme (Diomidis & Reddy 2022). (b) In-situ experiments at the Mont Terri rock laboratory confirm the low corrosion rates (Diomidis & Reddy 2022). (c) SF canister design (Nagra 2024d). (d) A corrosion depth of about 7 mm in 10,000 years under in-situ conditions is expected (Nagra 2024d).

Furthermore, the products of canister degradation (corrosion) are not detrimental to other engineered or geological barriers, because chemical interactions between corrosion products and the buffer do not significantly reduce its function as a hydraulic barrier (Section 8.2.3 and Appendix A in NTB 23-02 Rev. 1, Nagra 2024v).

6.1.5 Bentonite buffer (including interface with the lining)

The properties of the bentonite buffer (Fig. 6-5) have been extensively investigated (e.g., Nagra 2019, Leupin et al. 2014, Seiphoori 2015), resulting in a large body of evidence supporting safety-related claims. In particular, the following contribute directly or indirectly to the efficient retention of any radionuclides released following canister breaching:

- the inherently low hydraulic conductivity of the buffer at full saturation,
- its self-sealing ability,
- its marked sorption capacity,
- the micro-porous environment that it provides at full saturation, and
- its ability to transport gas without fracturing.

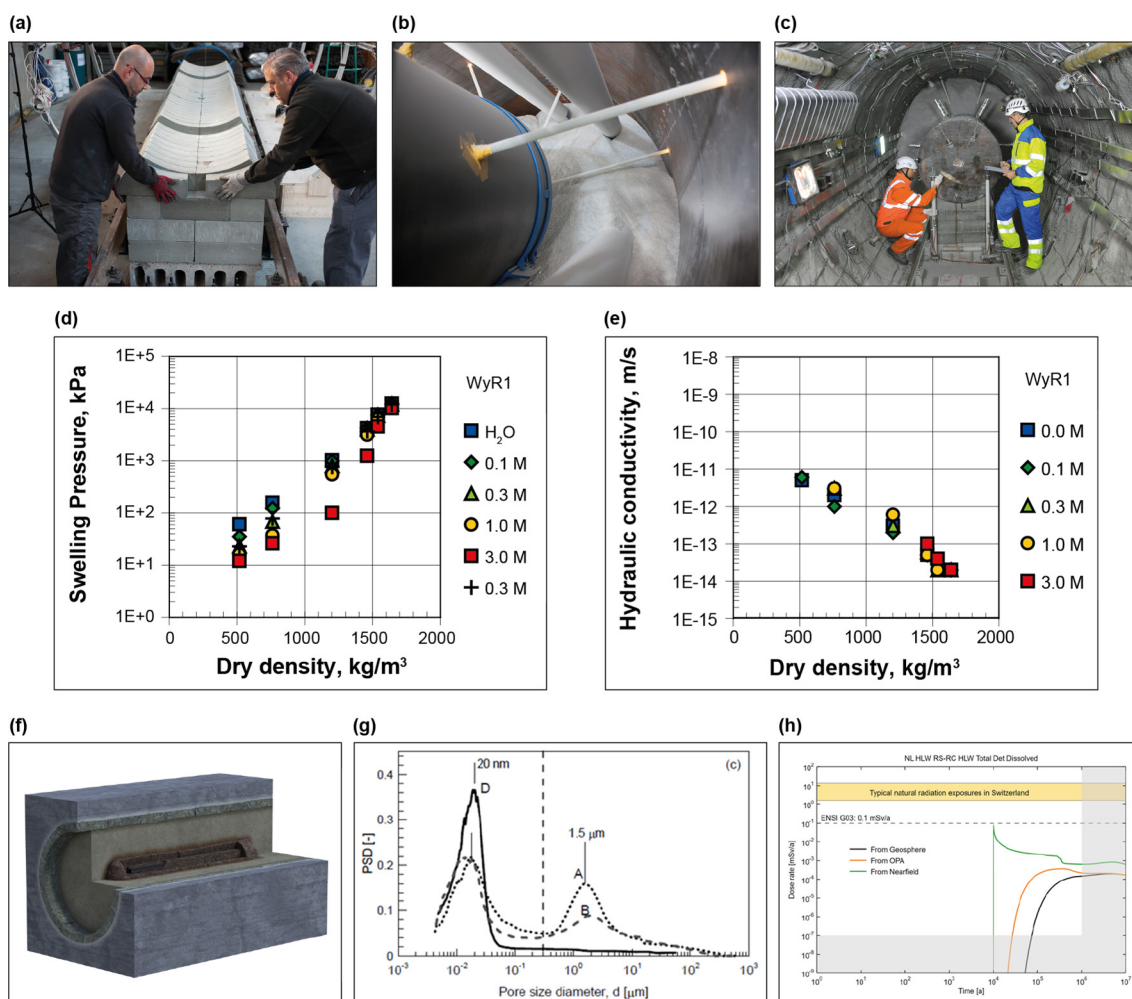


Fig. 6-5: Bentonite buffer, which has been extensively investigated in small- and large-scale experiments over the last few decades

The handling and emplacement of the bentonite buffer have been successfully demonstrated at a large scale (a-c). It has been shown that the hydraulic properties and the swelling pressure are related to the emplaced dry density (x-axis of (d) and (e); Karland et al. 2006) and that these properties are not significantly affected by the expected thermohydraulic evolution of the near field (Leupin et al. 2014). The interfaces with the canister and with the cementitious rock support (lining) evolve very slowly as these processes are limited by diffusion (f). The evolving interfaces do not impair the functionality of the bentonite buffer or of the host rock. Upon saturation the pore-size distribution is shifted due to the swelling processes towards nanopores (g) (Seiphoori et al. 2014), a key process defining macro properties such as hydraulic conductivity. The excellent retention properties of bentonite are an important contributor to limiting the radionuclide flux into the host rock (h).

The bentonite buffer has also been shown to provide favourable environmental conditions contributing to the longevity of the HLW canisters by providing a long-lasting, mechanically homogeneous, and chemically favourable interface. The microporous environment within the buffer provides sufficient limitation of microbial activity, and the buffer conducts heat sufficiently well. The waste package, canister, and buffer thus do not experience conditions detrimental to the performance of their safety functions (Nagra 2019).

The geomechanical and the chemical evolution of the cementitious lining around the HLW emplacement drifts do not significantly impair the functionality of any engineered and geological barriers. Specifically, the loss of strength of the lining does not significantly affect the safety functions of the buffer or the intended lifetime of the canister and does not significantly affect the extent and hydraulic properties of the EDZ (Geomechanica Inc. 2025). Indeed, experimental evidence shows that chemical interactions of the lining with the bentonite buffer are spatially limited and thus do not significantly affect the safety functions of the buffer (Chapter 3 in NAB 22-34, Prasianakis et al. 2022 and Yokoyama et al. 2021). These findings are supported by reactive transport modelling (Section 6.7.2 in NTB 23-02 Rev.1, Nagra 2024v, for details about the models see NAB 23-33, Kosakowski 2023).

6.1.6 L/ILW cementitious near field

The major components within the L/ILW near field are cement-based, compatible with each other and contribute to an alkaline environment upon contact with water from the Opalinus Clay (Sections 3.3 – 3.5 and Appendix A in NTB 23-03, Kosakowski et al. 2023 and Chapter 5 in NAB 20-11, Kosakowski et al. 2020 and references therein).

The cementitious near field contributes to the retention of radionuclides because, by design, the cement paste contains C-S-H phases that sorb radionuclides (Nagra 2021c, 2024u, Tits & Wieland 2023, Kosakowski et al. 2023).

The concrete disposal containers and cementitious container infill mortar contribute to the initial structural integrity of the waste packages because:

- by design, the concrete containers are reinforced and constructed of high-quality concrete, and
- filling of the disposal container with a container infill mortar contributes to limiting void space within the L/ILW cavern.

The grain-supported cavern backfill mortar (Fig. 6-6) contributes to the mechanical integrity of the L/ILW caverns by filling the void space and because, by design, the backfill mortar has grain-to-grain contact (Jacobs et al. 1994, Nagra 2021a, 2024u).

Finally, the porosity and chemical composition of the cavern backfill mortar contributes to the compatibility of the multi-barrier system. Specifically, the backfill mortar provides storage volume for repository-generated gas and a continuous porosity to transport that gas (Nagra 2024o). The highly alkaline conditions present in the backfill and throughout the near field also limit gas production by suppressing corrosion rates and microbial activity (Section 3.1.2 in NTB 21-02, Nagra 2021a; Section 3.3.1 in NTB 23-03, Kosakowski et al. 2023; Chapter 6 in NTB 13-04, Warthmann et al. 2013 and Section 3.4.3 in NTB 24-22 Rev. 1, Nagra 2024u).



Fig. 6-6: Grain-supported cavern backfill mortar for the L/ILW caverns

The figure shows results from a large-scale test of backfilling techniques and properties of the mortar.

6.1.7 Closure system

As described in Section 3.3, the V1-HLW seal ensures the mechanical and chemical integrity of the bentonite buffer, because, by design, the seal prevents buffer erosion and is constructed mostly of materials that are in chemical equilibrium with the surroundings (Section 4.2 in NAB 21-12 Rev. 1, Nagra 2021b and Chapter 6 in NAB 23-21, Martin et al. 2023). The V1-HLW seal also contributes to a retarded release of dissolved and volatile radionuclides, being designed in such a way that gas release along the EDZ and lining is limited (see Section 5.3 in NAB 21-12 Rev. 1, Nagra 2021b).

The V2-HLW seal contributes to a retarded release of radionuclides and prevents any safety-relevant interaction between the HLW and L/ILW repository sections (Section 2.1 in NTB 24-23, Nagra 2024o and Section 1.2 in NAB 23-21, Martin et al. 2023) due to a design that limits the transport of volatile and dissolved radionuclides after full saturation of the seal. The V1-L/ILW and V2-L/ILW seals also contribute to a retarded release of dissolved radionuclides from the L/ILW emplacement caverns. Since the permeability of the seals is low, transport of dissolved radionuclides across the sealing elements predominately occurs by diffusion (Chapter 5 in NAB 23-21, Martin et al. 2023).

Both, the V1-L/ILW and V2-L/ILW seals also contribute to the compatibility of the barrier components because:

- the low gas entry pressure of the sand-bentonite mixture used in the sealing elements ensures efficient gas release (see Section 3.5 in NTB 16-07, Manca 2016), preventing the occurrence of damaging gas overpressures (Chapter 6 in NTB 24-23, Nagra 2024o), and
- chemical interactions between the materials used in the closure system and the surrounding Opalinus Clay are limited (Section 2.3 and Chapter 3 in NAB 22-24, Prasianakis et al. 2022; Chapter 6 in NAB 23-21, Martin et al. 2023; Chapter 6 in NTB 23-02 Rev. 1, Nagra 2024v and Section 3.3.4 in NTB 23-03, Kosakowski et al. 2023).

The V3 shaft seal contributes to the isolation of radioactive waste from the biosphere because, by design, the hydraulic conductivity is so low that radionuclides are only released by diffusion across the V3 seal. Thus, the V3 seal limits the release of dissolved and gaseous radionuclides because, by design, the transport of gaseous and dissolved radionuclides is largely prevented after full saturation of the seal (Section 7.2.2 in NAB 23-21, Martin et al. 2023 and Chapter 6 in NTB 24-23, Nagra 2024o). Furthermore, the V3 seal together with the deep geological location, makes inadvertent intrusion unlikely and also contributes to isolating the repository from events at the surface.

The VF1 and VF2 backfills provide gas storage volumes that contribute to a low gas-pressure build-up, and, given their low hydraulic conductivity, delay and limit water flow into the operations tunnel. The stiff backfills also contribute to the mechanical integrity of the host rock by providing structural support. Finally, because of their clay content, the VF1 and VF2 backfills contribute to the retardation of radionuclide release along the tunnel system (Section 3.2 in NAB 21-12 Rev. 1, Nagra 2021b and Section 2.2.4 in NAB 23-21, Martin et al. 2023).

6.2 Assessment of the performance of the total system

Further PA claims are formulated for the performance of the repository system as a whole, including repository-induced processes and phenomena that develop over spatial scales covering multiple barriers. The most important claims, arguments and evidence in the assessment of the total system are presented in the following sections. A more detailed discussion of these and other claims, arguments and evidence is given in Chapter 4 of NTB 24-22 Rev. 1 (Nagra 2024u).

6.2.1 Approach and critical aspects

In NTB 24-22 Rev. 1 (Nagra 2024u), numerical simulations are used to judge the performance of the repository using predefined performance targets. Critical aspects considered in these simulations include, for example, the integrity of the multi-barrier system, scenarios for the potential release of radionuclides, thermal and gas-related factors, and geological attributes. The simulations rely on conceptualisations of geological and hydrogeological conditions. Based on the geological characterisation described in the Geosynthesis of Northern Switzerland in NTB 24-17 (Nagra 2024i) and references therein, it is assumed in these conceptualisations that diffusion is the dominant transport mechanism in the CRZ, with advective transport along steeply dipping faults not being relevant. Sedimentary features, such as hard beds in the upper and lower confining geological units, do not act as lateral release paths.

Critical aspects noted above are screened for their relevance to repository performance with a model-based assessment that uses a code capable of coupling thermal and hydraulic processes (e.g., TOUGH). Parametric uncertainty is considered probabilistically through Monte Carlo simulations. This necessitates a model setup with simplified representations of complex coupled processes to ensure that the demands on computational resources are manageable. Findings from

extensive 3D simulations confirm decoupling between HLW and L/ILW repository sections, allowing independent analysis. Consequently, separate model setups are employed for each repository section. Performance indicators include temperature (T) and pore pressure (P) at specific points, as well as the fluxes of tracers (F) representing conservative dissolved and volatile species, which are crucial for assessing safety functions related to radionuclide immobilisation and barrier integrity.

6.2.2 Expected thermohydraulic performance of the repository system

The total system performance mainly depends on the thermohydraulic (TH) evolution of the repository system. Detailed TH analyses are described in NTB 24-22 Rev. 1 (Nagra 2024u) and NAB 24-25 (Nagra 2024k), and the results are summarised briefly in the following paragraphs.

Fig. 6-7 shows temperature transients simulated with a 3D repository model over a one-million-year period, including the construction, ventilation¹⁴, and post-closure phases. In HLW emplacement drifts, temperature spikes due to heat-generating waste, then decreases back to ambient temperature. Sections of the operations tunnel near the emplacement area also see temperature increases due to their proximity.

In the L/ILW emplacement caverns, temperature returns from the relatively low values caused by ventilation to ambient within ~3,000 years and does not exceed it. The central repository area (not shown in the figure) shows no significant temperature changes due to heat generated by HLW.

¹⁴ The temperature drop from ~43 °C to 28 °C seen in the figure occurs during the ventilation period.

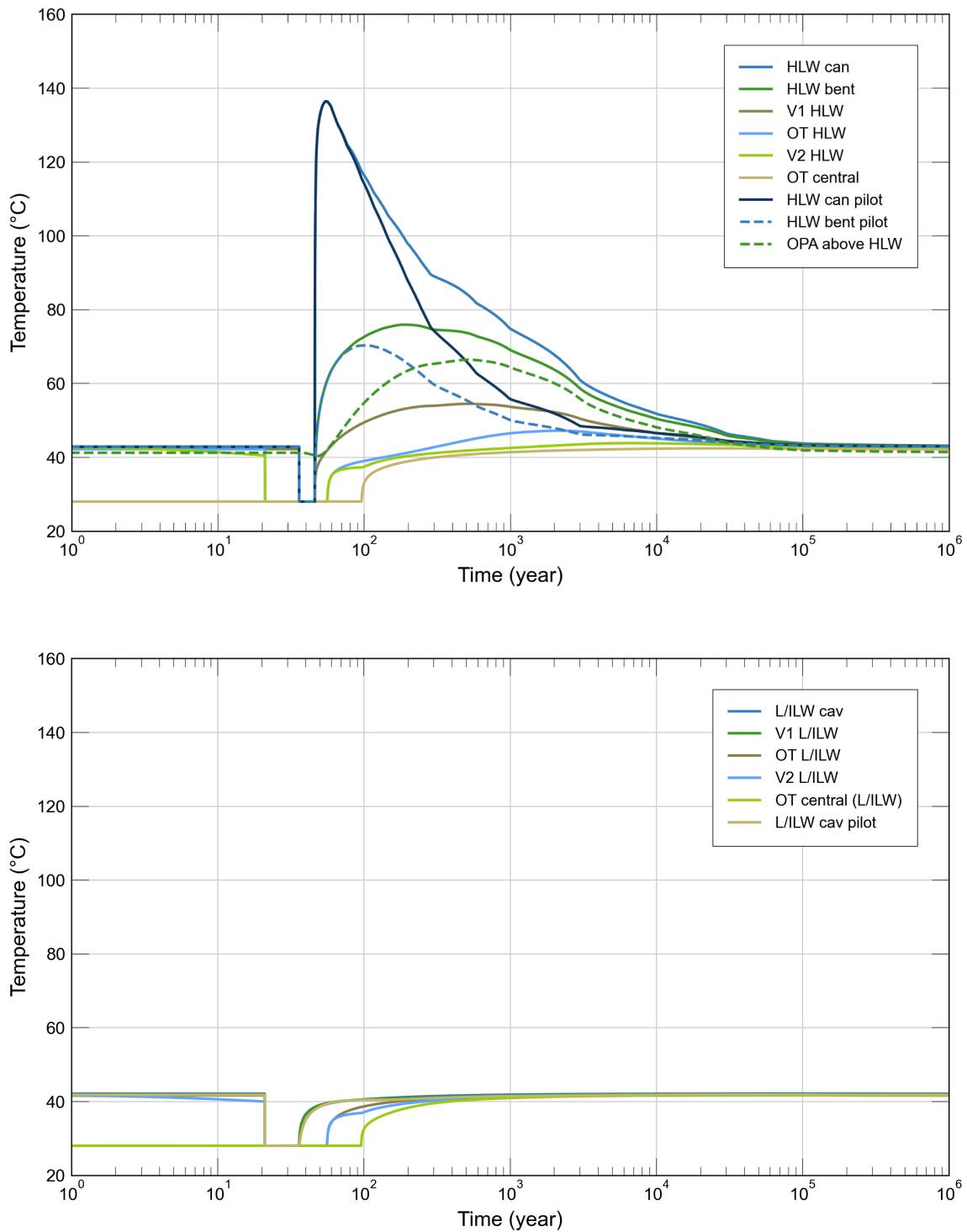


Fig. 6-7: Assessment of total system: transients of temperature in the HLW repository section (top) and in the L/ILW repository section (bottom)

V1: cavern or drift seal, OT: operation tunnel, V2: repository section seal, pilot: pilot facility. For more details see Section 5.3 in Nagra (2024o).

Fig. 6-8 shows gas saturation transients from the same simulation. In the HLW repository section (top figure), local gas saturation depends on the backfill material. The operations tunnel, back-filled with high-porosity material, shows increasing gas saturation over time, with the upper parts of this tunnel accumulating more gas. Sections backfilled with bentonite fully saturate with water. HLW drifts saturate with water between 200 and 300 years after emplacement, with the V1-HLW and V2-HLW seals fully saturating after ~500 and ~3,000 years, respectively. Gas generation is expected to increase after 10,000 years, when the disposal canisters must be assumed to breach and their inner surfaces start to corrode, leading to a gas phase formation in the buffer surrounding the canisters.

In L/ILW emplacement caverns (bottom figure), gas saturation decreases as porewater enters until, after ~10,000 years, it increases again due to the corrosion of metals and the degradation of organics. Gas saturation in the operations tunnel increases continuously after backfilling. The central repository area (not shown) also undergoes increasing gas saturation after closure. A continuous gas pathway forms along the repository level emplacement caverns, galleries, and seals over a one-million-year period. The V3 seal in the shaft, composed mainly of compacted bentonite, fully saturates with water after ~4,000 years.

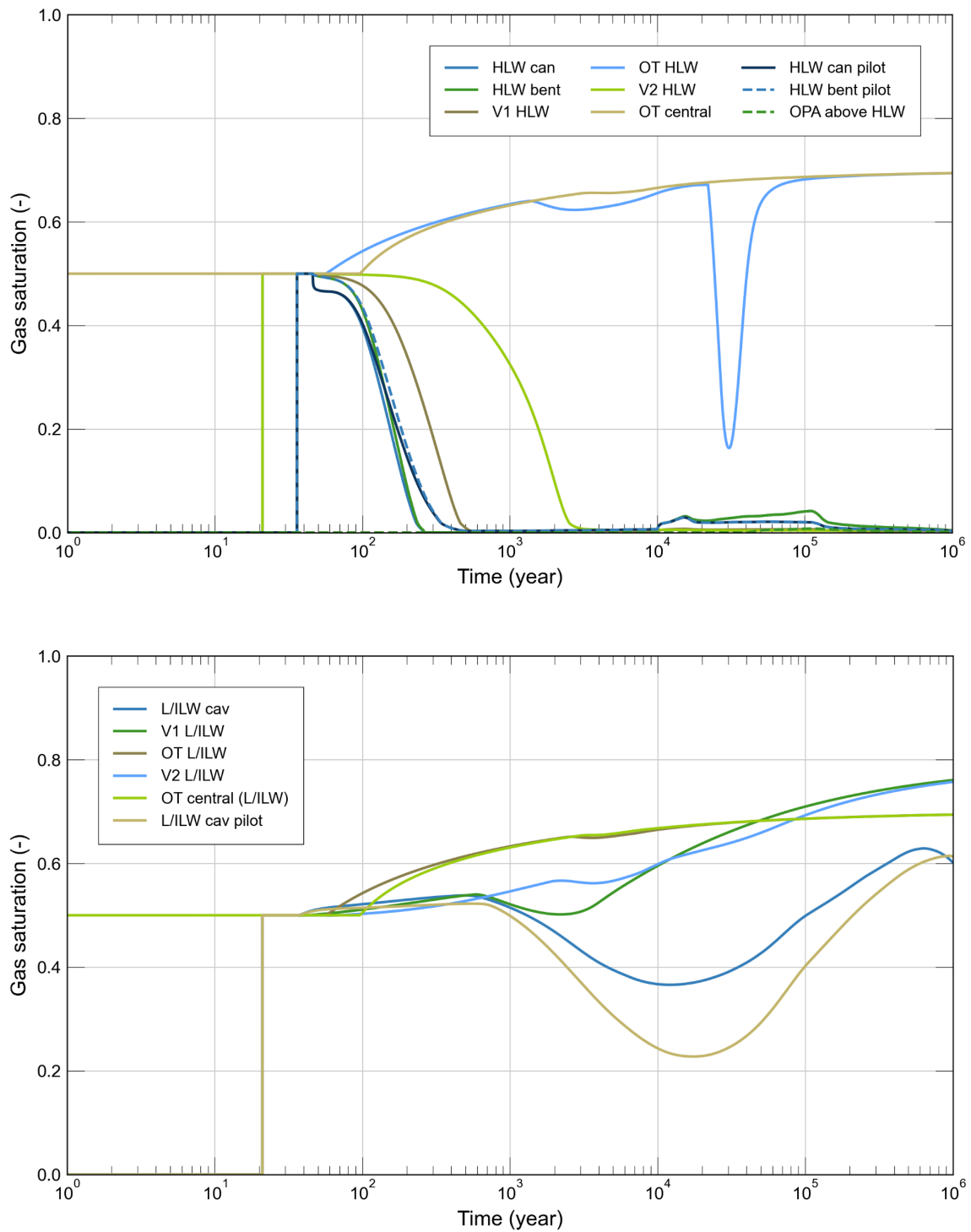


Fig. 6-8: Assessment of total system: transients of gas saturation in the HLW repository section (top) and in the L/ILW repository section (bottom)

V1: cavern or drift seal, OT: operations tunnel, V2: repository section seal, pilot: pilot repository. For more details see Section 5.3 in Nagra (2024o).

Transient fluid pressure evolution, again from the same simulation, is shown in Fig. 6-9. Note that pressure shown in the figure corresponds to gas pressure when a gas phase is present, and water pressure when only water is present. In the early post-closure phase (~1,000 years), pressure increases in the Opalinus Clay surrounding the HLW repository section due to differential thermal expansion caused by the heat emitted from the waste (Fig. 6-9, top), reaching a peak of ~13 MPa after 200 years at 30 m above the centre of the HLW repository section.

By contrast, thermally induced pressures within the HLW emplacement drifts do not significantly exceed hydrostatic values. After the thermal pulse has dissipated, pressure in the HLW drifts begins to increase due to gas accumulation from the corrosion of the canisters. Upon canister breaching, assumed to occur at 10,000 years, corrosion and gas generation rates increase, and gas pressure reaches a first peak of ~10.5 MPa after 15,000 years, followed by a peak of ~11.8 MPa after ~115,000 years, after which gas pressure decreases towards hydrostatic since the canister has corroded away and gas generation has ceased.

In the L/ILW emplacement caverns (Fig. 6-9, bottom), pressure starts to increase from about 36 years after emplacement due to the accumulation of waste-generated gas, reaching hydrostatic pressure after ~20,000 years and a peak pressure of ~9.3 MPa after ~70,000 years. Pressure in the operations tunnel and sealing sections of the L/ILW repository section follows a similar path of evolution, due to the hydraulic connection and continuous gas pathways in these repository structures. The same is observed in the pressure evolution in the central area of the repository (not shown in the figure). On the other hand, the fast restoration of hydrostatic pressure in the shaft above the V3 seal is indicative of the hydraulic separation of the shaft from the repository provided by the seal (Nagra 2024o).

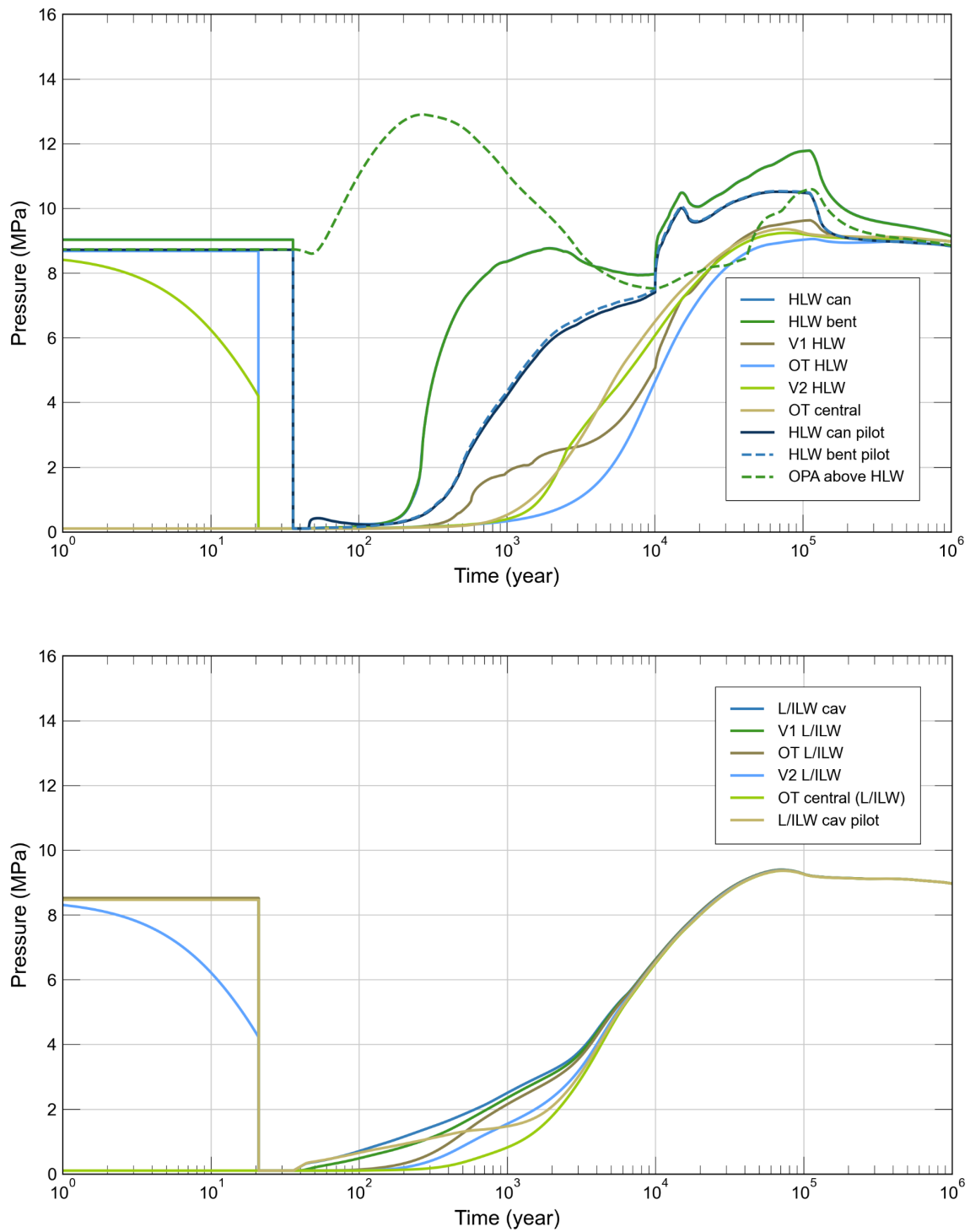


Fig. 6-9: Assessment of total system: transients of pressure in the HLW repository section (top) and in the L/ILW repository section (bottom)

V1: cavern or drift seal, OT: operations tunnel, V2: repository section seal, pilot: pilot repository. For more details see Section 5.3 in Nagra (2024o).

6.2.3 The production and fate of repository-generated gas

The assessment of the production and fate of gases in the repository is presented in detail in NTB 24-23 (Nagra 2024o) and is summarised in Section 4.3 of NTB 24-22 Rev. 1 (Nagra 2024u) and in the following paragraphs.

The assessment of gas pressure build-up consists of a description and quantification of (1) the potential gas sources, which include the waste, barrier components such as disposal canisters and other gas-generating repository components, (2) the processes leading to gas generation, including reactions related to the corrosion of metals and degradation of organics, and (3) gas transport mechanisms shaping the hydraulic evolution of the repository described in the previous section.

The evolution of the gas generation rate and of the cumulative volume of generated gas during the time period for assessment is calculated using a range of alternative assumptions and parameter values in a set of assessment cases that account for uncertainties (*deviation in performance*) (see Chapters 5, 6, 7 and 8 in NAB 24-25, Nagra 2024k for more details). The assessment cases correspond to “PA scenarios” related to the production of gas, the properties of the closure system and backfill, and the barrier function of the host rock (PA scenarios are discussed further in Section 6.4). These cases are additionally supported by probabilistic simulations to account for parametric uncertainty, as discussed in Section 6.3. The outcome of these calculations is used as input to the modelling of gas transport and the resulting build-up of pressure and associated fluid flows in the repository and surrounding rock, which are used as performance indicators.

The main outcome of the gas synthesis described in Chapter 6 in NTB 24-23 (Nagra 2024o) is a consistent hydraulic evolution of the repository adhering to the same principles and transport regimes for all calculations variants. This is the foundation for performance assessment of gas-induced effects, elaborated in NTB 24-22 Rev. 1 (Nagra 2024u), which demonstrates that gas production in the geological repository for both HLW and L/ILW does not compromise the post-closure safety functions of either the host rock or the engineered barriers. In all the cases explored, there is a safety margin with regards to the defined performance targets. Even under the most pessimistic assumptions, the safety functions of the repository system are not degraded.

A significant observation concerns the self-regulating nature of gas pressure in the repository: an increase in the amount of gas in the pores results in a higher effective permeability for gas, which increases gas transport along the repository structures and decreases pressure build-up. The performance assessment shows that partially saturated conditions will sustain gas pathways through the repository structures such that overpressures are mitigated. This is particularly relevant to the L/ILW repository section, where gas migrates axially along the tunnels. It is shown that fully water-saturated conditions, which could result in a loss of porosity and gas permeability due to potential interactions between materials and consequently lead to gas-induced overpressures, can be excluded. By contrast, the HLW near field saturates with water within few hundreds of years due to the high suction and thermal gradient. In all cases the saturated granular bentonite material limits the gas flow away from the corroding canisters. However, due to the low corrosion rate of the carbon steel canister under repository conditions, gas pressure build-up in the HLW drifts is not significant unless extremely conservative assumptions are made.

Another important insight from the results described in Chapter 6 of NTB 24-23 (Nagra 2024o) is that the V3 shaft seal saturates faster than it takes for waste-generated gas to reach the shaft. This leads to gas accumulation beneath the shaft, with gradual dissipation of gas through dissolution and diffusion. Over the course of tens of thousands of years, some of this gas eventually permeates to the Malm aquifer above the repository. By then, however, gaseous radionuclide-bearing species, notably methane incorporating ^{14}C , will have decayed within the repository system, such that there are no significant radiological consequences (see Chapter 8).

The reasons for the fast saturation of the V3 shaft seals are the abundance of water from aquifers above the Opalinus Clay and the high capillary suction of the compacted bentonite material of the sealing elements.

PA modelling makes use of conservative assumptions and simplifications. For example, the corrosion and degradation of materials are conservatively decoupled from the availability of water for these reactions. In reality, gas generation ceases in dry conditions, or at low relative humidity, as water availability is the most important limiting factor for all processes that generate gas (principally the degradation of organics and corrosion of metals). As a further example, it is conservatively assumed that gas generation from the L/ILW repository section starts immediately after waste emplacement, despite the highly alkaline near-field conditions, which, in reality, are expected to decrease corrosion and inhibit microbial activity. The calculated build-up of gas overpressure could be further reduced by including in the calculations the expected thriving gas-consuming microorganisms, which are not currently accounted for, and actual gas pressure build-up could, if it were to prove necessary, be reduced by further tailoring the closure system.

6.2.4 Probabilistic evaluation of performance indicators

The effects of parameter uncertainty on performance indicators are analysed probabilistically to identify any possible combinations of parameters that could give rise to a TH evolution of the repository that could damage or degrade the performance of the barriers and perturb the repository safety functions.

Fig. 6-10 outlines, as an example, a probabilistic workflow with the temperature in the bentonite buffer (T_B) as a performance indicator, calculated from simulations of the temperature distribution in the HLW near field. The buffer temperature is of concern, because, if it is too high, the possibility of thermally induced geochemical alteration cannot be excluded, which could alter key buffer properties, such as its swelling capacity and plasticity. Thermally induced geochemical alteration of the buffer is referred to as a deviation in performance, which, if it cannot be excluded, is propagated to safety scenario development and to the analysis of radiological consequences.

The workflow starts with the definition of uncertainty ranges for thermal and hydraulic parameters as explained in detail in Chapter 3 of NAB 24-25 (Nagra 2024k). Representative parameter values from these ranges are selected as a reference case. In addition, 100 Latin Hypercube samples are taken for the probabilistic realisations. TH simulations of the HLW near field are carried out for a model domain consisting of a 2D vertical cross-section through the repository centre (Fig. 6-10 central figure). Transients of the performance indicator T_B are monitored at a control point in the bentonite buffer, 7.5 cm above the canister. For each simulation, the maximum temperature, $T_{B,max}$, is calculated and is normalised by dividing by the performance target of $T_{max} = 150^\circ \text{C}$, yielding a dimensionless performance index, denoted by NBT. The performance target represents the maximum temperature in dry bentonite that can be maintained without causing irreversible geochemical alterations. A histogram showing both a probability density function (PDF) and cumulative density function (CDF) of NBT aids in calculating whether T_{max} could be exceeded. Additional details on the workflow are given in NTB 24-22 Rev. 1 (Nagra 2024u).

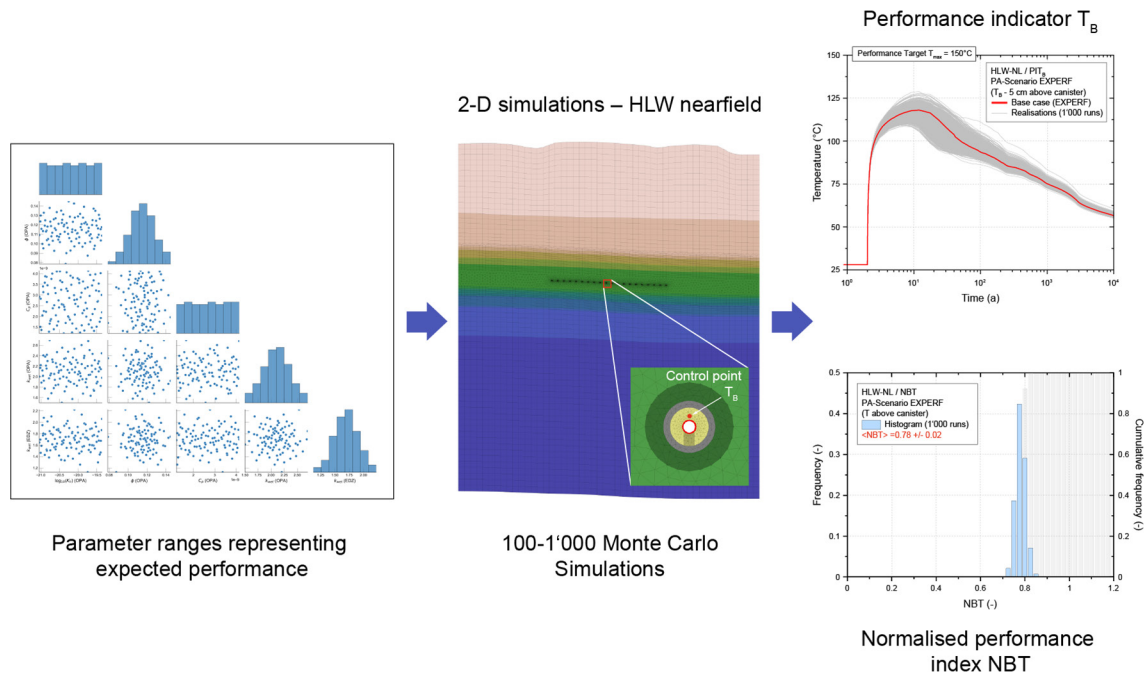


Fig. 6-10: Probabilistic workflow for the simulation of the temperature distribution in the HLW near field, with the bentonite temperature (T_B) as a performance indicator

Scatter-plot matrix of the 100 Latin Hypercube samples from distributions of thermal and hydraulic input parameters (left); Close-up of the 2D model through the centre of an HLW emplacement drift, with the control point representing the performance indicator T_B in the bentonite buffer (centre); Transients of the performance indicator T_B comprising the reference case and 100 probabilistic realisations, representing the uncertainty range of the performance indicator (right above). Finally, histograms with probability density function and cumulative density function of NBT (right below) aid in calculating performance margins and evaluation scales (see Nagra 2024k for further details).

The results show that, in this example, the performance target is met with a wide safety margin. The same is true for other performance indicators considered in the PA (Nagra 2024u), indicating that, over a wide range of conditions spanning all identified parameter uncertainties, the integrity of the barriers will be maintained, and the system is expected to perform its safety functions as designed.

6.3 Assessment of uncertainty and its propagation

The assessments described in the previous sections involve the use of multiple PA models and, in some instances, chains of models. The appropriateness of a model depends on its subjective representation of the real-world system and its intended use. Suitability for the intended purpose can be evaluated through a well-planned Code and Calculation Verification and Model Validation (V&V) programme, considering qualitative and quantitative performance measures.

In the probabilistic workflow described previously, uncertainty in the ranges of input parameters is propagated directly to uncertainty in model output and potentially further (e.g., if parameter uncertainty results in situations where a performance target may not be met) to the development of safety scenarios and to the analysis of radiological consequences. Uncertainty is, however, also associated with model abstraction, or conceptualisation, which inherently introduces inaccuracy

and additional uncertainty by simplifying real-world systems. Assessing such uncertainties within the PA workflow is integral to building confidence in the evaluation of post-closure performance and safety for the deep geological repository.

Chapter 5 of NTB 24-22 Rev. 1 (Nagra 2024u) provides a detailed discussion of model abstraction in the context of PA, including:

- geometric abstractions,
- the simplified representation of phenomena and processes, and
- homogenisation and the scaling of parameters.

Nagra's efforts in code-benchmarking and PA model validation are detailed in Section 5.7 of NTB 24-22 Rev. 1 (Nagra 2024u). In general, it is shown that the impact of abstractions is either small or errors on the side of conservatism; see, however, the discussion of the characterisation of the geological barrier below.

6.4 Screening of performance assessment scenarios

Section 6.2 above presents an analysis related to the thermally induced geochemical alteration of the buffer as an example of a deviation in performance, the potential for which are explored by modelling. Such deviations in performance could potentially arise in certain “PA scenarios”, e.g., upper bound heat production in the HLW canisters. A wide range of PA scenarios is considered, primarily through modelling including the expected performance of the repository, which is also categorised as a PA scenario.

The results from performance assessment (for details see NTB 24-22 Rev. 1, Nagra 2024u) indicate that most PA scenarios do not affect the broad descriptions of the evolution of the pillars of safety in any substantive way, i.e., performance targets are met even for combinations of unfavourable parameters explored using the probabilistic workflow, generally with a large safety margin, as illustrated by the example in Fig. 6-10. This shows that the repository system is highly robust regarding uncertainties related to the repository system performance and that the repository is likely to evolve according to its “expected performance” as described, e.g., in NAB 24-20 Rev. 1 (Nagra 2024m). PA scenarios that result in only minor deviations in performance are propagated, along with the PA scenario for expected performance, to the “reference safety scenario” and its variants, described further in Section 7.2.

There are several PA scenarios that relate to the properties of the CRZ. Compared to the engineered barriers, which are assumed to adhere to their design specifications, certain aspects of the CRZ involve some uncertainty. As noted in Section 6.2.1, PA modelling generally assumes that diffusion is the dominant transport mechanism in the CRZ and that advection along steeply dipping faults does not represent a relevant transport mechanism. Furthermore, it is assumed that sedimentary features, such as hard beds in the upper and lower confining geological units, do not act as lateral release paths. While these assumptions are well supported, some uncertainty remains due to inevitable incompleteness in geological characterisation. Thus, there are PA scenarios that assume the unlikely presence of an undetected, water-conducting fault or the presence of water-conducting sedimentary features within the confining geological units, which cannot be excluded *a-priori*, even in the repository construction phase. If they were to occur, such features could affect (at least locally) the conceptualisation of radionuclide transport in the geosphere as a diffusion-dominated process. These PA scenarios are propagated to alternative safety scenarios, as described in Section 7.3, or, if extreme and hypothetical properties for such features are assumed, to the “what-if” cases described in Section 7.4.

Finally, some PA scenarios, such as the creation of new faults, or the re-activation of existing faults, in the post-closure period by repository-induced effects, i.e., by the effects of repository heat and repository-generated gas, or by geological phenomena such as earthquakes, have been shown in the PA in NTB 24-22 Rev. 1 (Nagra 2024u) and in the assessment of earthquake scenarios in NAB 24-28 (Nagra 2024g) to be either implausible or exceedingly unlikely. It has also been shown that the excavation of the repository by erosion, within a million-year time frame, by erosive processes is hypothetical. These effects are, nonetheless, considered in the “what-if?” cases in safety scenario development, as described in Section 7.4.

Future human actions (FHA) and their potential radiological consequences do not fall within the scope of PA. Rather, these are considered separately, as reported in NAB 24-09 (Nagra 2024r). FHA safety scenarios are described in Section 7.5.

7 Safety scenario development

Aims of the chapter:

- To develop safety scenarios that capture uncertainty in the performance of one or more pillars of safety and the fulfilment of their respective safety functions.
- To connect performance assessment, wherein arguments are developed to support the claims of the safety concept, with the radiological consequence analysis that demonstrates the fulfilment of protection criteria.
- To show, specifically, how safety scenarios connect real world phenomena and processes addressed in performance assessment with the highly abstracted models of radiological consequence analyses.
- To describe the safety scenarios, their variants and the calculation cases that are propagated to radiological consequence analysis.

Safety scenarios describe the different possibilities for the initial state and subsequent evolution and performance of key features of the repository system (the pillars of safety). As such, they form an integral part of the treatment of uncertainty in the safety assessment (see Section 4.4). Safety scenarios provide a link between the THM process understanding achieved by modelling of the repository carried out by PA (Chapter 6) and the radionuclide-specific transport modelling carried out in the analysis of radiological consequences (Chapter 8); see Tab. 7-1. The development of safety scenarios is detailed in NTB 24-21 (Nagra 2024e). Safety scenario development “*leads to a systematic focusing of the safety assessment on the important conditions and phenomena relating to performance of the disposal facility*” (IAEA 2011b). Each safety scenario consists of a general description of the evolution of the repository system and its environment, starting from its initial state at the time of repository closure, and extending into the future, at least up to the end of the time period for assessment. A safety scenario can have one or more variants. Variants of a given safety scenario share largely the same general description of features, events and processes occurring over time, but may differ, for example, in the extent or timing of key phenomena associated with that safety scenario as well as positioning or properties of a key feature. In addition, each safety scenario or variant will typically have associated with it one or more “calculation cases”. The calculation cases define specific calculations that are carried out to analyse the radiological consequences of a particular safety scenario or variant, including key model abstractions. Care is taken in the selection of the set of safety scenarios to capture uncertainty in the broad evolution of the pillars of safety by bounding and representing the possible ways in which the repository might evolve. The definition of variants captures conceptual uncertainty, while parameter uncertainty is treated within the calculation cases (see Section 4.4 and Chapter 8).

7.1 Safety scenario development workflow

Safety scenario development, as part of the general safety assessment workflow, is illustrated in Fig. 7-1.

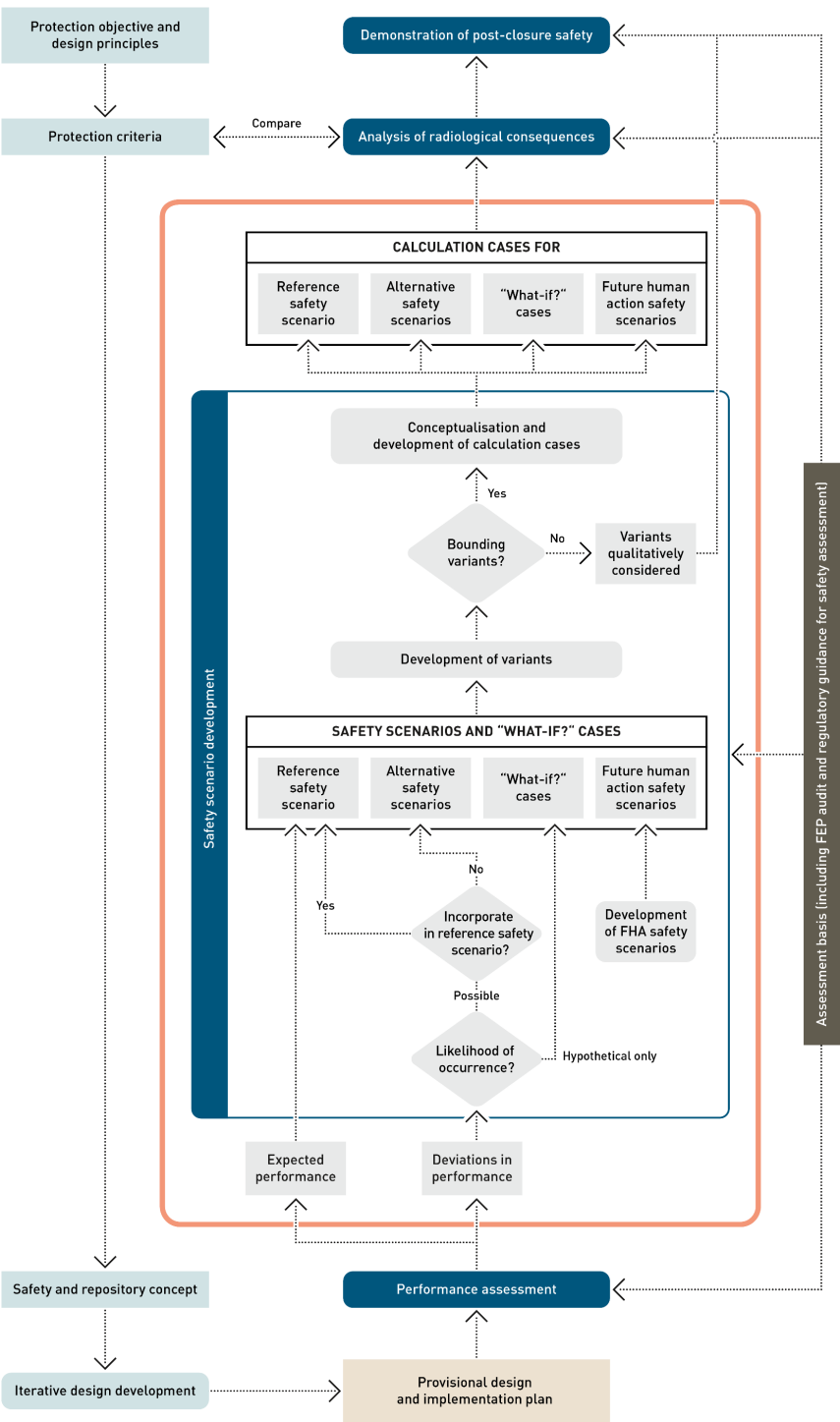


Fig. 7-1: Workflow for the post-closure safety case, highlighting, in the orange box, the main elements of the safety scenario development process

Consistent with international and national guidelines, safety scenarios are defined that describe both the expected evolution of the repository and the barriers over the one-million-year time period for assessment, and also less likely and hypothetical evolutions and deviations in performance as identified in the assessment basis and the PA as well as others resulting from future human actions. Specifically, safety scenario development involves the definition of:

- A reference safety scenario

The reference safety scenario is a general description of the expected initial state of the repository barriers, their expected evolution over time, including the release, retention and transport of radionuclides within the repository system, and the assumptions made regarding the biosphere for the analysis of radiological consequences.

- A set of alternative safety scenarios

The alternative safety scenarios include deviations in performance that are judged unlikely, or highly unlikely, but scientifically non-excludable evolutions, on the basis of the assessment and the findings of PA, and lead to a general description of the initial state of the repository barriers and their evolution over time that differs substantially from that of the reference scenario.

- A set of hypothetical “what-if?” cases

Deviations in performance that are judged to be impossible, at least in the one-million-year period for assessment, i.e., which are outside the range of possibilities supported by currently available evidence and knowledge, may nonetheless be propagated to the analysis of radiological consequences as hypothetical “what-if?” cases, designed to demonstrate the robustness of the repository system. As well as “what-if?” cases based on extreme assumptions regarding actual physical phenomena, additional hypothetical “what-if?” cases are also defined that postulate a degraded performance of each of the pillars of safety, unconnected to any physical phenomena. As noted in Section 4.5, they can mitigate possible “completeness uncertainty”, pre-empting possible criticism that some detrimental FEPs are either unknown or have been overlooked.

- A set of safety scenarios considering future human actions (FHAs).

The safety scenarios considering future human action concern the long-term impact of potential actions undertaken by human society on the performance and safety of the repository. They are handled in a broadly similar manner to the alternative safety scenarios but are treated as a separate class of safety scenarios, due to their particularly speculative nature. The main difference in their development compared with the alternative safety scenarios is that, instead of being derived from the assessment basis and deviations in performance identified in the PA, they are derived from FHA FEPs within the FEP database, which are screened to identify those with the potential to perturb the pillars of safety and their safety functions. A full description of Nagra’s methodology for FHAs in the general licence application, its application, and the assessment of the identified FHA safety scenarios can be found in Chapter 2 of NAB 24-09 (Nagra 2024r).

Having defined the safety scenarios, the main elements of uncertainty are identified that, depending on how they are conceptualised within a given safety scenario, could affect the radiological consequences of the scenario. These can include the extent or timing of key phenomena associated with a safety scenario as well as positioning or properties of a key feature. In general, the conceptualisations that are expected to give rise to the highest radiological consequences are taken as the scenario variants that are propagated to the analysis of radiological consequences.

Calculation cases are then defined for each safety scenario variant. This generally requires a degree of geometrical simplification, or abstraction. There may be different ways in which this abstraction can be performed, in which case, especially if the most conservative abstraction cannot be judged in advance of the calculations, more than one calculation case for a given safety scenario variant may be propagated to the analysis of radiological consequences. Additional calculation cases, including probabilistic analyses, are also defined to quantify the effects of parameter uncertainty; see Chapter 8 for details. Several individual calculations may be performed for each calculation case that is propagated to radiological consequence analysis. In particular, different model concepts and models are used for radionuclide transport pathways in the liquid phase on the one hand and two-phase flow and transport of ^{14}C on the other (see Chapter 8 in NAB 24-07 Rev. 1, Nagra 2024w and additional details given in Appendix B of NTB 24-18, Nagra 2024p).

Deviations in performance arising from almost all PA scenarios are incorporated within, or propagated to, one or more safety scenarios and “what-if?” cases, as illustrated in Tab. 7-2. The difference between safety scenarios and “what-if?” cases on the one hand and PA scenarios on the other is that safety scenarios and “what-if?” cases focus on the properties of the system and their evolution as they affect waste isolation and radionuclide release, retention, transport and radiological consequences. Thus, there is, for example, a “what-if?” case in which a fault is created at some time in the future, and the description given in the present chapter focusses on what this means in terms of radionuclide transport pathways, rather than what could lead to faulting in the first place. The PA scenarios, on the other hand, address events and processes that could potentially degrade the barriers and their safety functions leading to deviations in performance, including phenomena like earthquake, gas pressurisation and thermal stresses that could lead to faulting. Thus, several PA scenarios can be captured by just one (alternative) safety scenario or “what-if?” case, as can be seen in the table. There are also multiple PA scenarios that are captured by the reference safety scenario, through the ranges of parameter values considered in the analysis of this scenario, since the PA scenarios that are captured affect, e.g., the rates of certain processes, but do not fundamentally affect the broad evolution of the pillars of safety. Note that the hypothetical ZEROGAS PA scenario is not directly included in the reference safety scenario parameter ranges. Its impact is, however, considered implicitly through the conservative assumption of fully saturated conditions employed for the modelling of aqueous transport of radionuclides in the analysis of radiological consequences (see Chapter 8 and Appendix A in NTB 24-18, Nagra 2024p).

Further combinations of PA scenarios not captured by the present safety scenarios, or the combination of a PA scenario and an FHA FEP, could, in principle, be conceivable. However, the likelihood of such combinations, though not evaluated quantitatively, is so low that they could, in effect, be considered as “what-if?” cases. Furthermore, the consequences of PA scenarios and FHA FEPs considered individually can, in some instances, be greater if they are considered in combination. For example, the presence of an undetected fault, which is considered as an alternative safety scenario, will lead to reduced gas pressures in the repository, which would lessen the impact of gas-mediated transport of ^{14}C in the event of a borehole later being drilled and penetrating the repository underground structures. In practice, the consequences of such combinations are, in any case, bounded by the extreme “what-if?” cases mentioned above.

The comprehensiveness of the safety scenarios described in this chapter is assessed as part of the wider assessment of the comprehensiveness of the post-closure safety case carried out by means of the FEP audit (see Section 5.5).

Tab. 7-1: Incorporation of PA scenarios in safety scenarios and “what-if?” cases

Colouring indicates whether a PA scenario is incorporated in the reference safety scenario, an alternative safety scenario, or a “what-if?” case.

Performance assessment scenarios		Safety scenarios		“What-if?” cases			
		Reference safety scenario	Alternative safety scenarios		Related to physical phenomena		
			Hydraulically active confining geological unit	Undetected fault	Undetected fault with hypothetically high transmissivity	Future creation or reactivation of a fault	Repository excavation
EXPERF	Expected performance	RS					
MKALK	Exfiltration to Muschelkalk	RS-KEUPER INACTIVE					
UPFLOW	Upflow with artesian pressure in Keuper						
DWNFLW	Downflow with artesian pressure in Malm						
HETOPA	Heterogeneous Opalinus Clay						
PESSGAS	Pessimistic gas generation rate (L/ILW)						
UBGAS	Upper bound gas source term (HLW)						
V3PERM	Highly permeable V3 seals						WI-V3
HARDBED	Hard beds as release paths		AS-HA				
HERWIS	«Herrenwis Unit» as release path		AS-HA/CD				
FLTHLWP	Undetected fault parallel to HLW drifts			AS-FLT-HLW-Along	WI-FLT-HLW-Along		
FLTHLWN	Undetected fault normal to HLW drifts			AS-FLT-HLW-Across	WI-FLT-HLW-Across		
FLTLILW	Undetected fault affecting L/ILW caverns			AS-FLT-L/ILW	WI-FLT-L/ILW		
UBHEAT	See RIEHLW					WI-FLT/RA-HLW	

Tab. 7-1: Cont.

Performance assessment scenarios		Safety scenarios			“What-if?” cases			
		Reference safety scenario	Alternative safety scenarios		Related to physical phenomena			Hypothetical degraded performance of pillar of safety
			Hydraulically active confining geological unit	Undetected fault	Undetected fault with hypothetically high transmissivity	Future creation or reactivation of a fault	Repository excavation	
RIEHLW	Thermally- and gas-induced faulting (HLW)					WI-FLT/RA-HLW		
RIELILW	Gas-induced faulting (L/ILW)					WI-FLT/RA-L/ILW		
SEISHLW	Earthquake affecting HLW drifts					WI-FLT/RA-HLW		
SEISLILW	Earthquake affecting L/ILW caverns					WI-FLT/RA-L/ILW		
LOSSFLT	Fault activation due to overburden loss					WI-FLT/RA		
LOSSOB	Permeability increase due to overburden loss						not explicitly represented	
IRMAX	Instant release of entire I-129 activity							WI-SF-IRF
WPFail	Canister failure at time of repository closure							WI-HLW-CAN
ZEROGAS	No gas generation	not addressed						

7.2 Reference safety scenario and its variants

7.2.1 General description

In the reference safety scenario, the engineered barriers are assumed to be implemented according to the provisional repository design, as described in Section 3.3. The geological barrier (i.e., the CRZ) has the characteristics inferred from site characterisation and understanding, as described in Section 5.2. The engineered and geological barriers are further assumed to evolve and perform according to the phenomenological description of their expected evolution given in NAB 24-20 Rev. 1 (Nagra 2024m) and summarised in the present report in Section 5.3. This phenomenological description is supported by the results of the performance assessment as described in NTB 24-22 Rev. 1 (Nagra 2024u) and is also in accordance with the current safety and repository concept presented in Chapter 3 of NAB 24-18 Rev. 1 (Nagra 2024s) and summarised in the present report in Section 3.2. Human intrusion and other future human actions, either deliberate or inadvertent, that could impair the functionality of the repository are assumed not to occur

during the post-closure period in the reference safety scenario. This is justified by the placement of the repository in a deep underground location where human intrusion is unlikely and which is isolated from human actions occurring at or near the surface. Criticality safety in the post-closure phase is assumed, since it will be a design requirement of the repository, specifically, for the HLW final disposal canister and for the HLW canister loading plans. Criticality safety will be ensured through the implementation of technical and administrative measures (NAB 24-03, Nagra 2024j).

Based on the performance assessment, repository-induced effects associated with heat, gas and dissolved chemical species are expected to have minimal effects on the geological environment. Repository-generated heat could, however, lead to a narrow zone of altered bentonite around the HLW canisters, which is considered in the modelling carried out in the analysis of radiological consequences (see Appendix B, Figure B-1 of NTB 24-18, Nagra 2024p). Stress changes during the construction of the underground openings will result in the formation of Excavation Damaged Zones (EDZs) which will largely re-seal over time. However, their constant presence is conservatively accounted for in the modelling carried out in the analysis of radiological consequences. Repository-generated gas will permeate much of the closure system but will be largely retained underground by the V3 seals and slowly, over a period of tens of thousands of years or more, dissolve and diffuse into the host rock.

There will also be some changes associated with long-term climatic and geological processes. The repository is planned to be constructed deep underground, in a location where it is not susceptible to geological events and processes of relevance to post-closure safety. The future formation of permafrost and glaciers above the repository would affect hydraulic gradients, but the resulting variability in the gradients falls within the range of uncertainty covered by parameter variations in the analysis of the radiological consequences of this scenario. In addition, long-term glacial and fluvial erosion processes acting at the surface will gradually reduce the thickness of the overburden above the repository. Studies of the future erosion processes indicate that, considering the 5 – 95% probability range, between ~350 and 670 m of overburden are expected to remain above the repository (see, e.g., Section 6.4.4 of the Geosynthesis of Northern Switzerland, NTB 24-17, Nagra 2024i). It is also shown in Section 5.7.4 of NTB 24-17 (Nagra 2024i) that the minimum overburden of 200 m is sufficient for the Opalinus clay to maintain its self-sealing capacity. There may be some impact of the loss of overburden on the hydraulic conductivity of the Opalinus clay, but this again falls within the range of uncertainty covered by parameter variations in the analysis of the radiological consequences of this safety scenario. The impact of extreme erosion which leads to an excavation of the repository is discussed in the separate report NAB 24-08 Rev. 1 (Nagra 2024q).

The release of radionuclides from the repository during its expected evolution in the reference safety scenario and their retention and transport in the repository barriers are described phenomenologically in Section 5.4. Radionuclides will be retained within the HLW canisters until the canisters eventually become breached (see Section 6.1.4), and most will be released only slowly thereafter, as the waste packages degrade. Radionuclides will be released more rapidly from the waste packages and containers for L/ILW.

Radionuclides that are dissolved in water migrate away from the waste packages predominantly by diffusion and, upon reaching one of the deep aquifers, by advection. Given the geometry of the CRZ with large horizontal distances from the repository to potential exfiltration pathways, this transport can be approximated by 1D vertical transport (see Section 8.1 for more details on the implementation). Radionuclides that form volatile species, principally ^{14}C as methane, mix with, and are transported by, repository-generated gases through the engineered barriers and could also enter the CRZ. In contrast to the aqueous phase transport, the 3D characteristics of gas transport must be considered (see Section 8.1 for more details on the implementation). These radionuclides also eventually dissolve in porewater within the repository barrier system or within the deep aquifers. Radionuclides that do not decay in transit could eventually reach the biosphere.

Biosphere dose conversion factors (BDCFs) are calculated to convert the radionuclide release rate to dose rates by assuming different enveloping biospheres with respect to climate and geomorphology. Different biospheres are considered via calculation cases as described below in Section 7.2.3. Since it is not possible to make a statement about when which biosphere will occur in the far future, all of the different biospheres are evaluated for the whole time period for assessment (i.e., assuming constant climatic conditions) within the reference safety scenario.

7.2.2 Reference and variant conceptualisations of the CRZ

The starting point for the conceptualisation of the CRZ is the geological knowledge as described in Section 5.2.1, which is abstracted as illustrated below in Fig. 7-2 and Tab. 7-2. The CRZ, in the reference conceptualisation, is taken to extend between the upper deep aquifer above the repository (the Malm aquifer) and down below the repository to the Keuper aquifer. The geosphere modelling domain in the analysis of radiological consequences is also bounded, at the top, by the Malm aquifer, but extends below the Keuper aquifer to the deeper Muschelkalk aquifer.

Within the confining geological units of the Opalinus Clay, there are a number of thin competent (i.e., hard and brittle) horizontal rock units, including the hard beds, embedded between layers of more plastic clay. In the reference safety scenario, in line with current understanding of the site, these rock units are taken to be sparsely fractured and not to form horizontally connected pathways for radionuclide transport (the presence of a hydraulically active confining unit is considered as an alternative safety scenario; see Section 7.3.1).

Local conductive layers within the Keuper aquifer below the repository have been identified towards the area where the L/ILW repository section is planned; however, such features have not been observed in the area where the HLW repository section is planned. Furthermore, groundwater within the Keuper aquifer is characterised by very long residence times and it could be possible that the aquifer is hydrogeologically inactive (Section 4.5.5.4 in NTB 24-17, Nagra 2024i), at least below the HLW repository section. Thus, two conceptualisations are considered (see Fig. 7-3):

- **Reference conceptualisation:** it is conservatively assumed that the Keuper aquifer is hydraulically active and forms a significant advective transport path below the repository.
- **Variant:** the Keuper aquifer below the HLW repository section is inactive and radionuclides are only transported vertically across it by diffusion; the CRZ extends vertically downwards to the Muschelkalk aquifer.

Radionuclides reaching the hydraulically active Keuper aquifer can either be transported vertically across it, predominantly by diffusion, or be diverted into the aquifer and conveyed by groundwater flow to a regional aquifer (considered part of the biosphere), where they become substantially diluted. Radionuclides that reach the other deep aquifers, at the boundaries of the geosphere model, are also conveyed to, and diluted in, the biosphere. Any ^{14}C reaching the deep aquifers in the form of gaseous methane is assumed to dissolve. The significant transport time for dissolved radionuclides within these aquifers as well as between these beds and the biosphere is conservatively neglected. i.e., radionuclides that are diverted into deep aquifers are assumed to be directly and instantaneously transferred to the regional aquifer in the biosphere.

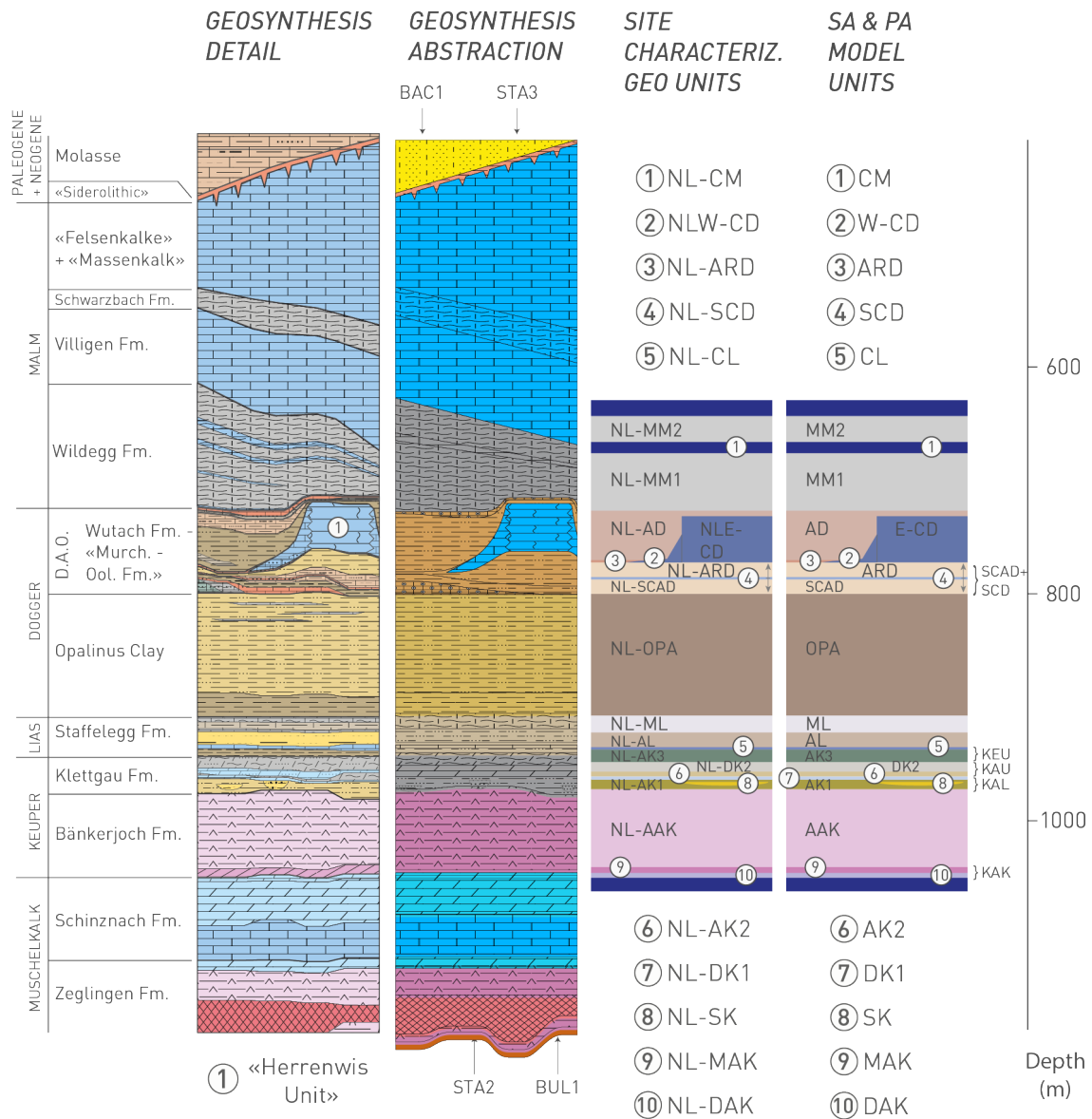


Fig. 7-2: Abstraction of the sedimentary Mesozoic sequence at the site into units of comparable transport parameters

The abstraction results in the generation of the local “layer-cake” model units for performance assessment and for the analysis of radiological consequences (“SA & PA model units”). D.A.O.: Dogger Group above Opalinus Clay.

Tab. 7-2: Stratigraphic units, geological abstracted units and abstracted model units for performance assessment and for the analysis of radiological consequences

Stratigraphic unit	Abstracted unit and full name		Abstracted model unit
Wildegge Fm.	MM2	Marly Malm	MM2
	CM	Calcareous Malm	CM
	MM1	Marly Malm	MM1
Dogger Group above Opalinus Clay	AD	Argillaceous Dogger	AD
	E-CD / W-CD	Calcareous Dogger («Herrenwis Unit»)	E-CD / W-CD
	SCD	Sandy-Calcareous Dogger	SCAD+SCD
	SCAD	Sandy-Calcareous-Argillaceous Dogger	
Opalinus Clay	OPA	Opalinus Clay	OPA
Staffelegg Fm.	ML	Marly Lias	ML
	AL	Argillaceous Lias	AL
	CL	Calcareous Lias	KEU
Klettgau Fm.	AK3	Argillaceous Keuper	
	DK2	Dolomitic Keuper	KAU
	AK2	Argillaceous Keuper	
	DK1	Dolomitic Keuper	KAL
	AK1/SK	Argillaceous Keuper / Sandy Keuper	
Bänkerjoch Fm.	AAK	Anhydritic-Argillaceous Keuper	AAK
	MAK	Massive Anhydritic Keuper	KAK
	DAK	Dolomitic-Anhydritic Keuper	

7.2.3 Definition of calculation cases

The calculation cases for the reference and variant conceptualisations of the reference safety scenario are shown schematically in Fig. 7-3. The calculation cases are propagated to radiological consequence analysis and are furthermore summarised in tables in Chapter 8, where the calculation results are presented and discussed.

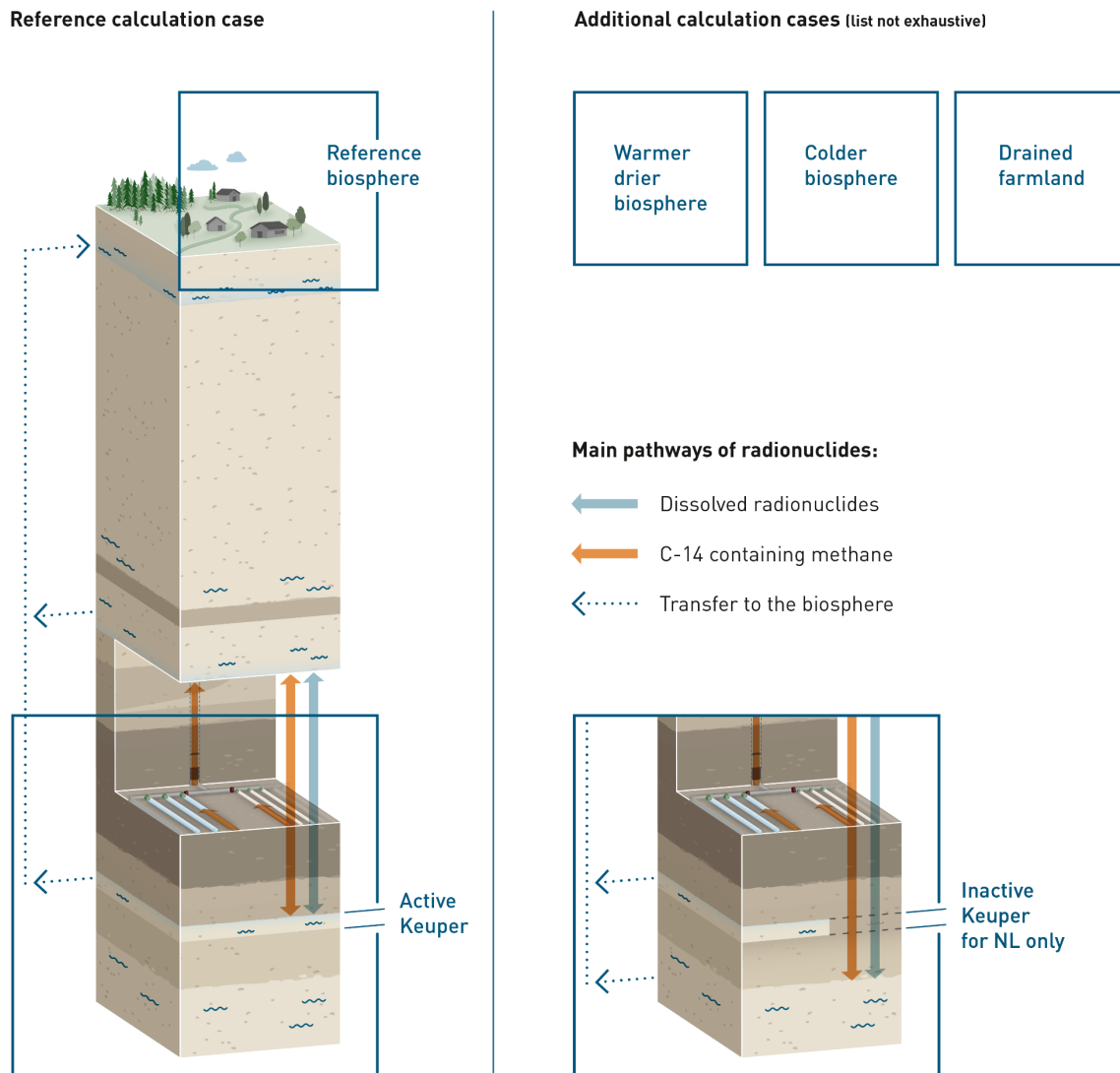


Fig. 7-3: Main pathways for radionuclide transport in the reference safety scenarios, showing, in the bottom left box, the reference conceptualisation with the hydraulically active Keuper aquifer and, the bottom right box, the variant with the inactive Keuper aquifer below the HLW repository section

The figure also illustrates the different biospheres which are handled as different calculation cases within the reference safety scenario.

Arrows in this and subsequent similar figures indicate, respectively:

Blue, bold: radionuclide migration in the aqueous phase, in which radionuclides are released to porewater that contacts the waste packages and are subsequently transported by aqueous diffusion and, where there is porewater flow, by advection,

Red, bold: radionuclide migration of ^{14}C in the aqueous and gas phases (two-phase flow), in which gaseous radionuclides mix with, and are transported by, repository-generated gas,

Dotted blue arrows: Instantaneous transfer of radionuclides into the regional aquifer within the biosphere, conservatively neglecting the significant transport time with associated radioactive decay.

The modelling approaches used to simulate these migration pathways are summarised in Section 8.1 and described in Appendix B of NTB 24-18 (Nagra 2024p), and in Chapter 4 of NAB 24-07 Rev. 1 (Nagra 2024w).

The reference case (RS-RC)¹⁵ is a single, deterministic calculation case, with the conservative assumption that the Keuper aquifer is hydraulically active. A reference biosphere based on present-day climatic conditions is assumed. Where there is quantified uncertainty in the properties of repository features, rates of processes and other model parameters, realistic values are assumed (other uncertainties are generally handled using conservative assumptions). Three additional deterministic calculation cases are defined that assume different, though still steady-state, climatic and geomorphological conditions:

- a warmer and drier climate (RS-BIOS/WARM_DRY),
- a colder climate (RS-BIOS/COLD), and
- a drained wetland surface environment (RS-BIOS/DRAINED_FARMLAND).

Further details of the reference and alternative biospheres, and of biosphere modelling in general, are provided in NAB 24-06 (Nagra 2024n).

Additional calculation cases are also defined that aim to identify, illustrate and quantify the impact of uncertainty in the parameters that control radionuclide retention and transport in the repository barriers:

- a probabilistic calculation case (RS-PROB) with multiple realisations in which all uncertain parameters are randomly sampled from probability distributions, derived from an assessment of parameter uncertainty, to illustrate the range of uncertainty in calculated consequences and to provide the data for a global probabilistic sensitivity analysis to identify those parameters to which the calculated consequences are most sensitive,
- a deterministic calculation case (RS-CMB) in which all parameters to which the radiological consequences are most sensitive, as identified in RS-PROB, are set to their most unfavourable value within their probability distributions.

Finally, a single deterministic calculation case (RS-KEUPER_INACTIVE) is defined for the variant in which the Keuper aquifer is hydraulically inactive below the HLW repository section.

7.3 Alternative safety scenarios

The alternative safety scenarios relate to uncertainty in the features of the CRZ, namely the possibility that there is either:

- a hydraulically active confining geological unit, or
- an undetected fault intersecting the repository.

¹⁵ To facilitate the presentation and discussion of the calculation results in Chapter 8, short keys are associated with the calculation cases, e.g., for the reference case RS-RC (Reference Safety Scenario – Reference Conceptualisation).

It is expected that further site investigations for future licensing milestones will reduce the uncertainty concerning these features, and it may then be possible to exclude these alternative scenarios and their variants in future iterations of the post-closure safety case. It is also likely that hydraulically active faults that are currently undetected would be identified during repository construction, and design measures would then be implemented, so as to avoid waste emplacement in their vicinity. However, this countermeasure is disregarded in the current definition of an alternative safety scenario.

The creation of new faults, or the re-activation of existing faults, in the post-closure period by repository-induced effects, i.e., by the effects of repository heat and repository-generated gas, or by geological phenomena, has been shown in the performance assessment in Section 6.3 of NTB 24-22 Rev. 1 (Nagra 2024u) and in the assessment of earthquake scenarios in NAB 24-28 (Nagra 2024g) to be either implausible or exceedingly unlikely. It has also been shown that the excavation of the repository by erosive processes is hypothetical within a one-million-year time frame. These effects are, nonetheless, considered in the “what-if?” cases in Section 7.4.

7.3.1 Hydraulically active confining geological units

As for the reference safety scenario, we consider the abstraction of the stratigraphic units as illustrated in Sections 5.2.1 and 7.2.2. In line with the geological evidence, the reference safety scenario considers no hydraulically active units within the CRZ. In contrast, in this alternative safety scenario, it is assumed that one of the confining geological units above (variant one) or below the Opalinus Clay (variant two) is hydraulically active. This might be due, e.g., to fractures within the thin, competent “hard beds” (see Section 5.2) having a higher degree of connectivity than currently understood. There are three potentially hydraulically active units within the upper confining geological units, namely Calcareous Malm (CM), Calcareous Dogger (CD), and Sandy-Calcareous Dogger (SCD), and two in the lower confining geological units, denoted by KEU, comprising Argillaceous Keuper and Calcareous Lias, and KAU, comprising Dolomitic and Argillaceous Keuper. In the calculation cases for this alternative safety scenario, each of the above-named units is considered, in turn, to be hydraulically active, while the remainder are inactive, leading to five calculation cases in all (AS-HA/xxx, with xxx being CM, CD, SCD, KEU and KAU).

The assumed hydraulically active unit is conceptualised so as to provide a horizontally connected advective pathway to the regional aquifer. In each of the calculation cases, radionuclides reaching the hydraulically active confining geological units can either be transported vertically across them, predominantly by diffusion, or be diverted into the units, in which case they are assumed to be conveyed instantaneously to the biosphere (see Fig. 7-4). Any ^{14}C reaching a hydraulically active geological unit in the form of gaseous methane is assumed to dissolve. The fractional distribution of radionuclides continuing vertically or being horizontally diverted is calculated based on flow rates and hydraulic gradients (see, e.g., Section B.1.1 in NTB 24-18, Nagra 2024p for details).

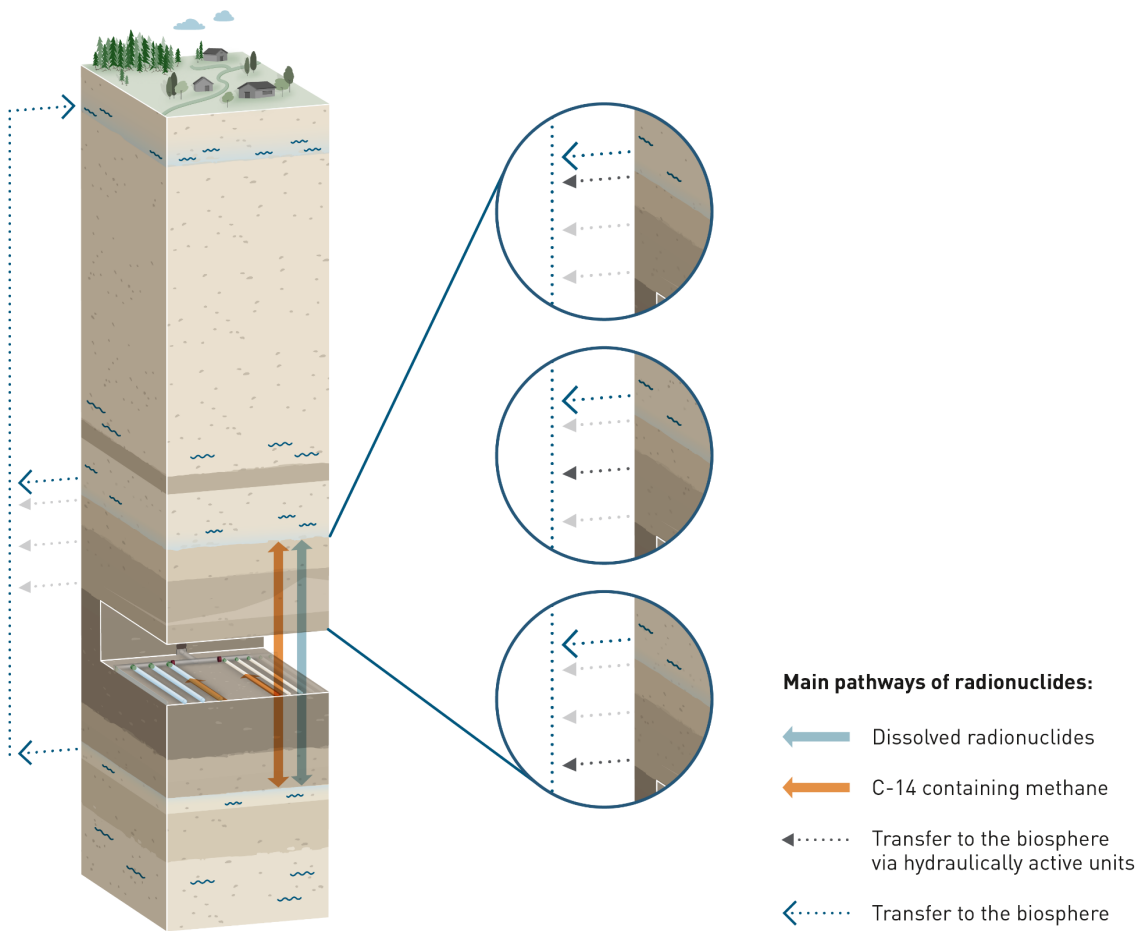


Fig. 7-4: Main pathways for radionuclide transport where it is assumed that one of the confining geological units above or below the Opalinus Clay is hydraulically active and connected to the regional aquifer and thus provides an additional pathway for the advective transport of radionuclides to the biosphere

7.3.2 Undetected fault

This alternative safety scenario assumes the existence of a fault in the vicinity of the repository at the time of its construction that was previously undetected in seismic surveys¹⁶. The fault is conservatively assumed to have a transmissivity of 10^{-10} m²/s. This is one order of magnitude above the highest observed fault transmissivity measured at relevant depths in Northern Switzerland, which occurs in a prominent thrust zone observed in the Schafisheim borehole, well away from the proposed site. Refer to Sections 5.6 and 5.7 of the Geosynthesis of Northern Switzerland in NTB 24-17 (Nagra 2024i), for detailed discussion.

The fault is conceptualised as a vertical planar feature with a transmissivity that is constant in space and time. Flow of water through the fault is assumed to occur vertically upwards from the deep aquifers underlying the repository (Keuper and Muschelkalk) to the deep aquifer above the repository (Malm). Conceptualising the fault as vertical planar feature with constant properties and connecting the upper and lower deep aquifers is a simplification of reality that is highly conservative.

The vertical Darcy flow in the fault is determined using PA models as described in NAB 24-25 (Nagra 2024k), with a hydraulic gradient of less than about 1 m/m. According to the hydrogeological knowledge described in Section 4.4 of the Geosynthesis of Northern Switzerland in NTB 24-17 (Nagra 2024i), this assumption is conservative.

Two variants are considered, where the fault intersects the emplaced waste in either the HLW or the L/ILW repository sections. For each variant, two calculation cases are defined. When the fault strike is parallel to the emplacement rooms (cases AS-FLT-HLW-Along and AS-FLT-L/ILW-Along), the fault is assumed to pass through one, single emplacement room. When the fault strike is perpendicular to the emplacement rooms (cases AS-FLT-HLW-Across and AS-FLT-L/ILW-Across), the fault is assumed to pass through all emplacement rooms in the respective disposal area (but not the emplacement rooms of the pilot repository). These different fault configurations are illustrated in Fig. 7-5.

¹⁶ The fact that such a fault would most likely be detected during construction and that technical measures would be implemented accordingly is disregarded in this scenario.

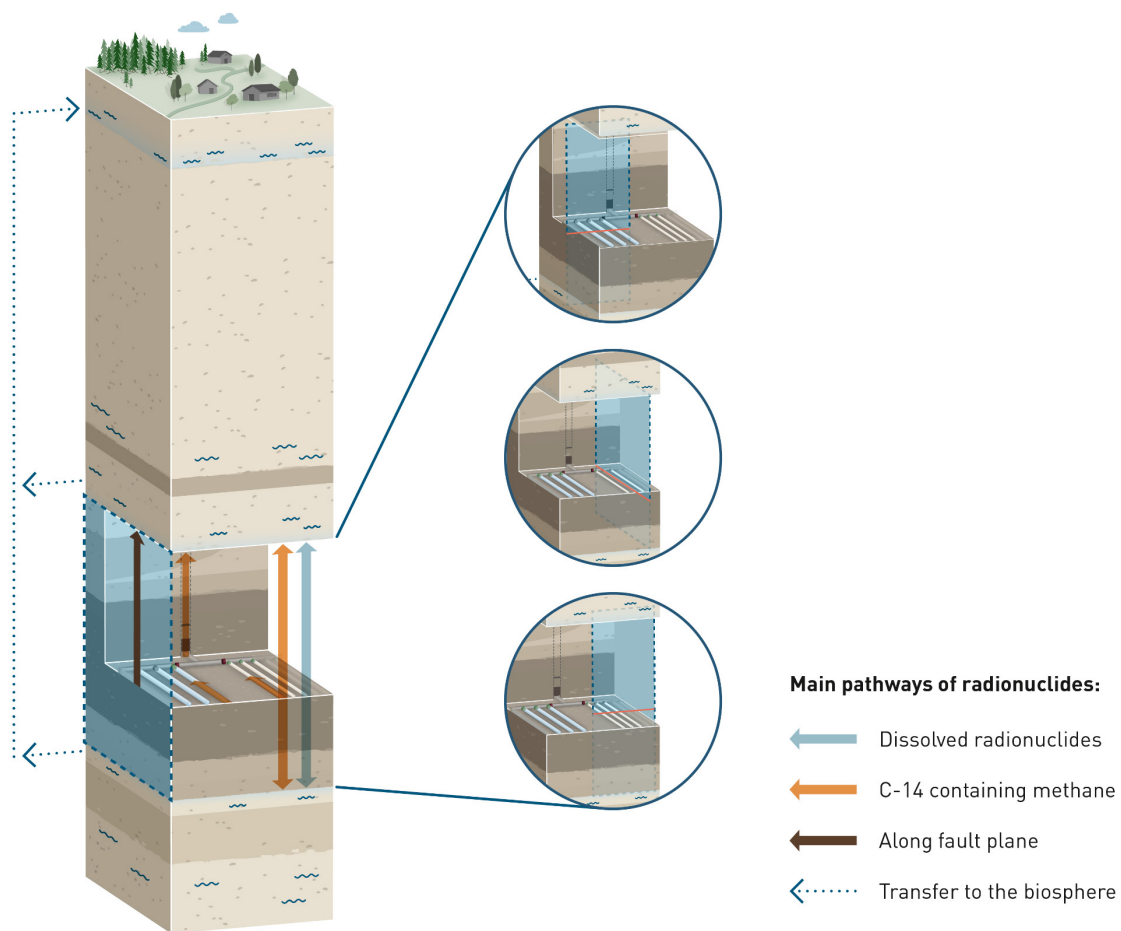


Fig. 7-5: Pathways for radionuclide transport where the existence of an undetected fault is assumed, which intersects one or more L/ILW emplacement caverns or HLW emplacement drifts

The small inset figures show different ways in which the fault is conceived to impact these repository features in the modelling carried out for the calculation of radiological consequences.

7.4 “What-if?” cases

Deviations in performance that are judged to be hypothetical, at least in the one-million-year time period for assessment, may nonetheless be propagated to the analysis of radiological consequences as hypothetical “what-if?” cases, designed to demonstrate the robustness of the repository system. On the one hand, “what-if?” cases can be related to real physical phenomena; such cases are described in Section 7.4.1. On the other hand, Section 7.4.2 presents additional “what-if?” cases that postulate a degraded performance of each of the pillars of safety, unrelated to any physical phenomena.

7.4.1 “What-if?” cases related to physical phenomena

Three groups of “what-if?” cases related to physical phenomena are described below and considered in the analysis of radiological consequences.

1. Occurrence of an undetected fault with a hypothetically high transmissivity in the vicinity of the repository

This set of “what-if?” cases is almost identical to the alternative safety scenario of an undetected fault but assumes the presence of an undetected fault in the vicinity of the repository with a transmissivity that is much higher than could reasonably be expected: $10^{-7} \text{ m}^2/\text{s}$, which is three orders of magnitude higher than that of the fault considered in the alternative safety scenario described in Section 7.3.2. This is similar to fault transmissivities observed in the thin competent, horizontal limestone features, e.g., the “hard beds”, embedded between more plastic clay-rock units within the CRZ; see Section 5.6 of the Geosynthesis of Northern Switzerland report NTB 24-17 (Nagra 2024i). It is four orders of magnitude above the highest observed fault transmissivity measured at relevant depths in Northern Switzerland, which occurs in a prominent thrust zone observed in the Schafisheim borehole, well away from the proposed site (Section 5.6 of NTB 24-17, Nagra 2024i).

As in the corresponding alternative safety scenario, the fault is represented, for reasons given earlier (Section 7.3.2), as a vertical planar feature with a transmissivity that is constant in space and time. Flow of water through the fault is assumed to occur vertically upwards from a deep aquifer underlying the repository (the Keuper or Muschelkalk) to the Malm aquifer above the repository.

As in the corresponding alternative safety scenario, two variants (fault intersecting either the HLW or the L/ILW disposal area) with two calculation cases each are considered: when the fault strike is parallel to the emplacement rooms (cases WI-FLT-HLW-Along and WI-FLT-L/ILW-Along), the fault is assumed to pass through one, single emplacement room. When the fault strike is perpendicular to the emplacement rooms (cases WI-FLT-HLW-Across and WI-FLT-L/ILW-Across), the fault is assumed to pass through all emplacement rooms in the respective repository section (but not the emplacement rooms of the pilot repositories). These different fault configurations are the same as those shown in Fig. 7-5.

2. Future creation or reactivation of a fault either by geological processes or as a repository-induced effect after repository closure

A second set of “what-if?” cases is defined that assumes the generation of a new fault or the reactivation of an existing, non-transmissive fault, during the post-closure period. Faulting or fault reactivation are assumed to occur as the result of stress changes induced either by geological processes, such as seismic activity, or by heat or gas generated within the repository. The transmissivity of the fault is again assumed to be $10^{-7} \text{ m}^2/\text{s}$.

As above, the fault is again represented as a vertical planar feature with a transmissivity that is constant in space and time. Flow of water through the fault is assumed to occur vertically upwards from a deep aquifer underlying the repository (the Keuper or Muschelkalk) to the Malm aquifer above the repository.

Again, two variants are considered, where the fault intersects either the HLW or the L/ILW disposal area. The fault is anticipated to be created or re-activated at 10,000 years post-closure in cases where the fault intersects the HLW repository section. The timing corresponds with the assumed canister breaching time, leading to conservative estimations of the consequences. Faulting or fault re-activation could also result in a degraded performance of the bentonite buffer which is considered in these calculation cases. In cases where the fault intersects the L/ILW repository section, 20,000 years is selected, as it is the earliest time when, according to perfor-

mance assessment, a gas overpressure could occur in the L/ILW emplacement caverns (Fig. 6-9), again leading to conservative estimations of consequences. There could be damage to the L/ILW near field as a result of faulting/fault re-activation, but this would be unlikely to affect radiological consequences, given that substantial mechanical degradation of the L/ILW cavern backfill cannot, in any case, be excluded by this time and is implicitly assumed in the analysis of radiological consequences, in which the contents of each cavern are assumed to be well mixed.

Finally, for each variant two calculation cases similar to those above are defined: when the fault strike is parallel to the emplacement rooms (cases WI-FLT/RA-HLW-Along and WI-FLT/RA-L/ILW-Along), the fault is assumed to pass through one, single emplacement room. When the fault strike is perpendicular to the emplacement rooms (cases WI-FLT/RA-HLW-Across and WI-FLT/RA-L/ILW-Across), the fault is assumed to pass through all emplacement rooms in the respective repository section (but not the emplacement rooms of the pilot facilities). These different fault configurations are again the same as those shown in Fig. 7-5.

3. Excavation of the repository by erosive processes

In the very long term, uplift and erosion by both glacial and non-glacial (fluvatile) processes will bring the repository closer to the surface; this phenomenon is included in the reference safety scenario. During the first one million years, the effects on the performance of the repository system are expected to be negligible (Chapters 3 and 6 of NTB 24-22 Rev. 1, Nagra 2024u).

In the more distant future, there is, however, the potential for repository material to be excavated by glacial and non-glacial erosion processes (Section 6.4 of NTB 24-17, Nagra 2024i). In Fig. 7-6, the erosion processes affecting the rock overlying the repository are shown, with fluvial incision due to the uplift and the evolution of the local topography (e.g., due to hillslope processes) comprising the non-glacial erosion processes.

Even though excavation of the repository by erosive processes is predicted to be possible only after the one-million-year time period for assessment, Nagra is, nonetheless, required by ENSI Guideline 33/649 (ENSI 2018) to analyse such events as part of the safety assessment. An “excavation of the repository by erosive processes” is thus categorised as a “what-if?” case and analysed in the safety assessment. The conceptualisations and the corresponding analysis of the radiological consequences are described in detail in NAB 24-08 Rev. 1 (Nagra 2024q).

The variants and calculation cases are grouped by the two main erosion processes, glacial erosion and non-glacial erosion:

- Three variants are defined for the glacial erosion, reflecting the three distinct post-glacial configurations: river, lake, and spring. The river variant is developed into two calculation cases, whereas lake and spring variants have one calculation case each.
- For non-glacial erosion two variants are defined: erosion of the repository on the hillside and erosion of the repository at the valley floor. The former has four calculation cases (reflecting different hillslope angles), whereas the latter has one calculation case.

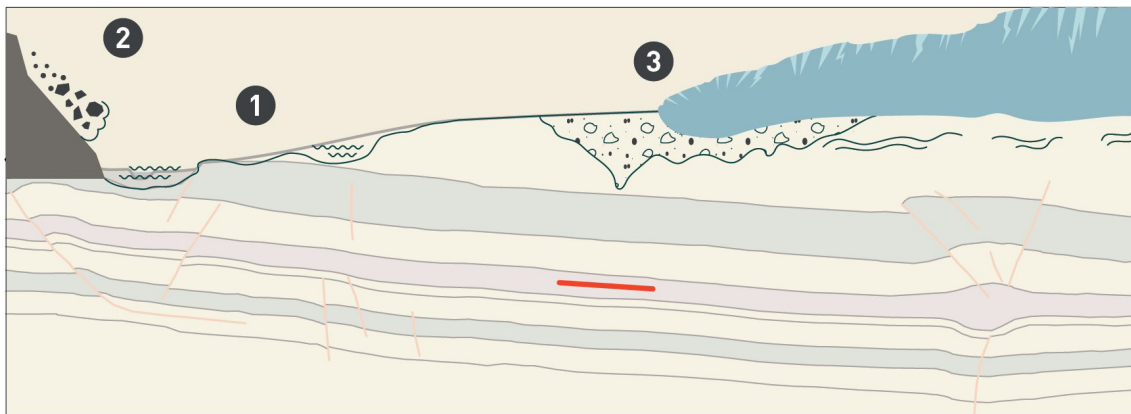


Fig. 7-6: Illustration of erosion processes affecting the rock overlying the repository (red bar), simplified from Figure 6-1 of NTB 24-17 (Nagra 2024i)

1: Fluvial incision; 2: Evolution of local topography; 3: deep glacial erosion.

7.4.2 “What-if?” cases postulating a hypothetically degraded performance of pillars of safety

Further “what-if?” cases are defined and analysed to test the robustness of the system, even in hypothetical situations in which, for no specific cause, certain key favourable features and processes fail to operate as expected or parameter values are assumed that are outside the range considered possible based on current scientific understanding. These cases are developed by systematically going through the pillars of safety¹⁷ and either removing them or assuming a degradation of the safety function they serve, as illustrated schematically in Fig. 7-7.

Each pillar of safety addressed can be considered a variant and, depending on the characteristic of each pillar, several calculation cases are defined and propagated to radiological consequence analysis.

For the CRZ these are:

- two calculation cases, one with no sorption of radionuclides in the CRZ (WI-CRZ-NOSORB) and another with hypothetically high diffusion coefficients in the CRZ (WI-CRZ-DIFF),
- two calculation cases with hypothetically high downwardly or upwardly directed hydraulic gradients (WI-CRZ-GRADdown and WI-CRZ-GRADup),
- one calculation case assuming no upper and lower confining geological units around the host rock (WI-CRZ-NOCONFINING),
- and a calculation case with hypothetically high radial dimension of the EDZ around the emplacement rooms (WI-CRZ-EDZ).

¹⁷ The only pillar of safety not considered here is a deep underground location in a stable geological environment. The “what-if?” cases related to physical phenomena described in Section 7.4.1 can, however, be seen as examples of the degradation of this pillar of safety. Specifically, if fault re-activation is caused by a large earthquake, this implies a less stable than expected geological environment. Excavation of the repository by erosion processes implies that the deep underground location no longer exists.

For the HLW matrices, the calculation cases assume a hypothetically high instant release fraction for radionuclides in spent fuel (WI-SF-IRF), hypothetically high matrix dissolution rates for spent fuel and the glass matrix for reprocessed waste (WI-HLW-DISS) and hypothetically high corrosion rates of spent fuel cladding and other structural parts (WI-SF-CORR).

For the HLW disposal canisters, a calculation case (WI-HLW-CAN) assumes the hypothetical early failure of all canisters after 1,000 years.

For the bentonite buffer in the HLW emplacement tunnels, no solubility limits (WI-HLW-NOSOL) or no sorption (WI-HLW-NOSORB) in the near field and hypothetically high diffusion coefficients (WI-HLW-DIFF) or a hypothetically high permeability of the bentonite buffer (WI-HLW-BUF) lead to four calculation cases.

For the L/ILW cementitious near field, no sorption (WI-L/ILW-NOSORB) and an instant release of all ^{14}C present (WI-L/ILW-C14) are considered.

Finally, the failure of the closure system leads to five calculation cases with hypothetically high permeabilities for the different types and locations of seals: WI-V1-HLW, WI-V1-L/ILW, WI-V2-HLW, WI-V2-L/ILW, WI-V3.

The calculation cases are also summarised in Tab. 8-8 in Section 8.4.2, where the results are presented and discussed.

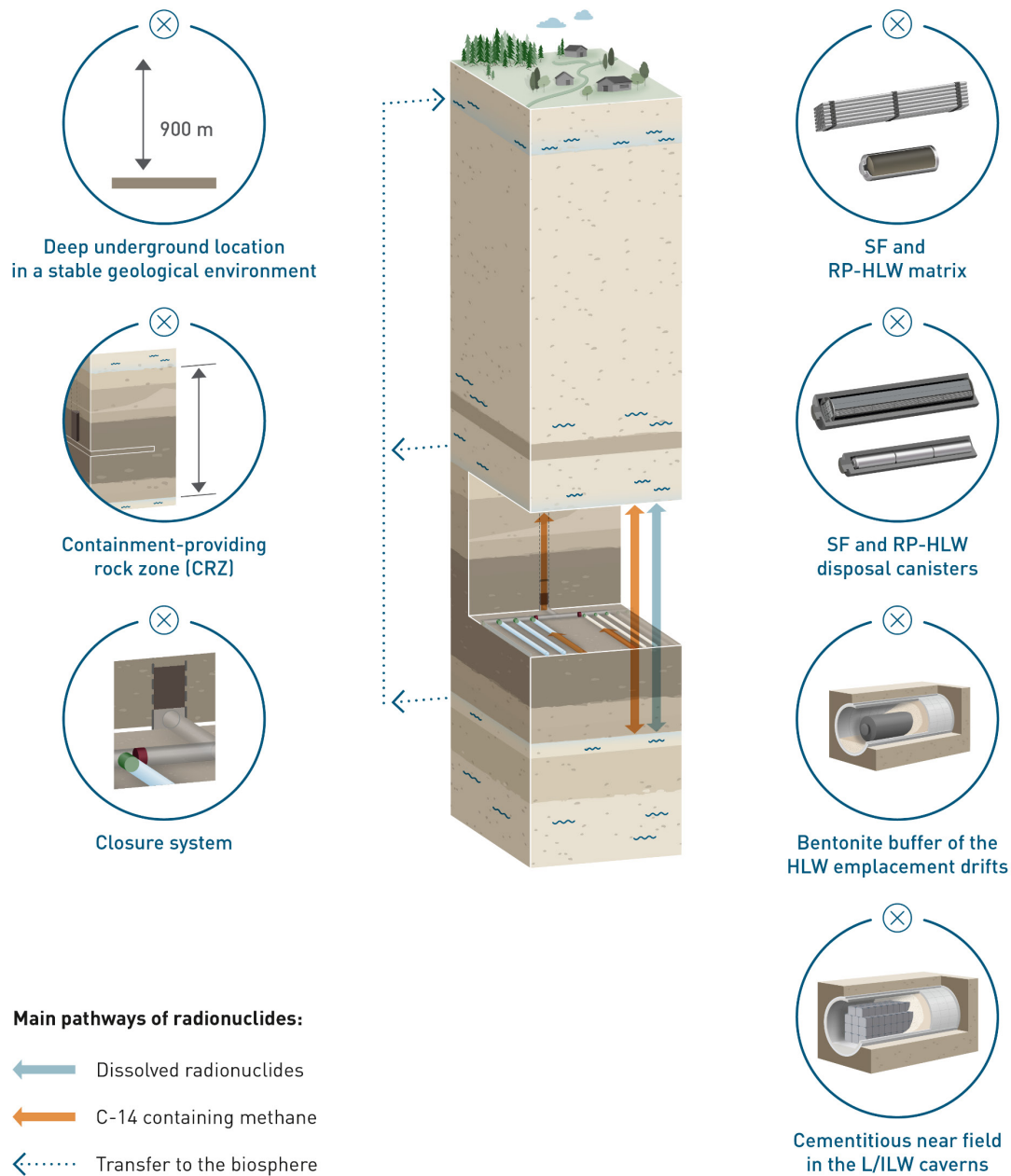


Fig. 7-7: Schematic illustration of the approach to develop “what-if?” cases postulating a hypothetically degraded performance of the pillars of safety (indicated by “x”)

7.5 Future human actions safety scenarios

The derivation and analysis of future human action (FHA) safety scenarios are described in detail in NAB 24-09 (Nagra 2024r). Two FHA safety scenarios are formulated and propagated to the analysis of radiological consequences based on a detailed FHA FEP review and discussion.

• Direct hit by a deep borehole

Drilling a deep borehole is deemed a plausible action at the site that could in principle put humans and the environment at risk. The propagated safety scenario (FHA-1) considers two variants, where the borehole penetrates either an HLW emplacement drift (FHA-1a) or an L/ILW emplacement cavern (FHA-1b), as illustrated in Fig. 7-8.

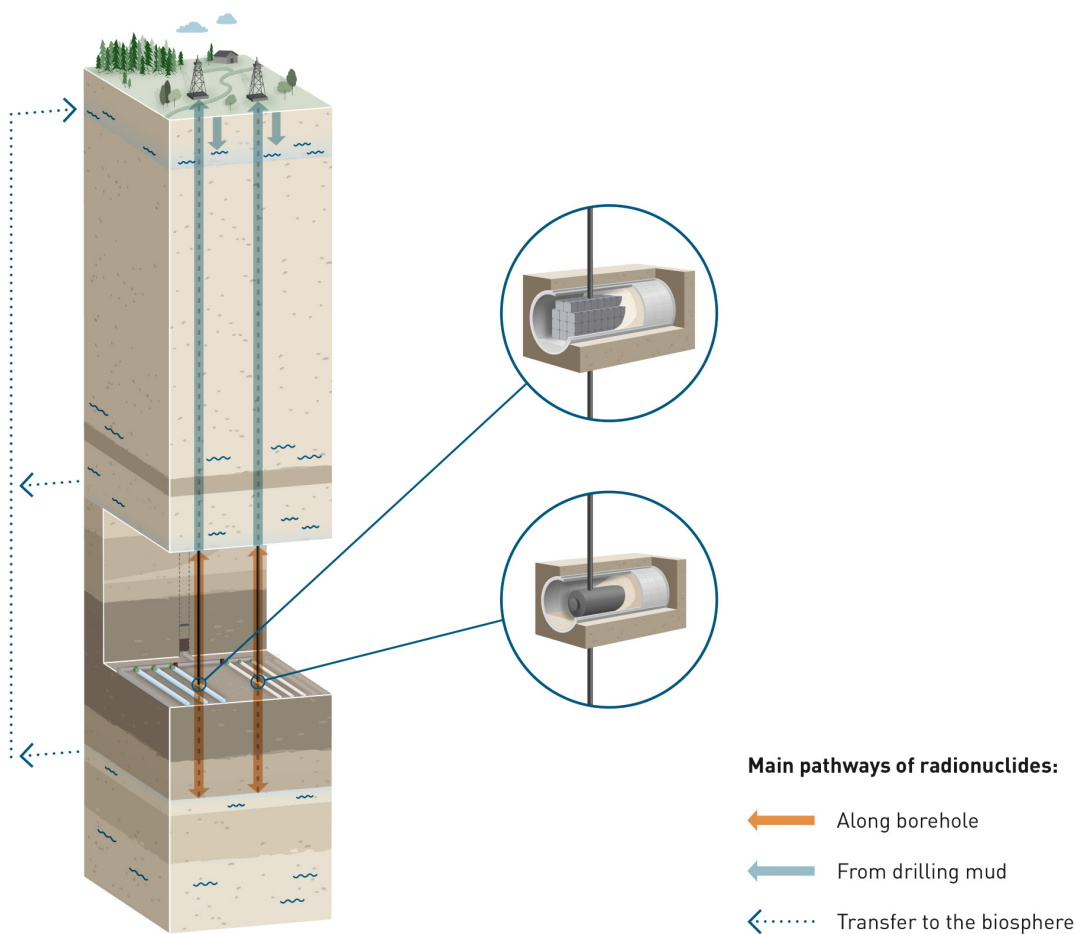


Fig. 7-8: Pathways for radionuclide transport along boreholes in the FHA-1 safety scenario

The borehole safety scenario assumes a continuous borehole through the CRZ and through either the HLW or the L/ILW repository section down into an underlying deep aquifer to allow for advective flow through the borehole. During drilling, contaminated material could be recovered with the drill mud and, with no knowledge of the possible danger, be disposed of in a gravel pit, from where dissolved radionuclides could enter the local biosphere aquifer. Additionally, deep groundwater and radionuclides could migrate through the borehole, which is assumed to remain open and is assigned various flow rates. Both exposure pathways are considered conservatively in a cumulated way.

For the HLW FHA-1 variant, the borehole is assumed to penetrate an SF disposal canister. The calculation case (labelled as FHA-1a-DH-Diss) assesses the subsequent transport of dissolved radionuclides via the aqueous phase. A two-phase flow calculation that accounts for ^{14}C in the form of methane is performed separately for the situation (FHA-1a-DH-Vol).

For L/ILW, the scenario assumes a borehole penetration of an L/ILW emplacement cavern. Three calculation cases are defined to account for the consequences of a waste package hit as well as for the consequences of penetrating a cavern. One calculation case focuses on a direct hit of a single waste package (FHA-1b-WP-Diss), from which contaminated material could be recovered with the drill mud and disposed of in a gravel pit, from where dissolved radionuclides could be transported to the local biosphere aquifer. The other two calculation cases assume that all radionuclides are homogeneously distributed within the cavern and could be released via a preferred pathway along the borehole; one case focuses on the aqueous phase (FHA-1b-PFP-Diss), while the other considers two-phase flow with a focus on ^{14}C in the form of methane (FHA-1b-PFP-Vol).

- **Repository abandonment**

This FHA safety scenario (FHA-2) addresses the abandonment of the repository during the monitoring phase, when all waste material is below ground but backfilling, sealing and closure of the tunnels and shafts are incomplete. It is assumed that groundwater entry through the open structures accelerates resaturation and leads to enhanced radionuclide transport through the shafts. Both dissolved radionuclides (FHA-2-Diss) and volatile radionuclides (FHA-2-Vol) are considered. The radiological impact is evaluated for the highly soluble, and dose-dominating, radionuclide I-129, and the volatile form (methane) of ^{14}C .

8 Analysis of radiological consequences

Aims of the chapter:

- To identify and account for uncertainties in radionuclide release, retention, and transport in each safety scenario.
- To compare the calculated main safety indicators of dose and risk with the protection criteria and to evaluate safety margins, and specifically:
 - to show that dose rates are well below the dose protection criterion for all the safety scenarios and calculation cases (except the case below),
 - to show that the risk associated with the “what-if?” case of excavation is well below the risk protection criterion for the time period for assessment.
- To assess sensitivities and the impact of uncertainties on radionuclide release, retention, and transport to demonstrate a sound understanding and illustrate the safety margins.

The analysis of radiological consequences assesses the potential impact that radionuclide releases from the repository to the surface environment could have on a representative individual from the population group most affected living in the far future. The analysis demonstrates that the quantitative protection criteria in the form of maximum individual effective dose or risk as specified in the ENSI Guideline G03 (Section 2.4) are adhered to. It furthermore illustrates the system behaviour with respect to radionuclide release, retention and transport and shows ample safety margins.

While annual individual effective dose is the most relevant measure and is evaluated and compared with the regulatory dose criteria for all calculation cases, the risk criterion is (solely) applied for the “what-if?” case of “excavation of the repository by erosive processes” during the time period for assessment.

In Section 8.1, we summarise the methodological aspect for the radiological consequence analysis and how it is embedded in the general safety assessment workflow. The main results are then presented for the safety scenarios and “what-if?” cases in Sections 8.2 to 8.5 and an overarching summary is given in Section 8.6.

Note that the full details of the radiological consequence analysis are, in general, documented in Chapter 7 of NTB 24-18 (Nagra 2024p); for the calculations linked to future human action scenarios see Chapter 4 of NAB 24-09 (Nagra 2024r) and for the “what-if?” case of an “excavation of the repository by erosive processes” see Chapters 3 and 4 of NAB 24-08 Rev. 1 (Nagra 2024q). Furthermore, the computation results are digitally available as an electronic input data and results application (EDR, Nagra 2024p).

8.1 Workflow for the radiological consequence analysis

The embedding of the analysis of radiological consequences in the general safety assessment workflow (see Chapter 4 and Section 5.4 in NTB 24-19, Nagra 2024t) is illustrated in Fig. 8-1. The starting point for the radiological consequence analysis is the set of calculation cases that has been defined in the safety scenario development as requiring quantitative evaluation, as discussed and illustrated in Chapter 7.

Each calculation case is then conceptualised in more detail. Numerical models are developed and parametrised, resulting in a set of calculations to be performed with input parameters for each of them. Care is taken that model abstractions are consistent with those made in PA and that related model uncertainty is bound and understood (see, e.g., NAB 24-25, Nagra 2024k and Appendix A and E of NTB 24-18, Nagra 2024p).

The numerical models are composed of near- and far-field components to compute radionuclide release, retention and transport through the engineered barriers and the geosphere (i.e., the Opalinus Clay and the CRZ) and a biosphere model that converts release rates from the geosphere to the biosphere into annual effective doses (Fig. 8-2).

In the near field, the focus is on the release (e.g., degradation and release rates, solubility) and retention (e.g., sorption) of radionuclides. The radionuclide source term is based on the expected inventory as described in Section 5.3.1. Due to the excellent retention properties of the repository barrier system, particularly the CRZ, only a small fraction of the radionuclides needs to be considered when calculating dose rates. These are determined using a screening process, which is based on a highly conservative and simplified transport model and, for the probabilistic calculations, in a second step based on the results of the deterministic calculations. Refer to Section 5.4.1 in NTB 24-19 (Nagra 2024t) for details.

For radiological consequence analysis, the stratigraphic units (host rock and confining geological units) are considered based on the abstraction shown in Fig. 7-2 in Section 7.2.2. The geological units are considered as individual transport media in the numerical modelling, whereby some further abstractions are made, such as merging Calcareous Lias CL and Argillaceous Keuper AK3 to a unit KEU. The Keuper aquifer is part of the KAL unit in the numerical abstractions.

For the migration of radionuclides to the boundary of the CRZ, two pathways are considered:

- radionuclide migration in the aqueous phase, in which radionuclides are released to porewater that contacts the waste packages and are subsequently transported by aqueous diffusion and, where there is porewater flow, by advection, and
- radionuclide migration of ^{14}C in both the liquid and gaseous phase¹⁸, in which ^{14}C in the form of methane mixes with, and is transported by, other repository-generated gas (hydrogen).

The near-field and geosphere conceptualisations and models for the two pathways differ as follows¹⁹:

- For the aqueous phase calculations, the near field includes either an individual SF or RP-HLW disposal canister, together with its surrounding bentonite buffer, or an individual L/ILW emplacement cavern. The HLW and its surrounding near-field barriers are represented using radially symmetric geometries, while a 2D symmetric representation of the LK90 disposal cavern as described in NAB 23-01 (Nagra 2023a) is used for the L/ILW near field. The near-field radionuclide release rates from SF and from RP-HLW are computed using the STMAN code (Robinson 2022). Releases from L/ILW are computed with the VPAC code. The main model output providing a source term for geosphere modelling is the total (positive) radial radionuclide flux across the emplacement drift or cavern into the surrounding rock. The geosphere retention and transport model is implemented in the PICNIC-TD code (Robinson & Chittenden 2022). The geosphere is modelled as a network of one-dimensional transport paths (called legs), which may represent porous media or discrete features such as fractures, cracks, or channels. Flow and transport parameter values, such as diffusion coefficient,

¹⁸ ^{14}C is the only radionuclide that can noticeably contribute to the dose rate by migrating in the gas phase (in the form of methane).

¹⁹ For a detailed description of the modelling approaches, we refer to Appendix B in NTB 24-18 (Nagra 2024p).

sorption coefficient, and hydraulic conductivity, are assigned to each leg and are assumed to be invariant in time and space within an individual leg. The radionuclide flux from different exfiltration points from the leg structure is captured and summed together to obtain the release rate from the geosphere into the biosphere.

- For the two-phase flow calculations, a 3D model combining both the near field and the geosphere is used. The implementation is based on the TOUGH3 code with its equation-of-state EOS75R module²⁰. It simulates two-phase flow of a gas phase and an aqueous phase and advective- diffusive transport of ¹⁴C as methane, with fully implicit thermo-hydro-mechanical coupling between primary variables (i.e., the state variables of pressure, gas saturation, temperature, mass fractions) and secondary variables (porosity, fluid density, fluid viscosity, capillary pressure) (Jung et al. 2018, Zhang et al. *in prep.*). In order to account for the different needs of dimensionality and discretisation in the near field and geosphere, a so-called “macro-element” approach is adopted, by which the number of the numerical elements is significantly reduced by simplifying the similar and (quasi) symmetrically positioned repository structures along with the surrounding rock to representative macro-elements. ¹⁴C fluxes from the model boundary are captured and summed together to obtain the release rate from the geosphere into the biosphere.

The biosphere model serves to convert radionuclide fluxes to an effective dose by providing so-called biosphere dose conversion factors (BDCFs). These BDCFs are derived from modelling of radionuclide transfer and accumulation processes in the surface environment and of radionuclide pathways leading to the exposure of humans to radioactivity. The biosphere modelling is carried out using a compartment modelling framework and is implemented with the SwiBAC code. For details of the biosphere modelling approach, see NAB 24-06 (Nagra 2024n). The biosphere model is identical for most calculation cases (use of reference biosphere), albeit for the reference safety scenario four different biosphere situations are considered, leading to four separate calculation cases (reference calculation case and three additional calculation cases, as shown in Fig. 7-3). Only the “what-if?” case “excavation of the repository by erosive processes” and some future human action safety scenarios require adaptations of the biosphere model due to potential input of radionuclides directly into the surface environment (for details see NAB 24-08 Rev. 1, Nagra 2024q and NAB 24-09, Nagra 2024r).

Note that in general, conservatively, radionuclide fluxes out of the CRZ are directly multiplied by the BDCFs. This corresponds to neglecting the transport time in between the boundary of the CRZ and the biosphere near-surface aquifer, that is, not taking credit for radioactive decay or for dispersion and dilution during this transport. This is illustrated in the scenario figures with the blue dashed lines (e.g., Fig. 7-3).

²⁰ The EOS75R module is an adaptation of the standard TOUGH3 module EOS7R, taking into account the presence of hydrogen gas.

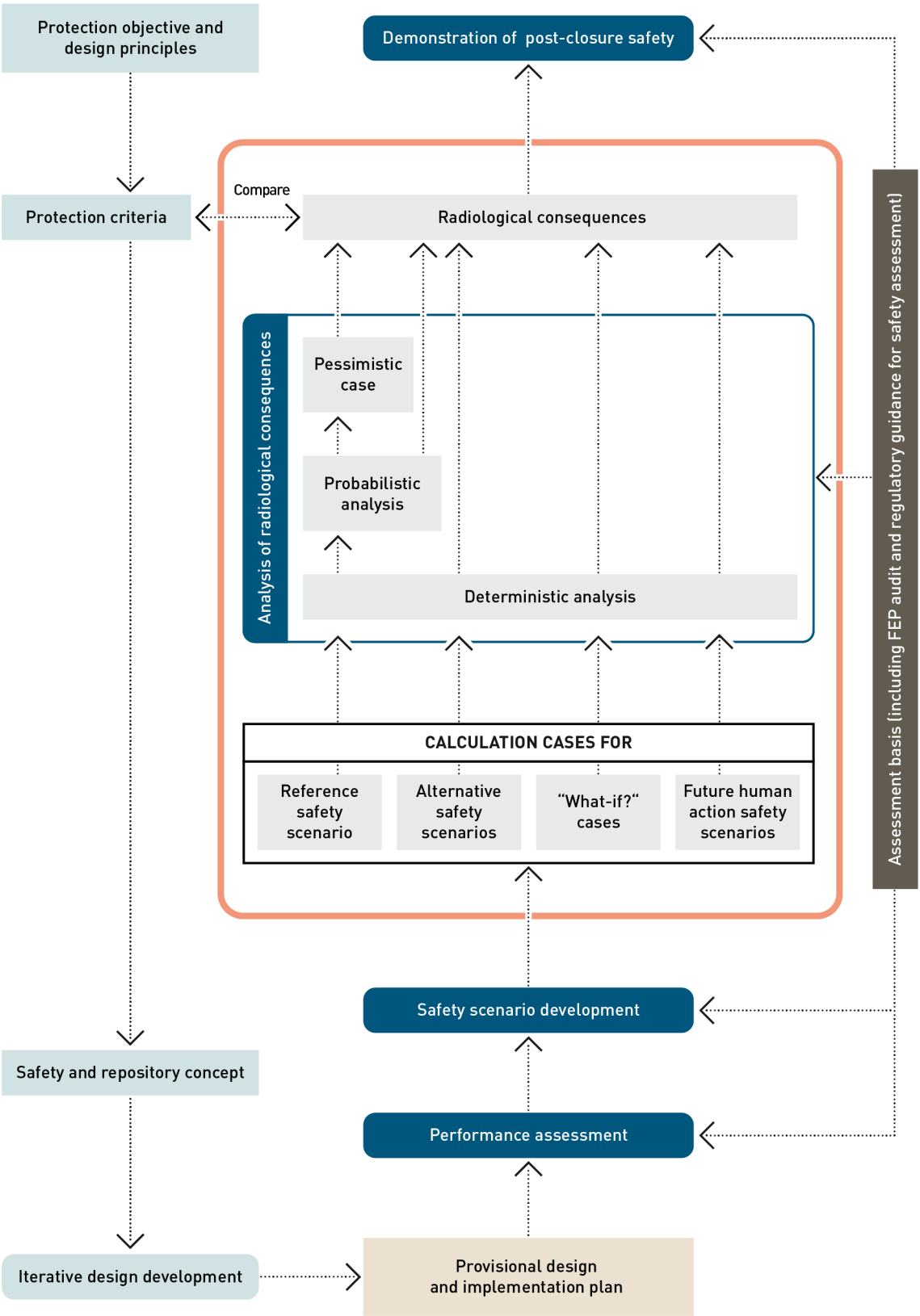


Fig. 8-1: Workflow for the post-closure safety case, highlighting, in the orange box, the main elements of the analysis of radiological consequences

When parametrising the models used in calculations, care is taken to distinguish between realistic (best estimate) values, conservative or pessimistic assumptions, and stylised approaches (see also Section 4.4). In general, when there is quantified uncertainty in the properties of repository features, rates of processes and other model parameters, realistic values and, for the probabilistic calculation case, realistic parameter bandwidths, are assumed. Other uncertainties are generally handled using conservative assumptions. Parameter assumptions and details of the treatment of parameter uncertainty can be found in Appendix C in NTB 24-18 (Nagra 2024p). Stylised assumptions are made where appropriate. They are used in the context of the biosphere modelling, the “what-if?” case of “excavation of the repository by erosive processes” and future human actions safety scenarios. Further, the assumption is made that the different biospheres (e.g., constant climatic conditions) are maintained throughout the time period for assessment.

Once the models and parametrisations are defined, simulations are performed and the results are grouped in the electronic input data and results application (EDR, Nagra 2024p) such that each result can be linked to its input parameters and model chain. Finally, the results are compared with the regulatory protection criteria and the safety margins are evaluated. For this, dose rates as a function of time and the maximum dose rate over the time period for assessment are analysed and compared. For the calculation cases linked to the “what-if?” case of “excavation of the repository by erosive processes”, the risk is also evaluated and compared with the regulatory risk criterion for these calculation cases.

Organic ^{14}C is considered as methane in both the aqueous phase calculations and in the two-phase flow calculations. Because the entire organic ^{14}C inventory is considered separately in both calculations, the results are also presented separately as alternative model assumptions, i.e., one in which all methane is assumed to be fully dissolved despite its volatility and the low solubility of methane gas and one in which methane is partitioned between the dissolved and gaseous phases. In fact, summing both results would correspond to counting the organic ^{14}C inventory twice. A further reason for presenting the results separately is that otherwise the distinct characteristics of the two-phase flow results would be barely visible, since the absolute values of dose rate contributions for these cases are small when compared with the aqueous phase contributions.

Note that, in the graphical representation of the radiological dose calculations shown in the following sections, the grey-shaded areas indicate two aspects:

- Dose rate: Many of the calculations presented in this chapter show very low dose rates. A dose of 10^{-7} mSv/a to an individual corresponds to an extremely low overall fatal risk coefficient for the exposed person of 5×10^{-12} per year²¹. This is reflected in the graphical presentation of such results, where grey shading is used below 10^{-7} mSv/a to emphasise that such results have no radiological significance.
- Extended time period: Shading is also used in graphs of dose-rate evolution, where results are presented beyond the time period for assessment of one million years, again to illustrate the behaviour of the models representing the disposal system, as well as to ensure that maximum dose rates are captured.

²¹ This estimate is based on the ICRP recommended factor for the conversion of dose to risk of 0.05 Sv per year (ICRP 2007a).

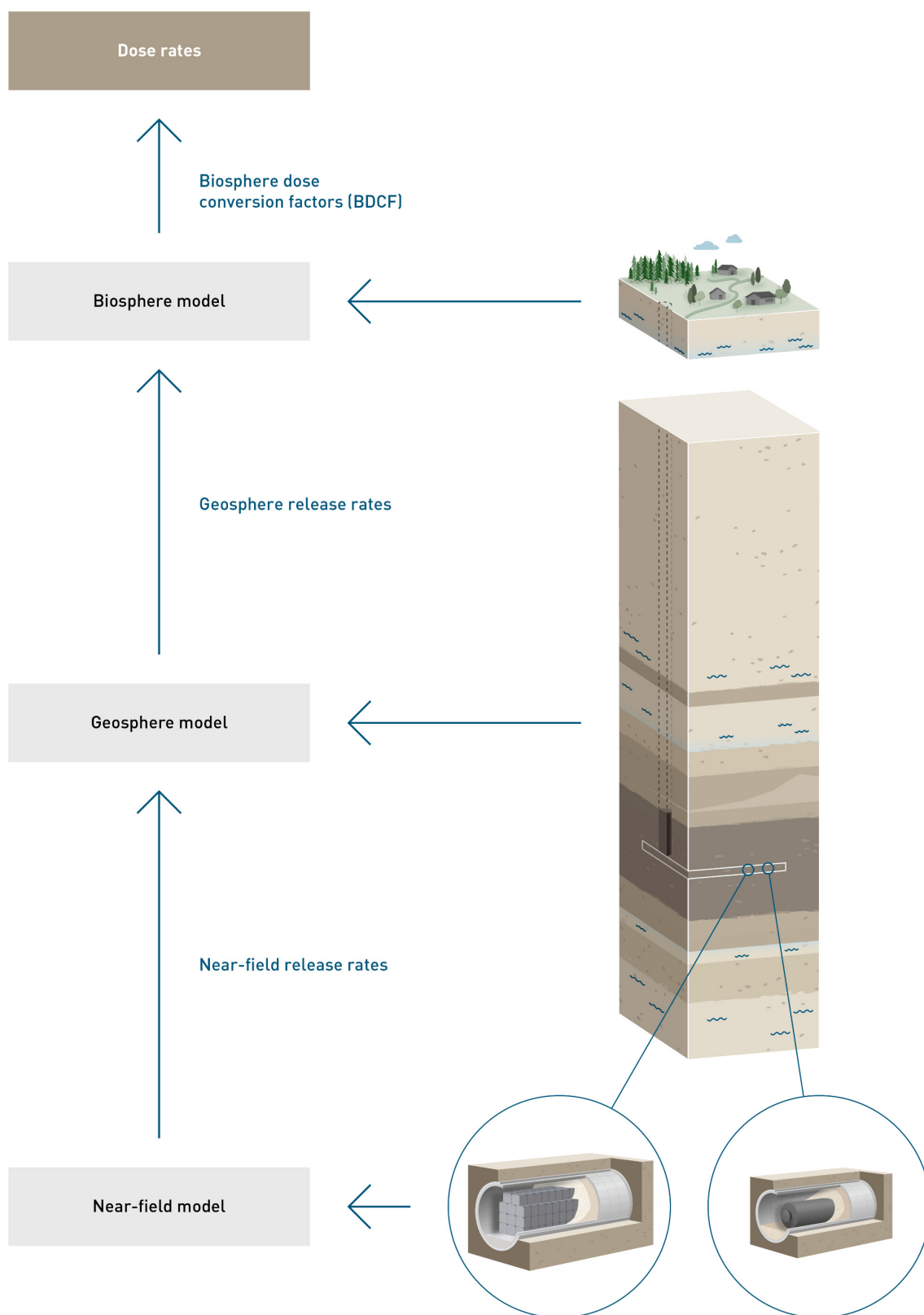


Fig. 8-2: Workflow for the computation of radiological consequences for an individual calculation

8.2 Results for the reference safety scenario

The reference safety scenario represents the expected initial state and expected evolution over time. Several deterministic calculation cases and one probabilistic calculation case, as summarised in Tab. 8-1 and described in Section 7.2, have been defined for the reference safety scenario. Calculation cases for the reference conceptualisation of the CRZ cover the reference case, alternative climatic conditions and one different geomorphology (different biospheres), and one calculation case addresses parameter uncertainty. An additional calculation case uses a variant conceptualisation of the CRZ with a hydraulically inactive Keuper aquifer to account for uncertainty in geological characterisation.

Computed dose rates for the reference calculation case RS-RC resulting from radionuclide migration in the aqueous phase are shown in Fig. 8-3, where the contributions of the HLW (top), L/ILW (middle) and the total dose rate for the repository (bottom) are distinguished. In each subfigure, the radionuclides that contribute significantly (and visibly) to the total dose rate are also plotted. It becomes obvious that only three radionuclides, namely I-129, Cl-36 and Se-79, reach the biosphere in sufficient amounts to contribute to the dose rates. Ca-41 only appears in the dose rate in which the contributions from both HLW and L/ILW are combined and has no radiological significance. All the other radionuclides decay to such a degree during their transport from the waste through the near field and the CRZ that their contributions are negligible, i.e., below 10^{-9} mSv/a, and thus not visible in the plots. The three contributing types of radionuclides are characterised by a sufficiently large inventory, low sorption coefficients, and comparably long half-lives.

While the relative contribution of the three radionuclide types differs between the L/ILW and HLW repository sections, and also changes over time, it is clearly the HLW repository section that dominates the overall dose rates, with I-129 being the radionuclide with the highest contribution.

The total dose rate reaches 1.5×10^{-4} mSv/a after one million years, which is about three orders of magnitude lower than the protection criterion of 0.1 mSv/a, indicating a large safety margin. Note that the total dose rate continues to increase slightly thereafter, reaching a maximum of 2.0×10^{-4} mSv/a after about 4.5 million years, but remains about four orders of magnitude below the typical present-day natural radiological exposures in Switzerland and as published by the Swiss Federal Office of Public Health in their annual reports on radiological monitoring of the environment (“Umweltradioaktivität”, e.g. BAG 2021).

A more detailed analysis of the radionuclide transport distance while considering decay is provided by analysing the dose rate contribution of radionuclides as a function of the distance from the near field (Fig. 8-4). For this, selected radionuclide fluxes were captured in the geosphere model every 10 m, and a hypothetical dose rate contribution per radionuclide was computed, assuming that the exfiltration pathway would be at these locations, i.e., the radionuclide fluxes at these hypothetical interfaces every 10 m were converted to dose rates using the BDCF from the reference calculation case. The excellent retention capacity of the near field as well as that of the Opalinus Clay (OPA) become evident. The same radionuclides as in Fig. 8-3 (I-129, Cl-36, Se-79 and Ca-41) leave the host rock and penetrate further into the upper (UCU) and lower confining geological units (LCU) and eventually reach the biosphere.

Tab. 8-1: Calculation cases for the reference safety scenario

All cases are calculated using both modelling approaches: aqueous-phase transport and two-phase transport of ^{14}C .

Variant	Case	Description
Reference conceptualisation of CRZ. Hydraulically active Keuper aquifer	RS-RC	Reference case: Realistic (best estimate) assumptions and parameters whenever available (else conservative) Present-day climatic conditions (reference biosphere) Note that one conservative assumption in RS-RC is that the Keuper aquifer is hydraulically active and is thus the main exfiltration layer in the geological units below the repository
	RS-BIOS/WARM_DRY	As the RS-RC but: Biosphere for warmer and drier climate
	RS-BIOS/COLD	As the RS-RC but: Biosphere for colder climate
	RS-BIOS/DRAINED_FARMLAND	As the RS-RC but: Biosphere assuming a drained farmland landscape
	RS-PROB	Based on the RS-RC but: A probabilistic simulation with multiple realisations in which parameters are sampled at random from probability distributions, derived from an assessment of parameter uncertainty. Illustrates the range of uncertainty in calculated consequences and provides the data for a global sensitivity analysis
	RS-CMB	Based on the RS-RC but: Deterministic case in which the parameters to which the radiological consequences are most sensitive, as identified in the probabilistic analysis (RS-PROB), are set to their most unfavourable values, or from the reference value within their probability distributions, respectively
Hydraulically inactive Keuper aquifer	RS-KEUPER_INACTIVE	As the RS-RC but: Hydraulically inactive Keuper aquifer below the HLW repository section of the repository

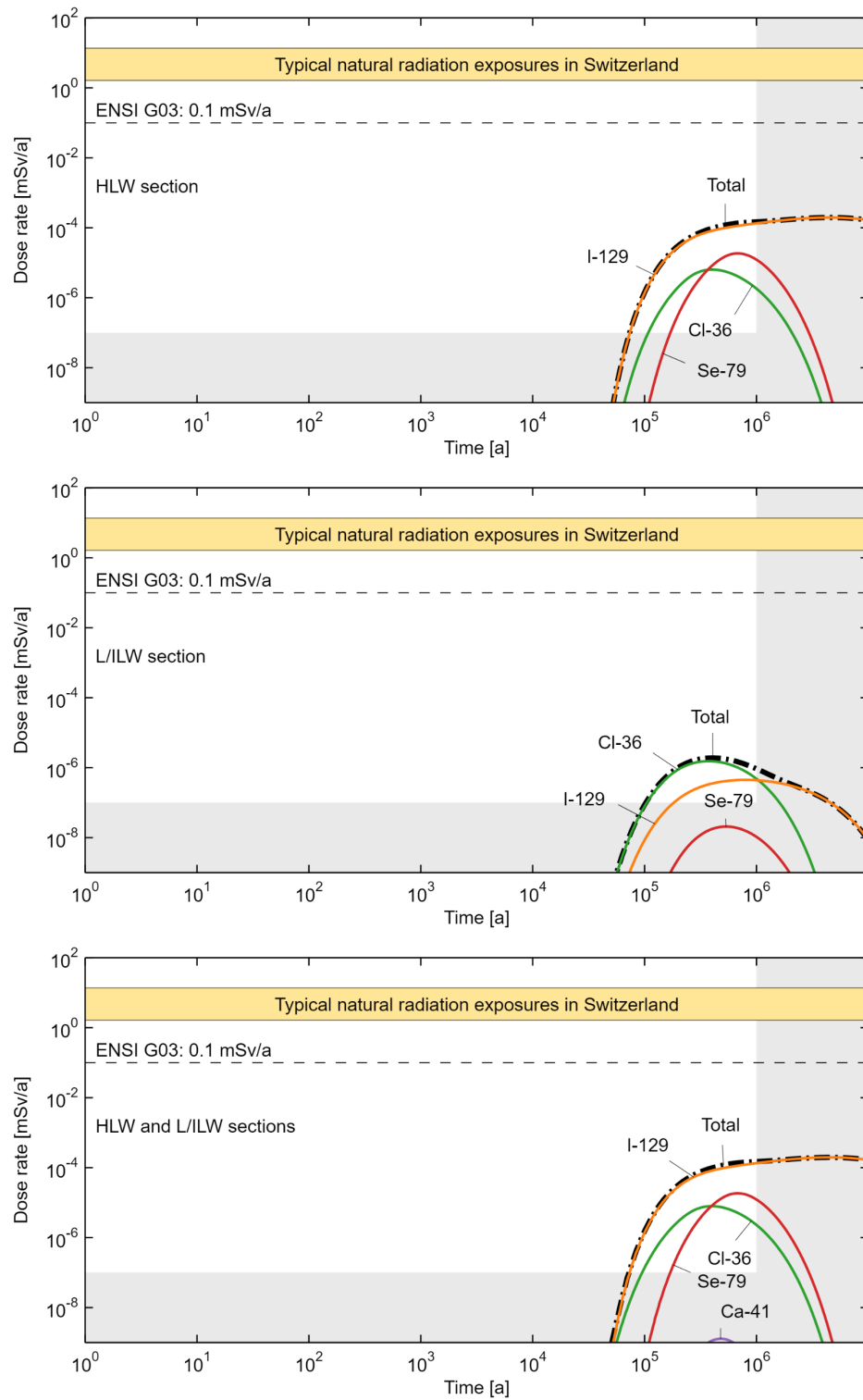


Fig. 8-3: Dose rates and main contributing radionuclides for the reference calculation case RS-RC assuming radionuclide transport in the aqueous phase

Contribution from HLW repository section (top); contribution from L/ILW repository section (middle); total dose rate for the repository (bottom).

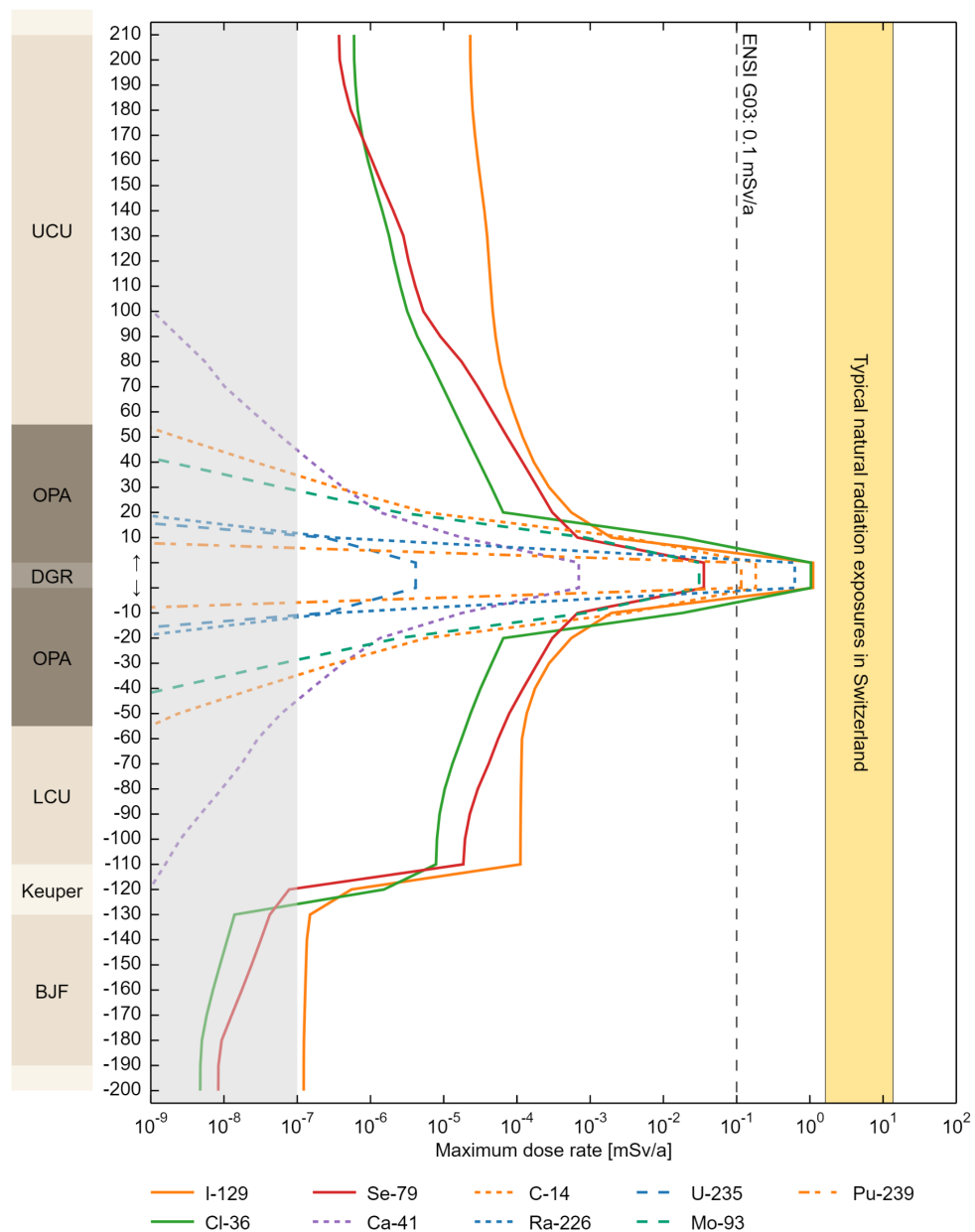


Fig. 8-4: Hypothetical dose rate contributions of individual radionuclide types for the reference calculation case RS-RC in the aqueous phase based on radionuclide fluxes captured in the near field (one value) and every 10 m in the geosphere model

UCU = upper confining geological units, OPA = Opalinus Clay, DGR = deep geological repository, LCU = lower confining geological units, BJB = Bänkerjoch Formation, groundwater markers indicate the deep groundwater aquifers: Malm (at the top of the UCU), Keuper (in-between the LCU and the BJB) and Muschelkalk (below the BJB). For a selected dose rate on the x-axis (e.g., 10^{-5} mSv/a), the reading of the distance on the y-axis (e.g., -30 m) indicates how far a signal of a given radionuclide, expressed as maximum dose rate contribution, reaches (e.g., Ca-41 would contribute ca. 10^{-6} mSv/a if the flux of Ca-41 at 30 m into the geosphere could be captured and directly transferred into the biosphere aquifer). Radionuclides with dashed lines would contribute less than 10^{-7} mSv/a at the boundary of the Opalinus Clay, ca. 50 m away from the centre of the repository into the geosphere. The sharp drop in the Keuper is due to dilution effects in the aquifer.

One key result of the probabilistic calculation case for the reference safety scenario, RS-PROB, is shown in Fig. 8-5. The bandwidth of calculated dose rates assuming radionuclide transport in the aqueous phase is shown using the statistical measures of median, 95/5 and 84/16 percentiles, and the maximum and minimum dose rates (individually computed at each time step as maximum and minimum across all the probabilistic calculations).

The dose rates for the probabilistic calculation case remain comfortably two orders of magnitude below the dose protection criterion of 0.1 mSv/a, thus demonstrating an ample safety margin with respect to parameter uncertainty.

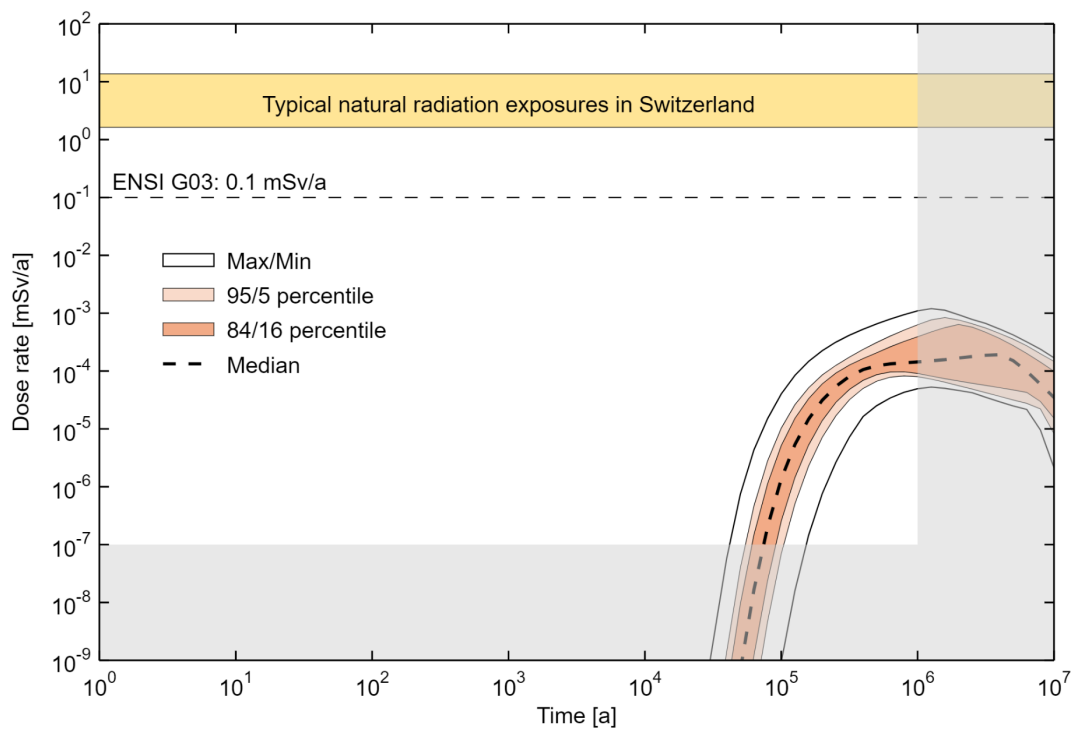


Fig. 8-5: Dose curves and bandwidths for the probabilistic calculation case assuming radionuclide transport in the aqueous phase

The statistical measures of median, 95/5 and 84/16 percentiles, and the maximum and minimum dose rates individually computed each time as maximum and minimum across all the probabilistic calculations are shown (i.e., the Max/Min lines do not correspond to one calculation case).

So far, we have summarised the results of the reference calculation case and the probabilistic calculation case for the reference safety scenario when considering radionuclide transport in the aqueous phase only. In fact, this is the pathway that contributes most to the dose rates and dominates them. This is illustrated in Fig. 8-6 which shows the maximum dose rates calculated for the probabilistic calculation case RC-PROB in the form of two boxplots, one where only aqueous phase transport is modelled for all the radionuclides (left side) and one for ^{14}C which, in the form of methane, is the only radionuclide that could significantly contribute to dose rates by migration in the gaseous phase (right side). The maximum dose rates are orders of magnitude smaller in the latter case.

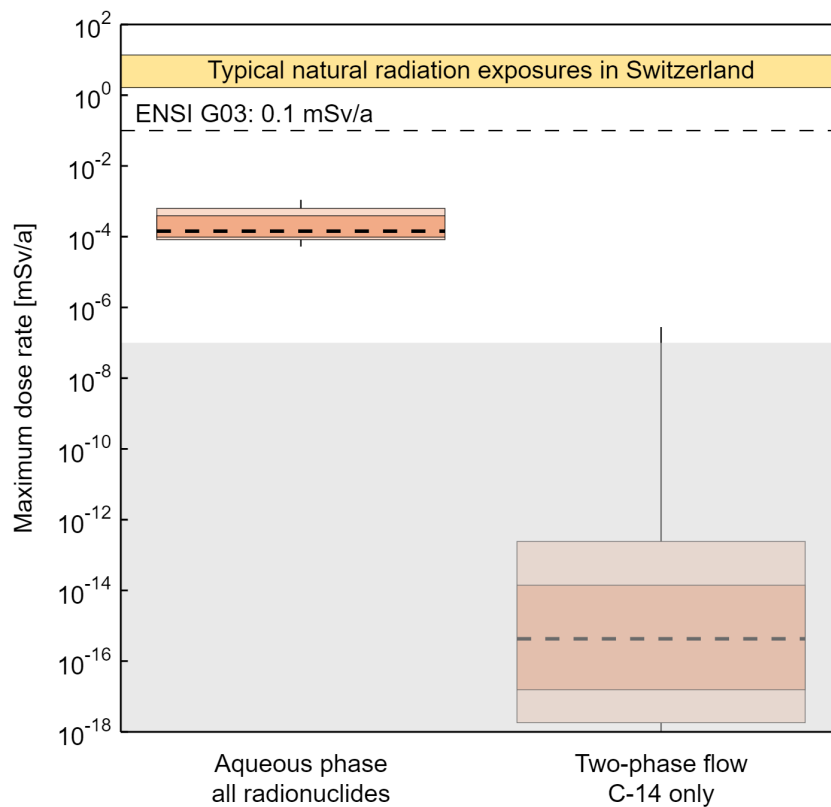


Fig. 8-6: Summary showing the maximum dose rates for the probabilistic calculation case for the aqueous phase transport (left) and two-phase transport of ^{14}C (right)

The maxima are calculated over a total calculation period of 10^6 years (see Fig. 8-5). The boxplots show the statistical measures of median (dashed horizontal line), 95/5 and 84/16 percentiles (light and dark shades of red) and the full extent of the spread between minimum and maximum (thin vertical line).

Both the ample safety margin and the dominance of the aqueous phase transport path are confirmed when analysing the dose rates for all the deterministic cases linked to the reference scenario (Fig. 8-7). For the aqueous phase transport (Fig. 8-7a), the results for the warmer and drier biosphere (RS-BIOS/WARM_DRY) and for the pessimistic calculation case (RS-CMB) lead to dose rates that are roughly within the probabilistic bandwidth of the calculation case RS-PROB. However, they are – expectedly – higher than the dose rates of the reference calculation case (RS-RC). The results of the calculation case with assumptions of the inactive Keuper aquifer below the HLW repository section (RS-KEUPER_INACTIVE) and of the two cases with a colder (RS-BIOS/COLD) or a drained farmland biosphere (RS-BIOS/DRAINED_FARMLAND) with lower BDCFs, lead to dose rates that are one to four orders of magnitude below those of the reference calculation case – leaving very large safety margins.

In Fig. 8-7b, the dose rate for the pessimistic calculation case RS-CMB is shown for the two-phase transport of ^{14}C (mainly transported as methane). It is larger than 10^{-7} mSv/a for a short time – but still very low. The resulting dose rates for the other deterministic calculation cases for the reference scenario and this transport pathway all remain below 10^{-9} mSv/a.

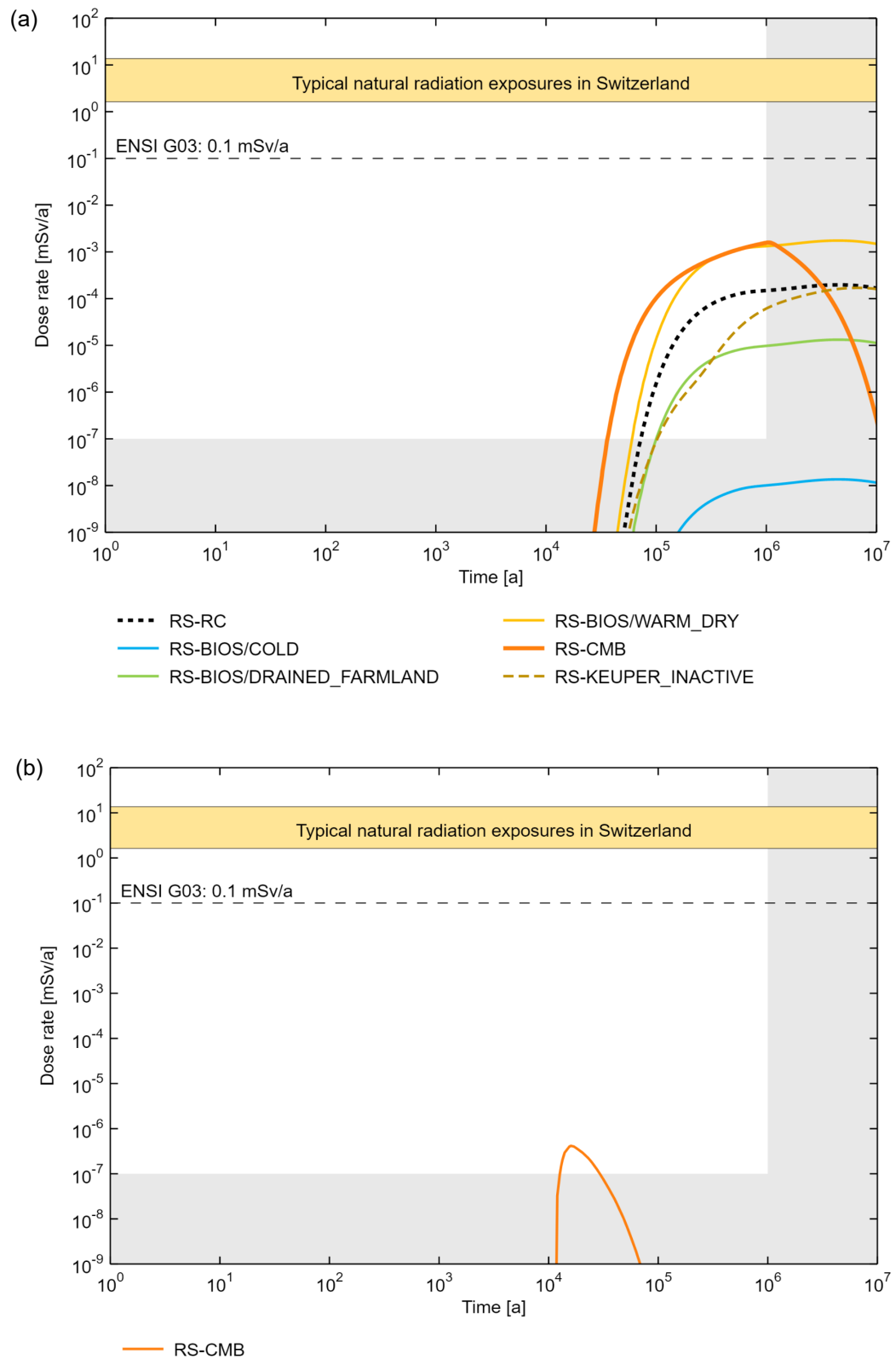


Fig. 8-7: Dose curves for the reference case and for other deterministic calculation cases within the reference scenario, showing the contributions of aqueous phase transport (a) and two-phase transport of ^{14}C (b)

In summary, both the probabilistic and deterministic results for the reference safety scenario demonstrate that the regulatory dose protection criterion is respected. The probabilistic results show a safety margin of more than two orders of magnitude for the 95 percentile and close to three orders of magnitude for the median. The most pessimistic deterministic case has a maximum dose rate occurring after roughly one million years that is still about two orders of magnitude below the regulatory protection criterion. Thus, for the expected evolution of the repository and considering parameter uncertainty within this evolution path, it can be demonstrated with a high level of confidence that humans and the natural environment will not be exposed to harm.

8.3 Results for the alternative safety scenarios

As described in Section 7.3, two alternative safety scenarios that address uncertainties in geological characterisation have been identified:

- one of the confining geological units above (variant one) or below the Opalinus Clay (variant two) provides a horizontally connected advective pathway connected to the regional aquifer, and thus provides an additional pathway for advective transport of radionuclides,
- an undetected fault, with a transmissivity that is in the range of what could reasonably be expected in such a scenario, exists in the vicinity of the repository at the time of its construction. The undetected fault could intersect either the HLW or the L/ILW repository section, leading also to two variants for this alternative safety scenario.

Finally, sets of calculation cases are considered for the two alternative safety scenarios as summarised in Tab. 8-2 and Tab. 8-3 (for details and illustrations see Section 7.3). Tab. 8-2 also indicates which radionuclide transport pathways are considered for which calculation case for the first alternative scenario. In fact, since gas-phase-mediated transport is directed upwards, the two-phase flow computation accounting for ^{14}C as methane is used – conservatively – for the closest hydraulically active unit above the repository.

Tab. 8-2: Calculation cases for the alternative safety scenario that assumes an additional hydraulically active unit within the CRZ above or below the Opalinus Clay

The transport modelling approach refers to the two modelling approaches: one that assumes all radionuclides dissolve and migrate through the repository barrier system in the aqueous phase, and one in which gaseous ^{14}C migrates, along with other repository-generated gas, in a two-phase flow regime. Only case AS-HA/SCD is modelled by the latter approach, since this is the most conservative case.

Variant	Case	Transport modelling approach		Description
		Aqueous phase	Two-phase	
Additional hydraulically active unit within the CRZ above the Opalinus Clay	AS-HA/SCD	✓	✓	Hydraulically active “SCD” unit (Sandy-Calcareous Dogger)
	AS-HA/CD	✓	✗	Hydraulically active “CD” unit (Calcareous Dogger)
	AS-HA/CM	✓	✗	Hydraulically active “CM” unit (Calcareous Malm)
Additional hydraulically active unit within the CRZ below the Opalinus Clay	AS-HA/KEU	✓	✗	Hydraulically active “KEU” unit (Calcareous Lias/Argillaceous Keuper)
	AS-HA/KAU	✓	✗	Hydraulically active “KAU” unit (Dolomitic and Argillaceous Keuper)

Tab. 8-3: Calculation cases for the alternative safety scenario that assumes an undetected fault

All cases are calculated using both modelling approaches: aqueous-phase transport and two-phase transport of ^{14}C .

Variant	Case	Description
Fault intersects the HLW emplacement drifts	AS-FLT-HLW-Along	Fault strike runs along the axis of a single emplacement drift
	AS-FLT-HLW-Across	Fault strike runs perpendicular to drifts, cutting each drift at its centre
Fault intersects the L/ILW emplacement caverns	AS-FLT-L/ILW-Along	Fault strike runs along the axis of a single emplacement cavern
	AS-FLT-L/ILW-Across	Fault strike runs perpendicular to caverns, cutting each cavern at its centre

The results in the form of dose rates are shown for all the calculation cases of the alternative safety scenarios in Fig. 8-8 for both (a) the aqueous phase and (b) two-phase transport. For the aqueous phase results, the reference case is also shown for comparison (for two-phase transport, the dose rate for the reference case is negligible). It can be easily noted that the presence of hydraulically active units has a limited effect on the results. Dose rates can increase minimally and appear only slightly earlier compared with the reference case.

In contrast, the presence of an undetected fault could lead to releases to the biosphere at significantly earlier times, especially in the cases when the fault intersects the L/ILW emplacement caverns. An undetected fault also leads to a noticeable – albeit still very low – dose rate contribution of ^{14}C as methane that migrates via a gas phase. However, for this alternative safety scenario, the overall dose rate maxima are again only marginally affected.

In general, similarly comfortable safety margins remain when compared with the reference safety scenario.

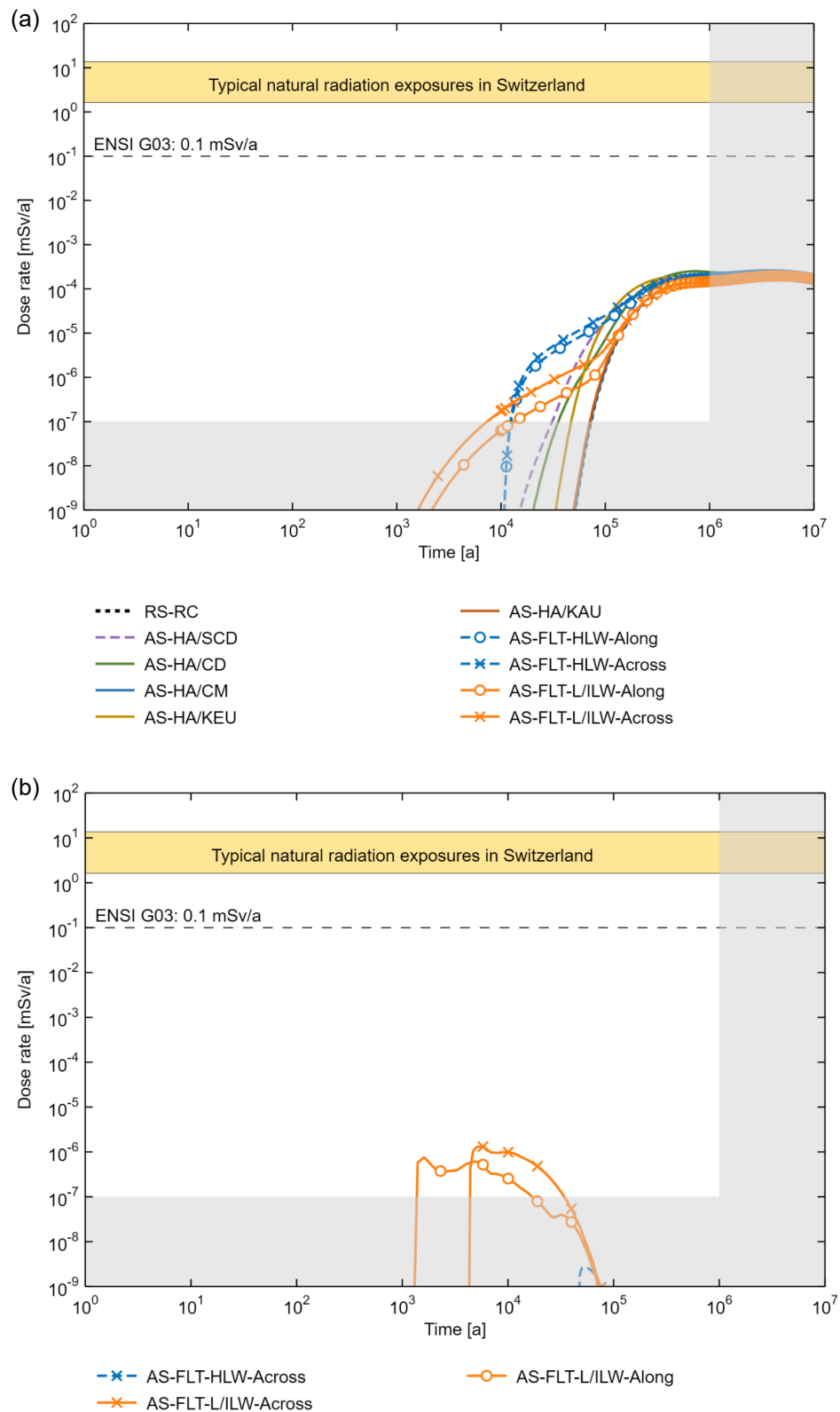


Fig. 8-8: Dose curves for the calculation cases of the alternative safety scenarios and radionuclide transport in the aqueous phase (a) and for the two-phase transport of ^{14}C (only 3 cases lead to dose rates higher than 10^{-9} mSv/a) (b)

8.4 Results for the “what-if?” cases

As discussed in Section 7.4, “what-if?” cases are split into those related to real physical phenomena and additional hypothetical cases with degraded or removed pillars of safety.

8.4.1 “What-if?” cases related to physical phenomena

Occurrence of an undetected fault with a hypothetically high transmissivity in the vicinity of the repository

Two “what-if?” cases related to the physical phenomenon of geological faults were identified (see Section 7.4.1):

- the occurrence of an undetected fault with a hypothetically high transmissivity in the vicinity of the repository,
- the future creation or re-activation of a fault either by geological processes or as a repository-induced effect after repository closure.

For each of these “what-if?” cases, two variants consider whether the HLW or L/ILW repository section is affected and two calculation cases per variant address the fault orientation. The calculation cases are summarised in Tab. 8-4 and Tab. 8-5. For details and illustrations see Section 7.4.1. Note that the different timing of fault re-activation for the HLW and L/ILW variants corresponds to conservative assumptions with respect to potential radiological consequences (see Section 7.4.1).

Tab. 8-4: Calculation cases for the “what-if?” cases for an undetected fault with hypothetically high transmissivity

All cases are calculated using both modelling approaches: aqueous-phase transport and two-phase transport of ^{14}C .

Variant	Case	Description
Fault with hypothetically high transmissivity intersects the HLW emplacement drifts	WI-FLT-HLW-Along	Fault, with a transmissivity of $10^{-7} \text{ m}^2/\text{s}$, with strike running along the axis of a single emplacement drift
	WI-FLT-HLW-Across	Fault, with a transmissivity of $10^{-7} \text{ m}^2/\text{s}$, with strike running perpendicular to drifts, cutting each drift at its centre
Fault with hypothetically high transmissivity intersects the L/ILW emplacement caverns	WI-FLT-L/ILW-Along	Fault, with a transmissivity of $10^{-7} \text{ m}^2/\text{s}$, with strike running along the axis of a single emplacement cavern
	WI-FLT-L/ILW-Across	Fault, with a transmissivity of $10^{-7} \text{ m}^2/\text{s}$, with strike running perpendicular to caverns, cutting each cavern at its centre

Tab. 8-5: Calculation cases for the “what-if?” cases for fault re-activation by geological processes or as a repository-induced effect

All cases are calculated using both modelling approaches: aqueous-phase transport and two-phase transport of ^{14}C .

Variant	Case	Description
Newly created or re-activated fault intersects the HLW emplacement drifts	WI-FLT/RA-HLW-Along	Fault, with a transmissivity of $10^{-7} \text{ m}^2/\text{s}$ created at 10,000 years, with strike running along the axis of a single emplacement drift
	WI-FLT/RA-HLW-Across	Fault, with a transmissivity of $10^{-7} \text{ m}^2/\text{s}$ created at 10,000 years, with strike running perpendicular to drifts, cutting each drift at its centre
Newly created or re-activated fault intersects the L/ILW emplacement caverns	WI-FLT/RA-L/ILW-Along	Fault, with a transmissivity of $10^{-7} \text{ m}^2/\text{s}$ created at 20,000 years, with strike running along the axis of a single emplacement cavern
	WI-FLT/RA-L/ILW-Across	Fault, with a transmissivity of $10^{-7} \text{ m}^2/\text{s}$ created at 20,000 years, with strike running perpendicular to caverns, cutting each cavern at its centre

The results in the form of dose rates are shown for the calculation cases of these “what-if?” cases in Fig. 8-9 for both (a) the aqueous phase and (b) two-phase transport. For the aqueous phase results, the reference case is also shown for comparison (for two-phase transport, the dose rate for the reference case is negligible).

The two calculation cases with an undetected fault with hypothetically high transmissivity intersecting L/ILW emplacement caverns (WI-FLT-L/ILW-Along and -Across) have highest dose rates for the aqueous phase, being about one order of magnitude below the dose protection criterion (Fig. 8-9a). There is not a big difference between the fault with strike running along the axis of a single emplacement cavern and the one striking perpendicular to caverns, cutting each cavern at its centre. The calculation cases of a newly created or re-activated fault after 20,000 years intersecting L/ILW emplacement caverns (WI-FLT/RA-L/ILW-Along and -Across) have much lower dose rates that are only one order of magnitude above the reference case.

The dose rates for similar calculation cases for faults intersecting the HLW emplacement drift(s) are more than one order of magnitude below the dose protection criterion. The calculation cases WI-FLT-HLW-Along and WI-FLT-HLW-Across are almost identical to the calculation cases WI-FLT/RA-HLW-Along and WI-FLT/RA-HLW-Across and therefore not visible in Fig. 8-9.

For the two-phase flow results (Fig. 8-9b), the highest dose rates come from the same calculation cases as for the aqueous phase, which are also about one order of magnitude below the dose protection criterion. WI-FLT-L/ILW-Across gives higher dose rates than WI-FLT-L/ILW-Along because the fault cuts each drift at its centre and the gas is able to escape from each cavern along the fault.

The HLW calculation cases for the two-phase flow are two orders of magnitude and more below the protection criteria. WI-FLT-HLW-Along is not visible in the figure since it is almost identical to WI-FLT/RA-HLW-Along.

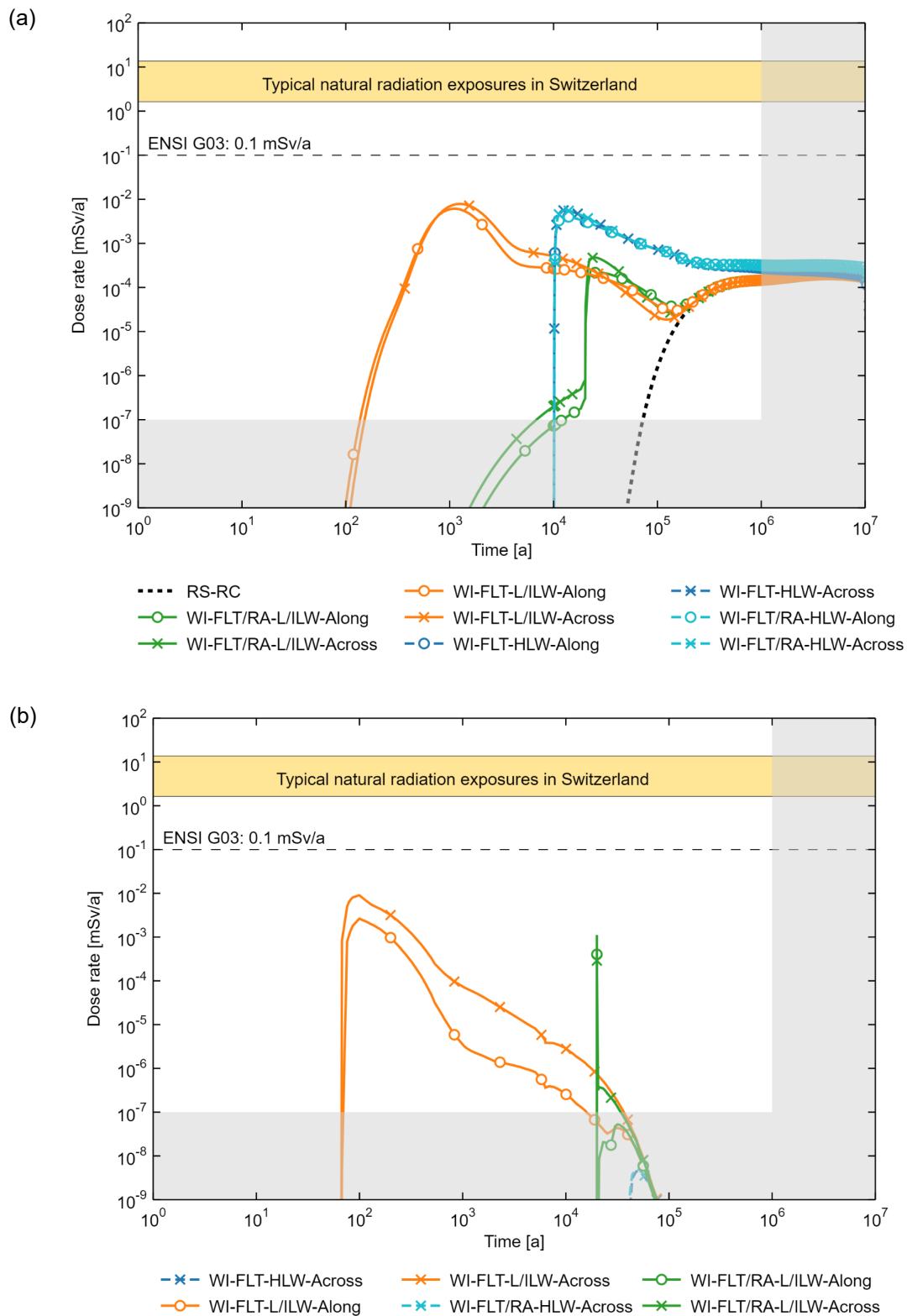


Fig. 8-9: Dose rates for the calculation cases of the “what-if?” cases related to physical phenomena and radionuclide transport in the aqueous phase (a) and for the two-phase transport of ^{14}C (b)

Excavation of the repository by erosive processes

An additional “what-if?” case related to physical phenomena concerns the excavation of the repository by erosion. During the time period for assessment, the occurrence of such an event has been demonstrated to be hypothetical (Nagra 2024i), thus positioning it as a “what-if?” case linked to the physical phenomenon of erosion. Note that, in the very distant future, beyond the one-million-year time period for assessment, there is a slow increase in probability with advancing time for repository material to be excavated as a result of erosion processes (NAB 24-08 Rev. 1, Nagra 2024q).

The variants and calculation cases for this “what-if?” case are shown in Tab. 8-6 and Tab. 8-7. The derivation of the variants and calculation cases, and also the radiological consequences of excavation of repository material by erosion, are fully discussed and analysed in NAB 24-08 Ref. 1 (Nagra 2024q). Uncertainties in key parameters, such as the assumed erosion rates, as well as the nature of the erosion (glacial and non-glacial), are taken into account and lead to the consideration of excavation occurring between 700,000 and four million years after repository closure.

Tab. 8-6: Calculation cases for the assessment of repository excavation by deep glacial erosion
All calculation cases are calculated for excavation occurring 700,000 and 1, 2, and 4 million years after repository closure.

Variant	Calculation case	Description
Ra	River base case	River variant with excavated material initially deposited in the shallow permeable sediments, groundwater connected to the river, water for drinking, animals and irrigation drawn from the shallow aquifer.
Rb	Alternative river case	River variant with excavated material initially deposited in the impermeable layer below the shallow permeable sediments, groundwater connected to the river, water for drinking, animals and irrigation drawn from the shallow aquifer.
L	Lake case	Lake variant with excavated material initially deposited in the impermeable sediments below the lake, water for drinking, animals and irrigation drawn from the lake.
S	Spring case	Spring variant with excavated material initially deposited in the permeable sediments on the hillside, water for drinking, animals and irrigation drawn from the spring.

Tab. 8-7: Calculation cases for the assessment of repository excavation by non-glacial erosion
All calculation cases are calculated for excavation occurring 700,000 and 1, 2, and 4 million years after repository closure.

Variant	Calculation case	Description
Hillside	Case A	Shallow valley with 50 m high hillslope, low erosion rate of 0.08 mm a^{-1} , considering long-term exposures and discrete landslide events.
	Case B	Average valley with 250 m high hillslope, average erosion rate of 0.2 mm a^{-1} , considering long-term exposures and discrete landslide events.
	Case C	Steep valley with 500 m high hillslope, high erosion rate of 25 mm a^{-1} , considering long-term exposures and discrete landslide events.
	Case D	Average valley with high-permeability limestone layer creating springs on the hillside below the repository.
Valley floor		Release from weathered repository below the local aquifer in a valley floor setting.

To demonstrate regulatory compliance for times within the time period for assessment (see Section 2.4), the radiological consequences of repository excavation prior to one million years after repository closure are expressed in terms of annual risk and compared with the risk protection criterion of 10^{-5} per year (Fig. 8-10), because dose rates exceed 0.1 mS/year for some calculation cases (Sections 3.7 and 4.7 of NAB 24-08 Rev. 1, Nagra 2024q). The annual risk is more than two orders of magnitude below the risk protection criterion and is so low that any impact during this period can effectively be considered irrelevant.

For excavation times beyond the time period for assessment, i.e., beyond one million years, calculated dose rates are shown in Fig. 8-11. The dose rates are below or within the range of natural radiation exposure in Switzerland, and therefore also compliant with the regulatory protection criterion for such long timescales (see Section 2.4).

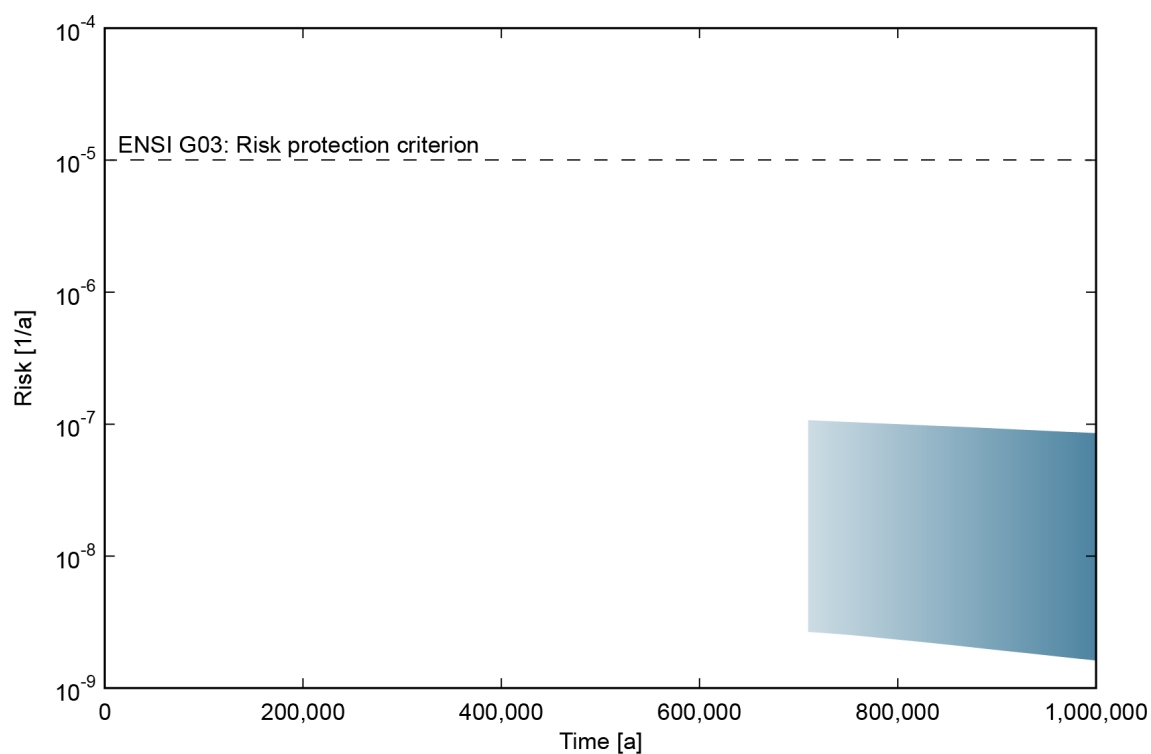


Fig. 8-10: Illustration of the range of the annual risk associated with repository excavation compared with the regulatory risk protection criterion, for excavation times of up to one million years

Based on the calculated dose rates for repository excavation after 700,000 and 1 million years from NAB 24-08 Rev. 1 (Nagra 2024q), conservatively using the probability of excavation at 1 million years (see Figure 6-44 of NTB 24-17, Nagra 2024i).

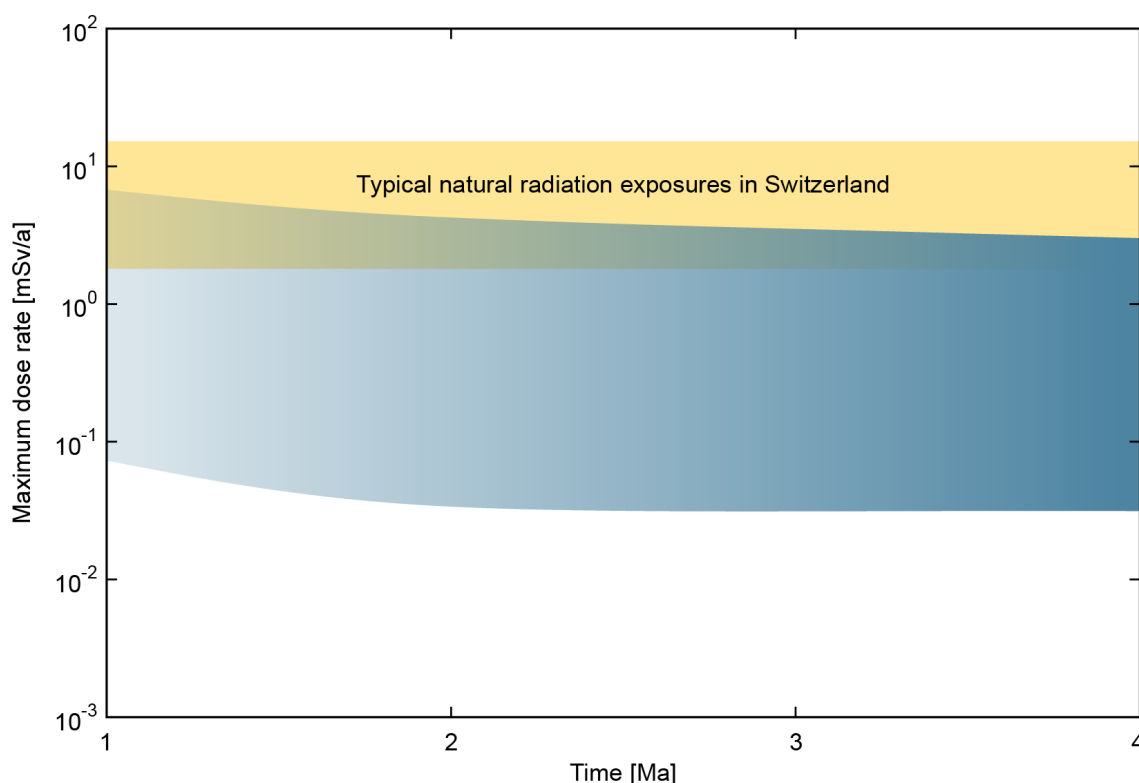


Fig. 8-11: Illustration of the range of the maximum dose rates for excavation times after the time period for assessment (1 to 4 million years)

Based on the range of maximum calculated dose rates for all the calculation cases for the assessment of repository excavation by erosive processes after 1, 2, and 4 million years. The shading indicates an increase of probabilities of the excavation at later times.

8.4.2 “What-if?” cases with hypothetically degraded performance of pillars of safety

Further “what-if?” cases are defined by systematically going through the pillars of safety and either removing them or assuming a degradation of the safety function they serve. The calculation cases as identified in the scenario development are summarised in Tab. 8-8 (see Section 7.4 for details and illustrations). Tab. 8-8 also indicates which radionuclide transport pathways are considered for which calculation case. On the one hand, calculation cases that impact mainly the flow along the underground structures, i.e., linked to the closure system or the permeability of the bentonite buffer, are only calculated for the two-phase flow and transport since the aqueous phase transport is dominated by diffusion through the CRZ. On the other hand, calculation cases that obviously do not significantly impact two-phase flow and release or the transport of ^{14}C as methane are only computed for the aqueous phase. This concerns hypothetical assumptions of no sorption or solubility limits, hypothetically high dissolution rates for spent fuel or the glass matrix of reprocessed waste, and also hypothetically high diffusion coefficients in the bentonite buffer. Finally, the calculation case with a hypothetically high upwardly directed hydraulic gradient would lead to lower doses for the aqueous phase transport since the dominating transport path for radionuclides is downwards.

Tab. 8-8: “What-if?” cases with hypothetically degraded performance of pillars of safety

Pillar of safety	Case	Transport modelling approach		Description
		Aqueous phase	Two-phase	
CRZ	WI-CRZ-NOSORB	✓	×	No geosphere sorption
	WI-CRZ-EDZ	✓	✓	Hypothetically high radial dimension of EDZ around emplacement rooms
	WI-CRZ-DIFF	✓	✓	Hypothetically high diffusion coefficients in the geosphere (In the modelling for two-phase flow, the diffusion coefficient for ^{14}C is hypothetically increased as well)
	WI-CRZ-GRADdown	✓	✓	Hypothetically high downwardly directed hydraulic gradient
	WI-CRZ-GRADup	×	✓	Hypothetically high upwardly directed hydraulic gradient
	WI-CRZ-NOCONFINING	✓	✓	Upper and lower confining geological units disregarded
HLW matrices	WI-SF-IRF	✓	✓	Hypothetically high instant release fractions (IRFs) for radionuclides in SF
	WI-HLW-DISS	✓	×	Hypothetically high matrix dissolution rates for SF and RP-HLW
	WI-SF-CORR	✓	✓	Hypothetically high corrosion rates of SF cladding and other structural parts
HLW disposal canister	WI-HLW-CAN	✓	✓	Hypothetical early failure of all HLW canisters at 1,000 years
Bentonite buffer (HLW)	WI-HLW-NOSOL	✓	×	No near-field solubility limits in HLW repository section
	WI-HLW-NOSORB	✓	×	No near-field sorption in HLW repository section
	WI-HLW-DIFF	✓	×	Hypothetically high diffusion coefficients in bentonite buffer
Bentonite buffer (HLW)	WI-HLW-BUF	×	✓	Hypothetically high permeability of bentonite buffer in HLW drifts
L/ILW cementitious near field	WI-L/ILW-C14	✓	✓	All ^{14}C present in L/ILW released instantaneously
	WI-L/ILW-NOSORB	✓	×	No near-field sorption in L/ILW repository section
Closure system	WI-V1-HLW	×	✓	Hypothetically high permeability of V1-HLW seals
	WI-V1-L/ILW	×	✓	Hypothetically high permeability of V1-L/ILW seals
	WI-V2-HLW	×	✓	Hypothetically high permeability of V2-HLW seals
	WI-V2-L/ILW	×	✓	Hypothetically high permeability of V2-L/ILW seals
	WI-V3	×	✓	Hypothetically high permeability of V3 seals

The results in the form of dose rates for the “what-if?” cases with hypothetically degraded pillars of safety are given in Fig. 8-12 for both (a) the aqueous phase and (b) two-phase transport. For the aqueous phase results, the reference case is also shown for comparison (for two-phase transport, the dose rate for the reference case is negligible).

The aqueous phase results in Fig. 8-12a show one calculation case with more than one order of magnitude higher maximum dose rates and considerably earlier releases than for the reference case; this is the calculation case of hypothetically high diffusion coefficients (WI-CRZ-DIFF).

Three calculation cases with hypothetically degraded performance of the CRZ lead to higher dose rates by up to about one order of magnitude and earlier releases than the reference case; this illustrates the importance of the geological barrier. These three calculation cases are (i) no geosphere sorption (WI-CRZ-NOSORB), (ii) the upper and lower confining geological units are disregarded (WI-CRZ-NOCONFINING) and (iii) a hypothetically high downwardly directed hydraulic gradient (WI-CRZ-GRADdown).

Two additional calculation cases also show approximately one order of magnitude higher dose rates compared with the reference case: (i) hypothetically high instant release fractions (IRFs) for radionuclides in SF (WI-SF-IRF) and (ii) hypothetically high matrix dissolution rates for SF and RP-HLW (WI-HLW-DISS).

All the other calculation cases for the aqueous phase have no impact on the dose rates.

For the two-phase flow results (Fig. 8-12b) only the dose rates from two calculation cases are visible; one with a hypothetically high diffusion coefficient for ^{14}C (WI-CRZ-DIFF) and one with a hypothetically high permeability of the V3 seals (WI-V3). Only WI-V3 gives rise to dose rates above 10^{-7} mSv/a.

Overall, the results of all the “what-if?” cases are well below the dose protection criterion, again confirming the robustness of the repository system.

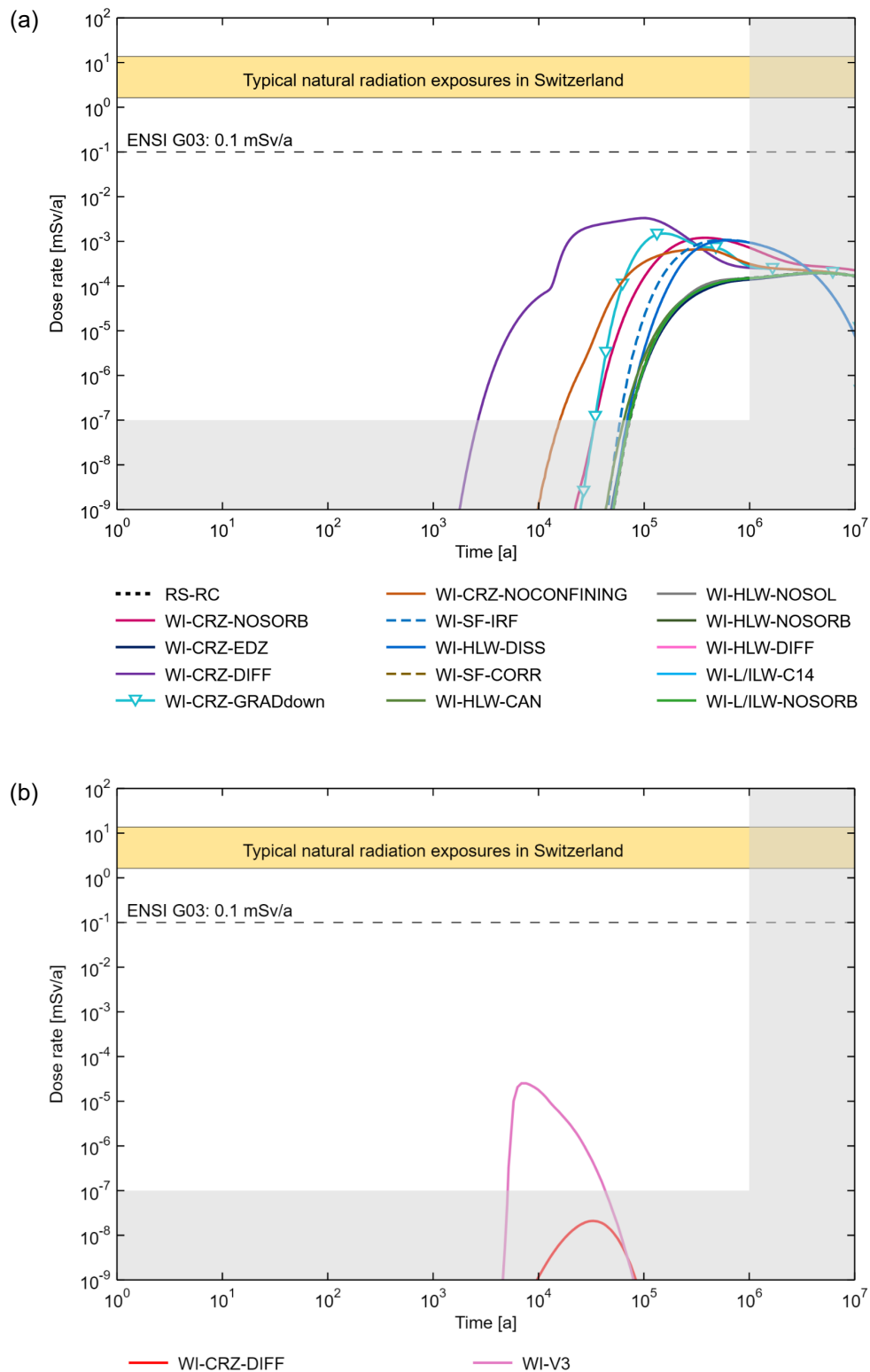


Fig. 8-12: Dose rates for the calculation cases for the “what-if?” cases with hypothetically degraded performance of pillars of safety and radionuclide transport in the aqueous phase (a) and for the two-phase transport of ^{14}C (b)

8.5 Results for future human actions safety scenarios

The following is a synthesis of the detailed results presented in NAB 24-09 (Nagra 2024r), where the derivation and the analysis of FHA safety scenarios are described in detail.

Two FHA safety scenarios have been identified for quantitative analysis (see Section 7.5):

- direct hit of the disposal canisters / containers by a deep borehole (FHA-1),
- abandonment of the repository (FHA-2).

For FHA-1, two variants with two or three calculation cases are propagated to radiological consequence analysis, while for FHA-2 only two calculation cases are propagated. A summary is provided in Tab. 8-9 (for details, see Section 7.5). Note that, within a calculation case, different parametrisations were used to explore sensitivities and the impact of remaining uncertainties (see Chapter 3 of NAB 24-09, Nagra 2024r for details).

Tab. 8-9: Calculation cases for the FHA safety scenarios

Within a calculation case, several parametrisations may be used to explore sensitivities and the impact of remaining uncertainties.

FHA safety scenario	FHA #	Waste	Calculation cases	Specifics	Transport modelling approach
Deep borehole	FHA-1a	HLW	FHA-1a-DH-Diss	SF only	Aqueous
			FHA-1a -DH-Vol	SF only	Two-phase flow
	FHA-1b	L/ILW	FHA-1b-WP-Diss, FHA-1b-PFP-Diss	L/ILW	Aqueous
			FHA-1b-PFP-Vol	L/ILW	Two-phase flow
Abandonment	FHA-2	HLW & L/ILW	FHA-2-Diss, FHA-2-Vol	SF and L/ILW	Aqueous, two-phase flow

The results of the FHA dose calculations are summarised in Fig. 8-13, which provides an overview of the maximum dose rates for the different calculations performed.

The spread of the results highlights the dependency of the radiological consequences on the details of a future borehole placement and penetration. In particular, the release of the rather short-lived ¹⁴C via a gaseous phase and the conservative assumptions of instantaneous and non-limited dissolution of the L/ILW inventory for the aqueous phase calculations lead to strong dependencies on assumptions such as, e.g., the penetration time or the flow regime along the borehole.

Nevertheless, and despite the conservative assumptions, the maximum dose rates for all the FHA calculation cases are considerably below the regulatory limit for FHA (i.e., 1 mSv/a).

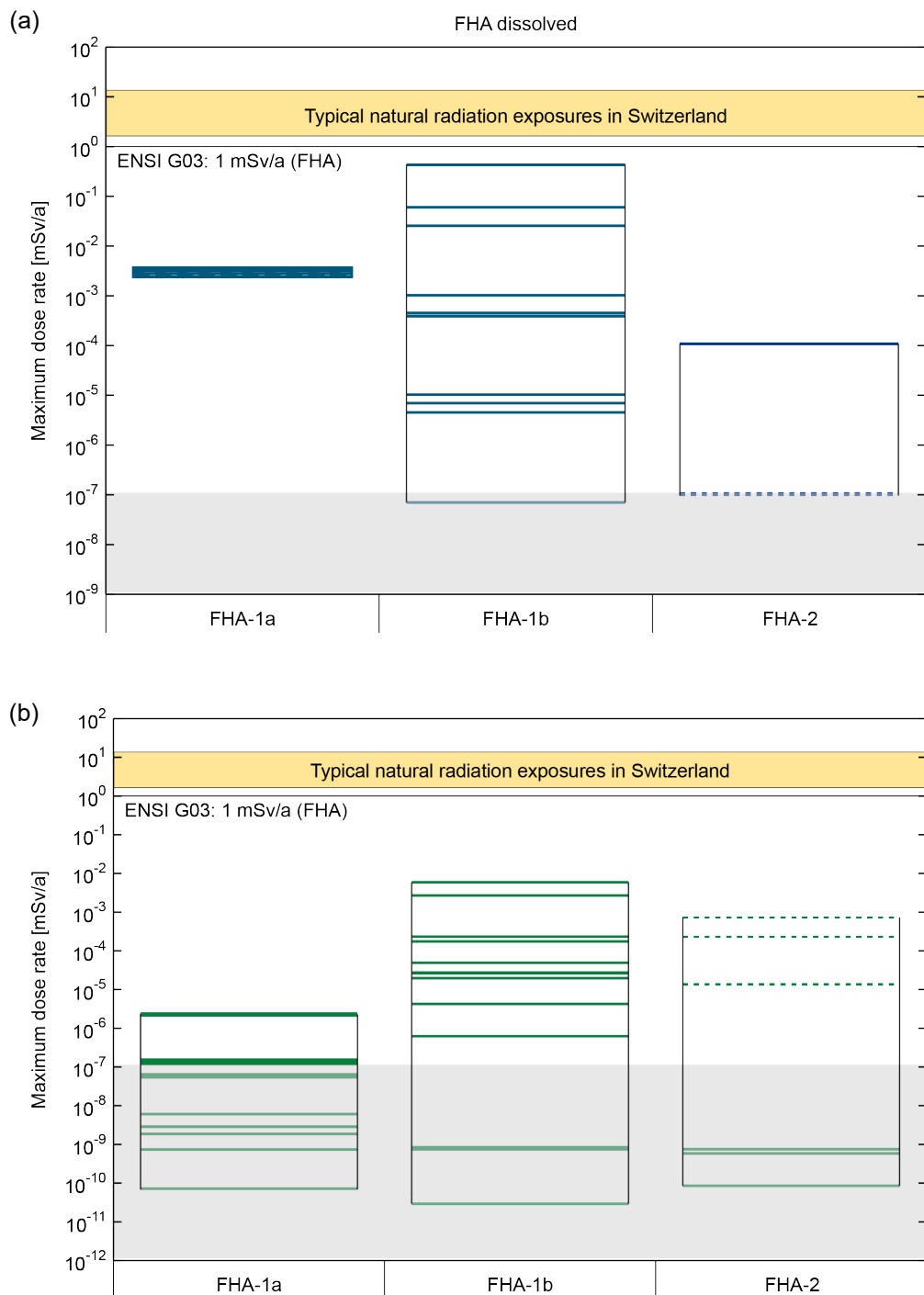


Fig. 8-13: Summary figure showing the maximum dose rates over 10^6 years for the future human action calculation cases for the aqueous phase transport of all radionuclides (a) and for two-phase transport of ^{14}C (b)

The spread of the results is due to the different calculation cases and, within the calculation cases, different parametrisations, whereby for the FHA-2 safety scenario permeabilities are varied for the L/ILW (dashed) and HLW (solid) access tunnels. Part of the spread of the aqueous phase results for the L/ILW direct hit scenario (FHA-1b) is moreover due to different waste packages considered in different calculations. See NAB 24-09 (Nagra 2024r) for details. Oxidising conditions are assumed for these calculation cases.

8.6 Summary of results and safety margins

The systematically deduced set of calculation cases considered in the radiological consequence analysis supports the argument of quality and comprehensiveness of the safety assessment (see Section 10.3). A summary is provided below.

8.6.1 Quality, comprehensiveness, and treatment of uncertainty

The starting point for the radiological consequence analysis is the set of calculation cases derived from the preceding steps of the safety assessment methodology, specifically performance assessment and safety scenario development.

The reference safety scenario corresponds to the expected initial conditions and evolution of the repository system. Uncertainty in geological characterisation leads to alternative safety scenarios or conceptualisations and to most of the calculation cases that relate to physical phenomena (Section 7.3) such as the presence of an undetected fault, or the presence of water-conducting sedimentary features within the confining geological units.

In the reference safety scenario, the repository system performs its safety functions as designed, even considering wide parameters ranges. The reference conceptualisation of the CRZ conservatively assumes that the Keuper aquifer is hydraulically active and forms a significant advective transport path below the repository. An alternative conceptualisation considers an inactive Keuper aquifer below the HLW repository section (see Section 7.2.2). Deterministic calculation cases for the reference safety scenario cover the reference case with best estimate parameters²² and a biosphere according to present-day climate, three different biospheres that account for alternative climatic conditions and geomorphology, and one case where all the sensitive parameters²³ are set to pessimistic bounding values. These cases are complemented with a probabilistic calculation case (Section 8.2).

In the two alternative safety scenarios, a hydraulically active geological unit within the CRZ and an undetected fault in the vicinity of the repository, are considered (Section 7.3). Conservative assumptions about the properties (e.g. parametrisation, geometry) of these geological units and features lead to upper bounds of potential radiological consequences. Calculation cases for the alternative safety scenarios consider different hydraulically active units within the CRZ and different orientations and locations of the undetected faults (Section 8.3).

To test the repository system robustness, several “what-if?” cases are postulated and analysed. The first category of “what-if?” cases relate to physical phenomena but that are shown to be hypothetical, i.e., de-facto impossible (Section 7.4.1). This category contains cases postulating either undetected or newly generated or re-activated faults with unrealistically high transmissivity. They are treated very similarly to the cases of the alternative safety scenario of an undetected fault, but with hypothetical parametrisations. The first category further contains cases analysing the excavation of the repository by erosive processes. Even though excavation of the repository by erosive processes is deemed to be hypothetical within the time period for assessment, the analysis is required by the regulatory guideline ENSI 33/649 (ENSI 2018). The second category of “what-if?” cases postulates the hypothetical degradation of the pillars of safety (Sections 7.4.2 and 8.4.2).

²² Best-estimate values for parameters are used whenever there is enough knowledge or arguments to do so (e.g., as for diffusion coefficients of the Opalinus Clay). If this is not the case, conservative values are assumed (e.g., no complete containment for L/ILW). Also, care is taken to avoid potential risk dilution, which introduces further conservatism (e.g., by assuming instantaneous failure of HLW disposal canisters after 10,000 years).

²³ Sensitive parameters are one result of the uncertainty and sensitivity analyses with the probabilistic calculation case.

Finally, calculation cases considering future human action safety scenarios (Sections 7.5, 8.5) complement the analysis, as required by regulation (Section 2.4).

An important conservatism throughout the consequence analysis is that radionuclides that leave the modelling domain of the geosphere (via deep aquifers) are instantaneously transferred to the regional aquifer of the biosphere. Thus, additional delay with decay and further dispersion is neglected.

8.6.2 Conformity with protection criteria and robustness

The results of the radiological consequence analysis demonstrate compliance with the regulatory protection criteria and, through safety margins and “what-if?” case results, underline the robustness of the repository system. The main results are summarised below and feed into the lines of argument for favourable results of the safety assessment (Section 10.2).

For any future evolution of the deep geological repository (except cases involving inadvertent human intrusion), either the dose protection criterion (0.1 mSv/a) or the risk protection criterion (10^{-5} /a) must be met during the time period for assessment (Section 2.4). After the end of the time period for assessment, the effects must not be significantly higher than the average current radiation exposure of the Swiss population.

Compliance with the dose protection criterion is illustrated in Fig. 8-14, where maximum dose rates over the simulation time of 10^6 years have been extracted for all the calculation cases except for the “what-if?” case of excavation and FHA cases²⁴.

The results that link to the reference safety scenario are well below the regulatory criterion of 0.1 mSv/a. The probabilistic results show a safety margin of more than two orders of magnitude for the 95 percentile and close to three orders of magnitude for the median. The most pessimistic deterministic case has a maximum dose rate occurring after roughly one million years. It is still about two orders of magnitude below the regulatory protection criterion. Its high maximum dose rate, which is not covered by the probabilistic results²⁵, is due to the cumulation of pessimistic assumptions, in particular pessimistic values of all the sensitive parameters that lead to the highest dose rates. Furthermore, the wide spread of the results (particularly the deterministic ones over 5 and 11 orders of magnitude for the aqueous phase and two-phase ^{14}C results, respectively) is mainly due to varying but strongly simplifying assumptions regarding the geosphere/biosphere interface, the biosphere itself, and the strong influence of the rather short half-life of ^{14}C for the two-phase results. This is further illustrated in Fig. 8-15, where dose rates over time are shown for all the calculation cases and thus the timing of the dose rate is visible.

For the alternative safety scenarios, very large safety margins also remain. Since the alternative safety scenarios are performed with the reference biosphere only, the spread of the results is narrower, and there is almost no impact on the maximum dose rates for the aqueous phase results when compared with the reference case. In conclusion, for the expected and any potential evolution of the repository system, the absence of any harm to humans and the natural environment is demonstrated with high confidence and an ample safety margin.

Furthermore, the results for the hypothetical “what-if?” cases shown in Fig. 8-14 are also below the protection criterion. That the maximum dose rates are generally higher than for the reference and alternative safety scenario cases is expected, since assumptions and parameters are varied in a pessimistic way to test system robustness. The highest maximum dose rates are those of the case

²⁴ For the “what-if?” case of excavation, conformity with the risk protection criterion is demonstrated, while for FHAs conformity with SSR-5 (IAEA 2011a) is shown.

²⁵ In general, the deterministic results have a wider spread, because combinations of extreme values are deterministically selected; it is thus highly unlikely that such combinations would be randomly sampled.

with an undetected fault with unrealistically high transmissivity that intersects the L/ILW section of the repository (see, e.g., also Fig. 8-9). The impact of cumulative conservative assumptions (e.g., fast radionuclide release and dissolution and the neglect of partial saturation for the aqueous phase calculations; full ^{14}C inventory as methane for the two-phase calculations) is at the source of these results that are thus significant overestimations.

It is worth noting that the “what-if?” cases that postulate a hypothetical failure of individual pillars of safety (see also Fig. 8-12) do not lead to dramatic increases in maximum dose rates, thus building a strong argument for the repository system robustness.

The results for the “what-if?” case of repository excavation by erosive processes (Fig. 8-10 and Fig. 8-11 in Section 8.4.1) are not reproduced here. Note, however, that these conform with the risk protection criterion with an ample safety margin during the time period for assessment, while afterwards, they lie well within the typical natural radiation exposure in Switzerland.

Finally, calculation cases for future human action safety scenarios also give results well below the relevant protection criterion, as has been shown in Section 8.5.

Note that the results of consequence analysis are, despite some novel approaches and more extensive calculations, highly consistent with analyses of previous stages in the disposal programme, namely the demonstration of disposal feasibility (Project *Entsorgungsnachweis*, Nagra 2002). Thus, confidence in the Opalinus Clay as a suitable host rock for radioactive waste disposal is confirmed.

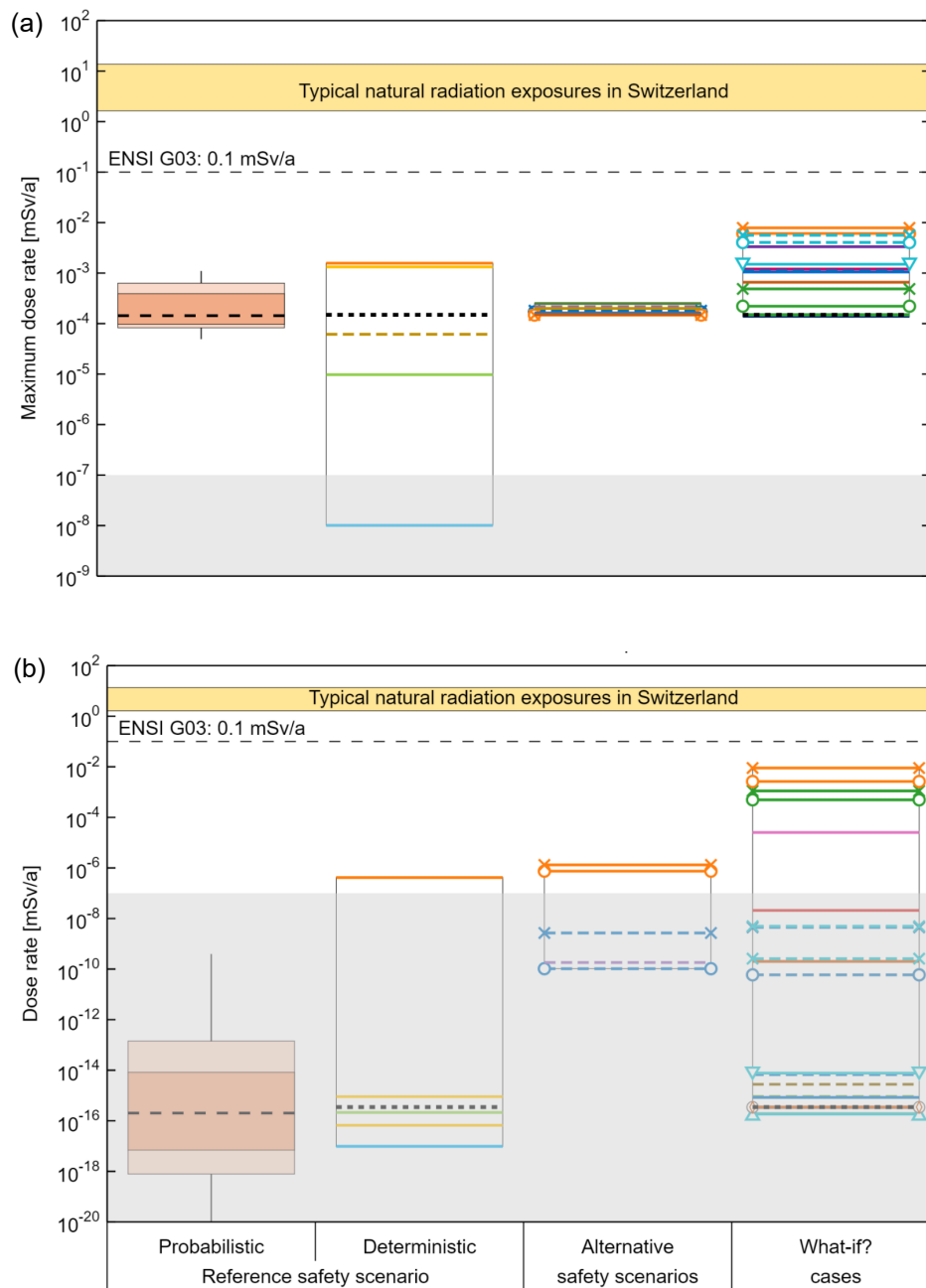


Fig. 8-14: Summary showing the maximum dose rates over 10^6 years for the aqueous phase transport of all radionuclides (a) and for two-phase transport of ^{14}C (b)

The reference safety scenario (probabilistic and deterministic calculation cases), calculation cases for the alternative safety scenarios and “what-if?” cases (except the “what-if?” case of excavation) are shown.

The boxplots show the statistical measures of median (dashed horizontal line), 95/5 and 84/16 percentiles (light and dark shades of red) and the full extent of the spread between minimum and maximum (thin vertical line); individual deterministic calculation cases are shown with coloured solid or dashed lines and markers which correspond to the calculation cases in Sections 7.1.5 and 7.2.5 of NTB 24-18 (Nagra 2024p).

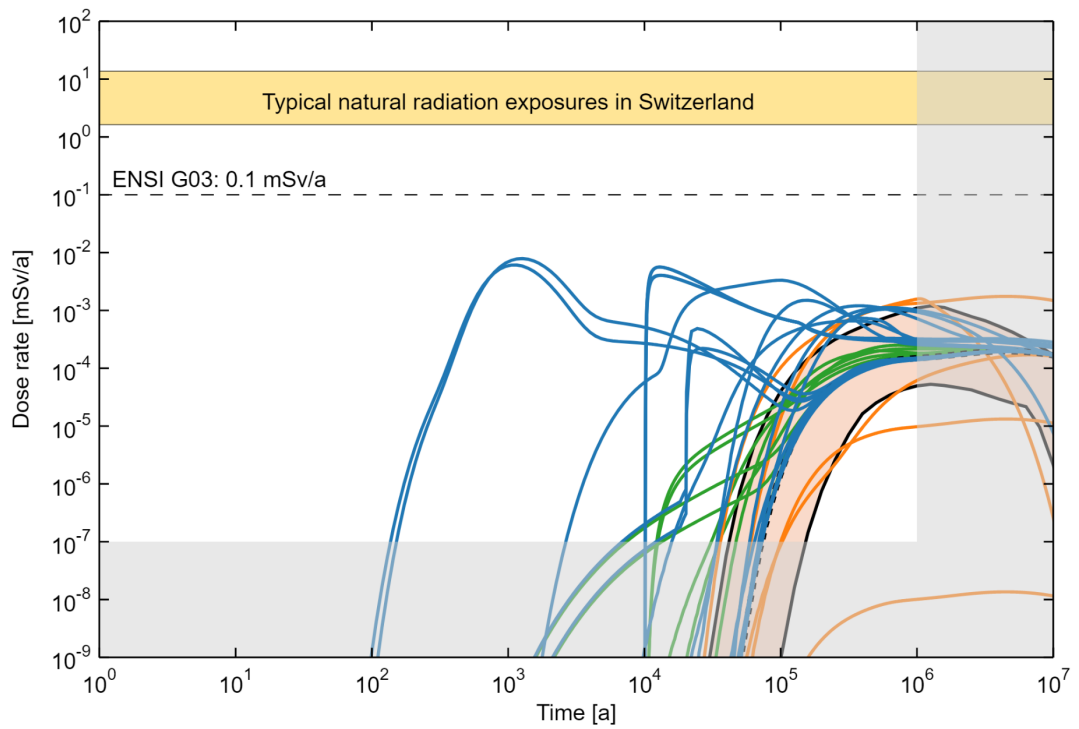


Fig. 8-15: Summary showing dose rates for all the calculation cases for which the dose protection criterion is relevant

The reference case in dashed black, the reference scenario in orange, the alternative safety scenarios in green and the “what-if?” cases in blue lines. The orange surface illustrates the minimum and the maximum of the probabilistic simulations.

9 Complementary lines of argument

Aims of the chapter:

- To describe natural and archaeological analogues that enhance confidence in phenomenological process understanding.
- To present complementary safety indicators to provide additional insights into the evolution of the hazard potential of the waste and its radiotoxicity.
- To show that numerous conservative assumptions and simplifications are incorporated in the models used to estimate the dose.

The safety margins, in terms of dose, calculated in the analysis of radiological consequences, are a key line of argument for the safety of the repository system. Other complementary lines of argument are presented in this chapter.

These consist of:

- Natural and archaeological analogues

Section 9.1 discusses studies of natural and archaeological analogues of features present, or processes occurring, within the repository system. These enhance confidence that the engineered and geological barriers are physically and chemically stable, at least to the extent assumed in safety assessment, and that all relevant processes are considered in the safety assessment models. Natural and archaeological analogues can be identified that are relevant to the geological barrier, to the engineered barriers and to key interfaces between the barriers.

- Complementary safety and performance indicators

Section 9.2 discusses how safety and performance indicators complementary to dose and risk can be defined that provide additional insights into the evolution of the hazard potential of the waste, how radiotoxicity (a measure of this hazard potential) is distributed within the repository barrier system and how it diminishes over time due to radioactive decay, and how the flux and concentration of radiotoxicity compare with natural, as well as man-made, radiotoxicity fluxes and concentrations.

- Conservatism of the calculated radiotoxicity and radiological consequences

The models used to calculate doses in the analysis of radiological consequences, as well as to evaluate radiotoxicity fluxes and concentrations, incorporate numerous conservative assumptions and simplifications, as summarised in Section 9.3, to handle uncertainties that are not addressed by other means (e.g., through parameter variations). The existence of these conservatisms means that doses, radiotoxicity fluxes and concentrations are generally overestimated and thus constitute another complementary line of safety-related arguments.

9.1 Natural and archaeological analogues

Incorporating natural and archaeological analogues as additional lines of argument within a safety assessment aligns with IAEA's guidelines for post-closure safety assessments for radioactive waste repositories. Following, e.g., Côme & Chapman (1986) and IAEA (1999), these analogues can be defined as natural or man-made environments and/or materials that share common features or similarities with, and thus can provide useful information on, the repository system and its evolution over time. They allow processes and their effects to be observed on large temporal and spatial scales, similar to, or even exceeding, those relevant to safety assessment, providing insights into the durability of system components and into the effective isolation of radioactive materials in a repository. They can showcase the resilience of chosen barrier materials to potentially detrimental repository-induced effects and to long-term degradation processes.

Analogues can usefully complement site-specific information, especially at the early stages of a programme where such information is limited. The three-stage Sectoral Plan for Deep Geological Repositories (SGT) in Switzerland has, since 2008 (BFE 2008), led to extensive exploration and studies of potentially suitable geological environments, yielding large amounts of geological and hydrological data, as well as experimental results elucidating the behaviour of relevant materials, including radionuclides, in such environments. Nonetheless, while this reduces reliance on natural and archaeological analogues to support the durability and performance of the repository barriers, information from analogues remains integral to the post-closure safety case and is summarised in the following sections.

9.1.1 The geological barrier

In most deposits and landfills both above and below ground, clay and clay barriers are used to prevent or mitigate waste seepage and contamination. It is therefore unsurprising that clays, such as bentonite and clay rocks, are considered worldwide as host rocks and/or engineered barriers for geological repositories. Clay rocks, known for their low hydraulic conductivity and high radionuclide retention capacity, are, for example, being considered as potential host rocks, not only in Switzerland, but also in Belgium, Canada and France. As a specific example, in the proposed concept for a deep geological repository at Bure, France, the 130-metre-thick clay-rich unit of the Callovian-Oxfordian argillite formation, serves as a substantial natural barrier against radionuclide release to the surface (Andra 2022).

Clay rocks such as the Opalinus Clay have proven to provide robust transport barriers, or seals, in many different environments. Numerous examples exist in the oil and gas industry where clay-based seals have demonstrably and comprehensively been used to prevent oil and gas transport over timescales of thousands to millions of years. These seals not only confine fluids beneath them, but also within them, as evident from the many clay formations that produce oil and gas around the globe. Examples are the Duvernay and Canol Formations in Canada (Biddle et al. 2021, Lee et al. 2019), the Eagle Ford Shale in the USA (Speight 2020), and the Vaca Muerta Formation in Argentina (Casala et al. 2019). These clay formations are often overpressured due to in-situ oil and gas generation during burial by in-situ thermal maturation of the original organic matter, consistent with their low permeability. The effectiveness of clay formations as long-term seals has been demonstrated for various deep (often 1,000 m) oil and gas reservoirs, including cap rocks for CO₂ disposal sites, even in the presence of significant tectonic faults. This suggests that clay formations, even when faulted, do not serve as major long-term conduits for flow due to their ductility and inherent ability to self-seal (Zhu et al. 2013). A comprehensive discussion and analysis of selected aspects of clay-based barrier performance in petroleum systems can be found in the work of Fisher et al. (2023).

Because of these properties, clay formations are frequently regarded as ideal cap rocks for shallow storage or for oil and gas reservoirs. Extensive exploration in oil-sand regions such as Venezuela and Canada have focused on understanding the thermo-hydro-mechanical behaviour of clay formations. Interbedded clay-rich units within sandy reservoirs can also act as discrete barriers to transport, influencing recovery performance due to their low permeabilities (Kumar & Hassanzadeh 2021, Gao et al. 2022).

Evidence from around the globe of slow diffusion-dominated transport in clay formations is complemented by a large body of isotope exchange (Aschwanden et al. 2024) and diffusion studies (Van Loon et al. 2023, Glaus et al. 2024b) from Northern Switzerland, particularly on Opalinus Clay (see Section 4.6 of NTB 24-17, Nagra 2024i). The studies reveal prolonged pore-water residence times and slow diffusion-dominated transport of solutes in the Opalinus Clay, where the repository will be located, in the overlying and underlying confining geological units and towards the bounding aquifers. Consistent aqueous concentration profiles of natural tracer profiles (Fig. 9-1) suggest low hydraulic conductivity and minimal hydraulic gradients. Moreover, tracer profiles are not noticeably affected by the contrasting properties of the limestone-rich beds of the «Herrenwis Unit» of the upper confining geological units, which represent a regional reef mount (Nagra 2024i). The clays also exhibit a high sorption capacity (Marques Fernandes et al. 2024b).

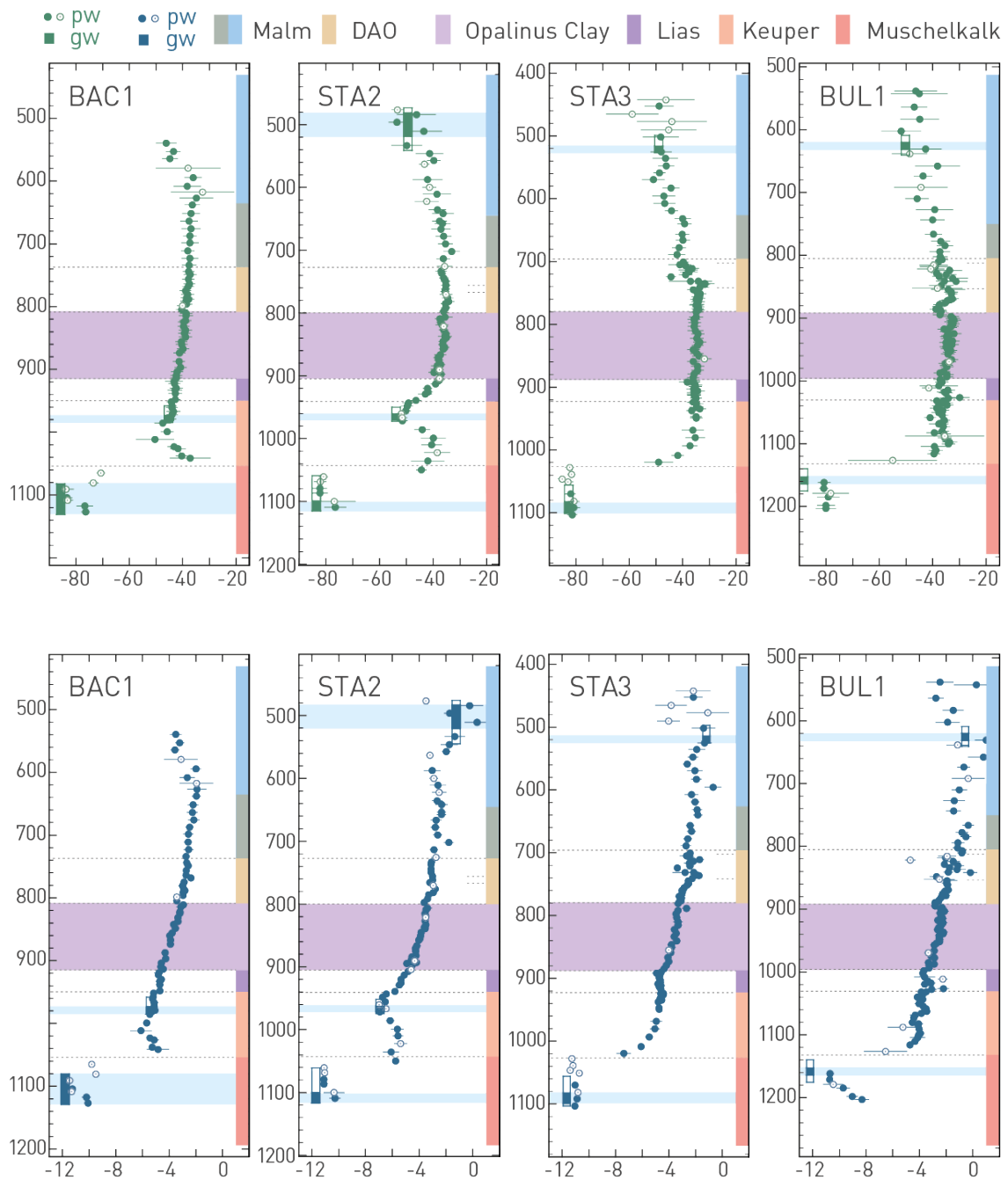


Fig. 9-1: Profiles of $\delta^2\text{H}$ [‰ VSMOW] (top row) and $\delta^{18}\text{O}$ [‰ VSMOW] (bottom row) in porewater (pw) and groundwater (gw)

In general, the porewater profile within the Opalinus Clay (purple band in the background) down to the lower part of the Keuper is linear, whereas the profile in STA2 is characterised by a clearly curved profile or profile parts especially towards the top of the Keuper. The steep drop towards the aquifers show that the original porewater within the clay rock experienced little exchange with modern meteoric water (simplified from Figure 4-119 of NTB 24-17, Nagra 2024i). Porewater data were derived from diffusive isotope exchange experiments; open symbols: less reliable (potential artefacts or contamination with drilling fluid). Where groundwater could be sampled, open rectangles show packer intervals, closed rectangles and light blue bands in the background most likely inflow zones. DAO: Dogger Group above Opalinus Clay. Light blue background indicates the location of aquifers.

9.1.2 Engineered barriers

Spent fuel

Several natural analogues for SF have been extensively studied since the late 1960s to obtain more information and data on the resistance of such material to oxidation and on degradation under natural, repository-relevant conditions. The best SF analogues are sedimentary uranium ore deposits formed under reducing conditions, as indicated in Chapter 6 of NAB 23-10 (Johnson et al. 2023). Examples include a range of uranium ore bodies such as El Berrocal in Spain, Poços de Caldas in Brazil, Oklo in Gabon, Koongarra in Australia, Cigar Lake in Canada and the Witwatersrand Au-U placer deposit in South Africa (Armstrong 1981). Researchers have explored these analogues to investigate in detail various different processes influencing the long-term behaviour of spent fuel (Smellie & Karlsson 1999, Smellie et al. 1997, West et al. 1992). According to Milodowski et al. (2015), naturally occurring uranium minerals such as uranite and, to a lesser extent, pitchblende, serve as excellent analogues for SF.

In the study of the Cigar Lake ore body in Canada, Smellie & Karlsson (1996) addressed important questions regarding radiolysis, fuel dissolution, and redox fronts (Section 6.4 in NAB 23-10, Johnson et al. 2023). They found that the ore body had remained essentially unchanged since its formation over 1,300 million years ago, thus supporting the long-term stability of SF. The most notable result is the evidence of UO_2 resistance against radiolytic oxidation. In particular, the Cigar Lake uranite ore maintained its predominately reduced state despite undergoing significant alpha radiolysis and experiencing prolonged hydrothermal activity spanning millions of years. Bruno & Spahiu (2014) concluded, based on a subsequent reanalysis of the original Cigar Lake data, that “*the long-term stability of the UO_2 matrix is warranted by the recombination of the radiolytically generated oxidants and reductants*”.

In the Oklo natural reactor in Gabon (Francewill Basin), uranium ore is mainly composed of UO_2 in direct contact with organic matter, filling the porosity of the sandstone (Section 6.5 in NAB 23-10, Johnson et al. 2023). Isotopic signatures indicated that criticality has occurred and was reached in regions of highest uranium enrichment and porosity. Within the reactor zones (which represent the fuel analogues) UO_2 remains the dominating phase (Gauthier-Lafaye et al. 1996). The environmental conditions remained reducing throughout the history of the reactor, due to the large amounts of associated hydrocarbons. In the analogy with the high-level waste near field, the organic matter takes the “chemical role” of the metallic canister. Despite its complex tectonic and hydrothermal history, UO_2 has been mostly preserved over 2,000 million years. This serves as evidence that extensive radiolytic oxidation of UO_2 , resulting in massive UO_2 dissolution and reprecipitation as U(VI) , did not occur (Johnson et al. 2023). The identification of H_2 -rich fluid inclusions (along with O-rich fluid inclusions, Dubessy et al. 1988) could be interpreted as the product of water radiolysis. The coexistence of these fluid inclusions with the well-preserved uraninite suggests that the radiolytically produced H_2 was chemically activated, providing protection against the oxidation of UO_2 .

The study of Johnson et al. (2023) concluded in Section 7.2 that “*The low fractional dissolution rates recommended to be used in safety assessment are based on multiple arguments, including natural analogue studies showing that uraninite remains stable over geologic time frames.*”

Hence, natural analogues complement the data derived from laboratory experiments, offering insights into spent fuel dissolution over geological timescales. However, it is important to acknowledge the limitations of the analogies, given the inherent differences in environmental conditions between the geological environment of the analogies and that of a deep geological repository (Section 7.2 of NAB 23-10, Johnson et al. 2023).

Reprocessed high-level waste

For over six decades, starting with Marshall (1961), natural glasses, such as volcanic glass and tektites, as well as archaeological glasses, such as uranium-bearing drinking vessels, slag or glass blocks of Roman age, have been the subject of extensive studies as analogues for reprocessed (vitrified) high-level waste. The primary focus has been on determining dissolution processes and rates, along with identifying the characteristics of secondary alteration products (see Miller et al. 2000 for specific examples).

Among these glasses, basaltic glasses exhibit the highest resemblance to contemporary RP-HLW formulations and are generally acknowledged as the most suitable analogues for borosilicate HLW glasses (Havlova et al. 2008). Due to the absence of high B_2O_3 in these glasses and radionuclides in general (Posiva 2023), both natural and archaeological glasses are of limited use as analogues of RP-HLW. Despite this limitation, the absorption of dissolved uranium on the surface of degradation products of natural glass, as observed by MacDougall (1977), suggests that radionuclides released from RP-HLW could potentially be sequestered by analogous secondary degradation phases on the waste package. Additionally, based on the examination of 425 natural glasses, it is suggested that over half of those were 2 million years old (Ewing 1979). Such information provides confidence that rapid degradation of RP-HLW is unlikely to occur during the time period for assessment.

Studies on Roman glass blocks buried in seawater-saturated mud provide additional insights despite the limitations of this chemical analogy (Section 4.4.5.2 in NAB 23-09, Curti 2022). Peripheral regions (rims) exposed to seawater exhibit higher degradation rates compared with internal regions. The long-term behaviour of basaltic glasses is extrapolated from laboratory experiments and natural occurrences, albeit with uncertainties. Degradation rates derived from natural basalts can underestimate true rates due to unrealistic assumptions and temperature uncertainties as summarised in Section 4.4.5.2 in NAB 23-09 (Curti 2022). Even though there are undisputable large uncertainties in estimating nuclear waste glass corrosion rates from natural and archaeological analogues due to varying conditions and compositions, these studies improve qualitative understanding of glass dissolution processes.

HLW disposal canisters

Mechanical and corrosion-related degradation processes will eventually lead to the breaching of the disposal canisters, which will then no longer provide complete containment of radionuclides. Along with copper, carbon steel is one of the few candidate canister materials for which there are natural and/or archaeological analogues that can help support long-term predictions of the corrosion behaviour (King 2021, Neff et al. 2010, Alexander & Reijonen 2023).

Various types of iron analogues, including natural analogues such as native iron (though rare) and archaeological analogues such as iron artefacts, have been extensively studied in the literature. Analogue studies can be used to support canister lifetime predictions by the provision of corrosion damage data, typically in the form of either corrosion rates (e.g., Crossland 2005) or pit depths (e.g., JNC 2000), over timescales unavailable to experiments, and by supporting the development of conceptual models and the validation of mechanistically based corrosion models.

Corrosion-related analogue studies often form part of a well-structured canister development programme that combines the results from short-term laboratory experiments, simulations and modelling, large-scale *in-situ* tests, and analogue studies (Dillmann et al. 2014). In general, analogues are used to support predictions of the rate and extent of uniform and localised corrosion rather than environmentally assisted cracking mechanisms, for which there are few, if any, analogues (King et al. 2023, Nagra 2024d). In addition, analogue studies are not widely used to support predictions of the mechanical stability of the canister.

Bentonite buffer

In line with the current safety and repository concept, bentonite is commonly used as a buffer material in repositories for the disposal of radioactive waste (Karnland et al. 2008, Karnland et al. 2007, Smellie 2001), and thus its thermal and long-term stability is critical. Examples of the long-term chemical and mineralogical stability of bentonite include those from Saskatchewan, Canada (75 Ma Avonlea bentonite), Wyoming, USA (95 Ma MX-80 bentonite) (Keto 1999) and Kiruna, Sweden (probably in the order of hundreds of Ma to 1 Ga old bentonite, Alexander et al. 2024), where all periods greatly exceed those required for bentonite in engineered barriers.

Other aspects of bentonite longevity have been addressed in a range of natural analogue studies and quantitative data are available for corrosion product interaction (Alexander & Reijonen 2023), low-alkali cement leachate interaction (Milodowski et al. 2016), illitisation (Leupin et al. 2014), host-rock interaction (Alexander & Reijonen 2023), and canister settling (Alexander et al. 2017). A dedicated study of the thermal alteration of bentonite has been initiated at an identified site, offering valuable data for future post-closure safety cases (Alexander & Reijonen 2023). Despite some shortcomings identified by Laine & Karttunen (2010) in natural analogue studies for bentonite applications in the framework of post-closure safety cases, there is confidence that bentonite can meet safety-related requirements for at least the entire time period for assessment.

Cementitious materials

An understanding of cement durability and degradation processes is needed to assess the evolution of the cementitious near field of the L/ILW repository. Cementitious materials are considered metastable products from a thermodynamic standpoint, implying that they will undergo changes and remineralisations over time due to internal and external cement degradation processes. These processes include aggregate-concrete interaction, alkali-silica reaction, mineralogical alteration by carbonatisation, leaching, and interaction with fluids containing concentrated ions (see Section 2.2 in NTB 23-03, Kosakowski et al. 2023).

Most information on the longevity of cement phases as present in Ordinary Portland cement (OPC) has been gathered during the Maqarin Natural Analogue Project of which Nagra was a partner (see Pitty & Alexander 2011, Martin et al. 2016). The natural clinker analogues at Maqarin, in northern Jordan, were formed by low-pressure and high-thermal heating of the organic-rich biomicrite and chalk host rocks around 2 Ma²⁶ ago, generating calcium silicate and calcium aluminate-ferrite mineral assemblages, which have similarities to clinker phases in modern OPC as summarised in Martin et al. (2016). Subsequent groundwater flow through these naturally generated clinker phases resulted in hyperalkaline fluids percolating through fractures and cracks created by tectonic fracturing of the biomicrite. This led to precipitations of cement minerals such as C-A-S-H, ettringite, thaumasite, and carbonates. The precipitations occurred mostly along the fractures which were infilled with such natural cement phases and underwent secondary alterations (see Milodowski et al. 2015, Martin et al. 2016 for details), which eventually sealed the former open fracture interfaces (see Fig. 9-2). Dating of the system suggests the fracture – biomicrite interfaces sealed within a very short period of only a few years.

²⁶ Even older (up to 58 million years old) natural concretes formed along similar lines have been described at Scawt Hill in Northern Ireland (Milodowski et al. 2015; 1989).

In distal regions along fractures further away from the natural clinker, diffusive exchange over sufficient time resulted in silicate dissolution and the precipitation of zeolites (or zeolite precursors) (Martin et al. 2016). Such secondary reaction phases are comparable to those produced when concrete is leached in the laboratory and in Underground research laboratory (URL) experiments (e.g., Mäder et al. 2006), clearly indicating the relevance of the analogy.



Fig. 9-2: Heavily fractured micrite host rock with abundant sealed fractures containing a comparable product to natural concretes at the Maqarin site in Jordan

For scale, see hammer in bottom left corner (Pitty & Alexander 2011).

9.1.3 Interfaces between barriers

Cement – clay interfaces

As in geological settings, interfaces of materials with contrasting chemical composition will evolve over time due to chemical interactions, giving rise to changes in transport properties at and across the material interfaces. In that sense, interfaces between cement and clay and the chemical interaction are of relevance, and process understanding was greatly enhanced by the CI field experiment in the Mont Terri rock laboratory, Switzerland, where cement – Opalinus clay or bentonite interactions, respectively, were investigated over several years (see Bernard et al. 2020, Yokoyama et al. 2021, Mäder et al. 2017). A detailed review on cement – clay interaction is presented in Chapter 2 of NAB 22-34 (Prasianakis et al. 2022).

Cement – clay interactions and long-term interface development have also indirectly benefited from the studies of the fractured rock complex at Maqarin, in northern Jordan, as a valuable natural analogue (see Maqarin Natural Analogue Site Study Group 1992, Linklater 1998, Alexander & Smellie 1998, Pitty & Alexander 2011 for details). There, materials around open fractures were assumed to have once been part of sealed fracture faces (Linklater 1998). Along these sealed fracture faces, mineral alterations into the limestone host rock (biomicrite) interface as a result of the hyperalkaline plume was observed to have limited penetration extent (mm) as demonstrated by Martin et al. (2016), i.e., the maximum alteration extent amounted to 25 mm. Importantly, generated interfaces were not totally clogged as, e.g., an alteration rim of a vein with microscopically clogged pores was identified to contain a remaining porosity of about 19 vol.-% (Martin et al. 2016).

These observations are well aligned with what Kosakowski et al. (2023) noted in Section 3.3.2 of NTB 23-03: *“Interfaces between cement and clay materials can be characterised by strong (geo)chemical differences, which influence the diffusive transport of solutes into both materials and may result in the formation of alteration fronts. Typically, dissolution of cement and clay phases is observed in combination with precipitation of secondary mineral phases at or near the material interface.”* This statement derived from a synthesis and assessment of various short and long-term laboratory and URL experiments (e.g., Lalan et al. 2016, Gaboreau et al. 2020, Yokoyama et al. 2021), as well as archaeological (e.g., Lee 2001) and natural (e.g., Pitty & Alexander 2011, Martin et al. 2016) analogues.

Iron – clay interfaces

Mineralogical and chemical studies of the interfaces between clay and iron (magnetite) ore have been carried out at the Kiruna bentonite natural analogue in the Kiirunavaara mine (Alexander et al. 2024). The smectite clay is made of significant amounts of a Fe-poor montmorillonite with a generally Ca-rich exchangeable interlayer composition. Preliminary results show that magnetite ore and smectite have been in direct contact at a temperature of between 50 and 150 °C for an extended period of time (ca. 350 Ma). Little or no uptake of Fe in the clay has been observed under these conditions, which are similar to those in a deep geological repository. Thus, the observations suggest minimal to no incorporation of iron into the clay matrix. Analysis of dioctahedral smectites demonstrates only marginal fluctuations in Fe₂O₃ content, ranging from 0.93 to 2.46 wt.-%. Interestingly, these variations do not appear correlated with the presence or absence of residual magnetite in the altered rocks or their proximity to magnetite ores.

9.2 Complementary safety and performance indicators

In addition to the calculation of the annual individual effective dose, the performance and safety of the repository system can further be illustrated using additional, complementary safety and performance indicators.

The indicators described in the following sections are based on the concept of radiotoxicity (RT) as a measure of the relative potential hazard of radioactive material. RT is defined as the effective dose received in the case of hypothetical direct ingestion of radionuclides in dissolved form into the body. RT, in units of sieverts (or millisieverts, where appropriate), is calculated from the activity $A_i(t)$ [Bq] of a nuclide i at a time t and the ingestion dose conversion factor F_i [Sv/Bq] of the nuclide i :

$$RT(t) = \sum_i A_i(t) \times F_i$$

RT fluxes and concentrations can be calculated in an analogous way to RT itself, by replacing the activity $A_i(t)$ in the equation above by activity flux or activity concentration, respectively.

The consideration of these complementary indicators can provide additional arguments for safety that avoid, to some extent, the difficulties faced in evaluating and interpreting doses and risks that may occur in the far future (OECD/NEA 2004, 2012b). A comprehensive review of complementary indicators used in post-closure safety cases, including the reference values for safety indicators, is reported in OECD/NEA (2012a).

In the following sections RT, RT fluxes and RT concentrations are used to illustrate:

1. the declining radiological hazard from the waste packages for SF, RP-HLW and L/ILW over time,
2. the evolving distribution and decay of RT in the repository barrier system, and
3. how the RT fluxes and concentrations associated with the repository system compare with other RT fluxes and concentrations in natural systems and from human activities, in order to place those associated with the repository system in perspective.

9.2.1 Declining radiological hazard from the waste packages

The evolution of the RT of the HLW and L/ILW disposed of in the repository is shown in Fig. 9-3, illustrating the declining radiological hazards from the waste packages with time.

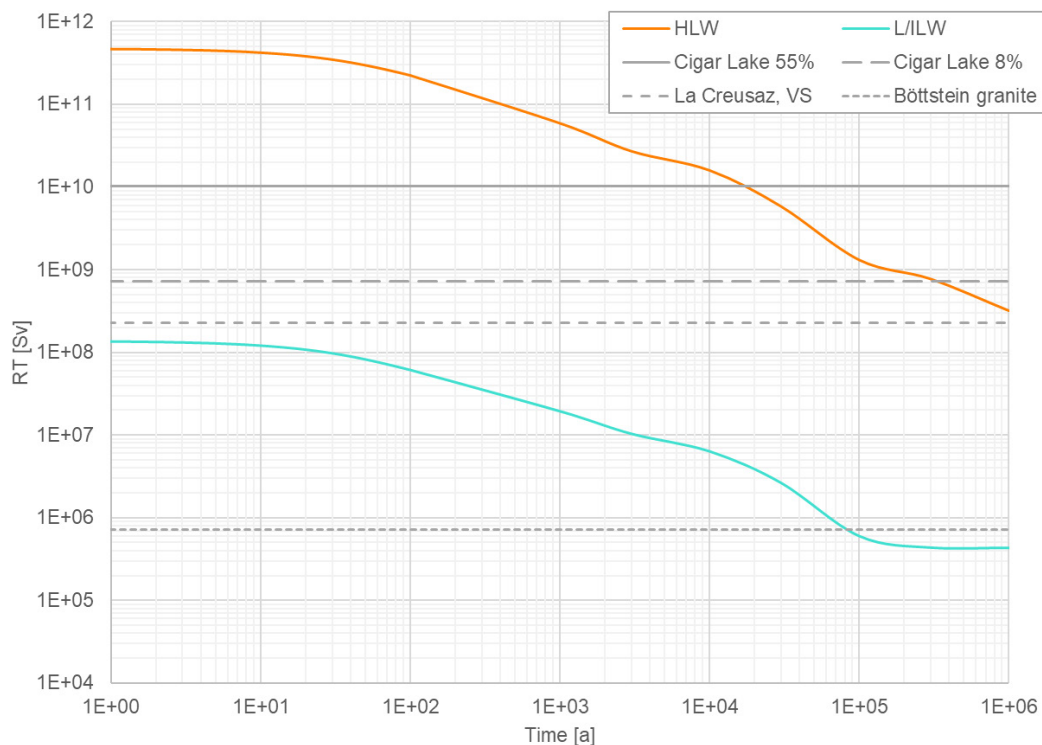


Fig. 9-3: Evolution of the radiotoxicity (RT) of the HLW and L/ILW disposed of in the repository

The horizontal reference lines correspond to the radiotoxicity of similar volumes of rocks and ores. Note that the uranium ore lines assume the approximate volume of the HLW repository section (100,000 m³), while the granite assumes the approximate volume of the L/ILW repository section (200,000 m³) (Nagra 2024a).

Fig. 9-3 also compares the RT of HLW and L/ILW disposed of in the repository as a function of time compared with the RTs of:

- Böttstein granite, a well-characterised, typical granite of Northern Switzerland,
- uranium ore from the Cigar Lake deposit in northern Saskatchewan, Canada, with two different uranium concentrations, and
- rock from La Creusaz, a small uranium deposit in canton Valais, Switzerland,

assuming a volume of these materials roughly equal to the volumes of the emplacement drifts and caverns (see figure caption for details). The estimation of the RT of ores from Cigar Lake and La Creusaz is described in Appendix A.1 of NAB 24-05 (Nagra 2024a). The estimation of the RT of Böttstein granite is described in Appendix A.2 of the same report.

After 100,000 years, the RT of the HLW has reached a level similar to that of low grade (8%) uranium ore from Cigar Lake, and, after 1 Ma, the RT is a little higher than that of the uranium-bearing rocks from La Creusaz.

The RT of L/ILW follows a similar trend to that of HLW, but with significantly lower RT values (roughly three orders of magnitude). At the time of emplacement in a deep geological repository, the radiotoxicity of the L/ILW is below that of both uranium ore from Cigar Lake and rock from the La Creusaz uranium prospect. After 100,000 years, L/ILW has a lower RT than Böttstein granite.

9.2.2 Distribution and decay of RT within the repository system

RT can also be used to illustrate how radiotoxicity is distributed within the repository system, how it diminishes over time due to radioactive decay, and how much reaches the biosphere. The following figures show that the great majority of the RT decays within the repository system and never reaches the surface environment.

The distribution of radiotoxicity in the various repository system barriers as a function of time in the reference case is shown in Fig. 9-4, Fig. 9-5, Fig. 9-6. Key observations are:

- SF: For times up to 10^6 years and beyond, the largest fraction of the radiotoxicity is contained within the fuel matrix. At 10^6 years, the radiotoxicity present outside the CRZ is still several orders of magnitude below that in the fuel matrix.
- RP-HLW: For times up to a few tens of thousands of years, the largest fraction of the radiotoxicity is contained within the glass matrix. At later times, the largest fraction is sorbed in the bentonite buffer. The radiotoxicity present outside the CRZ reaches a maximum after a few million years but is still orders of magnitude below that contained within the repository barrier system.
- L/ILW: For times up to about 10^5 years, the largest fraction of the radiotoxicity is sorbed in the cementitious near field, after which the largest fraction is contained within the Opalinus Clay. Radiotoxicity present outside the CRZ is too low at all times to be shown in the figure.

For comparison, as in the previous section, the RT of uranium ore from the Cigar Lake deposit in northern Saskatchewan, Canada, is also shown.

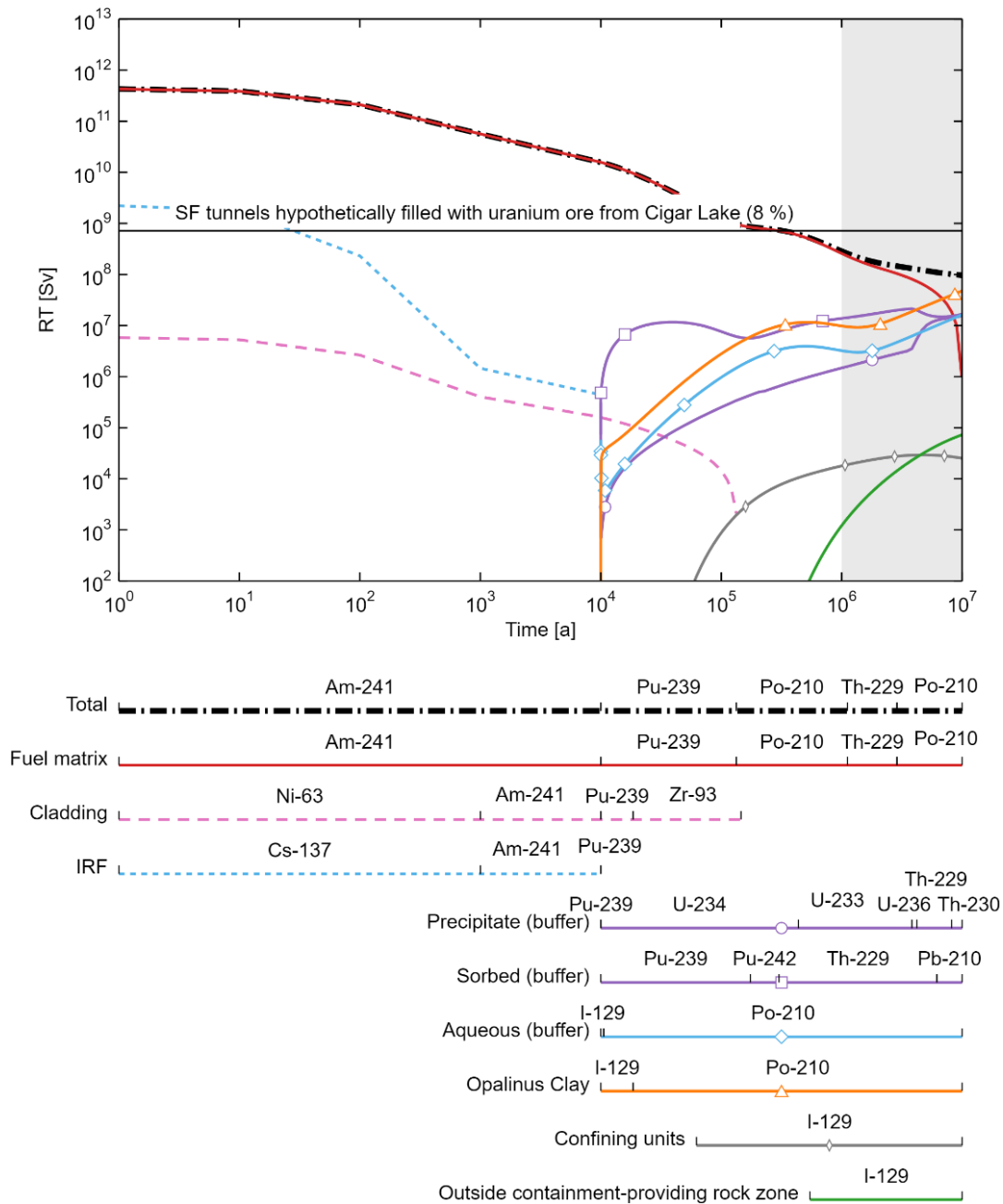


Fig. 9-4: Evolution and distribution of radiotoxicity (RT) from SF in the different components of the repository near field, the Opalinus Clay and its confining geological units, as well as RT that exists outside the containment-providing rock zone (CRZ)

The bars beneath the graph indicate the radionuclides that make the highest contributions to RT at any given time and in any particular part of the system.

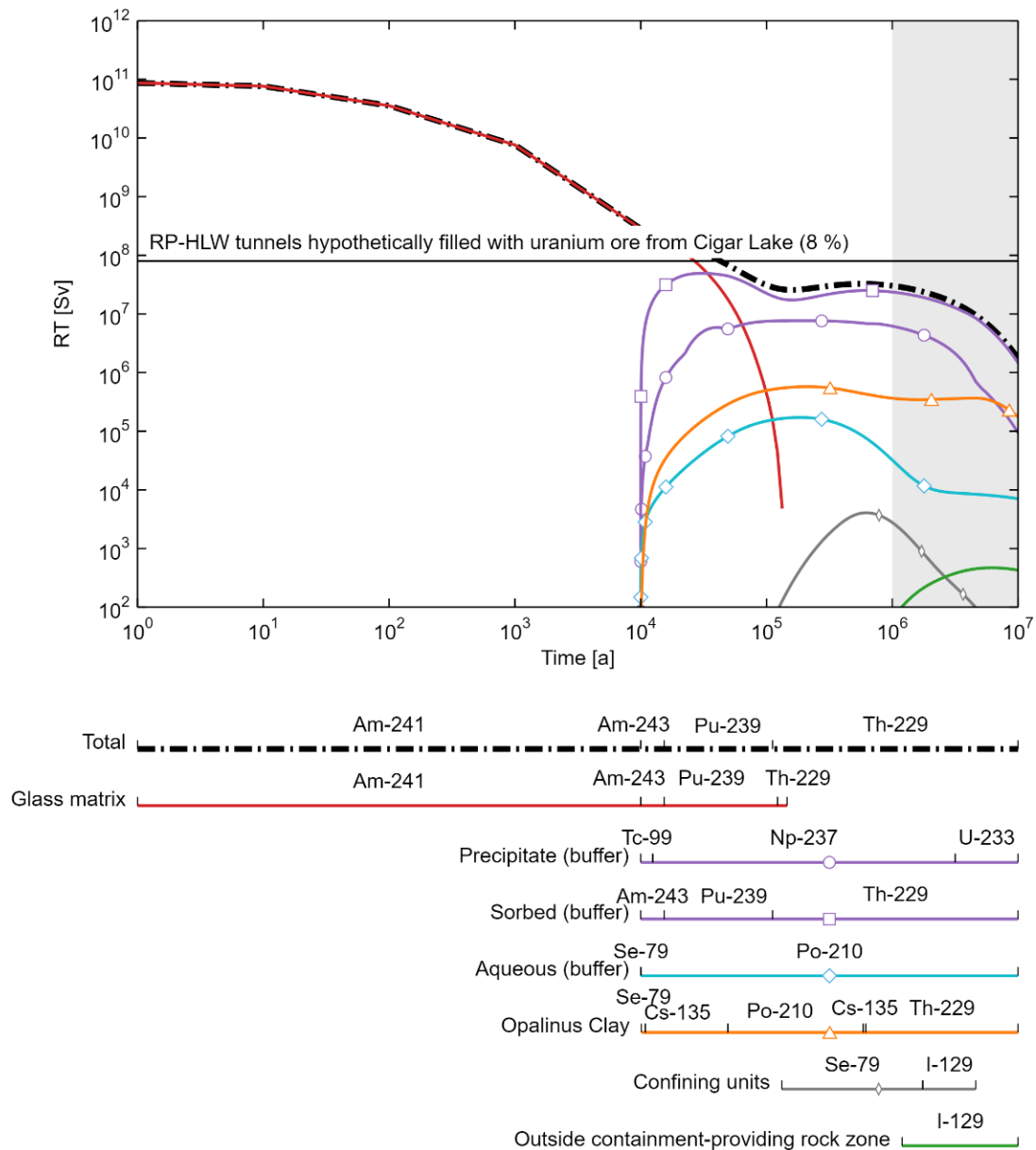


Fig. 9-5: Evolution and distribution of radiotoxicity (RT) from RP-HLW in the different components of the repository near field, the Opalinus Clay and its confining geological units, as well as RT that exists outside the containment-providing rock zone (CRZ)

The bars beneath the graph indicate the radionuclides that make the highest contributions to RT at any given time and in any particular part of the system.

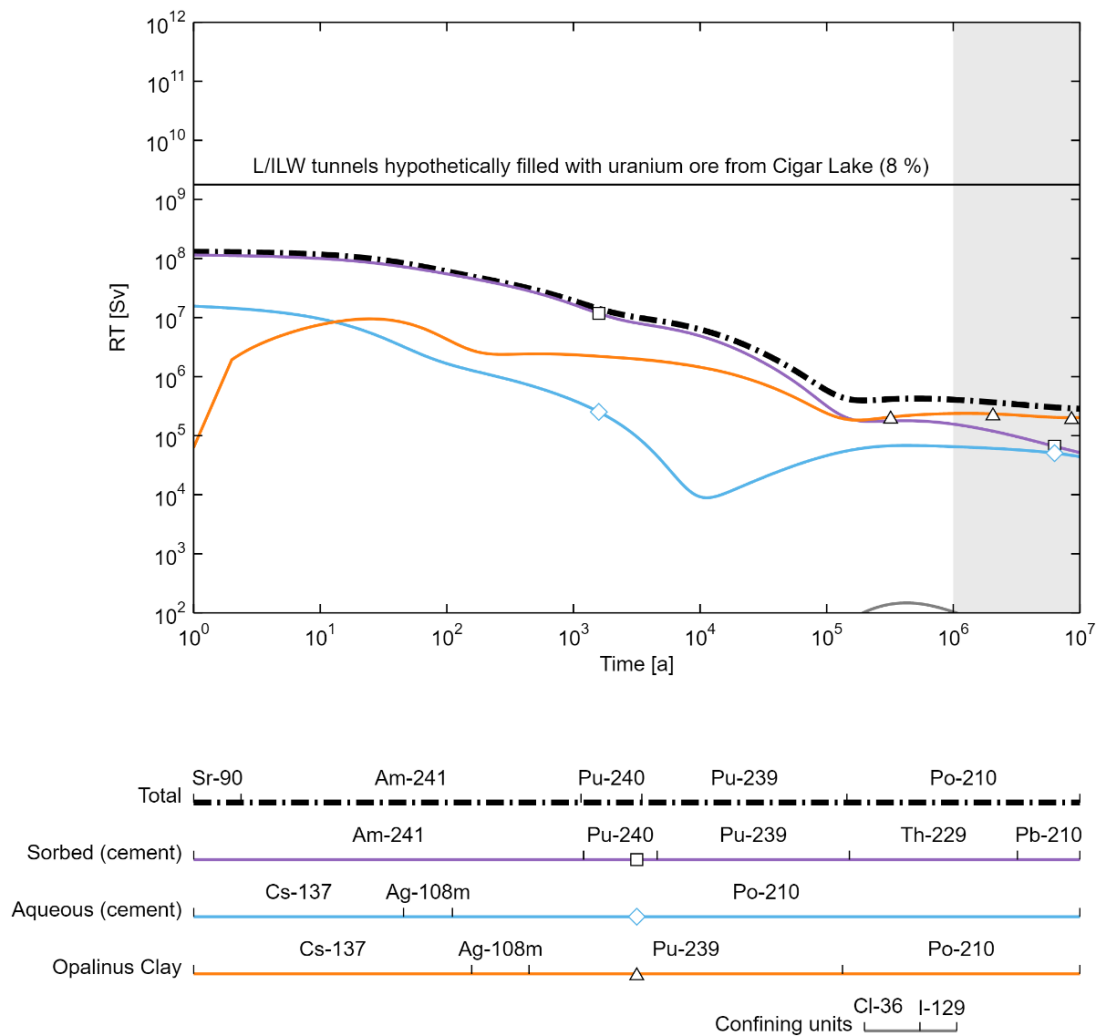


Fig. 9-6: Evolution and distribution of radiotoxicity (RT) from L/ILW in the different components of the repository near field, the Opalinus Clay and its confining geological units, as well as RT that exists outside the containment-providing rock zone (CRZ)

The bars beneath the graph indicate the radionuclides that make the highest contributions to RT at any given time and in any particular part of the system.

9.2.3 RT fluxes and concentrations in perspective

Fig. 9-7 shows the RT transfer rates, or fluxes, from the repository into the Malm aquifer above and into the Keuper and Muschelkalk aquifers below; these RT fluxes are assumed, in the analysis of radiological consequences, to be transferred directly to the biosphere. These transfer rates are compared with the following reference values, or “yardsticks”, which serve to place the transfer rates of RT from the repository in a broader perspective.

These yardsticks are based both on natural systems:

- RT fluxes in two major rivers in Northern Switzerland, the Rhine in the vicinity of the city of Basel and the Thur,

and on human actions:

- the RT of ash generated annually by the incineration of Swiss municipal waste.

The figure also shows the 60 Sv/a reference RT flux recommended in the EU SPIN project (Becker et al. 2002).

RT transfer rates to the biosphere via the three deep aquifers above and below the repository are orders of magnitude below all these yardsticks. They are, for example, three orders of magnitude or more below the SPIN reference value in the case of releases from HLW and five orders of magnitude or more below this reference value in the case of releases from L/ILW.

Fig. 9-8 shows the reference case aqueous RT concentrations due to releases from the HLW canisters, with an anticipated breaching time of 10,000 years, and from L/ILW at various locations within the repository barrier system, namely:

- the bentonite buffer, where the spatially averaged concentration is shown in the figure,
- the L/ILW cementitious near field, where again the spatially averaged concentration is shown in the figure and
- the lower boundary of the Opalinus Clay,
- as well as in the Malm aquifer above the repository and in the Keuper and Muschelkalk aquifers below it.

These concentrations are compared with the RT concentrations found naturally in Swiss drinking water and in Swiss mineral water; see Tab. 9-1 for information on these yardsticks and their sources. Fig. 9-8 also shows the 2×10^{-5} Sv/m³ reference RT concentration recommended in the EU SPIN project, which is based on K-40, Rb-97 and actinide concentrations naturally present in surface and shallow groundwaters in Finland, Switzerland, and the Czech Republic.

Aqueous RT concentrations in both the HLW near field and the L/ILW near field remain orders of magnitude above these yardsticks for all calculated times, emphasising why the geological barrier is needed to isolate the waste from humans and the surface environment.

The dominant pathway for radionuclide transport through the Opalinus Clay in the reference case is downwards from the repository towards the Keuper aquifer; see Chapter 8 or, for more detail, Section 7.1.1.3 in NTB 24-18 (Nagra 2024p). After a few tens of thousands of years, the RT concentration due to radionuclide releases from HLW in porewater at the lower boundary of the Opalinus Clay is also above the various yardsticks, but by a much lower margin (one to three orders of magnitude) than the aqueous RT concentrations in the HLW near field. The RT concentration due to releases from L/ILW in porewater at the lower boundary of the Opalinus Clay peaks at about 200,000 years and is similar to the upper bound value for Swiss drinking water, and around an order of magnitude above the SPIN reference value.

Aqueous RT concentrations are further greatly reduced when radionuclides reach the Malm, Keuper and Muschelkalk aquifers, due to decay during transport through the confining geological units and dilution within the aquifers themselves. The aqueous RT concentration in the Malm and Keuper aquifers due to releases from HLW are within the ranges of RT concentrations found in Swiss drinking water and in Swiss mineral water. The maximum RT concentration in the Malm aquifer is about an order of magnitude above the SPIN reference value, and the maximum RT concentration in the Keuper aquifer is a little below this reference value. RT concentration in the Muschelkalk aquifer is much lower, remaining about three orders of magnitude below the SPIN reference value.

Tab. 9-1: Descriptions, values and sources for the yardsticks presented in Fig. 9-7 and in Fig. 9-8

Yardstick	Description/value	Source
Rhine/Thur Rivers: water + sediment	RT flux due to radionuclides dissolved in water and due to radionuclides contained in sediments transported by the river; 3.5×10^5 Sv/a for the Rhine, 1.9×10^4 Sv/a for the Thur	A3.5.1 in Nagra (2002) and references therein
Rhine/Thur Rivers: water only	RT flux due to radionuclides dissolved in water; 2.7×10^4 Sv/a for the Rhine, 42 Sv/a for the Thur	
Ash from municipal waste incineration	Range of RT flux calculated for the whole of Switzerland; 4.7×10^3 Sv/a to 3.4×10^5 Sv/a	Radiological composition of ash from municipal waste incineration based on minimum and maximum values reported in Kallio et al. (2023); per-capita waste production in Switzerland is based on the Federal Office of the Environment (available online at https://www.bfs.admin.ch/bfs/de/home/statistiken/raum-umwelt/umweltindikatoren/alle-indikatoren/emissionen-und-abfaelle/siedlungsabfaelle.assetdetail.30645407.html); Size of Swiss population based on BFS (2022)
Drinking water	RT concentration in Swiss drinking water, ranging from 7.4×10^{-6} Sv/m ³ to 3.4×10^{-4} Sv/m ³	Radiological composition of drinking water based on Surbeck (2023) and on the Federal Office of Public Health (BAG 2023a)
Consumption of mineral water	RT concentration of mineral water, ranging from 1.1×10^{-5} Sv/m ³ to 1.4×10^{-4} Sv/m ³	Radiological composition of mineral water based on Baertschi (1995)
SPIN reference value	Based on several processes transferring naturally occurring radionuclides in the surface and subsurface environments, such as groundwater flow, weathering and erosion, and sediment transport, across an area of 200 km ² : 60 Sv/a	

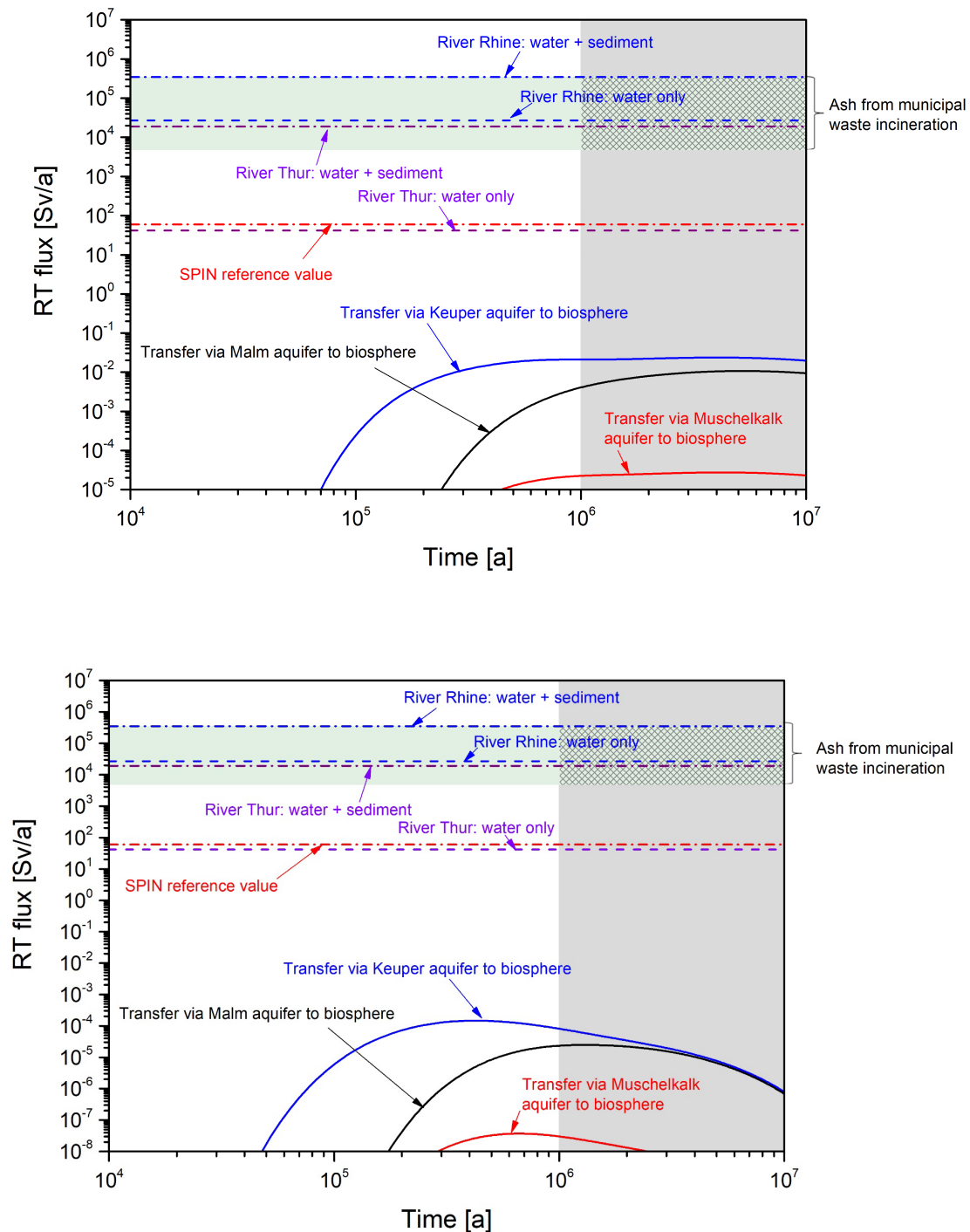


Fig. 9-7: Reference case RT transfer rates (flux) from HLW (upper figure) and from L/ILW (lower figure) into the Malm aquifer located above the repository and into the Keuper and Muschelkalk aquifers lying below it

These fluxes are assumed, in the analysis of radiological consequences, to be further transferred directly to the biosphere. These transfer rates are compared with RT fluxes in two major rivers in Northern Switzerland, with RT fluxes from human activities, i.e., the consumption of water for drinking and the incineration of municipal waste and with a reference RT flux from the EU SPIN project.

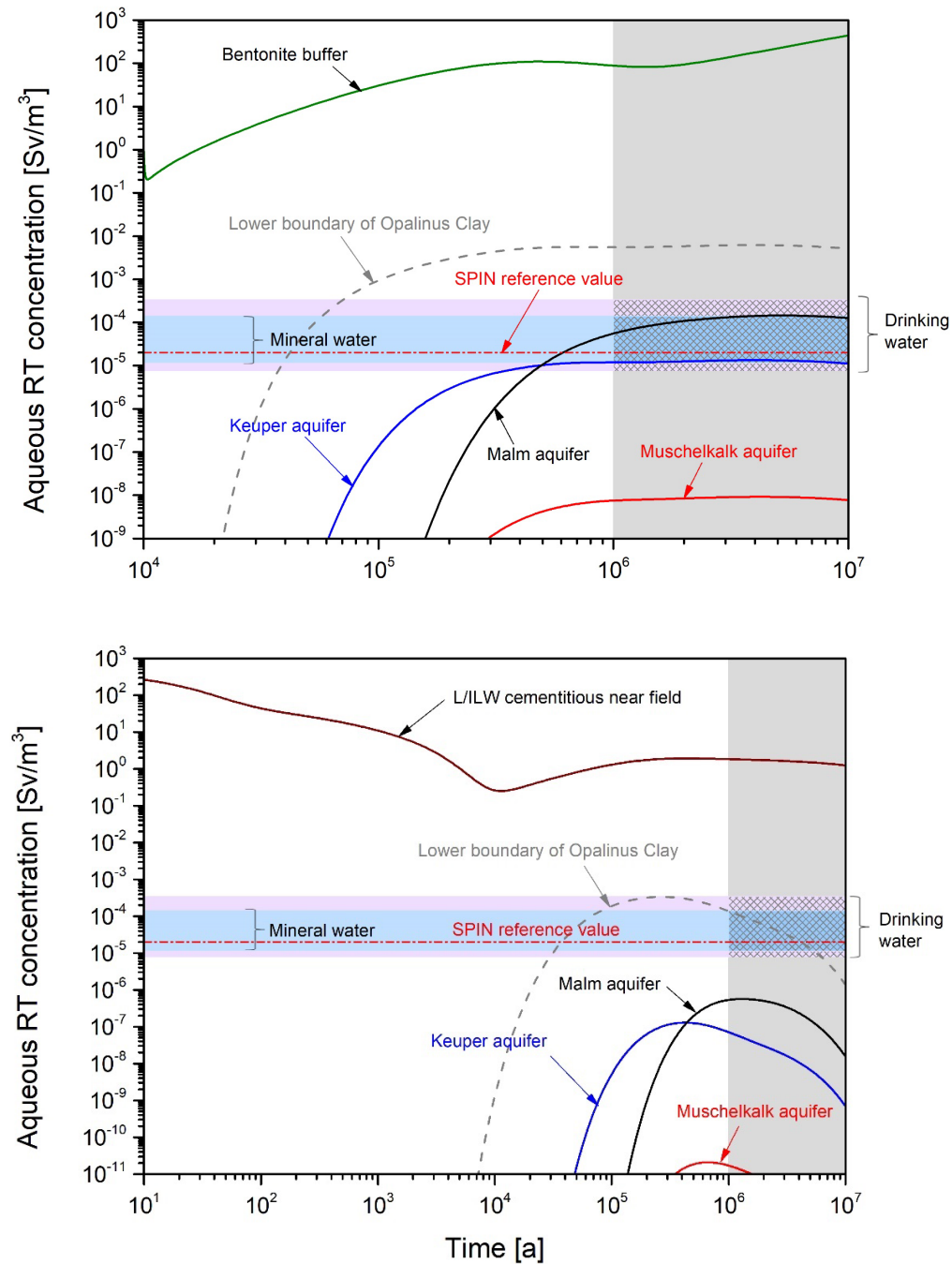


Fig. 9-8: Reference case aqueous RT concentrations due to releases from HLW (upper figure) and from L/ILW (lower figure) at various locations within the repository barrier system, as well as in the Malm aquifer located above the repository and in the Keuper and Muschelkalk aquifers lying below it

These concentrations are compared with the RT concentrations found naturally in Swiss drinking water and in Swiss mineral water and with a reference RT concentration from the EU SPIN project. Note that the concentrations in the aquifers are obtained by dividing calculated RT fluxes from the rock to the aquifers by assumed aquifer flow rates across the repository area. These flow rates are subject to substantial uncertainty and are estimated here for illustrative purposes only.

The aqueous RT concentration in the deep aquifers due to releases from L/ILW are lower still and remain at least an order of magnitude below all the above-mentioned yardsticks.

Many radionuclides released from the repository sorb strongly on various barrier materials, retarding their transport and providing more time for radioactive decay. Fig. 9-9 shows reference case RT concentrations, due to both sorbed radionuclides and radionuclides present in the aqueous phase, in barrier materials within the repository system, namely:

- the HLW bentonite buffer,
- the L/ILW cementitious near field, and
- the Opalinus Clay host rock.

In all three cases, spatially averaged concentrations are shown. For the Opalinus Clay host rock, this involves spatial averaging over the rock volume above and below the repository, up to the boundaries with the upper and lower confining geological units.

These concentrations are compared with the RT concentrations found naturally in various rocks, ores and average soil in Switzerland. The rock and ores used in this comparison are the same as considered in Section 9.2.1 in the context of the declining radiological hazard from the waste packages, i.e.:

- Böttstein granite, a well-characterised, typical granite of Northern Switzerland,
- uranium ore from the Cigar Lake deposit in northern Saskatchewan, Canada, with two different uranium concentrations, and
- rock from La Creusaz, a small uranium deposit in canton Valais, Switzerland,

as well as the Opalinus Clay itself, as currently characterised, prior to any releases from the repository.

The RT concentrations in barrier materials within the repository system lie within the range spanned by these naturally occurring materials and are generally below the RT concentration in uranium ores.

RT concentrations in the near-field materials are generally above those found naturally in the Opalinus Clay, Böttstein granite and in Swiss soils. However, the RT concentrations in the Opalinus Clay due to releases from HLW and L/ILW are generally within the same range as found naturally in these materials. In the case of releases from HLW, the RT concentration in the Opalinus Clay increases over several millions of years, reaching a plateau value (once all the SF has degraded and released its radionuclides) that is between that of the Opalinus Clay and that of Böttstein granite. In the case of releases from L/ILW, the RT concentration in the Opalinus Clay peaks relatively rapidly, after just a few decades, at a value between that of the Opalinus Clay and the Böttstein granite, and declines gradually thereafter, reaching a value similar to that of average soil in Switzerland after about 10,000 years.

Overall, the comparisons shown in this section demonstrate that calculated RT transfer via deep aquifers to the biosphere is often much lower than the RT fluxes that currently occur in some natural systems (e.g., major rivers in Northern Switzerland), and much lower than those arising from some current human actions (waste incineration, consumption of drinking water and mineral water in Switzerland). RT concentrations, both aqueous and sorbed, are relatively high in the repository near field, but, in the host rock, they are similar to those occurring in many natural systems and materials. Furthermore, in the deep aquifer water that could potentially reach the biosphere, aqueous RT concentrations due to repository releases are well below the RT concentrations occurring in many natural systems and materials. Such comparisons can place the RT

transfers due to the repository in a broader perspective, emphasising that radionuclide fluxes and concentrations that may arise due to the presence of the repository are often much lower than those that already exist in the human environment today.

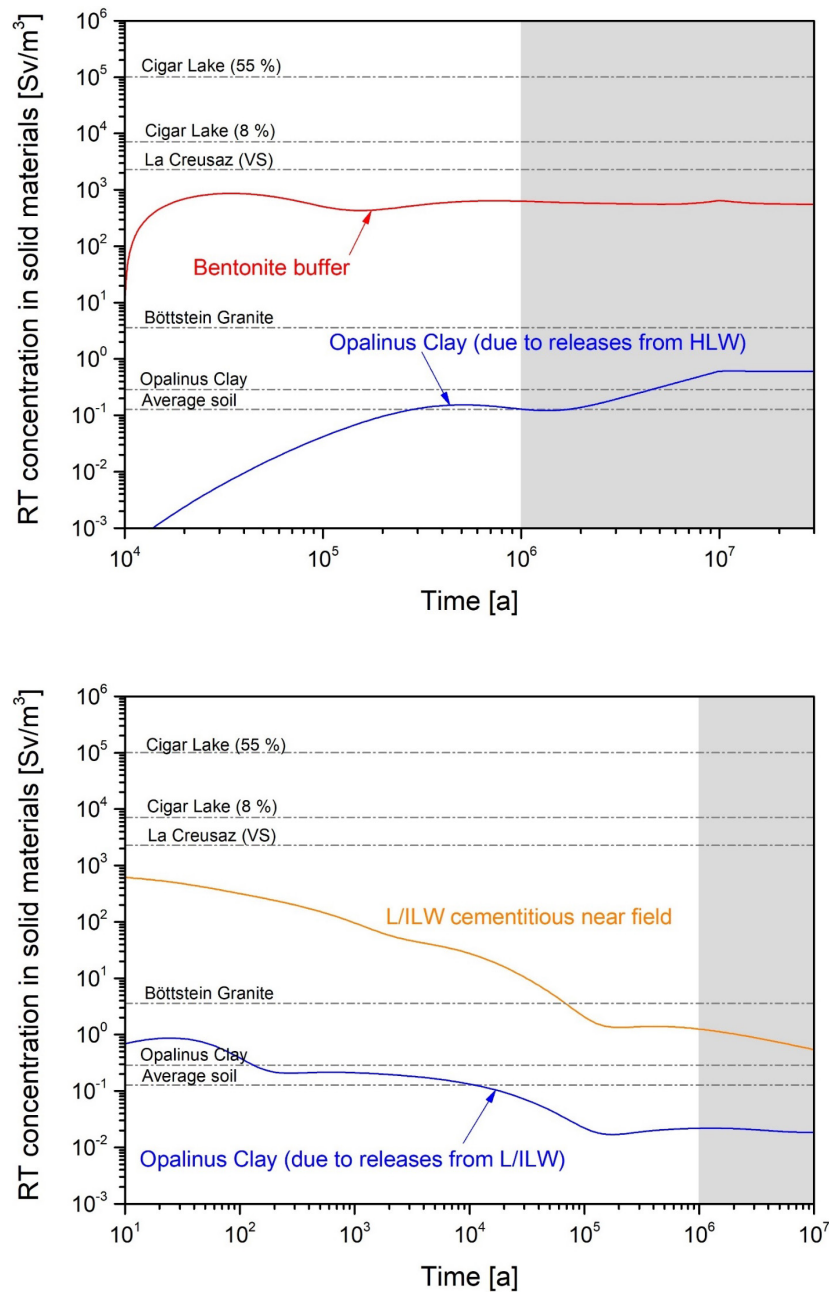


Fig. 9-9: Reference case RT concentrations, sorbed and aqueous, in materials in the HLW bentonite buffer and in the L/ILW cementitious near field, as well as in the Opalinus Clay host rock, due to releases from HLW (upper figure) and from L/ILW (lower figure)

These concentrations are compared with the RT concentrations found naturally in various rocks and ores, including the Opalinus Clay, and in average soil.

9.3 Conservatism of the calculated radiological consequences

The models used to calculate doses in the analysis of radiological consequences, as well as to evaluate the RT, RT fluxes and RT concentrations described in the previous sections, incorporate numerous conservative assumptions and simplifications to handle uncertainties that are not addressed by other means (e.g., through parameter variations).

Conservatisms include the deliberate omission of some phenomena that are considered likely to occur and that are beneficial to safety, from quantitative analysis because suitable models, codes, or databases are unavailable. Such phenomena are termed *reserve FEPs*, since they can be mobilised at a later stage of the waste disposal programme, provided the necessary models, codes, and databases are developed. Important reserve FEPs identified in the course of the Project *Entsorgungsnachweis* (Section 8.2.8.3 of NTB 02-05, Nagra 2002) that remain reserve FEPs in the present safety assessment are:

- the co-precipitation of radionuclides with secondary minerals derived from spent fuel, glass and canister corrosion (except for co-precipitation of radium, which is included in all cases),
- sorption of radionuclides on canister corrosion products,
- natural concentrations of isotopes in solution in bentonite porewater, which could further reduce the effective solubilities of some radionuclides,
- irreversible sorption of radionuclides in the near field or in the geosphere (surface mineralisation),
- long-term immobilisation processes (precipitation / co-precipitation) in the geosphere, and
- the delayed release of radionuclides, due to the slow corrosion rate of L/ILW metallic materials (e.g., hulls and ends), as well as a period of complete containment by L/ILW steel drums and emplacement containers.

Newly identified reserve FEPs that concern radionuclide release and transport in the aqueous phase are:

- the absence, for an extended period of time, of a continuous liquid phase connecting the L/LW cavern and the host rock, which would be needed for the release of radionuclides into the aqueous phase and further transport to the biosphere, and
- the increased sorption of some radionuclides due to elevated temperatures of the geological environment.

In addition, newly identified reserve FEPs that concern repository-generated gas and radionuclide release and transport in the gas phase are:

- the presence of microorganisms in the backfilled operations and construction tunnels that have the potential to reduce the hydrogen gas pressure to well below the hydrostatic pressure, and
- the impact on corrosion and degradation rates of the very slow resaturation of the repository, resulting in a very slow exchange of the original atmosphere with repository-generated gas such as hydrogen and slow build-up of gas pressures.

These reserve FEPs have the potential, in the future, to provide additional quantitative contributions to the evaluated performance of the disposal system. One specific reserve FEP that was identified in the *Entsorgungsnachweis*²⁷, but, due to site characterisation activities carried out in the intervening years, was mobilised in the present safety assessment, is:

- retardation in the confining geological units of the Opalinus Clay host rock.

This reserve FEP was conservatively neglected in the reference conceptualisation of the geosphere in the radionuclide retention and transport calculations for the *Entsorgungsnachweis* but is included in the present analysis of radiological consequences, which considers retention and transport through a broader CRZ.

Even when they are not mobilised, the presence of reserve FEPs constitutes, in effect, an additional qualitative argument for safety, since it indicates that the actual performance of the disposal system will, in reality, be more favourable than that evaluated in the analysis of assessment cases given in Chapter 8, or with more detail in Chapter 7 of NTB 24-18 (Nagra 2024p).

In addition to the reserve FEPs, there are further reserves due to several simplifying pessimistic or conservative assumptions that had to be made for the quantitative analysis of some assessment cases because of the limitations of the available models and codes as well as uncertainty in the underlying scientific understanding. Conceptual assumptions and simplifications of the models used in the analysis of radiological consequences are described in Chapter 3 and Appendix A of NTB 24-18 (Nagra 2024p). Many of those that are currently highly conservative are expected to be replaced by more realistic assumptions in future safety assessments, which will yield even lower calculated radiological consequences than in the present assessment.

²⁷ Retardation in the confining geological units was only taken into account in a few calculations' cases, due to limited data on the confining geological units that was available at the time.

10 Demonstration of post-closure safety

Aims of the chapter:

- To integrate the safety arguments in a holistic demonstration of post-closure safety.

This chapter integrates the arguments and evidence elaborated in the earlier chapters of this report in a synthesis of the claims that together constitute a demonstration of the post-closure safety of a repository for both HLW and L/ILW constructed in the Opalinus Clay at the proposed site. As illustrated in Fig. 10-1, these arguments justify claims concerning:

- the strength of the safety-driven processes for repository siting, design, and implementation,
- the favourable qualities of the site, the CRZ and the engineered barrier system,
- the quality and comprehensiveness of the safety assessment, including the quality of the assessment basis and its proper application when carrying out the main safety assessment processes, including performance assessment, scenario development, and the analysis of radiological consequences
- the favourable results of the safety assessment, and
- complementary lines of argument, including those based on natural and archaeological analogues and complementary performance and safety indicators.

Fig. 10-2 illustrates how these arguments combine to provide the level of confidence in post-closure needed in order to proceed with the disposal programme, while specific aspects are given in the following sections.

The safety demonstration is further supported by more general arguments for the strength and feasibility of deep geological disposal as a waste management option as presented in Section 2.1.

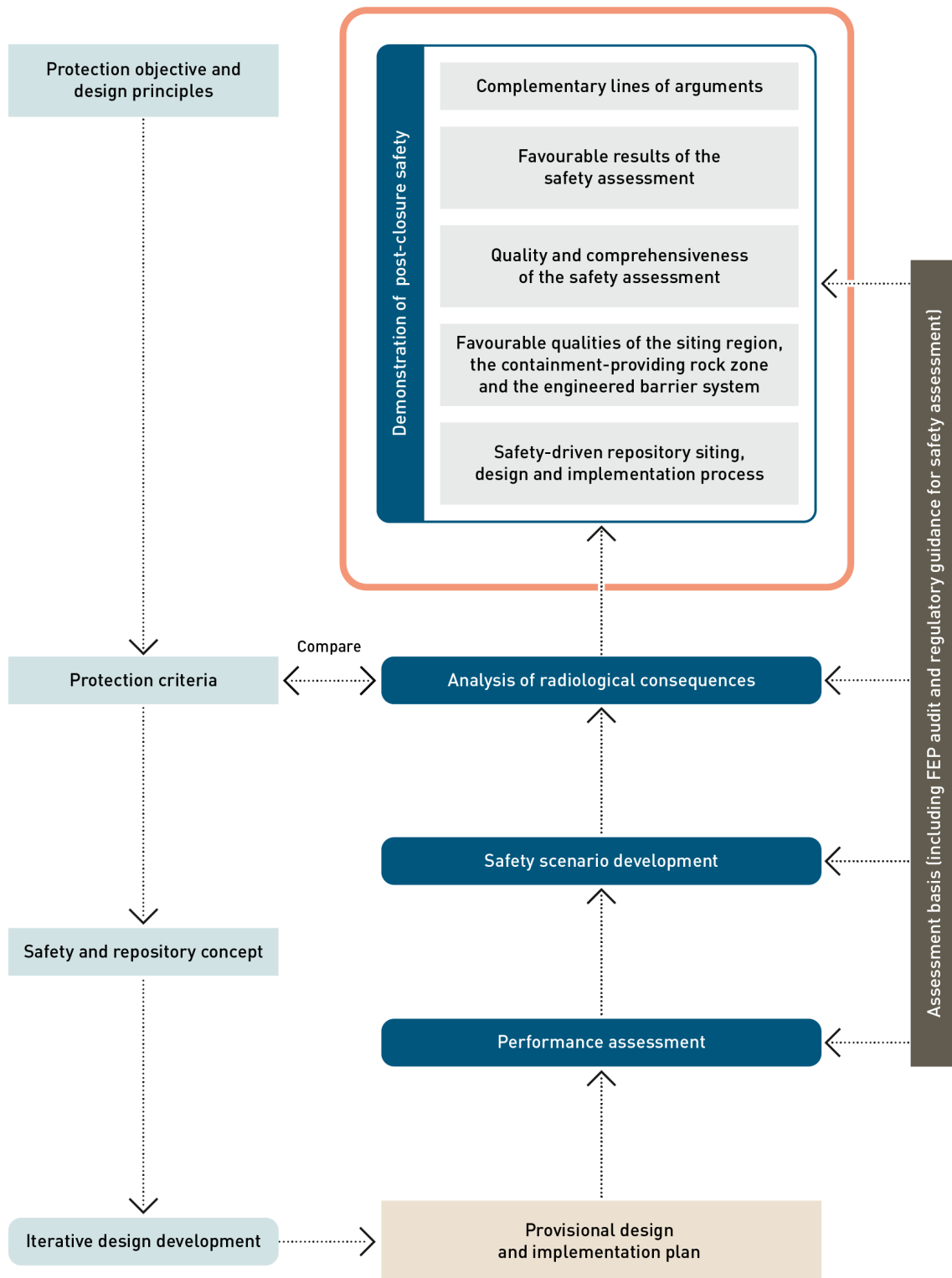


Fig. 10-1: Workflow for the post-closure safety case, highlighting the main elements of the demonstration of post-closure safety

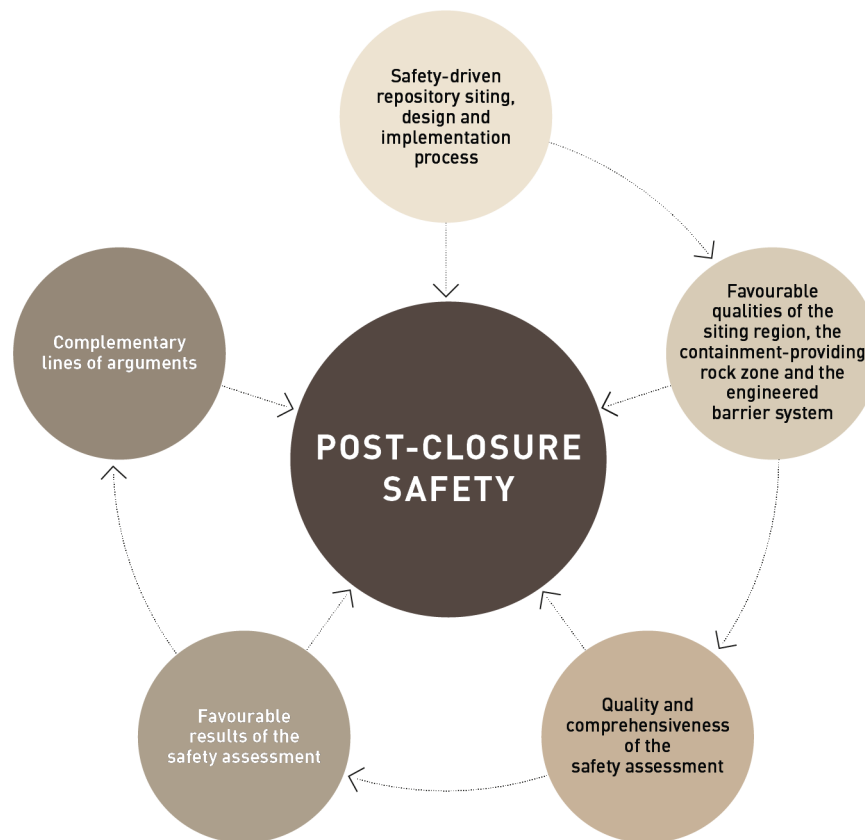


Fig. 10-2: Lines of argument in the present post-closure safety case

10.1 Safety-driven siting, design and implementation processes

Nagra has proposed a repository site and developed a provisional design and implementation plan resulting from transparent, safety-driven processes that have spanned over many years; *the repository siting, design and implementation processes generate confidence in the post-closure safety because they are safety-driven*. Specific evidence for this argument is summarised in Fig. 10-3 and elaborated in the following paragraphs.

1a: A safety-driven multi-stage siting process has been followed under the oversight of the regulatory authorities

The Sectoral Plan for Deep Geological Repositories (SGT) regulates the search for suitable siting regions for deep geological repositories in Switzerland. It is conducted in three stages (BFE 2008), Stages 1 and 2 have already been completed, and the general licence application, to which the present post-closure safety case belongs, is a result of the ongoing Stage 3. Clarifications on the safety-related specifications for this stage are provided in ENSI 33/649 (ENSI 2018), and have been fully followed by Nagra.

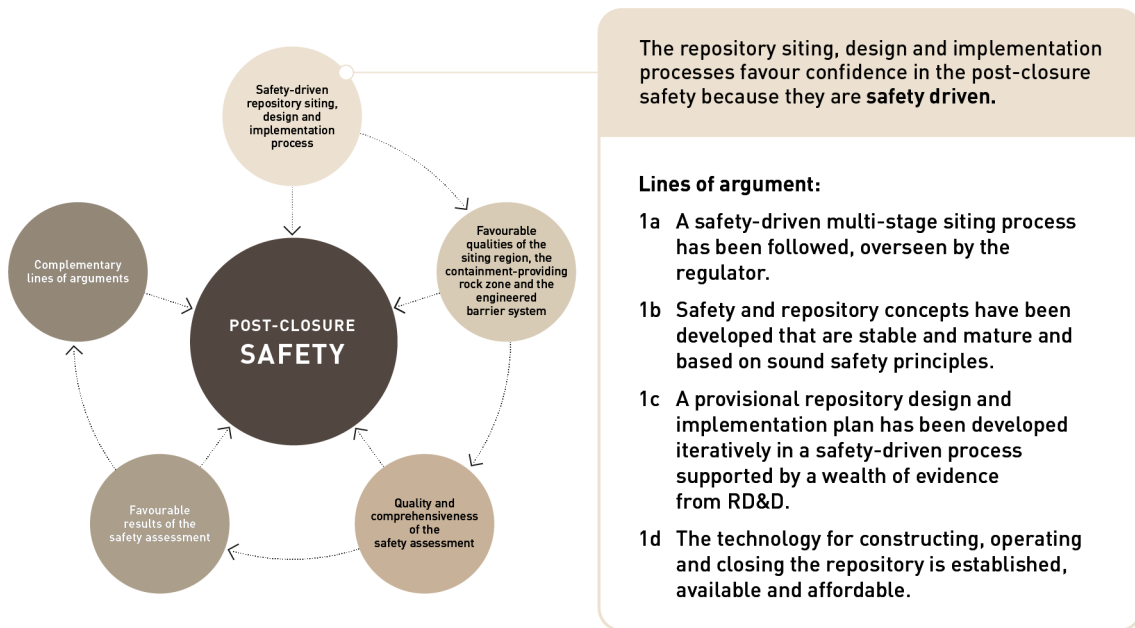


Fig. 10-3: Lines of argument for the claim that the repository siting, design and implementation processes favour confidence in the post-closure safety because they are safety-driven

Siting regions have only progressed from one stage to the next, once they have been shown to possess multiple attributes that favour post-closure safety (as well as engineering feasibility), based on criteria developed by the Swiss Federal Office of Energy (SFOE; BFE 2008), together with other agencies and organisations that are independent of Nagra. Provisional safety assessments have been used throughout the siting process to support the narrowing down of potential host rocks and siting regions. The siting process has thus been safety driven from start.

The methodology adopted by Nagra for the comparison of the three siting regions in Stage 3 of the Sectoral Plan is consistent with the safety assessment methodology used for the post-closure safety case, which is summarised in Chapter 4 and described in Chapter 6 of NTB 24-19 (Nagra 2024t). For a more detailed description of the methodology for site comparison and for the selection of the proposed site, the reader is referred to NTB 24-03 (Nagra 2025a) and its supporting reports.

1b: A safety and repository concept has been developed that is mature and based on sound safety principles

The safety concept explains how the repository system ensures the protection of humans and the environment through the safety barriers it provides and the safety functions they perform. The repository concept describes these repository safety barriers and their properties. Nagra developed the safety and repository concept for the disposal of all types of radioactive waste more than two decades ago, and the main features have remained largely unchanged throughout the intervening years. This concept, for example, formed the basis for Project *Entsorgungsnachweis* (Nagra 2002), which successfully demonstrated the feasibility of disposing of HLW and long-lived ILW in the Opalinus Clay of Northern Switzerland, as well as for the subsequent repository siting programme. The current safety and repository concept, though still provisional, is thus considered mature. It conforms with design principles and requirements for a deep geological repository set out in the Nuclear Energy Ordinance of 2004 (KEV 2004), including that the repository ensures post-closure safety through multiple passive safety barriers and that it is designed in such a way that, during operational phase, it can, if necessary, be closed within a period of a few years. It also conforms with international principles of passive safety, optimisation of protection, and defence-in-depth.

Nagra's current safety and repository concept is described in Chapter 3 of the present report and, in more detail, in NAB 24-18 Rev. 1 (Nagra 2024s). Design principles are described as part of the contextual and regulatory framework in Chapter 2.

1c: A provisional repository design and implementation plan has been developed iteratively over many years in a safety-driven process supported by a wealth of evidence from RD&D

The present post-closure safety case is based on a provisional repository design and implementation plan that is in line with the regularly updated Waste Management Programme NTB 21-01.

The development is safety-driven, meaning that safety is the prime concern when considering refinements or larger changes to the design and implementation plan. An important aspect of this are the system analyses that are performed to establish safety-related design requirements as illustrated in Fig. 10-4. For example, thermal analyses are performed to formulate provisional requirements on the loading of the HLW disposal canisters, their spatial separation along the HLW emplacement drifts and the lateral separation between the drifts, thus ensuring that the emplacement of heat-generating waste does not compromise the safety functions of the engineered barriers.

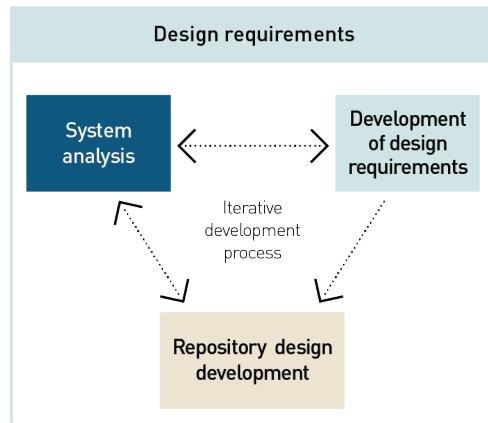


Fig. 10-4: Illustration of the iterative design development process, showing the interactions with system analysis and the development of design requirements

1d: The technology for constructing, operating and closing the repository is established, available and affordable

Based on national and international experience in radioactive waste management and in other industries, the state-of-the-art technology needed for constructing, operating and closing the repository (e.g., for further exploration of the site, excavation, handling of the waste, manufacturing / emplacing the engineered barriers, backfilling) is established, available and affordable. Furthermore, significant technological progress, e.g., in robotics and control, can be expected in near future.

The technology for constructing, operating and closing the repository is described in NAB 23-01 (Nagra 2023a) and NTB 24-11 (Nagra 2024b).

10.2 Robustness of the repository system

The repository system is robust, meaning that the site, the containment-providing rock zone and the engineered barrier system have many qualities that are favourable to safety, are well-understood, and that detrimental phenomena are, as far as possible, reduced or avoided. Specific evidence for this argument is summarised in Fig. 10-5 and elaborated in the following paragraphs.

2a: The site is geologically stable and provides high flexibility for repository placement

The site is located in a tectonically quiet and stable area with limited deformation. It is located well away from large-scale tectonic zones characterised by high uplift, enhanced seismicity or high tectonic complexity, such as the Alps. The occurrence of large flat-lying and undisturbed host-rock zones within the site has also been conclusively demonstrated, providing a high degree of flexibility regarding the repository location and layout.

The characteristics of the site are described briefly in Section 5.2 of the present report, while all aspects are fully described in the Geosynthesis of Northern Switzerland report NTB 24-17 (Nagra 2024i).

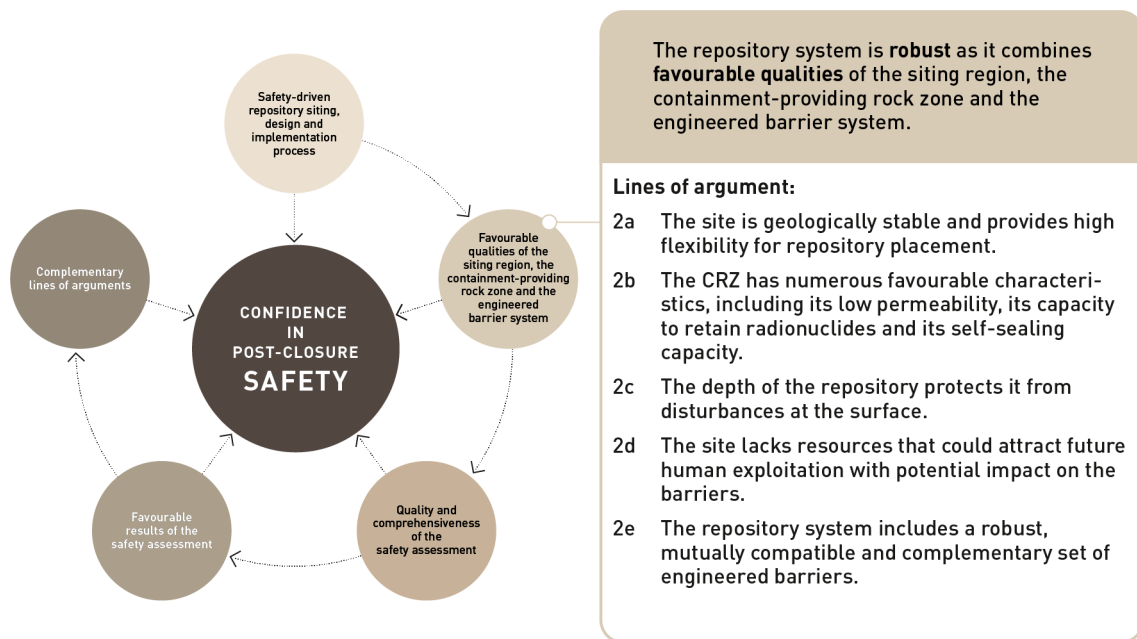


Fig. 10-5: Lines of argument for the claim that the repository system is robust

2b: The CRZ has numerous favourable characteristics, including its low permeability, its capacity to retain radionuclides and its self-sealing capacity

The geological barrier provided by the CRZ makes a powerful contribution to the robustness of the current safety concept due to its excellent qualities. The Opalinus Clay host rock is characterised by a fine and homogeneous pore structure, extremely low hydraulic permeability, a homogeneous and high clay-mineral content and the associated high sorption capacity for many radionuclides and high self-sealing capacity. Due to high clay contents, the confining geological units have similarly favourable properties and ensure a sufficiently large distance between the Opalinus Clay and the nearest water-bearing rock layers, contributing to the retention of any radionuclides released from the repository. The CRZ is also characterised by favourable properties for long-term stability, especially its self-sealing capacity, thus forming an effective and stable barrier against radionuclide transport and a suitable physical and chemical environment for the engineered barriers. This is explicitly demonstrated by the extremely long residence time for Opalinus Clay porewater, which indicates highly effective containment.

The properties of the CRZ are described briefly in Section 5.2 of the present report, while details are found in NTB 24-17 (Nagra 2024i).

2c: The depth of the repository protects it from disturbances at the surface

Host-rock depth at the site is located within an ideal depth window, sufficient to protect the repository from future erosion and other events and processes at the surface, including most conceivable future human actions, and shallow enough to ensure technical feasibility. Excavation of the repository within the next one million years, due, for example, to glacial and/or fluvial erosion, has been comprehensively assessed and is not expected to occur within this time period.

Isolation from geological and climatic processes at the surface is described in Chapter 6 of NTB 24-17 (Nagra 2024i). The negligible impact of most conceivable future human actions is discussed briefly in Sections 7.5 and 8.5 and in detail in NAB 24-09 (Nagra 2024r).

2d: The site lacks resources that could attract future human exploitation with potential impact on the barriers

No relevant natural mineral resources are located in the host rock that could attract future exploration and exploitation. Future exploration and exploitation of surface and shallow subsurface mineral and other resources (e.g., open limestone quarries) will have no impact on a deep geological repository within the Opalinus Clay layer as these resources are well above any confining geological units and thus have no impact on the barrier system of the repository. Although the deep Permo-Carboniferous troughs could potentially contain some hydrocarbons, there is no indication of concentrations that could justify drilling and exploitation. The potential for geothermal energy production is not significantly enhanced at the site.

At a depth of around 900 m below ground, the repository is also unlikely to be affected by shallow engineering or geotechnical construction projects, by surface water management activities, by surface waste disposal activities, by surface water and groundwater pollution, by archaeological studies, or by the exploitation of shallow groundwater systems in the future. The only exception to this is the potential disturbance of the uppermost part of the access shaft. However, this part of the shaft does not contribute to post-closure safety as multiple barriers isolate it from the main repository system. The absence of resources that could attract future human exploitation is described in Dossier VII of NTB 14-02 (Nagra 2014).

2e: The repository system includes a robust, mutually compatible and complementary set of engineered barriers

The geological 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 and gas generated by the waste and the disturbance to the rock caused by the excavation and ventilation of underground openings.

Engineered materials, such as steel, bentonite, and concrete are used, for which there is already a substantial knowledge base and experience in their application in many different fields, and which are chemically compatible with each other and with the surrounding rock, or for which interactions have limited impact. Furthermore, an appropriate distance between HLW emplacement drifts and L/ILW emplacement caverns counters possible thermal, hydraulic and chemical interactions between these two parts of the repository.

Key features of the engineered barriers that contribute to the safety functions and that are considered well understood are robust, and are thus classified as pillars of safety, including:

- the HLW disposal canisters, which are mechanically stable and corrosion-resistant in the expected environment and ensure complete containment of the radionuclides for a significant period of time,
- the bentonite buffer inside the HLW emplacement drifts, which provides a stable barrier against radionuclide transport, a suitable physical and chemical environment that enhances the longevity of the HLW disposal canisters and mechanical support to the host rock after the degradation of the tunnel support, and it reduces the potential impact of the heat release by the HLW on the radionuclide retention properties of the CRZ,
- the cementitious L/ILW near field, which promotes geochemical immobilisation and sorption of radionuclides, favours low rates of metal corrosion for metals and low rates of degradation of organic compounds in the repository; the low compressibility of the backfill also favours the mechanical stability of the host rock, even after the degradation of the tunnel support, and

- the closure system, which forms a stable barrier against radionuclide transport, provides mechanical stabilisation of the host rock and, together with the L/ILW near field, serves as a gas transport and gas storage system that prevents gas pressure build-up.

The performance of these pillars of safety over time and their robustness with respect to uncertainties and detrimental phenomena is evaluated in the performance assessment as summarised in Chapter 6 and in more detail in NTB 24-22 Rev. 1 (Nagra 2024u).

The design of the repository also ensures that, in the event of intrusion by drilling, only a small part of the repository is affected. This is achieved by compartmentalisation, e.g., each HLW disposal canister is completely surrounded by massive amounts of bentonite and thus forms an isolated compartment, with no shortcut from one to the next. The limited size of the L/ILW emplacement caverns has a similar effect. Solidification of the wastes ensures that only a small fraction of radionuclides released from the waste package are in solution and can be transported to the surface rapidly along any boreholes created by humans in the post-closure period, once records of the repository are lost.

Design aspects of the engineered barriers contributing to the repository safety functions are outlined in Chapter 3 and presented in detail in NAB 24-18 Rev. 1 (Nagra 2024s).

10.3 Sound assessment basis

The quality of the safety assessment rests on a sound assessment basis, including a sound scientific understanding of all relevant phenomena, a safety assessment methodology that respects national and international requirements, principles, guidance, and best practice, and models, codes and data to support both performance assessment and the analysis of radiological consequences that are fit for purpose. Specific lines of argument for the claim that a sound assessment basis, including an appropriate safety assessment methodology, has been developed and applied are summarised in Fig. 10-6 and elaborated in the following paragraphs.

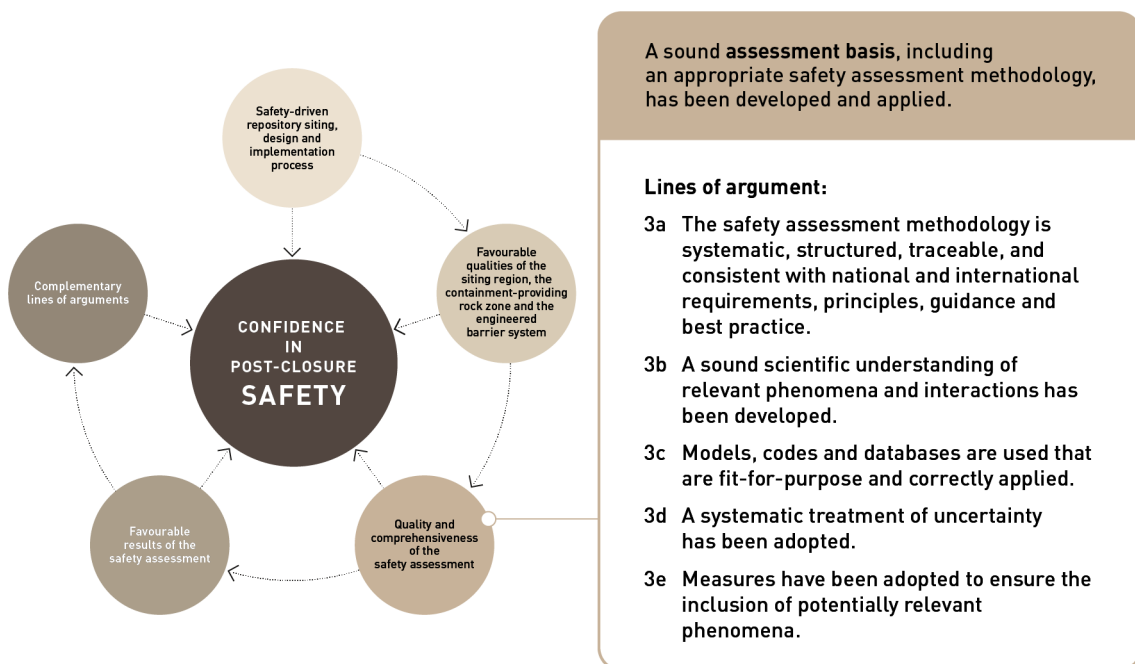


Fig. 10-6: Lines of argument for the claim that a sound safety assessment methodology and a sound assessment basis have been developed and applied

3a: The safety assessment methodology is systematic, structured, traceable, and consistent with national and international requirements, principles, guidance and best practice

The safety assessment methodology is described in NTB 24-19 (Nagra 2024t) and summarised in Chapter 4 of the present report. The safety assessment methodology comprises a set of distinct processes, each with its own systematic, structured, and traceable workflow; see Fig. 6-1, Fig. 7-1 and Fig. 8-1, which present the workflows for performance assessment, for scenario development and for the analysis of radiological consequences, respectively. The methodology conforms with a set of safety assessment principles, which are themselves based on Swiss legal and regulatory requirements, international developments and guidance, and Nagra's own experience from past safety assessments (see Section 4.1 as well as Chapter 3 of NTB 24-19, Nagra 2024t). Among these principles are the systematic and rigorous consideration and treatment of uncertainty, and the assurance of comprehensiveness, as described further below.

3b: A sound scientific understanding of relevant phenomena and interactions has been developed

A sound scientific understanding of the features, events and processes (FEPs) relevant to the disposal system and its evolution, and the interactions between these FEPs, has been developed through Nagra's site characterisation and RD&D programmes over many years. The quality of this scientific understanding is ensured, e.g., by publication in peer-reviewed reports and journals and participation in relevant international initiatives.

Current scientific understanding of the features, events and processes (FEPs) relevant to the disposal system and its evolution and the interactions between these FEPs have been synthesised in a supplementary volume of NAB 24-20 Rev. 1 (Nagra 2024l) and is summarised in Sections 5.1 to 5.4 of the present report.

3c: Models, codes and databases are used that are fit-for-purpose and correctly applied

Measures taken to ensure models, codes and databases used in performance assessment and in the analysis of radiological consequences are fit-for-purpose include comparison of model outputs with the results of experiments, covering a range of spatial and temporal scales and with observations of natural systems, verification of numerical codes, e.g., by benchmarking against analytical solutions and against other codes that address similar problems, and quality control of the data feeding into the safety assessment databases. Quality assurance and control are applied to all activities that produce or apply models, codes and databases.

Individual models, codes and databases are discussed, and their reliability described, in the specific reports that they correspondingly support; see also Section 5.6 of the present report.

3d: A systematic treatment of uncertainty has been adopted

The body of information that is contained within the assessment basis includes information gathered from a variety of sources, and the associated uncertainty represents a mixture of epistemic and aleatory types. Epistemic uncertainty is reduced, as far as possible, e.g., by site characterisation, research, and design optimisations. Remaining uncertainties are then identified and, if possible, quantified, e.g., by specifying either most likely values and ranges or probability density functions (PDFs) for associated parameters. The management of these uncertainties is a feature of all processes within the safety assessment methodology. Uncertainty in the broad evolution of the pillars of safety is handled by defining and evaluating a reference safety scenario, a set of alternative safety scenarios and a set of future human action (FHA) safety scenarios. Uncertainty in models used in both performance assessment and in the analysis of radiological consequences, where this may have an impact on results, is, generally, handled using simplifying, conservative

assumptions. The use of conservatism (see Section 9.3 for specific examples) implies that the actual performance of the repository will, in reality, be more favourable than that evaluated in quantitative analyses. Both deterministic and probabilistic techniques are used to handle parameter uncertainty. Hypothetical performance assessment scenarios and “what-if?” cases have also been systematically defined and analysed within performance assessment and the analysis of radiological consequences. These involve extreme and hypothetical assumptions, primarily aimed at demonstrating the robustness of the repository system. Such analyses may also pre-empt potential criticism that the selected ranges of parameter values are too narrow or that some detrimental FEPs are either unknown or have been overlooked.

The treatment of uncertainty is described in detail in Chapter 3 of NTB 24-19 (Nagra 2024t) and in Section 4.4 of the present report.

3e: Measures have been adopted to ensure the inclusion of potentially relevant phenomena

Understanding of the initial state and post-closure evolution of the repository and of the characteristics and evolution of the site has been developed and documented iteratively over many years, informed by Nagra’s extensive RD&D programme, and documented in peer-reviewed reports. The comprehensiveness of this understanding is further assured by effective information exchange among safety assessors, technical experts within Nagra, and the broader scientific community. In parallel, a FEP database (supplementary volume of NAB 24-20 Rev. 1, Nagra 2024l) has been developed independently of the safety assessment, and reviewed by both internal and external experts, drawing on similar databases developed internationally. A FEP audit has been conducted to verify the inclusion of all relevant FEPs in the safety assessment. The audit evaluates whether all FEPs in the database are adequately addressed in the safety assessment, either through inclusion in safety scenarios or via explicit or implicit consideration in performance assessment models. Excluded FEPs are shown as outside the assessment scope or irrelevant, ensuring none are overlooked without justification.

The assurance of comprehensiveness and the FEP audit are discussed further in Sections 4.5 and 5.5 of the present report. The FEP database and FEP audit are documented in a supplementary volume of NAB 24-20 Rev. 1 (Nagra 2024l).

10.4 Favourable findings of the safety assessment

Performance assessment analyses the evolution of individual system components and of the entire repository with respect to the various safety functions. The analysis of radiological consequences evaluates annual individual dose or risk for a range of safety scenarios, derived from the results of performance assessment. Since Project *Entsorgungsnachweis* (Nagra 2002) the body of knowledge has significantly expanded, and the assessment methodologies have evolved. Despite the advances made in the methodologies and tools used to build the safety case over the past 20 years, there is notable consistency in the safety-related findings. This consistency strengthens confidence in the Opalinus Clay as a suitable host rock for radioactive waste.

Specific lines of argument for the claim that the findings of these analyses confirm that the repository system provides a high margin of safety are summarised in Fig. 10-7 and elaborated in the following paragraphs.

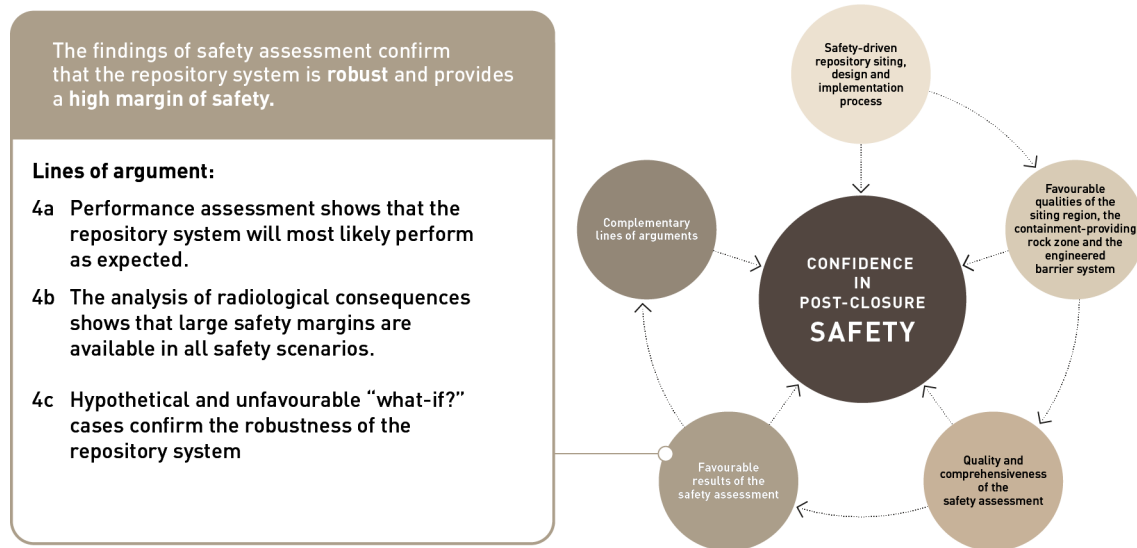


Fig. 10-7: Lines of argument for the claim that the findings of the safety assessment confirm the high margin of safety provided by the repository system

4a: Performance assessment shows that the repository system will most likely perform as expected

In performance assessment, the expected performance of the multi-barrier systems is expressed in terms of a set of claims concerning the contribution of each component of the multi-barrier system and of the overall repository system to the repository safety functions, in line with the current safety concept. Sound arguments are provided for each claim, which are themselves supported by convincing evidence, such that the claims are deemed robust. Overall, the performance assessment shows that, following a rigorous evaluation of the functioning of the evolving disposal system and considering all identified sources of uncertainty, the system will perform its

safety functions and conform to the reference safety scenario in most reasonably foreseeable circumstances. Deviations from the expected performance are captured in other, alternative, safety scenarios.

The performance assessment is synthesised in NTB 24-22 Rev. 1 (Nagra 2024u) and its findings are summarised in Chapter 6 of the present report. The development of safety scenarios, accounting for the results of the performance assessment, is described in Section 4.4 in NTB 24-21 (Nagra 2024e), and the safety scenarios themselves are summarised in the present report in Chapter 7.

4b: The analysis of radiological consequences shows that large safety margins are available in all safety scenarios

The annual individual effective dose is the main “safety indicator” used in the analysis of radiological consequences. The maximum dose rates are summarised in Fig. 8-14, where they are compared with both the regulatory guideline of 0.1 mSv/a and with the range of typical radiation exposure in Switzerland.

The coloured horizontal bars indicate individual calculations, e.g., for different variants within each safety scenario. For each of these variants, maximum dose rates are below both the regulatory guideline and the range of typical radiation exposures in Switzerland, often by several orders of magnitude. The figure also shows the results of a probabilistic uncertainty analysis carried out for the reference safety scenario, and the results of deterministic cases where the most sensitive parameters are set to their most pessimistic bounding values. Even in the most pessimistic cases, the maximum dose rate remains below the regulatory guideline, highlighting the robustness of the results with respect to remaining uncertainty.

The regulatory risk criterion has been applied when analysing excavation of the repository by erosive processes for the time period for assessment, i.e., up to one million years post-closure. The annual risk is found to be more than two orders of magnitude below the risk criterion of 10^{-5} per year and is so low that any impact during this period may be considered, in effect, hypothetical. Beyond one million years, the calculated dose rate is found to be within the range of natural background radiation exposure in Switzerland, which is again in accordance with the regulatory requirements.

The analysis of radiological consequences is documented in detail in NTB 24-18 (Nagra 2024p). Key aspects and results are presented in Chapter 8 and findings are summarised in Section 8.6 of the present report.

4c: Hypothetical and unfavourable “what-if?” cases confirm the robustness of the repository system

Fig. 8-14 also shows the maximum dose rates calculated for “what-if?” cases that involve extreme and hypothetical assumptions. These include the occurrence of hypothetical events, as well as the hypothetical degradation of each of the pillars of safety without any specified cause. Even in these cases, the calculated dose rates remain below the regulatory dose criterion, thus confirming the robustness of the repository system.

The analysis of “what-if?” cases is part of the wider analysis of radiological consequences documented in NTB 24-18 (Nagra 2024p) and is summarised in Chapter 8 of the present report.

10.5 Complementary lines of argument

Finally, lines of argument have been developed that are independent of, and complementary to, the main safety assessment processes. These are summarised in Fig. 10-8 and in the following paragraphs.

5a: Support from studies of natural and archaeological analogues

Indirect support for the safety of deep geological repositories comes from observations of natural and archaeological analogues, including the longevity of uranium ore deposits in many different geological environments around the world, as discussed in more detail in Section 9.1. The low mobility of many radionuclides in such environments and the demonstration that clay rocks provide robust transport barriers are of particular interest. Sedimentary uranium ore deposits and studies on natural or archaeological glasses provide information relevant to the stability of SF and RP-HLW. Indications of slow, diffusion-dominated transport in clay formations is gained from isotope exchange and diffusion studies around the globe. Moreover, numerous examples exist in the oil and gas industry where clay-based seals have been demonstrated to prevent oil and gas transport completely over timescales of thousands to millions of years.

Natural and archaeological analogues, which allow processes and their effects to be observed on large temporal and spatial scales, similar to, or even exceeding, those relevant to safety assessment, provide insights into the durability of system components and into the effective isolation of radioactive materials in a repository. They can showcase the resilience of chosen barrier materials to potentially detrimental repository-induced effects and to long-term degradation processes. Analogues have been identified that yield information on the stability and performance of the geological barrier, the individual engineered barriers, and the interfaces between these barriers.

Natural and archaeological analogues relevant to the proposed repository system are summarised in Section 9.1 of the present report.

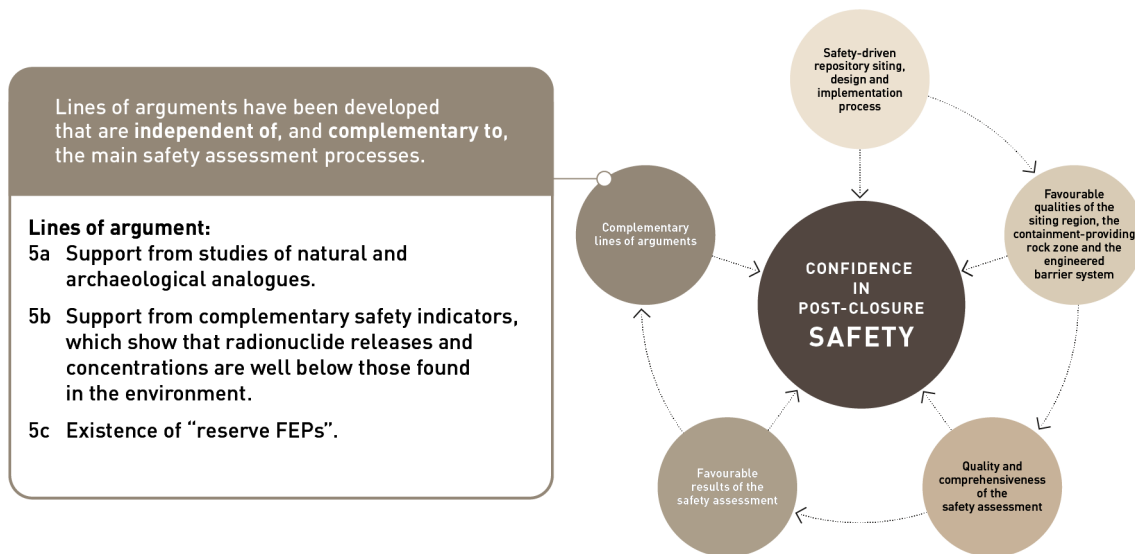


Fig. 10-8: Lines of argument that are independent of, and complementary to, the main safety assessment processes

5b: Support from complementary safety indicators, which show that radionuclide releases and concentrations are well below those found in the environment

Safety and performance indicators complementary to dose and risk provide additional insights into:

- the evolution of the hazard potential of the waste,
- how radiotoxicity (a measure of this hazard potential) is distributed within the repository system and how it diminishes over time due to radioactive decay predominantly within the repository system, with little reaching the biosphere, and
- how the fluxes and concentrations of radiotoxicity are generally much smaller than, or similar to, natural or other man-made radiotoxicity fluxes and concentrations.

Complementary safety and performance indicators are described further in Section 9.2 of the present report. They are computed from the results of radiological consequence analysis and are documented in more detail in Chapter 7 of NTB 24-18 (Nagra 2024p).

5c: Existence of “reserve FEPs”

Conservative assumptions and simplifications are widely used in the analysis of radiological consequences. These include the omission of several “reserve FEPs”, which have the potential, in the future, to provide additional quantitative contributions to the evaluated performance of the disposal system, although many of these would likely have only minor impacts on total dose, even if they were included. Conservative assumptions and simplifications mean that the doses calculated for each of the safety scenarios err on the side of caution, being higher than would be expected in reality. Many examples are provided in Section 9.3 of the present report.

10.6 Conclusions

The integration of the safety arguments presented above further reinforces the findings from Project *Entsorgungsnachweis* (Nagra 2002) and subsequent investigations. The geological barrier is of primary importance due to the excellent properties and long-term geological stability of the Opalinus Clay host rock and the wider containment-providing rock zone (CRZ). The extensive research over the last 20 years conducted by Nagra and other organisations further underlines the suitability of the Opalinus Clay as an outstanding host rock for the disposal of radioactive waste.

A key role of the engineered barriers is the minimisation and mitigation of disturbances to the CRZ. The current safety and repository concept at the chosen site has been shown to be well chosen, with the site itself providing highly favourable and stable properties, as well as significant flexibility for the placement of the repository. The remaining uncertainties have been assessed as insignificant to post-closure safety and are expected to be further reduced when additional underground information becomes available.

Post-closure safety is demonstrated based on the current mature and robust, but still provisional concept and design. In future stages of the waste management programme, up until construction and finally operation, further optimisation of the current safety and repository concept and design will be carried out as discussed in the next chapter.

11 Further development of the post-closure safety case in future programme stages

Aims of the chapter:

- To show how the present post-closure safety case is positioned in the context of the legally required licensing steps required for construction, operation, and closure of the deep geological repository in Switzerland.
- To show that future optimisation steps will be similar to those of other large infrastructure projects, but with specific requirements with regard to post-closure safety.
- To note that, in future stages of the disposal programme, the main drivers of the RD&D programme will be the optimisation processes.

11.1 Waste management programme roadmaps

The Sectoral Plan for Deep Geological Repositories regulates the search for suitable siting regions for deep geological repositories in Switzerland, which is conducted in three stages (BFE 2008). The Sectoral Plan process concludes with the end of Stage 3, i.e., with the approval of the object sheet²⁸ by the Federal Council. At the same time, the Federal Council decides on the granting of the general licence.

After the site has been approved in the general licensing procedure, construction and operation licenses are required for constructing and operating the repository (based on nuclear energy legislation). After completion of the operating and monitoring periods, a permit for closure must be applied for and issued by the Federal Council.

The background for preparing the application for future licences is provided by experience gained in preparing the general licence application, experience from international projects and information on newly developed technologies. However, the following main points are expected:

- **Construction licence:** The construction licence showcases a detailed plan of the future facilities and a plan for closure. The construction licence also specifies the key elements for technical implementation. This includes the test areas, which, together with the main repository and the pilot repositories, form integral parts of the geological repository.

With the framework of preparing the construction licence and its related permits, a stage-appropriate optimisation is carried out for the selection of the engineered barriers (disposal containers, backfilling, sealing) and the design of the underground repository architecture, as well as the technical implementation of the project.

The construction licence application also provides the possibility to optimise and adapt the repository infrastructure.

- **Operating licence:** The operating licence specifies the definitive protection zone²⁹ of the deep geological repository and the permissible capacity of the facility. It also sets out measures for monitoring the environment and the stages of commissioning, the start of which

²⁸ An object sheet is a standardised document used to present detailed information about a specific object in a clear and concise way.

²⁹ The protection zone defines the volume of rock protected from underground activities that could interfere with the CRZ integrity.

requires prior clearance by the regulatory authorities. The operating licence also specifies requirements relating to limits for the activity of the waste to be disposed of. The emplacement of waste packages in a repository requires approval by ENSI (ENSI 2023). The sealing technology for any sealing elements that have not been tested in advance is tested at this point, and their functionality during repository operation is demonstrated. Technology that would allow, if needed, the pre-closure retrieval of waste packages without undue effort must also be demonstrated, including the introduction and removal of backfill materials.

Once the operating licence has been granted, waste emplacement in the repository begins. Safety checks are carried out periodically during operation.

The monitoring period begins once emplacement is complete. The application for decommissioning of the facilities and structures at the surface is also prepared and submitted.

- **Closure of the repository:** After the end of the monitoring period, the Federal Council authorises closure work provided that the permanent protection of humans and the environment can be ensured. After final closure, the Federal Council may order further, temporary monitoring. At this time, or after expiry of the monitoring period, the Federal Council determines that the repository is no longer subject to nuclear energy legislation.

With the application for repository closure, the final decision on the preferred closure variant is made. For this, the repository closure plan and an updated project plan for the monitoring period are to be submitted.

11.2 Repository optimisation and design

Future work planned by Nagra will yield information that will allow further optimisation of the current provisional repository design and implementation plan, e.g., adapting it to the actual site-specific conditions that are encountered underground.

The waste producers have tasked Nagra with implementing safe deep geological disposal for all categories of radioactive waste in Switzerland at reasonable cost and in a timely manner. To fulfil this mandate, the safety of the deep geological repository must regularly be demonstrated to the authorities with a solid technical foundation and transparent documentation in line with the licensing procedure (KEG 2003).

The repository project must be realised within the framework of a holistically oriented optimisation process that is aligned with the following overarching optimisation goals and principles for post-closure safety and implementation:

- **Ensuring post-closure safety** for the protection of humans and the environment from ionising radiation without undue burdens for future generations.
- **Ensuring safety and feasibility in connection with construction and operation;** this includes occupational health and safety. Additional topics include security and safety in the case of incidents or accidents in terms of their impact on humans and the environment. Feasibility is defined as achieving reliable implementation of all required work steps with reasonable effort.
- **Considering resources and economic efficiency,** i.e., realising the project cost-efficiently and within an appropriate period.

While all major infrastructure projects must observe the latter three overarching optimisation goals and principles, the first refers specifically to deep geological disposal. This is highlighted in both ENSI Guideline G03 (ENSI 2023) and international standards as e.g., the IAEA Safety Standard SSR-5 (IAEA 2011a).

Optimisation is carried out regarding all optimisation goals and principles. When evaluating potential variants and options, their proportionality in terms of time and costs, as well as the opportunities and risks that each of these present, are thoroughly considered. This includes an evaluation of the options available for increasing system robustness as well as future developments. Conflicting requirements are balanced. In individual cases, it may be necessary to determine which solution is the most advantageous overall with regard to safety (ENSI 2023).

11.3 RD&D roadmaps

In future stages of the disposal programme, the main drivers of the RD&D programme will be the need for optimisation, which must be carried out from the perspective of safety (the updating of the safety cases at each milestone), feasibility of implementation (confidence underpinning the feasibility of implementing the design at each milestone) and resources (assessing cost efficiency up to completion). In anticipation of the construction and operating licences, technological developments are also foreseen. These technological developments will affect all barrier components as well as their implementation.

11.4 Anticipated evolution of the safety case

The approval of future licences will necessitate considering those safety aspects pertinent to the corresponding licence under application. This will in turn determine how a safety case is further refined and presented.

For the general licence application, the repository and the barriers have been developed to the level of a conceptual design. However, forthcoming licences will demand progressively more detailed technical specifications. Along with a steadily improving understanding of the geological environment, e.g., due to construction activities, and along with technological progress and confirmation of engineering solutions, e.g., through testing facilities, the repository project will become increasingly specific. In parallel, options for optimisation will be reduced. This evolution will be reflected in the increasing level of detail at which arguments are discussed in the safety case.

Moreover, future safety cases will be expanded beyond post-closure safety, to include, in particular, the case for operational safety. This is necessary to meet the needs of IAEA SSR-5 (IAEA 2011a), which states that a safety case needs to consider both the operational and post-closure phases. It is necessary to ensure that actions during operation will not be detrimental to post-closure safety. Furthermore, operational aspects will be important for future design optimisation.

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13 Glossary

Term	Definition
Alternative safety scenarios	Alternative safety scenarios arise from the potential for deviations from the expected evolution and performance of the <i>repository system</i> that have been identified in the <i>performance assessment</i> . They postulate a degraded functionality of one or more of the main components of the repository system compared with that assumed in the <i>reference safety scenario</i> . They include unlikely, or highly unlikely, but scientifically non-excludable evolution of the repository system and its environment in which the assumptions made in the reference scenario are significantly changed.
Analysis of radiological consequences	The use of quantitative models to analyse radionuclide release, retention and transport to the surface environment, which results in an evaluation of radionuclide release rates to the surface environment as a function of time, based on the assumed transport properties of the near field and geosphere. The release rates are then converted into annual individual effective doses, annual individual risks and complementary <i>safety indicators</i> . The outcomes are then compared with regulatory protection criteria (and other suitable measures) and the safety margin for each <i>calculation case</i> is evaluated.
Assessment basis	The methodologies, assessment tools, databases and all other evidence and knowledge developed or acquired in support of the <i>safety assessment</i> . Key elements of the assessment basis include the general scientific understanding of the waste inventory and engineered barriers, an integrated understanding of the <i>containment-providing rock zone</i> and its geological setting, and an understanding of the phenomenological evolution of the repository. The assessment basis also includes the assessment methodologies, and the models, codes and data used in the analyses.
Assessment perspectives	In the context of the qualitative evaluation of the <i>containment-providing rock zone</i> (CRZ) during site comparison, assessment perspectives are the different viewpoints of the siting criteria (classified into four groups), from which the CRZ should be evaluated in terms of safety and engineering feasibility.

Term	Definition
Calculation cases	Calculation cases define specific calculations that are carried out to analyse the radiological consequences of a particular safety scenario or “ <i>what-if?</i> ” <i>case</i> for which a conceptual model is formulated. A calculation case consists of a mathematical model and a set of input parameter values, solved using one or more computer codes or analytical solutions. Each safety scenario or “what-if?” case will typically have one or more calculation cases associated with it.
Combined repository	A repository comprising <i>disposal areas</i> for both HLW and L/ILW in the same siting region.
Compartmentalisation	Partitioning of waste, e.g., in different sections of the repository, through a designated layout of the repository and the installation of seals and backfill.
Complementary lines of argument	Argumentation that supports the <i>safety assessment</i> , in addition to those arguments developed in <i>performance assessment</i> , <i>safety scenario development</i> and the <i>analysis of radiological consequences</i> . This includes arguments based on natural and archaeological analogues, <i>performance and safety indicators</i> complementary to dose and risk, and <i>conservatism</i> in the calculations of <i>radiological consequence analysis</i> , including the existence of reserve FEPs.
Conservative / conservatism	Simplifications, assumptions, parameter values etc., that are confidently expected to lead to consequences that are less favourable than those anticipated under real conditions, taking uncertainty into account. Conservative choices can lie outside the range of what is physically possible (e.g., hypothetical assumptions). Conservatisms include the deliberate omission from quantitative analysis of some phenomena that are considered likely and beneficial to safety because suitable models, codes or databases are unavailable.
Containment-providing rock zone (CRZ)	The volume of rock that, together with the engineered barriers, ensures the containment and retention of radioactive substances contained in the waste during the <i>time period for assessment</i> considering the expected evolution of the deep geological repository. It comprises the host rock as well as the upper and lower barrier-effective confining geological units.
Defence-in-depth	The principle through which safety is defended at various levels, with layers of overlapping provisions so that, if a deficiency should occur in one layer it would be compensated for or corrected by others. This includes the <i>multi-barrier system</i> and how its <i>safety functions</i> ensure a robust <i>repository system</i> .

Term	Definition
Demonstration of post-closure safety	The demonstration of safety brings together the evidence, arguments and analyses of the three main processes of <i>safety assessment</i> (<i>performance assessment</i>, <i>safety scenario development</i> and <i>analysis of radiological consequences</i>), together with <i>complementary lines of argument</i>, to form the overall post-closure <i>safety case</i>. The demonstration comprises: • the safety-driven repository siting, design and implementation process, • the favourable qualities of the siting region, the <i>containment-providing rock zone</i> and the engineered barrier system, • the quality and comprehensiveness of the safety assessment, • the favourable results of the safety assessment, • complementary lines of argument.
Design requirements	Requirements placed on the design of system components such that they are expected to fulfil their respective <i>performance objectives</i> and <i>component-specific functions</i>. These design requirements are developed in association with the repository design and system analysis as part of an iterative design development process. For the <i>safety assessment</i> in support of the general licence application milestone, this process leads to the <i>provisional design and implementation plan</i>, for which safety is assessed.
Design specifications	Solutions to the <i>design requirements</i>, concerning technical, quality and organisational aspects of individual components or the whole <i>repository system</i>.
Deviation from expected performance	Occurs when an individual barrier component, or the repository as a whole, does not provide its intended <i>safety functions</i> or <i>component-specific function</i>. This may be quantitatively assessed using <i>performance indicators</i> evaluated against <i>performance targets</i>. The importance of such a deviation can depend on its relevance to safety and on the likelihood of occurrence.
Disposal area	Underground body of rock that encloses a group of adjacent <i>emplacement rooms</i>, including a minimum distance around these emplacement rooms. In addition to the emplacement rooms, a disposal area also has branch tunnels, unloading areas and sealing structures.
Emplacement rooms	Generic term for HLW emplacement drifts (HLW repository or repository section) and L/ILW emplacement caverns (L/ILW repository or repository section) for the long-term emplacement of radioactive waste.

Term	Definition
Envelope scenarios	Scenarios that cover potential lines of evolution.
Expected performance (of the multi-barrier system)	The situation when the <i>multi-barrier system</i> of the repository provides its intended <i>safety functions</i> and <i>component-specific functions</i> , both at the component level and at the total system level, and for which all <i>performance assessment</i> claims are expected to be met. The repository is designed such that, for the expected evolution (see <i>reference safety scenario</i>), it performs as required.
FEPs	<i>Features, events and processes</i> . Internal or external phenomena (or factors) that can either define the <i>repository system</i> or influence its evolution or radiological consequences.
Component-specific functions	A specific set of properties of, or actions to be performed by, one or more system features that contribute directly (or indirectly) to the <i>safety functions</i> .
Future human actions (FHA)	Potential actions, activities, behaviours and lifestyles undertaken by human society in the long term which can impact repository safety, particularly regarding human intrusion into the repository (e.g., by drilling deep borehole). Future human actions safety scenarios consider actions which appear credible from the viewpoint of today's society, i.e., considering current human activities in the regional vicinity of the site combined with state-of-the-art technology (e.g., drilling equipment and techniques etc.).
Matrix (waste)	In the case of L/ILW and RP-HLW, this comprises the solidified waste and the material in which it is embedded. In the case of spent fuel, this comprises the crystalline phases which make up the UO ₂ or mixed oxide fuel pellets.
Multi-barrier system	A system of staged, passive engineered and geological barriers that fulfil the overarching <i>safety functions</i> . In the context of the present <i>safety case</i> , the multi-barrier system includes engineered barriers consisting of the waste <i>matrix</i> , disposal canisters, backfill and seals, as well as the <i>containment-providing rock zone</i> as a geological barrier.
Optimisation of protection	A principle of radiological protection to keep radiological exposure as low as reasonably achievable (ALARA), which is also applied throughout the stepwise and iterative development of the repository design in view of post-closure safety.

Term	Definition
Performance assessment	An analysis of the thermal, hydraulic, mechanical and chemical evolution of the <i>repository system</i> . Performance assessment verifies whether the expected evolution of the repository system, as provided by the <i>assessment basis</i> , is indeed the expected evolution based on the arguments and evidence available. Performance assessment also identifies <i>deviations from the expected performance</i> that could degrade the functionality of the components of the repository system or of the repository system as a whole, as inputs to <i>safety scenario development</i> .
Performance assessment scenarios	Performance assessment scenarios address features, events and processes that could potentially degrade the repository barriers and their <i>safety functions</i> . They permit a quantitative assessment of the impact of <i>deviations from expected performance</i> on the performance of the <i>repository system</i> as a whole and enable the consideration of uncertainties related to the evolutionary path. This provides information on their relevance to safety, which facilitates the identification of those deviations that require an in-depth analysis of the radiological consequences.
Performance indicators	Quantitative metrics of the performance of individual or multiple system components that permit evaluation against a given <i>performance target</i> .
Performance objectives	Performance objectives describe how the system components, including both engineered and geological barriers, perform their individual, <i>component-specific functions</i> and thus, collectively, the <i>safety functions</i> . They are a part of the current <i>safety and repository concept</i> relating to the <i>pillars of safety</i> , safety functions and <i>component-specific functions</i> , that guide the design development process to achieve the <i>provisional design and implementation plan</i> .
Performance targets	Specified requirements for the performance of individual or multiple system components, or of the total system. A target value is set for a relevant <i>performance indicator</i> to allow quantitative evaluation of whether the <i>expected performance</i> is met, including the assessment of uncertainties.
Pessimism / pessimistic	The use of an assumption or parameter value that is possible but leads to consequences that are less favourable than the most likely consequences. The use of such a value means that not every single parameter variation needs to be carried forward into a <i>calculation case</i> to demonstrate safety. While both pessimism and <i>conservatism</i> lead to calculated consequences that are less favourable than those that are most likely, <i>conservatism</i> includes the use of hypothetical assumptions or parameter values and results in consequences that are less favourable than the most likely consequences, including uncertainty.

Term	Definition
Pillars of safety	The main features or characteristics of the current repository concept that contribute to the <i>safety functions</i> of the <i>multi-barrier system</i> which, together, ensure post-closure safety. These are: • deep underground location of the waste emplacement rooms in a stable geological environment • <i>Containment-providing rock zone</i> • Closure system (backfill and seals) • Spent fuel and RP-HLW <i>matrix</i> • Spent fuel and RP-HLW canisters • Bentonite buffer of the HLW emplacement drifts • Cementitious L/ILW near field. Each of the pillars of safety makes a direct and significant contribution to at least one of the safety functions but may also make indirect or more minor contributions to other safety functions.
Provisional design and implementation plan	The provisional layout, material choices and dimensions of all components of the repository, and the sequence of operations for repository construction, operation and closure. It results from the iterative design development process that is informed by the current <i>safety and repository concept</i>, system analyses and other design considerations, such as engineering practicality and operational safety.
Reference safety scenario	A general description of the initial state of the repository barriers, their expected evolution over time (including the release, retention and transport of radionuclides within the <i>repository system</i>), and the assumptions made regarding the biosphere for the <i>analysis of radiological consequences</i>. Essentially, in the reference safety scenario, the repository system will perform as expected and the <i>pillars of safety</i> are assumed to provide their assigned functions, in accordance with the current <i>safety and repository concept</i>.
Repository section	In the repository, in addition to the main HLW and L/ILW <i>disposal areas</i>, the HLW and L/ILW repository sections each include a pilot facility, in which the behaviour of the wastes, the backfill, the seals and the host rock is monitored until the end of a monitoring period.
Repository system	The repository system comprises the underground structures, the emplaced waste, the installed engineered barriers, and the <i>containment-providing rock zone</i>.

Term	Definition
Requirements	Conditions that must be fulfilled or attributes that must be possessed by the <i>repository system</i> or its components to achieve certain objectives. Requirements are organised hierarchically. At the highest level, these are overarching objectives and principles, including those specified in Swiss legislation and regulations. Low-level requirements are more detailed and specific and support the fulfilment of the overarching requirements.
Reserve FEPs	<i>Features, events and processes</i> considered likely to occur and beneficial to safety that are deliberately excluded from quantitative analysis because the level of scientific understanding is insufficient to support quantitative modelling, or because suitable models, codes or databases are unavailable. Reserve FEPs can be mobilised at a later stage of repository planning if the level of scientific understanding is sufficiently enhanced, and the necessary models, codes and databases are developed. The existence of reserve FEPs constitutes an additional, qualitative argument for reserves of safety beyond those indicated by the quantitative analysis.
Robustness principle	The principle of siting and designing the <i>repository system</i> such that it is insensitive to detrimental processes and events and that post-closure safety can be demonstrated irrespective of the remaining uncertainty.
Safety analysis (site-specific)	A synonym for <i>analysis of radiological consequences</i> in the context of site-specific analyses performed within the framework of site comparison. As for the more detailed <i>safety assessment</i> in support of the general licence application, this analysis also comprises a systematic and quantitative analysis approach, including a <i>reference scenario</i> and <i>alternative safety scenarios</i>, but does not include <i>future human action</i> safety scenarios or “<i>what-if?</i>” cases.
Safety assessment	The process of gathering the claims, arguments, evidence and performing associated analyses, regarding the safety of the <i>repository system</i> and its environment during the post-closure phase. It is the means by which the <i>safety case</i> is developed. Safety assessment includes <i>performance assessment</i>, <i>safety scenario development</i>, <i>analysis of radiological consequences</i> and the <i>demonstration of post-closure safety</i>.

Term	Definition
Safety assessment methodology	A description of how the four processes of <i>safety assessment</i> (<i>performance assessment</i>, <i>safety scenario development</i>, <i>analysis of radiological consequences</i> and <i>demonstration of post-closure safety</i>) are implemented to provide the <i>safety case</i>. Thus, it explains how the four processes of safety assessment are related to each other and how information flows between them. The safety assessment methodology is considered part of the <i>assessment basis</i>.
Safety case (post-closure)	The synthesis of claims, arguments and supporting evidence gathered, and analyses carried out, in the course of the <i>safety assessment</i> to make the case for the post-closure safety of a <i>repository system</i>.
Safety and repository concept	The safety and repository concept explains how, through the safety barriers and the <i>safety functions</i> they perform, the deep geological repository ensures the protection of humans and the environment. It provides information about the basic principles of how the deep geological repository is designed. This includes, for example, the concept of the <i>multi-barrier system</i> or the concept of storing disposal containers in long, layer-parallel <i>emplacement rooms</i> in a suitable host rock. Additionally, it includes the <i>performance objectives</i>, which describe how the system components, including both engineered and geological barriers, are envisioned to perform their individual, <i>component-specific functions</i> and thus, collectively, the <i>safety functions</i>.
Safety functions	Safety functions are the roles of the <i>multi-barrier system</i> that together ensure post-closure safety. Engineered and geological barriers and other system components contribute to one or more of these functions. They consist of: • S1: Isolation of radioactive waste from the surface environment. • S2: Complete containment of radionuclides for a period of time. • S3: Immobilisation, retention and slow release of radionuclides. • S4: Compatibility of the multi-barrier system elements and the radioactive waste types among each other and with other materials. • S5: Long-term stability of the multi-barrier system with respect to long-term geological and climatic processes.

Term	Definition
Safety indicators	Quantitative metrics of post-closure safety used for the <i>analysis of radiological consequences</i> . These include the annual individual effective dose and annual individual risk as the primary safety indicators, and radioactivity fluxes and radiotoxicity as complementary safety indicators.
Safety scenarios	The different possibilities for the initial state of key features, conditions and processes of the <i>repository system</i> and the ways in which they may evolve and perform their <i>safety functions</i> over time. They capture uncertainty in the broad ways the system may evolve, considering all relevant sources of uncertainty, and include: 1. a <i>reference safety scenario</i>, corresponding broadly to the expected evolution, 2. a set of <i>alternative safety scenarios</i>, arising from conceivable <i>deviations from the expected performance/evolution</i> 3. a set of <i>future human actions</i> safety scenarios. Note that “<i>what-if?</i>” cases are not defined as safety scenarios, although they are an output of <i>safety scenario development</i>.
Safety scenario development	The identification and description of a set of safety scenarios that capture uncertainty in the initial state of the <i>repository system</i> and the different ways in which it could evolve over time. Based on the knowledge provided by the <i>assessment basis</i> and the findings of the <i>performance assessment</i>, safety scenario development enables a systematic assessment of how key features, events and processes can impact post-closure safety. Safety scenario development also leads to a set of additional “<i>what-if?</i>” cases that are used to demonstrate robustness of the repository system.
Safety scenario variants	Variants of a given safety scenario (<i>reference, alternative or future human action</i>) that share largely the same general description of <i>features, events and processes</i> occurring over time, but may differ, for example, in the extent or timing of key phenomena associated with that safety scenario as well as positioning or properties of a feature. In contrast with <i>alternative safety scenarios</i> .
Scenario analysis	With respect to the regulatory guidance, this refers to a combination of (in Nagra’s terminology) <i>safety scenario development</i> and <i>radiological consequence analysis</i> for the defined <i>calculation cases</i> .
Sensitivity analysis	Technique to investigate how the variance in modelling results is impacted by the variance in input parameters. Thus, it identifies the parameters to which the computational model outcomes are most sensitive.

Term	Definition
Stylised approach	An approach that involves imposed, rather than scientifically derived, assumptions for handling poorly quantifiable or unquantifiable uncertainties in scenario development. This approach aims to limit arbitrary speculation on matters that are inherently unpredictable. Thus, no attempt is made to cover the full range of possible scenarios or to assign probabilities to them. Rather, a limited set of illustrative cases are analysed. In agreement with the regulations and international standards (e.g., SSG-23, IAEA 2012, see also ENSI Guideline G03, ENSI 2023), stylisation is used by Nagra for aspects of biosphere evolution, <i>future human actions</i> and lifestyles and for developing human intrusion scenarios.
Time period for assessment	The time frame over which assessment of post-closure safety is performed. For a repository for both HLW and L/ILW in the NL siting region, a 10⁶-year assessment period is considered. Consistent with the regulatory requirement that dose and risk calculations should extend up to the time of maximum radiological impact of the repository, dose calculations are generally performed through to 10⁷ years for HLW and 10⁶ years for L/ILW. Analyses for volatile radionuclides that focus on ¹⁴C, with a 5,730-year half-life, are carried out for a 10⁵-year period.
Time period under consideration	The time frame over which <i>safety analysis</i> calculations are performed in the frame of the site comparison in Stage 3 of the Sectoral Plan. This is defined as extending up to 10⁵ years for an L/ILW repository and up to 10⁶ years for an HLW repository.
Uncertainty analysis	Quantification of the influence of uncertainties in data, processes and model assumptions on calculation results.
Waste packages	Generic term for SF/RP-HLW disposal canisters and L/ILW disposal containers.
“What-if?” cases	“What-if?” cases are <i>calculation cases</i> that use a particular set of assumptions and/or parameter values that lie outside the range of possibilities supported by currently available evidence and knowledge. Such cases are primarily aimed at demonstrating the robustness of the <i>repository system</i> or to improve understanding of the system performance (e.g., by hypothetically removing a barrier or favourable process). Additionally, phenomena that could realistically occur, but whose occurrence is implausible during the one-million-year <i>time period for assessment</i>, can also be considered as a “what-if?” case.

App. A Safety case in the framework of Project *Entsorgungsnachweis* and the General Licence Application – a comparison

The safety case built in the frame of Project *Entsorgungsnachweis* (Nagra 2002) and the present report presenting the safety case for the general licence application are compared in Tab. A-1. Regardless of the methodology and tools used to build the safety case, there is a notable consistency in the findings, which strengthens confidence in Opalinus Clay as a suitable host rock for radioactive waste disposal. The table below is an addition to Section 1.2.

Tab. A-1: Safety case in the framework of Project *Entsorgungsnachweis* and the General Licence Application – a comparison

Safety case in Project *Entsorgungsnachweis* and in the General Licence Application – a comparison

Purpose of the safety case

Both Project *Entsorgungsnachweis* and the current safety case for a repository for both HLW and L/ILW at the Haberstal site have demonstrated the post-closure safety provided by a deep geological repository from the hazards presented by the waste.

Safety concept

The current safety concept, i.e., NAB 24-18 Rev. 1 (Nagra 2024s), is similar in Project *Entsorgungsnachweis* and in the present safety case. However, the definitions of the pillars of safety and the ways in which they contribute to the safety functions of the repository have been further developed and clarified.

Repository layout and engineered barrier design

Although the safety concept is largely the same, the details of the engineered barrier system and repository design have evolved since Project *Entsorgungsnachweis*. Of particular significance to safety assessment are:

- The use of dead-end emplacement rooms for both HLW and L/ILW, compared with the tunnels with access from two sides used for the disposal of HLW in Project *Entsorgungsnachweis*, which provides additional assurance of low water flows along the emplacement rooms.
- The closure system, including the system of repository seals and backfill, has been revised to ensure, for example, that there are no safety-relevant interactions between the HLW and L/ILW repository sections, and to ensure adequate storage volume for repository-generated gas, avoiding build-up of potentially damaging gas pressures.
- In Project *Entsorgungsnachweis*, the use of a concrete liner in the HLW emplacement tunnels was assumed to be unlikely and the possibility was not accounted for in the safety assessment. In the present provisional design, a concrete liner is now foreseen for the HLW drifts, as well as for the L/ILW emplacement caverns. This has motivated a more detailed evaluation of cement – clay interactions.

Safety case in Project *Entsorgungsnachweis* and in the General Licence Application – a comparison

- In Project *Entsorgungsnachweis* access would be provided, during construction and operation, by a spiral ramp. A vertical shaft, also used for construction during enlargement of the facility, would provide ventilation and the required emergency escape facility.
-

Safety arguments

The arguments for the post-closure safety, including arguments for both the intrinsic quality of the system and the quality of the safety assessment, are essentially the same in the present safety case as they were in Project *Entsorgungsnachweis* but are better supported by an improved assessment basis and a more thorough safety assessment, as indicated above.

Assessment basis (general)

Since Project *Entsorgungsnachweis*, the assessment basis has benefited from several further years of RD&D and site characterisation work by Nagra, providing a firmer basis for the arguments supporting the present safety case, including experiments, models and databases used to assess the performance of repository components, the repository system as a whole and the radiological consequences of safety scenarios (see below). The FEP database, while still serving primarily as a completeness check, has also been extensively reviewed, revised and reorganised.

Waste inventory and repository type

Project *Entsorgungsnachweis* considered a repository for the disposal of SF, HLW and long-lived ILW (primarily resulting from fuel reprocessing) at the “Zürcher Weinland” site. It was assumed that there would be a separate repository for the disposal of L/ILW including ATW arising from the operation and decommissioning of Swiss NPPs, from medicine, industry and research and from those operations in reprocessing that produce only low-level technological waste. The present safety case considers a repository for both HLW and L/ILW including ATW at the Haberstal site.

Geological understanding of the site / siting region

Project *Entsorgungsnachweis* basically relied on information from one deep borehole at Benken and a 3D seismic survey of the ZNO siting region. For the post-closure safety case presented here 4 deep boreholes in NL and 8 more in the other siting regions provide a wealth of information on rock properties. Additional 2D and 3D seismic surveys complete the picture. Drilling into the quaternary sediments adds information on the most recent geological evolution of Northern Switzerland for better understanding of possible future evolutions. A more detailed comparison of the two geosyntheses can be found in NTB 24-17 (Nagra 2024i).

Safety case in Project *Entsorgungsnachweis* and in the General Licence Application – a comparison

Reporting structure

A reporting structure has been adopted for the present safety case that reflects the safety assessment methodology. This includes, in addition to the present post-closure safety report which is a synthesis report of the overall safety case, dedicated reports on performance assessment, safety scenario development and the analysis of radiological consequences. In Project *Entsorgungsnachweis*, the performance assessment (then termed “system evolution”), safety scenario development (then termed “the safety concept and the identification of assessment cases”) and the analysis of radiological consequences (then termed “evaluation of the performance of the disposal system”) were documented primarily as chapters of the main post-closure safety report (Nagra 2002), though with many supporting reports including a report on FEP management for the Opalinus Clay safety assessment (Nagra 2003). The current structure allows these safety assessment processes to be documented in more depth, while keeping the post-closure safety report itself concise and accessible to a general technical audience.

Safety assessment methodology

The main processes of safety assessment, i.e., performance assessment, safety scenario development and the analysis of radiological consequences, feature in both Project *Entsorgungsnachweis* and in the present safety case. However, workflows for each of these processes and the interfaces between them have been further developed and illustrated diagrammatically. Specific developments include:

- The explicit link that is now made between the development of the repository concept and design and the safety assessment processes.
 - The greater emphasis on performance assessment as a key element of the methodology.
 - The more systematic performance assessment methodology, based around a set of claims, arguments and evidence at the system-component level and at the total system level, with each claim supported, where possible, by wide ranging arguments and evidence and linked to one or more safety functions.
 - The definition and use of quantitative performance indicators, indices and targets to assess the potentially detrimental phenomena and uncertainties and the margins by which the safety functions are provided, in support of safety scenario development.
 - The more extensive and systematic use of probabilistic methods for uncertainty assessment in both the performance assessment and in the analysis of radiological consequences.
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Safety case in Project *Entsorgungsnachweis* and in the General Licence Application – a comparison

Safety scenarios

While the safety scenarios of both the Project *Entsorgungsnachweis* and the present safety case cover the potential evolution paths of the repository, their exact formulation differs, accounting for the more profound assessment basis and the optimised safety assessment methodology. Specific differences include:

- The inclusion of gas-mediated radionuclide transport in the evaluation of all safety scenarios (in Project *Entsorgungsnachweis*, gas-mediated transport was considered as an alternative safety scenario).
 - Alternative safety scenarios solely linked to uncertainty in geological characterisation and evolution, namely linked to either hydraulically active confining geological units or to an undetected or reactivated fault.
 - The more systematic definition of hypothetical “what-if?” cases that test system robustness, based on the degraded performance of each of the pillars of safety.
 - More profound assessments of the “what-if?” case of excavation of the repository by glacial and non-glacial erosion in the far future.
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Process understanding and models

Improved process understanding and greater characterisation of the geological setting has been accompanied by a more thorough analysis of these features and processes in both the performance assessment and in the analysis of radiological consequences. Examples include:

- More profound characterisation and deeper process understanding for radionuclide retention and transport in the Opalinus Clay and the confining geological units.
 - More detailed and comprehensive THM modelling, allowing a better understanding of the rate of repository-generated gas and of the impact of repository-generated heat on the surrounding barriers.
 - A thorough examination of cement – clay interactions, including the development of arguments that such interactions will not lead to pore clogging or other detrimental effects.
 - The inclusion of the confining geological units in all radionuclide transport calculations (in the Reference Conceptualisation for Project *Entsorgungsnachweis*, the geological barrier was conservatively restricted to a transport path segment through the unperturbed host rock).
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App. B Key phenomena potentially relevant to repository evolution

The key phenomena that are potentially relevant to the evolution and performance of the repository system are considered in the performance assessment including those phenomena that have the potential to lead to deviations from expected repository performance.

These phenomena are identified in NAB 24-20 Rev. 1 (Nagra 2024m), which synthesises Nagra's understanding of the initial state and expected post-closure evolution of the repository. As noted in Section 4.5, this understanding has developed iteratively over many years, with each iteration peer-reviewed and informed by Nagra's extensive RD&D programme. The key phenomena identified in NAB 24-20 Rev. 1 (Nagra 2024m), their relevance to system evolution and performance, their potential to lead to performance deviations are listed in Tab. B-1 where the chapters of the performance assessment synthesis of NTB 24-22 Rev. 1 (Nagra 2024u) indicate where these phenomena are addressed.

Tab. B-1: Key phenomena potentially relevant to repository evolution

Key processes and/or phenomena potentially relevant to repository evolution		Relevance to evolution and performance (analysed in PA)	Potential performance deviations	Addressed in:
Brittle fracturing, shear failure and reactivation of pre-existing features in the drift and cavern near field (EDZ formation)	Table 5-1 of NAB 24-20 Rev. 1 (Nagra 2024m)	EDZ formation/ hydro-mechanical evolution	No potential for performance deviations identified	Included in Chapter 3 of NTB 24-22 Rev. 1 (Nagra 2024u) / assessment by component
Unloading of the rock and consequent changes in volumetric strain and pore pressure				Included in Chapter 3 of NTB 24-22 Rev. 1 (Nagra 2024u) / assessment by component
EDZ development due to pore pressure recovery as soon as tunnel support is emplaced				Included in Chapter 3 of NTB 24-22 Rev. 1 (Nagra 2024u) / assessment by component
Loading of the lining and convergence of the rock external to the lining as soon as the pore pressure recovers				Included in Chapter 3 of NTB 24-22 Rev. 1 (Nagra 2024u) / assessment by component
Onset of EDZ formation (L/ILW)	Table 6-2 of NAB 24-20 Rev. 1 Nagra (2024m)			Included in Chapter 3 of NTB 24-22 Rev. 1 (Nagra 2024u) / assessment by component

Key processes and/or phenomena potentially relevant to repository evolution		Relevance to evolution and performance (analysed in PA)	Potential performance deviations	Addressed in:
Swelling of buffer and possible re-compaction of EDZ (after degradation of rock support)	Table 7-1 of NAB 24-20 Rev. 1 Nagra (2024m)	EDZ formation/ hydro-mechanical evolution	No potential for performance deviations identified	Not explicitly included
Thermal heave and subsequent settlement				Included in Chapter 4 of NTB 24-22 Rev. 1 (Nagra 2024u) / total system analysis
Desaturation of the EDZ during active ventilation and cooling of the drifts and caverns	Table 5-1 of NAB 24-20 Rev. 1 Nagra (2024m)	Thermal and hydraulic evolution and repository-generated gas	Gas- or thermally-induced faulting or fault re-activation	Included in Chapter 4 of NTB 24-22 Rev. 1 (Nagra 2024u) / total system analysis
Initial heating of the bentonite buffer and redistribution of moisture due to heating (local drying close to canister, and diffusive and convective transport of water vapour possible)				Included in Chapter 4 of NTB 24-22 Rev. 1 (Nagra 2024u) / total system analysis
Resaturation of the (L/ILW) EDZ and self-sealing: self-sealing as water reaches any open fractures but also development of (or changes in) the EDZ due to loading, pressure recovery and resaturation (loss of apparent cohesion)				Included in Chapter 4 of NTB 24-22 Rev. 1 (Nagra 2024u) / total system analysis
Onset of (HLW) EDZ and buffer saturation	Table 6-1 of NAB 24-20 Rev. 1 Nagra (2024m)			Included in Chapter 4 of NTB 24-22 Rev. 1 (Nagra 2024u) / total system analysis

Key processes and/or phenomena potentially relevant to repository evolution		Relevance to evolution and performance (analysed in PA)	Potential performance deviations	Addressed in:
Self-sealing of EDZ due to saturation	Table 6-1 of NAB 24-20 Rev. 1 Nagra (2024m)	Thermal and hydraulic evolution and repository-generated gas	Gas- or thermally-induced faulting or fault re-activation	Included in Chapter 3 of NTB 24-22 Rev. 1 (Nagra 2024u) / assessment by component
Temperature increases in buffer, EDZ and host rock due to thermal output from canister				Included in Chapter 4 of NTB 24-22 Rev. 1 (Nagra 2024u) / total system analysis
Gas generation and pressure build-up due to anaerobic corrosion of metals	Table 6-2 of NAB 24-20 Rev. 1 Nagra (2024m)			Included in Chapter 4 of NTB 24-22 Rev. 1 (Nagra 2024u) / total system analysis
Partial decrease in oxygen partial pressure				Not explicitly included but conservatively accounted for in the calculations
Temperature peaks in EDZ and host rock due to reducing power output from canister. Cooling of buffer and EDZ with reduction in thermal stresses	Table 7-1 of NAB 24-20 Rev. 1 Nagra (2024m)			Included in Chapter 4 of NTB 24-22 Rev. 1 (Nagra 2024u) / total system analysis
Ongoing saturation of the EDZ and buffer. Self-sealing of EDZ due to resaturation				Included in Chapter 4 of NTB 24-22 Rev. 1 (Nagra 2024u) / total system analysis
Saturation of the deep geological repository (including ongoing gas production)	Chapter 6 of NAB 24-20 Rev. 1 Nagra (2024m)			Included in Chapter 4 of NTB 24-22 Rev. 1 (Nagra 2024u) / total system analysis
Corrosion of steel waste disposed of in MOSAIK containers	Chapter 6 of NAB 24-20 Rev. 1 Nagra (2024m)			Included in Chapter 4 of NTB 24-22 Rev. 1 (Nagra 2024u) / total system analysis

Key processes and/or phenomena potentially relevant to repository evolution		Relevance to evolution and performance (analysed in PA)	Potential performance deviations	Addressed in:
Cement – clay interactions and loss of anion-accessible porosity and thus permeability at the interfaces	Table 6-2 of NAB 24-20 Rev. 1 Nagra (2024m)	Chemical evolution of L/ILW near field Evolution of sealing performance	No potential for performance deviations identified	Included in Chapter 4 of NTB 24-22 Rev. 1 (Nagra 2024u) / total system analysis
Internal degradation of cement-based elements	Chapter 6 of NAB 24-20 Rev. 1 Nagra (2024m)			Included in Chapter 3 of NTB 24-22 Rev. 1 (Nagra 2024u) / assessment by component
External degradation of concrete elements: presence of water under partially saturated conditions	Chapter 6 of NAB 24-20 Rev. 1 Nagra (2024m)			Included in Chapter 3 of NTB 24-22 Rev. 1 (Nagra 2024u) / assessment by component
Canister breaching	Table 7-2 of NAB 24-20 Rev. 1 Nagra (2024m)	Breaching of HLW canisters and subsequent evolution of canister interior	No potential for performance deviations identified	Included in Chapter 3 of NTB 24-22 Rev. 1 (Nagra 2024u) / assessment by component
Bentonite porewater enters the canister				Not explicitly included but conservatively accounted for in the calculations
Release of radionuclides				Included in Chapter 3 of NTB 24-22 Rev. 1 (Nagra 2024u) / assessment by component

Key processes and/or phenomena potentially relevant to repository evolution		Relevance to evolution and performance (analysed in PA)	Potential performance deviations	Addressed in:
Short-term modulation of fluvial incision by glacial/interglacial cycles	Chapter 10 of NAB 24-20 Rev. 1 Nagra (2024m)	Geological evolution	No potential for performance deviations identified	Not included but conservatively accounted for in the calculations
Occurrence of permafrost				Not included but conservatively accounted for in the calculations
Long-term rock uplift, fluvial incision and remaining overburden			Faulting or fault reactivation due to earthquakes; extreme erosion	Included in Chapter 3 of NTB 24-22 Rev. 1 (Nagra 2024u) / assessment by component
The expected glacial scenario and deep glacial erosion				Not included
Earthquakes and faulting				Included in Chapter 3 of NTB 24-22 Rev. 1 (Nagra 2024u) / assessment by component

App. C Retention of non-radioactive hazardous substances

In addition to radionuclides, the waste contains non-radioactive but chemotoxic substances, which must also be safely confined in the deep geological repository. These non-radioactive, chemotoxic substances originate either from the waste inventory itself or from technical conditioning materials. Their long-term impact on humans, animals, ecosystems, and essential resources such as groundwater and soil must therefore be considered in environmental assessments (UVPV 1988). The legal framework governing these substances in Switzerland is provided by the Environmental Protection Act (USG 1983).

Within the context of a Swiss deep geological repository, such substances include a range of organic and inorganic compounds as well as metals such as cadmium, copper, and lead (Häner et al. 2014). It has to be ensured that these substances will not migrate into the CRZ and reach the groundwater in significant amounts. Generally, the behaviour of potentially chemotoxic metals does not differ from that of their radioactive isotopes with respect to transport through engineered and/or geological barriers. Therefore, the outcome of the radiological consequence analysis can also be used, by analogy, to indicate the consequences of non-radioactive substances (see Chapter 8).

In Stage 2 of SGT (BFE 2008), a risk assessment evaluated the potential release of chemotoxic substances from the waste inventory into surrounding groundwater systems (Häner et al. 2014). This assessment employed simplified, conservative scenarios that assumed transport by diffusion and no retention by either engineered or geological barriers. Even under these very conservative assumptions, the relevant toxicological thresholds for drinking water safety were largely met. It could thus be clearly demonstrated that chemotoxic elements would be of no safety concern.

Barrier properties and geological conditions

As described earlier in Chapter 3 and Section 5.2, the deep geological repository in Switzerland is planned to be constructed at a depth between 800 and 900 m below ground, within an approximately 105 m thick layer of Opalinus Clay (Section 4.2.6 of NTB 24-17, Nagra 2024i). This host rock, along with adjacent clay-rich formations, provides excellent containment for both radioactive and non-radioactive substances (Chapter 8 of NTB 24-18, Nagra 2024p). These formations isolate the repository from the nearest regional aquifers in the Malm or in the Keuper.

Consistent with earlier findings, the latest geoscientific investigations conducted within the framework of the general licence application confirmed that the clay-rich host rock and surrounding rocks possess large mineral surface areas (30 m²/g for the Opalinus Clay), resulting in excellent retention properties, an excess of negatively charged surfaces (illite-smectite minerals) that slow down diffusion of anionic species, low hydraulic permeability ($K = 5 \times 10^{-13}$ m/s), highly reducing conditions and excellent self-sealing properties. The small pore openings (distribution of diameters peaking at 2 – 3 nm) of the clay-rich rocks also reduce or prevent the transport of larger molecules in the Opalinus Clay (the relevant properties of the Opalinus Clay are described in detail in Chapter 5 of NTB 24-17, Nagra 2024i).

Following contact with water, a large proportion of the waste emplaced in the deep geological repository exhibits low solubility due to its material composition or conditioning. The material properties, along with the robust multi-barrier system, limit the release of substances into the porewater (Hummel et al. 2023, Tits & Wieland 2023, Hummel et al. 2022). As a result, the concentrations of substances that can be transported diffusively into and through the multi-barrier system are already limited. Additionally, many substances are strongly retained by sorption both in the backfilled repository tunnels and in the clay rocks. Calculations from the safety analysis

(Figures 6-9 and 6-61 in NTB 24-18, Nagra 2024p) illustrate that cationic radionuclides (and thus, by analogy, potentially water-polluting elements such as Pb, Cd, or Cu) are retained within the first 10 – 20 metres of the Opalinus Clay.

Conclusion

With an improved understanding of the performance of the repository barriers and more detailed site-specific geological data acquired in SGT Stage 3, previous conservative release models – such as those used in earlier assessments (e.g., Häner et al. 2014) – are now considered overly cautious.

Updated modelling approaches incorporating more realistic physicochemical parameters, including site-specific porewater chemistry (e.g., pH, redox conditions; for more details see, e.g., Glaus et al. 2024a, Marques Fernandes et al. 2024a, Hummel & Thoenen 2023), and enhanced knowledge of retention and degradation mechanisms, have confirmed earlier findings that a deep geological repository in the Opalinus Clay offers a robust multi-barrier system: it safely contains not only radioactive materials but also non-radioactive, chemotoxic substance.