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on Radioactive Waste Management

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Work Package 17

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| Authors | **Madalina Wittel (NAGRA), Anna Alvestav (SKB), Tamara Baldwin (GSL), Peter Buday (PURAM), Adrien Feuerle (ANDRA), Pauli Juutilainen (VTT), Rober Kilger (GRS), Liam Payne (NWS), Dimitri Rochman (PSI), Virginie Solans (NAGRA), Nadja Zeleznik, Barbara Horvat (EIMV)** |

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| Reviewed (KM) | Alexandru Tatomir, BGE | 04/07/2025 |
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# Executive Summary

The demonstration of post-closure criticality safety of radioactive waste in a deep geological repository is a regulatory requirement in many countries. This EURAD-2 deliverable provides the state-of-the-art compilation of national programs for the post-closure criticality safety demonstration for proposed deep geological repositories. Together with considered timeframes and applied criticality safety criteria, it relates to appropriate measures for ensuring criticality safety, summarizes approaches for evaluating criticality safety and describes the development of scenarios for post-closure criticality safety assessments. The document also discusses perspectives on communicating criticality safety for final disposal facilities, highlighting stakeholder input, current communication strategies, and key open questions that need to be explored to enhance public understanding and trust in these safety measures.

## Keywords

Post-closure criticality safety, Criticality safety assessment, Deep geological repository, Final disposal.

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# Glossary

| | |
|---|---|
| BGE | Bundesgesellschaft für Endlagerung mbH (Germany) |
| BUC | Burnup Credit |
| CNSC | Canadian Nuclear Safety Commission |
| CSA | Criticality Safety Assessment |
| DOE | U.S. Department of Energy |
| DGR | Deep Geological Repository |
| EBS | Engineered Barrier System |
| EOL | End of Life |
| EU | European Union |
| GRA | Guidance on Requirements for Authorisation (UK) |
| HAEA | Hungarian Atomic Energy Authority |
| HLW | High-Level radioactive Waste |
| IAEA | International Atomic Energy Agency |
| ICSBEP | International Criticality Safety Benchmark Evaluation Project |
| IFEP | International Features, Events and Processes (FEP) |
| IGSC | Integration Group for the Safety Case |
| ILW | Intermediate-Level radioactive Waste |
| IGD-TP | Implementing Geological Disposal of radioactive waste Technology Platform |
| $k_{eff}$ | effective neutron multiplication factor |
| LLW | Low-Level radioactive Waste |
| NAGRA | National Cooperative for the Disposal of Radioactive Waste (Switzerland) |
| NEA | Nuclear Energy Agency |
| NGO | Non-Governmental Organisations |
| NLC | Nuclear Law Committee |
| NWMO | Nuclear Waste Management Organization (Canada) |
| OECD | Organisation for Economic Co-operation and Development |

| | |
|---|---|
| ONDRAF/ NIRAS | Belgian National Agency for Radioactive Waste and Enriched Fissile Material |
| PCCS | Post-Closure Criticality Safety |
| PEP | Pathway Evaluation Process |
| PIE | Post-Irradiation Examination |
| PURAM | Public Limited Company for Radioactive Waste Management (Hungary) |
| RCA | Radio Chemical Analysis |
| RE | Research Entity |
| RWMC | Radioactive Waste Management Committee |
| QSS | Quasi-steady State |
| SOK | State Of Knowledge |
| SKB | Swedish Nuclear Fuel and Waste Management Company (Svensk Kärnbränslehantering Aktiebola) |
| SNF | Spent Nuclear Fuel |
| SSM | Swedish Radiation Safety Authority |
| STUK | Finnish Radiation and Nuclear Safety Authority |
| TSO | Technical Support/Safety Organisation |
| USL | Upper Subcritical Limit |
| WAC | Waste Acceptance Criteria |
| WMO | Waste Management Organisation |
| WPNCS | Working Party on Nuclear Criticality Safety |
| WP | Work Package |

# 1. Introduction

Deep geological repositories (DGR) are widely investigated as possible sites for final disposal of radioactive waste. The highly radioactive waste (HLW), e.g. spent nuclear fuel, as well as some low-level and intermediate-level waste types (LLW and ILW) contain fissile material. Under very specific circumstances, this fissile material could potentially lead to new fission chain reactions occurring in the DGR. The criticality safety of the DGR is thus a safety requirement in all national programmes that relate to the final disposal of radioactive waste containing fissile material.

The regulatory requirements on criticality safety in final disposal are defined in various national regulations and guidelines for the geological disposal of radioactive waste. There are different aims between different nations but generally the aim is in preventing a criticality event or demonstrating it is unlikely over the lifetime of the disposal facility. Criticality safety is typically to be ensured and demonstrated (through the safety case(s)) both in the operational and in the post-closure phase of the DGR. Although the operational phase requires consideration, some of the unique aspects are in the post-closure phase. This phase is associated with different boundary conditions and challenges posed by the long timeframes to be considered in the post-closure phase of the DGR (typically up to 1 million years).

The main international contributors to the regulatory framework on the geological disposal of radioactive waste are the European Union (EU), the International Atomic Energy Agency (IAEA) and the Nuclear Energy Agency (NEA) under the Organisation for Economic Co-operation and Development (OECD). The comprehensive contribution of these organisations to the safety of final disposal facilities comes in form of guidelines and safety standards, research and development programs, measures for ensuring criticality safety as well as the International Features, Events and Processes (IFEP) list for the DGR. In addition to international frameworks, national radioactive Waste Management Organisations (WMO) work to their own regulatory guidelines, usually inspired by the international standards and practices together with the outcomes of the multinational collaborative initiatives, as for instance under the Implementing Geological Disposal of radioactive waste Technology Platform (IGD-TP). The geological disposal of radioactive waste is thus regulated by a combination of international guidelines and national regulations in close exchange between the participating organisations.

As contribution to this EURAD-2 deliverable, the participating national radioactive WMOs were asked to complete the survey regarding the state of knowledge on the domestic post-closure criticality safety (PCCS) assessments. The questions of the survey mainly addressed the following aspects of the post-closure criticality safety demonstration:

- Description of boundary conditions regarding the national and international regulations or standards,
- Assessment timeframe,
- Criticality safety criterion,
- Waste management strategy,
- Data requirements for PCCS relevant waste streams,
- Development and assessment of scenarios for waste packages (“in-package” and “out-of-package” scenarios),
- Approaches to PCCS likelihood & consequence assessments,
- General approach regarding burnup credit (BUC) implementation and presently used methodology,
- Provision of key reference documents,
- Additional relevant comments.

In total, seven radioactive WMOs completed the survey form. The participating organisations are:

- French National Radioactive Waste Management Agency (Agence nationale pour la gestion des déchets radioactifs, Andra), France,

- Federal Company for Radioactive Waste Disposal (Bundesgesellschaft für Endlagerung, BGE), Germany,
- Nuclear Waste Management Organization (NWMO), Canada,
- Belgian National Agency for Radioactive Waste and Enriched Fissile Material (Organisme national des déchets radioactifs et des matières fissiles enrichies / Nationale instelling voor radioactief afval en verrijkte Splijtstoffen, ONDRAF/NIRAS), Belgium,
- Public Limited Company for Radioactive Waste Management (PURAM), Hungary,
- Swedish Nuclear Fuel and Waste Management Company (Svensk Kärnbränslehantering Aktiebolag, SKB), Sweden,
- United States Department of Energy (DOE), USA.

In addition, Nuclear Waste Services (United Kingdom) has contributed specific views in this deliverable.

The outcome of the survey is thoroughly compiled in the following Sections.

In this EURAD-2 deliverable, the overview of the international and national regulatory requirements and standards for the criticality safety assessment (CSA) for a final disposal facility for radioactive waste containing fissile material as well as the key criticality safety considerations with regard to the design of the final disposal facility, the timeframe to be considered, the criticality safety criterion (or criteria) and the pursued waste management strategy of participating organisations are given in Section 2. Measures for ensuring criticality safety for final disposal facilities, such as the implementation of burnup credit for irradiated fuel assemblies and the derivation of the safe amount of fissile material as well as the definition of acceptance criteria for waste packages, are discussed in Section 3. Section 4 deals with approaches for evaluating criticality safety for final disposal facilities and summarises presently assumed Features, Events and Processes (FEP) as well as the development of "in-package" and "out-of-package" scenarios together with the consideration of the likelihood and consequences of criticality event in post-closure criticality safety assessments. Perspectives on communication criticality safety for final disposal facilities are discussed in Section 5. A summary and the follow-up activities of WP-17 are given in Section 6.

Consequently, the findings and considerations in this report have a direct link and aim to contribute to the following EURAD themes and goals, as captured in the EURAD Roadmap [40].

Administrative measures to ensure criticality safety for final disposal typically aim to e.g. derive the maximum amount of fissile material that can be emplaced in a single waste package or to determine the minimum burnup level that the irradiated fuel assemblies must have before their joint loading in a final disposal canister. The latter refers specifically to the application of burnup credit (see Section 3) in the derivation of loading curves for spent fuel. In this context, several links with the following EURAD Sub-themes and Domains must be noted:

- The link to the EURAD Sub-theme 1.4 (National inventory) and, specifically, to the Domain 1.4.1. (National radioactive waste inventory) highlights the need to ensure that the level of detail captured in national inventory records for HLW as well as the scope of recorded spent fuel parameters should also be informed by criticality safety considerations. WP-17 aims to make contributions to this endeavour.
- The derivation of maximum acceptable fissile masses per waste package in view of criticality safety could have a direct link with Domain 2.1.2 (Waste Acceptance Criteria).
- The loading curve derivation methodology also requires detailed knowlegde of the source term. Consequently, a good characterization of the spent fuel would be highly beneficial. In this regard, the work envisioned in WP17 (and presented as example in Section 3) would contribute to Sub-theme 3.1 (Wasteforms).

Technical measures to ensure criticality safety for final disposal could comprise, for instance, the further optimization of the final disposal canister design to take into account aspects relevant for post-closure criticality safety. In this respect, a direct connection to Sub-theme 3.1 (Wasteforms), in particular Domain 3.1.1. (SNF) and the related State of Knowledge (SoK) reports, and Domain 3.2.1 (HLW and SF Containers) is evident.

Carrying out the criticality safety assessment, as part of the DGR safety case, requires a good understanding of the long-term evolution of the DGR system as well as a robust approach to defining post-closure scenarios to be analyzed. Thus, the aspects summarized in Section 4 have a direct link to the following EURAD Domains:

- Information from the DGR evolution model would inform the FEP and scenario definitions to be considered in the criticality safety assessment. Therefore, there is a direct interaction with the EURAD Domain 4.1.1. (Site descriptive model). In addition, there is also a clear link between this work and the SoTA report published by WP8 (SAREC), as details related to fuel dissolution also inform the type of scenarios to be considered, e.g. the definition of in-package or out-of-package scenarios, etc (see Section 4). Conversely, the work on burnup credit applications for final disposal may also inform the work of other WPs, such as WP8, e.g. regarding nuclides that remain in the fuel matrix over long timescales.
- Understanding the FEP in the long-term DGR system evolution that are relevant for criticality safety and carrying out the criticality safety assessment contribute directly to the EURAD Domain 7.3.1. (Performance and FEP analysis).

This overview, while not exhaustive, is intended to illustrate that the work envisioned under WP-17 as well as the key aspects of criticality safety for final disposal summarized in this report provide clear contributions and have direct links to many relevant Themes and Domains of the EURAD program.

# 2. Overview of national and international regulatory requirements for criticality safety in final disposal

## 2.1 Organisations issuing regulations and guidelines

The regulatory requirements on criticality safety in final disposal consist of a combination of international guidelines and national regulations that aim to ensure the long-term safety of a DGR. While criticality is an unlikely phenomenon in the operational phase of a geological repository ensuring that conditions in the repository do not lead to a criticality event remains a key safety requirement. Regulatory frameworks for geological disposal focus on preventing or demonstrating low likelihood of criticality by maintaining the conditions necessary for subcriticality.

Several international organizations play a role in setting regulations and guidelines for the geological disposal of nuclear fuel.

### 2.1.1 European Union (EU)

The EU has established regulations that govern the management and disposal of radioactive waste. The Euratom Treaty (Treaty establishing the European Atomic Energy Community) requires member states to ensure that the management of radioactive waste, including disposal, is safe. The EU Directive on Radioactive Waste (2011/70/Euratom [1]) specifically mandates that EU countries establish national programs for the safe management of spent fuel and radioactive waste, including geological disposal.

In 2011/70/Euratom [1] the necessity of remaining flexible and adaptable is noted, e.g. in order to incorporate new knowledge about site conditions or the possible evolution of the disposal system. The activities conducted under the Implementing Geological Disposal of Radioactive Waste Technology Platform (IGD-TP) could facilitate access to expertise and technology in this respect. With regards to research and development of competences it is noted:

- (38) Maintenance and further development of competences and skills in the management of spent fuel and radioactive waste, as an essential element to ensure high levels of safety, should be based on learning through operational experience.
- (39) Scientific research and technological development supported by technical cooperation between actors may open horizons to improve the safe management of spent fuel and radioactive waste, as well as contribute to reducing the risk of the radiotoxicity of high-level waste.

### International Atomic Energy Agency (IAEA)

The IAEA provides comprehensive safety standards and guidelines for the geological disposal of radioactive waste. Some key IAEA publications on this topic include:

- IAEA Specific Safety Requirements, *Disposal of Radioactive Waste* (SSR-5) [2]: This document provides a framework for the safety of geological disposal. It outlines the principles for the design, construction, and operation of geological disposal facilities for high-level radioactive
2.1.2 waste. The document covers the long-term aspects and post-closure safety, while acknowledging that in periods far in the future uncertainties can become so large that the criteria might no longer serve as a reasonable basis for decision making. It also points out the need to further the understanding of those aspects influencing the safety of the disposal system. Criticality safety can be found explicitly in two requirements:
    - Requirement 16 (Design of a disposal facility) includes that maintaining a subcritical configuration has to be part of the design considerations
    - Requirement 18 (Operation of a disposal facility) includes that placing fissile material in a disposal facility has to be done in a configuration that will remain subcritical. It also further states that assessments have to be undertaken of the possible evolution of the criticality hazard after waste emplacement, including after closure.
- IAEA Specific Safety Guide, *Criticality Safety in the Handling of Fissile Material* (SSG-27) [3]: details methods for preventing criticality accidents in facilities where fissile material is handled or stored. The guide describes approaches to ensuring criticality safety which includes macroscopic parameter and their analytical determination as well as subcritical limits and safety margins. It describes measures for ensuring criticality safety, for example design, operational limit, administrative measures and basic principles as defence in depth and passive safety. The recommendations provided in the Safety Guide applies to the design, operation and post-closure stages of waste disposal facilities. It is also discussed that credible degradation of the engineered barriers of waste packages, with consequential relocation and accumulation of fissile and non-fissile components should be taken into account.

### 2.1.3 OECD Nuclear Energy Agency (NEA)

The OECD-NEA provides significant support for the development of safe geological disposal practices. The Agency is an intergovernmental agency that facilitates co-operation among countries with advanced nuclear technology. Results of the cooperation are disseminated as published reports, technical guidance and research on the subject. Some activities and recently published documents are:

- Radioactive Waste Management Committee (RWMC): The RWMC supports members in the development of safe and economically efficient management of all types of radioactive waste including spent fuel considered as radioactive waste based on the latest scientific and technological knowledge. Under the RWMC are the Integration Group for the Safety Case (IGSC) and Working Party on Information, Data and Knowledge Management (WP-IDKM) as well as expert groups on communication and robotics.
- Working Party on Nuclear Liability and Radioactive Waste Disposal Facilities (WPLDF), a working party under the Nuclear Law Committee (NLC). The WPLDF works, among other issues, to enhance common understanding among legal and technical experts of the long-term risks presented by facilities for the disposal of radioactive waste and their relationship to nuclear liability regimes.
- 2.1.4 The Working Party on Nuclear Criticality Safety (WPNCS), a working party under the Nuclear Science Committee (NSC), that deals with technical and scientific issues relevant to criticality safety.
- International Features, Events and Processes (IFEP) List for the Deep Geological Disposal of Radioactive Waste, NEA/RWM/R (2024)2 [4]

### National Regulatory Frameworks

In addition to international guidelines, individual countries have their own regulatory frameworks for the geological disposal of radioactive waste. These national regulations often align with or are based on international standards. Examples include:

- United States: The U.S. Department of Energy (DOE) oversees the development of geological disposal facilities through the Yucca Mountain project (although currently on hold). The U.S. has specific regulations governing the disposal of HLW under the Code of Federal Regulations (CFR) Title 10, Part 63, which addresses the licensing of geologic disposal facilities.
- Finland: Finland has one of the first operational geological disposal facilities for high-level radioactive waste, known as Onkalo. The Finnish Radiation and Nuclear Safety Authority (STUK) is responsible for regulating the project, ensuring long-term safety.
- Sweden: Sweden's Swedish Radiation Safety Authority (SSM) regulates the disposal of spent fuel, with a proposed deep geological repository at Forsmark. Sweden is one of the leaders in developing geological disposal facilities and has an extensive regulatory framework for this purpose.

In the preparation for WP17 of EURAD-2 several of the participating organisations in the IGD-TP Post-closure Criticality Safety Project answered a survey about the state of knowledge on PCCS where information about the regulatory requirements was a part. The responses on specific national regulations are described in the following subsections.

#### 2.1.4.1 Andra, France

In France there is no prescriptive instruction concerning PCCS. Neither are there mandatory or specific safety criteria that must be used.

In 2008, the French nuclear authority published a safety guide relative to the final management of radioactive waste in a deep disposal – which is the admitted solution in France to deal with this waste. A mention to criticality issues is formulated in the "properties of the package" section translated as follows:

> *Taken into account the initial configuration and the degradations of the disposal facility components after the closure of the installation, the occurrence of a criticality excursion remains improbable and, if it cannot be definitively excluded, the consequences induced by such event are not unacceptable*.

#### 2.1.4.2 Bundesgesellschaft für Endlagerung mbH (BGE), Germany

In Germany there are specific regulations in:

- *Ordinance on the Safety Requirements for the Final Disposal of Highly Radioactive Waste* (German abbreviation: EndlSiAnfV): § 8; Annex (for § 8 subparagraph 2) [5]
- *Ordinance on the Requirements for the Execution of Preliminary Safety Analyses in the Site Selection Process for the Final Disposal of Highly Active Radioactive Waste* (German abbreviation: EndlSiUntV): § 9 subparagraph 1 No. 4 [6]

Furthermore, there are more indirect regulations, e.g.:

- German *Radiation Protection Act* (German abbreviation: StrlSchG) [7]:
  - Radiation protection supervisor needs to ensure that the necessary measures against an involuntary criticality are implemented (§ 72 subparagraph 1 No. 4 StrlSchG)
- German *Radiation Protection Ordinance* (German abbreviation: StrlSchV) [8]:
  - Radiation protection supervisor needs to ensure that criticality cannot occur during storage of fissile materials (§ 87 subparagraph 2 StrlSchV); radiation protection officer needs to ensure that the state of science and technology is taken into account for preventive measures against the occurrence of nuclear incidents (§ 104 subparagraph 1 StrlSchV)

There is no official guidance. However, it is mandated to take into account the state of science and technology (see above). This includes industrial standards, such as DIN 25472 ("Criticality safety for final disposal of nuclear fuels to be discarded"; 2012-08 [9]), or research reports, such as GRS-A-3707 ("Additional Treatment of Special Topics Regarding Criticality of Nuclear Fuels in the Post-Closure Phase of a Geological Repository for the Final Disposal of Nuclear Waste", German: "Weiterführende Bearbeitung spezieller Themen im Rahmen generischer Sicherheitsanalysen zur Kritikalität von

Kernbrennstoffen in der Nachverschlussphase eines geologischen Endlagers") [10]. DIN 25472 is currently under review and a new revised version will be published in the future.

Specific safety criteria have to be used: $k_{eff,calc}+\sigma_k$ < 0.95 (for the first 500 years after planned closure of facility) or 0.98 (for the remaining assessment timeframe) (§ 8 subparagraph 2 EndlSiAnfV; Annex (for § 8 subparagraph 2) EndlSiAnfV [5])

The regulations/guidance indicate that existing (national or international) standards must/should be used, e.g.:

- General requirement is to take into account the state of science and technology for the operation of federal facilities for safe-keeping and final disposal of radioactive waste (§ 8 subparagraph 2 No. 1 StrlSchG read in conjunction with § 4 subparagraph 1 sentence 1 No. 6 StrlSchG [7])
- Storage of nuclear fuels: preventive measures against damage based on the state of science and technology prerequisite for permit (§ 6 subparagraph 2 No. 2 AtG)
- Calculation programs and substance databases have to conform to the state of science and technology and have to be qualified in this regard (Annex (for § 8 subparagraph 2) EndlSiAnfV, Part A [5])

#### 2.1.4.3 NWMO, Canada

The Canadian Nuclear Safety Commission (CNSC) regulatory document REGDOC-2.4.3 "Nuclear Criticality Safety" [11] provides guidance on assessing criticality safety in general. There is no guidance specific to Deep Geological Repositories (DGRs).

The calculated multiplication factor plus associated uncertainties must not exceed an upper subcritical limit for all normal and credible abnormal conditions. There is guidance on determining these terms.

For this application and materials there are no specific standards that must or should be used.

#### 2.1.4.4 ONDRAF/NIRAS, Belgium

There are no specific regulations or guidance on what must be considered for PCCS in Belgium. Only for surface disposal are there specific regulations on what must be considered:

> *Criticality risks must be excluded within the repository and in its surroundings, during operational and post-operational periods under all reasonably foreseeable conditions, taking into account the associated uncertainties*.

No specific safety criteria are prescribed. Neither are there any specific standards that must or should be used.

#### 2.1.4.5 PURAM, Hungary

In Hungary there are specific regulations in:

HAEA decree 9/2022 [12]:

17.§ (4) e) During the handover of the radioactive waste the isotope composition – including the content of materials capable of chain-reaction – must be discoverable in a level of detail which enables to judge the fulfilment of the requirements stated in the safety case.

In II. Annex to HAEA decree 9/2022 [12]: Safety regulations; **Design, implementation, operation, closure and supervised control of repositories**

- 2.2.1.0400.: … If it's relevant in the given disposal facility, safety functions must be defined to ensure subcriticality, heat and gas-removal. …
- 2.2.1.0420.: Fundamental safety functions:
  - d) subcriticality
- 2.2.1.0710.: During the derivation of the operational terms and conditions measures have to be taken to
  - c) avoid criticality

- 2.2.8.1600.: It must be demonstrated [in the safety case] that accumulation of fissile material such as to sustain a nuclear chain reaction can be excluded.
- 2.2.8.1700.: If the possibility of a nuclear chain reaction cannot be excluded because of the long-term uncertainties of the disposal system, it must be demonstrated in a specific safety assessment, that an event like that will not jeopardize the long-term safety of the repository.

In III. Annex to HAEA decree 9/2022 [12]: Safety regulations; **Siting and implementation of a repository**

- 3.2.1.0400.: The operational period of the facility ends at the final closure of the repository, the timeframe of the long-term safety calculations of the repository must be derived based on the lifetime of the disposal system. The estimated migration-time of the dissolved radioactive isotopes trough the host rock to the biosphere also needs to be taken into account.
- 3.3.1.0100. d): Geological repository is allowed to be implemented only where the geological settings are able to prevent the migration of the dissolved isotopes and the critical accumulation of them in the geosphere. PURAM have noted that it's not clear, what the regulation means by "critical accumulation".
- 3.4.4.0300.: Requirements should be quantified and assessed over the lifetime of the waste disposal system, but at least over a time horizon of 100,000 years. The long-term development of features, events and processes, the derivation, examination and qualitative evaluation of the normal and alternative scenarios must be carried out for a time horizon of at least one million years, taking into account the expected migration time of the contaminants from the disposal area through the geological environment to the biosphere.
- 3.4.4.0600.: In the safety case for implementation license of a high-level or long-lived radioactive waste repository calculations have to be carried out to demonstrate that neither in the operational period, nor in the post-closure phase of the facility the critical accumulation of fissile material can be excluded both in the disposal area (near-field) and in the geosphere. PURAM have noted that this contradicts to 2.2.8.1700 and should be clarified with the regulatory body.

No specific safety criteria is specified in any regulation in connection with radioactive waste repositories. For L/ILW repositories the $k_{eff} + 3\sigma < 0.95$ requirement is used by PURAM.

In connection with the Interim Spent Fuel Storage Facility (operated by PURAM), and the last post-closure criticality safety assessment a more sophisticated criterion was used (see Table 2).

There are no specific guidelines or standards in the topic of PCCS.

#### 2.1.4.6 Swedish Nuclear Fuel and Waste Management Company (SKB), Sweden

There are no specific regulations or guidance on what must be considered for PCCS in Sweden. Swedish regulations are generally performance based and establishes the results and goals of the safety case and safety related activities.

Some main requirements are:

- Chapter 6 Section 2 SSMFS 2008:1 [13]:
  - Measures shall be undertaken to prevent criticality in connection with handling, treatment and storage of nuclear material at the facility. Such measures shall be specified in a safety analysis report in accordance with Chapter 4, Section 2.
- Guidance to Section 9 SSMFS 2008:21 [14]:
  - Particularly in the case of disposal of nuclear material, for example spent nuclear fuel, it should be demonstrated that criticality cannot occur in the initial configuration of the nuclear material. With respect to the redistribution of the nuclear material through physical and chemical processes, which can lead to criticality, it should be demonstrated that such redistribution is very improbable.
- Section 10 SSMFS 2008:21 [14]:
  - A safety analysis shall comprise the requisite duration of barrier functions, though a minimum of ten thousand years.
- Guidance to Section 10 SSMFS 2008:21 [14] discusses that the requirement:
  - should be a starting point for the safety analysis
  - that a relevant time period can be found by comparison of the hazard of the radioactive inventory of the repository with the hazard of radioactive substances occurring

- the next complete glacial cycle, currently estimated to be in the order of 100,000 years, should be particularly taken into account
- it should also be possible to take into consideration the difficulties of conducting meaningful analyses for extremely long periods of time, beyond one million years.

The specific safety criteria are determined in the safety analysis for system.

The guidance to the Swedish regulations indicates that international rules, standards and guidance, mainly from IAEA and NRC, can be used.

#### 2.1.4.7 Department of Energy, United States of America

In the US it is assumed that regulations similar to those that were applicable to Yucca Mountain (40 CFR 197 [15] and 10 CFR 63 [16]) would apply to any future repository that considered post-closure criticality. These standards require that all features, events, and processes that might affect the disposal system be identified. Those features, events, or processes that have less than one chance in 100,000,000 per year of occurring do not need to be included in post-closure performance assessment calculations. In addition, those features, events, and processes with a higher chance of occurring do not need to be included in post-closure performance assessment calculations if the results of the performance assessment would not be changed significantly in the initial 10,000-year period after disposal. These are the standards that provide the context for evaluating the probability and consequences of post-closure criticality.

There are no specific safety criteria for post-closure criticality.

The regulations do not cite national or international standards for post-closure criticality. However, the Disposal Criticality Analysis Methodology Topical Report (DOE 2003) [17] for Yucca Mountain considered five ANSI/ANS-8 standards. The approach developed in the topical report is consistent with some aspects of these ANSI/ANS standards and departs from other aspects, as described in DOE (2003):

- ANSI/ANS-8.1-1998. Nuclear Criticality Safety in Operations with Fissionable Material Outside Reactors [18]
- ANSI/ANS-8.15-1981 (Reaffirmed in 1995). Nuclear Criticality Control of Special Actinide Elements [19]
- ANSI/ANS-8.17-1984 (Reaffirmed in 1997). Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors [20]
- ANSI/ANS-8.10-1983 (Reaffirmed in 1999). Criteria for Nuclear Criticality Safety Controls I operations with Shielding and Confinement [21]
- ANSI/ANS-8.21-1995. American National Standard for the User of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors [22]

#### 2.1.4.8 Nuclear Waste Services, United Kingdom

In the UK environmental safety during disposal operations and after DGR closure is the regulatory responsibility of the relevant environment agency. Once the DGR has been closed, the risk of direct radiation exposure to operators or the public is removed due to the isolation and containment of the material deep underground in an engineered facility. However, if criticality occurs after DGR closure, it might affect the containment safety function provided by the DGR. In order to address this issue, the environment agencies' Guidance on Requirements for Authorisation (GRA) requires that the environmental safety case for the GDF demonstrates that:

*'The possibility of a local accumulation of fissile material such as to produce a neutron chain reaction is not a significant concern.'*

Furthermore, NWS is required to consider a 'what-if' criticality scenario by assessing *'The impact of a postulated criticality event on the performance of the disposal system.'*

There are no specific assessment timescales or safety criteria for post-closure criticality safety and the UK approach is based on an assessment timescale of up to 1 million years with a safety criteria of sub-critical (i.e. $k_{eff} < 1.0$).

## 2.2 Key Safety Considerations for Criticality in Geological Disposal

This subsection discusses some key aspects that usually are considered in the design of a final disposal and also are of regulatory interest. Long-term safety assessments must consider features, events and processes (FEP) that could happen based on the canister design and the waste form and it must be shown for a relevant timeframe, the assessment timeframe.

Geological disposal regulations aim to ensure that conditions in the disposal facility do not inadvertently lead to a criticality event. The design and operational controls for the repository are therefore focused on:

- Package Control: Ensuring that the initial state of waste packages are sub-criticality (i.e. through mass control of fissile material).
- Maintaining Sufficient Separation: Ensuring that spent fuel assemblies or waste packages are adequately spaced so that the waste cannot be accidentally arranged into configurations that could result in critical mass.
- Geological Stability: Ensuring that the geological formations selected for the repository do not change in ways that would lead to an unsafe concentration of fissile material.

Spent fuel is typically packaged in a way that significantly reduces the likelihood of criticality, even if the material were to be subjected to various environmental conditions. The main conditions that can be utilized and be of regulatory interest are:

- High Burn-up Fuel: High-burnup spent nuclear fuel (i.e., fuel that has been irradiated in a reactor) is less reactive than fresh nuclear fuel and is unlikely to reach criticality under any foreseeable condition. The reactivity changes during the long timescale due to decay of nuclides and build-up of more reactive nuclides, mainly decay of $^{241}Am$ or $^{240}Pu$.
- Canisters and Containers: The design of waste canisters and containers for geological disposal is intended to limit the potential for criticality by ensuring that the fuel is not placed in ways that could facilitate a chain reaction. These containers are typically corrosion-resistant and can include neutron-absorbing materials to mitigate the risk. In the long-term perspective the corrosion of the materials in the canisters and the stability of the neutron-absorbing materials are processes that needs to be considered.

Intermediate level waste typically contains less fissile material and is typically packaged in a way that significantly reduces the likelihood of criticality through immobilisation of the waste.

The criticality safety assessment shall be based on appropriate safety criteria and describe assumptions that have been made. And a key component that the development of the FEPs shall be based on is the assessment timeframe. In the following subsections these assumptions are further detailed as they are used by IGD-TP participants.

### 2.2.1 Assessment timeframe

The assessment timeframe depends on the contents of the waste as well as on the conditions of the final disposal (repository), where the latter can be subject to differences between nations depending on geological conditions. For the assessment timeframe the evolution of FEPs affecting the canister, its materials, the separation of radioactive waste or spent fuel and the reactivity of the radioactive waste or spent fuel shall be included. The assessment timeframe can reflect that the results from detailed models for safety assessment purposes are likely to be more uncertain for timescales extending into the far future [IAEA SSR-5 [2]].

In the survey about the state of knowledge on PCCS, information about the assumptions on the assessment timeframe was gathered. Responses are presented in Table 1.

| Participant | Assessment timeframe | Mandated or chosen | Other significant time steps | Any specific FEP |
|---|---|---|---|---|
| Andra, France | 1 million years | Chosen | - | 3rd version of the FEP list for deep geological disposal of radioactive waste [23] and the FEP Evaluation Catalogue for Argillaceous Media |
| BGE, Germany | 1 million years | Mandated | Period of required retrievability (500 years after planned closure) | Yes.<br>Annex (for § 8 subparagraph 2) EndlSiAnfV [5] |
| NWMO, Canada | 1 million years | Chosen | extend analyses (often to 10 Ma) to illustrate longer time trends and that there is no dramatic worsening of response | No |
| UK, NWS | 1 million years | Chosen | - | FEP screening performed to focus on:<br>FEPs that could result in water entry into a waste package (for example, the presence of ventilation openings or the occurrence of package failure mechanisms)<br>FEPs that could result in changes in reactivity following water entry into the waste package (such as degradation leading to the relocation of fissile material, neutron<br>absorbers, neutron reflectors and / or neutron moderators)<br>FEPs that could result in the migration and accumulation of fissile material outside a single waste package (for example, accumulation by precipitation, sorption, filtration or gravitational settling)<br>FEPs that could result in the migration and accumulation of fissile |

| | | | | |
|---|---|---|---|---|
| | | | | material from more than one waste package. |
| ONDRAF/NIRAS, Belgium | 1 million years | Chosen | EOL+~20ka and EOL+~1Ma | No |
| PURAM, Hungary | 1 million years | Mandated | No | IFEP NEA/RWM/R(2024)2 [4] |
| SKB | 1 million years | Chosen | Peak reactivity at EOL +1 yr | earthquake and corrosion of copper |
| USA | 1 million years | Mandated | 10,000 years | No |

*Table 1* – Assessment timeframe

All organisations that answered the survey have an assessment timeframe of 1 million years and whether it is mandatory or chosen differs. The motivation to use this is normally that this is timeframe at which total radioactivity returns to levels commensurate with a similar sized uranium ore body. Some organisations include other timesteps to capture for example: the peak reactivity after 20 000 years due to decay of $^{241}Am$ or $^{240}Pu$, or 500 years as a reasonable period of retrievability.

### 2.2.2 Criticality safety criterion

IAEA SSG-27 [3], 2.2.: *Subcriticality is generally ensured through the control of a set of macroscopic parameters such as mass, concentration, moderation, geometry, nuclide composition, chemical form, temperature, density, and neutron reflection, interaction or absorption. The determination of limits for these parameters is generally performed on the basis of the effective neutron multiplication factor ($k_{eff}$) of a system, for which nuclear data are needed. Because $k_{eff}$ depends on a large number of variables, there are many examples of apparently 'anomalous' behavior in which changes are counterintuitive*.

The safety criteria that are used to assess criticality safety can vary for different assessments or organisations but are generally based either on a set of parameters or the combined behaviour through $k_{eff}$. In the survey about the state of knowledge on PCCS, information about the safety criteria that is used was gathered. Responses are presented in Table 2.

| Participant | Safety criteria | Motive | Plans to reduce the safety margin | Constant for the entire assessment timeframe |
|---|---|---|---|---|
| Andra, France | $k_{eff} + 3\cdot\sigma < 1$ | Consistently with the criticality excursion exclusion, the under-criticality must be observed. As no human operational mistake can occur during this passive phase – and also no possible human exposure in this case – no additional margin to | Substantial margins exist in the configuration parameters. When a configuration does not allow securing the disposal feasibility, hypotheses are challenged to reach this objective | Yes |

| | | | | |
|---|---|---|---|---|
| | | the criteria is taken into account. | | |
| BGE, Germany | $k_{eff,calc}+\sigma_k$ < 0.95 (for first 500 years after planned closure of facility) or 0.98 (for remaining assessment timeframe) | Mandated | No | No |
| NWMO, Canada | $k_{eff}$ + 2 sigma < 0.95 | precedent in Canadian and international criticality safety community | No | Yes |
| NWS, UK | $k_{eff}$ < 1 | Direct radiation from a criticality event would be shielded by the surrounding rocks and there would be no direct risk posed to operators or members of the public therefore an additional safety margin is not required | No | Yes |
| ONDRAF/NIRAS, Belgium | - in-package scenarios: $k_{eff}$ + 2 σ < 0.95<br>- out-of-package scenarios: $k_{eff}$ + 3 σ < 0.95 | Arbitrary choice by the researcher | - | - |
| PURAM, Hungary | USL, based on ANSI/ANS-8.17 [20], with 0.05 administrative margin | Expert advice | - | Yes |
| SKB | $k_{eff}$ < 0.95 for flooded canister and 0.98 when corrosion has altered the geometry inside the flooded canister | international consensus | No, but efforts to reduce uncertainties on USL | Yes |

| | | | | |
|---|---|---|---|---|
| USA | no specific safety criteria for PCCS | - | - | |

*Table 2* – Criticality safety criterion

Most of the participating organisations that answered the survey uses the calculated multiplication factor $k_{eff}$ as the safety criteria to assess the complex behavior of the system. This is in line with criticality assessments for systems and nuclear facilities currently in operation. The numerical value of the $k_{eff}$ varies between countries and events or timeframe and the variation is usually in the application of uncertainties. Conservatism in the safety limit is treated in a couple of different ways in the participating countries: some use different $k_{eff}$ (0.95 or 0.98) depending on event or timeframe and some countries turn to more case specific analysis if a conservative approach is too limiting. For each respective application and country, the same criteria is normally used for the entire assessment timeframe.

## 2.3 Regulatory requirements or guidelines concerning BUC

In general, requirements on criticality safety in the final disposal aim at showing subcriticality and to do so with proper margins and uncertainties. Within this requirement it is often possible for licensees to use an analysis method that includes more or less conservatisms.

Margins regarding criticality are required but an overly conservative system will be both expensive and possibly use materials for the canister in a less sustainable way. Burnup credit (BUC) is an important tool in the criticality safety evaluation that is based on passive features and enables a realistic analysis with conservative assumptions. A large conservatism that is common is to base a criticality analysis on the assumption that the fuel is fresh, i.e. without burnup. In burnup credit the properties in the spent fuel after burnup is utilised, that the concentration of fissile nuclides has decreased, and the concentration of fission products that absorb neutrons has increased.

Among the participating organisations that answered the survey about the state of knowledge on PCCS no one have responded that there are national regulatory requirements or guidelines concerning BUC. However, requirements on the analysis method are applicable, for example regarding assuring the presence of the utilised nuclides, validation of the method and using well-motivated uncertainties and margins.

The U.S. NRC have issued many regulatory guidance documents for BUC implementation specially for storage and transportation of SNF such as NUREG/CR-7108 [24], NUREG/CR-7109 [25], NUREG/CR-6801 [26], NUREG/CR-7194 [27]. Interim Staff Guidance (ISG)-8, Revision 3 [28] provides review recommendations to the Nuclear Regulatory Commission (NRC) staffs for accepting PWR SNF storage and transportation systems using a BUC approach. Although these regulatory guidance documents are for SNF storage and transportation, they provide the basis for as-loaded criticality analysis using full BUC (actinides and fission products credit) that is currently being used to determine the likelihood that a loaded dual-purpose cask could achieve critical configuration in a repository.

## 2.4 Conclusions on regulatory requirements

The geological disposal of nuclear fuel is regulated by a combination of international guidelines and national regulations that aim to ensure the long-term safety and environmental protection of high-level radioactive waste. The goal is to isolate the waste from the biosphere for thousands to millions of years to prevent harm to human health and the environment. International organizations like the IAEA, NEA, and the European Union, as well as national regulatory bodies, play a critical role in establishing and enforcing these safety standards.

Criticality safety is an important regulatory consideration in the design and long-term safety assessments of geological disposal facilities. International guidelines and national regulations ensure that geological disposal facilities are designed with adequate safety margins to prevent conditions that could lead to criticality, such as ensuring proper spacing of waste packages, understanding material behaviour in the long timeframe and performing comprehensive safety assessments with appropriate uncertainties.

Some countries have a comprehensive regulatory framework while some countries have not specific regulations for the post-closure period. As was noted in the answers to the survey on the state of knowledge there are areas of difference between nations and there is not a common set of requirements, however there is a consensus between countries on several aspects even if specific safety criteria or assumptions are not defined by national regulators. A reason for the alignment can be found in the many international standards, guidelines, cooperation and research projects that supports the community.

The challenges in the assessments are connected with the evolution during the long time period that needs to be analyzed. During the assessment time frame all FEPs that act to increase the risk of criticality shall be identified and evaluated. Some analysis methods credit the reduction in reactivity from burnup of the fuel and the FEPs associated with that need to be evaluated just as rigorously. In addition to identifying the FEPs there are requirements on analytical methods that include these FEPs that they are verified and validated, and that uncertainties and margins are determined. Some safety programs also require that a knowledge management program be in place to take into account the state of science and technology and to ensure that future generations understand the potential risks of the repository, including the prevention of criticality.

# 3. Measures considered for ensuring criticality safety in final disposal

The approach taken to demonstrate criticality safety varies between Spent Nuclear Fuel (SNF), HLW and ILW due to the different properties and origins of the materials. In this Section an overview of the current status for these waste streams is given with SNF focussing on the application of burn-up credit and HLW/ILW focussing on package design and mass control limits.

## 3.1 SNF

The goal of loading curves is to indicate, given a number of assumptions, what can be safely included in canisters and how from a criticality safety perspective. The definition of safety criteria and conditions of applicability may vary in different WMOs, as discussed in Section 2, and some examples are provided in the following Section An example of a typical loading curve is presented in Figure 1, for PWR $UO_2$ fuel [29] with $^{235}U$ enrichment and the assembly burnup at the end of life. Two zones are identified: above the curve (representing the SNF assembly characteristics leading to a possible allowed load in a canister, given considered assumptions), and below the curve (representing the SNF assembly characteristics not allowed). The solid curve represents the limit, in terms of criticality safety, separating the allowed and not allowed loading SNF characteristics. Typically, the limit in criticality can be $k_{eff}$ = 0.95, but additional constraints often lower this value or depending on regulatory requirements it may be increased. In addition, examples of potential existing SNF assemblies are indicated in the figure as blue and red dots. The colours classify the SNF for possible different irradiation cycles (see Ref. [29] for more details).

As indicated, the derivation of such a loading curve depends on different assumptions. The principal one is to assume that the same SNF assemblies will be loaded in a canister. Typically, a PWR canister can contain four assemblies; in such a case, it is assumed that the four assemblies have the same characteristics, for instance in terms of initial enrichment and burnup. This assumption of similar cases is generally common to all national regulations. A different approach, based on so-called “mixed” loading, can present a number of advantages, but is in general not applied due to additional potential risks linked to the complexity of the method.

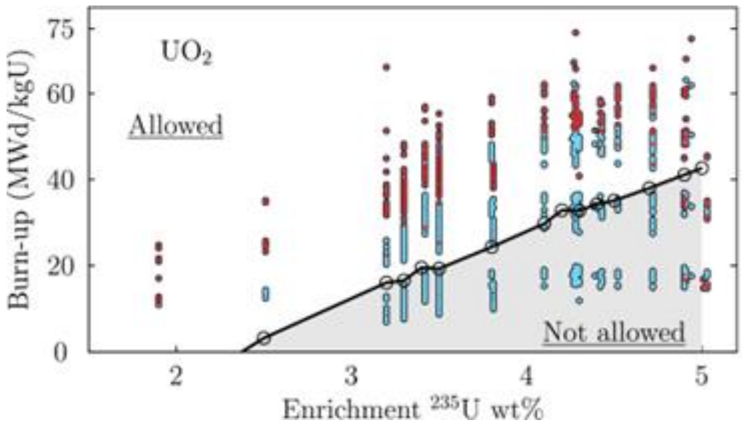


*Figure 1 – Example loading curve for specific assumptions and criteria* [29].

Generally, differences in loading curve derivation can arise from assumptions on:

- Canister design
- Burnup credit method
- Canister and SNF degradation scenarios

The impact of the canister design is not negligible on various aspects. An example of such design is the distance between SNF assemblies inside the canister. Naturally, the distance between the assemblies impacts the $k_{eff}$ value, and therefore the loading curve. Another example is the material considered for the canister. Depending on its neutronic characteristics, it can also influence the $k_{eff}$ value. Another aspect of the impact of the canister design is related to its behaviour over a long period of time. Some materials are more prone to fast oxidation than others. The oxidation can modify the physical integrity of the canister, and its neutronic aspect, for instance by modifying the amount of oxygen in the structure surrounding the spent fuel, or by changing the internal geometry of the canister, therefore potentially modifying the $k_{eff}$ value (and the loading curve).

The burnup credit method considered for the derivation of the loading curve can also impact the results. The estimation of quantities such as the upper subcritical limit or the use of selected nuclear data libraries, can have a direct impact on the loading curve, as indicated in [30]. Other quantities, for instance codes for transport and criticality calculations, benchmarks used for such code validation, and naturally the number and types of nuclides used for crediting the reactivity, are also of prime importance.

Finally, as the canisters are planned to be emplaced permanently in deep geological repositories, one also has to consider their possible degradation and the impact this has on criticality values.

In the following, some examples of burnup credit approaches for selected countries are presented. The origin of the information is the current state of knowledge, as indicated in the IGD-TP Post-closure Criticality Safety Project[1].

### 3.1.1 Sweden (SKB)

The BUC approach is considered in Sweden. The difference between PWR and BWR spent fuel is that in the last case, burnable absorbers are also taken into account for BUC. A total of 28 nuclides are adopted, including actinides and fission products, obtained from the NRC ISG-8 regulation [28].

Loading curves:

Loading curves are used for ensuring subcriticality, within the limit of 5% initial enrichment and 60 MWd/tHM. The safety criteria (0.95 or 0.98) are then reduced with respect to the uncertainties. For the determination of the loading curves, the SNF assemblies with their realistic (but conservative)

1 https://igdtp.eu/activity/pccs-post-closure-criticality-safety/

irradiation histories are considered. Examples of the parameters taken into account at the nodal level are the fuel and coolant temperatures, the power distributions, the void and control rod presence.

<u>Uncertainties and optimization:</u>

In terms of uncertainties, a vast range of sources are considered: for the fuel (manufacturing and operation data), burnup-induced changes (fuel bowing, changes in rod diameter, nuclide composition and burnup distributions), for the canister (corrosion), for the model (approximations and accuracy). In total, uncertainties are considered at 2 sigma.

In the future, SKB will consider mixing SNF assemblies with different burnup values. The loading optimization is mainly based on decay heat and thermal output.

<u>Verification:</u>

Due to safeguard reasons, each individual SNF assembly will be measured before encapsulation. The measurement technique and quantities will be finalized in the near future.

<u>Validation with PIE data:</u>

PIE data are considered for the validation of model and codes. This is done by evaluating bias and bias uncertainties, based on SFCOMPO data. Uncertainties on nuclide compositions are obtained from recommendations in the NRC ISG-8 document [28] and related ORNL references. In the future, not only SFCOMPO data will be used, but also from the MALIBU and LAGER program, as well as other public data.

### 3.1.2 Hungary (PURAM)

The BUC approach is considered in Hungary, unless the SNF assemblies originate from low initial enrichment (for instance 3.83%) and studies show that BUC is not necessary. Given the use of new fuel type and the increase of initial enrichment, the subsequent post-closure criticality safety studies should also include BUC. A total of 29 actinides and fission products were considered for BUC, as part of a sensitivity study. The list of nuclides is related to the NUREG/CR-6665 [31] isotope list.

<u>Loading curves, uncertainties and optimization:</u>

No loading curves are yet derived, and BUC will be applied in the revised PCCS assessment.

<u>Verification and validation:</u>

The verification and validation procedure will be defined in the future.

### 3.1.3 Canada (NWMO)

As the vast majority of the Canadian SNF assemblies are based on natural enrichment, it is highly probable that the BUC approach is not needed. There is currently no national regulatory requirements or guidelines, and there is not yet a definite method for the cases of enriched fuel assemblies.

<u>Loading curves, uncertainties and optimization:</u>

No loading curves are yet derived.

<u>Verification and validation:</u>

The verification is not presently used. For the validation of codes and models, some PIE data exist but are not available in SFCOMPO.

### 3.1.4 Germany (BGE)

The BUC approach is not required in Germany, however its consideration is possible.

<u>Loading curves, uncertainties and optimization:</u>

No details have been presented yet, and the derivation of loading curves might be done in the future.

Verification and validation:

The verification and validation procedure will be defined in the future.

### 3.1.5 France (ANDRA)

There is currently no national regulatory requirements or guidelines concerning BUC, as there is no need for existing SNF assemblies. In practice, different companies are applying BUC following a set of standard practices; in addition, BUC is considered for future SNF types. When applied, the approach of "actinides only" (without $^{241}$Am) is preferred, even if fission products can methodologically also be considered.

Loading curves:

Loading curves (to be differentiated from loading plan, which is related to the SNF position in canisters) are then derived from the above description.

Uncertainties and optimization:

The conservatism approach is applied for the derivation of the loading curves, for instance using flat irradiation profiles, or homogeneous mixtures during depletion calculations. In addition, no optimization is envisaged for the time being.

Verification and validation:

The verification is not yet described. For the validation, measurements of gamma emission are sufficient for the considered burnup level. This is compared with data sheets and ID controls.

### 3.1.6 UK (NWS)

The BUC approach is considered in the UK.

Loading curves:

Indicative loading curves have been produced based on consideration of evolution scenarios. These have been developed to assist in methodology development rather than providing final results. Further work is planned in the future to refine the methodology and scenarios to provide BUC limits.

Uncertainties and optimization:

The scoping curves produced are not optimised and do not take into account uncertainties.

Verification and validation:

Due to safeguard reasons, each individual SNF assembly will be measured before encapsulation. The measurement technique and quantities will be finalized in the near future.

Validation with PIE data:

The verification and validation procedure will be defined in the future.

## 3.2 Data in burnup credit methodology

### 3.2.1 Data for validation

Burnup credit and the loading curve determination is based on validation of the model used and the Upper Subcritical Limit (USL) where bias and bias uncertainty is included. A vital part of the validation are the experiments, and experiments that are suitable for the application. The burnup credit methodology has been developed by the U.S. NRC and ORNL initially for PWR fuel in transportation and dry storage, but the development has been continued for BWR fuel in the same applications and for other applications by other organisations such as a final disposal capsule by SKB. As more countries start using it and for other applications, new needs for experiments to validate the method are being recognised.

There are two main databases for experiments that are used in a large part of the world, both are coordinated and administered by OECD/NEA and more information about them can be found at the OECD/NEA website:

- International Criticality Safety Benchmark Evaluation Project (ICSBEP) [35] Handbook that contains criticality safety benchmark specifications that have been derived from experiments performed at various critical facilities around the world.
- SFCOMPO a database of measured nuclide composition of spent nuclear fuel, with operational histories and design data.

These two databases are continuously updated with new information and reviews of existing data. For the ICSBEP Handbook a new release was made in 2024 that included among other things, nine critical, near-critical or subcritical benchmarks. In 2023 a NEA report [32] was published describing Experimental Needs for Criticality Safety Purposes. This report concludes that there are needs for more experiments e.g. in the intermediate energy spectra for $^{240}$Pu and $^{238}$U and also for iron as a structural material. These experimental needs are relevant for some of the planed disposal systems internationally.

The validation of the model and method for the application to be analysed is based on critical experiments. This validation includes determination of bias, bias uncertainty and a margin for correlated experiments and results in a USL. The process of selecting experiments can be either by engineering judgement or by more sophisticated methods that compares sensitivities in nuclear data variation, the ck-method. Either method reveals some shortage in experimental data. When analysing an application that assumes fresh fuel the validation basis is good. But when assuming depleted fuel similarities between the experiment and the evaluated application was not as good, and not all of the important nuclear reactions could be covered in the same experiment. These needs have been noted by U.S NRC/ORNL in [26,27] and SKB in various studies [e.g. 33, 36].

For a copper canister filled with spent nuclear fuel it has been noted that to be able to reach a ck value of 0.8, experiments with both $^{239}$Pu and $^{56}$Fe is necessary but even then, there are no experiments that includes copper. This is, however, not strictly necessary in the ck-methodology.

Other aspects to cover in the burnup credit methodology is the determination of burnup and the nuclide composition at different burnups. With a large variety in operating conditions, it is difficult to have unambiguous correlations between burnup and nuclide composition, especially for BWR fuel with several fuel designs that can display different properties and have different void and control rod histories. Both aspects, burnup and nuclide composition, are measured and evaluated after reactor operation. Both of these aspects are covered by uncertainties in the BUC methodology and in studies [33] it has been noted that these uncertainties can be significant which also implies that there are large needs to increase the understanding of what affects the uncertainties.

Nuclide composition is determined through examinations of fuel by Radio Chemical Analysis (RCA). Several experiments exist but the majority of these are for PWR fuel and not as many for BWR fuel. The impact on nuclide composition from operating history is not straight-forward and more experiments are needed to get more consistent results, especially for BWR fuel.

**3.2.2**

Efforts are also being made to make better estimations on burnup, both discharge burnup and the uncertainty on burnup. These efforts include measurements from reactor operation such as cold critical measurements and evaluations of in-core measurements. This is another topic that is still under development.

### Sources of post-irradiation examination (PIE) data and its review

In the BUC methodology an essential part is to determine the burnup of the fuel and the nuclide composition, with a degree of confidence. This is done by post-irradiation examination (PIE) mainly with destructive RCA. The main international reference database for nuclide composition of spent nuclear fuel is SFCOMPO. Among the countries that answered the SOK-survey the organisations that use BUC responded that they rely mainly on SFCOMPO.

The PIE data have been noted to contain large uncertainties. For SFCOMPO the OECD/NEA has published a disclaimer on the SFCOMPO webpage describing that "*Any errors in measurements, omissions, or inconsistencies in the original reported data may be reproduced in the database*" and encourages the user of the data to "*consider and assess the potential data deficiencies*«. The validation basis does not completely cover the needs for a comprehensive validation. There are a few ways that the responders to the SOK-survey handles this, either by using all data or by performing a quality control of the data. To use all data in an experimental series can result in large uncertainties. To discard measurements after a quality control must be done with proper justification. It is relatively well understood that the majority of the uncertainty in the M/C values is driven by the uncertainty in the operating conditions used to determine the calculated value. Lack of data for example for nuclides with very few measurements a method using surrogate data as described in NUREG/CR-7251 [34].

In the responses to the SOK other sources of post-irradiation examination (PIE) to determine burnup in the nuclear fuel was noted, such as: proprietary RCA-examinations and gamma irradiation control.

### 3.2.3 Data to demonstrate compliance with loading curves

Since burnup credit and the use of loading curves for the fuel in a final repository is not very mature in the countries that answered the SOK-survey neither is the method to demonstrate compliance to loading curves or its associated data.

Sweden is one nation that have considered the verification of the fuel before loading it into the canisters but mainly from a safeguard's perspective. Measurements will be performed on individual fuel assemblies to verify that the physical characteristic of the fuel assemblies is in agreement with the safeguard declaration and the SKB data records. The exact process and measurement technology are under development.

The U.S. is using as-loaded analyses as opposed to loading curves. As-loaded analysis approach uses data collected using nuclear fuel data survey form (form GC-859). Additionally, detailed data from a few selected reactors are being collected to validate the as-loaded analysis using GC-859 data.

The U.S. has investigated misloading of fuel for disposal scenarios. Calculations assume that the utilities would get the correct fuel into the correct canister but could have placed all of this highly reactive assemblies in the middle of the canister, together. The assumption of having the correct fuel in the correct canisters is underpinned by the complete offloading of fuel from the spent fuel pool without any detected inconsistencies.

## 3.3 HLW/ILW

### 3.3.1 Measures considered for ensuring criticality safety: fissile material mass

Criticality safety studies need as input the characteristics of the fissile materials but also of its environment. This section focuses on waste package characteristics of the high level and intermediate level waste as the scenarios are discussed in section 4.

Like fuel assemblies, ILW/HLW packages have manufacturing specifications and associated controls to ensure their expected characteristics. The two main differences compared to fuel assemblies' packages are:

- The amount of fissile materials present is much more limited (typically of the order of several hundred grams or a few kilograms per package) as it contains residues of processes (either manufacturing, reprocessing or dismantling).
- The geometry and repartition of the fissile material in the package is subject to a higher level of uncertainty due to the conditioning process.

These differences lead to a need to consider a control on the fissile material mass as the preferred measure to ensure criticality safety in every situation. Mass as a criticality control measure is also relevant in final disposal for long term safety studies. Therefore, the amount of fissile material in each waste package is an important parameter of interest to validate the safety demonstration and must be known as accurately as reasonably feasible.

#### 3.3.1.1 Availability of the amount of fissile material per waste package

The amount of fissile material has always been a matter of concern and tracked to manage the risk in operational phases and to avoid loss of such material. This knowledge can be accessed either by historical data records or by measurements, or a combination of these means.

This validates the amount of mass as a relevant measure for ensuring criticality safety for ILW/HLW.

However, it is important to note the following:

- the uncertainties associated with a measurement of a constituted waste package will always be higher that the uncertainties for measurements prior to conditioning of the waste,
- historic measurements can be less reliable, new controls to confirm previous information can provide confidence,
- a performant data collecting system and maintained is a powerful tool.

#### 3.3.1.2 General approach to determine a safe amount of fissile material per waste package

Reminder: The scenarios leading to packages' environment evolution are developed in Section 4. This section focuses on packages' characteristics.

The general approach considers that only minimum accessible knowledge is available, even a long time after the production of the packages. This means that bounding assumptions are typically considered due to a lack of information.

To assist with the development of criticality safety assessment some typical data points are usually considered. As an example, three accessible data points have been identified:

- Initial outer dimensions of the waste package,
- Type of waste matrix,
- Initial disposal configuration
    - is the package alone on its level or among other packages?
    - are the packages disposed of over several levels?

Considering only minimum accessible knowledge leads to a theoretical situation which will probably never be encountered as other useful information are always available. Table 3 presents some examples of hypotheses that could be used as an alternative to a lack of information.

| **Missing data type** | | **General hypothesis which can be considered** | **Justification** |
|---|---|---|---|
| Fissile material characteristics | Isotopes | 100% $^{239}Pu$ | Non-rare in waste (unlike Cm, Cf) bounding material |
| | Physico-chemical form | Metallic | Most reactive form |
| | Density | 19,86 g/cm$^3$ | Theoretical maximum |
| | Repartition in the package | Concentrated in a sphere | Unfavourable geometry |

| Missing data type | | General hypothesis which can be considered | Justification |
|---|---|---|---|
| Moderation | Type | $CH_2$ | To cover presence of plastic materials. |
| | Amount | Variable | To find the optimum |
| Neutron absorbers | - | neglected | Would be favourable |
| Initial geometrical characteristics of the package | Material | Medium inside package assimilated as a water mist with variable density or the matrix material | Evaluate with both mediums either reflection or interactions between packages |
| | Thickness | None | Favors interactions between packages |

*Table 3 – General hypothesis without information.*

It should be also considered that instead of determining a safe fissile mass the demonstration can also consider the given fissile mass per package as an input data and verify the compliance with the applicable effective neutron multiplication factor safety criterion. The alternative hypotheses for missing data presented remain applicable in this case.

#### 3.3.1.3 Additional useful information to determine a safe amount of fissile material per waste package

Taking a bounding approach is a suitable starting point but there is potential that more information is available. When additional useful information is available, it can be considered in the criticality safety demonstration as either a refinement to modelling or further justification of the bounding nature of the assessment. If so, checks or reliability evaluations should be performed to ensure its validity at initial stage.

Also, the impacts of long-term evolution (see Section 4) must be evaluated on these parameters. These could be unfavourable impacts such as reduction of poisons by consumption, by separation or by decay like $^{240}Pu$, thickness reductions. They could also be favourable impacts such as, diffusion/dispersion of fissile materials, stabilization of the metallic fissile materials into oxide form.

### 3.3.2 Waste acceptance criteria

The criticality safety of disposal is demonstrated by the compliance of identified relevant parameters of a package to waste acceptance criteria (WAC). The acceptance criteria must be chosen to fulfil both demonstrations of criticality safety during the operational phase and long-term period. Also, these criteria must be verifiable at the moment of the waste package acceptance, which implies that long-term criticality safety relies on justifications provided by scientific knowledge of the disposal evolution with the associated uncertainties. As highlighted above, the approach to defining WAC will be based on the regulatory environment, the nature of the waste, the scenarios considered and the overall success criteria. Therefore, there is not one specific mechanism for defining WAC and future work is planned on understanding where there is a consensus and where approaches could/should be consistent.

## 3.4 Summary

For SNF disposal the majority of WMOs are, or will be, focussing on BUC as a key safety claim. The maturity of the methodologies, results and verification/validation varies between WMOs aligned to the status of the overall DGR process. However, there are key similarities in broad approaches and future needs and hence the rationale for this being a key area of interest for this Work Package (WP).

For HLW/ILW disposal the focus is on defining an appropriate definition of the waste package and contents that can be used as an input into criticality safety assessments. The approaches for this may vary but the broad principles of defining parameters that can lead to the generation of a fissile mass limit as input into future WAC are consistent. As with SNF, although there are similarities there are potential inconsistencies and hence the rationale for this being an area of focus in this WP.

# 4. Approaches for evaluating criticality safety for final disposal facilities

## 4.1 Features, Events, and Processes (FEPs)

FEP analysis is a tool which is used to help define relevant scenarios for safety assessment studies. For a radioactive waste repository, features would include the characteristics of the site, such as the type of soil or geological formation the repository is to be built in or under. Events would include things that may or will occur in the future, such as glaciations, droughts, earthquakes or formation of faults. Processes are things that are ongoing, such as erosion or subsidence of the landform where the site is located on or near.

Criticality safety results are sensitive to the environment and thus evolution of the disposal facility. FEPs are a relevant starting point to develop scenarios to demonstrate long-term criticality safety.

Several FEP catalogues are available as the type of site (clay, granite, salt…) greatly influences the content of the FEP to be considered. Also, FEP synthetises parameters that influence disposal evolution independently to their impact on the risks to manage.

An analysis of the existing FEP, by repository type and site, and for criticality safety purposes only can constitute a first basis to the definition of PCCS scenarios. The objective is to enrich this basis by feedback, current approaches and knowledge from post-closure criticality safety experts.

The first action is to identify and isolate in the current FEP catalogues those that are relevant to criticality scenarios. Knowledge about criticality is required to perform this identification. Also, this work must be shared between and undertaken by different specialists to make sure that the vocabulary is consistent.

The second step is to expand the FEP lists. To identify gaps, basic understanding of both criticality and disposal evolution is necessary. Review of already constituted scenarios in the different safety demonstrations can also provide insights about the generating events to be considered.

Once this basis is developed, options for translation into hypotheses to build criticality scenarios can be proposed.

## 4.2 Scenarios development and assessment

### 4.2.1 Stakes and approaches

At the beginning of the post-closure phase, the situation is, if not the same, very close to the situation during the operational phase. Indeed, as the objective of disposal is to safely contain and isolate the hazardous radioactive waste, stability over time of the selected site or materials is sought. Therefore, evolution of the environment of the fissile materials (e.g. natural site, manufactured components and the package) is slow. The methodology applied can then rely, for relatively short timescales, on the standards used for operational demonstrations.

Predicting the exact evolution of the disposal facility over 1 million years, which is a common duration for a criticality safety demonstration for deep geological disposal, is an impossible challenge. Indeed, it depends on many factors, some with large uncertainties and some correlated. This is a crucial point to develop as knowledge of the disposal facility evolution is the starting point to criticality safety scenario development. This evolution cannot be determined with certainty, and this should not be an objective.

It means that several situations may occur and are to be considered for scenario definition (rather than just one defined situation). And given the timescale, the associated uncertainties to each situation can lead to consideration of several scenarios to evaluate the different possibilities. This is a major difference compared to operational safety demonstrations; the expected configurations are multiple.

Despite the uncertainties, phenomenology studies provide ranges of possibilities and can assess them. From these learnings, criticality skills are applied to identify the relevant situations to study from a criticality safety perspective. The scenarios that could be developed to fully cover a period of 1 million years are infinite and therefore judgement is required on which ones to consider further.

An iterative approach can be selected, starting with simple cases (e.g. fewer variables) and implementing new ones to include knowledge improvements and/or to challenge former results if they generate excessive constraints. Indeed, in criticality it is still possible to compensate a lack of information by using bounding assumptions. If needed, sensitivity calculations and analyses over some parameters can help identifying the bounding case. However, limits to manage criticality risk will be more restrictive as they result from configurations with these bounding assumptions.

To illustrate this with a theoretical very simplified example, it is assumed that there are only two phenomena influencing system reactivity: water arrival and corrosion of metallic materials. With no more information about these phenomena, it is impossible to evaluate which one occurs before the other or if they are coupled. It is then possible to consider that these parameters are variable between the physically credible bounds e.g. for water arrival, the parameter ranges from no water to fully flooded and all the intermediate levels of filling. For corrosion of metallic material, the parameter ranges from the initial state and then through progressive corrosion until complete corrosion. All the assumptions are crossed to determine the acceptable limits. Doing this way is leading to something close to a worst-case scenario. But by reducing the uncertainties about these phenomena or with specific dispositions like using a most important thickness of material or some that are corrosion-resistant, it is possible to eliminate some scenarios. Of course, the reality is much more complex as there are more parameters than two and many of them are coupled, at least chemically.

Using an iterative approach also helps evaluating the stakes of refining PCCS demonstration and identifying for which parameter it is most efficient to acquire knowledge. On the opposite, if the design of the disposal or the package limits are imposed for example by thermal concerns, criticality refinement will only provide additional margins.

Finally, depending on the development status of the disposal, the objectives may not be the same. The more a concept is advanced, the more precise the scenarios can be. The same principle applies if waste packages are already constituted. As long as PCCS does not generate excessive constraints compared to the requirements of other fields or compared to the reality of the existing/expected waste packages, a 'worst-case' approach can be satisfactory. This approach is especially useful to ensure waste packages to be produced will be compatible with the disposal facility. However, application of an absolute 'worst-case' approach is not reasonable if it generates restrictive constraints or incompatibilities with existing waste packages.

### 4.2.2 In package scenarios

Most current PCCS demonstrations consider 'in-package' scenarios.

Evolution of the environment and degradation is considered or is planned to be considered in the future.

As examples, the following studies have been performed to evaluate their impact on reactivity:

- Corrosion of metallic components,
- Degradation of concrete,
- Movement compared to the initial geometry (due to mechanical failures),
- Water arrival,
- Behaviour of neutronic poisons.

It is important to note that the results of such studies are applicable in the context for which they have been conducted. Also, these studies are not necessarily relevant for all disposal types.

The examples studied to date describe the expected evolution but in scenario analysis it is not only the 'normal' conditions are studied. The potential impact of external events such as earthquake, glaciations, erosion or also of 'variant' scenarios of unexpected evolution is also evaluated.

Given the passive safety function of a disposal facility, scenarios must consider the credible accumulation of events over the 1-million-year period.

As exposed in the previous section, an approach is chosen to answer a current need with the available knowledge. Depending on their status, countries will select the most appropriate approach.

### 4.2.3 Out-of-package scenarios

Criticality safety analysis consists in defining the appropriate measures to manage the risk. To do so, the different sizing configurations must be determined. Qualitatively, out-of-package scenarios tend to disperse fissile materials as there is more space available which has a favourable influence on the risk management. Additionally, out-of-package situations are preceded by in-package situations. Therefore, if there are no areas outside of packages where the fissile materials can credibly reconcentrate, it is possible to exclude out-of-package situations as sizing configurations compared to in-package situations. If out-of-packages situations cannot be excluded as sizing configurations, scenarios must be evaluated.

Different parameters should be considered to develop the scenarios:

- Areas where the fissile material can gather and their characteristics,
- Physico-chemical evolution of the fissile material,
- Migration of the materials (corrosion products, concrete, host-rock, fission products, neutronic poisons),
- Reactions with remaining surrounding materials (corrosion products, concrete, host-rock),
- Identification of the timeframes relevant to study (can be only one),

With this information (or part of it), the approaches described in the section "stakes and approaches" can be applied to develop relevant scenarios for PCCS.

## 4.3 Approach to post-closure criticality scenario modelling

After disposal, the engineered barrier system of a deep geological repository will ensure that criticality is prevented for such time as the waste packaging affords a high level of containment. However, as waste packages begin to degrade, fissile and other materials may be mobilised, and this could affect the potential for criticality. Therefore, the potential for the identified criticality scenarios to lead to criticality requires consideration.

Separate and/or coupled computer models are developed to quantitatively assess the post-closure evolution of the disposal system and the associated neutron reactivity to determine the likelihood that the identified scenarios could lead to a critical system. Depending on national requirements, the intended approach may be to reduce the probability of criticality below a defined threshold or to eliminate the risk entirely for a defined period of time.

Depending on regulatory requirements and the approach to likelihood of criticality, some countries also consider the hypothetical consequences of criticality if such an unlikely event were to occur.

## 4.3 Approach to assessment of the likelihood of criticality

At the time of disposal, controls will ensure that all waste packages are sub-critical with substantial safety margins. Therefore, for post-closure criticality to occur, substantial degradation and relocation of waste form materials would be required. Criticality likelihood models take account of knowledge of the radioactive waste inventories, the disposal concepts for the different waste types and host rocks, and the expected evolution of conditions in the different DGR concepts, as well as associated uncertainties. The models may be developed to assess post-closure criticality scenarios involving rearrangement of materials in a waste package, accumulation of fissile material in the barriers outside a waste package and accumulation of fissile material from more than one waste package. These criticality scenarios could occur as a result of events and processes such as container breach due to corrosion, followed by uranium and plutonium dissolution, advection and sorption.

These criticality scenarios are analysed in different ways, depending on the fissile material contents of waste packages and the expected evolution of conditions in the DGR:

- If a waste package contains insufficient fissile material for criticality, even under the most favourable conditions that can be envisaged for criticality in the vicinity of the waste package, single package scale criticality is not credible.
- Where such judgments cannot be made, or where scenarios involving accumulation of fissile material from more than one waste package are considered, a more detailed analysis can be undertaken of waste package degradation and fissile material relocation in order to estimate the likelihood of criticality.

Where more detailed analysis is required, deterministic and/or probabilistic modelling approaches may be used to understand the long-term evolution of the system for a particular scenario. Both approaches may include arbitrary assumptions about the occurrence or otherwise of particular processes and geometrical configurations.

A deterministic calculation is undertaken for a particular combination of parameter values and assumptions, for example, to evaluate the best estimate value of the relevant parameters or a bounding combination. The probability of a deterministic calculation is often expressed qualitatively (e.g. realistic, cautious, bounding/worst-case).

A probabilistic modelling approach accounts for parameter value uncertainties by using probability density functions. Probability distributions are sampled over many model's runs (realisations) in order to understand the likelihood of critical concentrations or masses of fissile material developing after DGR closure.

The judgments made about the conditions required for criticality in different components of the DGR (in waste packages, engineered barriers and host rock) are important to the analysis. However, there are large uncertainties in the materials that might be involved in fissile material accumulation scenarios and the configurations of the accumulated material. In many cases, such uncertainties can be addressed by making bounding assumptions about fissile material accumulations, such as assuming that fissile material accumulates in optimal spherical or slab configurations and ignoring neutron absorbing materials that could be present. Data on minimum critical masses and concentrations of fissile material in such configurations are used to judge whether critical systems could develop in the different components of the DGR using the above-noted deterministic and probabilistic approaches. In other cases, neutron transport calculations are undertaken to determine whether the evolving systems remain sub-critical.

Figure 2 shows the results of one realisation from a coupled probabilistic system evolution and neutron transport model for a PWR SNF package. This example calculation considers a criticality scenario where the package, disposed of in a generic higher-strength rock environment, is assumed to undergo general corrosion. The presented realisation has sampled a value from the possible range of general corrosion rates and calculated how the volumes of container and fuel materials will vary as the package degrades and corrosion products form over time. The $K_{eff}$ of the system (how close it is to criticality) is calculated at selected points in time for the volumes of materials present, for an assumed model geometry. Two geometric configurations are considered, one where the degraded contents of the package are assumed to form a layer at the container base with water containing fissile material above (the "segregated" case) and one that assumes a uniform mixture of the waste and water (the "water mixed" case). The two configurations represent the two extremes of possible material distributions in the waste packages, but the highest reactivity conditions may occur for a configuration between the segregated and fully mixed cases.

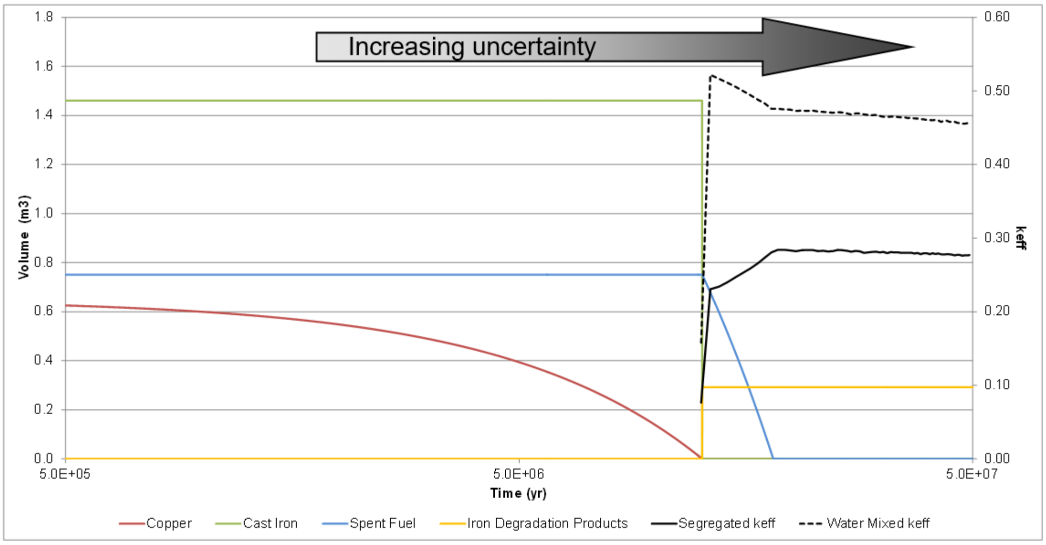


*Figure 2 – Volumes of package materials (left axis) and $K_{eff}$ (right access) over time for a PWR SNF package undergoing general corrosion disposed of in a generic higher-strength rock environment [37, Fig.4.17].*

Over such long timescales there are many possible evolutions of the system and many factors that may influence it. To make modelling computationally tractable, simplifications and assumptions are made. Given the necessary combination of so many parameters, simplifications and assumptions, it is not always clear that a bounding case has been identified. However, it is also important to ensure that resulting models are realistic and credible so that overly conservative and restrictive limits are not defined (which may lead to increased safety risks in other areas and increased cost). The sensitivity of nuclear reactivity to scenario uncertainties and model simplifications needs to be understood and the parameters identified that have the most significant impacts on calculation outcomes. This will support optimisation of assessment approaches by refining modelling methods and uncertainty treatment, especially where model simplifications are overly conservative.

#### 4.3.1.1 Summary of national approach

The survey performed within IGD-TP community suggests that most of the respondents have used deterministic approaches, with comments:

- BGE, Germany: deterministic approach mandatory.
- NWMO, Canada: probabilistic modelling may be used as complementary tool
- ONDRAF/NIRAS, Belgium: deterministic approach at least up till now
- USA Department of Energy: deterministic approach was applied for the Yucca Mountain project, but in the future, the approach will be decided when the site and disposal design are known

### 4.3.2 Approach to assessment of the consequences of criticality

Consideration of the consequences of criticality is not implemented by all nations as it is heavily informed by the relevant regulations (e.g. if criticality is required to be excluded then consequence assessments become irrelevant, whereas other nations are required to understand the potential consequences).

Following closure of a DGR, deterioration of the physical containment provided by the waste packages, movement of fissile material out of the waste packages and subsequent accumulation into new configurations could in principle lead to a hypothetical criticality event. Unlike at previous stages of the waste lifecycle, at this stage there will be no operators or public present, and any radiation produced during the criticality event would be safely shielded by the surrounding rock. The issue therefore then becomes the potential effects of a criticality event on the post-closure performance of the repository system.

When describing consequences of post-closure criticality and in the unlikely event that enough fissile material is brought together during the post-closure phase by some mechanism, broadly two types of criticality event are hypothetically possible, each possessing significantly different timescales and consequences, although it should be noted that the reality will likely lie somewhere between these two 'ends'.

When describing the criticality events the specific names may change but the broad characterisation is similar. In the first type of criticality event, referred to in UK reports as a quasi-steady state (QSS) criticality, an increase in temperature causes a decrease in the reactivity of the fissile material (a negative temperature feedback). Assuming that further fissile material is still accumulating (for example, from in-flowing groundwater) this allows a steady state to be reached, often with only a modest rise in temperature, in which a 'just-critical' configuration is maintained. This just-critical configuration could last for many millennia but would only yield physical consequences (temperature rise and power) that are typically limited to a few kilowatts of power, and a maximum temperature rise of a few hundred degrees Celsius. Therefore, consequences from a QSS criticality are not expected to significantly impact the surrounding geosphere (rock properties). Furthermore, it would only impact a highly localised region.

In the second type of criticality, referred to in UK reports as rapid transient (RT) criticality, an initial increase in temperature causes an increase in the reactivity (a positive temperature feedback). In these circumstances it is not possible to maintain a 'just-critical' configuration, so the neutron flux and power rise, leading to a rapidly escalating temperature. At some point the pressure will become sufficient to drive expansion of the critical region, leading to possible damage to the surroundings (such as possible void formation in the near field and cracking of the surrounding geosphere). This expansion may be sufficient to terminate the criticality. The timescale for a rapid transient event, from start to finish, is typically less than one second.

Importantly, the majority of hypothetical criticality events from fissile accumulation would only evolve as the low power QSS criticality and both types of events are far removed from an explosion type event.

Based on details in the survey about the state of knowledge on PCCS, information about the approach to consequence is summarised in Table 4.

| Participant | Are the consequences of a criticality event considered? | Are they a main part of the safety argument or supplementary arguments? | Key Reference s |
|---|---|---|---|
| Andra, France | No, since the approach is to eliminate post-closure criticality | Not Applicable | |
| BGE, Germany | No, since the approach is to eliminate post-closure criticality | Not Applicable | |

| | | | |
|---|---|---|---|
| NWMO, Canada | Not yet decided with a preference to argue that criticality is not credible. Therefore, no work performed. | If included, it is anticipated that they would be a supplementary argument. | |
| ONDRAF/NIRAS, Belgium | Not yet decided therefore no work performed. | Not Applicable | |
| PURAM, Hungary | Not yet decided therefore no work performed. | Not Applicable | |
| SKB | Yes. On request from the Swedish authority SKB has given an account of the consequences of nuclear criticality in the post-closure phase as a residual scenario. | A supplementary argument | SKB Public Memo 1417199 - What if criticality in the final repository? [36] |
| UK | Yes, a significant amount of work has been performed on transient analysis and also consequence assessments. This is as the regulatory expectation makes specific mention to the consequences of criticality | As part of the main argument, the UK approach is to demonstrate low-likelihood and low-consequence to demonstrate 'not a significant concern'. | |
| USA | For the performance assessment supporting the Yucca Mountain License Application, the consequences of a criticality event were studied but were ultimately not included in the license application. For the current study of direct disposal of SNF in DPCs, the consequences are currently being studied in two hypothetical geologic settings. No decision has been made with respect to the approach to be taken in any future license application. | Not yet confirmed due to status of programme. | |

Table 4 – *Approach to consequence*

As Table 4 demonstrates, the UK have performed a significant amount of work in the consequences of post-closure criticality. The key result from the UK, and similar results have been concluded from studies from other countries, is that the consequences of post-closure criticality are tolerable and do not significantly impact the performance of a DGR.

From a UK results perspective, for the local consequences of 'what-if' criticality, whether 'in-container' or 'out-of-container':

- Peak powers are expected to be no more than a few kilowatts using best estimate parameters
- Peak temperature increases are limited to a few hundred degrees Celsius within the critical region
- Local temperature rises of up to 10 °C above ambient are limited to within a few metres of the critical region.

- With more pessimistic parameters or assumptions, the local consequences of 'what-if' events could be larger.
- With regards to potential impacts of post-closure criticality, key conclusions of the UK work include, but are not limited to:
- While criticality events might have a significant localised impact, they are likely to affect only a limited part of a disposal facility
- Criticality events of the magnitudes considered credible would not significantly damage the geosphere barrier, such that the geological barrier will still act to effectively isolate the radioactive waste from the surface environment, even if localised parts of the Engineered Barrier System (EBS) are damaged
- The change in inventory associated with post-closure criticality is modest in comparison with the assumed original inventory intended for disposal
- Realistically, criticality is only considered credible a long time after disposal, when the inventory of some key radioactive isotopes such as $^{14}C$ would have decayed substantially and the inventory of reactive metals may also have decreased, reducing the consequences of a criticality on the gas pathway.

Table 4 demonstrates that the consequences of post-closure criticality have been studied extensively by a small number of countries with a variety of resource available that documents the specific details. The conclusions indicate that a post-closure criticality event would not impact the performance of a DGR. The use of such arguments in the safety case are driven by the regulatory environment and may not be available to use as a basis of the safety case. However, even if it is not part of the safety case an understanding of the consequences can help inform future work or support communication with stakeholders. Therefore, further work is planned on developing a broadly consistent methodology for assessing the impact of hypothetical criticality events on repository barrier systems and overall performance.

## 4.4 FEPs and Scenarios Summary

All PCCS assessments require the definition of scenarios to be considered for evaluation. A broadly similar approach is adopted across all WMOs that FEPs are reviewed and identified that are subsequently used to define the scenario for assessment. Once the scenarios are defined the assessments are performed in-line with the national regulatory requirements.

As discussed, there are differences between the approaches to scenario definition and assessments undertaken between different countries. It is unlikely that a single scenario identification and assessment method could be developed that is universally applicable without it being identification of the absolute worst-case and imposing significant constraints on disposal. However there has been a significant amount of work performed and collective understanding and consistency, as far as practical, would benefit the international PCCS community and hence the rationale for being a key area of focus in this Work Package.

# 5. Perspectives on communicating criticality safety for final disposal facilities

The establishment of final disposal facilities for radioactive waste is a complex process that requires different technical solutions. However effective, clear, independent, unbiased, and transparent communication of criticality safety is essential for building public trust, regulatory compliance, and ensuring long-lasting safety. Development of appropriate communication approach should be based on the needs of different stakeholders.

## 5.1 Overview Based on Input from National Programmes

The International Atomic Energy Agency (IAEA) provides a framework for public engagement and transparency through safety standards. IAEA safety guide [38] defines "interested parties" as individuals or groups concerned with, affected by, or having the potential to influence the safety of nuclear facilities and activities. These parties encompass a broad spectrum of groups, including the public, governmental bodies, professionals, Non-Governmental Organisations (NGOs), media, and others. A multi-tiered communication approach is recommended to ensure balanced information, two ways exchanges and broader engagement for good decision-making. Best practices from IAEA initiatives emphasise the need for early, continuous and adaptive communication strategies to address evolving public concerns and regulatory expectations. By implementing structured, transparent, and inclusive communication strategies, national programs can enhance public trust, streamline regulatory processes, and ensure the safe long-term management of radioactive waste.

Case studies from different countries [39] highlight the methodologies that enhance best practices in decision-making, regulatory frameworks, and stakeholder engagement. Early and continuous involvement with (identified) stakeholders, particularly civil society, regulators, technical experts, and policymakers is crucial. Moreover, stakeholder engagement has to be modelled for each specific group of stakeholders, i.e. their level of knowledge, concerns, and interests and has to start early to avoid later opposition.

Public engagement in safety case development [39] has faced diverse challenges shaped by historical, cultural, and political contexts. Various approaches have focused on regulatory requirements, safety functions, and post-closure safety assessments. Structured local partnerships, information committees, broader public involvement, and early engagement with communities have all played key roles in building trust. Tools such as technical reports, infographics, interactive Q&A sessions, and stakeholder workshops have helped clarify complex safety issues. Transparent communication and proactive, ongoing dialogue have proven essential in addressing community concerns and fostering acceptance.

## 5.2 Stakeholders

Key stakeholders in radioactive waste management include government, industry, scientific experts, the public, and NGOs. National regulators set safety requirements, review safety cases, and oversee licensing, while environmental agencies monitor impacts. Ministries of energy and environment develop waste policies, and local governments ensure compliance and address regional concerns. Elected officials and advisory bodies support public dialogue.

Industry and scientific actors play a central role. Waste Management Organisations (WMOs) implement disposal programs and collaborate with regulators and the public. Technical Support Organisations (TSOs) provide independent safety assessments and support regulatory bodies, and Research Entities (REs) contribute to solving research question, risk analyses and education.

Public and community stakeholders include the general public, local communities, and some other groups, like indigenous groups near proposed sites. They raise concerns about environmental impacts and long-term safety and participate in consultations. Media significantly shapes public perception.

NGOs and public interest groups advocate for transparency and environmental protection, influencing policy. While anti-nuclear groups often lack structured engagement, international observers help ensure global safety standards through peer review.

Nuclear power plant operators are responsible for waste production and operation of storage, ensuring safety and regulatory compliance. Ultimately, successful radioactive waste management depends on transparent communication and collaboration among all stakeholders.

## 5.3 Current Communication Strategies

The NEA's 2017 report [39] offers guidance on effective stakeholder engagement in radioactive waste management. It includes an event planning checklist based on audience characteristics like technical knowledge, demographics, role (e.g. regulators, NGOs, local communities), and nuclear field workers. The NEA outlines three communication strategies: proactive, passive, and interactive.

The report highlights the importance of two-way, transparent, and accessible communication to build trust and support informed public engagement. Proactive strategies, such as early involvement and educational outreach, help prevent opposition and build confidence. Interactive approaches, like workshops, expert dialogues, and stakeholder monitoring, foster collaboration and mutual understanding. Passive methods, including newsletters, reports, and media outreach, ensure transparency but are most effective when combined with more participatory efforts.

A balanced mix of proactive and interactive methods empowers communities to participate in decision-making. These approaches help address key challenges such as low public awareness, mistrust of authorities, and the complexity of nuclear topics. Presenting technical information clearly and accessibly is vital to overcome information overload.

Despite communication efforts, challenges persist. Many people only become aware of geological disposal when projects impact their area. Scepticism towards institutions, often due to perceived lack of regulatory independence, can hinder trust. To engage the public meaningfully, information must be communicated clearly, early, and through varied channels such as online platforms, community meetings, and site tours. Involving stakeholders in monitoring and direct dialogue helps strengthen trust and transparency in nuclear waste disposal.

## 5.4 Open Questions to Be Explored in View of Communicating Criticality Safety for Final Disposal

Addressing open questions in the nuclear field requires a combination of scientific, regulatory, and public engagement efforts, where tailored, transparent, and interactive communication approaches are essential for building public trust.

To enhance the communication of criticality safety for final disposal facilities, there are several areas needing additional efforts and will be addressed in the frame of WP17. The challenges in these areas have to be approached clearly and concisely:

1. How can the low probability of criticality be effectively communicated?
    - What technical explanations, analogies, and visual tools best convey safety mechanisms to non-experts?
    - How can radiation exposure risks be made relatable through comparison metrics?
    - How do historical case studies reinforce public confidence in nuclear safety?
2. How should uncertainties and long-term safety in deep geological disposal be addressed?
    - How can geological and engineered barriers be explained in a way that reassures stakeholders?
    - How can scientific uncertainties be communicated transparently without undermining trust?
    - How should worst-case scenarios be responsibly presented to avoid unnecessary alarm?
3. What role do regulators play in criticality safety communication?
    - How can regulators ensure transparency while maintaining independence and credibility?
    - What strategies promote consistency in messaging across different regulatory bodies?
    - How can international collaboration improve safety case communication across various nuclear policies?
4. How can transparency and public involvement in nuclear safety be improved?
    - What formats (e.g., events, town hall meetings, workshops) encourage meaningful participation from communities?
    - How can third-party assessments and independent experts enhance trust in safety measures?
    - How should security-sensitive information be communicated responsibly while maintaining transparency?
    - What ethical considerations should be addressed regarding nuclear waste disposal and community compensation?
5. How can technical accuracy be balanced with accessibility in communication?
    - How should messaging be tailored to different stakeholder groups without compromising accuracy?
    - What strategies ensure that simplifications do not lead to misinterpretations?
6. What tools and methods enhance public understanding of criticality safety for disposal?
    - How can digital tools (e.g., interactive websites, virtual reality) improve engagement?

- How should communication methods be customized for local communities versus international regulatory bodies?

7. How can trust be built while addressing misinformation about nuclear safety?
   - What strategies effectively counter false claims and differentiate legitimate concerns from ideological opposition?
   - How can engagement with journalists and media outlets ensure accurate reporting?
   - What proactive measures should organizations take to respond to misinformation effectively?
8. How can long-term knowledge preservation of nuclear safety be ensured?
   - What durable records, symbols, or cultural methods can maintain awareness of disposal site risks over generations?
   - How can sustained community interest and oversight be fostered even after project completion?

However, implementing all the bullet points above remains a not-yet-(fully)-resolved challenge regarding implementation, methods and the best approaches. Besides, the needed approaches in tackling public trust and engagement depend on several parameters like who is the targeted public, what is the current local policy, education system, political will and stability etc. One of the methods to solve the challenges in the above questions can be already developed Pathway Evaluation Process (PEP) method, where different professional profiles and interested parties gather and analyse potential solutions. The methodology's core aim is to encourage a comprehensive discussion on critical issues and perspectives, particularly among diverse actors, including technical experts and civil society. It was developed during the EU funded SITEX-II project (2015-2017), as an interactive tool for fostering structured, multi-stakeholder dialogue on complex topics like radioactive waste management and used already in EURAD 1, as well planned for EURAD 2 in several WPs.

In order to implement the PEP, further understanding of the raised questions should be obtained within the WP17 partners. Therefore, within this task 2 an approached will be developed and activities will be performed to obtain solutions for highly interconnected communication challenges.

# 6. Summary and Outlook

The criticality safety of the DGR is a safety requirement in all national programmes that relate to the final disposal of the radioactive waste containing fissile material. Criticality safety is to be ensured and demonstrated both in the operational and post-closure phase of the DGR.

The regulatory requirements on criticality safety in final disposal consist of a combination of international guidelines and national regulations that aim to ensure the long-term safety. The criticality event is an unlikely phenomenon in the operational phase of a geological repository but to control a criticality event in the closed repository over very long timescales is impossible. The regulatory frameworks for geological disposal thus mainly focus on demonstrating criticality safety in the post-closure phase of the DGR.

Several international organisations together with the national regulatory frameworks play a key role in setting regulations and guidelines for the geological disposal of nuclear fuel.

The main international contributors to the regulatory framework on the geological disposal of radioactive waste are the European Union (EU), the International Atomic Energy Agency (IAEA) and the Nuclear Energy Agency (NEA) under the Organisation for Economic Co-operation and Development (OECD). The comprehensive contribution of these organisations to criticality safety of final disposal facilities results in form of guidelines and safety standards, research and development programs, measures for ensuring criticality safety and well as the International Features, Events and Processes (IFEP) list for the DGR. In addition to international activities, some national radioactive Waste Management Organisations (WMO) work to their own regulatory framework, usually inspired by the international standards and practices together with the outcomes of the multinational collaborative initiatives, as for instance under the Implementing Geological Disposal of radioactive waste Technology Platform (IGD-

TP). The geological disposal of radioactive waste is thus guided and regulated by a combination of international guidelines and national regulations in close exchange between the participating organisations.

As a contribution to this report, the participating national radioactive WMOs were asked to complete the survey regarding the state of knowledge on the domestic post-closure criticality safety (PCCS) assessments.

The key considerations for criticality safety on geological disposal are accounted for in the design of a final disposal facilities. The long-term safety assessments consider FEPs that could happen based on characteristics of the geological site, the waste container design and the waste form.

The criticality safety assessments performed to date are based on appropriate safety criteria and describe the assumptions that have been made. A key component in the assessment is that the development of the FEPs is based on the assessment timeframe, which can be very long.  This very long assessment timeframe represents one of the main challenges for the PCCS assessments. The decision of scenario and discussion of results from the full assessment timeframe has to reflect the fact that the results from detailed models for safety assessment purposes are likely to be more uncertain for timescales extending into the far future.

Some key aspects that are unique to PCCS are the assessment timeframe and also the criticality safety criteria. In the survey about the state of knowledge on PCCS, information about the assumptions on the assessment timeframe was gathered. All organisations participated in the survey have an assessment timeframe of 1 million years. In some countries the assessment timeframe is subdivided into reasonable timesteps for example for accounting the peak reactivity or for taking into account the potential for retrieval (retrievability) of disposed waste from a geologic repository.

The criticality safety criterion for the criticality safety assessment of the DGR can be different. Most of the participating organisations that answered the survey use the calculated effective neutron multiplication factor $k_{eff}$ as the safety criterion to assess the complex behaviour of the entire system. This procedure is in line with criticality assessments for systems and nuclear facilities currently in operation. However, the target value of $k_{eff}$ varies between countries and event or timeframe and the variation is usually in the application of uncertainties, and it is even possible that a $k_{eff}$ of simply less than unity can be used (without administrative safety margins).  Requirements on criticality safety in the final disposal aim at demonstrating criticality safety with suitable margins and uncertainties. In the demonstration of criticality safety for PCCS an appropriate level of conservatism should be adopted to ensure that impacts elsewhere (e.g. cost, sustainability etc.) are proportionate.

A large conservatism that is common in wider criticality safety assessments is to base the analysis on the assumption that the fuel is fresh, i.e. without burnup. In burnup credit (BUC) the properties of the composition of the spent fuel after burnup are utilised. It accounts for the decrease in concentration of fissile nuclides and the increase in concentration of neutron absorbing fission products as a result of the fuel burnup. The burnup credit is thus an important tool in the criticality safety evaluation for PCCS that is based on passive features and enables a realistic analysis with conservative assumptions.

Among the participating organisations that answered the survey about the state of knowledge on PCCS no one have responded that there are national regulatory requirements or guidelines concerning BUC. Additionally, the participating organisations provided no information about alternative technical measures to ensure PCCS as for example the use of filler material prior to the final disposal.

Despite the fact that the U.S. NRC guidance documents for BUC implementation are issued for storage and transportation systems, they are already being used for the evaluation of the final disposal of the spent nuclear fuel, for example for the determination of the likelihood that a loaded dual-purpose cask could achieve critical configuration in a repository.

In contrast to spent nuclear fuel, for HLW/ILW packages the limitation of the fissile material mass is usually applied as an administrative measure for ensuring criticality safety. The general approach to determine a safe amount of fissile material per waste package considers that only minimum accessible knowledge is available, even long time after the production of the packages. This means that a bounding assumption is considered in the face of a lack of information every time it is possible. Considering only minimum accessible knowledge leads to a theoretical situation which will probably never be encountered as other useful information are always available.

For all waste types the criticality safety evaluation for the final disposal facility deals with FEPs as a starting point to develop scenarios to demonstrate the long-term criticality safety. Several FEP catalogues are available as the type of soil or geological formation of the repository greatly influences the content of the FEP to be considered.

As a part of the FEPs, the prediction of the evolution of the disposal facility over the relevant timeframe depends on many factors, some with large uncertainties and some correlated. This is a crucial point to develop as knowledge of the disposal evolution is the starting point to criticality scenarios development. This evolution cannot be determined with certainty. Nevertheless, the identification of a bounding case using sensitivity calculations and analyses over relevant parameters in an iterative approach is widely utilised.

The current PCCS demonstrations consider “in-package” scenarios, whereas the evolution of the “out-of-package” scenarios has only been considered in detail by some WMOs.

Design, construction and long-term evaluation of final disposal facilities for radioactive waste is a complex process that requires different technical solutions. Furthermore, effective, clear, independent, unbiased, and transparent communication of criticality safety is essential for building public trust, regulatory compliance, and ensuring long-lasting safety. Development of appropriate communication approach based on the needs of different stakeholders is inevitable as well.

Despite progress in communicating safety for the near-surface disposal facilities, several open questions remain, requiring further exploration, refinement, continued research, cross-sector collaboration, and public engagement. Addressing these challenges will enhance trust, transparency, and long-term sustainability of final disposal projects.

In summary, demonstrating the criticality safety of a final disposal concept in the post-closure phase, i.e. over very long timescales, is a complex and unique endeavour for many, if not all, WMOs that have to dispose of spent fuel and other radioactive waste. While certain matters are intrinsically related to the particularities of each individual disposal concept or facility, the WMOs do address many similar aspects. Therefore, the sharing of knowledge, experience and innovative ideas between them has clear benefits in ensuring criticality safety of final disposal facilities. To continue to build knowledge and understanding requires further continuation of national disposal programs through activities focusing on:

- Validation of long-term evolution scenarios for PCCS assessments,
- Verification of calculation model implementation for PCCS assessments,
- Validation of depletion and criticality codes for PCCS assessments,
- Investigation of alternative technical measures to achieve PCCS,
- Methodologies for post-closure criticality consequences assessments,
- Fissile waste package records as evidence supporting PCCS assessment assumptions,
- PCCS communication techniques.

These are the focus areas of this Work Package and will bring broad benefits to all project participants and their national programmes.

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