

Community research

PAMINA Performance Assessment Methodologies in Application to Guide the Development of the Safety Case

(Contract Number: FP6-036404)



Task reports for the first group of topics: Safety Functions Definition and Assessment of Scenarios Uncertainty Management and Uncertainty Analysis Safety Indicators and Performance/Function Indicators DELIVERABLE (D-N°:1.1.1)

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PU	Public	Х	
RE	Restricted to a group specified by the partners of the [PAMINA] project		
CO	Confidential, only for partners of the [PAMINA] project		



Foreword

The work presented in this report was developed within the Integrated Project PAMINA: **P**erformance **A**ssessment **M**ethodologies **IN A**pplication to Guide the Development of the Safety Case. This project is part of the Sixth Framework Programme of the European Commission. It brings together 25 organisations from ten European countries and one EC Joint Research Centre in order to improve and harmonise methodologies and tools for demonstrating the safety of deep geological disposal of long-lived radioactive waste for different waste types, repository designs and geological environments. The results will be of interest to national waste management organisations, regulators and lay stakeholders.

The work is organised in four Research and Technology Development Components (RTDCs) and one additional component dealing with knowledge management and dissemination of knowledge:

- In RTDC 1 the aim is to evaluate the state of the art of methodologies and approaches needed for assessing the safety of deep geological disposal, on the basis of comprehensive review of international practice. This work includes the identification of any deficiencies in methods and tools.
- In RTDC 2 the aim is to establish a framework and methodology for the treatment of uncertainty during PA and safety case development. Guidance on, and examples of, good practice will be provided on the communication and treatment of different types of uncertainty, spatial variability, the development of probabilistic safety assessment tools, and techniques for sensitivity and uncertainty analysis.
- In RTDC 3 the aim is to develop methodologies and tools for integrated PA for various geological disposal concepts. This work includes the development of PA scenarios, of the PA approach to gas migration processes, of the PA approach to radionuclide source term modelling, and of safety and performance indicators.
- In RTDC 4 the aim is to conduct several benchmark exercises on specific processes, in which quantitative comparisons are made between approaches that rely on simplifying assumptions and models, and those that rely on complex models that take into account a more complete process conceptualization in space and time.
- The work presented in this report was performed in the scope of RTDC 1.
- All PAMINA reports can be downloaded from http://www.ip-pamina.eu.
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Executive Summary

Pamina WP1.1 is devoted to the review of methods and approaches in the safety case used in the participant countries and in the other main geological disposal development programmes.

The work plan for WP1.1 is structured in 11 topics which all together encompass the scope of the safety cases. The programme is organised in three successive phases. The present report corresponds to the first phase, during which the following topics have been reviewed:

- safety functions
- definition and assessment of scenarios
- uncertainty management and uncertainty analysis
- safety indicators and performance/function indicators

This phase started at the inception of the project, in October 2006, and concluded with the edition of this report in March 2008.

The treatment of these four topics followed the steps defined in the Annex 1 to the Contract "Description of Work":

<u>First step</u>: Target definition. In this step the scope and the outstanding issues for each topic were clearly delineated and described in written guidelines.

<u>Second step</u>: Overview of methods and approaches. In this step the participants prepared their individual contributions, where the approaches and methods applied within their respective organisations, with appropriated references to the national and international contexts, were explained, first in preliminary, and later in final version. In order to harmonize the individual contributions, the participants held a technical meeting in June 2007.

<u>Third step</u>: Analysis and synthesis. The participants made a thorough discussion of the contributions on the four topics in a workshop hosted by Andra in October 2007. The synthesis of those contributions and of the discussions of the workshop is reported in the four topical reports included in this document, one for each of the topics.





The participants and the contributions made on the four topics included in the first phase of WP1.1 are the following:

	Scenarios	Safety functions	Indicators	Uncertainty management
ANDRA	Х	х	Х	Х
AVN	Х	Х	Х	
DBETEC			Х	
Enresa	Х	х	Х	Х
GRS – K	х	х	Х	Х
GRS – B			Х	Х
IRSN	Х	х	Х	Х
SCK-NIRAS	Х	х	Х	Х
NIREX-NDA	Х	х	Х	Х
NRG	Х	х	Х	Х
NRI	Х	Х	X	X
Posiva	X	X	X	X







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Part 1: Task report Safety Functions



PART 1: SAFETY FUNCTIONS

(Prepared by Jan Marivoet, SCK•CEN, Mol, Belgium)





1 Introduction

The concept of safety functions was already in use in 1980 for safety studies of nuclear power plants, e.g. [CORCORAN W.R. et al, 1981]. During the development of the defence-in-depth concept [IAEA, 1996] for nuclear power plants, the multi-barriers concept was complemented with an approach ensuring the fulfilment of three basic safety functions: controlling the power, cooling the fuel and confining radioactive material.

Around 1995 the possibilities to apply the defence-in-depth concept to radioactive waste disposal were examined within the Swedish radioactive waste management programme. The starting point for the development of safety functions related to a geological repository was the main objective of radioactive waste management, i.e., to protect man and the environment from exposure to ionising radiation from radionuclides, which are present in the waste, now and in the future. The strategy adopted to achieve this objective is to concentrate and contain the waste and to isolate it from the biosphere [IAEA, 2006]. It was felt necessary to complement the multi-barriers principle with a set of safety functions that are provided by diverse mechanisms and components. Early applications of the concept of safety functions in safety evaluations of radioactive waste disposal can be found in the Swedish [SKB, 1999] Belgian [DE PRETER P. et al, 1999] and French [ANDRA, 2001] radioactive waste management programmes.

2 Regulations and guidelines

As the concept of safety functions is relatively new in the context of safety evaluations of radioactive waste disposal systems, this concept has not yet been introduced in many published national regulations and guidelines. On the other hand the safety functions concept is already mentioned in a number of documents recently published by international organisations.

2.1 International level

The use of safety functions for the description of the contributions of the main system components in the presentation of the assessment basis is recommended in the NEA [2004] Safety Case brochure.

The IAEA [2006] Safety Standards for Geological Disposal of Radioactive Waste mention requirements for multiple safety functions: "The natural and engineered barriers shall be selected and designed so as to ensure that post-closure safety is provided by means of multiple safety functions. That is, safety shall be provided by means of multiple barriers whose performance is achieved by diverse physical and chemical processes. The overall performance of the geological disposal system shall not be unduly dependent on a single barrier or function." The Standards also mention





requirements concerning containment and isolation of the waste.

2.2 National regulations and guidelines

The regulations concerning radioactive waste disposal are currently being revised in various countries of the European Union. This means that in many cases the published regulations are becoming obsolete and that non-official information, e.g. discussion documents, presentations at workshops and minutes of working group meetings, is available on the new regulations that are in preparation.

The current French regulations [DSIN, 1991] are based on the multi-barrier principle, but they also mention, without using the term, a number of safety functions that have to be fulfilled by the repository system: "the site and the artificial barriers should play a double role: protect the waste by hindering flow of water in contact with the waste and intrusive human actions, and limit and retard the transfer of radionuclides released by the waste to the biosphere during a period necessary for a sufficient radioactive decay of the radionuclides". From the discussions between French nuclear safety authority ASN and other involved organisations, it can be expected that the new regulations will base the derivation of safety functions on the following key points:

- prevent the circulation of water in the repository;
- contain the radioactivity in the repository (by avoiding the dissemination of radioactivity, limiting the release of radionuclides and delaying and attenuating their migration);
- separate the radioactive waste from man and the biosphere so that the • repository system safety shall not be affected either by the erosion phenomena or ordinary human activities.

In Belgium, regulatory requirements and guidelines concerning long-term safety of high-level waste disposal are still in preparation. The so-called "Franco-Belge" document [FANC et al., 2004], which was prepared by the French and Belgian nuclear safety authorities and nuclear waste agencies, recommends that the principle of defence-in-depth should be implemented by multiple safety functions. The safety functions mentioned are isolation, containment and limitation and retardation. The concepts of multiple safety functions and of multiple barriers should be considered as complementary.

In Germany the management of radioactive waste is under review. Also the current regulations of geological disposal [BMI, 1983] are being reviewed by the Federal Ministry of the Environment. In this context, GRS Köln has prepared a discussion document on safety requirements [BALTES B. and RÖHLIG K.J., 2006]. This document proposes "confinement" as the primary safety function. Further it gives a number of basic and site specific safety requirements.

The Swedish regulations [SKI, 2002] mention barriers and their functions: "The function of each barrier shall be to, in one or several ways, contribute to the containment, prevention or retardation of dispersion of radioactive substances, either directly, or indirectly by protecting other barriers in the barrier system."

The Swiss regulations [HSK, 1993] mention a system of multiple passive safety





barriers, which have to contribute to the containment and retention of radionuclides.

In the UK, regulations [Environment Agency et al, 1997] require a multi-barrier concept: "The overall safety case for a specialised land disposal facility shall not depend unduly on any single component of the case". This, together with the types of waste in the UK, means that the safety case takes a multi-barrier approach, with explanation of the safety functions provided by the barriers and how these evolve over time. Also, the focus of UK regulations is on limited and delayed releases rather than on containment there is no regulatory requirement to contain radionuclides for a specified length of time.

3 Terminology

3.1 Definitions of safety function

The IAEA [2007] Safety Glossary gives as definition for a safety function: "a specific purpose that must be accomplished for safety" (the further explanations given in the IAEA Glossary are related to reactor safety and are not directly applicable to a geological repository).

The following definitions are used by waste agencies:

- ONDRAF/NIRAS [2007], Belgium: "function that the disposal system should fulfil to achieve long-term safety".
- SKB [2006], Sweden: "role through which a component contributes to safety".
- Andra [2005], France: "consists of meeting the safety objectives by implementing different type of actions that all contribute to the safety of the repository during the different phases of the repository".
- Nagra [2002], Switzerland: "a function relevant to long-term security and safety".

In Germany, the following definition is proposed by the technical support organisation GRS [BALTES B. et al., 2007]: "a safety function is a function, which takes over safety relevant requirements, in a safety related system, subsystem or single component".

From the above given definitions it appears that the differences in the definition of safety functions are small, and, consequently, that there is a common understanding of the safety functions concept among the different groups involved with safety cases of geological repositories.

3.2 Related definitions

Within the national waste management programmes various terms derived from or related to safety functions are used.

ONDRAF/NIRAS uses the term "sub-safety function": it forms a sub-category of a safety function.

SKB uses the term "safety function indicator": it is a measurable or calculable quantity



through which a safety function can be quantitatively evaluated. The use of this term has been discussed further in the topic 'Safety and Performance Indicators'.

ONDRAF/NIRAS uses the term "effective safety function": it is a safety function that is effectively fulfilled during a certain timeframe, and that can thus be relied upon in safety assessments.

ONDRAF/NIRAS and Andra use "latent safety function": it is a safety function that is available in the disposal system, but that only becomes effective if another safety function fails to be fulfilled.

ONDRAF/NIRAS uses the term "supplementary safety function", which is identical to the term "reserve safety function" used by Nagra: a safety function that could be effective during a certain timeframe, but whose performance cannot be properly evaluated because of a lack of knowledge.

Andra uses the terms "reserve safety function", which is a safety function that remains available, possibly in a degraded form, after the period assumed by the designer, and "performance margin" when the performance level is better than the one taken into account by the designer.

Methodology 4

4.1 Categories of safety functions

In the contributions given in the annexes, and in the Swedish (SKB, 2006) and Swiss (NAGRA, 2002) safety cases that were also considered, three main categories of safety functions can be distinguished:

- stability /isolation;
- containment (called "isolation" by SKB and POSIVA); •
- limited and delayed releases.

It has to be noticed that in France "prevention of water circulation" is also considered as a safety function [ANDRA, 2005]; it can be noticed that ONDRAF/NIRAS [2007] considers this term as a sub-safety functions belonging to the "limited and delayed releases" safety function.

The importance that is given to a specific safety function strongly depends on the host formation. In case of disposal in hard rock or salt formations, "containment" may be the primary safety function, whereas in the case of disposal in argillaceous formations the safety function "limited and delayed releases" may be at the same level of importance as "containment". The relative importance of the safety functions also varies with time.

a) Stability /isolation safety function

In this group of safety functions it is possible to distinguish two sub-groups:





- one sub-group is related to isolating the waste from future surface events and climate changes, and which thus contributes to the stability of the repositories' near field conditions and to the longevity of the natural barriers; this sub-group forms a boundary condition that ensures that the other safety functions can fulfil their role during the demanded periods; this sub-group is also called, e.g. in Germany, stability;
- the other sub-group is related to the reduction of the probability that future • human actions might result in inadvertent intrusions into the sealed repository.

b) Containment (called "isolation" by SKB and POSIVA)

This safety function prevents groundwater from coming into contact with the waste. It is considered by SKB and POSIVA as the primary safety function. In the case of disposal in hard rock or argillaceous formations this safety function is provided by a metallic canister (also called overpack or container in other waste management programmes). However, in the case of disposal in salt formations a "containment" function is provided by the host formation itself.

c) Limited and delayed releases

The containment function cannot be provided over all relevant times for all radionuclides. After failure of the "containment" function, the "limited and delayed releases" safety function will have to play its role. In the case of disposal in argillaceous formations and some hard rock formations this safety function is a very important one. Therefore, several waste management agencies considering disposal in clay have developed sets of sub-safety functions for this safety function, which are specific to the considered host formation and repository design. Andra developed the following set of sub-safety functions: limiting release of radionuclides, and delaying and reducing migration of radionuclides. ONDRAF/NIRAS considers the following sub-safety functions: limitation of releases, limitation of water flow, and retardation.

4.2 Demonstration that the safety functions will be fulfilled

As safety functions are playing an important role in the safety case, methods are developed to demonstrate that the expected set of safety functions will be available as long as required. For this purpose, SKB (2006) has developed a set of safety function indicators for its two main safety functions "isolation" (which we call "containment" in the present note) and "retardation". Those safety function indicators are based on the understanding of the properties of the components of the repository system and quantitative criteria have been defined for each safety function indicator. There is more detail on this topic in the 'Safety and Performance Indicators' task report.

ONDRAF/NIRAS has developed a system of so called "safety statements" which have to be fulfilled to ensure that the safety functions will be available at the required time periods.

For Andra, each safety function is characterised by a performance level, the period during which the function has to be available and the component(s) that have to fulfil the function. Some indicators allowing assessment of the performance of individual



components with respect of their safety functions have been defined.

4.3 **Role of dilution**

Dilution clearly plays a role in the estimation of radiological consequences (e.g. doses). However, it is not considered as a safety function because it cannot be controlled by design and only to limited extent by site selection, and, furthermore, it is expected to change considerably with time, e.g., due to the impact of the evolution of the climate on the hydrogeological system.

5 Applications and experience

Safety functions were initially introduced in safety cases for implementing the defencein-depth principle; therefore, the functioning of the repository system is analysed by identifying the role of the main components and processes of the system. As safety functions facilitate explanation of the functioning of the repository system in easily understandable terms, they soon appeared to be a very useful tool for communication to non-technical audiences. Later on, they started to play a central role in the safety case and they were also used for various applications such as determination of the safety strategy, development of the repository concept, structuring the safety case and identification of a comprehensive set of evolution scenarios.

The following list gives an overview of the applications of safety functions by a number of waste management agencies and technical support organisations:

- determination of the safety strategy: ONDRAF/NIRAS, Andra, SKB, and Nagra;
- developing the repository concept: ONDRAF/NIRAS, NRI/RAWRA, Andra, SKB, and POSIVA;
- analysis of the functioning of the repository system: ONDRAF/NIRAS, Andra, SKB, NRI/RAWRA, POSIVA, and NDA;
- testing the robustness of the repository system: Andra and GRS-Cologne;
- structuring the safety case: ONDRAF/NIRAS, Andra, Nagra, and SKB; .
- scenario identification: ONDRAF/NIRAS, SKB, Andra, and GRS-Cologne;
- uncertainty analyses: Andra;
- identification of performance indicators: ONDRAF/NIRAS, Andra, NRI/RAWRA, Nagra, and GRS-Braunschweig;
- communication: ONDRAF/NIRAS, SKB, Nagra, Enresa, and NRI/RAWRA. •

The use of safety functions in safety cases of geological repositories is relatively new. So in many national programmes the set of safety functions is not mature and comprehensive. In several cases, it appeared rather difficult to define generally acceptable reference values or criteria for the safety function indicators.





Developments 6

Various possible applications of safety functions within a safety case have been given in Chapter 5. A number of those possible applications are still in their early stages and require further development.

SKB (2006) mentions that some criteria used for the safety function indicators might be relaxed or that other criteria might be added. In Belgium, ONDRAF/NIRAS is underpinning the set of safety functions by sub-sets of safety statements. Those safety statements require further testing and checks for completeness, and they still have to be complemented with criteria for testing that the safety statements will be fulfilled by the repository system.

Safety functions have already been used by SKB (2006) for the identification of scenarios. Other organisations, such as ONDRAF/NIRAS, Andra and GRS, are still testing this possible application of safety functions. For instance, the treatment of the gas issue in the set of safety functions is not evident. A topic strongly related to scenario identification is uncertainty management. The use of safety functions for uncertainty management is being explored by, e.g., ONDRAF/NIRAS and Andra. Uncertainty management is a separate topic in PAMINA WP1.1.

Conclusions 7

The term safety function was already used in safety studies of nuclear power plant around 1980. In the defence-in-depth concept for nuclear power plants, safety is based on a set of safety functions. Around 1995 safety functions were introduced in safety cases for geological repository systems for radioactive waste disposal.

Most regulations published in European countries do not yet explicitly mention safety functions and they often refer to the multi-barriers concept. On the other hand they use terms such as containment, and limitation and retardation of releases, which we now call safety functions. Furthermore, it appears from available discussion documents that in several European countries new regulations are in preparation, and that it can be anticipated that many of those new regulations will make explicit use of the multiple safety functions concept.

Several definitions of the term safety function can be found in national or international documents, but they all have similar meanings. However, for the definitions of secondary terms derived from safety functions (such as the safety function indicators) some homogenisation might be desirable.

The sets of safety functions that are used by most waste management organisations as well as regulators are very similar. Three main categories of safety functions can be distinguished; these are stability /isolation, containment (which is called "isolation" by some organisations) and limited and delayed releases. The importance of a category of safety functions depends on the considered host formation and repository concept. Methods are being developed to demonstrate that the safety functions will be available when required. Dilution in the aquifers and biosphere is not considered as a safety function.





Safety functions are already widely used for various applications such as determination of the safety strategy, development of the repository concept, analysis of the functioning of the repository system, testing the robustness of the repository system, structuring the safety case, scenario identification, identification of performance indicators, and communication. There is a clear trend to increase the use of safety functions within the Safety Case, as can be seen in recent safety assessment exercises.

Topics that are still under development are the derivation of criteria to demonstrate that the safety functions will fulfil their expected role at the required times, and the application of safety functions for uncertainty analyses. These issues are covered in the separate PAMINA WP 1.1 topics on 'Safety and Performance Indicators' and 'Uncertainty management and analysis'.

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9 Appendices

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A5 IRSN (France)

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Part 1: Task report Safety Functions

Appendix A1: ANDRA (France)

A1 ANDRA (France)

[PAMINA] (D-N°: 1.1.1) – Task reports for the first group of topics Dissemination level: RE Date of issue of this report: 15/03/2008





Part 1: Task report Safety Functions

Appendix A1: ANDRA (France)





Project acronym:

PAMINA

WP1.1

OVERVIEW OF PAST EXPERIENCE IN SAFETY FUNCTIONS

(updated version 26/11/07)

Andra





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STRATEGY AND KEY ELEMENTS

This present contribution from Andra aims at giving an overview of methodologies that have been used by Andra in the framework of the Dossier 2005 Argile in the four topics selected by the steering committee: 1) safety functions, 2) scenarios, 3) safety indicators and 4) uncertainties management.

The first meeting hold in Amsterdam on June 12th, 2007 was an opportunity to review contributions and discuss them for the future workshop to be held in Paris in October. The present document completes the draft provided for the Amsterdam meeting and clarifies some points discussed during the October 2007 workshop at Andra. Its structure has been revised according to the DWG common structure.

The December 30, 1991 French Waste Act entrusted Andra, the French national agency for radioactive waste management, with the task of assessing the feasibility of deep geological disposal. The Basic Safety Rule RFS III.2.f of June 1991 [i], issued by the French nuclear safety authority, provides a framework for the studies to be conducted. The protection of man and the environment are to be demonstrated. Furthermore, studies should show the ability to limit potential consequences to a level as low as reasonably possible. The concept should include a multiple barrier system. and rely on passive repository evolution without institutional control beyond a given timeframe (500 years). The studies carried out within this framework are presented in the "Dossier 2005 Argile " (clay) [ii] and "Dossier 2005 Granite" [iii].

PRIMARY REFERENCES

In the present document, the « Dossier 2005 Argile » is used as reference. Primary references include the French Act and the series of reports submitted accordingly:

- The French Waste Act dated 30th December 1991 [iv] •
- The French Safety rules namely RFS.III.2.f, guidelines [i]. •
- Synthesis Report, Evaluation of the Feasibility of a Geological Repository, • Meuse/Haute-Marne Site (in English and French) [ii].
- Architecture and Management of a Geological Disposal System Report (TAG; ٠ C.RP.ADP.04.0001) (in English and French) [v].
- Phenomenological Evolution of the Geological Repository Report (TEP; C.RP.ADS.04.0025), (in English and French) [vi].
- Geological Repository Assessment of Safety Report (TES: • C.RP.ADSQ.04.0022) (in English and French) [vii]

Other references such as the presentation made at the symposium hold in Paris in January 2007 [viii], and the INTESC guestionnaire [ix] have been used when applicable.





STRATEGY AND KEY ELEMENTS

The feasibility assessment for the argillaceous site builds upon a number of key elements:

- Basic input: the inventory model of the waste and the geological site,
- Safety functions and requirement management [x and xi],
- Technical solutions based on industrial experience,
- Reversible management and monitoring,
- Phenomenological Analysis of Repository Situations (PARS) and detailed, coupled process modelling [xii],
- Qualitative Safety Assessment (QSA) [xiii], uncertainty management, and scenarios,
- ALLIANCES simulation platform and calculation results.

Although the process thus summarized may suggest a linear progression from basic input data to designing a "solution" and assessing its safety, the process is in fact highly iterative, with repeated feedback exchanged between the various processes (see Figure 1). In addition to the routine feedback common to parallel engineering, three main iteration loops have been identified since 1991, each corresponding to a major milestone of the program: License application for construction and operation of the underground research laboratory (in 1996), submission of the Dossier 2001 (in December 2001), and the recent submission of the Dossier 2005.



Figure 1: Dossier 2005 Argile; three iterations loops since 1991 (1996, 2001, 2005)

In view of providing sound feedback to design, research and development and to determine residual uncertainties, the following tools have been carried out: **the functional analysis (FA)** [x, xi] to determine the safety functions and associated requirements – what do we want? -; the **Phenomenological Analysis of Repository Situations (PARS)** [xii] providing a good scientific understanding based on scientific

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studies from surface and underground laboratory – what do we get? -; **the qualitative safety analysis (QSA)** [xiii] managing uncertainties and the quantitative assessment [safety and performance indicators] including sensitivity analysis –. What is the impact of a given uncertainty (or set of uncertainty factors) on the robustness of the system? – And eventually: does the concept meet the safety/acceptability criteria?

The following sections of the document describe in more details each of those topics according to the sequence of the various stages of activities conducted in the dossier 2005 (see Figure 2).

SAFETY FUNCTIONS

SECTION 1: BACKGROUND/INTRODUCTION

The safety function description constitutes one part of the Safety Tome of the dossier 2005 Argile as it is considered as one of the key input of the safety analysis (see Figure 2).

With respect to international guidance regarding the main elements of a safety case [xiv], Andra applied the notion of "multi-safety functions " (i.e. a system of controlling the safety of the repository by assigning safety functions) as a complement to the socalled "multi-barrier" approach. In many ways, the "multi-function" approach is a generalization of the "multi-barrier" concept relying on the geological layer (host rock), engineered components and waste containers and packages. The approach allows safety to rely on multiple functions performed by various components of the disposal system. Each function is characterized by: a performance level, the period during which the function has to be available and the component(s) (one or more) that have to fulfil the function. This approach acknowledges the fact that the components of a repository may not act as traditional "barriers" once the repository is closed, as total containment may not be guaranteed in the long run. Safety functions give access to a finer definition of the role of each component, making it possible to assess the contribution of each of them to the overall safety performance. It may allow us to identify features that are important for the global safety of the repository, even though they may not relate to a containment capacity.









Figure 2: Representation of the various stages of the analysis.

To ensure that safety considerations govern repository design, as well as construction and operating procedures, the safety functions are a basis for developing technical requirements imposed on design options (see Figure 2).

By identifying the functions that are to be performed in order to guarantee the postclosure safety of the repository, one makes a natural link between safety objectives, the features and processes that are critical as regards safety functions, and the engineering options that may fulfil safety functions. The near circular cross-section profile of the engineered structures, their dimensioning, their dead-end arrangement, their closure with low-permeability seals, the backfilling of all drifts and the choice of materials (concrete, steel, clay, bentonite) all indeed contribute to the three main safety functions.

With this approach, technical design solutions are presented for waste disposal packages, disposal cells, and for underground infrastructure. To assess the industrial realism of suggested design solutions, Andra has based its studies on existing industrial feedback, has conducted the design of underground facilities and operational equipment up to a reasonable level of detail, and has conducted specific tests (above ground), pertaining for example to the horizontal emplacement of C-type waste.

As mentioned in the first paragraph, Andra has implemented a system of controlling the safety of the repository by assigning safety functions as a method that complements the so-called « multi-barrier » approach. The latter, used in nuclear reactor safety, consists of placing several confinement barriers between the radioactive materials and the environment, as far as possible independent of each other. The development of this approach led to the establishment of the notion of defence in depth, which complements the « barrier » concept with that of « lines of defence », adding to the simple physical confinement barriers all the material and organizational provisions





enabling accidents to be prevented or their consequences reduced and managed. The functional approach to safety is another development of the « multi-barrier » strategy. Today, this approach is recommended at international level. It consists of meeting the safety requirement by asking oneself what are the objectives to be sought. Safety does not necessarily simply involve placing successive physical barriers between humans and radioactivity. In certain situations, particularly for a repository, such an approach is inappropriate.

The functional analysis was introduced early in the process, and benefits today from the iterative loops of 1996, 2001 and 2005. Overall methodology was not changed; mostly definition evolved according evolution of scientific knowledge and designing options.

SECTION 2: REGULATORY REQUIREMENT AND PROVISIONS

The functional analysis of Dossier 2005 was derived using only requirements from the Basic Safety Rule and the results from previous scientific studies.

The fundamental objective of the repository with respect to safety in the basic safety rules RFSIII.2 .f consists of "protecting the human being and the environment against hazards associated with the dissemination of radioactive substances" in the short and long term.

The RFS.III.2.f has given the conception basis using the "multi-barrière concept" i.e.

« Le site et les barrières artificielles de confinement devront jouer un double rôle :

- protéger les déchets en s'opposant à la fois aux circulations de l'eau au contact des déchets et aux actions humaines intrusives,
- limiter et retarder, pendant le délai nécessaire à une décroissance radioactive suffisante des radionucléides concernés, le transfert vers la biosphère des substances radioactives éventuellement relâchées par les déchets.

Le concept multi-barrières a pour mérite de ne pas faire reposer la sûreté du stockage sur une barrière unique, dont la défaillance pourrait, à elle seule, compromettre gravement les deux rôles assignés au stockage rappelés ci-dessus. Les barrières jouent, à cet égard, des rôles complémentaires. Néanmoins, à long terme et après décroissance d'une partie importante de la radioactivité contenue dans les déchets, la barrière géologique et les matériaux de scellement des puits devront pouvoir assurer à eux seuls le confinement ».

SECTION 3: KEY TERMS AND CONCEPTS

Safety Function: Each function is characterized by: a performance level, the period during which the function has to be available and the component(s) (one or more) that have to fulfil the function.

The internal functional analysis must not refer to the definition of the architectures, but only to elements of knowledge over which the implementer has no control (for example, regulatory requirements, or elements of phenomenological knowledge). For example, a correct way to express a function is "protecting vitrified waste from water" rather than



"maintaining the water tightness of the vitrified waste container". This way, the real objective is clearly expressed and technical options may vary without any change to safety functions.

By identifying the functions that are to be performed if one wants to guaranty the safety of the repository, one makes a natural link between safety objectives on the one hand, the features and processes that are critical as regards safety functions, and the engineering options that may favour safety functions. Therefore, the clear identification of safety functions, and a shared understanding of this notion among different teams within Andra, are the main tools to provide a natural link between engineering, phenomenological understanding, and safety.

SECTION 4: TREATMENT IN THE SAFETY CASE

METHODOLOGY

The derivation of the main safety functions, starting from general ones to more detailed ones, is then guided by a systematic methodology classically utilized in other industrial contexts such as aeronautics, and space industries. What was used for Dossier 2005 was a method of "flux management", that is to say to identify what "fluxes" (of matter, of energy, etc.) are important to be managed, and make sure that safety functions exist to perform such management. This method guided the breakdown structure of the three main safety functions while taking into account water and radionuclide fluxes. Indeed, in the "Dossier 2005 Argile", the flux of radionuclides through the repository was the most important one on the long term, although the flux of water may prove to be important also, even though only small fluxes are expected. In addition, the flux of mechanical constraints inside the repository may need to be considered, as the host rock may be damaged by it. One has to underline that the exact method is of a lesser importance than the result: various methodologies exist and may lead to a different expression of the safety functions. But as long as the methodology is systematic enough, the outcome is always very similar.

The reader will find a more detailed description, and explanations of the construction methods in the volume dedicated to this objective [v].

A functional analysis procedure was implemented (see Insert 1, Chapter 3.5 of the safety volume [vii]). Safety functions are established according to the life phases of the repository: distinction was made in the dossier 2005 between the operational and post closure phases.

Insert 1 Functional analysis procedure implemented

This insert explains the procedure in accordance with which the safety function breakdown structure was established. It is not essential to the understanding of the results of the analysis but helps to understand what level of systemisation it provides.

The establishment of the safety functions is the result of an internal functional analysis, both during the operational phase and in the surveillance and post-closing phases [x, xi]. In the first case, the análisis is based on the experience feedback from installations



required to manage high-level waste or spent fuel packages in order to define safety functions conventionally used in such cases.

During the post-closing phase, it was necessary to define a procedure that provided a systematic survey of the functions, within a context in which there is less experience feedback. ANDRA decided to apply a procedure based on the organisation of functions into a breakdown structure and the identification of « flows ». A function being a component's effect on its environment, this effect can always be interpreted in terms of flow management. For example, a confinement function consists of slowing down or blocking a flow of radioactive nuclides. A heat dissipation function controls a termal flow.

Identifying safety functions then comes down to identifying flows that have to be controlled. The flor of radioactive nuclides and chemical toxins within the repository is, of course, the most obvious flor but others also have to be taken into account :

- The flow of water within the repository, insofar as the concept of storing within clay is based on minimising water circulation ;
- The flow of thermal, chemical or mechanical stresses if they are likely to disturb the qualities of the components

Consequently, the procedure consists of following the major flows and ensuring that functions allow them to be controlled. This check is not suffic ient to guarantee that the functional arrangements cover all eventualities, since by definition they cannot be complete: it reflects a designer's choice amongst all the possible ways of defining and arranging safety functions. However, it makes it possible to confirm that the analysis is coherent.

In order to illustrate the procedure we can, for example, explain how the three main functions allowing the risk of dispersal of radioactive nuclides by water to be managed have been derived.

The risk is associated with the action of water. It is therefore a question, initially, of « harnessing this flow », i.e. controlling its onset. An initial function must therefore make it possible to ensure that any water circulation is under control and that the flows are limited (« resisting water circulation » function). This flow is then « transformed »: the water is liable to become charged with radioactive nuclides. The means of resisting this phenomenon must therefore be defined, i.e. of resisting radioactive nuclides from entering into solution and being transported by the water (« limiting the release of radioactive nuclides and immobilising them in the repository » function). Finally, all « incoming » flows must « exit » the system. This « exiting » must also be managed. A function makes it possible to ensure that the radioactive nuclides circulate as slowly as possible and that the flows are reduced (« delaying and reducing the migration of radioactive nuclides » function). The monitoring of flows using the « harness/transform/restore » method makes it possible to guarantee the systematic nature of the functional breakdown structure.

Furthermore, the design is « constrained » by elements that are beyond the designer's control. These include :

• The recommendations in the basic safety rule RFS III.2.f (limitation of water





flows, protection of waste packages etc.) which guide the design by directing the main choices ;

- Certain objectives that are not derived from the repository safety objective but which the designer deems necessary to meet in addition. For example, this may involve resisting to events which, without directly contributing to accelerating or increasing the flows of radioactive nuclides, may endanger or complicate the safety analysis. For instance, preventing a long term criticality accident within the repository means that it is not necessary to study in detail the potential consequences (as heat, radiation, ...) of the components performance in such a reaction configuration ;
- Requirements other than those concerning safety, for example those associated with reversibility.

All of these elements are considered as being « constraints ». The constraints are mentioned in the functional analysis for reference purposes. In certain cases, they direct the breakdown of functions into sub-functions.

Other methods would probably have led to a different arrangement of the safety functions and a different expression of the constraints. However, insofar as the functional analysis establishes the current state of knowledge and the designer's choices, the list of functions identified at the end of the analysis would have been similar.

Functional analysis methodology:

After the procedure of functional analysis was implemented, the function at the most general level were declined or broken down according to timescales and physical extent into sub-functions accomplished by specific repository components. The functional analysis methodology applied for instance for the internal functional analysis is described in the Insert 2, as given in the Safety Evaluation Volume Chap 3.5 [vii].

Determining the typical timescales over which the major repository components evolve (natural medium, waste, exogenic elements introduced by the repository) allows the designer, in particular, to structure his thought processes in time and space. It allows him to determine the technical solutions appropriate to each phase of the repository's life. The designer defines the safety functions to be fulfilled for each component, for each timescale. In order to do that, he takes into consideration the predictable behaviour of the components of his system. That allows him both to determine if it is realistic to assign such a function (« at that time, will the component still be in a physico-chemical condition that allows it to fulfil the expected safety function? ») and if it is necessary to add new functions applicable to the problems of each period (« at that time, will the extent of such a disturbance be of an order of magnitude such that it will be necessary to make provision for limiting its effects by means of some special arrangement or device? »).





Insert 2 : Functional analysis methodology

The purpose of this section is to describe the procedure for conducting the internal functional analysis of the repository, i.e. the manner in which the functions at the most general level, as described above, are broken down according to timescales and physical extent into sub-functions accomplished by specific repository components. That makes it possible to describe repository architecture and explain the requirement that each component has to fulfil. Each safety function can in fact be broken down into sub-functions and so on, to a level of detail that the designer considers sufficient with respect to these requirements, in order to characterise and specify the repository's components. The requirements themselves depend on the project's level of progress. The functions are broken down into technical solutions using a defined « system », i.e. within the limits in which the designer proposes to act. Apart from the system (including all « engineered works »), one must take into account its environment, as made up of all elements whose characteristics and behaviour are taken « as is ». The breaking down of functions into sub-functions is not, in principle, unique. It reflects the designer's choice. It is based on:

- The current knowledge of the behaviour of repository components, which provides confidence in their ability to fulfil their assigned functions.
- Experience feedback from earlier safety assessments, that have confirmed or not the benefits of certain safety functions compared to others and have, in particular, made it possible to identify external events or internal stresses which might endanger the correct operation of the repository, and against which it is possible to make constructive provisions.

The breakdown into functions therefore reflects the result of the designer's thought processes. It develops gradually as the design progresses. Once the overall functional context has been established, the design is revised and the fine detail added in order to enable the safety functions to be fulfilled. The research programme is aimed, in particular, at the phenomena that underlie the achievement of the functions (for example: corrosion for the container sealing function, the formation monitoring programme for its confinement properties, etc.).

As a minimum, each function is characterised by:

- <u>A performance level</u>, i.e. a quantification of the effectiveness of the action expected. However, it is not necessarily relevant in principle to fix a performance level. It is only valid if it can be used to establish the dimensions of the components that have to fulfil the function. If the function has to be fulfilled by at least one component that is beyond the designer's control (for example the geological medium) or if the link between dimensioning and performance depends on the functioning of the entire system (for example, the permeability of a given seal certainly influences the limitation of water flows but within a larger whole depending on other parameters), there is little point, in principle, in fixing a performance level ;
- <u>A period during which the function has to be available</u>;
- One or more of the components that have to fulfil the function and the physical



phenomenon or phenomena that enable these components to fulfil it. In the particular case of safety during the post-closing phase, given the long timescales involved, only the host formation, waste packages and engineered components introduced by humans (seals, containers, back-fill etc.) are considered to be components with a safety function. The other elements present in the repository, due to the operational conditions or to its natural evolution (functional clearances within disposal cells, corrosion gases generated within the repository, etc.) cannot fulfil a function as there are too many unknown factors concerning their long-term evolution.

Depending on the case, a function may:

- Be available, possibly in a downgraded form, beyond the period taken into account by the designer. We then talk about a « reserve function », the duration of this reserve not always being quantifiable. But identifying reserves gives confidence in the fact that the system has a better level of safety than that which is strictly predicted and quantified ;
- Be available with a performance level better than that taken into account by the designer. We then talk about a performance « margin », i.e. the designer does not use all the performance capability which could be expected to be available to him. The existence of margins also improves confidence levels. The existence of a phenomenon that improves safety but which is not taken into account as a function can be considered as either a reserve or a margin, depending on how you see it ;
- Finally, a function can be latent, i.e. it does not act due to the existence of another function. For example, the confinement provided by the matrix of a waste is latent as long as it has not been subjected to the action of water, i.e. as long as the container protects it. The existence of latent functions makes it possible to manage accidental losses of functions (for example, in this case, a loss of the container's sealing integrity).

An illustration of margins and reserve functions is given in Figure 3.

At this stage, it is not to justify the fact that the design is able to meet the safety objectives nor check the performance level of each function; that is the purpose of performance assessment. The aim is to explain the range of safety functions proposed by the designer within the repository system, check that they are complementary to each other and identify the existence of redundancy, margins, reserves and latent functions. We also explain the safety strategy used by the designer to guide his choices throughout the design development process. In the description of the function, we identify the design provisions and principal physico-chemical phenomena linked to each functions. These phenomena may be favourable (in which case the safety functions have to use them to best advantage) or unfavourable (in which case the functions must prepare to counter their effects). In some cases, they may be neither and simply have to be taken into consideration. Once designed, the check to ensure that the system can stand up to a wider range of disturbances and individual phenomena, without necessarily prejudging whether they are favourable or unfavourable, is the subject of later safety analyses.





Figure 3: Illustration of margins and reserve functions [vii]

APPLICATION

The so-called long-term safety functions, i.e. during the post-closing phase, constitute the repository's true specific character. Here, we limit ourselves to those at the highest level, very largely independent of the repository architectures eventually selected.

Relative to the post closure phase, one of the primary functions of repository safety was "**Isolating waste from surface phenomena and human intrusion**", which forms one of the principles of an installation in a deep geological formation. It consists of giving priority to a management solution in which the waste is kept out of reach of populations, in order to prevent them from being exposed to radioactivity (exposure to radiation the waste emits or risk of ingestion / inhalation), for periods linked to the decay of the radioactivity. The geological disposal principle is to carry out this function passively, i.e. without surveillance being required beyond a defined period.

Another long term safety function was: **Preserving the repository record**. The basic safety rule RFS III-2.f states that personnel protection must be provided « without relying on any institucional control on which it is impossible to count for certain beyond a limited period (...) (500 years) ». That does not contradict the desire to maintain the site record for as long as possible. The problem of maintaining a record of the site begins during the operational phase when it is a question of maintaining the knowledge and technical skills required to manage the installations. Secondly, after placing waste packages in the repository, the record forms an element of the defence in depth making it possible, in particular, to prevent the risk of intrusion within the repository or to enter it knowingly. On this alter point, it is linked to the previous function but also covers a broader objective of defence in depth. Nevertheless, forgetting about the repository in the long-term, which cannot be totally excluded, should not have an



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adverse effect on safety.

With respect to this method indicated in the previous paragraph, the fundamental safety function "**protecting the human being and the environment against hazards associated with the dissemination of radioactive substances**" can be declined into three high-level safety functions, that are at the core of the long-term safety assessment: (1) prevent water circulation in the repository (2) limit the release of radionuclides and immobilize them inside the repository, and (3) delay and reduce the migration of radionuclides toward the environment (Figure 4). In light of the great importance of the host rock properties for long term safety, a fourth high-level safety function was identified as: (4) Preserve favourable properties of the geological medium and limit disturbances.

With respect to the methodology described in Insert 2, diagram in Figure 5 illustrates the main longterm safety functions and their corresponding time scales.



Figure 4: High-level safety functions and components.







Legend : Operating functions, Latent functions Progressive need of a function Progressive disappearance of the function

Figure 5 : Safety functions over time

SECTION 5: LESSONS LEARNED

KNOWLEDGE/EXPERIENCE GAINED WITH THE DEFINITION AND USE OF SAFETY FUNCTIONS

The description of the safety functions points up the existence of three complementary lines of defence which last throughout the analysis : one relies on advection control inside the repository, another limits the release of radionuclides and immobilises them in the repository in the near-field, and the last delays and attenuates flows.

These functions enable us to characterise the role of the components more accurately than would be possible using only the notion of a « barrier ».

One of the aims of the analysis of operational hazards and of the qualitative safety analysis (QSA) in post-closure is to check whether there are causes of failure that can compromise the planned safety functions. The robustness of the system can nevertheless be affirmed at this point. It is based on:

• The different components : host formation, shaft seals, drift seals, cell plugs,





over-pack and disposal containers, waste matrices,

- The different types of measures : control of the construction of structures, general organization measures for repository, construction measures, natural properties of the site ;
- The redundancy of certain components, essentially seals installed in series ;
- The availability of reserve functions, for example the confinement capacities offered by metal containers, apparently greater than the reference capacity taken into account.

All these arrangements put in place to fulfil the safety functions make up a coherent process requiring a limited number of materials: clay, concrete, steel. Only the main design dimensioning components, directly dictated by safety, have been covered in this chapter. At this point it is necessary to verify that:

- The design measures selected make it possible to meet the safety objectives set by Andra. See chapters 4 for operational safety and 5 for post-closure of TES [vii];
- In a more detailed manner, by consolidating all the components which make up the repository installation, interactions of all types (thermal, hydraulic, mechanic, chemical and radiological), cannot interfere with the operation of the safety functions. See chapter 6 of TES and see the PAMINA topic on uncertainty management;

Beyond the expected evolution, the safety functions also make it possible to cope with situations of an incidental nature, whether it is during operation or in the long term.

WHAT HAS BEEN SUCCESSFUL AND DIFFICULT AREAS

One of the difficulties of performing such an analysis is to make sure that the set of safety functions that is finally obtained is "comprehensive", meaning that all functions that are relevant and may guide the design of the repository are clearly identified. Since a functional analysis is the expression of a certain state-of-the-art knowledge, it is expected that some functions may be overlooked at first go and added later on. But, at any given time, the functions should mirror the reflections of the implementer. One of the difficulties of performing such an analysis is to make sure that the set of safety functions that is finally obtained is "complete". "Completeness" here means that all functions that are important and may guide the design of the repository are clearly identified. Since a functional analysis is the expression of a certain state of the art, it is expected that some functions may be overlooked at one time and added later on. But, at any given time, the functions should mirror the reflections of the implementer. What was used for Dossier 2005 was a method of "flux management" (see Insert 1). Of course, one thinks of the flux of radionuclides through the repository, which is the most important one. But the flux of water may prove important also, even though only small fluxes are expected. The flux of mechanical constraints inside the repository may need to be considered, as the host rock may be damaged by it. On the other hand, it was judged at the time of Dossier 2005 that, for example, taking into account explicitly the flux of corrosion gases and to address it with specific safety functions, was premature. In the light of future knowledge, this position might be reconsidered [xv].





ON GOING OR PLANNED PROJECTS

The basic methodology of the functional analyses will be maintained. Based on the PDD [xvi] the functional analysis will be updated to take into account evolution of scientific knowledge, designing options, and recommendations from the different reviewing group such as « gas ». One first step will be 2009 when choices of safety options will be presented to safety authority.

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Appendix A2: AVN (Belgium)

A2 AVN (Belgium)

[PAMINA] (D-N°: 1.1.1) – Task reports for the first group of topics Dissemination level: RE Date of issue of this report: 15/03/2008





Appendix A2: AVN (Belgium)

European integrated "PAMINA" - Project

WP 1.1 – AVN Contribution "

Safety functions"

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Appendix A2: AVN (Belgium)

1 Introduction

As technical support of the Belgian safety authority, AVN has taken part in discussion concerning the safety approach of the Belgian concepts for near surface radioactive waste disposal and for deep geological high level waste disposal. As such AVN has also participated in the elaboration of two international documents. The first one has been jointly elaborated by the French and Belgian safety authority (ASN and FANC) and their technical support (IRSN and AVN) and by the French and Belgium waste operator (ANDRA and ONDRAF/NIRAS). The title of this document is "Geological Disposal of Radioactive Wastes: Elements of a safety approach". The second document has been elaborated in the framework of an European Pilot Group. The European Pilot Group is made of representatives of the safety authority from Belgium, France, Finland, UK, Sweden, Switzerland, Spain and some technical supports from Belgium, France and Germany. International Organisation like IAEA and EC TREN also participate to this European Pilot Group. The name of this second document, issued in March 2007, is "Report on the European Pilot Study on the Regulatory Review of a Safety Case for Geological Disposal of Radioactive Waste".

The strategy elaborated by the Belgian waste operator for its high-level waste program is described in the Safety Analysis and Feasibility Interim Report (SAFIR 2), issued in 2001 and will be further developed in its Safety and Feasibility Case (SFC-1) foreseen for 2013. This program strategy includes some important points like the choice of safety functions that have to be fulfilled by the disposal system and sub-systems. The implemented safety strategy should amongst others confirm in subsequent steps of the project, the chosen safety functions, and their allocation to different subsystems, components and subcomponents over time.

Although operational safety functions could be identified, in this document we limit our consideration to the long-term safety functions.

2. Definition of terms and used concepts

Reference [1] document provides the following definition of the safety function.

Safety Functions: A function can generally be defined as any action that a system or one of its components must carry out in order to achieve a given purpose. The functions of a disposal system contribute to fulfilling the different objectives assigned to it. Safety functions are those which make it possible to comply with the principles of safety and radiological protection as well as with the basic objective of protection during all stages of the life of the facility, while limiting the burden for future generations.

3. Regulatory Context

According to their role defined by the operator, safety functions concept has been initially developed in order to improve the understanding of the contribution of the different components to the demonstration of the long-term safety of the disposal system. Progressively, as much as the disposal program becomes mature and the operator acquires knowledge and a better understanding of the physico-chemical phenomena, safety functions become more and more a design tool of the disposal





system. Key components of the safety case, safety functions could not be dissociated from safety strategy and the final safety objective.

As such, safety functions, the physico-chemical phenomena they represent, the way they are fulfilled by the components, structures and systems of the disposal along time become a subject of interest for the safety authority.

In accordance to the evolution of the role of the safety functions in the disposal program of the operator, the level and the nature of the information required by the safety authority also evolve.

3.1 Regulations and guidance

In Belgium, no specific regulatory document on the safety function exists until now but as aforementioned above, FANC and AVN have deeply participated to the elaboration of the Franco- Belgium document [1] and to the regulatory review document [2].

3.2 Requirements and expectations

The IAEA safety Standard WS-R-4 is also in force in Belgium. Although the WS-R-4 document does not specifically define the safety functions, IAEA WS-R-4 identifies two safety functions: the containment and the isolation safety functions. IAEA WS-R-4 also states a safety requirement involving safety functions called "Requirements for multiple safety functions".

"Requirements for multiple safety functions": The natural and engineered barriers shall be selected and designed so as to ensure that post-closure safety is provided by means of multiple safety functions. That is, safety shall be provided by means of multiple barriers whose performance is achieved by diverse physical and chemical processes. The overall performance of the geological disposal system shall not be unduly dependent on a single barrier or function.

As a matter of fact, the concept of multiple safety functions directly results from the application of the 'defence-in-depth' concept to the particular case of radioactive waste disposal.

Even if there is no proper regulatory document on this specific topic, safety functions are subjected to regulatory concerns.

As previously said, the selection of safety functions is one of the key-points of the safety strategy and as such, this choice should be clearly explained and justified by the operator in the safety case.

Regulatory expectations evolve with the development of the project.

At early stage of the program, it is expected that an explanation would be provided of how the characteristics and properties of each component are intended to provide for the allocated safety functions and how this will evolve with time. This must be supported by: an overview of the technical feasibility of the proposed design options, by an investigation on how the components of the disposal system will function together in a complementary manner to ensure that

[PAMINA]



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there is adequate defence in depth so that safety is not unduly dependent on a single safety function.

- At siting stage, it is expected that the basic characteristics of the natural and engineered components and proposed design options be described in such a way that they present how the safety functions and the performances expected for each component will be achieved for the site(s) under consideration. This must be supported by consideration again of how the disposal system components together will play a complementary role to ensure that there is adequate defence in depth so that safety is not unduly dependent on a single safety function.
- At design stage, it is expected for example, that the design rules should assist in demonstrating that the likelihood of a components of the disposal system failing is low and that, in the event of degradation, the loss of a safety function of one component does not jeopardize the safety of the whole system, considering the normal evolution of the facility and disturbing events both anticipated and less likely.

Independently of the considered stage, allocation of safety functions to components should provide information on how such allocation evolves with time. Such analysis should be done for each safety-related component, both individually and in an integrated perspective for the expected behaviour of the disposal and for degraded evolutions. Evolution of the status of the safety functions through time should be clearly identified.

3.3 Experiences and lessons learnt

In the framework of the pre-project of LLW disposal or of the deep geological disposal, some discussions between ONDRAF/NIRAS and safety authority address specifically safety functions. In a first stage, documents provided by the operator address more specifically the identification of different safety functions. Although different levels of safety function could be distinguished; fundamental ones were the subject of exchanges. In addition to this identification their evolution and their status in the perspective of the timeframe were also discussed.

Feedbacks from these discussions could be summarized as follows:

- Both operator and safety authority consider that the identification of the safety function should be done early in the disposal program. Safety authority considers that safety function identification should be integrated in the presentation of the proposed safety strategy. Otherwise safety function identification could only be considered in a technical perspective but not in a safety one.
- There is a common agreement on the necessity to develop in a dedicated chapter of the safety case the Safety functions and safety strategy.
- General agreements exist between operator and safety authority on the identification of the main safety functions.
- On some specific topics, Operator and Safety authority have a different approach, e.g. when looking at the "prevention" or "protection" function developed by the operator. This function addresses the water flux inside the





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repository or on how to consider the isolation function in an integrated safety concept.

The way to integrate the isolation function in an integrated safety concept remain • a topic of further discussions.

Through the evaluation experiences conducted so far, the Belgian Safety Authority considers that the use of the safety functions concept should be complementary to the use of multi-barriers concept.

From a regulatory point of view, the review of SAFIR 2 report for deep geological disposal enabled to identify that a key step in the use of safety function is the identification of the different components of the disposal system that contribute to the implementation of the safety functions. Associating a safety function to a disposal component will determine the safety classification of this component. Preliminary discussions on association of safety function to components highlight that if a first guess has to be done in an early stage of the disposal program, the final converged association will be the result of an iterative process where the design, the disposal environment, as well as a preliminary safety assessment will be discussed and presented.

3.4 Developments and trends

Due to their increasing importance in the safety strategy and their international recognition of their roles and based on the experience feedbacks described in §2.3, Safety Authority will increase its review on the role of the safety functions in the perspective of:

- 1. The development of the safety strategy;
- 2. The identification of the different safety functions;
- 3. The implementation of the safety function through disposal component;
- 4. The complementary role of the multi-barrier approach;

5. The elaboration of a classification of safety-related components based on safety functions and the review of its implementation;

6. Their assessment in the radiological impact assessment through the concept of robustness

4. Analysis and synthesis

4.1 Advantages and potential difficulties

Many advantages could be associated to the use of safety functions in a safety case as defined by NEA/OECD.





- Advantages:
 - Improve the communication with the stakeholders as it explains clearly how the safety of a disposal facility could be reached;
 - Ensure that during the long time scale of the disposal project and during each phase of the licensing procedure, the safety strategy will be implemented;
 - Ensure that a structured and sound methodology will be used for the determination of safety related structure, system and components;
- Possible implementation difficulties:
 - The implementation of the iterative approach inside a licensing step
 - The link with the multi-barrier approach.
 - The confidence on the technical feasibility of the safety functions and safety approach

4.2 Feasibility

Two types of feasibility could be distinguished: the Conceptual feasibility and the Technical feasibility.

- Conceptual feasibility addresses how the safety strategy will be implemented.
- Technical feasibility is more linked to the ability of the safety components, structures and systems to fulfil the safety functions assigned to them.

The current status of radioactive waste disposal project in Belgium (Cat.A and Cat. B&C) does not provide sufficient information on the safety functions feasibility. It is thus premature to provide any comments on those two kinds of feasibility.

4.3 Integration in a step-by-step process

The use of safety functions suits very well with the step-by-step approach. No particular comment has to be made on this specific point. At each new licensing step, the operator should taking into account the new data on site characteristics and engineered barriers, the progress on the design and the better knowledge on the disposal facility components, re-evaluate its safety approach and how the safety functions are allocated to system components.

4.4 Data requirements

In a first step, as part of the safety strategy, the development of the safety function does not require any specific data. In the final step "construction licensing step", a complete and sound information on the design, site environment, type of waste to be disposed of and radiological impact assessment results have to be provided.

4.5 Uncertainties

As concept, no specific investigations on uncertainties related to safety functions have to be undertaken. However, the association of safety functions to design components and the confidence that these components will ensure the properties assigned to them



should be subjected to an investigation on uncertainties.

4.6 Improvement potential

No further development is expected on the concept of safety functions. The further improvements are more related to their feasibility.

4.7 Regulatory compliance

Future regulatory compliances will mainly not address the safety functions by themselves. Safety authority expects to receive information on a sound and feasible safety strategy and they should assess, judge and finally approve it. In a next step, safety authority has to receive at the key point of the licensing process, information on key parameters for which they have to determine some criteria or reference values.

4.8 Harmonization – Integration

The safety functions are a tool for harmonisation. No particular comment has to be made on this specific point. However as integration is more related to management considerations, safety authority will focus their review on how the safety strategy and thus the safety functions are integrated through the different applicant teams involved in the project.

5. Referentes

- 1. Bruno, G., De Preter, P., Grevoz, A. Nys, V., Raimbault, Ph., Geological Disposal of Radioactive Waste: Elements of a Safety Approach, Report of a French-Belgian working group gathering ASN, ANDRA, AVN, IRSN, FANC, ONDRAF, AVN, September 2004
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- 5. De Preter, P., Lalieux, P., Cool, W. et al., Safety Assessment and Feasibility Interim Report 2 (SAFIR 2), ONDRAF/NIRAS NIROND 2001-06 E, December 2001





Appendix A2: AVN (Belgium)

6. Annexes

From reference [1 chap 7]

The primary safety functions of the disposal system are established by the implementer during the design of the disposal system. Subsequently, this allows the implementer to optimise its design in terms of long-term safety through the successive iterations of its safety case. In Belgium and France, the primary safety functions of the disposal system are identified as the functions of "isolation", "containment" and "limitation and retardation".

During the development of the safety case the iterative process that associates the safety functions to the various components of the disposal system is conditioned by the implementation of Safety Principles ("Defence-in-depth" and 'Demonstration") and Radioprotection Principles (cf. ICRP) and by the integration of external constraints imposed on the programme. The application of safety principles, in particular through the concept of robustness, is one of the driving forces of the iterative process for the association of the safety functions to the components of the system.

The radiological impact depends on the properties of the disposal system and its environment. The role of the disposal system's environment is distinguished from the safety functions linked to the disposal system itself by the fact that the environment capacity to reduce the peak flux of the radionuclides is not optimised during the design of the disposal system for two main reasons. The characteristics of the disposal system environment, and therefore its role on the peak flux, are an indirect consequence of the site selection process; as such this role is considered to be imposed and cannot be optimised.

The interpretation of the results of the safety assessments from the safety function point of view leads to the following conclusions: the safety functions do not at all participate at one and the same time in the safety of the disposal system. So, various possible states of safety functions can be considered depending on whether they participate actively or latently in safety, or whether they are considered not to be an effective part of the safety case (reserve safety function).

A "latent safety function" can be defined as a function that becomes partially or totally active only when other safety functions do not or no longer achieve the expected performances.

A "reserve safety function" is a function that, at a given time, is not sufficiently well characterised to be fully relied upon in the safety case, but whose existence contributes to confidence in the overall safety of the repository.

It is now recognised that the safety of a repository relies more on concepts of the complementarity and redundancy of functions, than on the concept of the redundancy of barriers.

Each component of the repository can contribute to fulfilling one or more safety functions with a certain level of performance for each one. The assigning of these functions to different components depends on the choices made by the implementer,





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the phenomenological knowledge available and the understanding of the functioning of the overall disposal system. Functions are defined in terms of well-known phenomena or characteristics and operate over long periods of time. A component can, at a given time, fulfil a latent safety function, then go on for a certain period of time to fulfil an active safety function and finally reach a point where this is no longer fulfilled. All of the functions together must at all times ensure the protection of man and the environment.

In the framework of the iterative design approach and in the more advanced stages of the programme, the safety functions could be used in advance for the revision and optimisation of the designs studied.





Part 1: Task report Safety Functions

Appendix A3: ENRESA (Spain)

A3 ENRESA (Spain)

[PAMINA] (D-N°: 1.1.1) – Task reports for the first group of topics Dissemination level: RE Date of issue of this report: 15/03/2008





Part 1: Task report Safety Functions

Appendix A3: ENRESA (Spain)





Proposal/Contract no.: FP6-036404

Project acronym: PAMINA

Project title:

PERFORMANCE ASSESSMENT METHODOLOGIES IN APPLICATION TO GUIDE THE DEVELOPMENT OF THE SAFETY CASE

Instrument: Integrated Project

Thematic Priority:

Management of Radioactive Waste and Radiation Protection and other activities in the field of Nuclear Technologies and Safety

WP1.1 Safety functions ENRESA contribution

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- PU Public PP Restricted to other programme participants (including the Commission Services) PE Postricted to a group apositied by the conservitium (including the Commission
- RE Restricted to a group specified by the consortium (including the Commission Services)
- CO Confidential, only for members of the consortium (including the Commission Services)

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Х



1 Background and introduction

This document describes the experience of Enresa regarding the use of Safety Functions in the Performance Assessment (PA) of HLW repositories in granite and clay. The scope of the present document is circumscribed to the use of Safety Functions in Enresa's most recent Performance Assessments of spent fuel repositories: ENRESA 2000 [2] for a granitic formation and ENRESA 2003 for a clay formation [3].

2 Regulatory requirements

The acceptance criteria for radioactive waste disposal facilities was set in 1987 by the following statement of the regulatory authority (CSN): "to ensure safety individual risk should be smaller than 10⁻⁶yr⁻¹, that is the risk associated to an effective dose of 10⁻⁴ Sv/vr".

There are no specific requirements on safety functions.

3 Key terms and concepts.

The term Safety Function is not used explicitly in Enresa's PA exercises [2] and [3].

4 Treatment in the Safety Case

4.1 Methodology

The term Safety Function is not used explicitly in Enresa's PA exercises. The concept of Safety Function is not used to structure the evaluation, but some functions or properties of the barriers were identified a posteriori as relevant for safety, and have been included here as safety functions.

4.1.1 Broad Safety Functions

Overall, two broad safety functions in a repository system are identified:

- **Isolation of repository components:** consists in the protection of the • repository components against negative environmental conditions, which could impair their intrinsic condition and/or their performance.
- Containment of radionuclides: consists in avoiding or limiting the transport





and release of the hazardous materials disposed of in the repository.

Barriers are those repository components that perform the safety functions.

There are processes that have influence on the performance of a repository system, but which are essentially independent of a repository concept, and are not amenable to optimisation. Examples of these are the dilution capacity of water bodies in the surface or near the surface, and the transfers between biosphere compartments. They are accounted for in the safety assessment in order to estimate the relevant performance indicators but the physical features responsible for those processes are not considered as barriers.

Isolation is most frequently associated with providing favourable boundary conditions for the longevity and performance of the inner barriers, and avoiding the potential adverse effects of external actions. The most prominent case in this category is the host rock assuring protection of the engineered barriers against the chemical, hydraulic, mechanical, thermal and biological conditions prevailing in the biosphere. But in this category of functions can be placed for example the protection (mechanical, chemical, hydraulic and biological) provided by the buffer to the waste package (to the canister, and to the waste form after failure of the former).

Containment may be accomplished by a physical barrier, which cannot be crossed by contaminants, as it is that provided by the canister wall (absolute containment). A second category of containment is the slow release of radionuclides from the waste form (UO_2 matrix and structural materials of the fuel). The third class of containment is provided by the retardation of the transport of radionuclides, which has two useful effects:

- Retardation leads in general to a decrease of the fluxes leaving the near field or the far field, which magnitude is a function of the travel time through the barrier to half-life ratio. Obviously, the retention of radionuclides in a barrier allows for the accumulation of contaminants in it, increasing the inventory available for release at a later stage if environmental conditions change.
- Retardation helps to spread releases over long time periods time, and in this way reduce the maximum release rates, even in absence of decay. For instance, the bentonite buffer acts in this way limiting the release rates to the host rock of the radionuclides in the instant release fraction (IRF) of the spent fuel.

We may think of isolation assessment as dealing with the barrier evolution and the boundary conditions, while containment assessment deals with the performance of the barriers regarding radionuclide release, transport and impact. Both types of assessments are necessary to support the safety case.

4.1.2 Safety functions and safety requirements

The disposal system is a multi-barrier system where both engineered and natural barriers contribute to the overall safety through a diversity of safety functions, so that the uncertainties and variability (both in time and space) in the performance of any barrier are compensated with the margins available in the performance assured by the others. This feature of geological repositories corresponds with the principle of





defence-in-depth, which is paramount in nuclear safety.

The multi-barrier concept contributes to system robustness, which in turn confers robustness to the safety analysis. They both are mirror concepts. System robustness can be understood as the capability of the repository system to comply with safety criteria with wide safety margins and low sensitivity to the performance of individual barriers. Robustness of the assessment includes the use of reasonable pessimistic assumptions in the safety analysis and the existence of additional safety functions which are not actually accounted for in the assessment (reserve safety functions).

Regulations generally only specify safety requirements for the total system in the form of individual dose (or risk) constraint. Usually, there are not specific regulatory requirements applicable to any individual barrier.

Robustness of the reference system is usually illustrated through analyses consisting in assuming arbitrarily that a single (containment) safety function of the system is lost or seriously degraded. These analyses are tools to understand the robustness of a given repository system, and in no way should be understood as:

- verifications of any safety criteria (that do not exist), nor
- meaning that any safety function is unimportant in itself, since all of them are part of the multi-barrier concept and, beyond that, contribute to the confidence in the safety of the system.

A clear identification and justification of the many safety functions provided by repository can be a strong argument in support of the safety case.

4.1.3 Safety functions of the different barriers of a repository

Typically a spent fuel repository in granite or clay comprises the following physical barriers:

- the waste form,
- the canister,
- the bentonite buffer, and
- the host rock.

On the base of Enresa PA exercises in granite [2] and clay [3] the safety functions fulfilled by the different barriers in a repository have been identified and are presented in the next sections.

4.1.3.1 The waste form (spent fuel)

The spent fuel is the first barrier to retain the radionuclides, which are released at the rate of degradation of the different parts that make up the fuel assemblies. Only a part of the activity in the spent fuel (the so called Instant Release Fraction) is assumed to be released at the time of canister failure.

The only safety requirement relative to the different spent fuel parts is their own rate of



degradation, which is assessed separately for the metallic parts and for the UO2 matrix. Degradation only starts once the absolute containment provided by the container is lost.

The spent fuel provides the following safety function that is included in the PA:

Slow release of the great majority of the radionuclides in the waste form, with the • exception of the IRF.

In addition, the zircaloy cladding provides two reserve safety functions, which have been identified but not included in the PA:

- The cladding of most fuel rods is expected to remain watertight when the • canister fails, providing an additional barrier that must be breached before the IRF inventory is released and the matrix alteration begins.
- Even after failure of the cladding, the long zircalloy tubes can represent a useful • physical barrier for the transport of radionuclides from the UO₂.

4.1.3.2 The canister

The carbon steel canister provides the following safety functions, which are included in the PA:

- Provide absolute containment of the waste during the operational phase (operational safety) and the thermal transient phase.
- Ensure a reducing environment in the near field due to the high amount of iron • present.
- Delay the start of the leaching of the UO_2 matrix, allowing alpha activity to decay. With the UO₂ matrix alteration model used in Enresa's exercises, the UO₂ alteration rate when the canister fails decreases with the duration of the canister.
- Gradual failure of the canisters along a significant time period. Spreading canister failures is a useful mechanism to avoid the simultaneous release of the IRF inventories in all the canisters. Although this safety function usually is not considered in PA exercises, it has been included in Enresa's exercises and found to be useful

In addition, three reserve safety functions of the canister have been identified but not included in the PA:

- Provide a physical barrier against radionuclide transport after canister failure. In • Enresa's PA exercises no credit is given to the canister after failure: canister "disappears" after the failure.
- Provide a high hydrogen partial pressure around the waste, leading to a • dramatic reduction of the UO₂ matrix alteration rate. This hydrogen is produced in the anaerobic corrosion of the carbon steel container.
- The thick layer of corrosion products formed from canister material will have a very small porosity and provide strong sorption for many radionuclides. This layer has the potential to further retard the transport of radionuclides.





In the conceptual design of the canister, glass beads are used as filling material that provide two reserve safety functions that have been identified but not included in the PA:

- Provide a considerable supply of silica, which may contribute to favourable reactions (i.e. formation of insoluble uranium compounds).
- Reduce the empty space inside the canister that could be filled by water or bentonite when the canister fails. This reduces the risk of criticality and the loss of bentonite by intrusion into the canister cavity.

4.1.3.3 The bentonite buffer

The bentonite buffer is required to perform a large diversity of safety functions, which can only be fulfilled once the bentonite saturates and swells. As the safety functions provided by the buffer are accounted for the full duration of the quantitative safety assessment (in the scale of the million years) its properties have to be preserved at a sufficient level for commensurable periods of time.

The buffer acts as an **isolation barrier** that protects and limits the disturbances to the inner barriers, through different safety functions:

- Isolate mechanically the canister from limited shear displacements in the disposal drift walls. In the reference case of ENRESA2001 [2] shear faults are not expected, so this is a reserve function.
- Keep the canister in place in the middle of the disposal drift (prevent canister sinking).
- Inhibition of microbial growth that could contribute to canister corrosion.
- Limitation of the flux of external reactant that can reach the canister contributing to its corrosion. For instance, the bentonite buffer limits the flux of carbonate from natural groundwaters reaching the canister surface, and the formation of siderite is limited too (although other corrosion products can be formed).
- Avoid the build up of excessive gas pressure in the near field, without undue impairment of other safety functions.
- Transfer radiogenic heat from the waste package to the host rock, avoiding excessive temperatures (due to good thermal conductivity).
- Buffer groundwater composition to ensure that the water that contacts with the canister first, and the waste later, is not aggressive for those barriers.

The buffer acts as a **containment barrier** that limits radionuclide transport on the base of its properties, through different safety functions:

- Elimination of advection in the buffer (due to low hydraulic conductivity). As a consequence, only diffusive transport is possible in the buffer.
- Buffer groundwater composition so that the composition and other characteristics of the water that contacts the waste ensure very small solubilities for many radionuclides (such as actinides).
- Filtration of colloids and large complex molecules (due to the small size of the





pores)

- Inhibition of microbial growth (due to the small size of the pores and low water activity). Radionuclide transport by micro-organisms (or colloids) can significantly reduce the retardation of radionuclides in the bentonite buffer.
- Retardation of transport due to good sorption properties for many radionuclides.

4.1.3.4 The host rock

The far field rock acts as an **isolation barrier** that protects and limits the disturbances to EBS through different safety functions:

- Provide a favourable chemical environment to the EBS.
- Provide mechanical protection to the EBS.
- Transfer radiogenic heat out of the near field, avoiding excessive temperatures (due to good thermal conductivity).
- Avoid the build up of excessive gas pressure in the near field, without undue impairment of other safety functions.
- Limit water flows close to the near field (for a repository in granite) to avoid damage to the buffer.

The far field rock acts as a **containment barrier** that limits radionuclide transport on the base of its properties, through different safety functions:

- Small water flows close to the near field (for a repository in granite) to limit radionuclide releases from the near field.
- Slow water transport through the formation for a repository in granite.
- Small groundwater flows through the formation in the case of a repository in clay. Small hydraulic conductivity of the clay ensures that radionuclide transport is controlled by diffusion, and the effect of advection is small.
- Retardation of transport due to good sorption properties of the host rock (or fracture coating/infill in granitic formations) for many radionuclides.

4.2 Related topics

Safety functions can play a central role in the "assessment strategy and the safety approach" of deep geological repositories and can be useful in the "identification of scenarios".

4.3 Databases and tools

Not applicable.





4.4 Application and experience

The concept of Safety Function is not used explicitly in Enresa PA exercises, but some functions or properties of the barriers were identified a posteriori as relevant for safety, and have been included here as safety functions (section 4.1).

4.5 On going work and future evolution

Enresa is not doing any in-house developments on this topic.

Recent Safety Assessments done in France, Sweden, Belgium and Switzerland use the concept of safety functions for different purposes within the assessment. There is a clear trend to increase the role of the safety functions concept in the Safety Case and Enresa is strongly interested on international developments on this topic.

5 Lessons learned

Although the concept of "safety function" is not explicitly used in Enresa's PA exercises, these exercises have allowed identifying several properties or functions of the repository system that are relevant to safety.

A clear identification and justification of the different safety functions in the repository can be very useful to support the Safety Case, showing the role of the different barriers and the intrinsic robustness of the disposal system. Safety functions can be very useful when presenting the results to different audiences, too.

Not only the safety functions considered in the Reference Scenario should be included in the Safety Case, but also other "reserve safety functions" should be identified. These reserve functions are useful to show that there are additional safety margins, although some of these reserve safety functions can be hard to model or demonstrate.

6 References

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- [3] Enresa 2003. Evaluación del Comportamiento y de la Seguridad de un Almacén Geológico Profundo de Residuos Radiactivos en Arcilla. 49-1PP-M-A1-01 Rev.0. February 2004. (In Spanish)





Part 1: Task report Safety Functions

Appendix A4: GRS-K (Germany)

A4 GRS-K (Cologne, Germany)





Part 1: Task report Safety Functions

Appendix A4: GRS-K (Germany)





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Project acronym: PAMINA

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Thomas Beuth, GRS

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PU	Public	
PP	Restricted to other programme participants (including the Commission Services)	Х
RE	Restricted to a group specified by the consortium (including the Commission Services)	
СО	Confidential, only for members of the consortium (including the Commission Services)	

[PAMINA] (D-N°: 1.1.1) – Task reports for the first group of topics Dissemination level: RE Date of issue of this report: 15/03/2008





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1 Background/ Introduction

In recent years safety functions got more and more essential roles e.g. for repository design, safety assessments, and as additional or further qualitative arguments in the safety case concerning deep geological disposal of radioactive waste. At the outset, safety functions were primarily used to explain and describe the complex mechanisms of a deep underground repository. In the course of time it was realised that safety functions have quite more to offer, than only to be used for illustration reasons.

From expert opinion, approaches based on safety functions have the potential to overcome certain drawbacks of the multi-barrier approach. Therefore, safety functions are taken into account in several national programmes /NEA 06/.

In Germany, the requirement for an isolating rock zone implies a safety concept in which the main safety functions are preferably carried out by the natural (geological) components together with geotechnical barriers (e.g. shaft seal) of the repository system.

As part of the development of criteria and guidelines for demonstrating the safety case, an assessment system based on safety functions is currently being developed. The safety of the repository system will be appraised in consideration of the respective defined safety functions. This can be done both in a qualitative and quantitative way. For the quantification of a safety function a measure and corresponding value is needed for the evaluation whether the safety function under consideration fulfils the respective task or not. Therefore, appropriate measurable properties (safety function indicators) have to be found as well as their corresponding quantitative limits (safety function indicator criteria).

As outlined in the Annex I "Description of Work" of the Integrated Project PAMINA the tasks in WP 1.1 will be carried out by bringing together and by including the perspectives from both the "developers" and the "evaluators". For this reason each task will be addressed by the "development working group" (DWG) and by the "evaluation working group" (EWG) whereas the latter group will be the working platform for GRS Köln.

Thus the present draft document includes the background, fundamentals, and the regulatory basis as well as recent developments in revising the existing regulations "Safety Criteria" /BMI 83/ from 1983 concerning the topic "Safety Function".

2 Definition of terms and used concepts

The defined terms and used concepts in the frame of safety functions are as follows /BAL 07/:

Safety Function

Safety function is a function, which takes over safety relevant requirements, in a safety related system, subsystem or single component. Through interaction of such functions the confinement (isolation) as the primary safety function of the **repository system** is guaranteed as well as the compliance with safety principles and protection objectives





both in the operational phase and post closure phase of the repository.

Repository system

The repository system comprises the **repository** and its geological environment, which in turn includes all rock areas that have to be considered for the compliance proof of the safety principles and protection objectives for final disposal.

Repository

The repository is part of the repository system in which high active waste will be placed. It comprises the repository mine, the host rock and the **isolating rock zone**.

Isolating rock zone

The isolating rock zone is part of the geological barrier which at normal development of the repository and together with geotechnical barriers (shaft seal) have to ensure the confinement of the waste.

It should be noted that the term safety function is also part of the definition of scenarios (cf. 2. of the contribution of GRS Köln to the topic "Definition and assessment of scenarios").

3 Regulatory context

Presently, the management of radioactive waste in Germany is under review. It is the policy of Germany that radioactive material should be concentrated and contained rather than released and dispersed in the environment. According to the international consensus that long-lived radioactive waste has to be disposed of in deep geological formations in order to guarantee that man and the environment are protected in the long run from the effects of ionizing radiation by isolation of the radioactive waste. In Germany all types of radioactive waste have to be disposed of in a deep repository.

Amongst the important cornerstones of the new waste management plan is a revision of the "Safety Criteria for the Disposal of Radioactive Waste in a Mine" /BMI 83/ (in the following named as "Safety Criteria") which were issued in 1983 /BAL 06/.

As indicated, the German "Safety Criteria" are at present under revision on behalf of the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU) in order to account for the progress in safety-related developments and procedures, e.g stepwise approach, constrained optimisation, and "Safety Case" methodology. The revision of the "Safety Criteria" as well as the development of supporting guidelines is carried out by the Final Disposal Department of GRS Köln with the support of a number of experts from Germany and abroad. The revision accounts for the ideas and requirements given in the OECD/ NEA report "Post-closure Safety Case for Geological Repositories" /NEA 04/ and in the IAEA safety requirements guide WS-R-4 (formerly known as DS-154) /IAE 06/.

In the following sections the regulatory framework and the ongoing work concerning safety functions will be shown. Specific topics which strongly relates to safety functions like "Safety Function Indicators" are described regarding their context but will be addressed in detail separately as part of the topic "Safety Indicators and Performance/





Function Indicators".

3.1 Regulations and guidance

The legal basis for licensing is the "Plan Approval Procedure" required by the German "Atomic Energy Act" for federal installations for the safekeeping and final disposal of radioactive waste. The "Plan Approval Procedure" has a so-called "concentrating effect" for several fields of law and will generally lasts for the whole duration of a project.

In application of the "Plan Approval Procedure" in respect of deep geological repositories the formulated "Safety Criteria" /BMI 83/ have to be considered. These "Safety Criteria" do not stress the term "Safety Function" directly, due to the fact that this term was no subject at this time at all. However, the topic "Safety Function" was addressed in the figurative sense, insofar as the overlying rock and the adjoining rock have to fulfil a barrier function. This barrier function should have a low conductivity and a high sorption capacity in order to avoid unduly concentrations of radionuclide releases from the repository mine into the biosphere. Furthermore, it was recognised that water paths between the biosphere and the operated repository constitutes a potential for radionuclide releases. Such potential paths may be for disposal host rocks at the most so low, that the protection function of the geological and technical barriers persists. These statements from the formulated "Safety Criteria" in 1983 show, that the basic idea of safety functions has already exists. However, these first indirect steps towards safety functions were far away from the approach considered today. At present the "Safety Criteria" are being revised.

Recent results of the revision work of GRS Köln were documented in a draft report "Safety requirements for the disposal of high active wastes in a deep geological formation" /BAL 07/ (in the following named as "Safety Requirements") and discussed on a workshop held on 6 and 7 March 2007 in Hannover, Germany. The proposal for the criteria revision is however still being reviewed by advisory bodies and might therefore undergo further changes. A final draft for the proposal of the revised "Safety Criteria" is not available so far. Earlier stages of the development are reflected in several published documents /BAL 04a, BAL 04b, BAL 05a, BAL05b, EUS 06/.

The statements presented in the following sections relate to a large extent to the above mentioned "Safety Requirements" /BAL 07/. As indicated before this proposal has a draft status and should therefore be seen as a preliminary work with no binding regulatory basis. However, the document includes the recent developments in the field of regulatory requirements on the basis of broad and thoroughly performed discussions and exchange of information and experience with experts from Germany and abroad.

3.2 Requirements and expectations

Primary safety function

Central safety function of the repository system in all stages of the development of the repository is the confinement (isolation) of the high active wastes.

Basic requirements

The repository system has to be robust in terms of its central safety function, i.e. the



sensitivity against effective events and processes as well as uncertainties, has to be small.

The repository system has to guarantee the safety, by a system of graduated defence measures and their safety functions both in the operational phase and in the post closure phase.

Components, without approved technical rules, that exert a safety function in the repository system, have to be tested. If the testing is not practicable, the suitability has to be technically scientifically justified and corresponding safety reserves have to be provided.

The safety functions have to be described and assessed in a long-term prognosis, taking into account the determined potential developments of the geological barrier system from the geological long-term prognosis. It has to be presented that the safety functions of the technical barriers are effective over the demanded time periods.

Site specific requirements

The repository should be established in a sufficient depth, to the protection against implications of future evolutions on the site, e.g. glacial periods or uplift with erosion, so that the safety function of the isolating rock zone will not be affected in the demonstration period.

The site must allow a good predictability of the long-term evolution of the site conditions and site characteristics. The dynamic of geological processes, which the site underlies today and was underlying in the past, have to be clear so far, that out of it a geoscientific long-term prognosis for the site and especially for the isolating rock zone, in terms of its safety function for the demanded demonstration period of one million years, can be derived. Thereby the geoscientific long-term prognosis has to identify, describe and in terms of safety assess the potential future developments of the geological barrier system and its safety functions, due to internal and external causes. The influence of the geological barrier system and its safety functions, due to the construction of the repository mine and the emplacement of high radioactive wastes, has to be taken into account.

Safety functions in connection with grouping scenarios

Scenarios with similar developments taking place may be summarised to scenario groups and shown by a representative scenario. Prerequisite for it is that the effects from the representative scenario on the safety functions of the repository system cover the effects of the group.

3.3 Experience and lessons learned

Due to the pretty new development and the unfinished discussion of the revision document no experience exists until now, regarding the implementation and application of regulations in terms of safety functions.





3.4 Development and trends

BMU has to implement the regulatory guidance in consideration of the state of the art. According to this task, GRS is involved in a R&D-Project called "Comparative Safety Analyses for Repository Sites for the Assessment of Methods and Instruments" (VerSi) since September 2007. This project consists of four subprojects which cover conceptual work, scenario development, long-term analysis and evaluation. The overall objective of the project is the provision of appropriate methods and tools for the comparison of repository concepts in different host rocks e.g. clay and salt.

In the framework of the subproject scenario development the derivation of scenarios in consideration of safety functions is one of the main tasks. The proposed procedure comprises the following steps:

- Identification and selection of potentially relevant FEPs in consideration of existing databases (national, international) extended by the choice of site and concept specific FEPs.
- Definition of safety functions taken into account the repository system including repository design, disposal concept, isolating rock zone and geology.
- Assignment of safety functions to the repository system, subsystems or single components.
- Concentration of the compiled FEP database by selection of relevant influencing FEPs on defined safety functions.
- Development of scenarios in conjunction with the results from the previous step.
- Combining of developed scenarios to representative scenarios and classification
 of combined scenarios according to the scenarios classes (likely scenarios, less
 likely scenarios, scenarios that need not be considered any further) described in
 the topic "Definition and assessment of Scenarios".

4 Analysis and synthesis

This section describes the frame of safety functions as a basic element in the safety case from a regulatory perspective, which is still in discussion /BAL 07/.

Application areas of safety functions

As above mentioned, safety functions seem to be very useful in a multiple way. Some common areas of application are:

- In support of the repository design and planning phase.
- Identification of key safety issues.
- As a means to communicate safety aspects with different stakeholders.
- Illustration of complex connections between systems, subsystems and single components of the repository concept and the geological environment.
- Identification of requirements for R&D- work.





- Approach for the derivation of scenarios.
- Construction of What-If-Cases for the representation of the robustness of the repository system.
- Check list for developed scenarios.
- Identification and collection of supplementary gualitative arguments for the safety case.

There are no regulatory requirements for using safety functions and how they should be used in the listed areas above. It is left to the implementer to decide, whether or not safety functions should be included for whatever reason. However, all aspects of regulatory requirements related to safety functions described in section 3.2 have to be taken into account by the implementer provided the current discussed requirements get a legal status.

Focus of the repository concept

The preferred option in Germany is a repository concept based on a favourable overall geological setting for which the isolating rock zone as the main geological barrier and the shaft barrier will take the main part of isolation as the primary safety function while other technical barriers have a supplementary function. Thus, a system of multiple safety functions with emphasis on the main barrier have to be defined, settled and analysed. This process might undergo several iterations.

Development of safety functions

For each developed safety function a comprehensive description is required. The description has to comprise the background and motives of respective developed safety functions, and should be documented in a transparently, reasonable, consistently and clearly manner. Furthermore, the potential connection to subsystems, components or single component of the repository system has to be specified.

Break down of safety functions into sub safety functions

A common procedure in the development of safety functions is to set up so called primary and secondary functions such as isolation and retardation, which complies the main safety objective of the repository system. Due to the scope and complexity of the repository system further subdivision of safety functions is required for the investigation of subsystems, components and finally for the proof of the fulfilment of main functions. This process can be repeated as often as necessary according to the investigation level in guestion. In principle the subdivision of subordinate safety functions can be almost continued indefinitely. There are only practical reasons for a limitation.

Features, interaction and dependencies of safety functions

The characteristics of safety functions should be taken into account in the context of the safety case. Some essential characteristics of safety functions are listed in the following:

Safety functions can be assigned to repository components or subsystems, whereas several components form a subsystem.





- More than one system component can contribute to a single safety function.
- A single system component can contribute to more than one safety function.
- Some safety functions contribute to safety at all times considered, while others contribute over limited time frames.
- The loss of one or more safety functions does not necessarily mean that the safety of the entire repository system is compromised.
- The fulfilment of a safety function may depend on the fulfilment of other safety • functions.
- Some safety functions will only appear if one or several safety functions have lost their effectiveness.
- Safety functions can have a more or less strong dependence in the following forms:
 - Safety function is dependent from another safety function.
 - Safety functions are interdependent.
 - Safety function is dependent from another safety function via a third safety function.

Completeness and comprehensiveness

In case of scenarios derived on the basis of safety functions, the question of a complete and comprehensive consideration of relevant safety functions arises. Like for other aspects, e.g. FEPs, the completeness and comprehensiveness cannot ultimately be proved. However, it has to be represented credibly, that all essential safety functions were defined and analysed.

Assessment of fulfillment of safety functions

The assessment of the fulfillment of safety functions requires the involvement of quantifiable measures and values. Such quantifiable measures and values can be termed according to the "SR-Can" report safety function indicators and safety function indicator criteria. Both safety function indicators and safety function indicator criteria are an essential part for the safety analyses. The derivation of safety function indicators is sometimes very difficult if not impossible. In such cases an alternative safety function, e.g. by subdividing the respective safety function or a substitute for the safety function indicator is needed.

The following example should illustrate the derivation of a safety function indicator:

The safety function of the isolating rock zone as a subsystem of the repository consists in the retardation of radionuclides. Is the isolating rock zone composed of porous rock, which allows the migration of radionuclides, it is possible to draw conclusions on the isolation capacity of the isolating rock zone in consideration of the concentration of the migrated radionuclides uranium and thorium from the emplacement spot to the point of the border area, which indicates the transition from the isolating rock zone to the overlying rock and adjoining rock. The safety function indicator in this example is therefore the concentration of the radionuclides uranium and thorium at the border between the isolating rock zone and the overlying rock and adjoining rock. A limiting





value for the concentration determines whether the fulfillment of the safety function is given or not, indicates the safety function indicator criterion.

Further details are included in the contribution of GRS Köln to the topic "Safety Indicators and Performance/ Function Indicators".

Robustness of the repository system

Robustness of the repository system is the insensitivity of safety functions of the repository system against internal and external effects and disturbances as well as uncertainties.

The loss of one or more safety functions should be presumed for analysing the robustness of the repository system. This might contribute to increase the confidence in safety of the repository system.

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Part 1: Task report Safety Functions

Appendix A5: IRSN (France)

A5 IRSN (France)

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Appendix A5: IRSN (France)

Safety functions

1. Background/ Introduction

Originally developed for the nuclear reactors, the principle of defence-in-depth consists of the implementation of complementary or redundant levels of protection applied to all the nuclear activities. Within reactor context, this principle is fulfilled with the implementation of successive containment barriers disposed to compensate for the potential loss of one of them allowing protecting man and the environment. Moreover, the principle of defence-in-depth adapted to the nuclear reactor introduces the possibility of human intervention. On the contrary, when applied to a deep geological disposal system and due to the impossibility of human intervention, the derivation of the principle of defence-in-depth leads to the implementation of "multiple safety functions" provided by different mechanisms and disposal components. Therefore, the functions contribute to the development of a strategy, developed by the implementer, notably by allocating the safety functions to the different components of the disposal system and the mechanisms and by justifying the allocations in the safety case.

2. Definition of terms and used concepts

As described in the report [5], the safety functions are defined as functions allocated to disposal components must contribute to fulfilling the different safety objectives assigned to the disposal system. Thereby, the safety functions, with respect to the objectives of the post-closure phase, must allow complying with the principles of safety and radiological protection, while limiting the burden for future generations.

The principle of multiple safety functions is based on the fulfilment of several different functions by a single component. In other words, a component may fulfil several functions in the same time or successively over different time scales. Moreover, a same function may be fulfilled by several different components in a complementary manner or in a redundancy manner. The design must also integrate that a component can fulfil a safety function at a given time, and then, due to the slow degradation, isn't able to fulfil its allocated function. Designing and developing a disposal system on the basis of the multiple safety functions aims at not compromising the safety of the system by losing one component's function. All of the functions together must at all times ensure the protection of man and the environment. That purpose leads to introduce the notion of backup or latent safety functions replacing the loss of safety function due to component degradation or minimizing their effects on the system performance. As an example of latent safety function, the confinement ability of the glass matrix is put to the test after the corrosion of the overpack.

For the management strategy of "containment and concentration", the safety functions





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defined are: isolation, containment, limitation and retardation. In the frame of the French disposal project, the safety functions, jointly defined by French implementer, assessor and regulator in the current release of the BSR 3.2.f (Basic Safety Rule [2]), are: isolation, containment and prevention of the water circulation. The "limitation and retardation" function coming from "concentration and containment" strategy is included in the "containment" function as a sub-function.

3. Regulatory context

Today, the regulatory requirements are provided by the BSR3.2.f (Basic Safety Rule relating to the disposal of radioactive wastes in deep geological formations [2]) edited in June 1991. The new version under progress completes the latter by introducing the notions and safety approaches developed partly in the 2005 Clay Dossier of ANDRA, particularly the notion of safety functions. It is currently issued jointly by ANDRA, ASN and IRSN. The approach recommended by the current BSR3.2.f as well as the new issues under discussion in the framework of the release of BSR3.2.f are used as a basis for developing present IRSN approach.

The allocation of the safety functions to the disposal components is considered to be part of the safety case, and then is taken into account in the strategy used for the development of the evolution scenario. In fact, as described in the under progress BSR3.2.f [2], the allocation of the safety functions is involved in the two first steps of the iterative approach concerning respectively the verification of the performance of the disposal components and the assessment of the disturbance due to the interactions between components. The development of the scenarios is involved in the third step regarding the modelling of the disposal system behaviour.

a) Regulations and guidance

The containment system described in the current BSR 3.2.f consists of three barriers (the waste packages, the engineered barrier and the host rock) placed between the radioactive wastes and the biosphere.

The following three barriers are defined in the current BSR 3.2.f:

- The waste packaging. This generally consists of a matrix in which the waste is • incorporated, placed in a container and possibly in an over-pack.
- Engineered barriers. These consist of the materials used for plugging the • disposal chambers and shafts, backfilling the drifts and sealing the access shaft.
- The host rock. This consists of the low permeable geological formations hosting • the disposal.

Those barriers of the containment system are supposed to play complementary roles, the main one being the host rock, particularly in the long term.

The approach based on the "multi-barrier" must being better adapted to a repository concept since the components can't provide a full containment all along the postclosure and since a failure of a barrier won't necessarily reduce the safety of the





repository system. Therefore, the "multi-barrier" approach is currently being modified by the notion of safety functions associated to repository components [2].

The approach discussed currently by French organizations and nuclear safety authority to derive the future safety functions that will be described in the new release of the BSR3.2.f is based on the following safety key points:

- To prevent the water circulation inside the repository
- To contain the radioactivity in the repository by :
 - Avoiding the dissemination of the activity (or a part of the activity) contained in a component outside of this component
 - Limiting the release of the radioactive substances and immobilizing them into the repository
 - Delaying and attenuating the radionuclide migration
- To separate the radioactive wastes from man and the biosphere so that the repository system safety shall not be affected either by the erosion phenomena or by ordinary human activities.

The "prevention of the water circulation in the repository" consists of limiting the regeneration of water in the vicinity of the canisters since fresh water contributes to degrade them. Therefore, the components (host rock, engineered components) and the design of the repository have to limit the velocity of the water flow and have to avoid advective dominated transport regime inside the repository.

The "containment of the radioactivity" means that the containment system has to avoid or limit the transfer of the radioactivity outside of the components assumed to fulfil this safety function. However, regarding the time scale associated to the post-closure phase, the full containment is hardly conceivable due to the degradation of the components and the potential human intrusion. As a matter of fact, the containment is ensured by avoiding the dissemination of the radioactivity, and also by limiting the radionuclide release from waste matrix and immobilizing them or delaying and attenuating the radionuclide migration. In order to allow radioactive decay of soluble radionuclides, it is therefore important that the total containment be insured by the tightness of the waste canisters (typically more than 500 years).

The host rock itself contributes to delay and attenuate the transfer of radioactive plume thanks to adequate chemical and hydrogeological properties, including stability. In addition to those properties, the depth of the host rock must allow separating the wastes from the biosphere; it shouldn't be degraded by the geomorphological processes as long as the dissemination of radioactivity allows reaching significant dosimetric impact.

b) Requirements and expectations

As is done in the multi-barrier approach, the principle of the multiple safety functions must lead, at least, to the development of a design preventing the simultaneous failures of several different components by analysing their causes and their potential





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consequences. The implementer must establish the safety functions allocated to the disposal components in order to optimize the design in terms of long-term safety on the basis of a stepwise process by demonstrating, at each step of the safety case, the fulfilment of those safety functions.

Within the iterative process, safety functions should conduct to describe easily and succinctly the functioning of the disposal system and allow the implementer to analyse this functioning in a more systematic manner. Each component of the repository can contribute to fulfilling one or more safety functions with a certain level of performance for each one. The choices of the safety functions are made by the implementer with respect to the available phenomenological knowledge and the understanding of the functioning of the overall disposal system. In order to keep the expected level of performance, certain safety functions could also avoid the components fulfilling specific safety functions to be degraded.

c) Experience and lessons learned

Within the "2005 Clay Dossier" [4], ANDRA used the approach of the safety function for guiding design and scenario development. ANDRA associated a certain number of functions to the different disposal components for the exploitation phase and for the post closure phase. Those functions were described in the IRSN assessment report [3].

ANDRA defined safety functions for the exploitation:

- To preserve people from irradiation
- To confine radioactivity
- To control criticality risk
- To dissipate the residual thermal power out of the disposal canisters
- To dissipate radiolytic gases from certain canisters

ANDRA defined safety functions for post-closure phase:

- to separate the wastes from surface mechanisms and human intrusions
- to save the repository memory
- to prevent water circulation
- to limit the radionuclide releases and to immobilize them in the repository
- to retard and to attenuate the radionuclide migration.

ANDRA adds that the implementation of those safety functions conducts to sub-function statement.

The application of that approach needs to identify the components contributing to the implementation of the safety functions and then to clarify their characteristics. Those steps allow a better understanding of the plausible evolution of the components considering interactions. From this understanding gained in the evolution of the




components, the levels of performance possibly reached are evaluated through different environmental settings. Consequently, this approach allowed ANDRA, within the "2005 Clay Dossier", to design the architecture of the disposal system and to elaborate disposal concepts ensuring the safety of this repository.

4. Analysis and synthesis

Implementation of safety functions associated to the components complies with the principle of defence-in-depth lauds by the international organisations, and particularly with the level 1 and 3 edited by IAEA [6]. The objective of the first level is the prevention of abnormal operation and system failures and the third level ensures that safety functions prevent accidental evolution of the system by providing specific safety systems or other safety features. In fact, the principle of multiple safety functions allows ensuring, in the event of the loss of a function or the failure of a component, the disposal system preserves safety margins and ensures the expected level of performance. The complementarity notion relates notably to the various time scales of the fulfilment of the safety functions by the different components because of their progressive degradation. On the contrary, the host rock has an important role for the system safety and will not lose its containment functions as long as the dissemination of the residual activity leads to unacceptable individual exposures.

The principle of demonstrability is to be followed for the design of the disposal system. This principle consists of dealing with methods allowing, in a first hand, to demonstrate the preserving of safety functions within the post closure phase and the achieving of the expected performance of the system components. In a second hand, the methods allow to appreciate the quality of the demonstration by adding relevant arguments conducting to the robustness of the components and the simplicity of the conception of the disposal system. As an example, the level of quality actually reached should be assessed in situ for the various components of the repository. Long term performances should therefore depend on the initial and real state of the components during operational phase (comprising canisters design and manufacturing).

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Part 1: Task report Safety Functions

Appendix A6: NDA (United Kingdom)

A6 NDA (United Kingdom)





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Project acronym: PAMINA

WP 1.1 – Safety functions





Section 1: Background/ Introduction

Documentation of the safety function of each component of the disposal system is one of the most important aspects of a safety case.

Section 2: Regulatory requirements and provisions

The multi-barrier principle is fundamental to the choice of design concept, in fact in the UK it is a Regulatory requirement to demonstrate that the overall safety case does not depend unduly on any single component of the case. Another principle relevant to the approach of selecting barriers is the cautionary principle, that is generally erring on the side of caution. For example, this leads to a strategy to develop a concept based on the use of well-characterised materials and using established engineering techniques wherever possible.

However regulations do not define or require any numerical safety function indicators other than the overall 10⁻⁶ per year risk criterion.

Section 3: Key terms and concepts.

None

Section 4: Treatment in the Safety Case

Section 4.1: Methodology

In NDA's next assessment, the focus will be on describing the safety functions of each of the multiple barriers and the timescales over which they are most important. A range of arguments will be included to build confidence in our understanding of the function and evolution of these safety functions, including direct references to research and comparisons with natural and anthropogenic analogues. These arguments will be supported by quantitative performance indicators for each of the main safety barriers. For the NDA repository concept for ILW, in which grouted wasteforms are packaged in stainless steel or concrete containers and placed in vaults which are eventually backfilled with a cementitious material, the main barriers (and their associated safety functions) are as follows:

• **Containment**. The waste container is mechanically and structurally intact. Only gaseous releases (via container vents) are possible, all other materials are completely contained within the waste packages.





Appendix A6: NDA (United Kingdom)

- **The Package**. The physical containment afforded by the waste packages, including the wasteform itself, continues to retard the release of radionuclides by the groundwater pathway, even though localised corrosion may have reduced the integrity of some containers.
- **The Chemical Barrier**. The release of radionuclides continues to be retarded by the reducing, alkaline conditions established in the repository backfill porewater.
- The Geological Barrier. The geological barrier provides a long travel time to the surface, gives substantial dispersion and dilution and retards sorbing radionuclides. This prevents most radionuclides that leave the near field from returning to the surface environment and ensures that any radionuclides that do reach the surface do so in very low concentrations that do not pose any significant health risk. The long-term stability of the geosphere continues to provide safety at very long times in the future, even under significant external change.

There is no specific 'prioritisation' of the safety barriers, rather it is recognised that the barriers play different and complementary roles and that their relative significance will vary over different timescales and for different radionuclides. For example, at early timeframes the container provides effective physical containment for all radionuclides (with the potential exception of gaseous emissions via the vent). The chemical barrier is very effective at containing the release of sorbing and/or less soluble radionuclides, but has little impact on soluble, mobile radionuclides such as chlorine-36 or iodine-129. The geological barrier has important roles in isolating the wastes, protecting the engineered barriers and delaying the return to the surface environment of those radionuclides that cannot be completely contained by the engineered barriers. In this way the safety functions are 'nested'.

Section 4.2: Related topics

None

Section 4.3: Databases and tools

None

Section 4.4: Application and experience

This is currently on-going work. We have developed the methodology but have not yet produced a safety case based on the proposed methodology. The first generic safety case based on this approach is due for publication in 2009.





Part 1: Task report Safety Functions Appendix A6: NDA (United Kingdom)

Section 4.5 On going work and future evolution

This work will continue to be developed as it is implemented.

Section 5: Lessons learned

Too early to say as the methodology has not yet been implemented. NDA will actively seek and respond to feedback.

Section 6: References

Nirex, Generic Post-closure Performance Assessment, Nirex Report N/80, 2003.

L.E.F. Bailey, Performance Assessment in Context, Nirex Report N/117, 2005.





Part 1: Task report Safety Functions

Appendix A7: NRG (Netherlands)

A7 NRG (Netherlands)

[PAMINA] (D-N°: 1.1.1) – Task reports for the first group of topics Dissemination level: RE Date of issue of this report: 15/03/2008







note

to :	Topic coordinator 'safety functions'
from :	J.B. GrupaPetten/015.017
copy :	J. Hart, A.D. Poley
date :	03 December 2007
reference :	21951/07.86197 RE/JG/ES
subject :	NRG Final contribution to topic 1 'Safety functions'

Section 1: Background/ Introduction

In the late 1980's the VEOS study (Safety evaluation of disposal concepts in rock salt) has been performed in the Netherlands [1, 2, 3, 4]. The aims of this study were the evaluation of the post-closure safety of some possible disposal concept and the determination of relevant characteristics. VEOS used a scenario approach followed by a deterministic consequence analysis and several deterministic sensitivity studies. The analyses resulted in a number of release scenarios with estimated exposure. For some scenarios with a relatively high exposure the probability of occurrence was also calculated. The resulting risk defined as the product of this probability and the health effect of the exposure was below the risk levels set in neighbouring countries and the IRCP.

In the early 1990's a generic probabilistic safety analysis (PROSA, [5]) of the Dutch generic reference disposal concept has been performed. In this study a systematic approach to scenario selection has been used that ultimately leads to a set of selected scenarios that covers all aspects relevant for the long term safety. The method used a FEP catalogue to show comprehensiveness of the obtained set of scenarios.

Section 2: Regulatory requirements and provisions

There are presently no regulatory requirements and provisions that directly relate to safety functions.

Section 3: Key terms and concepts.

Presently there is no specific usage of 'safety functions' in the Netherlands. Depending on the still to be adopted definition of safety functions, maybe a similar entity can be found in the





scenario identification strategy used in the Dutch (probabilistic) safety study.

In the PROSA study, mentioned above, the various safety functions of a given barrier in the disposal system have been treated implicitly. In the FEP analysis procedure, so-called primary FEPs have been identified. These primary FEPs distinguish from others because they actually disturb one or more of the safety functions of the barrier. The methodology will improve if the safety functions of each barrier are explicitly identified before the FEP analysis starts.

Section 4: Treatment in the Safety Case

Section 4.1: Methodology

The adopted methodology for the scenario selection was based on the idea that the repository is a multi-barrier system which' evolution can be characterized by the state of the four barriers:

- 1. the engineered barriers;
- 2. the isolation shield of salt around the repository;
- 3. the overburden, and
- 4. the biosphere itself.

It was assumed further that the first three barriers can have in principle two possible states: i) present and ii) by-passed. In the safety assessment the biosphere was not considered to be by-passed. This implies that there are 8 possible states of the multi-barrier system. For each barrier state a number of FEPs, the so-called primary FEPs, can be found which are defining the state of the barrier. These primary FEPs are used to define the scenarios. The other FEPs are the so-called secondary FEPs which describe the transport and state of the nuclides. The methodology implies that for each FEP one has to think whether it is of importance and if so how the role will be and in which part of the repository the FEP is applicable.

In the PROSA report, no specific safety functions were explicitly mentioned to characterize the above-mentioned four barriers.

We expect however that the PROSA procedure for identifying scenarios will be extended by the application of 'safety functions' for future safety studies.

Section 4.2: Related topics

General framework for scenario identification. Performance assessment.





Section 4.3: Databases and tools

FEP database and the procedure for FEP analysis.

Section 4.4: Application and experience

No applications and experiences yet concerning the topic of safety functions.

Section 4.5 On going work and future evolution

A specific topic that may be stressed from the Dutch point of view is the performance indicator "closure times" of plugs and seals in a salt-based repository, which are defined as the times for which compacted salt reaches the percolation limit (1% porosity), for which the possible water flow paths in the compacted salt are cut off through the ongoing compaction process. If the percolation limit is reached any water inflow or outflow from a sealed compartment (borehole, gallery) is considered impossible, which also greatly reduces if not terminates the transport of radionuclides from the disposal zone. Within the EU NF-PRO project, the University of Utrecht and NRG have improved the modelling of the compaction behaviour of compacted-salt borehole plugs. This modelling effort is continued within the THERESA project.

Taking this into account, and in the framework of Safety Functions, the performance indicator "time to reach percolation limit" in the case of a salt-based repository could be opted for as a measure to characterize the safety function "isolation".

We expect that the PROSA procedure for identifying scenarios will be extended by the application of 'safety functions' for future safety studies.

Also we expect that it will be very useful to present the results of PA-calculations along the lines of safety functions.

Section 5: Lessons learned

No applications and experiences yet.

Section 6: References

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- [5] PROSA Study Prij, J.,B.M. Blok, G.M.H. Laheij, W. van Rheenen, W. Slagter, G.J.M. Uffink, P. Uijt de Haag, A.F. B. Wildenborg and D.A. Zanstra, PRObabilistic Safety Assessment, Final report, of ECN, RIVM and RGD in Phase 1A of the OPLA Programme, 1993.





Part 1: Task report Safety Functions Appendix A8: NRI, RAWRA (Czech Republic)

A8 NRI, RAWRA (Czech Republic)



Part 1: Task report Safety Functions



Appendix A8: NRI, RAWRA (Czech Republic)





Proposal/Contract no.: FP6-036404

Project acronym: PAMINA

Project title:

PERFORMANCE ASSESSMENT METHODOLOGIES IN APPLICATION TO GUIDE THE DEVELOPMENT OF THE SAFETY CASE

Instrument: Integrated Project

Thematic Priority:

Management of Radioactive Waste and Radiation Protection and other activities in the field of Nuclear Technologies and Safety

WP1.1 Safety functions

NRI and RAWRA contribution

S. Konopásková and A. Vokál

Due date of deliverable: 03.31.07

Actual submission date: 26.10.2007

Start date of project:10.01.2006Duration:36 months

Revision: 1

Project co-funded by the European Commission within the Sixth Framework Programme (2002-2006) Dissemination level

 PU
 Public

 PP
 Restricted to other programme participants (including the Commission Services)
 X

 RE
 Restricted to a group specified by the consortium (including the Commission Services)
 X

CO Confidential, only for members of the consortium (including the Commission Services)

[PAMINA] (D-N°: 1.1.1) – Task reports for the first group of topics Dissemination level: RE Date of issue of this report: 15/03/2008





This document describes the work of NRI and its Associated partner RAWRA (Radioactive Waste Repository Authority) regarding the use of safety functions in the Performance Assessment (PA) of HLW repositories in granite.

1.1 Role and Application of Safety Functions

Safety functions in Czech disposal programme support the process of geological repository development in the measure of

- Description of performance of the disposal system and its components
- Justification of the decision process in the repository design
- Optimization process
- Definition, qualification and quantification of safety indicators

1.2 Criteria and Identification of Safety Functions

By regulations (Regulation No. 307/2002 Coll. on radiation protection), the potential individual dose raised by repository existence, has not to exceed 0.25 mSv/yr for normal evolution scenarios and/or 1 mSv/yr for emergency scenarios. There exists no other quantitative limitation postulated by nuclear legislation.

Two principal qualitative requirements have been stated by regulation:

- Disposal place has to be dry during the operational period
- Disposal place has to be protected from water infiltration during the operational period

To facilitate the application of the radiohygienical criterion, there were defined safety functions of the repository and its components - spatially and time dependent.

Safety functions of the repository had to be developed respecting following criteria:

- The dose limit has not be exceeded during all the periods of repository existence
- Disposal system has to show sufficient potential for radionuclide isolation, retention and dilution
- Disposal system performance has to be robust with respect to long term changes in near field, far field and biosphere and/or potential initial project faults that could affect disposal system performance

Regarding this, the system of safety functions has been divided into following subgroups:

- Disposal system safety functions
- Near field safety functions



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- Host rock safety functions
- Biosphere safety functions

In each of the group, there exist one ore more components, qualitatively or quantitatively defined. As a whole, the system of safety functions has to assure the compliance of disposal system performance with the principal radio-hygienical criterion.

1.3 Historic evolution of safety functions

Regarding the stage of geological repository development in the Czech Republic and the state of legislation framework that is not very explicit in the terms of safety postulations, the history of safety functions evolution is not too long. The functions are formulated following the repository programme steps. At present, the near field studies are in a systematic progress and far field studies are expected to start. Siting activities have a research character, i.e. they are in a descriptive stage, without substantial or systematic relation to repository safety. Repository safety cannot be plausibly evaluated in the case of the lack of data from site. In every case, there have been finished following projects of smaller extent that gave rise to a description of a background of safety functions definition:

- safety and sensitivity evaluation of the reference project
- a test case constructed on a reference project
- near field parameters sensitivity study
- far field sensitivity study

Further specifications are awaited after finishing of near field and far field project whose results shall be used as inputs to the optimization phase of the reference project.

1.4 Description of the present set of safety functions

Disposal system

Dose limit

The disposal system has to assure through its isolation, retention and dilution capacities the compliance with the individual dose limits.

Time frame: operational period, post closure period 0 – some tens of thousands of years

Stability

The disposal system has to show stabile properties from a long term point of view to defend the assumptions accomplished in the safety analysis and to assure the long term isolation capacity of the system





Time frame: operational period, post closure period 0 – some tens of thousands of years

Robustness

The disposal system has to be robust with regard to potential adverse initial events including inadvertent project events, and to uncertainties in safety assumptions and determination of input parameters in safety assessment

Time frame: operational period, post closure period 0 – some tens of thousands of years

Near field

Waste form

The waste form has to provide the physical containment for the waste in the period of interim storage and to immobilize the waste in the first period after disposal. Waste form is one of the components of the multibarrier system. Principal process affected by this function is radioactive decay. Gas production and heat production are processes that have to be thoroughly followed.

Time frame: operational period, post closure period 0 – some tens to hundreds of years, generally. The period, in which the repository is resaturated.

Container

The container will provide the physical containment to the final waste form and will prevent radionuclides release and/or retard it in the period when the repository has been resaturated. Gas production, heat production, corrosion and initial faults are the principal processes that have to be followed for assurance of this safety function. Human intrusion has to be assessed.

Time frame: post closure period – some hundreds to thousands of years. The first period after the repository resaturation, previous to the start of chemical prevention.

Backfill

Backfill (including sealing) will provide barrier after the isolation of potential of waste form and container are strongly disabled or have passed at all. The chemical protection is based on reducing conditions in the backfill material that have to correspond to the groundwater properties of the host structure. Chemical conditions retard the radionuclides migration and delay the radionuclides release to the hydrogeological environment, but releases from the near field are possible in dependence on the backfill properties and geometry of the near field. Heat production and transport have to be evaluated with respect to potential changes of the backfill material dependent on heat production.

Time frame: post closure period – some thousands to tens of thousands of years

Engineered Barrier System

The EBS has to provide isolation and retention of radionuclides in the near field for a period that has to be evaluated with regard to transport times in the far field and transfer in





biosphere. The EBS construction shall manage the transport process in the way that diffusion transport is the principal process in the near field during the whole period of backfill safety function persistence. All potential effects as waste form and container life times, heat production and effects, releases from the waste for/container/EBS, human intrusion has to be taken into account.

Time frame: post closure period – some tens of thousands of years

Water Flow Rates

Water flow rate in the repository has to be minimized. It will allow achieving resaturation times as long as possible and retarding potential releases. Near field conditions have not to allow water flow through the repository in the operational phase and in the phase of physical containment. Stability of potential water flow rates has to be documented.

Time frame: operational period, post closure period 0 – some tens of thousands of years

Stability

Stability of the system waste-EBS will provide evidence of safety assumptions in the long time frames.

Time frame: operational period, post closure period 0 – some tens of thousands of years

Robustness and Impact of Initial Events

The disposal system has to be robust with regard to potential adverse initial events including inadvertent project events, and to uncertainties in safety assumptions and determination of input parameters in safety assessment

Time frame: operational period, post closure period 0 – some tens of thousands of years

Far field – geological barrier

Dilution and retention

The geological barrier has to ensure dilution and retention of radionuclide releases from the near field in the measure necessary for the compliance with dose criteria. Flow and transport through far field has to be evaluated.

Time frame: tens of thousands of years. In this period, the release from the repository (EBS system) displays as a homogenous source term.

Travel Times

Travel times of critical radionuclides have to be long enough to assure that the concentrations in environmental components are in compliance with dose criteria. Flow and transport through far field has to be evaluated.

Time frame: tens of thousands of years. In this period, the release from the repository (EBS





system) displays as a homogenous source term.

Robustness

The geological barrier has to be robust with regard to potential predictable adverse initial events, and to uncertainties in safety assumptions and determination of input parameters in safety assessment. Flow and transport through far field has to be evaluated.

Time frame: tens of thousands of years

Stability

Stability of the host structure has to provide safety for a very long period in the future and supports the robustness of the disposal system. After the period of tens of thousands of years, quantitative assessment seems to be irrelevant, qualitative argumentation has to support the safety assumptions.

Time frame: tens of thousands of years to millions of years

Biosphere

Transfer

Transfer of radionuclides in the biosphere has to assure compliance with dose criteria. Biosphere is not considered as a safety media, but it provides dilution of radionuclide released from geological barrier. Geological barrier is considered as a homogenous source term for the biosphere input.

Time frame: tens of thousands of years

1.5 Preliminary results of current project safety function analyses

Safety functions described above lack a strict hierarchical structure that could assure that no important functions were neglected. There is also no connection with interactions identified between individual parts of the system, i.e. features, events and processes (FEPs). A more systematic approach for safety functions development started therefore in a project initiated two years ago by RAWRA with the main aim to establish the scientific and technical basis for evaluating the safety function "containment and minimisation of release of the near field" of the reference design of Czech DGR.

This function was divided into two daughter functions:

- Contain (isolate) wastes in waste packages (Containment function)
- Minimise release of radionuclides after waste packages failure from near field (Release function)

The first function is active in time before containers failure and is allocated to waste





packages. The second one is active after container failure and is allocated on the repository system in the state after container failure. Further decomposition of these functions was based on identification of interactions of the reference EBS system and surrounding systems (buffer, backfill, construction materials, geosphere, biosphere).

Containment function analysis

Containment function allocated on waste packages can be assured if the following "daughter" safety functions are met:

- To resist to mechanical stress.
- To resist to chemical (microbiological) conditions of geosphere.
- To resist to the effect of wastes (radiation, temperature).
- To resist to corrosion products generated by degradation of waste packages materials

It is also evident, that this Containment function allocated on waste packages will work only under some assumptions concerning thermal, hydrological, mechanical and chemical (microbiological) effects of surrounding systems. Accordingly, the following safety functions allocated to surrounding systems have been identified:

- To conduct heat from waste packages (Thermal effect)
- To limit water flux to and from waste packages (Hydrological effect)
- To prevent mechanical stress on waste packages (Mechanical effect)
- To provide favourable chemical and microbiological conditions (Chemical effect)

All these functions require further analyses and further decomposition to higher level of detail.

Release function analysis

After waste package failure, the safety function "to minimise the release of radionuclides to geosphere" is based primarily on:

- Low degradation rates of waste form.
- Low solubility of radionuclides.
- Low permeability of surrounding materials,
- High sorption of radionuclides on EBS materials.

While the Containment function was limited only for some certain time depending on materials of waste packages selected, the Release function must work for hundred thousand or even million of years. The following daughter functions were identified:

- To limit contact of waste form with water
- To limit degradation rates of waste form





- To limit solubility of radionuclides in near field
- To retard migration of radionuclides by sorption

These safety functions can also be met in long term only under some assumptions allocated on other part of the system. These assumptions could be the same as mentioned above for Containment function, but they must work in other timescales. The need of some of them will gradually disappear. For example, the function " to conduct heat from the waste forms is not further active after decay of the most heat generated radionuclides (after about 500 to 1000 years). The functions allocated to surrounding systems of near field after failure of canisters can be formulated as follows:

- To limit the flux of water to disposal units (Hydrological effect).
- To limit the mechanical stress on disposal units (Mechanical effect).
- To provide favourable chemical conditions for low degradation rates of waste forms and long term low solubility of radionuclides (Chemical effect).

1.6 Lessons learned

The approach given above is more systematic than "judgemental" system of safety functions identification based on literature review and/or experts judgement, but it can be easily recognised that it is not reasonable to start the function analysis only from near field function analysis without taking into account the top functions related to the whole disposal system. In the framework of this project (WP 3) and projects supported by RAWRA mentioned above, it was proposed to start a systematic top-down approach starting from a top function for the whole disposal system. The top-down approach selected will be based on so-called FRAT (Function, Requirements, Answers, Test) system developed by Prof. Morais from Synergistic Applications, Inc.[1], characterized by the following steps (see Figure 1 too):

- 1. Anything with parts that interact to achieve a common purpose whether it is a product a process, organization, or a thought, can be viewed as a system.
- A system can be described by four views what the system does (functions), how well the system performs its functions (all types of requirements including constraints), what the system actually is (answers), and verification and validation activities the provide the proof that the actual system satisfies the intended functions and requirements (tests).
- 3. It is important to define and understand the three interacting systems: the product system, the program system that creates the product system, and everything else that interacts with the product and program system.
- 4. To define a system at any level of decomposition, you need as an input a definition of the next higher level. If this upper level definition does not exist, the first step is to establish this in terms of the four views defined above. Once this is available, it can be decomposed into lower level functions. Once the functions are available the requirements for these functions can be established. Given the function/requirement descriptions the search for alternative answers can begin and trade studies used to select the better answer. Finally, definition and results of tests for verification and







validation of the answer are generated.



Figure 1: Sorting and using frat data

This FRAT system belongs to top-down approaches, which have been used in a number of countries for scenario development [2]. It enables to better quantify requirements on the repository and its components in a structured manner. It forces performance evaluators not only to identify functions, but quantify them in the form of quantitative requirements. The great advantage of this system is also that all activities can be well documented in a structured manner, which enables linking performance assessment with QA system. In the case of analyses of already known system, it is possible to change FRAT to AFRT (Answer-Function-Requirement-Test), in which safety functions are allocated to already proposed components of the system.

The primary step in this approach is to determine the main objective and top function of the whole disposal system. A top safety function of the reference disposal system (A) after closure can be formulated in agreement with IAEA documents and Czech legislative requirements as follows:

(F) To isolate all spent fuel assemblies and other radioactive waste not acceptable to surface repositories generated in the Czech Republic from the human environment and to ensure the long term radiological protection of humans and the environment so that the releases from a repository due to "gradual" processes or from disruptive events shall be less than the dose or risk upper bound apportioned by national authorities from an individual dose or risk limits.

(R) The main requirement is effective dose of 250 μ Sv/yr. This level is considered as sufficiently evidenced, as far as there is evidenced that neither the foreseen deviation from the normal operation the given guidance level can be exceeded.

(T) The main task of test programme is to prove that this reference repository will meet the



requirement of 250 $\mu Sv/yr$ under all possible gradual or disruptive processes that can occur in the disposal system.

Further decomposition of this top function, requirement and test for Czech reference design of the repository will be performed in similar way as shown above in the framework of WP3.

Literature

[1] B.G. Morais, Systems Engineering Fundamentals Course, Prague, 2 – 6, October, 1995

[2] NEA/OECD, Systematic Approaches to Scenario Development, OECD 1992

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Part 1: Task report Safety Functions

Appendix A9: POSIVA (Finland)

A9 POSIVA (Finland)

[PAMINA] (D-N°: 1.1.1) – Task reports for the first group of topics Dissemination level: RE Date of issue of this report: 15/03/2008







Appendix A9: POSIVA (Finland)

PAMINA

WP1.1 Safety functions

Posiva Oy – Contribution – 2nd Draft

PAMINA. WP1.1 Comprehensive Review of Methodologies and Approaches in the Safety Case

Safety functions

1 Background

The term safety function has been introduced recently in Posiva's Safety Case, but there is overlap with the term safety requirement, which has been used much earlier. Safety function was introduced along with the definition and application of the Safety Case concept (TKS-2003; Vieno & Ikonen 2005). What is nowadays called safety functions was described and defined as role in long-term safety of each of the components of the multi-barrier system (e.g. TVO YJT-85-30 Report). Proposals for safety requirements and criteria for the disposal of high level radioactive waste were prepared in co-operation by the radiation and nuclear safety authorities in the Nordic countries (Ruokola 1990). The proposals of the working group were based on the recommendations given by the International Commission on Radiation Protection (ICPR), the International Atomic Energy Agency (IAEA), and the Nuclear Energy Agency (NEA) of the OECD.

1.1 Current understanding

The criteria for identification and definition of the safety functions have been developed during the disposal programme since the early 80's as part of the system design and design requirements (a robust system maintaining the long-term isolation of spent fuel has been the leading principle since the very beginning). The regulator states: "the long-term safety of disposal shall be based on redundant barriers so that deficiency in one of the barriers of a predictable geological change does not jeopardise the long-term safety. The barriers shall effectively hinder the release of disposed radioactive substances into the host-rock for several thousand of years" (STUK 2001).

2 Repository system components and safety functions

A summary description of the safety functions is presented in Figure 1. All the safety functions for the canister should be kept up to 100 000 years. The safety functions of the other system components are expected to be kept up to the same time and even further.





Appendix A9: POSIVA (Finland)

In discussing the evolution of the repository and site (Pastina & Hellä 2006) the description on the long-term behaviour of the system components starts from the safety functions from each of the components.



Figure 1. Long-term safety functions of the bedrock and engineered barrier system in the KBS-3V disposal concept (Vieno & Ikonen 2005)

3 Safety functions in process description

The description of the processes affecting each of the system components (fuel, canister, buffer, backfill, geosphere) starts with a statement on the safety functions of the component. Then individual processes are qualitatively rated as for their importance (high, medium, low) in jeopardizing the safety functions in particular and the performance of the repository in general (POSIVA 2007).

Rating the importance of the processes in this way serves as a guide to prioritise and focus research studies.

4. Challenges and recommendations

The estimation of the time frames for which the safety functions must be kept is not always





Appendix A9: POSIVA (Finland)

straightforward and misunderstandings happen. For example, as long as the canister keeps its safety functions, it shouldn't matter if the buffer does not. However most processes are coupled and the canister integrity is linked to the behaviour and safety functions of the bentonite. Gathering the evidence on the fulfilment of the safety functions and validity of the safety concept ("safety approach") is the core of the safety case that should be kept clear; the main future tasks are the streamlining of the discussion and crystallising the conclusion, and analysing the system and its alternative evolution paths in more detail.

5. References

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Posiva 2003. Nuclear Waste Management of the Olkiluoto and Loviisa power plants: Programme for research, development and technical design for 2003-2006. Posiva Oy, Olkiluoto, Finland. Report POSIVA TKS-2003.

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Part 1: Task report Safety Functions

Appendix A10: SCK-CEN, ONDRAF-NIRAS (Belgium)

A10 SCK-CEN, ONDRAF-NIRAS (Belgium)





Part 1: Task report Safety Functions

Appendix A10: SCK-CEN, ONDRAF-NIRAS (Belgium)

SCK•CEN EHS/IPA/PAS 07/C092010/JMa/N-7 (rev. 1) 05-09-2007

PAMINA PROJECT

WP 1.1 Review of Methodologies

The Role of Safety Functions in the Belgian HLW Disposal Programme

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²ONDRAF/NIRAS, Brussels, Belgium

1. Background / Introduction

Safety functions were introduced in the Belgian high-level waste (HLW) disposal programme in 1999 (De Preter et al. 1999). One of the first main reasons for this was that safety functions allow to explain the functioning of a geological disposal system to various stakeholders in a relatively easily understandable way. The safety functions were also applied in the SAFIR 2 report for explaining the role and justifying the choice of the main engineered barriers and for identifying performance indicators (ONDRAF/NIRAS, 2001). This was to a large extent an *a posteriori* application, i.e. after the design work and the safety evaluations.

After SAFIR 2, safety functions started to play a key role within the Belgian HLW disposal programme: they were also used to facilitate the communication between the three main teams involved in the development of the safety cases, i.e. repository development, phenomenology and safety assessment, as well as for structuring the R&D work (including the designing of the facility). In the Safety and Feasibility Case 1 (SFC 1), which is scheduled to be finalised in 2013, the safety functions are underpinned by safety statements.

It has always been the intention of developing a set of safety functions that is applicable to all types of disposal systems under consideration in the Belgian programme (surface disposal for the short-lived waste and deep disposal for the high-level and long-lived waste).

While this document is only dealing with the long-term safety functions, ONDRAF/NIRAS is also considering and developing operational safety functions.





2. Regulatory requirements and provisions

Regulatory requirements and guidelines concerning long-term safety of high-level radioactive waste disposal are still in preparation in Belgium.

3. Key terms and concepts

The following definitions of a safety function and related terms are used within the Belgian high-level radioactive waste disposal programme.

Long-term safety function: a function that a disposal system should fulfil to achieve its fundamental objective of providing long-term safety through the concentration and confinement strategy, while limiting the burden for future generations.

Multifunctional system: a disposal system that provides long-term safety by means of multiple safety functions. These safety functions are fulfilled by multiple barriers, in such a way that the overall safety of the system does not depend unduly on a single barrier or function.

Effective safety function: a long-term safety function that is fulfilled effectively during a certain time frame by at least one component of the disposal system and that can thus be relied upon in safety assessments.

Latent safety function: a long-term safety function that is available in the disposal system but that only becomes effective if another function that is supposed to be effective actually fails to be fulfilled properly. The level of performance of a latent function, once it becomes effective, can largely depend on the moment of its activation. Depending on the expected level of performance, on the knowledge available and on the adopted safety, a latent function will be effectively relied upon in safety assessments or will be considered a supplementary safety function.

Supplementary safety function: a long-term safety function that could be effective during a certain time frame, but whose performance cannot be properly evaluated because of a lack of knowledge. A supplementary safety function can become an effective safety function or a latent safety function if the uncertainties on its effective operation can be sufficiently reduced.

4. Treatment in the Safety Case

4.1 Methodology

4.1.1 Derivation of safety functions

The derivation of the long-term safety functions of a disposal system is based on two considerations. In the first place, the disposal system must be intrinsically able to protect





man and the environment. This implies that its functional integrity must not be unduly jeopardized by external disturbances.

a) Protecting man and the environment

In order to protect man and the environment, and given the adopted strategy to "concentrate and confine", two measures must be taken:

- 1) the prevention of exposure as a result of inadvertent human intrusion in the repository or as a result of insufficient shielding of the waste;
- 2) the prevention of exposure as a result of the release of contaminants, i.e. radionuclides and non-radioactive toxic substances, from the disposal system into the environment.

The first protection measure can be implemented through isolating the waste durably from the environment by disposing of it in a place that is and will remain difficult-to-access and well-shielded, and that is not likely to attract human activities. This is achieved through the long-term safety function *isolation*.

The second protection measure can be implemented through containment of the contaminants, where containment implies designing for a minimal release of contaminants (IAEA, 2006). This can be done at two successive levels:

- first, at the level of the disposed waste, i.e. the waste form and its engineered containment barrier, by preventing any dispersion of the contaminants from the disposed waste. This is achieved through the long-term safety function *engineered containment*.
- then, at the level of the disposal system, i.e. before the contaminants are released into the environment of the disposal system, by hindering and retarding dispersion of the contaminants towards man and the environment as much as achievable, so that the release rates of contaminants from the disposal system into the environment remain at all times limited to an acceptable level. This is achieved through the long-term safety function *delay and attenuation of the releases*.

b) Protecting the disposal system

In order to ensure that the functional integrity of the disposal system is not unduly jeopardized by external disturbances, two measures must be taken:

- 1) the prevention of disturbances resulting from external processes and events other than inadvertent human intrusion;
- 2) the prevention of inadvertent human intrusion.

These protection measures can be implemented through, respectively:

- disposing of the waste in a setting that is stable in both geomorphological and physicochemical terms and that is well isolated from the environment;
- disposing of the waste in a setting that is not likely to attract human activities.

This is achieved through the long-term safety function *isolation*.





4.1.2 Present set of long-term safety functions

The three long-term safety functions and their sub-functions constitute a basic tool for designing a disposal system and for assessing its safety, as well as for structuring the research and development work. Depending on the type of waste considered and, so, on the type or design of the disposal facility, some functions or sub functions may not need to be explicitly taken into account for developing the facility or may not be taken into account when assessing its safety (ONDRAF/NIRAS, 2007).

a) Isolation (function I)

The isolation function I consists of *isolating* the waste durably from man and the environment, by (1), preventing direct access to the waste and (2), protecting the disposal system from potentially detrimental processes occurring in the environment of the disposal system.

The I function can be divided into two sub-functions.

1) Reduction of the likelihood of inadvertent human intrusion and of its possible consequences (I-1)

The I-1 sub-function consists of *limiting the likelihood of inadvertent human intrusion and,* in case such intrusion does occur, of *limiting its possible consequences* in terms of radiological and chemical impact on man¹ and the environment.

Reducing the likelihood of inadvertent human intrusion is possible through:

- disposing of the waste in a place that provides substantial physical separation from man and the environment. Gaining access to the waste would require special technical capabilities, beyond the reach of individuals.
- locating the repository away from areas of underground mineral resources.

Limiting the possible consequences of inadvertent human intrusion is possible through:

- enhancing the resilience of the system, namely enhancing its ability to maintain high performances after it has been disturbed by inadvertent human intrusion. The resilience of the system can be enhanced, for instance, by dividing the repository into compartments or, in case of geological disposal, by locating it in a self-healing host formation.
- 2) Ensuring stable conditions for the disposed waste and the system components (I-2)

The I-2 sub-function consists of protecting the waste and the components of the disposal system from changes and perturbations occurring in the environment of the system, such

¹ Individual inadvertent intruders cannot necessarily be protected or do not necessarily have to be protected to the same extent as the general public (IAEA, 2006; ICRP, 2000). So, the consequences of human intrusion to be assessed are those on people living near the disturbed repository and further away.



as climate changes, erosion, uplifting, seismic events or relatively rapid changes in chemical and physical conditions.

Ensuring stable conditions for the disposed waste and the system components is possible through:

- selecting a stable geological setting to avoid or limit geomorphological processes leading to denudation of the waste;
- selecting a buffered physicochemical environment for the components of the disposal system that contributes to the fulfilment of the other safety functions (see below).

b) Engineered containment (function C)

The *engineered containment* function C consists of preventing for as long as required the dispersion of contaminants from the waste forms and the escape of gaseous substances, by using one or several appropriate impermeable barriers.

Engineered containment can be achieved by placing one or several impermeable barriers around the waste forms, so as to prevent contact between the contaminants and the infiltrating water or already present water, which is the main vector of contaminant dispersion. These barriers will also prevent gaseous escapes.

As long as the C function is effective, there is virtually no dispersion of contaminants from the disposed waste and the radioactive decay of the radionuclides within the waste forms reduces the total potential radiological impact of the waste. When the C function is no longer effective, which is inevitable with time, or if it was not required in the first place, another safety function *delay and attenuation of the releases* (cf. Section 4.1.2 c)) must take over.

c) Delay and attenuation of the releases (function R)

The function of *delay and attenuation of the releases* R consists of retaining the contaminants within the disposal system for as long as required, by (1), limiting contaminant releases from the waste forms, (2), limiting the water flow through the system and hence the quantity of contaminants migrating and ultimately leaving the system and (3), retarding contaminant migration.

The R function can be divided into three sub-functions.

• Limitation of contaminant releases from the waste forms (R-1)

The R-1 sub-function consists of *limiting and spreading in time the releases of contaminants from the waste forms*. In addition, it limits and spreads in time the release of contaminants from the waste canisters (and overpacks if present).

The limitation of contaminant releases from the waste forms is the result of various processes, properties and phenomena: physicochemical processes such as slow dissolution mechanisms and properties such as low solubility limits of the waste matrix and of the imbedded radionuclides, which translate globally into a "resistance to leaching", and phenomena such as the spreading in time of waste container failure or overpack failure (e.g. by corrosion) and geometric limitations for the transport of contaminants (e.g. if the perforation of the container and/or overpack remains limited at





first to small holes, as a result of pitting corrosion for instance).

• Limitation of the water flow through the disposal system (R-2)

The R-2 sub-function consists of *limiting the flow of water through the disposal* system as much as possible, thus preventing or limiting the advective transport to the environment of the contaminants released from the waste forms and from the waste containers (and overpacks if present).

The limitation of the water flow through the disposal system can be achieved by locating the repository in a low-permeability host formation (in case of geological disposal) and through using low-permeability engineered barriers. A slow advective or a diffusive transport of the contaminants through the engineered and natural barriers of the disposal system spreads the release of the contaminants from the system in time due to their dispersion during their transport in a porous medium.

The R-2 sub-function determines also the amounts of water that actually come into contact with the barriers that fulfil the engineered containment function (C) or the limitation of contaminant releases from the waste forms function (R-1).

• *Retardation of contaminant migration (R-3)*

The R-3 sub-function consists of *retarding and spreading in time the migration to the environment of the contaminants* released from the waste forms and from the waste containers (and overpacks if present).

The retardation and spreading in time of contaminant migration is the result of processes such as contaminant precipitation and sorption within the disposal system.

4.2 Related topics

4.2.1 States of a long-term safety function and multiple safety functions

A long-term safety function can be in one of the following three states:

- A long-term safety function can be *effective*, which means that at least one component of the disposal system fulfils the safety function during a certain time frame in an effective manner. An effective safety function can be relied upon in the safety assessments of the disposal system.
- A long-term safety function can be *latent*, which means that the function is available in the disposal system but that it will only become effective if another function that is supposed to be effective actually fails to be fulfilled properly. The level of performance of a latent function, once it becomes effective, can largely depend on the moment of its activation. Depending on the expected level of performance and on the knowledge available, a latent function will be effectively relied upon in safety assessments or will be considered a supplementary safety function (see below). This depends upon the adopted safety strategy (see also Chapter 5).
- A long-term safety function can be considered a *supplementary* safety function, which





means that it could be effective during a certain time frame, but that there is a lack of knowledge to evaluate its performance. A supplementary safety function can become an effective safety function if the uncertainties on its effective operation can be sufficiently reduced.

Multiple safety functions and multiple barriers are required to ensure long-term safety in such a way that long-term safety does not depend unduly on a single function or component (IAEA, 2006). Safety functions are defined in terms of well-known phenomena or characteristics and operate over a certain time frame. Not all of them have to operate in each time frame. It is the contribution of all functions together that must be taken into account to ensure the protection of man and the environment. A component of the disposal system can contribute to fulfilling one or more safety functions with a certain level of performance and within a certain time frame. If a single component's contribution to the safety functions is overshadowing the contributions of all other components, it has to be demonstrated that the total loss of this component during the period that it has to fulfil its safety functions is extremely unlikely.

4.2.2 The role of "dispersion and dilution in the environment"

The environment of a disposal system may disperse and dilute the contaminants released from the disposal system, and as such contributes to long-term safety, because the impact of the disposal system on man and the environment is inversely proportional to the reduction in contaminant concentrations.

The processes of *dispersion and dilution* are considered a role of the environment, as opposed to a safety function, since all efforts made to maximize or optimize them would lead to a "disperse and dilute" strategy, instead of the chosen strategy to "concentrate and confine".

"Dispersion and dilution" mainly reduce the potential individual impact (dose), but not necessarily the total potential impact.

The capacity of the environment of the disposal system to dilute and disperse can be affected by changes and perturbations occurring in the environment of the system, such as climate changes and geomorphological processes. Their effect on the dilution and dispersion capacity has to be evaluated in safety assessments and in the safety case.

Finally, it should be mentioned that in case of geological disposal, the environment of the disposal system with its geological strata overlying the host rock (overburden) also contributes to the safety function "isolation". These overlying strata create a physical barrier between the waste and man, that contributes to the I-1 safety function and that is supplementary to the physical barrier of the host rock itself. With its buffering capacity with respect to changes and disturbances occurring at the surface it can also contribute to the I-2 safety function.

4.3 Databases and tools

Not applicable







4.4 Application and experience

Safety functions are intensively used within the Belgian HLW disposal programme for various applications:

- communication: safety functions have been successfully applied to explain in easily understandable terms the functioning of the surface and geological disposal systems to various, including non-technical, audiences;
- safety strategy: from the functional analysis of the repository system and the results of existing safety assessments strategic choices for the development of the repository concept have been made;
- repository development: the fulfilment of the selected basic safety functions was one of the key elements for the development of the engineered barrier system;
- structuring the safety case: a safety case is based on a comprehensive research and development programme; the safety functions are used as "keywords" within the Belgian safety case programme;
- identification of scenarios: starting from a functional analysis of the disposal system in case of the expected evolution scenario, it is examined for each selected scenarioinitiating event which safety function(s) might be affected; the possible impact of the considered event on the functioning of the disposal system is illustrated with a functional diagram; scenarios with similar functional diagrams are grouped as far as possible, and result in the so-called altered evolution scenarios; failures of safety functions not yet considered in previously identified altered evolution scenarios can be treated as "what-if" scenarios;
- identification of performance indicators: in the SPIN project (Becker et al., 2001) safety functions have been successfully applied to identify performance indicators; those performance indicators have also been used within the SAFIR 2 report.

4.5 On going work and future evolution

Safety functions are used for structuring the safety case. Therefore a system of safety statements, underpinning the safety functions, is being developed. Safety functions are also used for the identification of altered evolution and what-if scenarios. This application will be further developed within WP 3.1 of PAMINA.

Hitherto, safety function indicators have not been applied within the Belgian HLW disposal programme; those indicators were introduced within the Swedish HLW disposal programme (SKB, 2006). Within WP3.3 of PAMINA, SCK•CEN and ONDRAF/NIRAS will test the applicability of safety function indicators for a repository located in a clay formation.




Appendix A10: SCK-CEN, ONDRAF-NIRAS (Belgium)

5. Lessons learned

Safety functions have been introduced in 1999 within the Belgian HLW disposal programme. They have been applied for a large range of applications. Successful applications are communication of the functioning of the repository system to various stakeholders, identification of performance indicators, and structuring the interfacing between the different teams involved in the development of the safety case (repository development, phenomenological research and safety assessments).

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PART 2: DEFINITION AND ASSESSMENT OF SCENARIOS

(Prepared by Thomas Beuth, GRS, Germany)





1 Background/ Introduction

1.1 General Information

The topic "Definition and Assessment of Scenarios" is one of overall 12 topics which have to be dealt with in the framework of RTDC-1 of the integrated project PAMINA. The main goal of RTDC-1 is to provide a current and comprehensive overview of safety assessment methodologies, tools and experiences along the identified Safety Case topics.

This task report summarises the main facts, aspects, and views regarding scenario development. The basis for the task report was primarily the contributions to the topic from several participating organisations and the findings gained in a workshop of participants. Table 1.1 comprises in alphabetical order the organisations which provided a report to scenario development. Submitted reports are enclosed in the appendix.

Acronym	Organisation	Country
ANDRA	Agence Nationale pour la Gestion des Déchets Radioactifs	France
AVN	Association Vinçotte Nuclear	Belgium
ENRESA	Empresa Nacional de Residuos Radioactivos S.A.	Spain
GRS-K	Gesellschaft für Anlagen- und Reaktorsicherheit mbH	Germany
IRSN	Institute de Radioprotection et de Sureté Nucléaire	France
NDA	Nuclear Decommissioning Authority	United Kingdom
NRG	Nuclear Research & Consultancy Group	Netherlands
NRI, RAWRA	Nuclear Research Institute Rez plc., Radioactive Waste Repository Authority	Czech Republic
POSIVA	Posiva Oy	Finland
SCK•CEN, ONDRAF/ NIRAS	Studiecentrum voor Kernenergie - Centre d'Etude de l'Energie Nucléaire/ Nationale Instelling voor Radioactief afval en verrijkte splijtstoffen	Belgium

Table 1.1: List of participating organisations

This report includes several sections and an appendix with the following content:

Some general information concerning this document and an introduction into the subject scenario development are given in section 1.

Section 2 addresses existing regulations and guidelines in terms of scenario development. Different aspects concerning relevance, usefulness, and expectations to regulations and guidelines are considered. A detailed overview of definitions regarding "scenario" and "scenario development" and used terms are presented in section 3. Section 4 presents the underlying methodologies for scenario development in different countries and section 5 sets the focus on the application of the methodologies and lessons learnt. New developments, possible trends, and altered views are the subject of section 6. Section 7 summarises the essential aspects of the previous sections. Section 8 contains references which are of





common interest e.g. international documents or important reports from countries that did not prepare a contribution. For more details, the reader is referred to the papers and respective references presented in the appendix.

The topic "Definition and Assessment of Scenarios" is related to other topics in RTDC-1 such as "Human Intrusion", "Biosphere", "Analysis of the Evolution of the Repository System", and "Safety Functions" which will be handled separately. These topics are addressed in this task report in so far as they are essential for further understanding. Details concerning the related topics are subject of the respective task reports.

Relations to the contributions in the following sections are indicated as follows:

- These signs "<< " and ">> " indicate the beginning and end, respectively, of an extract or quotation of text from a contribution.
- The acronym of an organisation enclosed in brackets indicates the reference of the contribution.

1.2 Scenario Development

One of the first steps towards safety assessment is the identification of all relevant factors in terms of the long-term safety of the repository as well as their combination to develop scenarios. A systematic and transparent way for this work is vital in order to demonstrate compliance with regulations and to increase the confidence that all essential factors have been taken into account [OECD/NEA, 2001].

Most of the participating organisations have a lot of experience with systematic scenario development due to the former and / or current application of own, modified or adapted methodologies in safety assessments. Previous international projects have also increased knowledge and experience amongst participants. In the following are some examples that underline the detailed work in the field of scenario development by different organisations:

- << Definition and assessment of scenarios were carried out in the Performance Assessment (PA) of HLW repositories in granite and clay. >> [ENRESA]
- << Definition of scenarios is being dealt within the Process report (POSIVA 2007) scheduled by the end of 2007. >> [POSIVA]
- << Definition, scenario development and assessment of scenarios were carried out in the Safety Evaluation of HA and MAVL repositories in clay (Dossier 2005) >> [ANDRA].
- << Systematic scenario development in Czech geological disposal programme started in 1996 by analysing broad approaches, primarily Sandia Scenario Selection Procedure and SKI/SKB scenario development approach. The scenario development in Czech programme in further years was affected by participation of Czech specialists in Performance Assessment Advisory Group (PAAG) of the Radioactive Waste Management Committee (RWMC) of NEA and by consequent NEA publications. >> [NRI, RAWRA]
- <<Nirex (now the NDA) undertook extensive identification and development of scenarios for an ILW repository concept, with a series of reports published and reviewed by the OECD-NEA in 1999.>> [NDA]





- << For scenario development three main phases can be distinguished in the Belgian radioactive high-level waste (HLW) disposal programme:
 - phase 1 (period 1978 1990): a number of less systematic approaches were applied; these approaches will not be discussed in the present paper;
 - phase 2 (period 1992 1999): a systematic approach based on a catalogue of features, events and processes (FEPs) was introduced; this approach was used in the SAFIR 2 (safety and feasibility interim report) report (ONDRAF/NIRAS, 2001);
 - phase 3 (period 2004 2012): the new approach is still in development, partially within PAMINA, and will be applied for the Safety and Feasibility Case 1 (SFC 1).
 >> [SCK•CEN, ONDRAF/NIRAS]
- << In the late 1980's the VEOS study (Safety evaluation of disposal concepts in rock salt) has been performed in the Netherlands. VEOS used a scenario approach followed by a deterministic consequence analysis and several deterministic sensitivity studies. In the early 1990's a generic probabilistic safety analysis (PROSA) of the Dutch generic reference disposal concept has been performed. In this study a systematic approach to scenario selection has been used that ultimately leads to a set of selected scenarios that covers all aspects relevant for the long term safety. >> [NRG]

The term scenario development is used to describe the compilation and arrangement of both scientific and technical information as a fundamental basis for the assessment of long-term safety for a radioactive waste repository. This includes the identification of relevant FEPs, the modelling of the scientific basis, and the derivation of calculation cases. Therefore scenario development constitutes the overall framework for the discussion of the evolution of the repository, calculation cases and their results, as well as failures or weakness of models, attributed to unknown or less known mechanisms [OECD/NEA, 2001].

In the following are some selected statements from the contributions (see appendix) that reflect in principle the mentioned context, role, and essential elements of scenario development and the common opinion of the participants.

Context of scenario development in the frame of safety assessments:

<< Scenario development is a key topic in the frame of the safety analysis, since it has an important role in capturing uncertainties and quantifying their influence, in verifying fulfilment of safety functions associated with disposal components, and in quantifying the dosimetric impact due to the disposal system. >> [IRSN]

<< Safety assessments for radioactive waste repositories in deep geological formations are an integral part of the comprehensive demonstration of the safety of the repository in the post-closure phase. The demonstration will be conducted on a site specific basis in consideration of the geological, geochemical, and geotechnical state of the repository system, and its long-term predictions as well. The safety assessment includes the scenario development, consequence analysis with uncertainty and sensitivity analyses, and the demonstration of the compliance of prescribed protection objectives. >> [GRS-K]

Role of scenario development:

<< The need for carrying out a scenario development in safety assessment of radioactive waste disposal facilities arises from the fact that it is virtually impossible to predict exactly





what will be the evolution of the disposal system through time.

A scenario describes one possible future of the disposal system, corresponding to a combination of events and processes together with their characteristics and their chronological sequence. The expression scenario development is used both for the identification of the set of scenarios that will be representative of the different states of the disposal facility and for the identification, general description and selection of the possible safety-relevant features of the disposal system for one defined disposal evolution. >> [AVN]

Consideration of the evolution of the repository as an essential element:

<< The possible evolution of a repository system can be addressed in terms of a base scenario that provides a broad and reasonable representation of the natural evolution of the system and its surrounding environment, and a number of variant scenarios that represent the effects of probabilistic events. >> [NDA]

<< "Scenarios" are simplified descriptions of the repository. The system representation for the safety model thus developed is based on a "Normal evolution scenario" (SEN), which purpose is to provide a bounding value for all likely or probable future evolutions. Beside that, some altered evolution scenarios (SEAs) were defined in principle. >> [ANDRA]

<< By a stepwise process, the scenario development aims at choosing a limited number of different scenarios that, taken together, illustrate the behaviour of the system and its safety and improve the understanding of mechanism of the system by testing the reactions of the system under certain stresses. In other words, a relevant strategy of scenarios should allow defining all the situations to be considered and should allow classifying them by their occurrence in order to structure the performance assessment and the safety case by identifying the need for further work to avoid, mitigate or reduce uncertainties and to evaluate their effect. >> [IRSN]

Common Opinion:

A consensus was reached among the participating organisations regarding the key role of scenario development in safety assessments. In this context, scenario development constitutes the fundamental basis for consequence analysis. The scenario development has to indicate in a reasonable manner that all relevant FEPs have been taken into account. Furthermore, compliance with the appropriate regulations has to be shown.

2 Regulations and Guidelines

Regulations and guidelines are, in general, a worthwhile basis for both the developer and the evaluator. The developer benefits from guidance which indicates how the compliance with provided requirements could be demonstrated. The evaluator can draw on a framework given by regulations that facilitates the review work, assessments etc. of relevant documents in the licensing procedure.

Therefore regulations should not only include requirements that have to be fulfilled by the developer but also acknowledge inevitable uncertainty about future developments. Moreover it should offer guidance in areas where there is great uncertainty about the future that makes uncertainty management difficult, for example in consideration of the biosphere and human





activities.

In principle, international guidance is addressed in the respective legal national frameworks. The international guidance does not consider scenario development explicitly. However, it constitutes an initial basis for the elaboration of specified rules or guidelines with regard to the handling of scenarios. Actually, only two countries Finland and France have implemented specific regulations or guidelines concerning the handling of scenarios. Whereas the existing Basic Safety Rule RFS III.2.f from 1991 in France is currently revised. Some essential aspects of the regulations are shown in the following:

- << According to STUK's regulatory guide, scenario analysis shall cover both the expected evolutions of the disposal system and unlikely disruptive events affecting long-term safety. The scenarios shall be composed systematically from features, events and processes, which are potentially significant to long-term safety and may arise from
 - mechanical, thermal, hydrological and chemical processes and interactions occurring inside the disposal system
 - external events and processes, such as climate changes, geological processes and human actions. >> [POSIVA]
- << Basic Safety Rules RFS III.2.f. recommend that in the framework of a safety analysis, should be considered :
 - A reference situation (i.e. normal evolution scenario), considering the foreseeable evolution of the repository covering situations considered certain or highly probable.
 - Hypothetical situations (i.e. altered evolution scenarios) covering uncertain events.
- The event recommended in the Basic Safety Rules RFS III.2.f. for considering the effects, are the following situations:
 - Major climatic changes (including changes due to human activity, greenhouse effect)
 - Exceptional vertical movements or earthquakes.
 - Various possible forms of human intrusion
 - Geological barrier defects.
 - Waste package defects.
 - Engineered barrier defects (seal defects). >> [ANDRA]
 - << Implementer develops its own set of evolution scenarios taking into account the potential evolutions of the disposal system and their related uncertainties in agreement with the RFS III.2.f. However, regulators can recommend including specific situations in the development of the scenarios or integrating technological uncertainties in the normal evolution scenario.
 - The post-closure safety assessment must cover the assessment of the future behaviour of the repository and checking that individual exposure is acceptable. The approach adopted shall consist in considering a limited number of situations representative of the different families of events or sequences of events such that the associated consequences are the greatest among those of the situations of the same family. The families of events or sequences of events adopted shall be those





considered to be conceivable among all those which are a priori possible.

The events and processes constituting the situations adopted for the purposes of the safety analysis must be modelled and characterized. This characterisation shall be essentially iterative insofar, in particular, as the determination of situations considered is liable to be refined on the basis of a better understanding of the barriers and their behaviour. >> [IRSN]

Regulations in a more general sense were formulated by Czech Republic and UK which include the following:

- << Legislative regulation supposes that performance assessment evaluators will</p> describe behaviour of the system and its components and determine under all possible sets of events and processes which occur in the future, under all possible scenarios. Development of scenarios is thus an implicit requirement of legislation, but with no exact guide.
- It has been defined by regulations of Czech regulatory body (State Office for Nuclear • Safety) that the potential individual dose raised by repository existence, has not to exceed 0.25 mSv/yr for normal evolution scenarios and/or 1 mSv/yr for emergency scenarios. There exists no other quantitative limitation postulated by nuclear legislation or some other concerning scenarios. >> [NRI, RAWRA]
- << UK regulatory guidance specifies that: "After control is withdrawn, the assessed radiological risk from the facility to a representative member of the potentially exposed group at greatest risk should be consistent with a risk target of 10⁻⁶ per year (i.e. 1 in a million per year)." This specification includes all situations (scenarios) that could give exposure: "Radiological risk to a representative member of a potentially exposed group is the product of the probability that a given dose will be received and the probability that the dose will result in a serious health effect, summed over all situations that could give rise to exposure to the group.">> [NDA]

Belgium is developing general regulations on radioactive waste disposal while Germany, is currently preparing detailed regulations for scenario development. Listed below are some examples of intended specific regulations:

- << In Belgium, it is up to the operator to define for each project of disposal a relevant • list of scenarios adapted to the considered case. The aim is to establish a limited (e.g. ten or so) but relevant list of scenarios that correctly enables to appraise the possible extent of the evolution of the system along time until the very-long term, from the scenarios the most "realistic" up to the scenarios the most "pessimistic" (and less likely), also taking into account possible disruptive events.
- The strategy followed by the operator for the scenario selection should be clearly explained in the Safety Case.
- The list of scenarios should then be discussed with the regulator, and eventually • approved by him.
- With such a position taken by the nuclear safety authority, the necessity for the operator to clearly justify the reasons for the choice of the selected scenarios is crucial.
- It is not the intention to impose a particular methodology to the operator for developing





the scenarios.

- The regulatory approach concerning scenario development should consider, on the one hand, the different categories of scenarios which need to be developed and, on the other hand, how to appraise them. >> [AVN]
- << The long-term safety analysis has to comprise, the scenario development and the consequence analysis for the proof of compliance of protection objectives. The consequence analysis must underlie scenarios obtained from the scenario development. Strategy and methodology of the analyses have to be shown.
- It is to carry out a scenario development for the repository system. Here the potential evolutions of the repository system according to scientific findings, which are caused by endogenous and exogenous processes, have to be considered. Furthermore, the relevant scenarios for the safety case, with the exception of human intrusion, have to be identified.
- The scenario development has to be documented in a transparent and comprehensible manner. Each individual step has to be justified, and relevant decisions have to be explained clearly.
- Scenarios have to be assigned into the scenario classes "Likely scenarios", "Less likely scenarios", and "Scenarios that need not to be considered any further". This classification has to be justified.
- There are no requirements regarding the choice or use of a certain method, procedure and approach for the development of scenarios. It is left to the implementer to decide which tools, programmes or instruments are useful or not for the task of scenario development. >> [GRS-K]

<< There are presently no regulatory requirements and provisions of the remaining countries, Netherlands and Spain, which directly relate to the definition and assessment of scenarios. >> [NRG], [ENRESA]

The following conclusion to the issue "Regulations and guidance" by the participants, take into account the compiled facts from above and the findings from the workshop:

There are different states regarding regulations in terms of scenario development of the participating organisations and countries respectively. Some countries have established regulations, others are currently developing specific regulations or revising existing ones, and others in turn do not have any specific regulations concerning scenario development at all. Therefore, no consensus whether regulations are needed or not from the view of developers exists. For some participants, guidance in general and regulations in terms of human intrusion and the biosphere are seen as helpful instruments. Others in turn, consider guidance and regulations as a necessary basis. Different opinions exist also regarding the question if the regulator should provide a set of scenarios which have to be investigated by the implementer.

3 Terminology

Scenario development plays a key role in many technical fields and in particular in safety assessments for radioactive waste repositories across all concerned countries. Given the fact that there are different methodologies, approaches, procedures etc. for addressing





scenarios in safety assessments the meaning and also the number of terms and terminologies varies significantly. The use of different terms and additional concepts accompanying the process of scenario development does not actually facilitate the situation and might lead to some confusion. At least, it makes the communication on a national as well as international basis more difficult. In order to aim for a common harmonized terminology, international bodies like IAEA and OECD/NEA issues, glossaries and definitions of appropriate terms. In this context the following definitions were given:

OECD/NEA (Definition for scenario development) [OECD/NEA, 1992]:

Scenario development is defined as "the identification, broad description, and selection of potential futures relevant to safety assessments of radioactive waste repositories."

IAEA (Definition for scenario) [IAEA, 2007]:

Scenario is defined as "a postulated or assumed set of conditions and/or events. Most commonly used in analysis or assessment to represent possible future conditions and/or events to be modelled, such as possible accidents at a nuclear facility, or the possible future evolution of a repository and its surroundings. A scenario may represent the conditions at a single point in time or a single event, or a time history of conditions and/or events (including processes)."

As indicated above, the definition and use of terminologies is dependent on specific national frameworks and safety case methodologies. This is reflected in the contributions from the participants. The main observations made by the review of these contributions can be summed up as follows:

- Positions and content regarding definition of terms and concepts used differ widely.
- Only a few contributions contain a definition for "scenario development".
- Some deliver no definitions, neither for "scenario development" nor for "scenario".
- Some refer to definitions from other organisations and documentations respectively (IAEA, NEA, WIPP).
- Lots of additional concepts in connection with scenarios and also synonyms were used.

In the following, the different aspects of scenario development with respect to definitions, terminologies, additional concepts, use of synonyms etc. corresponding to the contributions (see appendix) are discussed:

Obsolete terms, new terms and modified definitions:

Formerly the term "scenario analysis" was often used similarly as "scenario development". Some organisations or countries have used "scenario analysis" also for the calculation of consequences with respect to defined scenarios. In the meantime the term "scenario development" has become generally accepted for the derivation and definition of scenarios. Another similar aspect is given by different concepts, which have the same meaning but one or more of them were used in former times, e.g. initial scenario and base scenario, and thus still exist in respective documentations. This situation might not only lead to some confusion but also have to be taken into account for "information preservation" another relevant subject that relates inter alia to "human intrusion". The same applies for introduced new terms such as "safety functions" which might play an essential role for scenario development in future times. It is also conceivable that definitions, e.g. scenario development, be subject of







changes in the course of time. This is a natural process in an evolving technical field. Therefore it is essential to be aware of such aspects in order to avoid misinterpretations and to take action with respect to the information and documentation of future generations.

Common approach:

As stated before and as can also be seen from the contributions is that there are many concepts and synonyms for the term "scenario". It should be noted here that there are possibly no common features concerning the used terms and concepts, e.g. the same term can have different meanings in different nations. However, a common approach exists regarding the general consideration of scenarios which can be divided in principle into two groups. That might help to differentiate the great number of used concepts in a rough manner.

All organisations consider a base case which describes a starting point for scenario development. The participants called this overall case "central scenario" that represents one of the mentioned groups. Provided scenarios that can be assigned to the "central scenario" are normal evolution scenario, base scenario, reference scenario, initial scenario, main scenario and expected evolution scenario.

Remaining scenarios were assigned to the group "other scenarios". To this class belong scenarios like altered evolution scenarios, variant scenarios, disturbance scenarios, disruptive scenarios, scenario representations, representative (umbrella) scenarios, assessment scenarios, additional scenarios, what if scenarios, what if cases, conventional scenarios, situations, human induced scenarios, human intrusion scenarios and stylised scenarios.

Further dividing in groups is presumably feasible e.g. scenarios like representative scenarios which indicate an overall group for similar scenarios, but was not intended and has actually no influence for the conclusion.

Taking into account the above mentioned aspects and the discussion on the workshop, the participants came to the following conclusion:

A wide range of definitions and concepts related to scenario development and scenarios exists. Use and meaning of terms differ significantly from country to country. But all of them have in common, that a central, or reference scenario is considered as a starting position with appropriate additional scenarios. It was stated, that definitions provided by IAEA and OECD/NEA regarding the terms "scenario" and "scenario development" constitute a valuable initial basis which can be modified / adapted according to the respective national conditions.

Another outcome of the discussion was:

There is no need to harmonise the terminology across the different countries, but a common understanding is necessary for communication and for avoiding misleading discussions.

4 Methodology

The methodology for scenario development provides the procedure for the description, definition, derivation and identification of scenarios which might have an influence on the performance of the repository for the assessment period. Developed scenarios by the





methodology build the basis for calculation cases that have to be considered in safety analyses.

Up to now, a number of methodologies and techniques were applied, e.g. directed diagrams, event and fault trees, matrix diagrams, influence diagrams, bottom up approach, top down approach, and judgemental approach [OECD/NEA, 1992], [OECD/NEA, 2001].

The considered wide range of methodologies can be also confirmed by the different contributions (see appendix), where the following observations were made:

- << Used approach is mainly deterministic according to regulations >> [ANDRA]
- << Addressing possible future evolutions of the repository by defining a base scenario and variant scenarios >> [NDA]
- << Approaches based on the Sandia Methodology and on the SKI/SKB scenario development procedure were used >> [ENRESA], [NRI, RAWRA].
- << Approaches on the basis of expert judgement were applied >> [NRI, RAWRA].
- << Top down approach was used or currently being formed >> [POSIVA], [NRI, RAWRA]
- << Scenario development on the basis of FEP classification taking into account the "barrier state" caused by the FEP (PROSA method) was performed >> [NRG], [SCK•CEN, ONDRAF/NIRAS].
- << Some are planning or taking into account safety functions for scenario development >> [ANDRA], [SCK•CEN, ONDRAF/NIRAS], [NRG], [NRI, RAWRA], [GRS-K].

It could be also observed, that the approaches of scenario development differ to some extent widely but the underlying basic approach is nearly the same in all countries. This more or less common approach comprises the description of a central or reference scenario (terms used for this scenario are normal evolution scenario, base scenario and reference scenario), and the definition of so called alternative developments which are described by scenarios named as altered evolution scenario, variant scenarios, disruptive scenarios etc. (cf. Section 3).

The procedure itself has also some components which are widely used in the respective methodologies. These components are:

- Collection of FEPs
- Screening of FEPs
- Combination of FEPs to scenarios or grouping of phenomenological situations based on repository evolution towards a normal evolution scenario (in that case, checking of results to FEPs database)
- Grouping of scenarios to representative scenarios

Although this seems a logical sequence of steps to develop scenarios, in practise the process of developing scenarios is iterative. E.g. screening of the FEPs requires some knowledge of the central evolution scenario, and will also depend on identified altered evolution scenarios.





Along the different components the involvement of expert judgement is also a common feature. Additional common features are the use of the international NEA FEP database as a basis for the collection of FEPs which are then screened and / or enhanced by specific FEPs depending from national requirements, repository sites and disposal concepts.

An essential key topic in scenario development is the question of whether the proposed methodology will be able to deliver a complete, comprehensive or sufficient set of relevant scenarios. In this context the application of systematic methodologies for organising the information and the collected FEPs might help to identify gaps and shortfalls and therefore provide more confidence of reasonable or sufficient completeness. A formalised approach for the clear, transparent and accurate documentation of screened FEPs or grouped scenarios might also support the ultimate goal of completeness.

As a result from the observations and the discussion on the workshop the participants summarises the following:

Similarly to the issue "Terminology" a wide range of methods and approaches in terms of scenario development are in use. Some of them are currently revised or will be replaced by new methods and approaches respectively. The general basis for many of the procedures is the international OECD/ NEA FEP database. Another fixed element of scenario development constitutes expert judgement. In this context, the general opinion arose that systematic approaches should be used whenever possible. It was also recognised, that expert judgement implies some subjective influences which finally cannot be avoided. Therefore, traceability of decisions by expert judgement is of paramount importance. Regarding the matter of comprehensiveness in terms of scenarios and / or FEPs it was concluded, that comprehensiveness can be achieved but it cannot be proved.

5 Application and Experience

Since scenario development is intrinsically linked to safety analyses the subject has been addressed in international and national projects for a long time. The involved organisations in PAMINA took also part in several former international projects, e.g. EVEREST, SPA, and BENIPA, or contribute to international databases or catalogues such as OECD/NEA FEP database and FEPCAT, wherein scenario development or influencing factors as well as the handling of scenarios were of great interest. Moreover, the organisations participate in international studies, working groups like the former PAAG and current IGSC, and workshops.

Hereafter some examples from national projects and working programmes in conjunction with gained experience are listed:

<< The dossier 2005 Argile considered a normal evolution scenario aiming at verifying that the repository fulfils the safety objectives assigned to it. Results of the reference calculation showed that the main barrier for the confinement of the radionuclides (except four radionuclides) is the Callovo-Oxfordian. In addition, a series of sensibility studies were performed taking into account phenomenological analysis and associated uncertainties. >> [ANDRA]

<< The underlying methodology for scenario identification was applied by Enresa in the most





recent Safety Assessment. Enresa has developed its own FEP databases for repositories in granite and clay using NEA FEP database as starting point. FEPs from other Safety Assessment exercises were also included. >> [ENRESA]

<< A matrix diagram was used to examine the interactions between FEPs. The matrix diagram

- addresses FEPs at the conceptual model level and all potential interactions were considered in a systematic manner.
- is particularly helpful for identifying second-order interactions (i.e. where FEP A influences FEP B via FEP C).
- has been used to define modelling requirements for new software modules and to assist in packaging assessment work by identifying potential impacts of specific FEPs.
 > [NDA]

<< The extended PROSA method has been applied for the safety study underlying to the license application for the closure of the Asse salt mine and the Morsleben Repository for radioactive waste. >> [NRG]

<< In preliminary safety analyses, which have been performed in the Czech Republic so far, conservative parameters more characteristic to altered scenarios then to normal evolution scenario have been used. >> [NRI, RAWRA]

<< The latest safety assessment of Posiva is TILA-99. TILA-99 did not use the concept scenario as defined in the IAEA (2003). The scenarios in TILA-99 were in fact calculation cases that could be grouped to fit within a few scenarios using "scenario" as defined in the IAEA (2003) >> [POSIVA]

<< The PROSA methodology has been applied in the SAFIR 2 report (ONDRAF/NIRAS, 2001). It appeared necessary to develop a much more detailed assessment basis and an up-to-date scenario development methodology for the Safety and Feasibility Case 1, which is scheduled to be published in 2013. >> [SCK•CEN, ONDRAF/NIRAS]

As indicated above the participating organisations have gained a lot of experience in scenario development from different activities. The lessons learnt from the experience and activities as presented in the contributions can be summarised as follows:

- Use of more realistic data in future work concerning the evaluation of the normal evolution scenario is envisaged.
- Derivation of altered scenarios in considerations of safety functions as a new option.
- There is a strong influence of expert judgement concerning the results of scenario development.
- Creating of comprehensive FEP lists is very time consuming, large lists are difficult to manage, using and implementing existing FEP list in own database is not a straightforward process.
- Significant effort exists regarding expert judgement of FEPs.
- Interpretations of other national programmes are difficult due to different usage of the terms.





Remarks and lessons learnt from the evaluator view which relates understandably to the work of their respective country are as follows:

<< The main remark regarding the SAFIR 2 report was, that the used approach is oversimplified and does not correctly reflect the reality when considering only two states for addressing the performance of a safety component: either fully-efficient or fully non-efficient. A more accurate approach, considering possible partial degradations of the safety functions has been recommended. >> [AVN]

<< "2001 Clay Dossier" and "2005 Clay Dossier" were provided by ANDRA and reviewed by IRSN concerning the deep geological disposal. Several remarks arose from the reviews, e.g. the safety analysis doesn't clearly highlight the key engineered components and their performance levels expected in relation with the safety of the disposal system. >> [IRSN]

<< Scenario development is largely based on expert judgement which is partly accompanied by subjective influence. These subjective influences should be reduced as far as possible. >> [GRS-K]

Finally, it has to be stated, that we can learn a lot from each other and should participate from the developments of our partners abroad, both in a positive and negative sense. Furthermore, it is important to document success and failures with respect to the evolution of a repository along the different stages such as siting, licensing, construction etc. which can have a strong influence on scenario development. Since the evolution of the repository can take a period of several decades, numerous generations will be involved in the entire process, so that comprehensive, suitable, and transparent documentation of successful or unsuccessful developments are vital in order to avoid same mistakes or redundant work.

6 **Developments**

Developments are mostly the result of gained experience from former work and projects, reviews or changed conditions and frameworks. In case of scenario development it was not different. Identified developments from the contributions are listed in the following:

<< International and national reviews of the dossier 2005 considered that the methodology for scenario development was quite interesting and should be pursued. Furthermore, it was recommended to develop QSA (Qualitative Safety Analysis) prior to scenario development as it was acknowledged that it could be useful for identification of calculation cases. Further safety activities will consider such a methodological development. >> [ANDRA]

<< Enresa does not intend to make a new Safety Case exercise of a deep geological repository in the near future. Enresa follows the international developments in this field (scenario development) and other fields related to the Safety Case, and can take part in EC R&D projects, but no indigenous work is being done on this topic. >> [ENRESA]

<< NDA has recently carried out work with Bristol University on the application of Bayesian Belief Networks to variant scenarios connected with climate change. Identification of variant scenarios is a basis for future work in this area. >> [NDA]

<< It is expected, that the PROSA procedure for identifying scenarios will be extended by the application of 'safety functions' for future safety studies.





And it is also expected, that it will be very useful to present the results of PA-calculations along the lines of safety functions. >> [NRG]

<< Currently top down system described in the document devoted to safety functions is being formed. This system is strictly going from top functions to daughter functions and requirements. At each level of system decomposition it will be tested whether the identified safety function is fulfilled under all external effects from outer systems. >> [NRI, RAWRA]

<< Currently a Safety Case is being performed whereas the definition of scenarios is part of the process report. >> [POSIVA]

<< From national and international (NEA, 2003) peer reviews as well as from internal discussions, it appeared necessary to develop a much more detailed assessment basis and an up-to-date scenario development methodology for the Safety and Feasibility Case 1 (SFC 1). Therefore, it was decided to base the identification of altered evolution scenarios on the availability or non-availability of the safety functions instead of on the intactness or failure of the main barriers of the repository system. >> [SCK•CEN, ONDRAF/NIRAS]

Developments from the perspective of the evaluators are:

<< Guidance has to be developed in Belgium. The guidance should first address the purpose and the role of scenarios in a SC for disposal facility (deep geological disposal or near surface disposal). In parallel, some guidance on specific topics has been developed or has to be developed. >> [AVN]

<< The new release of the RFS III.2.f is evolving in the following notions: implementation of the safety functions, reversibility and definition of a disposal concept considering spent fuel. The scenario development must take into account these new trends having a role on the possible performance of the disposal system. >> [IRSN]

<< Currently the "Safety Criteria for the Disposal of Radioactive Waste in a Mine" from 1983 are revised. The revision comprises several requirements in the context of scenario development and dealing with scenarios.

In the framework of a recent launched project the development of scenarios in consideration of safety functions is one of the main tasks. >> [GRS-K]

7 Conclusions

The findings from the workshop plus underlying facts, descriptions, and examples from the contributions constitute the foundation for this task report. The main conclusions to the topic "Definition and Assessment of Scenarios" as addressed in the respective sections, are listed below:

General aspects:

Consensus exists, in terms of the key role of scenario development in safety assessments. In this context, scenario development constitutes the fundamental basis for the further work like the consequence analysis. The scenario development has to indicate in a reasonable manner that all relevant FEPs have been taken into account. Furthermore, the compliance with the regulations has to be shown.







Regulations:

There are different states regarding regulations in the various countries. Some countries have established regulations, others currently work on specific regulations or revise existing ones, and others in turn do not have any regulations at all. Therefore, no consensus whether regulations are needed or not from the view of developers exist. For some participants, guidance in general and regulations in terms of human intrusion and the biosphere are seen as helpful instruments. Others in turn, consider guidance and regulations as a necessary basis. Different opinions exist also, regarding the question of whether the regulator should provide a set of scenarios which have to be investigated by the implementer.

Terminology:

A wide range of definitions and concepts related to scenario development and scenarios exist. Use and meaning of terms differ significantly from country to country. But all of them have in common, that a so called central scenario is considered as a starting position, with appropriate additional scenarios. It was stated, that the definitions provided by the IAEA and OECD/NEA regarding the terms "scenario" and "scenario development" constitute a valuable initial basis which can be modified / adapted according to the respective national conditions. Another outcome of the discussion was, that there is no need to harmonise the terminology across the different countries, but a common understanding is necessary for communication.

Methodology:

Similarly to the issue "Terminology" a wide range of methods and approaches in terms of scenario development are in use. Some of them are currently revised or will be replaced by new methods and approaches respectively. The general basis for many of the procedures is the international OECD/ NEA FEP database. Another fixed element of scenario development constitutes expert judgement. In this context, the general opinion arose that systematic approaches should be used whenever possible. It was also recognised, that expert judgement implies some subjective influences which finally cannot be avoided. Therefore, traceability of decisions by expert judgement is of paramount importance. Regarding the matter of comprehensiveness in terms of scenarios and / or FEPs it was concluded, that comprehensiveness can be achieved but it cannot be proved.

Application and Experience:

A great deal of experience exists due to the several international projects, studies, working groups and initiatives as well as national projects and working programmes with respect to scenario development. One of the outcomes on the basis of gained experience and cognition were, that safety functions seem to play a great role in connection with scenario development in future. Furthermore the role of expert judgement appears to be a subject for discussion in some nations concerning high effort as well as strong and subjective influence.

Developments:

The main developments identified focus more or less to the consideration of safety functions either in existing methodologies by modifications or by developing new approaches. Developments related to regulation comprise the current revision of existing safety criteria and safety requirements, respectively.

8 References

IAEA, "IAEA SAFETY GLOSSARY; TERMINOLOGY USED IN NUCLEAR SAFETY AND





RADIATION PROTECTION", Vienna, 2007 EDITION, Austria

OECD/NEA, "Systematic Approaches to Scenario Development"; Paris, 1992, France

OECD/NEA "Scenario Development Methods and Practice", An Evaluation Based on the NEA Workshop on Scenario Development, Madrid, Mai 1999, Spain, OECD 2001

9 Appendices

A1 ANDRA (France)

A2 AVN (Belgium)

A3 ENRESA (Spain)

A4 GRS-K (Cologne, Germany)

A5 IRSN (France)

A6 NDA (United Kingdom)

A7 NRG (Netherlands)

A8 NRI, RAWRA (Czech Republic)

A9 POSIVA (Finland)

A10 SCK·CEN, ONDRAF-NIRAS (Belgium)



A1 ANDRA (France)





Part 2: Definition and Assessment of Scenarios

Appendix A1: ANDRA (France)





Project acronym:

PAMINA

WP1.1

OVERVIEW OF PAST EXPERIENCE IN SCENARIOS DEVELOPMENT

Updated version 26/11/07

Andra





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STRATEGY AND KEY ELEMENTS

This present contribution from Andra aims at giving an overview of methodologies that have been used by Andra in the framework of the Dossier 2005 Argile in the four topics selected by the steering committee: 1) safety functions, 2) scenarios, 3) safety indicators and 4) uncertainties management.

The first meeting hold in Amsterdam on June 12th, 2007 was an opportunity to review contributions and discuss them for the future workshop to be held in Paris in October. The present document completes the draft provided for the Amsterdam meeting and clarifies some points discussed during the October 2007 workshop at Andra. Its structure has been revised according to the DWG common structure.

The December 30, 1991 French Waste Act entrusted Andra, the French national agency for radioactive waste management, with the task of assessing the feasibility of deep geological disposal. The Basic Safety Rule RFS III.2.f of June 1991 [i], issued by the French nuclear safety authority, provides a framework for the studies to be conducted. The protection of man and the environment are to be demonstrated. Furthermore, studies should show the ability to limit potential consequences to a level as low as reasonably possible. The concept should include a multiple barrier system, and rely on passive repository evolution without institutional control beyond a given timeframe (500 years). The studies carried out within this framework are presented in the "**Dossier 2005 Argile**" (clay) [ii] and "**Dossier 2005 Granite**" [iii].

PRIMARY REFERENCES

In the present document, the « Dossier 2005 Argile » is used as reference. Primary references include

the French Act and the series of reports submitted accordingly:

- The French Waste Act dated 30th December 1991 [iv]
- The French Safety rules namely RFS.III.2.f, guidelines [i].
- Synthesis Report, Evaluation of the Feasibility of a Geological Repository, Meuse/Haute-Marne Site (in English and French) [ii].
- Architecture and Management of a Geological Disposal System Report (TAG; C.RP.ADP.04.0001) (in English and French) [v].
- Phenomenological Evolution of the Geological Repository Report (TEP; C.RP.ADS.04.0025), (in English and French) [vi].
- Assessment of Geological Repository Safety Report (TES; C.RP.ADSQ.04.0022) (in English and French) [vii]

Other references such as the presentation made at the symposium hold in Paris in January 2007 [viii], and the INTESC questionnaire [ix] have been used when applicable.

STRATEGY AND KEY ELEMENTS

The feasibility assessment for the argillaceous site builds upon a number of key elements:





- Basic input: the inventory model of the waste and the geological site,
- Safety functions and requirement management,
- Technical solutions based on industrial experience,
- Reversible management and monitoring,
- Phenomenological Analysis of Repository Situations (PARS) [x] and detailed, coupled process modelling,
- Qualitative Safety Assessment (QSA), [xi] uncertainty management, and scenarios,
- ALLIANCES simulation platform and calculation results.

Although the process thus summarized may suggest a linear progression from basic input data to designing a "solution" and assessing its safety, the process is in fact highly iterative, with repeated feedback exchanged between the various processes (see Figure 1). In addition to the routine feedback common to parallel engineering, three main iteration loops have been identified since 1991, each corresponding to a major milestone of the program: Lic ense application for construction and operation of the underground research laboratory (in 1996), submission of the Dossier 2001 (in December 2001), and the recent submission of the Dossier 2005.



Figure 1 : Dossier 2005 Argile; three iterations loops since 1991 (1996, 2001, 2005)

In view of providing sound feedback to design, research and development and to determine residual uncertainties, the following tools have been carried out: the functional analysis (FA) [xii] to determine the safety functions and associated requirements – what do we want? -; the Phenomenological Analysis of Repository Situations (PARS) [xiv] providing a good scientific understanding based on scientific studies from surface and underground laboratory – what do we get? - ; the qualitative safety analysis (QSA) [xi] managing uncertainties and the quantitative assessment [safety and performance indicators] including sensitivity analysis –. What is the impact of a given uncertainty (or set of uncertainty factors) on the robustness of the system? – And eventually: does the concept meet the safety/acceptability criteria?

The following sections of the document describe in more details each of those topics





according to the sequence of the various stages of activities conducted in the dossier 2005 (see Figure 2).

DEFINITION AND ASSESSMENT OF SCENARIOS

SECTION 1: BACKGROUND/INTRODUCTION

"Scenarios" are simplified descriptions of the repository. The system representation for the safety model thus developed is based on a "Normal evolution scenario" (SEN), which purpose is to provide a bounding value for **all likely or probable future evolutions**. For example, the event of a few early waste package failures is included in its description. Calculation results based on this SEN are at the core of the performance assessment of the repository (see Figure 2).

Under the logic on which the Dossier 2005 is based, the altered evolution scenarios (SEAs) were first defined based on feedback from Andra's experience, analysis of situations taken into account internationally, and the recommendations of basic safety rule RFS III.2.f. The main types of "situation" to be covered and the main calculation cases were established on the basis of this definition (see section 3).

In addition to this definition, the SEN answers several distinct objectives. Its main aim is to verify that the repository, as designed and to the extent that its evolution over time is understood by contemporary science, fulfils the safety objectives assigned to it. This general objective can be broken down into several inter-related goals:

- Confirm that the performance achieved, as indicated by the chosen indicators, is consistent with the predefined threshold values. This safety objective implies the need to present a vision that exaggerates the repository's potential impact ;
- Provide an overall simulation of the repository's expected evolution, in order to assess the expected behaviour in global terms, in the form of a necessarily simplified and partially conventional representation that nevertheless aims to be as representative as possible. The aim is to assess the relative importance of the main phenomena and the performance of the safety functions. This understanding-oriented objective precludes the use of overly simplistic representations, which would make the models less representative.
- Provide a basis on which to judge the sensitivity of the level of safety to changes in the environment and the behaviour of repository components, and to use the sensitivity analyses as a tool for quantifying the repository's robustness.

In fine the reference scenario aimed at verifying the performances of the three safety functions (as listed in topic 1) using appropriate indicators (see topic 3). The SEAs are assessed by a safety model derived from the SEN safety model but taking into account the particular features of the evolutions in question (see sections 2 and 3). These SEAs allow better understanding the role of the different components of the concept. For instance:

- Waste matrices and waste packages can contribute to limit radionuclides releases in case of human intrusion (borehole),
- Seals limit the hydraulic influence of boreholes and can contribute in limiting the propagation of radionuclides in case of waste packages defects (control of the





hydraulic transitory).



Figure 2 : Representation of the various stages of the analysis.

SECTION 2: REGULATORY REQUIREMENT

The Basic Safety Rule RFS III.2.f. recommends evaluating quantitatively the following situations:

« 2.4. Situations prises en compte

Dans le cadre de l'analyse de sûreté, on retient :

- une situation de référence, correspondant à l'évolution prévisible du stockage au regard des événements certains ou très probables. ...
- des situations correspondant à l'occurrence d'événements aléatoires, d'origine naturelle ou associées à des actions humaines, qui se superposent à la situation de référence et qui peuvent conduire à des transferts préférentiels de radionucléides entre le stockage et la biosphère...

Ces situations sont précisées dans le chapitre 5 et l'annexe 2. » :

« 5.3.1. Situation de référence

Les événements à considérer sont :

• les événements liés à la présence du stockage l'impact de ce dernier se traduira par la





mise en jeu de processus associés à l'émission de chaleur, à des modifications mécaniques, physico-chimiques ou encore à la désaturation du milieu naturel autour du stockage. L'ensemble des processus de dégradation progressive des barrières artificielles (corrosion des conteneurs et des matrices de confinement, vieillissement des barrières ouvragées et des scellements, ...) devra être considéré,

- un ensemble d'événements naturels très probables (changements climatiques, subsidence et surrection). Les changements climatiques (géodynamique externe) s'accompagnent de processus tels que les cycles d'érosion/sédimentation, les modifications de l'hydrologie de surface et des circulations en profondeur.
- 5.3.2. Situations hypothétiques correspondant à des événements aléatoires

Les événements pris en compte dans ces situations seront, soit des événements de même nature que ceux retenus dans la situation de référence, mais d'ampleur exceptionnelle, soit des événements très incertains quant à leur date d'occurrence et leur déroulement. Ces événements seront répartis en deux catégories, ceux d'origine naturelle et ceux liés à l'activité humaine... ».

The event recommended in the Basic safety rule RFS III.2.f. for considering the effects, are the following situations:

- Major climatic changes (including changes due to human activity, greenhouse effect)
- Exceptional vertical movements or earthquakes.
- Various possible forms of human intrusion
- Geological barrier defects.
- Waste package defects.
- Engineered barrier defects (seal defects).

SECTION 3: KEY TERMS AND CONCEPTS

The basic Safety Rules, RFS III.2.f, require safety to be quantitatively evaluated by the means of "situations" and so as to avoid confusion with PARS, Andra uses the word "**scenario**" that encompasses all possible evolutions of the repository and that are judged as the most unfavourable in terms of consequences, among all possible evolutions that can be reasonably foreseen.

"Scenarios" are simplified descriptions of the repository [xiii]. The system representation for the safety model thus developed is based on a "Normal evolution scenario" (SEN), which purpose is to provide a bounding value for **all likely or probable future evolutions**.

PARS: In parallel with the repository definition approach and in strong interaction with it, a detailed process of description of its evolution over time is carried out. This work is based on a breakdown of the repository into situations, with each of these situations corresponding to a space and time interval within which a few major phenomena dominate the evolution of the components. This description is the object of the phenomenological analysis of repository situations (PARS) in a normal evolution situation [xiv]. Thermal, hydraulic, mechanical, chemical and radiological phenomena are recorded in this context.





For definition of scenarios, the behaviour of the repository's various constituents and its environment is represented by models [xiv]. This is the conceptualisation of the repository, whose results are presented in the dedicated documents (see the complete list in the volume titled the « phenomenological evolution of the geological repository » [vi]). The results of the conceptualisation and the performance calculation are used to confirm the safety objectives are being met, as well as to feed back the design and the knowledge acquisition approach.

This conceptualisation is itself « tainted » by uncertainties which are described in these documents. In order to proceed with a global assessment, the models are selected and concatenated to form a global safety model, which represents the normal evolution scenario. This latter can have variants and separate calculation cases in order to cover the normal evolution domain. The definition of the scenario and the results of the performance calculation are given in chapter 5 of the safety evaluation volume of the Dossier 2005 [vii].

The scenario is made up of a series of calculation cases, as follows :

- A « reference calculation », called normal evolution scenario (SEN) that sets out Andra's current knowledge of the repository's foreseeable evolution, in an approach that considers both the fruits of scientific research and the safety strategy. The purpose of this calculation is to assess factors that would increase the impact of creating a repository. To this end, it includes a series of parameters and models, choosing those based on the best available scientific knowledge, and incorporating a degree of conservatism that varies according to the uncertainties, being less conservative where the parameters or models have been validated in detail, and more conservative where substantial questions remain outstanding ;
- A series of single or multi-parameter **sensitivity analyses** that set out to rank the parameters and models by determining the ones that, if they were to vary, would have the greatest consequences for the overall assessment.

The normal evolution scenario is defined as a set of evolutions that appear probable enough to be treated as normal, rather than as a single linear scenario. Therefore, in addition to the deterministic elements, it also comprises some events defined with a high occurrence probability. For instance, the welding of the caps of the canisters is a very accurately monitored process, but it has been considered that a certain percentage of faulty quality checks would be unavoidable. Then, considering the present nuclear industry standards, a deterministic assumption of one canister's default per each waste type was considered within the SEN.

The SEN and its sensitivity studie s form a non-dissociable whole. The following points should be noted :

- Several coexistent phenomenological models can be used to account for a given phenomenon, according to the state of progress of the studies, or the accuracy with which environmental conditions are taken into account.
- Models may depend on parameters fitting and adjustment. Such adjustments are based on available experimental data ; in numerical terms, this data may not be sufficiently representative to allow a mean and standard deviation to be calculated, which leaves a degree of leeway in the choice of the model's parameters ;
- In some cases, chaining the selected models together to form the overall calculation model can result in an exaggeratedly complex representation of the repository that





causes prejudice to the good understanding of the fundamental mechanisms.

For all these reasons, certain choices must be made in order to position the « safety model », which forms the basis of the SEN assessment, in relation to the available conceptual models. They must be made in such a way that they do not result in the repository's impact being underestimated. To this end, it is important to define standard terminology for qualifying the models and parameters proposed by scientists, to ensure that the « safety » choices are made on a standardised basis common to the science and safety engineers.

Depending on the knowledge acquired for each phenomenon or material, four different types of models might be available at a given stage of the project development:

- A so called "modèle phénoménologique", or "best estimate model", is either, the model that is based on the most comprehensive understanding of the phenomenon to be modelled, and whose ability to account for direct or indirect measurements has been confirmed, or in comparison with the other available models it might be the one offering the best match between the reality that it is supposed to represent and the numerical results that it generates in the impact calculation. Examples of the former include basic physical models (Coulomb's law, etc.) and mechanistic models representing Fick's law or Darcy's law for example. Examples of the latter include all models subject to a broad-reaching experimental validation and/or a solid international consensus among experts in the field.
- A so called "modèle conservatif", or "conservative model", addresses a case in which it is possible to demonstrate that its use, all things being equal otherwise, tends to overestimate the repository's impact, compared with the results that would be obtained by taking into consideration all the relevant phenomena in the chosen parameter variation range. For example, selecting a transport model that ignores chemical retention could, in situations where retention has a potentially significant effect, be deemed "conservative".
- A so called "modèle pénalisant", or "pessimistic model", designates a model that is not based on phenomenological understanding, however empirical, but that definitely overestimates the repository's impact. For example, making an assumption that waste packages immediately release radionuclides is, except in special cases, a pessimistic choice.
- Finally, an "alternative" model stands for a model that can't be classified according to this three items list but offers a different perspective. Examples might include models that don't have an unequivocal effect on the impact, or models that appear more comprehensive than the selected reference model but have been less thoroughly validated.

A parallel classification is defined as regards parameter values:

- A "*phenomenological*" value is considered to offer the best match between the model's results and the measured results. This choice must be supported by detailed arguments which may include a representative number of measurements, a physical reasoning that demonstrates that the chosen value is the most representative based on reliable data, or a judgement by recognised experts unambiguously designating it as the most appropriate value for the study context.
- The "conservative" value is chosen among those generated by the studies and measurements which give a calculated impact in a range of high values, all other





parameters being equal. In the simplest case, where the impact increases (or conversely, decreases) as the value of the parameter increases, a value in the highest (or lowest) range of available values. "Conservative" values cannot be defined if the variations in impact are not monotonic with changes in the parameter.

- A "*pessimistic*" value is one that is not based on a state of phenomenological understanding, but is chosen by convention as definitely yielding an impact greater than the impact that would be calculated using possible values. Such values can represent physical limits. A pessimistic value can also be equal to the conservative value plus (or minus, where applicable) an appropriate safety factor that places it significantly beyond the range of measured values. A value cannot be described as "pessimistic" if the variation in impact in response to a variation in a parameter cannot be characterised.
- In order to explore the possible parameter variation ranges, one or more so-called "*alternative*" values can be suggested as a means of investigating the effect of contrasting values.

The SEAs are situations covering several altered evolutions due to various causes (e.g. a waste package failure scenario may be due either to a manufacturing defect or to the container corroding much faster than normal).

The SEA represents these different situations in a « bounding » way, i.e. it provides a description that generally overestimates the different possible effects. In the example given, the SEA would imagine the total « disappearance » of the container after 200 years. While one can assess the plausibility of each altered situation, it is a more delicate matter to assess the plausibility of a scenario that may represent several such situations in the form of stylised hypotheses. An altered evolution scenario may not represent any physically possible situation: in this case one speaks of a « conventional » or « what if » scenario. As an example, a situation such as a whole series of defective containers resulting from a quality control error however used as the « what-if » basis for the « package failure » altered scenario evolution, which considers very early loss of the functionalities of the metal containers on a series of containers and for the entire inventory. This extremely « what-if » scenario finally covers all forms of uncertainty concerning the corrosion conditions.

SECTION 4: TREATMENT IN THE SAFETY CASE

METHODOLOGY

In accordance with the French Safety Rule RFS.III.2.f, the kind of approach, which has been adopted for the safety analysis, is mainly deterministic. This is implemented at two different stages; first for the definition of the SEN (normal evolution scenario) and SEA (altered evolution scenario), and then during the scenarios modelling computation and analysis itself.

Normal evolution scenario

The definition of the normal evolution domain is progressive and is made interactively with the repository's design studies. It allows specifying the performances which can be expected from the functions. Once this domain is defined, the objective is to check through a performance assessment, first component by component, and then globally that the normal operation domain complies with the set safety objectives.





The description is not univocal: because of the space and time scales considered, uncertainties exist over the time frame of the phenomena, their spatial extension, and possibly even their nature. Therefore, it is not a matter of presenting a sure evolution of the repository, but rather a set of possible evolutions. These evolutions belong to the normal evolution domain, which combines all the likely evolutions, as well as possibly other less likely evolutions, whose consequences have no impact on safety. For example, if a container was sized to last ten thousand years, it is possible that its service life will be longer if it is placed under favourable conditions: all the service lives greater than ten thousand years belong to the normal evolution domain. On the other hand, a shorter service life which could jeopardise the repository's safety does not belong to this domain.

Because there are uncertainties relating to the repository's evolution over a one-million year period it is not possible to unequivocally define a single sequence of processes as being the reference evolution. There may be variants in the very nature of the physical and chemical interactions occurring inside the repository, and the length and spatial extent of the various phenomena are liable to vary. The concept of a « normal evolution domain » was introduced as a result of this uncertainty ; this domain represents the set of evolutions that appear probable enough to be treated as « normal ». The normal evolution scenario must represent these evolutions in a bounding manner, i.e. presenting a standard evolution having safety-related effects that are equivalent or unfavourable as compared to the situations in the normal evolution domain.

The SEN is inextricably linked with a safety calculation model that is used to evaluate the SEN, yielding a quantified impact. The model is based on :

- The internal functional analysis of the repository, conducted in order to represent the components that perform safety functions. This representation is either direct (with the component and its characteristics being modelled directly), or via the safety function's effects (for example, a component whose function is to protect the host formation may not be represented but the characteristics of the host formation used for the calculation take the aforementioned protection into account). Certain components without a safety function may be modelled (in particular, components that act as a transfer path for radionuclides are modelled, even if they have no safety functions);
- The record of phenomena liable to occur inside the repository, from a chronological and geographical perspective, as described in the phenomenological analysis of disposal situations in normal evolution, which describes the reference phenomenology (PARS). Thanks to the systematic cataloguing of the phenomena at work, this analysis can be used to determine how the normal evolution scenario unfurls [xiv];
- The repository's detailed conceptualisation which considers the initial definition of the phenomena and components to be included and proposes appropriate conceptual models and representations. The models adopted for the purpose of calculating the SEN are a subset of the proposed conceptual models, chosen from a safety perspective (see PAMINA topic on Modelling, a summary is given below);
- An initial uncertainty analysis, performed continuously via the phenomenological analysis of repository situations (PARS) and the conceptual models, which allows them to be included in either the scenario description or the choice of sensitivity studies. This analysis, in the form of a discussion of the proposed phenomena, models and parameters, is not claimed to be comprehensive at this stage. The purpose of the qualitative safety analysis (see PAMINA topic on uncertainties management) is to systematically run through the listed uncertainties and confirm that the SEN is part of a





coherent whole.

Note that the reference calculation itself has a variant, inasmuch as it uses two different hydrogeological models. In broad terms, sensitivity studies can be treated as « variants » of the SEN if their configurations are deemed to be generally less representative of current knowledge than the reference calculation (whether because they are excessively conservative or because they include anticipated research results) while nevertheless relating to the « normal evolution domain ».

Regarding the critical group living in the surface environment – the « biosphere » -, predicting the evolution of the surface environment over long periods is an exercise fraught with uncertainty. Consequently, the concept of « standard biospheres » was introduced in Basic Safety Rule RFS III.2.f. These are defined on the basis of lifestyles as they are known today, without attempting to anticipate their evolution, as this cannot currently be reliably predicted. The major determinants of climate change and surface geodynamic evolution, to the extent that they can be predicted by models, are however taken into consideration when defining the model (for example, allowance is made for the possibility of cold periods and the natural evolution of the surface hydrographic system).

Altered evolution scenarios

Under the logic on which the Dossier 2005 is based, the altered evolution scenarios (SEAs) were first defined in principle, based on feedback from Andra's experience, analysis of situations taken into account internationally, and the recommendations of basic safety rule RFS III.2.f [i]. The main types of situation to be covered and the main calculation cases were established on the basis of this definition.

The SEAs are assessed by a safety model derived from the SEN safety model but taking into account the particular features of the evolutions in question. To describe the corresponding sequences of events, sufficient knowledge of the evolution of the repository « outside the scope of normal evolution » is required. Phenomenological analyses have therefore been performed for altered situations corresponding to the scenarios.

Only after completion of the qualitative safety analysis [xi] it was possible to ensure that the defined altered evolution scenarios cover all the situations, Andra has identified as being beyond the scope of the normal evolution scenario and its sensitivity analyses.

Some sensitivity analyses may be induced by the will to evaluate the influence of a parameter, and have no direct connection with the QSA. Once the SEAs have been defined and their bounding characteristics verified by the QSA, they still have to be quantified.

The initial consideration that led to the definition of the altered scenarios is based on a breakdown by function. The intention was to define an exemplary failure situation for each of the three main safety functions, regardless of the probability of the situation described.

For the function of « limiting water circulation », the shaft, drift and module seals are important. It seemed natural to build a seal failure scenario for failure of combinations of seals [xv], differentiating those equipped with a hydraulic cut-off. Although C cell plugs do not formally have a « limiting water circulation » function, they are also included as defective components in this SEA because they are of a similar nature to the other seals.

The function of « limiting the release of radionuclides and immobilising them in the repository





» is fulfilled by different components at different time periods : containers at first (for vitrified waste and spent fuel), waste matrices, physical-chemical form of the elements released, chemical and hydraulic conditions in the disposal cells. It is difficult to define a scenario to cover the failure of all these components. A scenario involving failure of thermal waste containers [xvi] was chosen ; this would allow early release of radionuclides and their diffusion in a thermal environment, which in principle would accelerate migration beyond the near field.

The « delay and attenuate radionuclide migration » function mainly relies on the host formation, though the seal cores and disposal cells are also involved. The features involved are the predominantly diffusive conditions in the host formation and the spatial dispersal that these conditions allow, supplemented by measures to preserve the dispersal capacities of the surrounding formations. It therefore seemed useful to consider an intrusive borehole intercepting the geological formations and the repository at various points, defined in such a way as to short-circuit all barriers including the aquiferous horizons [xvii]. The aim was to disrupt the spatial dispersion and encourage advection.

These three particular situations were intended to illustrate cases of function failure, not to cover all possible situations in theory. The safety analysis tells us whether the causes envisaged are plausible, and whether other phenomena than those initially considered could cause the effects covered by the scenarios. This work was presented in Chapter 6 of TES [vii].

It seemed useful, to complement the SEAs defined above, to define a fourth one that would take into account a generalised failure of all safety functions. This is based neither on feedback from scenarios defined by Andra's counterparts, nor on altered situations identified through the QSA. It is a « severely degraded evolution » scenario that consists of systematically reducing the performance of the safety functions to exceed the scope of the normal evolution scenario. The first three SEAs serve to test the degree of redundancy between the safety functions : the idea is to minimise or eliminate the contribution of one function, and then study whether the others are sufficient to comply with the safety objectives. The « severely degraded evolution » scenario assesses the complementary nature of these functions : by degrading all of them at once and comparing the results with the results of a normal evolution scenario, one can observe whether minimal performance levels, below what is normally expected, complement each other sufficiently well to control the impact.

Modelling

As regards the modelling and computation of the scenarios, the approach is also mainly deterministic. Usually, computation cases are carried out with a given set of fixed parameters. Comparisons are made by changing only one parameter at a time, or in any case a limited number (See details about the models and parameters selection and use in section 3). By testing the influence of a set of determined parameters on the performances of the repository system, the results of the SEN and SEA calculations enabled to identify the most influential elements and to deduce the lessons learnt on the role of the components with regard to the main safety functions.

In addition to these results, a probabilistic study was carried out taking into account the simultaneous variation spectrum of the various parameters [xviii]. That consists in a sensitivity analysis exercise, conducted by way of an illustration on the iodine and selenium of C1/C2 glass, which is designed to back up the lessons learnt of the deterministic studies



and assess the effects of joint variations of several parameters. From this type of calculation, it is possible to deduce information on the uncertainty of the result by situating the position of the various deterministic calculations on an overall distribution curve. It is however difficult to draw direct lessons from this type of assessment as it depends on the probably distribution laws that were adopted. Consequently, the objective adopted by Andra at the stage of this initial methodological exercise is first and foremost to identify the parameters which, due to their uncertainty, have the greatest influence on the uncertainty of the result. This does not mean proceeding with a probabilistic treatment of the impact of the repository. In accordance with RFS.III.2.f [i], the safety approach remains deterministic. The calculation is limited to the indicators as such the molar flow rate out of the Callovo-Oxfordian and access structures and the distribution of radiological impact is not assessed. The altered evolution scenarios and their results are presented in chapter 7 of TES [vii].

APPLICATION

The dossier 2005 Argile considered a normal evolution scenario aiming at verifying that the repository, as designed and to the extent that its evolution over time is understood by contemporary science, fulfils the safety objectives assigned to it. However, it was not in the sense of the CIPR 81 a real prediction of the impact of the repository.

The reference calculation results showed clearly that the main barrier for the confinement of the radionuclides is the Callovo-Oxfordian, which is the host formation of the repository. It attenuates and delays all the radionuclide flows. It only allows four radionuclides to exit over a time-scale of several hundred thousand years.

In addition to the reference calculation, a series of sensibility studies was performed in order to evaluate the influence of the parameter choices or of models that are different from those chosen for the reference calculation. The majority of the studies correspond to models or sets of parameters that are less favourable than those selected for the reference calculation. In this way, any remaining uncertainties about the values selected for the reference calculation, which were already cautious, will be covered.

Other sensitivity studies were also conducted with alternative models or with sets of parameters that are less pessimistic than those of the reference calculation. These studies are conducted either to establish predictions in order to evaluate a potential margin, or in order to integrate recent results that are less cautious than those of the reference calculation.

The sensitivity analyses also make it possible to classify the parameters and models according to their influence on the safety indicators (the impact, or any other intermediary indicator). The presented sensitivity analyses thus provide useful information for topic 4, which gives details about existing uncertainties concerning phenomena inside the repository. The sensitivity studies are summarised in Table 1. They have been divided into three main categories:

Sensitivity studies concerning parameters for the Callovo-Oxfordian, swelling clay, and concrete

The majority of the considered parameter values are conservative, such as for the following:





- permeability in Callovo-Oxfordian
- the hydraulic, transfer, and retention parameters in the EDZ
- the transfer and retention parameters in the Callovo-Oxfordian, coupled with those of the swelling clay and concrete

Some studies were conducted with values that appear to be less pessimistic.

• Sensitivity studies concerning the kinetics of release by waste packages

Only spent fuels, C waste, bituminised sludge packages, and inorganic packages that do not release hydrogen (reference packages disposed in B1x type cells) are included in sensitivity studies, with parameters that are less favourable than those used in the reference calculations. Since the other reference packages were represented by a labile source term in the reference calculation, they do not require such studies.

Furthermore, a sensitivity test is performed for the spent fuels using a model based on the conventional dissolution of the matrix (and not the radiolytic dissolution used in the reference calculation); this model results in slower release kinetics.

And finally, a sensitivity study was conducted to make predictions in order to evaluate the potential advantages of adopting durable concrete overpack for waste packages disposed of in B1x cells.

• Sensitivity studies concerning the overall calculation model.

This final set of sensitivity studies tests transfer methods for radionuclides other than those considered in the reference calculation. This category includes the following studies:

- Study of a radionuclide transfer under hydraulic transient influence, and thus of stresses caused by gas in particular. Studied for B1x wastes and CU11 spent fuels.
- Study using different properties for the overlying formations, in order to take into account a slower diffusion in the Kimmeridgian and in the C3a horizon of the Oxfordian. Studied for iodine-129 from CU1 spent fuels.
- Study testing the influence of the hydrogeological model of the overlying formations on the impact. Studied for iodine-129 from CU1 spent fuels.





Sensitivities		
1. Sensibility to parameters of Callovo-Oxfordian, bentonite, and concrete in cells		
1.1 – Conservative Callovo-Oxfordian permeability (factor of 10)		
1.2 – EDZ. Pessimistic fractured zone : K = 10-6m/s, diffusion coefficient of the water in water (Dp = $2.10-9m^2/s$), no delay, no solubility Conservative micro-fissured zone : K = $5.10-9m/s$, (De = $1.10-11m^2/s$; w= 0.04); degraded retention capacities		
1.3 – Conservative transfer and retention parameters (engineered barrier and Callovo-Oxfordian) : sorption, diffusion and solubility limit for the Callovo-Oxfordian, sorption and solubility limit for the bentonite and the concrete		
1.4 – Partition coefficient for lodine of 10 ⁻³ m ³ /kg in the Callovo-Oxfordian		
1.5 – Callovo-Oxfordian thickness of 160 m		
1.6 – Phenomenological thermal evolution		
2 – Sensitivity to waste packages		
2.1 -B1x type packages : conservative release rate parameters		
2.2 -B2 type packages : bituminised sludges : rate = 10-3/year		
2.3 -B1x type package : durable container used for B wastes		
2.4 – C waste packages : conservative parameters of model $V_0.S \rightarrow V_r$		
2.5 -C waste packages : pessimistic model V ₀ .S		
2.6 – Spent fuels : Conservative parameters (radiolytic dissolution)		
2.7 – Spent fuels : Control model via the solubility of uranium		
3 – Sensitivity concerning the overall calculation model		
3.1 – Transfers during hydraulic transient		
3.2 – Diffusion properties of semi-permeable layers of overlying formations		
3.3 – Pathways in the overlying formation model		

Table 1 : List of sensitivity analyses of the normal evolution scenario

Four altered scenarios were considered:

- the « waste package failure » scenario [xvi], •
- the « seal failure » scenario [xv], •
- the « borehole » scenario [xvii], •
- and the « severely degraded evolution » scenario (worst-case scenario) [vii]. •

The SEAs are situations covering several altered evolutions due to various causes (see section 3 and see topic 4). They were compared to RFS.III.2.f recommendations:





- Major climatic changes. No significant effects are expected due to the depth of the repository installation. Possible consequences departing from evolution situation might be similar, in the worst case, to significant modifications in the living and feeding habits of individuals residing on the site and in the characteristics of surrounding formations. These scenarios are covered:
 - in a preparatory study based on a critical group associated with a cold biosphere and demonstrating that said group is less pessimistic than the reference group chosen,
 - by selecting 'deep' outlets close to the site and not sensitive to the possible damage of surrounding formations near the surface,
- Exceptional vertical movements or earthquakes. The tectonic risks in the Meuse / Haute-Marne site are low. The effects of a possible earthquake were taken into account in the qualitative analysis (see section) and shown to be negligible in the engineered structures and in the rock ;
- Various possible forms of human intrusion, covered by the 'borehole' altered evolution scenario;
- Geological barrier defects. The basic safety rule proposes considering sedimentary hiatuses in the form of sand lenses for the argillaceous sites. Such structures are excluded in the Callovo-Oxfordian argillite. The safety analysis conducted in chapter 6 (see section) indicates that the only undetected structures possibly present are minor structures with limited extent and release. The effects associated with such structures are theoretically very limited. They are covered by the borehole scenario as a last resort
- Seal failures, covered in a specific altered evolution scenario "seal failure"
- Waste package defects, at least for sensitivity analysis purposes. This possibility is covered in the normal evolution scenario and in a specific altered evolution scenario.

Sensitivity calculations may be performed on the SEAs in order to :

- cover variants of the situations envisaged in the main calculation, usually variants that constitute aggravating circumstances ;
- cover phenomenological uncertainties on the parameters.

The results of the SEA calculations (basic case and sensitivities) must be compared with thresholds. Basic safety rule RFS III.2.f. gives no such thresholds, since it seems difficult to define the acceptability of the results of SEAs generically. After all, as already stated, the SEAs are situations covering several altered evolutions due to various causes (e.g. a waste package failure scenario may be due either to a manufacturing defect or to the container corroding much faster than normal). The SEA represents these different situations in a « bounding » way, i.e. it provides a description that generally overestimates the different possible effects. In the example given, the SEA would imagine the total « disappearance » of the container after 200 years. While one can assess the plausibility of each altered situation, it is a more delicate matter to assess the pla usibility of a scenario that may represent several such situations in the form of stylised hypotheses. In some cases, an altered evolution scenario may not represent any physically possible situation : in this case one speaks of a « conventional » or « what if » scenario.






Appendix A1: ANDRA (France)

SECTION 5: LESSONS LEARNED

KNOWLEDGE/EXPERIENCE GAINED WITH THE APPLICATION OF SCENARIO DEVELOPMENT IN THE CONTEXT OF SAFETY ASSESSMENT

The repository performance studies highlight a significant number of results for safety analysis. Safety functions are guaranteed a good level of performance, in both the reference calculation and in the sensitivity studies.

For the « resisting water circulation » function, the diffusive transport regime dominates in all configurations within the Callovo-Oxfordian host rock, and in most of the structures. It should be noted that this is not solely due to the efficiency of the seals: even when this is degraded in the sensitivity study, the flows remain limited overall, since the water from the Callovo-Oxfordian is insufficient to supply them.

For the function of « limiting the release of radionuclides and immobilizing them in the repository »: The low solubility of many radionuclides in the cells means that their impact is heavily restricted; this is especia lly the case of Selenium-79. The containers and over-packs contribute an element of confinement, helping to delay the occurrence of dose maxima, but without strong influence on their magnitude. The properties of the Callovo-Oxfordian attenuate the flows even in the case of transfer in a thermal environment.

For the function of « delaying and reducing the migration of radionuclides », the diffusion times are slow in the Callovo-Oxfordian and enable a decay of all the radionuclides that could contribute to the impact, except for iodine-129, chlorine-36 and selenium-79. The last two are, however, significantly reduced. The transport parameters prove sensitive in terms of the impact of these three radionuclides. In the argillites, the results reveal that the most influential factors are the diffusive transport parameters for the soluble, unsorbed elements like iodine and chlorine.

This function analysis shows that the Callovo-Oxfordian is a particularly important component, whose characteristics ensure a good level of safety function performances, even in the event of mediocre operation of other components (defective containers, inefficient seals) or even of degraded properties of the geological medium itself.

ON GOING OR PLANNED PROJECTS

International and national reviews of the dossier 2005 considered that the methodology for scenario development was quite interesting and should be pursued. Furthermore, it was recommended to develop QSA analysis prior to scenario development as it was acknowledged that it could be useful for identification of calculation cases. Further safety activities will consider such a methodological development.





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Appendix A2: AVN (Belgium)

A2 AVN (Belgium)





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European integrated "PAMINA" - Project WP 1.1 – AVN Contribution "Scenario development"

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1. Introduction

The need for carrying out a scenario development in safety assessment of radioactive waste disposal facilities arises from the fact that it is virtually impossible to predict exactly what will be the evolution of the disposal system through time.

A scenario describes one possible future of the disposal system, corresponding to a combination of events and processes together with their characteristics and their chronological sequence. The expression scenario development is used both for the identification of the set of scenarios that will be representative of the different states of the disposal facility and for the identification, general description and selection of the possible safety-relevant features of the disposal system for one defined disposal evolution.

Within the safety assessment of a radioactive waste disposal facility, for reference evolution or altered scenario, scenario development then aims in a first step to modelise the applicant's understanding of future situations that might "*realistically*" happen by giving a concrete illustration of the way the disposal system may evolve through time until the very long-term, considering either probable or less likely evolution and/or assumptions. In this sense, it aims to examine the reaction of the disposal system or one of its component to different assumptions in order to increase understanding of how the system functions. In a second step, scenario development aims to quantify the ability of the disposal concept to fulfil the main safety functions assigned to it (isolation and confinement properties).

Furthermore, the scenario development is also an important step for communicating with the public and the different stakeholders, as it gives a concrete illustration of the expected evolution of the disposal system through time, considering some realistic assumptions (for the reference scenario), as well as its possible evolution considering more pessimistic assumptions (for the altered evolution scenarios).

2. Definition of terms and used concepts

The Belgian safety authority distinguishes between the following categories of scenarios:

the "reference evolution scenario"² is aimed to illustrate what is the expected evolution
of the system, and thus to give a general sight of the global level of safety of the
system. The reference evolution scenarios correspond to the foreseeable evolution of
the repository with respect to the most likely effects of certain or very probable events
or phenomena (Note that there can also be a set of several reference evolution
scenarios). This type of scenario takes into account the relevant Features, Events and
Processes (FEP's) that are present or will take place with certainty or near-certainty.
The reference evolution scenario therefore describes the most likely sequence of
events to take place after the closure of the repository. This is the undisrupted
performance of the disposal system, in which natural processes will lead to a slow and



² In some countries, the "reference evolution scenario" is sometimes named "normal evolution scenario"



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gradual degradation of its containment capability according to the design and used technologies.

- the "altered evolution scenarios"³ represent less likely but plausible modes of repository evolution (e.g. degradation of components occurring more rapidly than expected). They should enable to design the different components of the facility so that the overall level of safety of the system remains consistent with the main objective of "protection of the man and the environment, now and in the future", even in case of occurrence of some possible low probability events. Consequently, they are bounding for a well-identified set of events, process ou features.
- "Beyond Design" scenarios portray extreme and very unlikely events (e.g. extreme ice-age or a major seismic event), for which it appears that it is not reasonably possible to thwart the occurrence or the consequences. By definition, these scenarios cannot be used for the repository design. However, the development of this type of scenarios is important for confidence building in the safety of the repository. The justifications leading to the classification of the scenarios to "Beyond Design Scenarios" should be provided in the safety case as well as their scope and limitations. This substantiation should be part of the approval by the safety authority.
- Imposed or conventional scenarios known as "What if" scenarios should also be considered. For such scenarios, the occurrence of an event or random phenomenon is postulated although it seems possible to exclude it through design or the level of knowledge available (e.g. of "what if" scenario: postulated failure of a confinement barrier for undefined reasons). These scenarios are not meant to represent a realistic situation. They are used mainly for assessing the relative importance of the components of the disposal system, for exploring the robustness of the system, for helping to provide multiple lines of reasoning and hence for building confidence in the safety case,
- Finally, scenarios involving unpredictable future human actions leading to the partial or full degradation of the confinement properties of the disposal facilities (e.g. human intrusion) should be considered independently from the other types of categories. Moreover, for the particular case of human intrusion scenarios, a distinction should be made between the direct effects and differed effects of human intrusions:
 - For deep disposal, due to the nature of the waste to be disposed of (high level and long-lived radionuclides), the long-term protection of the intruder himself (voluntary inadvertent intrusion) could not be ensured by any reliable system of protection. For this reason, direct effects of human intrusion are considered to belong to "Beyond Design Scenario" class.
 - On the contrary, differed consequences of human intrusions can be regarded as a particular set of accelerated degradation of the confinement properties of the disposal system. Consequently, they are considered to belong to "altered evolution" scenarios.

Examples of such postulated scenarios are drilling for water, exploratory drilling with the extraction of cores, operation of a mine near the repository or direct physical human intrusion into the disposal facility. For defining the list of human intrusion scenarios to be developed for



³ "Altered evolution scenarios" may also be named "degraded scenarios"



a particular disposal facility, one has to take into account the regional context of the repository (e.g. presence of natural resources...).

3. Regulatory context

The national regulatory framework in Belgium is developed by Federal Agency for Nuclear Control (FANC) and its technical support AVN starting from international regulations issued by the IAEA, the ICRP and the European Union (Directive 96/29/Euratom).

3.1 Regulations and guidance

The main principles set out by IAEA in its publication 111-F [1] are derived in 8 principles applicable to the management of radioactive waste on the whole Belgian territory. They are defined in a document named "strategic note related to the licensing procedure for radioactive waste disposal facilities", issued in March 2007.

The national regulation applicable to radioactive waste disposal facilities is still under preparation in Belgium. The more recent developments that provide some feedbacks for scenario development have covered the following fields:

- Seismic hazard assessment for radioactive waste disposal projects (earthquakes belong to repeating events that could affect the confinement properties of a disposal concept),
- Management of human intrusion risk for near-surface disposal facilities (human intrusion is a very specific type of scenarios, see further).

3.2 Requirements and expectations

Whereas the regulations issued in some countries tend to impose to the operator to study a fixed list of scenarios in the safety assessment of a disposal facility, it is not intended to define such definite list in the future Belgian regulation. In Belgium, it is up to the operator to define for each project of disposal a relevant list of scenarios adapted to the considered case. The aim is to establish a limited (e.g. ten or so) <u>but relevant</u> list of scenarios that correctly enables to appraise the possible extent of the evolution of the system along time until the very-long term, from the scenarios the most "realistic" up to the scenarios the most "pessimistic", also taking into account possible disruptive events. In any case, the strategy followed by the operator for the scenario selection should be clearly explained in the Safety Case.

The list of scenarios should then be discussed with the regulator, and eventually approved by him.

With such a position taken by the nuclear safety authority, the necessity for the operator to clearly justify the reasons for the choice of the selected scenarios is crucial.

Likewise, it is not the intention to impose a particular methodology to the operator for developing the scenarios: two possible methods can be envisaged, the first one starting from the list of features, events and processes (FEP's) that may affect the system and the alternative one starting from the safety functions of the disposal system.

The regulatory approach concerning scenario development should consider, on the one





hand, the different categories of scenarios which need to be developed and, on the other hand, how to appraise them. These two considerations are further detailed in the following paragraphs.

For judging the acceptability of the impacts calculated for the various studied scenarios, it is intended to fix two types of values in the future Belgian regulation:

- For the reference evolution scenario, which corresponds to the <u>expected</u> evolution of the system (i.e. with high probability of occurrence), some additional legal <u>limits /</u> <u>constraints</u> to be strictly met will be fixed, such as dose constraints (e.g. a fraction of the dose limit for public) or risk constraints.
- For altered scenarios with lower probability of occurrence, some <u>reference values</u> may also be mentioned in the regulation to define the acceptable level of impact, but without prejudice to the fact that higher values may not necessarily be a cause of rejection (for those scenarios, the resulting calculated dose could be significant, and higher than the dose constraint. It may still remain acceptable, as the probability of the assumptions taken into account is lower).

For "what-if" scenarios, no specific comparaison values are set up. Variations of some representative parameters of the confinement properties are evaluated through the use of "what-if" scenarios. More than a strict comparaison with a fixed value like a criterion or reference value, the concern is on the amplitude of the considered parameters in order to detect common failure or specific sensibility to some components failure.

From a regulator's point of view, the aim of the scenario development is to tackle a number of possible, less possible or postulated situations through a limited number of different scenarios. Scenarios taken together, illustrate the behaviour of the system and its safety in a variety of circumstances, from the more expected ones to the less probable ones. The scope covered by scenario development should address the whole spectrum of possible evolution as described by the five types of scenarios listed above. The determination, by the operator, of scenarios thus constitutes a major point of interest for the Nuclear Safety Authority when assessing a safety case of a disposal facility.

It is also expected that the assumptions made for each scenario and their scope are clearly described and justified, and that the various types of uncertainties attached to the scenarios are deeply discussed. It implies to discuss the "degree of belief" or "likelihood of occurrence" of the various scenarios when developing them.

The classification of each scenario in one of the five categories listed before should also be explained and justified by the operator.

The expectations of the Safety Authority concerning the scenario development also evolves through the different stages of the licensing project (see in [2]):

 At the conceptualisation stage (very early in the project development), the scenario development should mainly present a generic list of scenarios gathering the different scenarios intended to be developed for assessing the long-term safety of the disposal system (main assumptions for the definition of the reference evolution scenario according to the selected concept, description of the altered scenarios which will be considered...). The rationale for establishing this relevant list of scenarios should be presented, and the reasons for rejecting particular types of scenarios from this list





should be clearly expressed and justified.

• At the siting stage, the generic list of scenarios defined at the previous step should be refined to take account of the specific site characteristics, according to the knowledge acquired during the preliminary site investigations for each possible site.

The need for developing some new scenarios may also arise at the siting stage, if the particularities of the site characteristics require it.

As concerns the decision-making process, the scenario development at the siting stage is also particularly important since it may be used for assessing the ability of a particular site to comply with the safety requirements.

- At the design stage, it is expected that the reference evolution scenario be precisely defined, as well as the altered evolution scenarios which have been chosen for designing the various components of the disposal system according to their allocated performance in the safety demonstration. In complement to the reference evolution scenario and the various altered evolution scenario, the development of more pessimistic scenarios ("what if" scenarios and "beyond design" scenarios) at the design stage enables to explore the robustness of the disposal system, which is essential for building the confidence in the safety case.
- At later stages in the repository lifetime (construction, operation, closure, postclosure), as the level of knowledge of the system characteristics progressively increases, the improvement in the scenario development mainly consist of reducing the level of uncertainties attached to the scenarios by using the return of experience gained from the earlier stages (results of measurements acquired through the monitoring programme, experience acquired during operation...).

3.3 Experiences and lessons learnt

In the "Safety Assessment and Feasibility Interim Report" issued by ONDRAF/NIRAS for geological disposal [3], the "altered evolution scenarios" have been built using a systematic approach, where the disposal system and its environment have been reduced to the two main barriers (namely the engineered barriers and the geological barrier) and the hydrogeological component. All possible states of the disposal system have then been analysed by the operator assuming that each of these three main components can either be present (active) or absent (not effective). As a result, a matrix presenting the eight possible states of the disposal system has been built. In the pre-project for Near Surface Disposal facility, a similar approach based on three main safety-related components has been followed.

The main remark which has been issued during the review / assessment of this report by both the Belgian Safety Authority (FANC / AVN) and an international peer review team [4] was that this over-simplified approach does not correctly reflect the reality when considering only two states for addressing the performance of a safety component: either fully-efficient or fully non-efficient. A more accurate approach, considering possible partial degradations of the safety functions has been recommended.

3.4 Developments and trends

Safety Authority and its technical support are convinced that guidance on scenarios has to





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be developed in Belgium. The guidance should first address the purpose and the role of scenarios in a safety case for disposal facility (deep geological or near surface disposal). In parallel, some guidance on specific topics have been developed or have to be developed.

For instance, in view of the development of a project of near surface repository for low-level waste, the Belgian nuclear safety authority has recently worked on the preparation of a particular guidance related to the assessment of human intrusion scenarios.

This project of guidance sets the assumption that the probability of occurrence of human intrusion is equal to the unity at the end of the institutional control period. Consequently, a particular dose constraint will be fixed that will have to be met for the period following the release of regulatory control of the site. For the periods before, reference values may also be fixed, as comparison points for judging the acceptability of the calculated impacts of human intrusion scenarios occurring during the period of institutional control.

In this project of guidance, different critical groups have been considered for studying the direct or indirect effects. The interest of considering each exposure route independently from the others, as it is done in [5], has also been recognized.

Other specific guidances have to be developed for exemple, on robustness assessment.

4. Analysis and synthesis

4.1 Main advantages / possible difficulties

The main advantage of the scenario development is to provide to the stakeholders a concrete illustration of the foreseen evolution of the disposal system, which enables to better appraise the resulting potential impact on the environment.

Each scenario describes one potential evolution of the disposal system, making assumptions on a number of elements (parameters, influences of some processes, disruptive events...). In the scenario development process, the main difficulty will probably lie in demonstrating that the set of selected scenarios adequately addresses all the possible ranges of the various parameters.

It is therefore essential for all the parties involved (researchers in the various fields, safety authorities etc.) to be able to verify that the safety assessment has given due consideration to the most relevant FEP's and scenarios. An emphasis in scenario development must therefore be on transparency of the methodology and traceability so that input can come from all the relevant fields of expertise.

4.2 Feasibility

From a conceptual point of view, the feasibility of developing a set of scenarios for describing the possible evolution of a disposal facility is obvious, whatever the type of facility and whatever the stage of development of the project.

However, the technical feasibility of the scenario development stage highly depends on the acquired level of knowledge on the FEP's, amongst which the site characteristics, the main physico-chemical processes affecting the radionuclide transport and on the performances of the different components of the system. The key point is on the iteration approach on







scenarios and their improvements. If the acquired knowledges imply to modify a scenario, the applicant has to provide the needed justifications.

4.3 Selected approach

On the proposed approach adopted by the applicant, no specific requirements are defined. It is up to the applicant to justify the underlying reasons of his approach. Up to now, depending of the nature, the scope and the final role of the scenario in the safety case, the appraisal should be more seen as a case-by-case basis. Some considerations for this case-by-case appraisal are developed below:

- Quantitative assessments of scenarios may be wholly deterministic or may seek to capture a range of uncertainties within them using probabilistic methods. Quantitative assessments may, for example, calculate doses, probability distributions of doses, or risks;
- The underlying assumptions may be best estimate, conservative or stylised, or some combination of all three. A stylised approach may be used either for a whole scenario or for only part of a scenario (e.g. for representation of the biosphere or for representation of the very long term evolution of the repository system);
- The shortcomings of building all scenarios into a single overall probabilistic assessment have been highlighted in [2], which reports that the attempts made in the past to use this approach have proved unsatisfactory for a number of reasons like the burden of scenario development is not avoided, issues with the generation of probabilities, difficulties of interpretation of the results, low probability investigation and the lack of flexibility;
- The way the uncertainties are managed in the safety approach.
- The deterministic scenarios might also be of the type that seeks to represent the expected evolution of the repository system. Moreover deterministic scenarios could be used for modelling extreme events that are still within the range of realistic possibilities (bounding cases), and those that do not aim to be realistic but rather explore the robustness of the system ("what-if" cases, cf. section 2.2). At the end of the day, the best solution probably lies in a safety analysis involving a limited number of different scenarios, some possibly being wholly deterministic and others seeking to capture a partial range of uncertainties. Uncertainties not included within scenarios would be captured by differences between scenarios.

4.4 Integration in a step-by-step process

Scenario development helps to structure the review of the safety case and is a valuable tool to identify where further work should be directed to avoid, mitigate or reduce uncertainties and to evaluate their effect. There is a requirement to establish and maintain a clear structure for the safety analysis throughout its development, including the presentation of scenarios. In particular, the developer of the disposal facility should not present isolated pieces of work (such as isolated scenarios) to the regulator, or to a wider audience, divorced of a clear statement as to how these relate to the safety analysis as a whole: **it is essential to maintain the link between safety assessment and safety strategy.** The scenario development also represents an important tool for designing the facility as it enables to fix the level of performance of the different safety components so as to maintain the radiological impact to the critical group at a sufficiently low level. Safety authority will focus also their







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review on the way these links are established and maintained during the project. The way these links are working is a point of attention from the safety authority.

Consequently, scenario development is an important step in the safety assessment of a radioactive waste disposal facility and it plays an important role in demonstrating the robustness of the disposal system, since it is established that the concept of robustness of a disposal system component means that the component's characteristics associated with its safety function(s) is (are) preserved when faced with a spectrum of reasonably foreseeable stresses despite any residual uncertainty associated with this component. The same content can be extended to a group of components.

4.5 Data requirements

There is broad international consensus as to the methodology to be applied for scenario identification, based on FEP's (features, events and processes) lists (but there is no formal requirement on using FEP's, as mentioned before). The selection of scenarios thus entails a good qualitative understanding of the features, events and processes that significantly affect the evolution of the disposal system, in order to reduce as much as possible the uncertainties attached to the parameters and to the modelisation in the scenario development. These FEP's lists are compiled at international level (NEA, IAEA) and regularly reviewed. When FEP's are considered, it is essential to consider both generic FEP's arising from the literature as well as site-specific FEP's, to take account of the particularities the project may present.

4.6 Uncertainties

An initial cause of uncertainty in the disposal system is associated with the actual scenario descriptions, especially the uncertainty whether all relevant scenarios have been thoroughly included in the safety assessment.

Significant uncertainties can arise with altered scenarios: uncertainty about the form and scope of the considered phenomenon and of its impact on one or more components of the disposal system and on the probability and time of occurrence. With some scenarios this can lead to the consideration of a large number of possible variants. One class of uncertainty in the scenario description is the uncertainty caused by possible evolutions of the system, e.g. as a result of climate changes or changes in the geology of the site.

The definition of the "critical group" in the scenario development may also constitute an important source of uncertainty attached to scenario. When considering long-term periods, the characteristics of the biosphere and the critical group (in particular its eating habits and lifestyle) can only be hypothetical. So, a <u>stylised</u> approach becomes the most appropriate when dealing with this topic. Stylisation is thus a way of bypassing unquantifiable uncertainties, especially those attached to the scenarios.

Treatment of the different types of uncertainties (uncertainties attached to parameters, models and scenarios) belongs to the most important methodological elements that can help build the confidence in the long-term safety assessment. Treatment of uncertainty in the description of the scenarios can be made by:

• Elaborating a structured and transparent scenario-development so that experts from the various relevant research fields can appreciate which processes and phenomena have been considered – or not – and for what reasons;





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- Analysing the disposal system in the various periods of time, and using different types of calculations, lines of reasoning and safety indicators for each period;
- Using internationally established and verified FEP databases;
- Proceeding to international peer review of the scenario development.

4.7 Improvement potential

Most promising future improvements within the scope of scenario definition and assessment include the comprehensiveness of the developed scenarios, the treatment of uncertainties attached to these scenarios and the assessment of the "degree of belief" of each scenario.

4.8 Harmonization – Integration

An international agreement on a definite list of scenarios to be studied for disposal facilities of the same type would be valuable in order to be able to compare different concepts, especially in the case of human intrusion scenarios for near-surface disposal facilities.

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A3 ENRESA (Spain)





Part 2: Definition and Assessment of Scenarios

Appendix A3: ENRESA (Spain)





Proposal/Contract no.: FP6-036404

Project acronym: PAMINA

Project title: PERFORMANCE ASSESSMENT METHODOLOGIES IN APPLICATION TO GUIDE THE DEVELOPMENT OF THE SAFETY CASE

- Instrument: Integrated Project
- Thematic Priority: Management of Radioactive Waste and Radiation Protection and other activities in the field of Nuclear Technologies and Safety

WP1.1 Definition and Assessment of Scenarios ENRESA contribution

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- PUPublicPPRestricted to other programme participants (including the Commission Services)XRERestricted to a group specified by the consortium (including the Commission Services)XCOConfidential, only for members of the consortium (including the Commission
- CO Confidential, only for members of the consortium (including the Commission Services)

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1 Background and introduction

This document describes the experience of Enresa regarding the definition and assessment of scenarios in the Performance Assessment (PA) of HLW repositories in granite and clay. The methods and results presented correspond to Enresa's second and most recent cycle of performance assessment exercises consists of one performance assessment for a repository in granite [1] and a second PA for a repository in clay [2].

2 Regulatory requirements and provisions.

The acceptance criteria for radioactive waste disposal facilities was set in 1987 by the following statement of the regulatory authority (CSN): "to ensure safety individual risk should be smaller than 10^{-6} yr⁻¹, that is the risk associated to an effective dose of 10^{-4} Sv/yr".

There are no specific requirements on the definition and assessment of scenarios. No set of scenarios to be analyzed has been defined by the regulators either.

3 Key terms and concepts.

No systematic definition of the concepts related with scenario definition is done in [1] and [2]. The different terms are used with the common meaning in this field of knowledge.

4 Treatment in the Safety Case

Enresa's programme for geological disposal is at the stage of feasibility studies. The siting studies were discontinued in the late 1990's. There are many data available for Spanish granite and clay formations which appear to be favourable in principle, but no detailed characterisation has been made for any formation. As a consequence, the performance assessment studies were done for generic sites defined on the base of the real data available, complemented as required with plausible data obtained using expert opinion, on the base of data available from different places, or derived from general geological knowledge. To tackle the foreseen span of characteristics of the diverse potential sites, the performance assessment studies considered several alternative sets of geological data, which were used as inputs for corresponding alternative cases of the reference scenario (definition given below).

For the same reason, the objectives of the scenario analysis were not to define actual scenarios, nor to predict the evolution of the hypothetical site, but rather to first identify qualitatively the different scenarios that would likely have to be addressed at more advanced stages, which were then parameterized on the base of expert judgement. This approach would allow, on the one hand, testing the ability of the modelling and calculation tools, and on the other hand to have an insight on the response of the repository system under different







constrains.

4.1 Methodology

4.1.1 Definition and types of scenarios

In a certain sense, the use of the term scenario by Enresa encloses some ambiguity. The broad meaning is a set of data which provides boundary conditions for the definition of the repository system initial condition and for its evolution along the time. It should be noticed that this definition does not imply uniqueness at any point in time for a given scenario; indeed, the system is characterized by uncertainties within each scenario,

Typically, in probabilistic performance assessment pdf's are defined for uncertain parameters, and alternative conceptual models are used to represent not fully understood processes which control the evolution of the different components of the system. In the calculation for each scenario, the parameter values are sampled many times, and the alternative models can be sampled too, but in Enresa's PA exercises only parameters are sampled for a given scenario. Alternative conceptual model are analysed through variants of the Reference Scenario.

In deterministic performance assessment the set of data defines a unique starting point and a unique evolution along the time.

The former definition of scenario does not take in account the probability. This is nevertheless very important for the assessment of the acceptability of the repository system. In Enresa's PAs the probability of the scenarios is not estimated qualitatively. There is only a broad qualitative classification as plausible or not plausible, and then the consequences of the former class are quantified and compared to the regulatory acceptance criteria; we will refer to this category, when there may be ambiguity, as assessment scenarios. When the attributes of a would be scenario are unexpected, or they do not match the actual system, the regulatory criteria are not applicable; this category of scenarios is used for other purposes that verifying compliance with safety criteria, as for example: analysing the system robustness or improving the understanding of the role of a given feature or hypothesis in the system response; they are usually referred to with the term "what-if". A case in point in Enresa's PAs, which belong to the later category, are the calculations made for cases arising when the elements which characterize an assessment scenario are systematically changed, once at a time (either parameter values or pdf's, hypothesis, conceptual models, events, etc) the scenario; the scenarios defined in this way are usually referred to as "variants", and as it has been explained, they are not required to comply with the acceptance criteria.

The assessment scenarios are defined by sets of processes, events and features (FEP's), and an accompanying set of numerical values which quantify them. The first set of FEP's is formed by selecting from a comprehensive list of FEP's, those which are considered to have a high probability. Nevertheless typically a decision is made to simplify the initial scenario and some FEP's with a high probability are excluded or otherwise simplified, and then the deleted FEP is considered in a later scenario. A typical example of the later is the climatic change, which is considered in a separated scenario.

That initial scenario is called reference scenario. Other scenarios, formed with different combinations of FEPs, are called altered evolution scenarios. In fact, the scenarios defined in this way are scenario classes which are further unfolded in one or more scenarios through a





process which progressively specify the characteristics more in detail, considering secondary alternative branches, down to a point where calculation cases may be specified unambiguously; along this procedure are identified i) different alternative models that represent the system, in order to address the conceptual uncertainty, and ii) relevant different sets of parameter values; in this case, the reason is not to tackle parameter uncertainty, which is dealt with through the probabilistic approach, but to cover different features of alternative repository systems (in particular different potential sites, since no site has been selected yet).

4.1.2 Scenario selection methodology

The scenario analysis has as general objective to identify plausible future evolutions of both the Near Field and the Far Field of the repository system. In Enresa's approach the equivalent analysis for the biosphere is a distinct activity, which will be described in a different report of the WP1.1 of PAMINA. In the following the methodology implemented in Enresa 2000 is described [1]. Enresa 2003 was built on the same foundations, but because of time and resources constraints the methodology was simplified.

The approach was based on the Sandia Methodology [3] and on the further developments made in the joint SKI/SKB scenario development project [4]

The main phases of the methodology are:

- 1 Identification of FEPs.
- 2 Classification of FEPs
- 3 Screening of FEPs
- 4 Grouping of external FEPs in classes with similar consequences
- 5 Formation of scenario classes: reference scenario and altered evolution scenarios (see above).
- 6 Development of scenario classes in specific scenarios or calculation cases.

1 Identification of FEPs

The main tool used in the three first steps above are the FEP lists, which eventually are consolidated in a Project FEP Data Base.

Initial lists are compiled by the different teams taking part in the PA exercise, including performance assessors, and scientists of different disciplines (geologists, geochemists, hydrogeologists, materials science experts, etc.) involved in the PA project. They use both their own experience and also relevant references. Among the later, are the FEP lists made in other programmes and by international organisations [5], [6], [7], [8], [9], [10], [11], [12]. These initial lists are based on the responses of the expert teams to a questionnaire prepared by the performance assessment team. Experts are asked to provide:

- a complete list of FEP of potential relevance for geological disposal, without exclusions.
- the classification as feature, event or process



- the definition or description of each FEP in the list
- causes and effects. Potential significance.
- degree of understanding
- comments and references.

Before drafting their individual initial FEP lists, the experts are called by groups to topical workshops were the FEPs of the different subsystems are identified using the RES methodology [13]. In this application the main features or component of a subsystem are represented in the principal diagonal of a square matrix, and later the interactions between any couple of elements of the principal diagonal are entered in the related element of the matrix. For example: the influence of element "m,m" on the element "nn" is represented in the element "m,n" of the matrix. Within Enresa 2000 projects these workshops have been hold for the "source term", the near field, the far field and the biosphere.

In parallel with the previous lists, the performance assessment team compiled their own lists on the base of the FEP lists of SITE 94 [7]. and Kristallin [6].

In the next step all the former lists were consolidated in a single list, after a review of several teams which analysed redundancy and nesting of the entries in the different lists. A further activity consisted in the mapping of the consolidated FEP list with the International FEP list of NEA [5] (the available draft at the time was actually used). The FEP list of the project was then frozen.

2 Classification of FEPs

In order to make a classification of FEPs useful for the later steps in the methodology, the concept of Reference System was adopted, in line with the SITE 94 methodology [7]. The reference system encompasses all the FEPs with a high likelihood.

The fundamental FEP classification was i) belonging to the Reference System and ii) external to the reference system.Several other schemes of classification were used (in particular one based on the physical domain; Near Field, Far Field; Biosphere, External to the System).

3 Screening of FEPs

During the first step the more obvious screening criteria were applied after the compilation of the initial lists: i) FEPs not related to the system under analysis, ii) redundancy, iii) nesting of more detailed FEPs in a more general FEP.

In the third step new screening criteria are applied: iv) outside the scope of the project, v) screened out because required data is not available and vi) very unlikely. The screening by likelihood was qualitatively judged on the base of siting criteria, analogy with similar projects in other national programmes and general scientific knowledge.

4 Grouping of external FEPs in classes with similar consequences

The external FEPs which can have similar effects on the repository system (for example: reduction of radionuclides travel time in the geosphere) are grouped in families. This procedure is facilitated by diagrams where the qualitative effect of each external FEP on the





performance of the Near Field and on the Far Field is plotted.

5 Formation of scenario classes: reference scenario and altered evolution scenarios

All the FEPs belonging to the Reference System form the Reference Scenario. On the other hand, each of the groups of external FEPs formed in the previous step is represented by the envelope of the individual effects on the repository system of the FEPs belonging to that group ("hyperfep"). The combination of a hyperfep with the reference system gives rise to an altered evolution scenario class.

6 Development of scenario classes in specific scenarios or calculation cases.

This last step is actually performed at a later stage of the PA , once the system models are developed and as a previous step to consequence calculation. An analysis of the data available is done to identify the alternative conceptual models which are plausible for the same processes, and sets of parameters characteristic of alternative potential repository systems

4.1.3 Scenarios identified for evaluation

The methodology previously explained led to the identification of the following scenarios in the Safety Assessment of a repository in granite ENRESA 2000 [1]:

- Reference (normal evolution) scenario: represents the expected evolution of the disposal system and today climatic conditions. Radionuclides are discharged to a stream which water is used by the critical individual.
- Climatic scenario: identical to the reference scenario, but with much different climatic conditions foreseen for the future (colder and drier weather).
- Geodynamic scenario: hydraulic conductivities of the main fractures increase a factor 10.
- Human intrusion: an exploratory drilling intersects a canister and a fraction of the waste in the canister is homogeneously distributed along the column of the drilling. The remaining canisters are not affected. Doses are calculated for the same critical individual of the reference scenario, NOT to the workers that perform the drilling. The purpose of the scenario is to analyse the degradation in the repository performance produced by such intrusion.
- Shallow well scenario: the receptor uses water from a well drilled in the upper layer of the altered/fractured outcropping granite.
- Deep well scenario: the receptor uses water from a deep well, but only for drinking.
- Poor backfill/sealing scenario: a preferential pathway for water movement is created along repository disposal drifts, access galleries and shafts, due to a great increase in hydraulic conductivities of the buffer/backfill and the seals. Receptor uses water from a shallow well.

The methodology previously explained led to the identification of the following scenarios in





the Safety Assessment of a repository in clay ENRESA 2003 [2]:

- Reference (normal evolution) scenario: represents the expected evolution of the disposal system and today climatic condition. The receptor (critical individual) uses water from a well drilled in the aquifer above the clay formation.
- Climatic scenario: identical to the reference scenario, but with much different climatic conditions foreseen for the future (colder and drier weather).
- Deep well scenario: identical to the reference scenario, but the receptor uses water from a well drilled in the aquifer below the clay formation.
- Poor backfill/sealing scenario: a potentially preferential pathway for water movement is created along repository disposal drifts, access galleries and shafts, due to an increase in hydraulic conductivity. In order to maximize consequences, groundwater flow through the clay formation is ascendant (while in the reference scenario it was descendant).

4.3 Related topics

The scenario identification described in this document is focused on the identification plausible future evolutions of both the Near Field and the Far Field of the repository system. In Enresa's approach the equivalent analysis for the Biosphere is a distinct activity, and will be described in a different report of the WP1.1 of PAMINA.

4.4 Databases and tools

NEA FEP database v2.1 is a useful starting point for any organisation that intends to make its own FEP list from scratch. Regrettably, NEA FEP database is not based on the most recent Safety Assessment exercises and as a consequence it is somehow outdated.

Enresa has developed its own FEP databases for repositories in granite and clay using NEA FEP database as starting point. FEPs from other Safety Assessment exercises not included in the NEA database were included in the databases too. In addition, Enresa R&D groups were requested to identify the FEP's relevant in their fields of knowledge, and provide information on those FEP's.

4.4 Application and experience

In section 4.1 the methodology for scenario identification followed by Enresa in the most recent Safety Assessment exercises performed has been presented. After these exercises, Enresa has been the coordinator in EC project BENIPA (Bentonite Barriers in Integrated Performance Assessment) [14], that was carried out between September 2000 and August 2003.

One of the tasks in WP2 "FEP analysis" was the generation of lists of FEPs relevant for the bentonite barrier in repositories in granite and clay. Enresa was the WP leader for the case of a repository in clay, Within WP2, two structured FEP lists for the bentonite (one for granite and other for clay) were produced following a "top-down" approach.

The FEP lists have three levels of detail following a logic tree with 3 levels:





Level 1: The following five generic FEP groups are used:

- Barrier properties.
- Boundary conditions.
- Barrier evolution.
- Radionuclide transport.
- External FEPs.

Level 2: A second level develops the first level without reaching the level of individual FEPs.

Level 3: FEPs from level 2 are further broken down into several more detailed FEPs (that correspond to the usual concept of FEPs).

In BENIPA project it was found useful to include some structure in the FEP list (level 1 and level 2) where the individual FEPs (level 3) can be fit. This structure makes easier to understand the consequences of each FEP and to identify if a given FEPs is missing or already included in the list.

The table of bentonite FEP's for a repository in clay obtained in BENIPA project [14] is presented here:





Level 1	Level 2	Level 3		
		COMPOSITION		
		P1.1	Clay minerals	
		P1.2	Accessory minerals	
	P1	P1.3	Water content	
BARRIER PROPERTIES		P1.4	Organics	
		P1.5	Additives	
		P1.6	Modifications during elaboration	
	P2	PROPERTIES		
		P2.1	Thermal properties	
		P2.2	Mechanical properties	
		P2.3	Hydraulic properties	
		P2.4	Chemical properties	
		P2.5	Gas transport properties	
		P2.6	Solute transport properties	
	P3	EMPLACEMENT		
		B1.1	Swelling of the corrosion products	
		B1.2	Breach of the canister	
		B1.3	Degradation of the engineering confining elements	
		B1.4	Deformation of rock cavity	
		B1.5	Drift wall discontinuities	
		B1.6	Host rock stress	
	В2	REPOSITORY MATERIALS		
		B2.1	Rock material	
		B2.2	Canister material	
		B2.3	Waste form	
		B2.4	Rock supporting materials	
		B2.5	Stray materials	
		B2.6	Other materials of the design	
CONDITIONS	В3	RADION	IUCLIDE SOURCE TERM	
		B3.1	Radionuclide inventory	
		B3.2	Radiation field	
	B4	HEAT INPUT		
		B4.1	Radionuclide decay heat	
		B4.2	Geothermal gradient	
	B5	GAS GENERATION		
	B6	WATER	WATER FLOW AT THE BENTONITE/ROCK/LINING INTERFACE	
	В7	GROUNDWATER CHEMISTRY		
		B7.1	Natural groundwater chemistry	
		B7.2	Modified groundwater chemistry due to reactions with repository materials	
	B8	DESIGN	AND OPERATION OF THE REPOSITORY	

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Level 1	Level 2	Level 3		
	E1	THERM	AL EVOLUTION	
		CHEMICAL EVOLUTION IN THE PORE-WATER		
	E2	E2.1	Speciation	
		E2.2	Radiolysis of water	
		E2.3	Electrochemical gradients	
		E2.4	Chemical interactions between groundwater and solid	
	E3	MECHANICAL EVOLUTION		
	E4	GAS TRANSPORT		
	E5	HYDRAULIC EVOLUTION		
EVOLUTION OF THE BARRIER		E5.1	Bentonite resaturation	
		E5.2	Water flow through bentonite	
	E6	BENTONITE DILUTION		
		E6.1	Loss of bentonite material through outer boundary	
		E6.2	Loss of bentonite material through inner boundary	
	E7	CRITICALITY		
	E8	MICROE	MICROBIAL ACTIVITY	
	E9	LONG-TERM STABILITY OF THE BENTONITE		
		E9.1	Mineralogical alteration	
		E9.2	Bentonite cementation	
		E9.3	Radiation effects	
	Т1	RADIONUCLIDE TRANSPORT		
		T1.1	Transport by diffusion and advection	
		T1.2	Radioactive decay	
		T1.3	Solubility, precipitation and co-precipitation	
RADIONUCLIDE TRANSPORT		T1.4	Radionuclide sorption	
		T1.5	Effect of multiple releases	
		T1.6	Coupled transport phenomena	
		T1.7	Anionic exclusion	
		T1.8	Surface diffusion	
		T1.9	Colloids mediated transport	
		T1.10	Complexes mediated transport	
EXTERNAL FEP's	EX1	HUMAN ACTIONS. BOREHOLE DRILLING		
	EX2	THERMAL CHANGES		
		EX2.1	Volcanism	
		EX2.2	Geothermal alterations	
	EX3	HYDRAULIC CHANGES		
	EX4	MECHA	MECHANICAL CHANGES	
	EX5	CHEMICAL CHANGES		
		EX5.1	Changes in the salinity of the groundwater	
		EX5.1	Changes in the Eh of the water	

[PAMINA]





4.5 On going work and future evolution

In all the performance assessments performed by Enresa, scenarios have been identified following a systematic approach based on a catalogue of Features, Events and Processes (FEPs).

Enresa does not intend to make a new Safety Case exercise of a deep geological repository in the near future. Enresa follows the international developments in this field (scenario development) and other fields related to the Safety Case, and can take part in EC R&D projects, but no indigenous work is being done on this topic.

5 Lessons learned

Making a comprehensive FEP list for a HLW repository is a time consuming task. Since FEP lists are prepared at the beginning of a PA project, if it takes much longer than expected there exists a clear risk of reducing the time available for the following phases of the PA. The effort and time spent in the FEP list must be controlled because they can easily last much longer than scheduled.

When making the FEP list we found some difficulties to communicate with experts, due to the different approaches. It is important to be sure that experts have understood what the PA team expects of them.

If the number of entries in the FEP list becomes too great, handling the list can become difficult. Including some structure in the FEP list has been found useful. For example, checking the completeness of the list is much easier if the FEPs are already classified into groups than if an unstructured list of several hundred FEPs is used.

Creating your own FEP lists adding several FEP lists already existing is not a straightforward process due to differences in criteria followed to produce them, differences in the level of detail, overlapping, redundancy and nesting of FEPs. Structured lists as the one developed in BENIPA [14] can be useful to tackle these problems.

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Part 2: Definition and Assessment of Scenarios



Appendix A4: GRS-K (Germany)

A4 GRS-K (Germany)



Part 2: Definition and Assessment of Scenarios

Appendix A4: GRS-K (Germany)





Proposal/Contract no.: FP6-036404

Project acronym: PAMINA

Project title: PERFORMANCE ASSESSMENT METHODOLOGIES IN APPLICATION TO GUIDE THE DEVELOPMENT OF THE SAFETY CASE

- Instrument: Integrated Project
- Thematic Priority: Management of Radioactive Waste and Radiation Protection and other activities in the field of Nuclear Technologies and Safety

WP1.1 Topic "Definition and Assessment of Scenarios" GRS Köln contribution to the EWG

Due date of deliverable:	30 September 2009
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Thomas Beuth, GRS

Revision: 2

Project co-funded by the European Commission within the Sixth Framework Programme (2002-2006) Dissemination level

- PU
 Public

 PP
 Restricted to other programme participants (including the Commission Services)
 X

 RE
 Restricted to a group specified by the consortium (including the Commission Services)
 X

 CO
 Confidential only for members of the consortium (including the Commission
- CO Confidential, only for members of the consortium (including the Commission Services)

[PAMINA] (D-N°: 1.1.1) – Task reports for the first group of topics Dissemination level: RE Date of issue of this report: 15/03/2008





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- 2 Definition of terms and used concepts
- 3 Regulatory context
- 3.1 Regulations and guidance
- 3.2 Requirements and expectations
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- 3.4 Development and trends
- 4 Analysis and synthesis
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1 Background/ Introduction

Safety assessments for radioactive waste repositories in deep geological formations are an integral part of the comprehensive demonstration of the safety of the repository in the postclosure phase. The demonstration will be conducted on a site specific basis in consideration of the geological, geochemical, and geotechnical state of the repository system, and its long-term predictions as well.

The safety assessment includes the scenario development, consequence analysis with uncertainty and sensitivity analyses, and the demonstration of the compliance of prescribed protection objectives. According to the scenario development the potential evolution of the repository system has to be investigated. A variety of potential changes in system behaviour has to be taken into account due to the long time frames. Derived scenarios from the scenario development constitute the fundamental basis for the further work like the consequence analysis. Furthermore, scenarios essentially determine the subsequent phases of the repository evolution e.g. planning, design, operation, and post-closure. Finally, the scenarios provide an important basis for the dialog between the different involved parties in radioactive waste disposal. And therefore contributes decisively in the process of confidence building.

As outlined in the Annex I "Description of Work" of the Integrated Project PAMINA the tasks in WP 1.1 will be carried out by bringing together and by including the perspectives from both the "developers" and the "evaluators". For this reason each task will be addressed by the "development working group" (DWG) and by the "evaluation working group" (EWG) whereas the latter group will be the working platform for GRS Köln.

Therefore the present draft document includes the background, fundamentals, and the regulatory basis as well as recent developments in revising the existing Safety Criteria from 1983 concerning the topic "Definition and Assessment of Scenarios".

1 Definition of terms and used concepts

The defined terms and used concepts in the frame of scenario development are as follows /BAL 07/:

Scenario development

The scenario development represents an identification and selection of relevant alter-native developments (**scenarios**) of the **repository system** for further treatment in safety analyses.

<u>Scenario</u>

Scenario describes a postulated evolution of a **repository system** and its **safety functions**, specified by a combination of **relevant factors** that characterise or influence the **repository**





system.

Repository system

The repository system comprises the **repository** and its geological environment, which in turn includes all rock areas that have to be considered for the compliance proof of the safety principles and protection objectives for final disposal.

Repository

The repository is part of the repository system in which high active waste will be placed. It comprises the repository mine, the host rock and the **isolating rock zone**.

Isolating rock zone

The isolating rock zone is part of the geological barrier which at normal development of the repository and together with geotechnical barriers (shaft seal) have to ensure the confinement of the waste.

Safety Function

Safety function is a function, which takes over safety relevant requirements, in a safety related system, subsystem or single component. Through interaction of such functions the containment (isolation) as the primary safety function of the **repository system** is guaranteed as well as the compliance with safety principles and protection objectives both in the operational phase and post closure phase of the repository.

Relevant factors

Relevant factors comprise site and system specific features, events and processes (FEP's) which have or might have an influence on the **repository system**.

3 Regulatory context

In Germany all types of radioactive waste have to be disposed of in a deep repository. It is the policy of Germany that radioactive material should be concentrated and contained rather than released and dispersed in the environment. According to the international consensus that long-lived radioactive waste has to be disposed of in deep geological formations in order to guarantee that man and the environment are protected in the long run from the effects of ionizing radiation by isolation of the radioactive waste. In Germany all types of radioactive





waste have to be disposed of in a deep repository.

Presently, the management of radioactive waste in Germany is under review. Amongst the important cornerstones of the new waste management plan is a revision of the "Safety Criteria for the Disposal of Radioactive Waste in a Mine" /BMI 83/ (in the following named as "Safety Criteria") which were issued in 1983 /BAL 06/.

As indicated, the German "Safety Criteria" are at present revised on behalf of the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU) in order to account for the progress in safety-related developments and procedures, e.g stepwise approach, constrained optimisation, and "Safety Case" methodology. The revision of the "Safety Criteria" as well as the development of supporting guidelines is carried out by the Final Disposal Department of GRS Köln with the support of a number of experts from Germany and abroad. The revision accounts for the ideas and requirements given in the OECD/ NEA report "Post-closure Safety Case for Geological Repositories" /NEA 04/ and in the IAEA safety requirements guide WS-R-4 (formerly known as DS-154) /IAE 06/.

In the following sections the regulatory framework and the ongoing work concerning scenario development will be shown. Specific topics which strongly relates to scenario development like "Human Intrusion" and "Safety Function" are described regarding their context but will be addressed in detail separately.

3.1 Regulations and guidance

The legal basis for licensing is the "Plan Approval Procedure" required by the German "Atomic Energy Act" for federal installations for the safekeeping and final disposal of radioactive waste. The "Plan Approval Procedure" has a so-called "concentrating effect" for several fields of law and will generally lasts for the whole duration of a project. A stepwise approach is not explicitly implemented. Nevertheless it is the opinion of GRS that such an approach could be applied within a "Plan Approval Procedure" if the stakeholders would commit themselves on a voluntary basis. Within such an approach, a safety report based on the knowledge achieved so far would be produced at well-defined decision points, communicated to regulators and other stakeholders, and utilised to support decisions about how to proceed ("Safety Case").

In application of the "Plan Approval Procedure" the formulated "Safety Criteria" /BMI 83/ have to be considered. The "Safety Criteria" from 1983 addressed the subject "Definition and Assessment of Scenarios" insofar as disruptive scenarios are part of the required safety analyses. According to the "Safety Criteria" potential disruptive scenarios have to be justified in detail and fixed in their constraints. Such disruptive scenarios have to be taken into account in safety analyses in consideration of scientific methods. Safety analyses are required in terms of the operational phase, decommissioning phase and post closure phase. There are no further requirements in the "Safety Criteria" regarding scenario development.

Recent results of the revision work of GRS Köln were documented in a draft report "Safety requirements for the disposal of high active wastes in a deep geological formation" /BAL 07/ (in the following named as "Safety Requirements") and discussed on a workshop held on 6 and 7 March 2007 in Hannover, Germany. The proposal for the criteria revision is however still being reviewed by advisory bodies and might therefore undergo further changes. A final draft for the proposal of the revised "Safety Criteria" is not available so far. Earlier stages of the development are reflected in several published documents /BAL 04a, BAL 04b, BAL 05a,





BAL05b, EUS 06/.

It should also be noted that site selection is not and siting is not in detail addressed in the proposal for the criteria revision. The revision is based on the understanding that a site has be chosen in accordance to the requirements of a siting procedure as outlined e.g. in the AkEnd recommendations on site selection /AKE 02/.

The statements presented in the following sections relate to a large extent to the above mentioned "Safety Requirements" /BAL 07/. As indicated before this proposal has a draft status and should therefore be seen as a preliminary work with no binding regulatory basis. However, the document includes the recent developments in the field of regulatory requirements on the basis of broad and thoroughly performed discussions and exchange of information and experience with experts from Germany and abroad.

3.2 Requirements and expectations

Scenario development as a component of long-term safety analysis

The long-term safety analysis has to comprise, the scenario development and the consequence analysis for the proof of compliance of protection objectives. The consequence analysis must underlie scenarios obtained from the scenario development. Strategy and methodology of the analyses have to be shown.

Scenario development as a requirement

It is to carry out a scenario development for the repository system. Here the potential evolutions of the repository system according to scientific findings, which are caused by endogenous and exogenous processes, have to be considered. Furthermore, the relevant scenarios for the safety case, with the exception of human intrusion, have to be identified.

Requirements for scenario development

The scenario development has to be documented in a transparent and comprehensible manner. Each individual step has to be justified, and relevant decisions have to be explained clearly.

Human activities in knowledge of the closed repository are not considered. These are left to the acting society's own responsibility.

Concerning the assessment of long-term safety, the scenarios have to be assigned to the following scenario classes, and this classification has to be justified:

• Likely scenarios: Scenarios which are highly probable to occur during the demonstration period of one million years.





- Less likely scenarios: Scenarios which compared to the likely scenarios are much less probable to occur during the demonstration period of one million years.
- Scenarios that need not be considered any further: Scenarios with a very low occurrence probability or with primary effects that exceeds the secondary consequences of the repository by far, e.g. impact of meteorites.

Option for grouping of scenarios

Scenarios with similar developments taking place may be summarised to scenario groups and shown by a representative scenario. Prerequisite for it is that the effects from the representative scenario on the safety functions of the repository system cover the effects of the group. Likely scenarios and less likely scenarios may not be summarised in a group.

Consideration of representative scenarios in consequences analysis

The determination of consequences from the emplacement of high active wastes, i.e. the potential release and migration of harmful substances in the repository system, must be performed for all representative scenarios.

Method, procedure and approach for scenario development

In fact, there are no requirements regarding the choice or use of a certain method, procedure and approach for the development of scenarios. It is left to the implementer to decide which tools, programmes or instruments are useful or not for the task of scenario development. However, the implementer has to demonstrate, that all above mentioned requirements were taken into account and fulfilled. The demonstration has to be done in a transparent, reasonable, consistently and comprehensible manner.

3.3 Experience and lessons learned

Subjective influence

The scenario development is largely based on expert judgement. Expert judgement is borne by experience, knowledge, expertise, opinions etc. of individuals or groups in the respective technical fields. However, the derivation of scenarios is determined to some extent by subjective influences. Resulting from the evaluation of the relevance, coherence, occurrence, or likelihood of potential factors that might have influences on the repository system. A striven aim will be to work out a sound approach and acknowledged procedure which reduces subjective influences due to inevitable expert judgement or other sources as far as possible.

3.4 Development and trends

Human Intrusion

The systematic investigation of potentials, procedures and effects of human intrusion into a





deep repository requires the prediction of the environment of societies, social structures and the state of the art of future generations. A broad consensus exists, that the development of human societies and human behaviour are not predictable. Due to this fact the issue of human intrusion into a deep repository will be treated separately and outside of a systematic scenario development. A detailed discussion of this issue will be a subject of the contribution of GRS Köln in the frame of WP 1.1 GRS Köln to the topic "Human Intrusion" of the PAMINA project.

Safety Functions

In the opinion of GRS, the use of "Safety Functions" in terms of scenario development constitutes a promising approach on two counts. Firstly, the definition of safety functions and the assignment to repository components or subsystems helps to subdivide the repository system into more manageable areas. The focus of the investigation regarding influencing factors shifts from a complex system to individual safety functions. Relations between safety functions, also in consideration of their temporal effectiveness, can be better investigated and identified. In conclusion, the entire system will be more understandable and explainable. Secondly, it is expected that due to the investigation of specific safety functions the number of potentially influencing factors that have to be discussed and finally taken into account in safety analyses will decrease.

In recent years the consideration of "Safety Functions" in safety assessments plays a more and more important role. Examples for that can be found in safety reports from different countries, e.g. "SAFIR2" (Belgium), SR-Can" (Sweden), "Dossier 2005 Argile" (France) and "H12" (Japan). Currently some countries like Belgium intend to identify altered evolution scenarios on the basis of safety functions.

Since September 2007 GRS is involved in a R&D-Project called "Comparative Safety Analyses for Repository Sites for the Assessment of Methods and Instruments" (VerSi) under the auspices of the BMU. This project consists of four subprojects which cover conceptual work, scenario development, long-term analysis and evaluation. The overall objective of the project is the provision of appropriate methods and tools for the comparison of repository concepts in different host rocks e.g. clay and salt.

In the framework of the subproject scenario development the derivation of scenarios in consideration of safety functions is one of the main tasks. The proposed procedure is described in more detail in the contribution of GRS Köln to the topic "Safety Function" in the frame of WP1.1 of the PAMINA project.

4 Analysis and synthesis

This section describes the frame of scenario development in Germany from a regulatory perspective, which is still in discussion, as a basic component for safety assessments /BAL 01, BAL 07/:

Safety objective and ethical principle

The fundamental radiological safety objective of the permanent disposal of radioactive wastes is the protection of man and the environment from ionising radiation. The protection of future generations is achieved through measures that isolate the radioactive wastes within


deep geological formations and does not depend on active measures in the future. The basic principle is that the same protective objective should apply to future generations as for current generations.

Optimisation process

For developed scenarios, except inadvertent human intrusion, adequate protection of humans and the environment is achieved through a optimisation process in connection with a dose constraint of 0.3 mSv/a. The process of optimisation means that all meaningful measures are laid hold of for the reduction of the individual dose estimated for the future during an iterative procedure for site selection, planning, development of the construction and operation of the repository.

The process of optimisation is fulfilled when the repository is completed with state-of-the-art in science and technology, technical and managerial principles are realised, the 0.3 mSv/a constraint is adhered and actions on meaningful measures against inadvertent human intrusion are taken.

Basis for scenario development

The scenario development represents an identification and selection of relevant alter-native developments of the disposal system for further treatment in safety analyses. The scenario development thus requires adequate knowledge of the disposal system which makes possible a description and characterisation of the entire system, its behaviour and evolution up to now and also in the future. Basis for this work is a comprehensive identification of the relevant site- and system specific factors influencing the system (features, events and processes, FEPs). The understanding of the system, for example of the geological and geotechnical situation, must be such that a prediction of the potential evolution of the system can be given with reasonable certainty. For this purpose exploration of the site and accompanying laboratory and in-situ studies have to be carried out. Appreciable attention in the investigations of the site is given to the interpretation of the history of the geological evolution of the site itself. This should provide a basis for predictions within a timeframe which is relatively short in terms of the geologically interpretable history of site-evolution. Further observations of nature e.g. natural analogues are essential to understand the system and its evolution.

Potential evolutions of the disposal site

The possible evolutions of the disposal site originate on the one hand with natural i.e. endogenous and exogenous processes involving the entire system and on the other hand in the evolutions induced by human activities.

Natural processes are disposal system evolutions which are of natural origin. These comprise normal as well as disturbed evolutions in the disposal system; they include hypothetical initiating events and occurrences which involve bypassing or damaging of barriers. Basis for these considerations is the status of the subsystem, the components and barriers, as well as of the disposal system at the beginning of the post-operational phase. Thus, for example, the state of the engineered barriers at the beginning of the post-operational phase represents the starting situation and description of conditions for the scenario development. This includes consideration of uncertainties in the design and construction of engineered barriers and in the same way consideration of human failings in





the manufacture, installation, and quality assurance of technical components.

By human activities is meant all those activities which intentionally or inadvertently alter the effectiveness of the barriers of the disposal system. These on the one hand are activities which have an influence on the effectiveness of the barriers or the site situation as, for example, the building of a dam that brings about a change in the ground-water flow regime, and on the other hand such activities that bypass the barriers and constitute a short-circuit between the repository and the biosphere. Examples of such direct intrusions are borehole drillings or mining activities. The latter activities are called "Human Intrusion" and will be handled separately, i.e. outside from the systematic scenario development.

Furthermore, only those human activities are studied in safety analyses which inadvertently affect the isolating property of the disposal system. These activities are such that knowledge about the existence and whereabouts of the repository is lost to the memory of the living, or the potential danger from the activities presumably cannot be known. For the intentional intrusion into the repository or an intentional risk-taking in regard to influencing the whole disposal system, the intruders themselves should accept responsibility. These scenarios are therefore not considered further in the safety analyses.

General steps for scenario development

The following general steps are distinguished for the scenario development:

Firstly, relevant factors essential to characterising the behaviour of the system under consideration (mostly: "FEPs") are gathered together. For this reason, generic data bases (e.g. the OECD/NEA FEP database) as well as site-specific information can be reverted to. The process of selecting phenomena regarded to be relevant for the analysis is partly based on subjective decisions. This holds as well when the decision process is stringently formalised or even automated because in such cases the (possibly subjective) decision is made by the definition of the selection criterion. Were the selection of e.g. the probabilities of occurrence of the phenomena drawn upon, then the question by which procedure this likelihood of occurrence was determined is brought up.

Secondly, the phenomena are then combined to potential evolutions (scenarios). There exist several possible methodologies for combining the phenomena (FEPs) to scenarios whereby none is distinguished through having advantages in comparison with the others. A new approach concerning the derivation of scenarios in consideration of "Safety Functions" is under development.

The development of scenarios depends on the purpose of the analysis to be carried out. So, for example, processes that describe natural site evolutions can be especially important for the site selection, while processes pertaining to the disposal system can be drawn upon for the safety analysis. The developed scenarios as a basis for safety analyses are divided as described above (cf. 3.2) in the following classes according to the likelihood of occurrence: Likely scenarios, less likely scenarios and scenarios that need not be considered any further. For the latter, decisive reasons for this have to be stated e.g. very low likelihood of occurrence.

Finally, the remaining scenarios are grouped and differentiated with respect to further procedures regarding their place in an analysis and to the purpose of the analysis itself. The remarks concerning the subjectivity of such a decision-making process are valid here as well.





The process of scenario development must be transparent, i.e. it must be reproducibly documented for the licensing procedure. Hence, the individual steps must be well founded and the decision made by the experts traceable presented.

Assessment and handling of scenario classes

Likely scenarios:

The compliance of protection objectives is guaranteed by the isolation capacity of the repository system. For the assessment of the isolation capacity of the repository system the consequences of the likely scenarios will be analysed. The assessment orientates itself as far as possible at the thought that the isolation is guaranteed, if the existing natural system is disturbed as little as possible. Thereby also the protection of the environment is fulfilled beside the protection of the mankind.

Less likely scenarios:

In the consequences analysis for less likely scenarios the consequences (as a result of migrated radionuclides) will be determined in the respective subsystems. As assessment factor the conditions and consequences will be included, which can be determined due to natural from the repository unaffected circumstances. Requirements are regarded as fulfilled, if the determined consequences due to released radionuclides from the repository are not greater than those which results from natural of the repository unaffected circumstances.

Scenario uncertainty:

Both for the likely and for the less likely scenarios the consequences will be determined in consideration of data uncertainties. Under the use of stochastic methods, the calculated 95-percentile of the indicator, on the basis of a 95 % confidence interval, for the assessment of the results has to be considered.

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A5 IRSN (France)





Definition of scenarios

1 Background/ Introduction

The principle of passive disposal system conducts to the necessity to demonstrate that man and the environment are adequately protected without human control or intervention. Consequently, the assessment strategies must focus on potential radionuclide releases from the repository to the biosphere and evaluate their consequences by calculating dosimetric impacts for various plausible situations of evolution of the disposal system. Understanding those possible evolutions may be gained by illustrative calculations under different assumptions about key events or properties of the system. By a stepwise process, the scenario development aims at choosing a limited number of different scenarios that, taken together, illustrate the behaviour of the system and its safety and improve the understanding of mechanism of the system by testing the reactions of the system under certain stresses. In other words, a relevant strategy of scenarios should allow defining all the situations to be considered and should allow classifying them by their occurrence in order to structure the performance assessment and the safety case by identifying the need for further work to avoid, mitigate or reduce uncertainties and to evaluate their effect.

Among numerical modelling activities performed, IRSN studies focus on the understanding of transient processes as chemical and thermal interactions, dehydration/rehydration occurring during drilling and after closure of the repository, and long term behaviour of the EDZ in indurate clay. New and high-performance numerical methods are also under implementation in the MELODIE software, currently used in the framework of Euratom exercises (EVEREST, SPA, BENIPA and on-going NF-PRO, PAMINA and MICADO exercises), to improve resolution of coupled flow and transport equations for highly heterogeneous systems.

The assessment approach describes in the present topic is derived from the BSR III.2.f issued in 1991. In addition the notion of safety functions, which are included in the release of the BSR currently being discussed by ASN, ANDRA and IRSN, are also used, because of their involvement in the strategy of the scenario development.

2 Definition of terms and used concepts

The safety assessment through the scenario development is built to prove the favourable behaviour of the repository considering the possible consequences of disturbances. Typically, a normal evolution scenario (NES) is first developed considering the expected performance of the components of the repository. Then, altered evolution scenarios (AES) are built in order to assess the role of components assumed, either to be containment barriers or to fulfil specific safety functions, and to quantify the influence in case of failure of those components. Moreover, sensitive calculations are performed so that the influences of the uncertainties on the performance of a component or the lack of knowledge relating to



physical or chemical mechanisms are assessed.

3 Regulatory context

ASN (Nuclear Safety Authority) develops the regulatory framework for the safety of the deep geological disposal. This framework follows the principles and recommendations provided by the international organisations being technically competent (IAEA, ICRP, OECD). IRSN acts as the technical support organism of ASN and participates to the regulatory framework definition. IRSN performs studies and research to support its technical appraisal for ASN. IRSN is deeply involved in international working groups so that the regulatory and assessment approaches developed in France are consistent with international guidance.

In June 1991, the Basic Safety Rule 3.2.f (BSR3.2.f) was edited by ASN as guidance for defining the situations providing demonstration of safety through evolution scenarios. A new version of this was released in 2007 in order to introduce the notions and the safety approaches developed in the 2005 Clay Dossier edited by ANDRA.

Scenario development is a key topic in the frame of the safety analysis, since it has an important role in capturing uncertainties and quantifying their influence, in verifying fulfilment of safety functions associated with disposal components, and in quantifying the dosimetric impact due to the disposal system.

a) Regulations and guidance

In a current manner, implementer develops its own set of evolution scenarios taking into account the potential evolutions of the disposal system and their related uncertainties in agreement with the BSR3.2.f. However, regulators can recommend including specific situations in the development of the scenarios or integrating technological uncertainties in the normal evolution scenario.

To verify that the objectives of the repository are reached, the post-closure safety assessment must cover the following three complementary sides in an iterative process:

- verification of the favourable behaviour of the performance of the disposal components associated to safety functions when no interactions are expected,
- evaluation of the disturbances caused by the creation of the repository and checking that they remain acceptable in terms of the safety level chosen for each of the safety functions with respect to the preventive and palliative options of design,
- assessment of the future behaviour of the repository and checking that individual exposure is acceptable. The approach adopted shall consist in considering a limited number of situations representative of the different families of events or sequences of events such that the associated consequences are the greatest among those of the situations of the same family. The families of events or sequences of events adopted shall be those considered to be conceivable among all those which are a priori possible.





The events and processes constituting the situations adopted for the purposes of the safety analysis must be modelled and characterized. This characterization shall be essentially iterative insofar, in particular, as the determination of situations considered is liable to be refined on the basis of a better understanding of the barriers and their behaviour.

As concerns the timing of these situations, reference shall be made to the following periods:

- an "initial" period of 500 years in which records of the repository would be kept, making human intrusion in the repository area extremely unlikely. It would also correspond to substantial decay of the activity of the short and intermediate-lived radionuclides,
- an intermediate period of 50,000 years, characterized by the absence of extensive glaciations,
- a subsequent period after 50,000 years in which allowance for extensive glaciations shall in particular be made.

For the reference situation, the events to be considered are:

- events associated with the presence of the repository: the impact of the latter will consist of the initiation of processes associated with the emission of heat, mechanical, physical and chemical changes, as well as desaturation, as a consequence of the excavation or due to gas, of the natural medium around the repository. All the processes of gradual degradation of the artificial barriers (corrosion of the containers and the containment matrixes, aging of the engineered barriers and seals etc...) must be taken into consideration.
- series of highly probable natural events (changes in climate, subsidence and uplifting). The climatic changes (external geodynamics) are accompanied by processes such as erosion/sedimentation cycles, and changes in surface hydrology and ground-water movements.

For the hypothetical situations corresponding to random events, those events allowed for in these situations shall be either events of the same nature as those considered in the reference situation but of exceptional amplitude, or events which are of high uncertainty as to when and how they will take place. Such events are divided into two categories, those of natural origin and those associated with human activity:

• events of natural origins to be taken into consideration shall include at least the following: major climatic changes, seismic activities, subsidence and uplifting of an exceptional nature, diapirism, magmatic activity and meteorite impact. Some of these events may, depending on the site, be dismissed only after justification by analysis.

As regards seismic activity, allowance will be made for a level of seismic activity liable to be encountered during the periods studied. There are uncertainties concerning the seismic levels possible long before the historical period. The existence of a physical limit for the seismic levels in a given region could constitute a limiting value in view of the seismotectonic context.

 events associated with human activity relate to direct and indirect human intrusion (drillings, mines, cavity forming and surface and sub-surface construction), defects in packages (unexpected degradation or failure to comply with specifications), defects in





the engineered barriers (improper sealing resulting from a failure to comply with the specifications for emplacement or fabrication, and design errors), climatic changes associated with human activity (greenhouse effect), defects in the geological barrier resulting in anomalies in it (imperfect knowledge of the site, earlier intrusions etc...).

b) Requirements and expectations

The long-term safety of radioactive waste repository (over periods of time of several thousands years) is based on the "concentration and containment" strategy. Achievement of this strategy relies partly on design options which must contribute to minimize disturbances caused by the repository in order to preserve containment properties of the different components. Among main disturbances are chemical and mechanical interactions between different exogenous materials (cement, metallic components, bentonite), host rock and disposal facilities that may cause damage to the host rock, the different barriers and the canisters. Another important issue for long-term safety is the feasibility of seals and plugs to close the repository in order to limit advective water flux and mitigate possible by-pass of the host rock.

c) Experience and lessons learned

"2001 Clay Dossier" and "2005 Clay Dossier" were provided by ANDRA and reviewed by IRSN concerning the deep geological disposal. In both reports, ANDRA has developed a significant gathering of evolution scenarios of the disposal system devoted to simulate the radionuclide migration through engineered components and geological layers and to evaluate dosimetric impacts

The main remarks arisen from the evaluation of the "2005 Clay Dossier" made by IRSN were about the abandon by ANDRA of the allocation performance approach and the significant progress concerning several topics:

- the setting up of the normal evolution scenario based on phenomenon accounting for • technological and phenomenological uncertainties,
- the choices of the parameters representing the possible evolutions of the components,
- the relevance of the assumptions on the failures conceived within altered evolution scenarios and their influences,
- the variety of sensitivity analysis aiming at highlighting the importance of each components and of the concept design.

Nevertheless, contrary to the importance of the host rock, the safety analysis doesn't clearly highlight the key engineered components and their performance levels expected in relation with the safety of the disposal system. By the way, it will be important to consolidate the assumptions contributing to the design (dimensionnement?) of the disposal engineered components.







d) Developments and trends

The BSR3.2.f issued in 1991 showed relevance to guide the elaboration of ANDRA report on the feasibility of a possible HLW repository in a clay formation. The new release of the BSR3.2.f is evolving in the following notions: implementation of the safety functions, reversibility and definition of a disposal concept considering spent fuel. The scenario development must take into account these new trends having a role on the possible performance of the disposal system.

In a practical point of view, IRSN focus on the assessment of the level of quality to be reached in situ for the various components of the repository. As a matter of fact, the long term performances depend on the initial and real state of the components during operational phase (comprising canisters design and manufacturing) and then different questions are arisen from these thoughts:

- methods, process, quality control to detect defects (e.g. of canisters...)
- what will be the criteria, function indicators upon which (below which) the long term performance of the component should lead to an altered evolution of the repository?
- derivation and classification of evolution scenarios according to the level of confidence in the specified characteristics of the components, tolerance, deviations from specifications...
- How to measure the performance of in situ component?
- Effects of natural heterogeneities and defects due to in situ manufacturing,

The setting up of the notion of timescale in the strategy of scenario development is a topic of interest. In a common manner, concerning the normal evolution scenario, the data, translating the expected performance of the components, are fixed at the beginning of the simulation and are not modified during this simulation. However, the performances of the components are progressively degraded by the disturbances due to the presence of the disposal facility and the environment. The evolution of the disposal system is more difficult to plan and the uncertainties associated to the level of performance increase with time. These decreases of the performance are generally studied through altered evolution scenarios, although those degradations are involved in the normal evolution of the disposal system. The notion of time in the scenario development, and particularly for the normal evolution scenario, seems to be a key point of the strategy to be applied concerning the uncertainties management and, at last, in the evaluation of the radiological impact.

4 Analysis and synthesis

To back up its technical appraisal, IRSN carries out numerical modelling activities aiming, on the one hand, at quantifying physical processes and interactions possibly occurring in an underground repository and, on the second hand, at quantifying containment capabilities of the different components.

Modelling approach aims at providing quantitative outputs regarding:

• The intensity, extension and characteristic timescales of different disturbances due to hydraulical, chemical, mechanical and thermal interactions between wastes,





exogenous materials of disposal components and host rock.

• The expected containment capabilities of the different barriers accounting for disturbances listed above and for various hydrogeological settings governing flow patterns.

The normal evolution scenario must be based on a gathering of reasonably penalizing values selected with respect to confidence degree associated to the knowledge of the phenomenon contributing to the evolution of the components. The second scenario named altered evolution scenario assumes that drifts are not properly sealed. The aim of this scenario is to assess the influence, on radionuclide transfer, of the failure of the narrow bentonite-filled trench supposed to interrupt the damaged zone.

IRSN develops also an "what if" evolution scenario based on uncertainty in geological survey considering the presence of a secondary fault, away from the disposal zones but crosscutting the access drift. This fault could be an advective radionuclide pathway and could reconsider the safety functions associated to the host formation by providing water inside the repository system and by disseminating the radionuclide inventory contained by the nearest disposal tunnels.

The waste packages are assumed to ensure the safety by containing the radioactivity in the repository. The influence of the UOX matrix dissolution is covered by a sensitivity scenario, since this dissolution mechanism is bad-known. The influence of the various mechanisms occurring is assessed by sampling different degradation rates and allows evaluating the impact of those rates regarding the transport mechanisms (sorption, solubility, advection...). These calculations are performed at the scale of the disposal tunnels and at the scale of the host rock identifying which is the impact of this mechanism.

Sensitivity scenarios are also developed considering assumptions on the hydrogeological settings and require the definition of the outlets (artificial "well drilling" zone or at the ground surface) requires the characterisation of groundwater flow regimes based on advective movement of water in the aquifers and possible conductive discontinuities. The results obtained for the different flow simulations have highlighted the fact that the computed hydraulic heads in the different layers could be calibrated by using different groundwater flow patterns (and then defining different outlets) where the identified or suspected structures within the studied area either play a hydraulic role or not. Additional hydrogeological field studies would be necessary in order to reduce the number of flow models that could be considered in the repository area. The influence of remaining groundwater flow schemes on the transfer times and concentrations of the radionuclide plumes at the outlets should be assessed (by the mean of a sensitivity study) to select the flow scheme leading to major release of activity at the outlets.

The simulation of radionuclide transport through the permeable and semi-permeable formations also requires the quantification of the diffusive transfer process. While this phenomenon is recognised as being dominant through the homogeneous Callovo-Oxfordian formation and has given rise to studies aimed at specifying it, the modelling exercise has shown that such phenomenon must also be more accurately characterised in the aquifers in order to discriminate the importance of more widely scattered outlets (in areas which might be potential water resources) away from hydraulic discontinuities. Else, the uncertainties resulting from a lack of knowledge regarding this process must at least be taken into account by means of a sensitivity study.





From a practical point of view, numerical calculations are split into two kinds of calculations associated respectively to "process level" modelling and "integrated level" modelling.

Process level modelling is mainly performed at the vault scale and aims at understanding and quantifying (extension, intensity and duration) processes playing a role in the evolution of the containment properties of the different components of the repository (waste packages, containers, engineered barriers, plugs and seals) and of the "near field" (part of the host rock submitted to interactions). In complement, such modelling must provide data that enable simulating transport of radionuclides through the repository and host rock to the biosphere.

Radionuclide transport is mainly devoted to simulate radionuclide transport through the total disposal system. Such modelling is a convenient tool to assess ability of the design options to compensate weaknesses of site features or the degradation of some key components. More precisely, the importance of the components containment capability with regard of the whole disposal system is appraised by quantifying the attenuation of radionuclide flux for various component performances and for various evolutions of the repository system, accounting for the disturbances studied at process level. To better highlight the assets or drawbacks of the investigated components in limiting radionuclides releases, dysfunctions of the repository as well as unfavourable features are postulated in complement of the expected evolution of the system.

Integrated calculations of the radionuclide transport are performed at two scales: near field scale and far field scale. The modelling of the near field allows evaluating the level of performance to be reached and verifying the fulfilment of the safety functions. Through sensitivity analysis, the assessment of the performance of the components influenced by disturbances due to the disposal facility (alkaline plume on bentonite, gas migration...) improves the understanding of the evolution of those components. Accounting for conclusions from near field scale and the hydrogeological settings, the far field modelling is used to assess the global performance of the disposal system and to quantify dosimetric impact at the biosphere. Those scales are complementary in terms of safety assessment, since they participate at two different steps in the understanding of the evolution of the disposal system.

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Part 2: Definition and Assessment of Scenarios

Appendix A6: NDA (United Kingdom)

A6 NDA (United Kingdom)









Project acronym: PAMINA

WP 1.1 – Definition and Assessment of Scenarios



Appendix A6: NDA (United Kingdom)

Section 1: Background/ Introduction

NDA considers that the possible evolution of a repository system can be addressed in terms of a base scenario that provides a broad and reasonable representation of the natural evolution of the system and its surrounding environment, and a number of variant scenarios that represent the effects of probabilistic events.

Section 2: Regulatory requirements and provisions

The regulatory guidance indicates that there is a need to consider all situations potentially giving rise to risk. Paragraph 6.15 of the regulatory guidance states:

6.15 Radiological risk to a representative member of a potentially exposed group is the product of the probability that a given doses will be received and the probability that the dose will result in a serious health effect, summed over all situations that could give rise to exposure to the group.

Section 3: Key terms and concepts

FEPs: Features, Events and Processes that might affect the performance of the repository system.

Base scenario: This provides a broad and reasonable representation of the natural evolution of the system and its surrounding environment (i.e. includes all those features, events and processes (FEPs) that are considered more likely than not to persist for a significant part of the assessment period.

Variant scenario: These represent the effects of probabilistic events (i.e. those FEPs which may or may not occur. FEPs which inititate a variant scenario are termed scenario-defining FEPs.

Section 4: Treatment in the Safety Case

Section 4.1: Methodology

As noted above, the possible evolution of a repository system can be addressed by defining a base scenario and variant scenarios. Any FEPs not considered within the base scenario must either be screened from the assessment basis (with a justification for their irrelevance or insignificance) or considered within a variant scenario. Consideration within a variant scenario does not necessarily imply explicit representation of a specific FEP, many FEPs have a similar impact on system performance and hence can be represented by a single 'scenario representation'.

The scenarios approach leads to an understanding of what is important in terms of the performance of a repository system and hence allows resources to be focused on those aspects most important to safety.

In previous studies screening of scenarios has been carried out using expert judgement on the basis of certain scenarios being physically unreasonable or having an insignificant impact. In order to make such judgements it is necessary to have a suitable framework to ensure that a consistent view is taken in the decision-making process. Where a scenario is considered to be immaterial to the system performance it will be regarded as screened from





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the assessment basis and the justification for this decision will be documented and will form part of the auditable record of the assessment.

If a scenario cannot be screened, it may be possible for it to be subsumed into another scenario that has an equivalent or more serious consequence. The overall aim is to apply a principle of caution to subsume scenario representations at the highest possible level (for example, into the base scenario whenever appropriate) and hence to treat explicitly only those scenario representations which cannot be subsumed. All subsuming decisions are based on the principle of caution, while reserving the option to revisit a decision if it becomes too onerous. This philosophy has the advantage of making the assessment tractable and focusing effort on the most important areas in terms of safety implications. All subsuming decisions are fully justified and will form part of the auditable record of the assessment.

Subsuming of scenario representations involves considering a specific scenario representation in relation to a more general case. If the specific scenario representation has a conditional risk which is similar to or lower than the general case it can be subsumed into the general case. For example, any variant scenario with a conditional risk less than or equal to the base scenario can be subsumed into the base scenario. This will always be conservative, regardless of the probability of occurrence for the variant scenario, as the base scenario is taken to have probability one.

For our generic post-closure performance assessment (GPA), the base scenario includes risks arising from the groundwater pathway and from the generation of repository-derived gas. Variant scenarios that were considered were two human intrusion scenarios.

Section 4.2: Related topics

Uncertainty Management, Scenario Uncertainty.

Section 4.3: Databases and tools

We aimed to be comprehensive in its identification of all relevant FEPs. This was achieved by eliciting FEPs in a structured way using a wide range of appropriate experts. The FEPs were structured on a Master Directed Diagram (MDD) that has the performance indicator, radiological risk, as the top-level FEP. The development of the next level requires identification of those FEPs required to determine the top FEP, i.e. radiological dose and radiotoxicology, which are linked to the top FEP by an 'AND' logic gate. Each of these second level FEPs was developed in the same fashion, and so on to increasingly lower levels of details as the FEPs become more and more specific. The lowest level FEPs on the MDD reflect an appropriate level of detail to form the basis of model development.

In constructing the MDD no FEPs were excluded on the basis that they were insignificant. All FEPs were included, although some were later screened from inclusion in assessment models where there was good and agreed justification to do so.

All non-screened FEPs were associated with one or more conceptual models. The software platform on which the MDD was developed allows 'influence audits' to be created from any FEP, allowing the construction of 'spider diagrams' in which all FEPs can eventually be traced to a conceptual model. This is one tool that facilitates the demonstration that all relevant FEPs have been addressed in the safety case.





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Section 4.4: Application and experience

A matrix diagram was used to examine the interactions between FEPs. The matrix diagram addresses FEPs at the conceptual model level and all potential interactions were considered in a systematic manner. The matrix diagram is particularly helpful for identifying second-order interactions (i.e. where FEP A influences FEP B via FEP C). The matrix diagram has been used to define modelling requirements for new software modules and to assist in packaging assessment work by identifying potential impacts of specific FEPs.

The final strategy by which we aim to ensure a comprehensive modelling approach is through the use of peer review at all key stages. For example, as well as being directly compared with the NEA FEP database, the MDD, matrix diagram and the model development strategy utilising them, were all reviewed by an international expert team; and NDA has an on-going commitment to peer preview and review of all aspects of its safety case development.

Section 4.5 On going work and future evolution

NDA has recently carried out work with Bristol University on the application of Bayesian Belief Networks to variant scenarios connected with climate change. Identification of variant scenarios is a basis for future work in this area.

Section 5: Lessons learned

NDA's approach for development of scenarios received a favourable review by the NEA in 1999. Due to the political changes in the UK programme, however, the methodology has not yet been fully implemented as all safety assessments produced since that date have been generic, i.e. not related to a specific site.

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note

to :	Topic coordinator 'Definition and Assessment of Scenarios'	
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copy :	J. Hart, A.D. Poley	
date :	03 December 2007	
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subject :	NRG Final contribution to topic 2 'Definition and Asses	ssment of Scenarios'

Section 1: Background/ Introduction

In the late 1980's the VEOS study (Safety evaluation of disposal concepts in rock salt) has been performed in the Netherlands [1, 2, 3, 4]. The aims of this study were the evaluation of the post-closure safety of some possible disposal concept and the determination of relevant characteristics. VEOS used a scenario approach followed by a deterministic consequence analysis and several deterministic sensitivity studies. The analyses resulted in a number of release scenarios with estimated exposure. For some scenarios with a relatively high exposure the probability of occurrence was also calculated. The resulting risk defined as the product of this probability and the health effect of the exposure was below the risk levels set in neighbouring countries and the IRCP.

In the early 1990's a generic probabilistic safety analysis (PROSA, [5]) of the Dutch generic reference disposal concept has been performed. In this study a systematic approach to scenario selection has been used that ultimately leads to a set of selected scenarios that covers all aspects relevant for the long term safety. The method used a FEP catalogue to show comprehensiveness of the obtained set of scenarios.

Section 2: Regulatory requirements and provisions

There are presently no regulatory requirements and provisions that directly relate to the definition and assessment of scenarios.

Section 3: Key terms and concepts.

<u>Scenario</u>: Considering the set of all possible futures of the system, a scenario is a subset that contains similar future occurrences (definition taken from the WIPP documentation).

A scenario provides a broad brush description of the relevant events and processes and their sequencing.

In the <u>Normal Evolution Scenario</u> all barriers are functioning as expected and are only attacked slowly by natural processes.

In <u>Altered Evolution Scenarios</u> one or more barriers are compromised.

For a proper disposal system, the probability of the Normal Evolution Scenario is practically





one, while Altered Evolution Scenarios have small probabilities.

Section 4: Treatment in the Safety Case

Section 4.1: Methodology

Method proposed in PROSA

An important aim of the PROSA study was the determination of the sensitivity of the radiological consequences and the derivation of safety relevant characteristics of a disposal concept. So a systematic procedure to account for the variability and uncertainty was used to reach this aim. The scenarios used in the VEOS project [1] were critically reviewed to assure that the important scenarios and the most relevant processes have been accounted for in the consequence analysis. Therefore the starting point of the scenario development should be a comprehensive list of potentially important FEPs. A screening procedure has been applied in order to result in a manageable number of representative scenarios. This screening is a crucial step in each procedure for scenario selection and has to be done in an easy and transparent way with a minimum number of consequence analyses. As this screening is a difficult on the repository system as a whole it was proposed to perform this screening on a number of well defined states of the barriers in the multi-barrier system. In a particular state of the multi-barrier system it is easier to screen the FEPs for several reasons:

- i) In bypassed barriers transport related FEPs can be neglected;
- ii) Each multi-barrier state implies a relevant time scale for the nuclides to arrive in the biosphere. If for instance the isolation shield in the salt formation is not bypassed it takes very long times before the nuclides leave the salt formation and consequently short time FEPs can be neglected.

Having defined the possible states of the multi-barrier system, the screening now consists of identifying the relevant FEPs for each of the multi-barrier states. Not only the FEPs which can cause the state of the barriers but also the FEPs which transport the nuclides in that state of the barriers have to be identified. The methodology proposed to select the scenarios and to find the processes needed in the consequence analysis contains the following steps:

- 1. Identification of FEPs which might influence the state of the barriers, the release, transport, and state of radionuclides. The list should be comprehensive and not be restricted to FEPs induced by nature or the waste but also contain human induced FEPs.
- 2. First screening of the list of FEPs. The first screening of this list is performed with respect to the type of host rock (repository in a rock salt formation) and the probability of occurrence.
- 3. Classification into primary and secondary FEPs. A primary FEP directly attacks or bypasses one or more of the barriers from the multi-barrier system. The primary FEPs consequently define the state or evolution of the repository .In particular they lead to a change in the size or the short circuiting of the barriers. The remaining FEPs are defined as the secondary FEPs. These FEPs influence the transport and the state of the radionuclides. The secondary FEPs define the transport and the state of the nuclides for a given state or evolution of the repository and should be included in the







transport model and/or code.

- 4. Definition of possible multi-barrier states (MBS). In the definition of the state or evolution of the barriers in the multi-barrier system a simple division into attacked or by passed was proposed (see Table 1). In addition a relatively small number of barriers was proposed to limit the number of possible MBS. The main reason for the use of the MBS is the simplification of the further screening prior to the combination of primary FEPs.
- 5. Assignment of the primary FEPs to each of the multi-barrier states taking into account that some processes attack more than one barrier. Table 2 is an example of such an assignment for Multi-barrier state 1.
- 6. Screening of the FEPs for each of the multi-barrier states. In this screening a classification of FEPs with respect to time is very helpful.
- 7. Definition and selection of the scenarios to be analyzed further. This step also includes the selection of the processes to be taken into account in the consequence analysis.
- 8. Determination of the secondary FEPs for each of the multi-barrier states.

Engineered Barriers	Isolation Shield	Overburden	State Number	State Symbol
Present i	Present ii	Present iii	1	Qqq
Present i	Present ii	Bypassed III	2	qqQ
Present i	Bypassed II	Present iii	3	qQq
Present i	Bypassed II	Bypassed III	4	qQQ
Bypassed I	Present ii	Present iii	5	Qqq
Bypassed I	Present ii	Bypassed III	6	QqQ
Bypassed I	Bypassed II	Present iii	7	QQq
Bypassed I	Bypassed II	Bypassed III	8	QQQ

 Table 1 Possible states of the multi-barrier system [5]





Table 2 Primary FEPs related to Multi-barrier state 1 (qqq), the "normal evolution scenario" – in this state of the repository all barriers are present and only attacked slowly by natural processes.

1.2.5	Fault activation	Ρ			iii
1.3.5	Glaciation	Ρ			iii
1.4.4	Denudation	Р			iii
1.4.10	Subrosion	Ρ		ii	
1.5.4	Groundwater discharge	Ρ			iii
2.1.1	Canister defects	Ρ	i		
2.1.2	Common cause (canister) failure	Ρ	i		
2.1.5	Material effects	Ρ	i		
2.1.7	Seal failure	Ρ	i		
2.1.10	Undetected geological features	Ρ	i		
2.1.10	Undetected geological features	Ρ	i	ii	
2.3.1	Archeological investigation	Ρ			iii
2.3.2	Attempt of site improvement	Ρ			iii
2.3.3	Exploitation drilling	Ρ			iii
2.3.4	Exploratory drilling	Ρ			iii
2.3.5	Geothermal energy production	Ρ			iii
2.3.6	Groundwater abstraction/recharge	Ρ			iii
2.3.7	Injection of fluids	Ρ			iii
2.3.9	Recovery of repository materials	Ρ			iii
2.3.10	Resource mining	Ρ			iii
2.3.12	Underground construction	Ρ	i		
3.2.4	Gas generation, explosions	Ρ	i		
3.2.8	Metallic corrosion	Ρ	i		
3.3.4	Fracturing	Ρ	i		
3.4.6	Release of stored energy	Ρ	i		

P: Primary FEP

i: Engineered barrier – present

ii: Isolation shield - present

iii Overburden - present

In the described methodology it is assumed that the evolution of the repository can be defined in terms of barrier states and therefore the methodology can be considered to be a top-down approach in which the different states of the barriers are used as scenario elements.

The PROSA method leads to three families, or distinct grouped sets, of scenarios:

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- 1 The subrosion scenarios;
- 2 The flooding scenarios
- 3 The human intrusion scenarios

This method was fit for its purpose. However, extending the scope of the method to abandonment scenarios (i.e. a different start condition), the method failed and a modification had to be introduced [6, 7]. This modification reflects a more elaborate approach to 'barrier state'. We recognise that the use of safety functions may be a more elegant method to account for the barrier state in different scenarios.

As the PROSA consequence analysis has been performed on the repository as a whole the methodology is not in conflict with the total systems approach.

Section 4.2: Related topics

The ISAM scheme [8] gives a way of handling and management of developed scenarios in safety assessments, taking into account iterative processes and interactions in developing a safety case.

Other related topics are safety functions and probabilistic analysis.

Section 4.3: Databases and tools

FEP database and the procedure for FEP analysis.

Section 4.4: Application and experience

The extended PROSA method [7] has been applied for the safety study underlying to the license application for the closure of the Asse (D) salt mine including the experimental disposal facilities (29. January 2007 [9]) and for a review on behalf of the Ministry of Agriculture and Environment of Sachsen-Anhalt (MLU) of two supporting reports issued in 2002 in preparation of the licensing process for the Morsleben Repository for radioactive waste (Endlager für radioaktive Abfälle Morsleben - ERAM) [10].

Section 4.5 On going work and future evolution

We expect that the PROSA procedure for identifying scenarios will be extended by the application of 'safety functions' for future safety studies.

Also we expect that it will be very useful to present the results of PA-calculations along the lines of safety functions.

Section 5: Lessons learned

Usage of the FEP catalogue leads to more transparency. However, an enormous amount of expert judgement is needed to evaluate all FEPs for all scenarios and subsystems.

Comparison with approaches in other national programmes shows that the overall approach is often similar, but this is obscured by different usage of the same terms. Even a common





definition for 'scenario' could not be established [11].

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Part 2: Definition and Assessment of Scenarios



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Part 2: Definition and Assessment of Scenarios

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Proposal/Contract no.: FP6-036404

Project acronym: PAMINA

Project title:PERFORMANCE ASSESSMENT METHODOLOGIES IN
APPLICATION TO GUIDE THE DEVELOPMENT OF THE
SAFETY CASEInstrument:Integrated ProjectThematic Priority:Management of Radioactive Waste and Radiation
Protection and other activities in the field of
Nuclear Technologies and Safety

WP1.1 Definition and Assessment of Scenarios in the geological repository development Programme NRI and RAWRA contribution

A. Vokál

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Х

Project co-funded by the European Commission within the Sixth Framework Programme (2002-2006) Dissemination level

- PU Public
- PP Restricted to other programme participants (including the Commission Services)
 RE Restricted to a group specified by the consortium (including the Commission Services)
- CO Confidential, only for members of the consortium (including the Commission Services)

[PAMINA] (D-N°: 1.1.1) – Task reports for the first group of topics Dissemination level: RE Date of issue of this report: 15/03/2008





This document describes the work of NRI and RAWRA regarding the scenario development in the Performance Assessment (PA) of HLW repositories in granite.

1 Objectives for selecting and analysing scenarios.

1.1 Compliance with regulations, confidence building, system testing

The waste management legislation in the Czech Republic follows the recommendations of IAEA Safety Standards [1] that the main objectives of underground disposal of high level wastes are to isolate high level wastes from the human environment and to ensure the longterm radiological protection of humans and the environment inasmuch as the releases from a repository due to "gradual" processes or from disruptive events shall be less than the dose or risk upper bound apportioned by national authorities from an individual dose or risk limits taking into account all gradual and disruptive processes that may occur in a repository. Gradual processes are considered to include all evolutionary processes affecting the disposal and disruptive processes are those processes that occur as random events and may have a disruptive effect on the repository and its environment. The similarly defined requirements are included in Czech legislative regulations. According to the Czech Atomic Act [2], approved in January 1997, and relevant regulations all practises resulting in exposure shall maintain such level of radiation protection that the risk to life and health of persons and to the environment is as low as reasonably achievable from economical and social viewpoints. All physical, chemical and biological properties of radioactive wastes must be taken into account and this must be demonstrated in the credible way taking account the site of locality and all risk that can occur in the post-closure period. It is not, however, exactly defined what is meant by all processes, all properties or by all risk. It means that legislative regulation suppose that performance assessment evaluators will describe behaviour of the system and its components and determine under all possible set of events and processes which occur in future, that is under all possible scenarios. Development of scenarios is thus an implicit requirement of legislation, but with no exact guide.

1.2 Requirements from regulations

It has been defined by regulations of Czech regulatory body (State Office for Nuclear Safety)[3] that the potential individual dose raised by repository existence, has not to exceed 0.25 mSv/yr for normal evolution scenarios and/or 1 mSv/yr for emergency scenarios. There exists no other quantitative limitation postulated by nuclear legislation or some other concerning scenarios.

2 Methods for the development of scenarios

Systematic scenario development in Czech geological disposal programme started in 1996 by analysing abroad approaches, primarily Sandia Scenario Selection Procedure [4] and



SKI/SKB [5, 6] scenario development approach. Under the influence of these approaches the following elements, which seemed to be the most important at that time for DGR concept in granite host rock, were defined:

- Engineered Barrier System
 - Waste form
 - Container
 - Buffer
 - Backfill
 - Seals
- Host rock
 - Groundwater (chemistry)
 - Fractures (flux)
 - Mechanical stress (tectonic changes)
- Technology
 - Selected disposal and excavation technologies
 - Layout of the repository
 - Construction materials

These elements were placed in diagonal boxes of interaction matrices and specialists from different fields (chemistry, geotechnics and geology) were asked to prepare literature review and to classify and discuss interactions between the elements in the interaction matrices. The classification assessment was in the range from 1 to 5, where 1 was negligible interaction and 5 critical one. It was found, however, that the results of this classification were strongly affected by major fields of specialists that answered questionnaires. They focused primarily on discussing and evaluating the interactions pertinent to their fields and primarily those interactions regarded as the most important ones. This "bottom up" approach was therefore abandoned.

The scenario development in Czech programme in further years was affected by participation of Czech specialists in Performance Assessment Advisory Group (PAAG) of the Radioactive Waste Management Committee (RWMC) of NEA and by consequent NEA publications [7] The following scenarios were selected and accepted in Czech programme for reference concept by performance assessment specialists mainly on the basis of the study of international FEPs database [8, 9]:

- Normal evolution scenario covering all processes with high probability of occurrence
- Altered scenarios initiated by unfavourable initial conditions
 - Premature container defect at manufacture it can lead to earlier contact of water with waste. (Calculations are the same as in normal scenario, but with other parameters, depending on assumed number of containers with premature defect)
 - Damage backfill it can lead to increased hydraulic conductivity and possible movement of container in a borehole etc. (Calculations are the same as in normal





scenario, but with other parameters for buffer and backfill, and other distances between containers and host rock)

- Wrong container emplacement it can lead to contact of container with higher amount of water than supposed, higher corrosion rates and higher release rates of radionuclides (Calculations are the same as in normal scenario, but with other distances between container sand host rock)
- Stray construction materials left in the repository it can lead to change of chemistry and properties of engineered barriers and higher corrosion rates (Calculations are the same as in normal scenario, but with other parameters container lifetime, for porewater composition, etc.)
- Presence of higher amount of microbes (Calculations are the same as in normal scenario, but with other parameters container lifetime, for porewater composition, etc.)
- Work in host rocks it leads to changes of stress in disposal sites or generation of fractures (Calculations are the same as in normal scenario, but with other parameters container lifetime, for porewater composition, etc.)
- Altered scenarios initiated by climatic changes
 - Glaciation it can lead to change of water fluxes and chemistry (The impact depends on the time of glaciation, calculations are the same as for normal scenario, but with other parameters for container lifetime, porewater composition, etc.). It was agreed that in the Czech Republic the changes connect with glaciation will not be significant in next 10 000 years.
 - Permafrost it can lead to change of water fluxes and chemistry (The impact depends on the time of permafrost, calculations are the same as for normal scenario, but with other parameters container lifetime, for porewater composition, etc.).
 - Seismic changes due to climatic changes, e.g. seismic changes after glaciation period (The impact depends on the time of permafrost, but calculations are the same as for normal scenario, but with other parameters container lifetime, for porewater composition, etc.).
 - Global warming and other less significant climatic changes It can be expected only small changes in host rock and repository itself. The major impact will on biosphere conversion factors
- Human induced scenarios
 - Human intrusion
 - Drilling of borehole in a repository leading to the change of hydraulic conditions in the repository and possibly preferential way for radionuclide release (calculations are the same as for normal scenario, but with other parameters container lifetime, porewater composition, flux of water etc. depending on time of drilling.) – The probability of this scenarios is presumably very low and must be discussed
 - Drilling through disposal units and taking samples out on surface (This is a special scenario requiring another way of calculations based on exposure of workers, which perform drilling and analyses. The probability of this scenario will be presumably very low)





- Excavation work on surface it leads to major changes in flux of groundwater (Calculations are the same as for normal scenario, but with other parameters)
- Change of chemistry of site due to human action (dumping of waste near to surface, intensive agriculture) (Calculations are the same as for normal scenario, but with other parameters)

Some processes evidently not pertinent to granite or Czech Republic geography, such as salt diapirism and dissolution, hydrothermal activity, volcanic and magmatic activity or meteoric impact, were excluded from scenarios considered. The possibility of occurrence of criticality inside or outside of waste packages was discussed and it was concluded that this scenario cannot be excluded from scenarios without a more detail calculations and will be treated in future.

3. Consideration and estimation of probability

3.1 Probability of unfavourable conditions in a repository

The probability of unfavourable initial conditions in the repository can be estimated using approaches for assessment of reliability of components in modern nuclear power plants, where the frequency of failure of components must be lower than 10⁻⁵ in year. Since DGR for spent fuel assemblies and HLW is considered as nuclear facility, the all requirements common in NPP will have to be also applied for DGR and its components.

The components in nuclear power plants contrary to DGR components are not exposed to changing conditions, as it is the case in nuclear power plants. It can be therefore supposed - in agreement with reliability experts - that failures of DGR components, be it canister or buffer backfill due to some hidden defect in the first hundred of years after closure will be very low. If we conservatively supposed that this value would be 10^{-6} in a repository with 5000 canisters, then it can be calculated that after 200 hundred years only one canister would fail due to some hidden defect. The same approach can be applied to other components of DGR.

3.2 Probability of natural events

Probability of some natural events, such as glaciation or permafrost on the territory of candidate sites in the Czech Republic has not been performed so far. It is planned to estimate it on the basis of expert judgement in future projects. The probability of some events is, however, reduced to minimum by exclusion criteria given by decree of Czech regulatory body on siting of nuclear facilities including repositories for radioactive wastes [9]:

- The occurrence of karstic phenomena in the extent endangering the stability of the rock massif in the bedrock and in the rock cover of the land selected for the siting.
- The manifestation of post-volcanic activity such as the escapes of gases, thermal, mineral and mineralised waters, found on the lands or area of the supposed siting and in their site vicinity zones.
- The achievement or exceeding of the value of intensity of the maximum calculated earthquake 8 °MSK (scale of Medvedev-Sponheuer-Karnik for estimation of the





macroseismic effects of earthquakes) on the lands of supposed siting.

• The occurrence of the capable and seismogenic faults with the recent surface deformations of area and with the possibility of origination of secondary faults, found by a geological survey on the land of supposed siting.

3.3 Probability of human induced scenarios

Probability of some human activities on the site of DGR is reduced by the following exclusion criteria by mentioned decree of Czech regulatory body:

- The existence of the significant underground waters supply or mineral waters in the site vicinity zones.
- The occurrence of the raw material mining in the site vicinity zones.

No attempt has been made to quantify probabilities of human induced scenarios in Czech DGR programme

3.4 Events or processes frequency quantification approach

It is very difficult to estimate frequencies of some events, features or processes. But some events or processes are possible to describe by words as highly, medium or low probable or incredible. The frequencies allocated to these expressions given in the Table 1 have been considered to be applied.

Expression describing an event or process	Nominální value of frequency per year			
Highly probable	Pe > 10 ⁻²			
Medium probable	Pe = 10 ⁻² až 10 ⁻⁴			
Low probable	Pe = 10 ⁻⁴ až 10 ⁻⁶			
Incredible	Pe < 10 ⁻⁶			

 Table 1: Frequency quantification [10]

4 Tools for scenario development

Only expert judgement approach based on studying FEPs relevant to Czech concept has been applied for scenario development. Currently top down system described in the document devoted to safety functions is being formed. This system is strictly going from top functions to daughter functions and requirements. At each level of system decomposition it will be tested whether the identified safety function is fulfilled under all external effects from outer systems. On the top level there is only one disposal system defined by some boundary conditions and outer systems, such as human environment, natural environment, or climate changes with all its impacts on repository. In testing the disposal system all interactions between disposal and outer systems must be identified and analysed on each





Part 2: Definition and Assessment of Scenarios

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level of system decomposition. If the impact is considered not to be negligible then further functions and requirements (or constraints) can imposed on the disposal system or possibly on the site. For example, if it was concluded then glaciation or permafrost can have at some time in future an impact on the repository in the Czech Republic then also all subsystems and components on lower level of decomposition must take into account the change of the system under glaciation or permafrost.

5 Experience with the application of scenario development in the context of safety assessment

In preliminary safety analyses, which have been performed in the Czech Republic so far, conservative parameters more characteristic to altered scenarios then to normal evolution scenario have been used. A good example is lifetime of canisters, which is conservatively taken to be about 1000 years. Our experimental results, however, suggest that this value is unrealistically low. Lifetime of carbon steel canisters will be an order of magnitude higher. The value of 1000 years is therefore more characteristic to an altered scenario initiated by hidden defects of canisters. Also the parameters used in preliminary scenarios for site are more characteristic to some altered scenarios initiated by selection of the site with unrecognised defects or defects caused by some natural event or human activity. In future safety analyses, values used for the normal evolution scenario evaluations will be based on more realistic data and altered scenarios will be identified using the approach outlined in the document devoted to safety functions.

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Appendix A9: POSIVA (Finland)

A9 POSIVA (Finland)





Appendix A9: POSIVA (Finland)

PAMINA WP1.1 Definition and Assessment of Scenarios Posiva Oy – Contribution – 2nd Draft

PAMINA. WP1.1 Comprehensive Review of Methodologies and Approaches in the Safety Case

Definition and Assessment of Scenarios

1 Background

The latest safety assessment of Posiva is TILA-99 (Vieno & Nordman 1999). TILA-99 did not use the concept scenario as defined in the IAEA (2003). The scenarios in TILA-99 were in fact calculation cases. The uncertainties were treated varying the parameters to be used in the calculations and by "adding" the effect of parameters related to processes that were uncertain to occur at the same time. The calculation cases in TILA-99 could be grouped to fit within a few scenarios using "scenario" as defined in the IAEA (2003)

Currently in Posiva's Safety Case the forthcoming radionuclide transport report (safety assessment report) is under work and no results are yet available. However the definition of scenarios is being dealt within the Process report (POSIVA 2007) scheduled by the end of 2007.

2 Definition and types of scenario

According to the IAEA (2003) definition, scenario is defined as "a postulated or assumed set of conditions and/or events. They are most commonly used in analyses or assessments to represent possible future conditions and/or events to be modelled, such as possible accidents at a nuclear facility, or the possible future evolution of a repository and its surroundings".

According to the STUK's regulatory guide "a scenario analysis shall cover both the expected evolutions of the disposal system and unlikely disruptive events affecting long-term safety. The scenarios shall be composed systematically from features, events and processes, which are potentially significant to long-term safety and may arise from

- mechanical, thermal, hydrological and chemical processes and interactions occurring inside the disposal system
- external events and processes, such as climate changes, geological processes and human actions.

The base scenario shall assume the performance targets defined for each barrier, taking account of the incidental deviations from the target values. The influence of the declined overall performance of a single barrier or, in case of coupling between barriers, the combined effect of the declined performance of more than one barriers, shall be analysed by means of variant scenarios. Disturbance scenarios shall be defined for the analysis of unlikely disruptive events affecting long-term safety."





Appendix A9: POSIVA (Finland)

Posiva's safety case considers scenarios as relatively complete descriptions of future developments. In practical work, the scenarios are quantitatively evaluated using calculation cases that – after conceptualisation of scenarios – handle them in various entireties. Thus calculation cases represent more restricted sets of assumptions than scenarios. In addition, calculation cases are used for handling uncertainties within the defined scenarios e.g. by varying the values of calculation parameters.

3 Methodology and Scenarios in Posiva's Safety Case

The method for developing scenarios follows a top down approach since most of the scenarios to have into consideration come from regulatory requirements. This means that we first select or define the scenarios to be analysed and then use FEP lists/databases, complemented with expert judgement, to check that nothing important has been left out of consideration.

3.1 Definition/description of main scenario, defective canister scenario, additional scenarios, and variants

The scenarios considered in the Posiva's Safety Case portfolio have been partly defined in the Evolution Report (Pastina & Hellä 2006). In the main scenario of that report all system components are expected to behave as designed to keep their long-term safety functions over all time frames required by regulations (YVL 8.4) and the time frames defined in the two climatic scenarios (Weichselian-R and Emissions-M) to be taken into account (see Chapter 5 in Pastina & Hellä 2006). No major disruptive events giving place to radionuclide releases are expected within the main scenario.

Following STUK's recommendations (STUK 2001), the defective canister scenario (DCS) has also been defined in the Evolution Report. In the expected evolution of the repository no release of radionuclides occur within 100 000 to 1 000 000 years. Two variants are considered within this scenario, DCS-I and DCS-II. For the purpose of radionuclide transport calculations in the main variant (DCS-I) it is assumed that the canister has no initial penetrating defects and that release of radionuclides does not occur within the first 10 000 years after closure of the repository. In the main alternative (DCS-II) it is assumed that the canister has an undetected penetrating defect and that release of radionuclides may start immediately at the repository closure.

Because of the uncertainties in the occurrence and timing of disruptive features (e.g. site properties), events (e.g. rock block movements) and processes (e.g. corrosion), additional scenarios (AD) are defined for the purpose of radionuclide transport calculations and to comply with specific regulatory requirements. Three variants are considered within additional scenarios: AD-I considers the failure of one or more canisters as a consequence of a sudden rock block movement along a fracture intersecting one or more deposition holes. AD-II considers disruptive events both in the initial conditions of the buffer and its emplacement leading to large corrosion rates. AD-III considers that gas expels the radionuclides of instant release fraction (IRF) from the deposition hole.

A major requirement of the regulator is the human intrusion scenario (HI) where two variants


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are to be considered, HI-I assumes boring a deep water-well at the disposal site and HI-II assumes core drilling, hitting a canister.

Table 1 summarizes the set of scenarios and variants described above.

Table 1. Scenarios in Posiva's Safety Case

Scenarios in Posiva's Safety Case	Descriptions	
Main (base) scenario	No release of radionuclides within safety-relevant period of time	
Defective canister scenario (DCS)	- DCS-I : Main option (no penetrating defect – no radionuclide release within the first 10 000 years after closure of the repository)	
	- DCS-II : Main alternative (penetrating defect – radionuclide release any time after closure of the repository)	
Additional scenarios (AD) come from deviations in initial conditions and timing of processes (whatever internal/external)	AD-I: Earthquake / Rock shear	
	AD-II : Disruptive events both in the initial conditions of the buffer and its emplacement leading to higher corrosion rates than expected.	
	AD-III : Gas expels IRF from the deposition hole. This case requires an initial penetrating defect at the bottom of the canister as a prior condition.	
Human intrusion scenario (HI)	HI-I: Boring deep water well at the disposal site	
	HI-II: Core-drilling penetrating into a canister	

4 Assessment of Scenarios – organization of calculation cases and variants for safety assessment (radionuclide transport analyses)

Since the main scenario is tied in the expected evolution, where no releases of radionuclides will occur within safety-relevant period of time, no calculation cases are needed for its handling. On the other side, the defective canister scenario (DCS), additional scenarios (AD) and human intrusion scenarios (HI) in Table 1 are called assessment scenarios and are appraised by means of quantitative analyses (see Figure 1). The scenario variants will be conceptualised and several calculation cases will be derived that do not aim to be realistic but rather explore the robustness of the system.

The latter ones include what in TILA-99 (Vieno & Nordman 1999) were called "What if" cases and "sensitivity cases". For example the calculation cases for DCS-I are defined based on the timing of corrosion process and the physico-chemical conditions at the time (e.g. the flow rate during ice sheet formation or melting is significantly different; Pastina & Hellä 2006). The calculation cases for DCS-II are defined combining the size of the penetrating defect, the time of release of radionuclide, the buffer and backfill conditions, and the groundwater physico-chemical conditions at the time of release.





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Figure 2-1. The hierarchy of scenarios (1), conceptualization (2) and derivation of calculation cases (3).

Figure 2-2 shows the derivation calculation cases for DCS-II as an example. A complete description of the calculation cases derived from the scenarios will be given in the radionuclide transport report scheduled to spring 2008.



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Figure 2-2. Derivation of calculation cases in the Defective Canister Scenario DCS-II or Main alternative.

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Part 2: Definition and Assessment of Scenarios



Appendix A10: SCK-CEN, ONDRAF-NIRAS (Belgium)

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PAMINA PROJECT WP 1.1 Review of Methodologies

Definition and Assessment of Scenarios in the Belgian HLW Disposal Programme

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1. Background / Introduction

For scenario development three main phases can be distinguished in the Belgian radioactive high-level waste (HLW) disposal programme:

- phase 1 (period 1978 1990): a number of less systematic approaches were applied; these approaches will not be discussed in the present paper;
- phase 2 (period 1992 1999): a systematic approach based on a catalogue of features, events and processes (FEPs) was introduced; this approach was used in the SAFIR 2 (safety and feasibility interim report) report (ONDRAF/NIRAS, 2001);
- phase 3 (period 2004 2012): the new approach is still in development, partially within PAMINA, and will be applied for the Safety and Feasibility Case 1 (SFC 1).

2. Regulatory requirements and provisions

Regulatory requirements and guidelines concerning long-term safety of high-level radioactive waste disposal are still in preparation in Belgium.

3. Key terms and concepts

Scenario development is defined as "the identification, broad description, and selection of potential futures relevant to safety assessments of radioactive waste repositories" (definition



taken from NEA (2001)). The main objective of the scenario development is to show in a traceable and transparent way that all potentially important features, events and processes of the repository system have been taken into account in the safety assessment.

Within the Belgian HLW management programme we distinguish two essential steps in scenario development:

- *identification* of the main evolution scenarios aiming at identifying a sufficiently representative set of scenarios: the base case is the expected evolution (or normal evolution) scenario, which is complemented by a set of altered evolution scenarios; if felt necessary, a number of "what-if" cases can also be considered in the evaluations;
- *description* of the identified scenarios in such a way that it is clearly shown how the retained features, events and processes are treated within the considered scenario.

The scenarios are grouped in 3 families of scenarios:

- *Expected Evolution Scenario (EES)*: the expected future evolution of a disposal system after facility closure, which is consistent with current understanding; in the SAFIR 2 report (ONDRAF/NIRAS, 2001) this scenario was called *normal* evolution scenario.
- Altered Evolution Scenarios (AES): the assumed future evolutions of a disposal system after facility closure taking account of less likely disturbing events or processes that are capable of significantly altering the system if they do occur. The altered evolution scenarios do not cover inadvertent human intrusion, which is treated separately.
- *Human intrusion scenarios (HIS)*: a number of future human actions (e.g. borehole drillings) can result in an intrusion in the sealed repository. The human intrusion scenarios that have to be analysed will be discussed with the radiological protection authorities.

Beside the 3 above mentioned families of scenarios, a number of "whar-if?" cases can also be defined:

• *"What if?" cases*: cases set up to test the robustness of a disposal system. "What if?" cases are outside the range of possibilities supported by scientific evidence and may seem physically impossible to occur. They are restricted to those that test the effects of perturbations on key contributors to safety.

Features, events and processes (FEPs): the features, events and processes that can have an impact on the behaviour of a disposal system and its environment.

Reserve FEP: a FEP that is considered likely to occur and to be beneficial to safety, but that is deliberately excluded from scenarios, or at least from their analysis, when the level of scientific understanding is insufficient to support quantitative modelling, or when suitable models, codes or databases are unavailable. Reserve FEPs may be mobilized at a later stage of repository planning if the level of scientific understanding is sufficiently enhanced, and the necessary models, codes and databases are developed.





4. Treatment in the safety case

4.1 Methodology

4.1.1 Methodology used in SAFIR 2

After the publication of the NEA report on scenario development (NEA, 1992), it was decided to elaborate a study on systematic identification of altered evolution scenarios (Marivoet, 1994) starting from a catalogue of features, events and processes (Bronders et al., 1994) for the case of geological disposal of high-level radioactive waste in the Boom Clay layer at the Mol site. For the description and analysis of the identified scenarios an approach based on "a robust repository concept" was used (see section 4.1.1 b).

a) Identification of altered evolution scenarios

The applied approach was developed in the framework of the EVEREST project (Gomit et al., 1997) in collaboration with J. Prij from ECN (Petten, the Netherlands), who developed the PROSA approach for the case of disposal in salt (Prij, 1993).

The main steps of the PROSA approach are:

- identification of relevant FEPs;
- classification of FEPs according to their occurrence probability;
- classification of FEPs according to the state of the repository system;
- identification of altered evolution scenarios.

1) Identification of relevant FEPs

For the preparation of the catalogue of FEPs relevant for geological disposal in the Boom Clay formation at the Mol site, Bronders et al. (1994) started from the FEP list of the NEA (1992) report. This list was complemented with a few FEPs specific for the case of disposal in clay:

- decrease of the plasticity of the clay;
- oxidation of the host rock during construction and operation;
- excavation effects.

The catalogue gives a short description of each FEP and discusses its relevance for the case of disposal in the Boom Clay at the Mol site.

The considered FEPs were screened by applying the following elimination criteria:

- probability lower than 10⁻⁸ per year;
- negligible consequences;
- not relevant for the considered waste types;
- not relevant for the considered repository design;





- not relevant for a clay formation;
- not relevant for the Mol site;
- responsibility of future generations;
- multiple entries or similar effects.

The FEPs that only have impact on the biosphere were considered in the development of the reference biosphere and were not to be taken into account for the scenario development of the repository system.

The FEP catalogue considered 134 FEPs, 58 FEPs were eliminated as irrelevant and 16 only affected the biosphere. Thus, 60 FEPs were retained for treatment in the scenario development.

2) Classification of FEPs according to their occurrence probability

In the case of geological disposal in clay formations, groundwater will penetrate into the near field of the repository after a relatively short period and the migration of radionuclides is expected to start immediately after the perforation of the overpacks or canisters. The expected evolution scenario was introduced as the scenario that considers the expected evolution of the repository system. It should take into consideration all the FEPs that are certain or about certain to occur and that have the potential to significantly influence the performance of the essential repository components.

The retained FEPs were classified on the basis of their probability of occurrence into two groups: those to be treated in the expected evolution scenario and the others that can lead to altered evolution scenarios.

However, for a number of FEPs, e.g. glaciation and gas mediated transport, this classification depends on the severity or magnitude of the considered FEP. Glaciations comparable to the three most recent glaciations of Quaternary are expected to occur on the basis of Milankovitch's orbital theory. However, the occurrence of a very severe glaciation, i.e. an ice-cap reaching the Mol area, cannot be completely ruled out in this early phase of the scenario development. In the case of disposal of vitrified high-level waste, the amount of metals or other materials that can contribute to the generated gas can be evacuated by diffusion in the interstitial clay water. In this case, gas mediated transport will only occur if the gas generation is higher or the evacuation rate lower than expected. On the other hand, in the case of disposal of, e.g., medium-level waste, the expected gas generation are expected.

Of the 60 retained FEPs 45 were treated within the expected evolution scenario, and 17 were considered for the identification of the altered evolution scenarios.

3) Classification of FEPs according to the state of repository system

A top-down approach, called the PROSA methodology (Prij, 1992) was developed in the early nineties. The PROSA methodology can be considered as a variant of the SKI/SKB top-down approach (Andersson et al., 1989). The repository system is partitioned into three compartments or main components: the near field, the host clay layer and the aquifer system. As mentioned above, the biosphere is treated separately. Each main component can







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be in two possible states: intact or by-passed. The repository system can thus be in 8 possible states (see Table 1).

Table 1: Definition of the possible states of the repository system(i: intact component; b: by-passed component)

State number	Near field	Clay barrier	Hydrogeology
1	i	i	i
2	i	i	b
3	i	b	i
4	i	b	b
5	b	i	i
6	b	i	b
7	b	b	i
8	b	b	b

The altered evolution FEPs were classified according to the state of the repository system to which they lead. FEPs that affect the same component could in many cases be treated together or could be considered as variants within one group of scenarios.

4) Identified scenarios

The expected evolution scenario corresponds to state 1.

The following altered evolution scenarios were identified:

- exploitation drilling (state 2): this scenario considers the drilling of a water well in the aquifer underlying the host formation; it has to be noticed that the drilling of a well in the overlying aquifer is already considered in the analysis of the expected evolution scenario;
- green-house effect (state 2): this scenario takes into account the possible impact of global warming on the aquifer system and, of course, on the biosphere;
- poor sealing of the access shafts and main galleries (state 3): it is assumed that, owing to a human error, the access shafts and main galleries have not been successfully sealed and the poorly sealed galleries and shafts might create a preferential pathway for the migration of radionuclides through the clay layer;
- fault activation (states 3 and 7): it is assumed that an active tectonic fault crosses the repository affecting the confinement provided by the host clay layer;
- severe glaciation (states 4 and 8): this might lead to the occurrence of an ice-cap in the Mol area; subglacial erosion can reach depths up to 400 m, and, as a consequence, can severely affect the clay barrier; as an extreme case, it might bring remnants of the disposed waste to the surface;
- early failure of the engineered barriers (state 5): many variants can be considered in this group of scenarios; however their consequences are strongly limited by the





presence of the intact host clay barrier; one of the more severe variants is an early failure of the overpack in the case of heat generating high-level waste: this will lead to migration of radionuclides while considerable thermal gradients occur in the near field; various coupled thermo-hydro-mechanic-chemical transport processes might occur;

- gas driven transport (states 3 and 7): if the gas production rate is higher than the gas evacuation rate, a gas bulb will be formed in the near field and pressure will build up; when the gas pressure exceeds the effective stress of the host formation, a disruption of gas into the clay layer will occur; the expelled gas bulbs can contain radioactive gases and they might also convey a fraction of the near field ground water, containing dissolved radionuclides, into the host clay layer;
- exploration drilling (state 8)⁴: it is assumed that a borehole is drilled through the repository; three variants of this group of scenarios are analysed: (1) the examination of a core containing radioactive waste by a geologist (cf. scenario described by Smith et al., 1987); (2) the borehole cuttings contain fragments of the disposed radioactive waste and are dumped on the surface in the neighbourhood of the drilling site; (3) the walls of the borehole are left open and ground water is flowing through the borehole, where it comes in contact with the disposed waste.

A comparison of the outcome of different approaches to systematic scenario development was carried out in the framework of the EVEREST project (Gomit et al., 1997). The French organisations ANDRA and IPSN (now IRSN) applied the independent initiating events methodology, while SCK•CEN applied the approach described above. It appeared that both approaches led to the identification of very similar scenarios for the case of disposal in clay formations. This conclusion strengthened the confidence that the most relevant scenarios were identified.

b) Description of evolution scenarios

The description of the expected evolution scenario (Marivoet, 1999) started from the list of FEPs that are about certain to occur. The description and, as a consequence, the analysis of this scenario are strongly simplified by the introduction of the robust repository concept (NAGRA, 1994). This means that the complex real repository system is reduced to a simpler system that can be modelled with a higher level of confidence. In the robust repository concept the disposal system is reduced to the safety-relevant characteristics and processes in which there is a high level of confidence and to those processes which can be detrimental to safety. On the other hand, some components and processes that give a positive contribution to the confinement can be conservatively neglected.

4.1.2 Methodology being developed for SFC 1

From national and international (NEA, 2003) peer reviews as well as from internal discussions, it appeared necessary to develop a much more detailed assessment basis and an up-to-date scenario development methodology for the Safety and Feasibility Case 1 (SFC 1).



⁴ This scenario is a typical human intrusion scenario. However, in the period 1991-1994, when the PROSA methodology was developed, future human actions were treated in the same as FEPs of natural origin or FEPs induced by the repository or the disposed waste.



Safety functions were successfully introduced in the Belgian HLW management programme in 1999 (De Preter et al., 1999). Therefore, it was decided to base the identification of altered evolution scenarios on the availability or non-availability of the safety functions instead of on the intactness or failure of the main barriers of the repository system. For the scenario descriptions it was agreed that they would be based on detailed phenomenological descriptions of the expected (or considered altered) evolution of the repository system.

a) Description of evolution scenarios

The reports describing the assessment basis will present a phenomenological description of the expected evolution of the disposal system and will present the information in terms of the safety functions. These reports will also provide an opportunity to identify potential initiating events. In a second phase, phenomenological descriptions of the altered evolution scenarios will also be prepared.

b) Identification of altered evolution scenarios

In many safety cases the identification of altered evolution scenarios was done in a rather arbitrary way; e.g., one of the recommendations of the International Peer Review of SKB's SR-Can Interim Report (SKI/SSI, 2005) was "a clearer, more structured approach to scenario identification would help to make the logic of the process, and the role of supporting arguments in scenario screening and selection, more visible". Therefore, it was decided to develop for the SFC 1 a systematic approach for scenario identification aiming at combining the advantages of the PROSA methodology, with the use of safety functions.

Starting from the functional analysis and a list of potential scenario initiating events, both available from the assessment basis (cf. section 4.1.2 a), it is proposed to apply the following methodology for the identification of altered evolution scenarios, and possibly of "what-if" scenarios:

- 1. examine which safety function can be affected by which scenario-initiating event;
- 2. construct functional diagrams to illustrate the impact of the considered event on the functioning of the disposal system;
- 3. group scenarios with similar functional diagrams as far as possible;
- 4. check if failures of safety functions not yet considered in steps 1 and 2 should be treated as "what-if" cases.

4.2 Related topics

4.2.1 Estimation of probabilities

It appeared that only for a very limited number of scenario-initiating events, e.g. meteorite impact, it was possible to estimate a probability of occurrence (d'Alessandro and Bonne, 1981). For the identified altered evolution scenarios, we only evaluate radiological consequences (doses) and we do not try to estimate risks. The discussion of the likelihood





remains qualitative.

4.2.2 Use of stylised approaches

For the biosphere model, a stylised biosphere based on present practices is used. The possibility to include the impact of climate changes on the biosphere is taken into consideration. It is foreseen to develop, partially within PAMINA, stylised human intrusion scenarios during the next years.

4.2.3 Time scales

The following time scales are considered in the Belgian HLW disposal programme:

- thermal phase: the thermal output of the disposed HLW heats up the near field and the • host clay formation; this thermal phase lasts about 600 years in the case of vitrified HLW disposal and 2000 years in the case of spent fuel disposal; the near field gets resaturated during the first decades of this phase;
- engineered containment phase: during this phase the intact overpack prevents contact • of groundwater with the disposed waste; this phase should last at least as long as the thermal phase; for the current repository design, this phase is estimated to last about 10 000 years;
- system containment phase: after breaching of the container, groundwater comes in • contact with the disposed waste and radionuclide migration through the buffer and the host clay formation will start; however, it will take a few tens of thousands of years before significant amounts of non-retarded radionuclides will be released into the surrounding aquifers and eventually into the human environment; retarded radionuclides will remain confined in the disposal system during hundreds of thousands of years;
- stable geological barrier phase: the well functioning of the repository system requires • stable conditions that ensure that the main barriers and processes of the repository system can fulfil the safety functions that were attributed to them; this phase is assumed to last about 1 million years.

4.3 Databases and tools

A FEP catalogue, for which the NEA FEP database (SAM, 2006) and the FEPCAT report (Mazurek et al., 2003) were the main input documents, in database format is used for completeness checking of the phenomenological descriptions.







4.4 Application and experience

The PROSA methodology has been applied in the SAFIR 2 report (ONDRAF/NIRAS, 2001). It appeared necessary to develop a much more detailed assessment basis and an up-to-date scenario development methodology for the Safety and Feasibility Case 1, which is scheduled to be published in 2013.

4.5 On going work and future evolution

See section 4.1.2.

5. Lessons learned

A systematic approach for scenario identification was introduced in the Belgian high-level radioactive waste disposal programme during the first half of the nineties. This approach was applied for the SAFIR 2 report (ONDRAF/NIRAS, 2001). Although based on the functioning or non-functioning of components of the repository system, the applied approach had as advantage that it showed in a traceable way how the considered altered evolution scenarios had been selected. The scenario descriptions appeared to be insufficiently developed, especially the description of the near field evolution.

In 2003 it was decided to develop a new approach based on phenomenological descriptions of the evolution of the repository system and on safety functions. This new approach is still in development. The detailed phenomenological descriptions of the evolution of the repository system are expected to show in a traceable way that all relevant FEPs have been taken into account. They will also identify potential scenario-initiating events. Furthermore, completeness checks will be organised, in which a review team will verify whether all the FEPs of the FEP catalogue, which was independently developed mainly on the basis of the FEP catalogues that are available in the NEA FEP Database (SAM, 2006), were taken into account. The identification of altered evolution scenarios, and possibly of "what-if" scenarios will be based on the availability or non-availability of the safety functions. Recently a somewhat similar approach for scenario identification has been developed by SKB (2006). We expect that the proposed methodology for the identification of altered evolution scenarios of the safety functions scenarios on the basis of safety functions will give satisfying results.

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PART 3: UNCERTAINTY MANAGEMENT AND UNCERTAINTY ANALYSIS

(Prepared by J. Alonso. Enresa, Spain)





1 Introduction

Uncertainty is inherent to all kind of safety assessments. In general, uncertainties arise from imperfect knowledge of the system to be assessed and its evolution. In the case of geological disposal, there are specific characteristics which enhance the relevance of uncertainties for the post-closure safety assessment.

In the European Pilot Project [Vigfusson *et al.* 2007] it is stated that "uncertainties concerning the safety of repositories are unavoidable due to the complexity of the phenomena of concern and the scales in time and space under consideration, and their management is central when developing a repository system and assessing its safety".

In the first place, the time scales to be considered in the safety assessments of geological repositories are very long Typically, the assessment is extended to hundreds of thousands of years or more. These long time scales introduce further sources of uncertainty, make some uncertainties larger and exclude the assumption that after some time (a few hundred years) human actions may be accounted for preventing, detecting, mitigating or otherwise reacting to the deviation from the expected evolution of the repository. The long time scales have also as a consequence that the design and the long term safety assessment cannot build on the experience of previous facilities of the same kind.

Another important characteristic of geological repositories in regard of safety assessment uncertainty is the variability of the natural media in which the repository is placed. Natural media are essential components of the repository system, for two reasons. Firstly the natural barrier plays a role in the confinement of the contaminants disposed of and also in the protection of the inner components of the repository system (i.e. the engineered barriers). Secondly the natural environment controls the background conditions in which processes influencing the performance of the different safety barriers take place. Variability of the natural media occurs both in time and in space. Initial variability of the natural media ("the site") is the object of site characterisation and site monitoring; nevertheless the space scales of the site put practical limits to exhaustiveness and the level of resolution at which the relevant features of the site can be known. The characteristics of a site are not constant: on the contrary, they will evolve under the influence of factors both external and internal to the repository system, including the interactions with the man made components and with the waste. Furthermore, as it has been pointed out above for the time scales, the space scales make impossible the direct test of the repository system in order to verify its performance.

For the reasons explained above, the means to assess the long term safety are necessarily indirect. Tests and experiments are only possible over short duration and in contexts which can only approach those expected in reality. Experimental data need to be extrapolated to the required scales; this is done typically using models, based on the understanding of the rules that control or bound the evolution of the physical entities in the future. This extrapolation at very different scales is another source of uncertainty.

The end point of the quantitative safety assessment of a geological repository is the mathematical calculation of a safety indicator, and its comparison with a relevant numerical criterion, as defined by the regulations (see Pamina WP1.1 report on Safety and performance indicators). Nevertheless, this comparison is meaningless if there is not an analysis of how the different uncertainties affect confidence in the safety indicator. The aim of uncertainty analysis is to provide confidence in the bases and arguments developed to





support the claim that the repository is safe, and that the mathematical estimation of the safety indicators does not misrepresent the expected performance of the repository.

In most regulations and guidance documents developed for geological repositories, it is emphasised that uncertainty analysis is a key component of the safety assessment. An essential activity within the safety assessment is the identification of uncertainties that have the potential to undermine safety. Thus, safety assessment needs to be integrated within the management strategy. In the safety case, the connection needs to be made between key uncertainties that have been identified and the specific measures or actions that will be taken to address them [OECD/NEA, 2004]. The importance of this aspect is recognized by the agencies involved in the development of geological disposal programs. In the safety assessments made so far can be observed a clear trend can be observed towards a more extensive and structured consideration of the uncertainty issue, which involves two aspects:

- i) the control of the uncertainties in the overall development programme, through strategic provisions, at both the technical and organizational levels, in order to reduce and improve the basis for identifying, controlling and analysing the uncertainties; and
- ii) the actual identification and handling of the uncertainties in the safety assessment. These two aspects are related to the double title of this report: Uncertainty Management and Uncertainty Analysis respectively.

The issue of uncertainties has received a lot of attention on the part of both the Regulators and Developers, and has been a focus of international activities [OECD/NEA, 2004] [OECD/NEA, 2006]

The scope of the present report is to summarize the work done within Pamina WP1.1 on this topic.

2 Regulations and guidelines

Most of the national regulations and international guidance emphasize the importance of uncertainty analysis in the safety assessment.

In [IAEA, 2006] a requirement of geological disposal is that there must be sufficient confidence in the results of the safety assessment. This will be facilitated by identifying the features and processes that provide safety and also the features, events and processes that might be detrimental to safety, showing that they are sufficiently well characterized and understood. Where there is uncertainty, it will be taken into consideration in the estimation of safety. The understanding of the performance of the disposal system and its safety related features and processes will evolve as more data are accumulated and scientific knowledge develops. Early in the development of the concept, the data and understanding need to be sufficient to provide the level of confidence necessary to commit the resources to further investigation. Before the start of construction, during emplacement and at closure, the understanding must be sufficient to support the safety case in satisfying the applicable regulatory requirements. In establishing these requirements, it is important to recognize the multiple components of uncertainty that are inherent in modelling complex environmental systems and that there will inevitably be substantial uncertainties associated with projecting the disposal system performance. Furthermore, it is required that the post-closure safety case and supporting assessment identify and present an analysis of the associated uncertainties.







[SKI, 2001] requires that the following shall be reported (in the safety assessment): "how uncertainties in the description of the functions, scenarios, calculation models and calculation parameters used in the description as well as variations in barrier properties have been handled in the safety assessment, including the reporting of a sensitivity analysis which shows how the uncertainties affect the description of barrier performance and the analysis of consequences to human health and the environment".

Furthermore in the general recommendation on the former regulation is stated that *"these uncertainties can be classified as follows:*

- scenario uncertainty: uncertainty with respect to external and internal conditions in terms of type, degree and time sequence,
- system uncertainty: uncertainty as to the completeness of the description of the system of features, events and processes used in the analysis of both individual barrier performance and the performance of repository as a whole,
- model uncertainty: uncertainty in the calculation models used in the analysis,
- parameter uncertainty: uncertainty in the parameter values (input data) used in the calculations,
- spatial variation in the parameters used to describe the barrier performance of the rock (primarily with respect to hydraulic, mechanical and chemical conditions).

There are often no clear boundaries between the different types of uncertainties. The most important requirement is that the uncertainties should be described and handled in a consistent and structured manner.

The evaluation of uncertainties is an important part of the safety assessment. This means that uncertainties should be discussed and examined in depth when selecting calculation cases, calculation models and parameters values as well as when evaluating calculation results.

The assumptions and calculation models used should be carefully selected with respect to the principle that the application and the selection should be justified through a discussion of alternatives and with reference to scientific data. In cases where there is doubt as to a suitable model, several models should be used to illustrate the impact of the uncertainty involved in the choice of model.

Both deterministic and probabilistic methods should be used so that they complement each other and, consequently, provide as comprehensive a picture of the risks as possible.

The probabilities that the scenarios and calculation cases will actually occur should be estimated as far as possible in order to calculate risk. Such estimates cannot be exact. Consequently, the estimates should be substantiated through the use of several methods, for example, assessments by several independent experts. This can be done, for example, through estimates of when different events can be expected to have occurred.

Based on scenarios that can be shown to be especially important from the standpoint of risk, a number of **design basis cases** should be identified.

Together with other information, such as on manufacturing method and controllability, these cases should be used to substantiate the design basis such as requirements on barrier





properties.

Particularly in the case of disposal of nuclear material, for example spent nuclear fuel, it should be shown 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 shown that such a redistribution is very improbable.

The result of calculations in the safety assessment should contain such information and should be presented in such a way that an overall judgement of safety compliance with the requirements can be made".

[SSI, 1998] endorse the above referenced SKI's regulations, and specifies that "the different categories of uncertainties, which are specified there, should be evaluated and reported on in a systematic way and evaluated on the basis of their importance for the result of the risk analysis. The report should also include a motivation of the methods selected for handling different types of uncertainties, for instance, in connection with the selection of scenarios, models and data. All calculation steps with appurtenant uncertainties should be reported on.

Peer review and expert panel elicitation can, in the cases where the basic data is insufficient, be used to strengthen the credibility of assessments of uncertainties in matters of great importance for the assessment of the protective capability of the repository".

In France, [ASN, 1991] requests that uncertainty ranges be provided for the radiological consequences of the repository. In addition, sensitivity analysis should be carried out in order to identify priority areas for further effort, and to help in the assessment of the uncertainties affecting the results of the safety assessment. A very similar statement regarding the information to be provided in the safety assessment on the uncertainties is found in the Swiss HSK-R-21/f [HSK & KSA, 1993].

In Finland, the Government Decision on the safety of disposal of spent nuclear fuel (478/1999) requires that "the data and models introduced in the safety analysis shall be based on the best available experimental data and expert judgement. The data and models shall be selected on the basis of conditions that may exist at the disposal site during the assessment period and, taking account of the available investigation methods, they shall be site-specific and mutually consistent. The computational methods shall be selected on the basis of safety analysis, with high degree of certainty, overestimate the radiation exposure or radioactive release likely to occur. The uncertainties involved with safety analysis and their importance to safety shall be assessed separately". On this issue, the Radiation and Nuclear Safety Authority [STUK, 2001] specifies that the safety analysis shall include "uncertainty and sensitivity analyses and complementary discussions on the significance of such (unlikely disruptive events) impairing long-term safety phenomena and events which cannot be assessed quantitatively". And further on:" the computational methods shall be selected on the basis that the results of the results of the safety analysis, with high degree of certainty, overestimate the radiation exposure or radioactive release likely to occur.

In a similar way, the so-called "Franco-Belge" document [FANC et al, 2004] states that the consideration of uncertainties is a central element of a safety case. It can be undertaken, among other ways, by the use of conventional deterministic or probabilistic uncertainty evaluation tools.

In the UK [Environment Agency et al, 1997] the regulators have set out guidance on the





principles and requirements against which any application for authorisation of a radioactive waste repository will be assessed. It asks that the information provided by the developer includes, among other things: "...overall results from probabilistic risk assessments of the disposal system which explore the relevant uncertainties; suitable breakdowns of such risk assessments to show, for example, the probability distribution of doses and the contribution of important radionuclides; [and] a comprehensive record of the judgements and assumptions on which the risk assessments are based...". The expectation value of risk has to be compared with the regulatory risk target. The expectation value of risk is obtained by averaging the calculated risk from each probabilistic realisation (Annex 6).

In the US, detailed and comprehensive regulations have been implemented for the licensing of the WIPP disposal facility. These regulations provide the developer with a detailed, prescriptive path for the conduct of supporting assessments, and include the assessment period to be covered (10,000 years), limits on the cumulative release of radionuclides to the accessible environment, assumptions to be used in assessing particular Features, Events and Processes (FEPs), and requirements on the treatment of uncertainties. In addition to complying with radionuclide release limits, WIPP must comply with individual and groundwater protection standards [Galson D. A. et al, 2007].

Also in the U.S., the EPA and the NRC are currently developing the standards that will apply to the disposal of HLW and spent fuel in the potential repository at Yucca Mountain (proposed 40 CFR Part 197 and 10 CFR Part 63). The requirements of the proposed rule in the matter of uncertainties are described by the DOE-YMP in its contribution to PAMINA Work Package 1.2 as follows [Galson D. A. et al, 2007]:

"In the Supplementary Information published with the rule, the NRC has stipulated the application of a probabilistic framework for total system performance assessment (TSPA):

'Demonstration of compliance with the postclosure performance objective specified at § 63.113(b) requires a performance assessment that quantitatively estimates the expected annual dose, over the compliance period and weighted by probability of occurrence, to the average member of the critical group. Performance assessment is a systematic analysis of what can happen at the repository after permanent closure, how likely it is to happen, and what can result, in terms of dose to the average member of the critical group. Taking into account, as appropriate, the uncertainties associated with data, methods, and assumptions used to quantify repository performance, the performance assessment is expected to provide a quantitative evaluation of the overall system's ability to achieve the performance objective. (64 FR 8640)'

Note that the NRC not only anticipates that there will be significant uncertainties (proposed 10 CFR 63.101), but the NRC also requires the TSPA take into account uncertainties in characterizing and modeling the barriers (proposed 10 CFR 63.114). Furthermore, proposed 10 CFR 63.113(b) (64 FR 8640) requires a demonstration of compliance by calculating an expected annual dose, defined as follows:

'The expected annual dose is the expected value of the annual dose considering the probability of the occurrence of the events and the uncertainty, or variability, in parameter values used to describe the behavior of the geologic repository (the expected annual dose is calculated by





accumulating the dose estimates for each year, where the dose estimates are weighted by the probability of the events and the parameters leading to the dose estimate). (64 FR 8640)' "

In Canada, [CNSC, 2006] includes the following guidance on the treatment of uncertainty:

"The strategy used to demonstrate long term safety may include a number of approaches, including, without being limited to:

- 1. Scoping assessments to illustrate the factors that are important to long term safety;
- 2. Bounding assessments to show the limits of potential impact;
- 3. Calculations that give a realistic best estimate of the performance of the waste management system, or conservative calculations that intentionally over-estimate potential impact; and
- 4. Deterministic or probabilistic calculations, appropriate for the purpose of the assessment, to reflect data uncertainty.

Probabilistic models can explicitly account for uncertainty arising from variability in the data used in assessment predictions. Such models may also be structured to take account of different scenarios (as long as they are not mutually exclusive) or uncertainty within scenarios"

In the Netherlands a safety report has to show that risks and individual doses are below the regulatory limits. However, a license application will also include an EIS (Environmental Impact Statement), which follows more or less the ICRP principles for Radiation Protection, i.e.: (1) justification, (2) optimisation, and (3) compliance with limits. The EIS uses the safety report to show compliance. For optimisation the EIS needs more indicators to be able to compare with alternative options. Presently the only indicators are dose and risk, for which there are reference values and constraints (Annex 7).

The Czech State Office for Nuclear Safety (SUJB) issued in 2004 a methodological guide for compilation of a safety report in support of siting application for a radioactive waste repository. This guide addresses the evaluation of uncertainties stemming from insufficient knowledge and complexity of the natural environment (Annex 8).

With the notable exception of the U.S., where detailed requirements are set, the regulatory approach to treatment of uncertainties that many countries are taking is not prescriptive, and is defined through the publication of non-binding guidance or "expectations" with respect to scope and methods for performing the assessments, coupled with licensing procedures at local and national levels. For example, this approach has been discussed in the European Pilot Project [Vigfusson *et al*, 2007] and is also the way followed in Canada [Galson D. A. et al, 2007].





3 Terminology

3.1 Formally defined terms

There are not official or generally accepted definitions for some of the terms used in documents dealing with uncertainty analysis in the field of geological disposal. For these, working definitions or explanations on the way they are understood within WP1.1 of Pamina Project are discussed in section 3.2 below. Other terms are defined either in national regulations or in international references.

In [IAEA, 2007] *uncertainty analysis* is defined as an *analysis* to estimate the uncertainties and error bounds of the quantities involved in, and the results from, the solution of a problem. More specifically in the field of geological disposal uncertainty analysis is a component of the safety assessment that analyses how the uncertainties which affect the different elements (data, assumptions, etc.) of the assessment propagate along it and affects the uncertainty of (or conversely the confidence in) the results (the safety indicators).

Sensitivity analysis is defined by IAEA as a quantitative examination of how the behaviour of a *system* varies with change, usually in the values of the governing parameters. A more specific common meaning of this term is analysis to investigate the dependencies of the result of the assessment on the alternative input elements (data, assumptions...) and in particular the dependencies of the uncertainties of the results on the uncertainties of the input elements to the assessment.

The definition given for **risk** in the same reference is: The probability of a specified *health effect* occurring in a person or group as a result of *exposure* to *radiation*. In [SSI, 1998] the health effects considered are cancer (fatal and non-fatal) as well as hereditary effects in humans, *"in accordance with paragraphs 47-51 in Publication 60, 1990, of the International Commission on Radiological Protection"*. In quantitative risk assessment the risk associated with an exposure is the product: consequence of the exposure times the probability of occurrence, and the total risk is the sum of this product extended to all the exposures (sum of probabilities equals one). The consequence of the exposure is calculated multiplying the dose by a conversion factor.

A close concept to risk, as it combines dose and probability in an aggregated indicator is **expected dose**, which is the dose times the probability of its occurrence; this indicator is used in the Finnish regulations for the long term safety of geological disposal to set constraints for unlikely events (deep well, rock movement, glacial climate) [STUK, 2001]. In the U.S. Yucca Mountain Project, the regulations establish safety criteria in probability weighted doses over the full spectrum of expected future situations, which is the total expected dose. For a single unlikely event, this concept coincides with the Finnish guidance that has been referred to. In both cases, there is a constant ratio to risk (the conversion factor).

3.2 Working terms

As explained in the former section, some of the terms used in the field of uncertainty are not universal, or official definitions that are generally accepted are not available. Users usually define them in their documents as required. Relevant terms used within the WP1.1 of Pamina Project or by the organisations which have made developments of interest for this





task report are introduced below.

Risk dilution is an issue which has been discussed for long time. In [OECD/NEA, 1997] it is said that "this term is used to describe a situation in which an increase in the uncertainty of the input parameters of a model (while holding the mean of the distributions constant) leads to a decrease in the mean of an output quantity (...). If overestimation of uncertainty results in mean consequences being reduced, the unfortunate effect is that what appears to be a conservative step (overestimating the degree of uncertainty) leads to an overoptimistic assessment of mean system performance". In [OECD/NEA, 2004a] risk dilution is "an issue for both risk based and non-risk based approaches. The concept of an annual risk criterion (which can be expressed as taking 'the peak of the means') can lead to an apparent lowering of risk - risk dilution. One concern appears to be averaging the consequences of events with short duration but with uncertainty as to their time of occurrence. Using the 'mean of the peaks' (also termed "total risk") is one way to get around this problem (although currently no regulations provide guidance on this issue). The use of the mean of the peaks is comparable to the use of a dose criterion, which gives the same level of protection for all individuals irrespectively when they are exposed. This is also compatible with the concept of sustainable development in that allows the exploitation of natural resources at any time. The mean of the peaks approach can, however, lead to misleading results by effectively combining results from events that are in fact independent. Some countries have therefore decided not to take this approach".

The very title of this report refers to *uncertainty management* and *uncertainty treatment*, which are worthy of a definition. In [OECD/NEA, 2004a] the analogue concept of "risk management was interpreted as the whole sequence of risk assessment, decision making and consecutive actions that affect the realisation of the risk". The working understanding in WP1.1 is that the focus of uncertainty management is the strategy followed in the overall repository programme to control the uncertainties which may influence the performance of the long term safety functions of the repository system; it includes the full range of actions and measures taken in the stepwise repository programme. Uncertainty treatment is a subset of the former, and refers to the way uncertainties are handled in the safety assessment. This is in line with Annex 10, where it is stated that "uncertainty analysis is the analysis by different methods and tools that aims to quantify the uncertainty in the considered output variable (e.g. calculated doses or radionuclide fluxes)"...whereas "uncertainty management is the broader activity of deciding on the level of the disposal programme how to deal with the uncertainties, i.e. what measures have to be or will be taken in the disposal programme to systematically identify the uncertainties and decide for each of the identified uncertainties the way to treat them (e.g. reduction of uncertainties through additional design modifications or site and waste characterisation actions, conservative assumptions in assessments)".

Several terms appear in the classification of uncertainties made in different programmes. Uncertainties are often classified as **epistemic** which are knowledge-based and, therefore, reducible, and **aleatoric** uncertainties, which are random and irreducible. It has been claimed that there are very few purely aleatory uncertainties. This classification is not often very useful, as most uncertainties are a mixture of both types. The epistemic character, however, is dominant in most cases (Annex 3).

From a methodological point of view the classification of uncertainties is of special interest [Galson D. A. et al., 2007]:

1. Uncertainties arising from an incomplete knowledge or lack of understanding of the behaviour of engineered systems, physical processes, site characteristics and their





representation using simplified models and computer codes (for example, the incorrect establishment of initial conditions/boundary conditions, of the dimensionality or the level of resolution (discretisation) (Annex 10)). This type of uncertainty is often called "**model**" uncertainty.

- Uncertainties associated with the values of the parameter that are used in the implemented models. They are termed "parameter" or "data" uncertainties (e.g., for SKB "data uncertainty concerns all quantitative input data used in the assessment" [OECD/NEA, 2007])
- 3. Uncertainties associated with significant changes that may occur within the engineered systems, physical processes and site over time. These are often referred to as "**scenario**" or "**system**" uncertainties. [Galson D. A. et al, 2007]

The former classification is used by many organisations. But others apply somewhat different terms and/or nuances in the definitions. Enresa (Annex 2) and Posiva (Annex 9) utilize the term "**conceptual**" uncertainty which is related to the term model uncertainty referred to above. It is the same for SKB, for which "*conceptual uncertainty essentially relates to the understanding of the nature of processes involved in repository evolution. This concerns not only the mechanistic understanding of a process or set of coupled processes, but also how well they are represented in a possibly considerably simplified mathematical model of repository evolution*"[OECD/NEA, 2007].

Posiva uses the term "**numerical**" uncertainty instead of "data" or "parameter" uncertainty used by most organisations.

Some organisations prefer alternative terms to scenario uncertainty. Enresa uses the term "**system evolution**" uncertainty (Annex 2) which is similar, but emphasises the fact that given a scenario, which is affected by uncertainty in its characteristics, there are uncertainties in the way it influences the characteristics of the system. This view approaches SKB's term "**system**" uncertainty [OECD/NEA 2007] which "concerns comprehensiveness issues, i.e. the question of whether all aspects important for the safety evaluation have been identified and whether the analysis is capturing the identified aspects in a qualitatively correct way, e.g. through the selection of an appropriate set of scenarios. In short, have all factors, FEPs, been identified and included in a satisfactory manner?" [OECD/NEA, 2007]. In the same context, NRG uses the term "future developments".

Some organisations devise more distinct uncertainty classification schemes:

Andra distinguishes the following classes of uncertainties:

- Independent of the repository behavior (e.g.: waste inventory)
- Intrinsic characteristics of the repository components (this class may be related to parameter uncertainty)
- Affecting processes controlling repository evolution:
 - Affecting the prediction of long term behavior
 - Based on short term observations
 - Limited validity of models (this class may be related to conceptual uncertainty)
- Technological uncertainty: a) alternative operating methods (e.g. excavation





method), b) limited knowledge on the application of technologies in the underground

- *External events*: a) natural (tectonic, climate), b) human actions (anthropogenic effects, h. intrusion.
- Design provisions

NRI and RAWRA use the following classification (Annex 8):

- *Time uncertainty* we do not know the behaviour of barriers over thousands of years.
- **Structural uncertainty** we do not know the effect of some factors (temperature, radiation, microbial) on the behaviour of barriers.
- *Metric uncertainty* we do not know whether the physical of chemical data have been well determined.
- Translation uncertainty We cannot explain causes of some effects.

A "what if scenario" is generally understood to be a scenario that is not physically impossible, but outside the range of expected possibilities supported by scientific evidence [OECD/NEA, 2006]. Such scenarios are not expected, so they are not factored in the measure of performance of the repository system, but they are frequently used as a method of analysis to improve the understanding of the system and its evaluation. They can also be used as part of sensitivity analyses, particularly in deterministic studies. In the Belgian programme the term "what if cases" is preferred to imply that they are representative of calculation cases rather than resulting from a series of physically possible events (determining the scenarios).

Enresa uses the tem "**variant**" to refer to "what if" scenarios characterised by specific deviations from the scenarios considered in the evaluation, usually by defining a single modification of an assumption or model in the original scenario (Annex 2). In the Belgian programme, the term "**variants**" are used in a different way: they "are considered within a specific scenario such as the distinction made between different possible evolutions of climate (undisturbed natural evolution or greenhouse effect) in the expected evolution scenario" (Annex 10)

Upscaling is often cited as one of the causes of uncertainty in the safety assessment. Upscaling relates to either spatial or time extrapolation. It occurs in particular for:

- Transfer of data from a context to another (e.g. data from an experiment to the repository system)
- Attribution of data obtained at a point in space to a larger domain (e.g. site feature measured at a point extended to a model cell in a mesh)
- Extrapolation of short term observation to long timescales





4 Methodology

4.1 Uncertainty management

The overall strategy for uncertainty management can be synthesised by four words: **Identify**, **avoid**, **reduce**, **assess**. The safety strategies of the repository development programs in the different countries have numerous features that are relevant for uncertainty management, even if they are not explicitly intended for, or not exclusively for that purpose. In the contributions made on the topic of Uncertainty Management and Uncertainty Treatment within WP1.1 of Pamina (see annexes 1 to 10 to this document), it is possible to identify many such features. Also, in the case of SKB, uncertainty management measures are mentioned in [OECD/NEA, 2007]. Below there is a compilation of the relevant uncertainty management features that can be identified in the national repository development programs which explicitly refer to each feature as part of their uncertainty management scheme are mentioned; in many others such features are also accounted for implicitly

- Stepwise development process of the repository programme: at each step the uncertainties are identified, analysed and ranked: priorities are defined to systematically reduce and/or address remaining uncertainties in the next step of the programme (Enresa, Posiva, IRSN, NDA). In the case of Andra, safety is the driving objective of the programme from the initial stage.
- *Regulatory framework*, independence of the Regulatory Authority, openness and participation of multiple stakeholders in the development process, which introduce cross-scrutiny (Enresa, NDA)
- Long timescales of the project, from the initial planning phase to the closure of the repository, which provides opportunity for i) multiple opportunities for re-assessment of the acceptability of the repository ii) the involvement of different individuals (Enresa)
- *Robust repository concept* (i.e. low sensitivity to uncertainties), for example by the use of sound principles, as the multi-barrier and multi-function system, passive safety (Andra, SKB, NRG, Nagra, Enresa, SCK-CEN, ONDRAF-NIRAS).
- Flexibility of the repository development programme: i) to accommodate changes in the amounts and quantities of waste, ii) to deal with new site data, iii) to take decisions (in particular on technological issues) when sufficient knowledge is available, keeping alternative options available until a decision is needed. Flexibility provides room for inclusion of results from technical developments, new R&D results, and more detailed site understanding (SKB)
- Intrinsically sound repository components (e.g. use of reliable materials and technologies for EBS (Andra, SKB, NDA), excellence of site characteristics (Andra, SCK-CEN, ONDRAF-NIRAS)
- Specific design provisions to avoid or mitigate certain sources of uncertainty, and ample margins to counter their effects (e.g. avoiding problematic materials (IRSN), durable containers, limiting temperatures (SCK-CEN, Andra, Enresa, ONDRAF-NIRAS, SKB), compartmentalisation of the repository into zones to prevent interactions (Andra).





The discussions on this issue within WP1.1 of Pamina revealed a large consensus on the majority of the points above, but some organisations did not assign themselves to some of the points as they do not have yet clear views on them, or work in those areas has not been carried out yet in their programme.

4.2 Uncertainty treatment

The treatment of uncertainty in the safety assessment needs a systematic and structured approach clearly established in the project basic strategy and implemented following strict procedures. The whole process of uncertainty analysis has to be thoroughly reported. QA and expert review are essential for building confidence in the analysis.

The basic Strategies for handling uncertainty tend to fall into the following broad categories (Annex 6):

- 1. Demonstrating that the uncertainty is irrelevant, i.e. uncertainty in a particular process is not important to safety because, for example, safety is controlled by other processes.
- 2. Addressing the uncertainty explicitly, for example using probabilistic techniques.
- 3. Bounding the uncertainty and showing that even the bounding case gives acceptable safety.
- 4. Ruling out the uncertain process or event, usually on the grounds of very low probability of occurrence, or because other consequences, were the uncertain event to happen, would far outweigh concerns over the repository performance (for example a direct meteorite strike).
- 5. Explicitly ignoring uncertainty or agreeing a stylised approach for handling an uncertainty (for example the 'reference biospheres' approach developed by the IAEA BIOMASS project).

In France the management of uncertainties is at the centre of the safety analysis [Andra 2005] (Annex 1) "Qualitative safety assessment (QSA) methodology was developed for detailed consideration of FEPs in the Dossier 2005 Argile. The qualitative safety analysis is a method for verifying that all uncertainties in particular in FEPs and design options have been appropriately handled in previous steps of the analysis, thereby justifying post hoc, e.g., the selection of altered evolution scenarios. It also led to the identification of a few additional calculation cases and has, in principle, the potential to inform design decisions and the derivation of additional scenarios. Some uncertainties can have a direct influence on the confidence that can be had in a given safety function. For example, if the uncertainty about the permeability of the host formation is too great, this could call into question the performance of the function « prevent water circulation ». Uncertainty is the subject of a systematic study that identifies:

- which component is concerned by this uncertainty, with if relevant the effects caused by one component on another by means of a perturbation;
- which performance aspects of which safety function can become altered. A qualitative, but argued assessment, including the use of special calculations if relevant, is





conducted on the risk of a significant reduction in the expected performances ;

• _ if applicable, and if such information is useful, the time period involved"

Sensitivity analysis is a tool generally utilized in the uncertainty analysis to investigate the significance of the different uncertainties (WP 1.1 task report on Sensitivity Analysis). The main objective is to prioritize the uncertainties for future work. But the problem lies often in the characterisation of the uncertainties, which may prove to be extremely complex. So, it is sensible to try simplified strategies first to reduce the problem.

In some occasions it may be shown that a particular uncertainty is irrelevant because either it is unlikely, or because the impact is not significant. Both strategies may be used in combination. The later may be done, for example, by considering an extreme assumption for the uncertain element (the scenario, model or parameter), and verifying that the influence on the outcome of the assessment is not significant. This strategy must be applied carefully, making sure that the assumption made is conservative.

In other cases, the uncertain element may be represented in the assessment by a conservative assumption (e.g.: a conservative model, a conservative parameter value, a pessimistic scenario).

It may be interesting to combine detailed uncertainties in an uncertainty at a higher level, which encompasses all of them, for example, assuming the total or partial loss of a safety function. In the case of the engineered components, in a first assessment stage, uncertainties may be bound by postulating deterministic failures of these components with varying degrees of severity.

The approach to uncertainty analysis may be either essentially deterministic, as it is the case in many countries of continental Europe, or probabilistic, as it is the case in particular in the US and UK. In some probabilistic safety assessments each scenario is assessed separately, and its probability is not quantified (Annex 2). In fully probabilistic approaches, the probability is thoroughly considered and mathematically aggregated with uncertainty.

In both deterministic and probabilistic approaches, conservative assumptions are often made to deal with uncertainties. "In performance assessment modelling, it is often necessary to make a number of simplifying assumptions, either because insufficient data are available or the modelling capability cannot represent some feature of the system in full detail. The aim is to address issues as realistically as possible, whilst erring on the side of caution. Therefore, some simplifications involve taking a conservative view, i.e. assumptions are made such that radiological risk will tend to be over- rather than under-estimated. Conservative assumptions are often the best way of addressing issues without introducing unnecessary complexity into the models.

However, this approach of making conservative assumptions can sometimes lead to models which, although robust from a safety point of view, are physically unrealistic. Also, it is important to note that the probability that all parameters in a system take their most pessimistic values is, in general, negligible, so that a calculation that assumes this would give a significant overestimate of the consequence and therefore provide a poor basis for making decisions. In particular, when optimising the design of a repository, it is important to have as realistic a view of the repository system performance as possible" (Annex 6).

The methods used in the treatment of uncertainties in the safety assessments are, in





general, specific for each type of uncertainty. In the discussion which follows, the terminology used in [Galson D. A. et al, 2007] is used, but the actual classification of uncertainties made by the different organisations must be borne in mind.

4.2.1 Scenario uncertainty

Systematic scenario methodologies address directly the issue of scenario uncertainty, which is the focus of a different task report within WP1.1 ("Definition and assessment of scenarios"). A sufficient set of scenarios has to be defined in order to encompass the range of plausible future evolutions of the repository system.

One important issue for scenario analysis is comprehensiveness, in particular, the consideration of all relevant factors (FEPs). In many programmes, this is verified by using FEP databases (preferably site specific) in the construction of scenarios. The FEP database itself has to be comprehensive; diverse measures are taken to secure this: use of expert judgement, peer review, audit against international established FEP databases (e.g. the International FEP Database and the FEPCAT of OECD/NEA). In Enresa's assessments (Annex 2) several independent teams of experts constructed independent lists of FEPS relevant for the formation of scenarios. In Andra's methodology, scenarios are formed on the basis of systematic structured analyses of the phenomena affecting the repository system (PSARS) (Annex 1); the resulting scenarios are compared later with the results obtained with Qualitative Safety Assessment following a FEP database based approach; both approaches are developed by different expert teams.

In safety assessments one (or a few) scenario(s) is defined to describe the most likely evolution of the repository; this is called in different ways in the different programmes: reference scenario, normal evolution, base scenario, etc. This scenario(s) may be complemented by several cases which address alternative likely evolutions (e.g. in the Belgian programme, this is done for alternative future climatic evolution; these alternative cases are called "variants").

Several scenarios are usually selected which address the unlikely evolutions ("altered scenarios" in [Andra, 2005]) or are defined pessimistically to encompass the uncertainties in future developments. The latter receive sometimes names as "worst case scenario" (Annex 3) or "very degraded" scenario [Andra, 2005].

The definition of the scenarios needs several categories of information, each of which is the subject of uncertainty: i) the initial conditions, ii) the internal FEPs and the couplings among them, iii) the external FEPs, iv) the time scales where the various elements of the scenario definition are relevant.

Andra's "qualitative safety assessment (QSA) consists of identifying uncertainties and studying their influence on repository evolution, thus analyzing the limits of validity of the given scenarios" (Annex 1).

In [SKB, 2006] the purpose of the methodology is "the selection of a sufficient set of scenarios, through which all relevant FEPs are considered in an appropriate way in the analysis. The selection of scenarios is a task of subjective nature, meaning that it is difficult to propose a method that would guarantee the correct handling of all details of scenario selection. However, several measures have been taken to build confidence in the selected









set of scenarios:

- A structured and logical approach to the scenario selection;
- The use of safety function indicators in order to focus the selection on safety relevant issues;
- The use of bounding calculation cases to explore the robustness of the system to the effects of alternative ways of selecting scenarios, including unrealistic scenarios that can put an upper bound on possible consequences;
- QA measures to ensure that all FEPs have been properly handled in the assessment;
- The use of independent reviews". [OECD/NEA, 2007]

Sensitivity analysis is used to check the importance of the different elements that characterise the scenarios. "What if" scenarios are often used to analyse the significance of scenario attributes. In some programmes, bounding scenarios are defined.

It is important not only that all relevant factors are considered and appropriately represented. Their intensity and time of occurrence have also to be adequately described (Annex 10).

In general it may be difficult to quantify the probability of unlikely scenarios. It may not be necessary to quantify probability, for example, if the assessment shows that the consequences are acceptable. But in other cases that result in non-negligible consequences, quantification may be required to show compliance with the regulations. In a fully probabilistic approach the scenarios and scenario attributes are defined probabilistically.

In some cases, of extensive irreducible uncertainties, some scenarios are defined in a "stylized" manner, applying logical decisions, based on expert judgement or in regulatory guidance. This is especially the case for human intrusion scenarios. The same approach is often taken for the description of the biosphere in the different scenarios.

4.2.2 Model uncertainty

The quantification of the system evolution and performance require the use of models that allow its mathematical representation. This itself is a source of uncertainty, due to:

- Poor or incomplete knowledge or understanding
- Simplified or incomplete representation of the system or processes
- Human errors in the execution of the models (SKB [OECD/NEA, 2007])

One issue which has to be addressed is the influence between processes (i.e. couplings). The ignorance or misrepresentation of the mutual influences between processes is a typical cause of model uncertainty.

The handling of model uncertainties requires the use of structured procedures. Uncertainties are identified by assessing the level of knowledge achieved for the different models used in the assessment. This is usually done in the first instance by the scientific experts and by the safety assessment experts involved in the programme development, and in a second instance by external experts, peer reviews, etc. Comparison of laboratory, *in situ* experiments or natural analogues with *blind* simulation results, is an important way of testing





models and of evaluating the confidence in quality. The latter process of confidence building is often referred to as "validation". (Annex 10).

Several possible options are open to deal with model uncertainties:

- the use of good models i.e., verified, checked against experimental evidence, based on expert judgement, benchmarked, well documented.
- the use of alternative plausible models. As an example, in the U.S. Yucca Mountain Project, "a structured approach has also been established to take account of alternative conceptual models (ACM). If two or more ACMs show different subsystem performance, abstractions are developed for both and used in TSPA calculations to determine any differences in system-level performance. If there are significant differences, the options are to include multiple ACMs in TSPA with a weighting, or to consider a conservative choice of model" [OECD/NEA, 2004a].
- the use of conservative or bounding models. These may be highly simplified models, in some cases just parameters, for which the uncertainty may be defined (ranges, pdfs) (Annexes 3 and 9).
- in the case of highly complex external phenomena, by the use of stylized approaches (e.g.: SKB treatment of conceptual uncertainty for external influences, essentially through the definition of a sufficient set of scenarios and by state-of-the-art models) [OECD/NEA, 2007].
- some conceptual uncertainties (such as certain uncertainties on site characteristics) may be handled as alternative scenarios (e.g., alternative discharge zones, unknown discrete features).

Human errors in the handling of models may be avoided by a number of measures:

- Good documentation of the model and of the computer code where it is implemented (for example, specifying the domain of validity).
- Formal procedures to guide and control the use of the models in the safety assessment
- Comparing the results obtained with simplified models (e.g. "scope calculations")
- By the use of QA procedures, peer review, benchmark exercises, etc.

4.2.3 Parameter uncertainty

Parameter uncertainty is addressed in most if not all safety assessments, even in the programmes at an early stage of development. Parameter uncertainty may arise for different reasons:

- The value is dependent on conditions not well established
- Insufficient knowledge
- Data is dependent on circumstances external to the repository or on future decisions (e.g. inventory, waste characteristics, technological options)
- Inaccuracies in measuring techniques



- The value in the future may change
- Natural variability (aleatoric); values only known at discrete places that have to be extended to larger domains
- Measurements are taken in a context different to the repository system (e.g. in experiments)
- Not well established correlations between data

The treatment of parameter uncertainty in the safety assessment may be either deterministic or probabilistic. The choice of one or other approach is primarily dictated by regulations. When the regulatory acceptance criterion is expressed in terms of risk, or the probability and the consequences are required to be aggregated, the approach must be probabilistic. When the regulatory criterion is dose, a probabilistic approach may be useful in the analysis of uncertainties; i.e. it can give an indication of the range of the expected doses; deterministic based programmes often use probabilistic methods as a complement to deterministic ones [**OECD/NEA**, **2007**].

Probabilistic methods (e.g. Monte-Carlo) for data uncertainty (including spatial variability) process uncertainty characterised inputs (e.g. pdfs). In some programmes the number of uncertain parameters is reduced, in order to facilitate the calculations (Annexes 3 and 8), in particular to reduce the computing time; this is done through a screening process (Annex 7) where some parameters, for example those for which uncertainty is not influential and are not correlated with other uncertain parameters, are treated deterministically (e.g. they are assigned a conservative value). In some occasions the uncertainty of the parameters may be based on the data available (for example on the basis of statistical distributions), but very often expert judgment (formal or informal) is needed (WP1.1 task report on Criteria for input and data selection).

Care is needed to identify and avoid "risk dilution"; *this effect can be identified by comparing the peak of the calculated mean (dose or risk)* (for each point in time) *with the mean of the individual peaks* (irrespective of the time they occur) [Galson D. A. et al., 2007].

The results of probabilistic calculations are shown by means of different statistics: percentiles, peak value, mean, median, etc. In a fully probabilistic assessment, the results for the various scenarios are aggregated, considering their probability. The results may be presented in the form of risk, or expected dose (probability weighted mean dose). In German and US (WIPP Project) studies performed in the past, the complementary cumulated density function (CCDF) for the maximum was plotted. That means that the maximum output values of all runs, regardless of their time of occurrence, are evaluated together. Plots of maxima frequency density have also been used (Annex 3)

"The programmes in the US have played a significant role in the development and use of probabilistic methods for conducting PA. For example, PA calculations for the WIPP project involve using the results from a set of deterministic, process-level models to construct response surfaces that are subsequently used by a probabilistic, process-level code (CCDFGF) to estimate potential releases [DOE 1996]. Uncertainty in the process-level models is considered epistemic and is associated with the lack of knowledge about the precise values of the model parameters. This uncertainty is represented by sampling 300 sets of parameter values (using Latin Hypercube Sampling) for the parameters and running the models for each set. PDFs for each parameter are derived from data, where available, and/or by using subjective methodologies. The level of information on which to base the





assignment of the distributions of possible values varies greatly among the parameters. The level of knowledge is an important consideration in assigning both the shape and the variance of a distribution. When knowledge about parameters is small and these parameters have been identified by the regulator or modellers as potentially significant to the performance of the disposal system, a conservative approach is sometimes taken. Bounding assumptions have been made in these instances" [Galson D. A. et al. 2007].

In the U. S. Yucca Mountain project, internal and external (including international and regulatory) reviews of the earlier TSPA that supported the Site Recommendation considered that there was an inconsistent treatment of uncertainties across the disciplines feeding into the TSPA. The project's response to these reviews was to prepare an "Uncertainty Analysis and Strategy" document and guidance. The overall goal of the uncertainty strategy is an analysis approaching realism. However, the focus on a realistic treatment of uncertainty is not necessarily the same as a full understanding of realistic performance. It is therefore appropriate to use simplified models, broad uncertainties and conservatisms providing these are justified and explained. A team approach of scientists and analysts is a key element of the uncertainty strategy. A formal process has been established for selecting parameter values and distributions for TSPA. For uncertain parameters that are important to system performance, the goal is to represent the "full range of defensible and reasonable parameter distributions rather than only extreme physical situations and parameter values". [OECD/NEA, 2004a].

In the deterministic calculations the probability is not aggregated with the calculated indicators. In some occasions, calculations are just done for one parameter value (e.g. conservative values) and the effects of uncertainties are discussed in a more qualitative way. Conservative values cannot always be straightforwardly established (Annex 3). Often in deterministic approaches the calculation of the indicators is repeated for one single scenario using several sets of input data, intended to represent the domain of uncertainty. In this approach, the results are series of values of the calculated indicators, which are not aggregated each other. (Annex 5).

In some programmes, parameter uncertainty is handled explicitly at both a detailed (process) level, and at an integrated (e.g. global safety assessment) level. The traceability of uncertainties from individual elements to the overall safety assessment should be made transparent to build confidence in the Safety Case (Annex 9). In other programmes, parameter uncertainty is more informal at the process level (e.g., only "best estimated" or "conservative" calculations are made) and the formal comprehensive treatment of parameter uncertainty is reserved for the integrated safety assessment.

5 Applications and experience

There is extensive experience in the application of uncertainty analysis methods to safety assessment. In the most advance programmes, the analysis of uncertainty is documented in safety assessments made in compliance with legal requirements and submitted to the scrutiny of regulatory authorities. The NEA project INTESC [OECD/NEA, 2006] gives ample attention to the experience gained in the treatment of uncertainty in the safety case.

In France, the management of uncertainties is at the centre of safety analysis of the Dossier 2005 Argile. The QSA methodology was developed specifically for Dossier 2005 Argile. It was based on previous attempts and on the comments that these attempts generated,





especially from the 2003 NEA peer review of "Dossier 2001 Argile". The aim was to provide traceability in the management of uncertainties. The reader of Dossier 2005 Argile, and especially safety evaluators, have a direct access to a list of the uncertainties that have been managed in the dossier, explaining how they have been managed and what consequences they might have on safety. This proved useful when discussing the management of uncertainties with the various evaluators." (Annex 1). The approach of Dossier 2005 to uncertainty analysis is deterministic.

In Sweden," the current safety case is a preparatory step for a safety case in support of a licence application". The strategy applied for the management and treatment of uncertainty is documented in "several reports produced in the SR-Can assessment project (...) All these are primary references of central importance for the assessment and are published together with the SR-Can main report [SKB, 2006]

- Data report
- FEP report
- Initial state report
- Process reports
- Climate report
- Model summary report
- FHA report. (deals with future human actions)

Most of the calculations in SR-Can are deterministic. Probabilistic calculations are used essentially as a means of handling data uncertainty and spatial variability in the modelling of radionuclide transport and dose". [OECD/NEA 2007]

ONDRAF-NIRAS and SCK-CEN have a long experience in the application of probabilistic methods in uncertainty analysis. In the PAGIS safety assessment, in the late eighties, a first series of limited uncertainty analyses were conducted, mainly focussing on parameter uncertainty by making stochastic calculations, and on the analysis of a first set of scenarios, derived on the basis of expert judgement. In the SAFIR 2 report [ONDRAF/NIRAS 2001] more detailed uncertainty and sensitivity analyses were conducted and a first attempt in the direction of uncertainty management was made to discuss in a more systematic manner the different types of uncertainties and their impact on the level of confidence in the safety and feasibility of the studied disposal system and on the future activities of RD&D. SAFIR 2 led to a positive conclusion on the feasibility and safety, but also to the identification of the key remaining uncertainties and of the main priorities for the current RD&D phase of the disposal programme (Annex 10).

Enresa's probabilistic approaches to uncertainty analysis have been implemented in the safety assessments ENRESA 2000 and ENRESA 2003, for repositories in granite and in clay, respectively. Each scenario is analysed individually; the results are expressed in terms of mean dose. They meet in all cases they meet the dose constraint (10^{-6} Sv) proposed by the Safety Authority (CSN). There has been no attempt to quantify scenario probabilities (Annex 2).

In Germany, GRS has a long experience in probabilistic safety analysis, which was carried out for the Pagis study. In later studies it was applied in the same form and using the same tools, recently for the LAW repository near Morsleben (ERAM) and the experimental







LAW/MAW repository in the salt mine at Asse (Annex 3).

The probabilistic approach is used to address most of the uncertainties in NDA's post-closure assessments of the radiological risk from the groundwater pathway. Nirex's Generic post-closure Performance Assessment (GPA) does not consider time-dependencies explicitly. Rather, the possible variation of a parameter in time is included implicitly in the uncertainty (in probabilistic calculations) for that parameter. Some stakeholders have challenged this approach and dynamic models will be used in future assessments.

In the late 1980's the VEOS study (Safety evaluation of disposal concepts in rock salt) has been performed in the Netherlands, which used a scenario approach followed by a deterministic consequence analysis and several deterministic sensitivity studies. For some scenarios with a relatively high exposure the probability of occurrence was also calculated. In the early 1990's a generic probabilistic safety analysis of the Dutch generic reference disposal concept has been performed. The results obtained show that for well-designed disposal systems, quantitative use of uncertainty (e.g. by probabilistic analyses) generally leads to the observation that for all different scenarios regarded in the uncertainty study, the regulatory limits for dose and risk are met (Annex 7).

Due to the initial stage of deep disposal programme in the Czech Republic, the total performance assessments were based on simplified, deterministic models. Only the effect of limited number of parameters (e.g. solubility) has been also tested in probabilistic mode. It was concluded that both sensitivity or "what-if" deterministic analyses and probabilistic analyses could contribute to demonstration that all uncertainties have been taken into account in safety assessments. For calculating the uncertainty of migration parameters (such as distribution coefficients K_d), an approach that stems from chemical analytical measurement calculations has been applied (Annex 8).

6 **Developments**

The development of methods and tools to improve the treatment of uncertainty in safety assessments is actively pursued in practically all the national programs and is in the focus of international programmes. The Pamina Project RTDC2 "Treatment of Uncertainty" focus entirely on uncertainty methods and strategies. Participants had proposed activities for RTDC2 on the basis of the needs they have their national programmes.

In France, the different national and international peer reviews of the Dossier 2005 agreed that the QSA method appeared to be an interesting tool, and was quite efficient at managing uncertainties. However, it was recommended to better explain such a method, especially the link between QSA, Safety Functions, and PARS. It was also suggested to develop the QSA method ahead of the "definition and description" of the scenarios. On going work will include feed back on this methodology, and exchange on an international level, in order to consolidate for the future safety assessments that Andra will have to produce, not only for the on-going geological repository project, but also for other future projects or existing disposal facilities (Annex 1).

Also in France, a new version of the relevant safety rule [ASN, 1991] is currently in progress, to account for the developments accomplished in Andra's Dossier 2005. IRSN explores the possibility of deriving simplified models from the 3D model in order to perform probabilistic analysis. A probabilistic approach is judged by IRSN to be complementary to the deterministic modelling approach, which remains the reference approach. IRSN develops





studies aiming at evaluating the means for predicting the performances actually reached in situ by the engineered components which depend on the initial and real state of the components reached after the construction and the operational phase (Annex 5).

In Germany, it is planned to create a basis for more systematic uncertainty management. This comprises unique rules for establishing appropriate probability distribution functions according to the degree of knowledge, as well as applying standardised criteria for evaluation of the results (Annex 3).

In the U.K., NDA has an on-going programme of work to develop the treatment of uncertainty as the safety case is developed for the forward programme. NDA continues to keep a watching brief on developments in the treatment of uncertainty to ensure awareness of new methodologies and their possible application. For example, collaboration with Bristol University on the application of Bayesian Belief Networks to variant scenarios connected with climate change has been carried out recently. Future assessments will use a more sophisticated treatment of the time-variation of parameter values, rather than treating time variation within parameter uncertainty (Annex 6).

In the Czech Republic, NRI experience has shown that the best way to express uncertain data is using probability distribution functions (pdfs), but it is felt that this approach makes it difficult to explain the results in a simple way. For this purpose, it seems to be more convenient to apply variation sensitivity analyses. Therefore in future analyses it is proposed to use both probabilistic and deterministic approaches (Annex 8).

In Finland, Posiva is developing methods for a systematic treatment of uncertainty (Annex 9).

Ondraf-Niras is developing a new safety strategy methodology in order to treat the uncertainties (of all classes) in a more systematic and pragmatic way (Annex 10).

7 Conclusions

The regulations and guidance at both international and national levels identify the need for a systematic and structured management of uncertainties in the repository development programmes, and require their treatment in the safety case.

The national agencies and research organisations responsible for the repository development programmes have recognised the importance of these requirements, and have devoted since early stages in their programmes significant effort to develop and implement appropriate measures and methods to deal with uncertainties. Experience has been gained in the application of uncertainty analysis methods in safety assessments. In the most advanced programmes, the treatment of uncertainties in recent published safety assessments has reached a high level of maturity and comprehensiveness.

Both probabilistic and deterministic methods are available for uncertainty analysis. The choice between them is primarily driven by regulations. Many programmes consider that these approaches complement each other. More generally, in several programmes alternative methods are applied in parallel to increase the confidence in the results obtained.

Aspects deserving further efforts have been identified in the various programmes. They are




being actively pursued within national and international R&D programmes, in particular within Pamina RTDC2.

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9 Appendices

A1 ANDRA (France)

A2 ENRESA (Spain)

A3 GRS-B (Braunschweig, Germany)





- A4 GRS-K (Cologne, Germany)
- A5 IRSN (France)
- A6 NDA (United Kingdom)
- A7 NRG (Netherlands)
- A8 NRI, RAWRA (Czech Republic)
- A9 POSIVA (Finland)
- A10 SCK·CEN, ONDRAF-NIRAS (Belgium)



A1: ANDRA (France)





Part 3: Uncertainty management and uncertainty analysis

Appendix A1: ANDRA (France)





Project acronym:

PAMINA

WP1.1 OVERVIEW OF PAST EXPERIENCE IN UNCERTAINTIES MANAGEMENT

Updated version 26/11/07 Andra





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STRATEGY AND KEY ELEMENTS

This present contribution from Andra aims at giving an overview of methodologies that have been used by Andra in the framework of the Dossier 2005 Argile in the four topics selected by the steering committee: 1) safety functions, 2) scenarios, 3) safety indicators and 4) uncertainties management.

The first meeting hold in Amsterdam on June 12th, 2007 was an opportunity to review contributions and discuss them for the future workshop to be held in Paris in October. The present document completes the draft provided for the Amsterdam meeting and clarifies some points discussed during the October 2007 workshop at Andra. Its structure has been revised according to the DWG common structure.

The December 30, 1991 French Waste Act entrusted Andra, the French national agency for radioactive waste management, with the task of assessing the feasibility of deep geological disposal. The Basic Safety Rule RFS III.2.f of June 1991 [i], issued by the French nuclear safety authority, provides a framework for the studies to be conducted. The protection of man and the environment are to be demonstrated. Furthermore, studies should show the ability to limit potential consequences to a level as low as reasonably possible. The concept should include a multiple barrier system, and rely on passive repository evolution without institutional control beyond a given timeframe (500 years). The studies carried out within this framework are presented in the "**Dossier 2005 Argile**" (clay) [ii] and "**Dossier 2005 Granite**" [iii].

Primary References

In the present document, the «Dossier 2005 Argile» is used as reference. Primary references include the French Act and the series of reports submitted accordingly:

- The French Waste Act dated 30th December 1991 [iv]
- The French Safety rules namely RFS.III.2.f, guidelines [i].
- Synthesis Report, Evaluation of the Feasibility of a Geological Repository, Meuse/Haute-Marne Site (in English and French) [ii].
- Architecture and Management of a Geological Disposal System Report (TAG; C.RP.ADP.04.0001) (in English and French) [v].
- Phenomenological Evolution of the Geological Repository Report (TEP; C.RP.ADS.04.0025), (in English and French) [vi].
- Assessment of Geological Repository Safety Report (TES; C.RP.ADSQ.04.0022) (in English and French) [vii]

Other references such as the presentation made at the symposium hold in Paris in January 2007 [viii], and the INTESC questionnaire [ix] have been used when applicable.

Strategy and key elements

The feasibility assessment for the argillaceous site builds upon a number of key elements:





- Basic input: the inventory model of the waste and the geological site,
- Safety functions and requirement management,
- Technical solutions based on industrial experience,
- Reversible management and monitoring,
- Phenomenological Analysis of Repository Situations (PARS) [x] and detailed, coupled process modelling,
- Qualitative Safety Assessment (QSA) [xi] uncertainty management, and scenarios,
- ALLIANCES simulation platform and calculation results.

Although the process thus summarized may suggest a linear progression from basic input data to designing a "solution" and assessing its safety, the process is in fact highly iterative, with repeated feedback exchanged between the various processes (see Figure 1). In addition to the routine feedback common to parallel engineering, three main iteration loops have been identified since 1991, each corresponding to a major milestone of the program: License application for construction and operation of the underground research laboratory (in 1996), submission of the Dossier 2001 (in December 2001), and the recent submission of the Dossier 2005.



Figure 1 : Dossier 2005 Argile; three iterations loops since 1991 (1996, 2001, 2005)

In view of providing sound feedback to design, research and development and to determine residual uncertainties, the following tools have been carried out: **the functional analysis (FA)** [xii] to determine the safety functions and associated requirements – what do we want? -; the **Phenomenological Analysis of Repository Situations (PARS)** [x] providing a good scientific understanding based on scientific studies from surface and underground laboratory – what do we get? -; **the qualitative safety analysis (QSA) [xi]** managing uncertainties and the quantitative assessment [safety and performance indicators] including sensitivity analysis –. What is the impact of a given uncertainty (or set of uncertainty factors) on the robustness of the system? – And eventually: does the concept meet the safety/acceptability criteria?

The following sections of the document describe in more details each of those topics according to the sequence of the various stages of activities conducted in the dossier 2005 (see Figure 2).





UNCERTAINTY MANAGEMENT AND ANALYSIS

SECTION 1: BACKGROUND / INTRODUCTION

Following the operational phase, safety assessment must be conducted over a very long period of time (up to a million years). Uncertainties regarding the behaviour of the repository over such periods are significant. Feedback on the evolution of natural or artificial systems on time scales of hundreds of years is limited to archaeological analogues, or to natural analogues that in turn give access to periods representative of geological time scales. But this does not mean that these uncertainties cannot be mastered with a sufficient degree of confidence. They must be tackled in a very systematic way, their effects analysed and taken into account in assessments.

Uncertainties are not the same from one period to an other, nor the components of the repository or its environment that are considered. Thus, by way of example:

- In the near field, i.e. in the immediate environment of the repository structures, uncertainties regarding the behaviour of the materials and the rock are going to decrease over time, when thermal, mechanical and hydrological processes due to disturbance of the repository dwindle or reach equilibrium. However, the time of attaining equilibrium and the exact nature of this equilibrium are subject to uncertainties;
- Uncertainty regarding the surface environment and the surface layers of the geosphere will increase overall, especially when major climatic changes such as periodic glaciations are included in the assessment.

The assessment of a repository feasibility assumes that a sufficient knowledge of the behaviour of the repository components has been acquired, in particular, thanks to the composition of a large corpus of scientific knowledge and development of a repository architecture down to a sufficient level of detail, and taking into account unavoidable uncertainties when considering evolution over hundred of thousand of years. Over such timescales, no feed back is available other than by means of natural and archaeological analogues. This does not mean, however, that these residual uncertainties related to the long durations, specific to the dossier, cannot be managed with a sufficient degree of confidence:

- Provisions are taken with regards to the repository conditions which would allow overcoming uncertainty consequences: choice of a very stable geological medium hardly affected since its deposition (155 million years ago), compartmentalisation of the repository into zones to prevent interactions between various kinds of waste, use of simple materials whose behaviour is well-known.
- In addition, to ensure the control of uncertainties, safety is integrated upstream the design phase in order to orient the choices toward the most robust solutions with respect to a possible lack of knowledge.

Finally, uncertainties are systematically investigated, and taken into account in the safety assessment. Their potential effects are examined; particularly in qualitative safety analyses (see Figure 2).

To conduct that investigation, Andra implemented three complementary approaches





(FA/PARS/QSA) to synthesise the knowledge, describe the repository evolution and manage the uncertainties:

- Knowledge reference documents were made up in order to provide a complete view of the scientific understanding on the following studied components: geological medium, engineered materials, packages, etc. They describe indeed the state of knowledge, correlatively identify the lack of knowledge and thus contribute in determining the sources of uncertainty and orienting the actions to reduce them.
- Once a good level of knowledge is reached on each component and the global architecture is defined, the evolution of the repository over space and time is described as finely as possible: this is the purpose of PARS, which describes the phenomena (thermal, mechanical, hydraulic, chemical, radiological) and their coupling throughout the repository evolution and specifies the phases of this evolution from its construction up to 1 million years.
- The systematic work accomplished with APSS/PARS led to a list of uncertainties (on phenomenology, models, data, component characteristics...).



Figure 2: Representation of the various stages of the analysis.

SECTION 2: REGULATORY REQUIREMENTS AND PROVISIONS

Management of uncertainties is an important recommendation of the RFS III.2.f. Particularly for sensitivity analyses to be conducted, it is indicated:

...«Les analyses de sensibilité permettent d'identifier les points sur lesquels devrait porter en



priorité l'effort de définition (situations prises en compte), de compréhension et de hiérarchisation des processus mis en jeu (modèles) ou de caractérisation (paramètres) pour accroître la crédibilité des résultats des évaluations.

Elles contribuent à l'appréciation de l'incertitude sur les résultats des évaluations des expositions individuelles à partir des incertitudes sur l'ensemble des facteurs (scénarios, modèles, techniques numériques, valeurs des paramètres, ...) entrant dans la démarche ayant conduit à ces résultats.»

SECTION 3: KEY TERMS AND CONCEPTS

Prior to identification of uncertainties and their treatment, a general typology was drawn up. The classification of uncertainties forms part of the basis of their management (see for example [xiii]). The following were distinguished:

- Uncertainties regarding repository project input data, i.e. in this instance the packages' inventory data and characteristics, independently of their behaviour in the repository (e.g. uncertainty about quantities or the radiological and chemical inventory);
- Uncertainties regarding the intrinsic characteristics of a repository component. These are of several kinds:
 - they may be linked to inaccuracies in measuring techniques ;
 - they may also be tied to a number of variables that are not directly accessible to measurement, and for which reference is then made to data available in the bibliography, with an uncertainty concerning the relevance of their application, for example;
 - They may be due to the variability of the component in space with regard to a necessarily limited sampling. In particular, in the case of the geological medium, the data used are acquired at different scales and from a limited number of measurements. The information acquired must then be extended to larger spaces or volumes, while managing the changes in scale. This applies to characterizing the rock from samples;
 - They may be linked to the model underpinning the definition of a parameter that we seek to characterize. The characteristics of a repository component are only defined within the framework of a given model. Thus, the permeability of the geological medium, a medium whose structure is complex on the microscopic scale, corresponds to an overall property of this medium at the macroscopic scale with regard to water transfers. In some instances, if the model is too simplified to assess the physical reality, the associated variables may only be defined with a margin of error. This is one form of uncertainty regarding models, which links up with uncertainties regarding the processes themselves;
- Uncertainties regarding the processes governing repository evolution. Once the data is acquired concerning all the system components, and the phenomenological representation given in detail, it remains to understand and represent the way in which these various elements interrelate and act on the system's evolution. The complexity of the phenomena does not necessarily provide a detailed understanding of each





interaction and necessitates adopting an overall representation of the medium that best describes the operation of the system. Representation by a model is subject to uncertainties since it simplifies a more detailed representation of the phenomena. There is also some uncertainty here regarding the choices of models. This especially applies to coupled phenomena, which are generally harder to represent. Several kinds of uncertainties come into this category :

- those due to the necessity to predict long-term behaviour, sometimes up to a million years, based on observations that are generally made over much shorter periods ;
- Those due to the validity limitations of the models or to the existence of several models characterizing the same set of empirical findings. As example, are the uncertainties regarding the behaviour models of some waste packages, such as vitrified waste. In addition to measurement uncertainty regarding the model's parameters (the initial rate of dissolution, for example), there is uncertainty regarding the very nature of the phenomena governing the dissolution of the vitreous matrix;
- **Technological uncertainties**. At the feasibility stage, the technological measures to be implemented are not finalized, and there are choices between different solutions that may not have all the same consequences for the repository's long-term safety. Moreover, within the framework of repository management in stages, it is extremely unlikely that, even once the repository has been more fully defined; it should remain operated in the same way throughout its expected duration. It is therefore a matter of taking into account:
 - Uncertainties due to the variability of possible repository operating conditions, either because different techniques exist (e.g. excavation), or because the operating scheme may vary (order and rate of delivery of packages, duration of observation and reversibility phases). This variability is only an uncertainty in respect of safety analysis if it leads to indetermination regarding the initial state of the repository in the post-closure phase. If, for example, the variability of possible excavation techniques or the operating period cause indetermination regarding the extent or nature of the damaged zone, it is important to take it into account;
 - Those due to limited knowledge about the conditions of implementing a particular technology in an underground context. Since the concepts proposed by Andra use well-known, proven technologies, these uncertainties are very limited and only relate to very specific operations, which do not have their exact equivalent in other industrial sectors;
- External events. These form a special type of uncertainty regarding repository evolution. In general, a distinction is made between naturally occurring surface phenomena (climatic, tectonic events, etc.) which are, in principle, predictable but often subject to great uncertainty, and events due to human action (intrusion, anthropogenic effects) which are, in most cases, unpredictable after a reasonable lapse of time. These events are considered as to uncertainties, because of the disturbances that they cause. Partly conventional approaches are generally adopted to restrict the scope of the uncertainties to be taken into account. In accordance with the Basic Safety Rules III.2.f [¡Error! Marcador no definido.], it is assumed in particular that future human behaviour will be overall the same as today. However, it is possible to adopt a predictive approach, based on past evolution, for most natural phenomena. Even in this case, uncertainties regarding the far future should be taken into account.
- Design provisions as discussed in the topic of functional analysis (see chapter 3 of







TES for details [vii]).

SECTION 4: TREATMENT IN THE SAFETY CASE

METHODOLOGY

The management of uncertainties is at the centre of safety analysis of the dossier 2005 argile (Figure 2). It directs the design, participates in the definition of the normal evolution domain [xiv], and lays the foundation for risk analyses. It may therefore appear in all the topics discussed within PAMINA's project; however, more details are given in this topic.

A qualitative safety analysis (QSA) methodology was developed for detailed consideration of FEPs in the Dossier 2005 Argile [xi]. The qualitative safety analysis is a method for verifying that all uncertainties in particular in FEPs and design options have been appropriately handled in previous steps of the analysis, thereby justifying *post hoc*, e.g., the selection of altered evolution scenarios. It also led to the identification of a few additional calculation cases and has, in principle, the potential to inform design decisions and the derivation of additional scenarios. Some uncertainties can have a direct influence on the confidence that can be had in a given safety function. For example, if the uncertainty about the permeability of the host formation is too great, this could call into question the performance of the function «prevent water circulation». Uncertainty is the subject of a systematic study that identifies:

- which component is concerned by this uncertainty, with if relevant the effects caused by one component on another by means of a perturbation;
- which performance aspects of which safety function can become altered. A qualitative, but argued assessment, including the use of special calculations if relevant, is conducted on the risk of a significant reduction in the expected performances;
- if applicable, and if such information is useful, the time period involved.

The first objective is to identify whether the uncertainties are correctly covered by the normal evolution scenario (SEN), either in its reference version, or in the sensitivity studies considered. If some of the uncertainties are not, it must be confirmed that they would have little impact on the repository, or that they refer to very unlikely situations.

As a second stage, if the uncertainty is not covered by the SEN, the function(s) and component(s) that could be affected must be identified. A systematic component-by-component analysis is used in particular to identify the shared causes of the loss of several functions: for example, an incorrect assessment of the long-term behaviour of a material can affect all the components that contain it, even though these could have different functions. The qualitative safety analysis provides an assessment of the degree of independence of safety functions, by identifying the possible uncertainties affecting several functions.

The effect of taking each uncertainty into account is described (i.e. the behaviour of the repository if the worst-case value of the parameter in question was the actual value, or if the risk envisaged actually occurred), in terms of the repository's evolution. This is done on the basis of the functions that are likely to be lost. For example, if a series of uncertainties can call into question the function «regulate the pH in the vitrified wastes cells», the corresponding situation is described, i.e. the effects of an uncontrolled increase in pH. If the design can cancel this effect, or if this is taken into account in the SEN or in its sensitivity calculations, the analysis stops at this stage. If a safety function can be affected and the







evolution of the repository could start to diverge from normal, with a possible impact on other components, this effect is then specifically identified. If the effect is not covered by any situations in one of the four altered evolution scenarios (the seals failure scenario, the package failure scenario, the bore-hole scenario and a severely degraded scenario which radically lower performances of safety functions), then a new calculation case is created and quantified. Specific qualitative analyses of external events were also conducted [xv].

The qualitative safety analysis was conducted by Andra engineers who were not involved in writing the scientific documents. In this way, the safety analysis is given a certain degree of independence, since the people in charge of analysing the uncertainties and the possible altered situations (the safety engineers) are not the same as those who established the phenomenological plan for normal evolution.

The comparison between the FEPs databases and Andra's own analyses was an important exercise for the qualitative safety analysis, and provided supplementary information on several aspects, to finally end with consistency between the approaches [xvi]. It proved to be very useful to safety engineers in ensuring that no fundamental characteristic of the components and no phenomenological process likely to have an influence on the repository had been forgotten.

APPLICATION AND EXPERIENCE

The qualitative safety assessment (QSA) consists in identifying uncertainties and studying their influence on repository evolution, thus analyzing the limits of validity of the given scenarios [xi]. It systematically confronted the design options of each major repository component with the functional analysis, PARS and supporting simulation results. It makes it possible to highlight uncertainties significant with regard to safety. It then verified whether design options are robust in light of these uncertainties or not, the latter situation meaning uncertainties may affect the safety functions.

The QSA methodology was developed specifically for Dossier 2005 Argile. It was based on previous attempts and on the comments that these attempts generated, especially from the 2003 NEA peer review of "Dossier 2001 Argile" [xvii]. The aim was to provide traceability in the management of uncertainties. The reader of Dossier 2005 Argile, and especially safety evaluators, have a direct access to a list of the uncertainties that have been managed in the dossier, explaining how they have been managed and what consequences they might have on safety. This proved useful when discussing the management of uncertainties with the various evaluators. The uncertainties are (see previous section):

- Uncertainties on the initial data of the project as such waste inventory.
- Uncertainties on the characteristics of components as such measurement uncertainties, validity of the use of data taken from the literature, limitations due to changes of scales, limitations on the definition of the features (e.g.: the notion of Kd)...
- Uncertainties on processes as such validity of models used to represent them, validity of the use of these models over very long timeframes.
- «Technological» uncertainties covering the reliability of components, their implementation and their quality control before implementations such as a poorly manufactured container.





• External risks as such seismic event.

With regards to the Functional Analysis and to the PARS, the QSA consisted in a much more systematic identification throughout scientific documentation (PARS, Reference and conceptual notes) of uncertainties, by safety engineers who have not participated in the scientific work. The QSA analyses each uncertainty (on component's characteristics, its evolution, and its interaction with other components) that may either (i) affect its ability to perform a safety function, (ii) or have an influence on another component's ability to perform a safety function, or (iii) modify the component's environment in a way that could affect the way the component fulfils its functions. This analysis permits to check if the uncertainty is taken into account either by design or by the way the normal evolution scenario 'SEN' it represented. In the framework of the dossier 2005, it allowed to identify the uncertainties that were accounted for by the SEN and the related sensitivity studies. The uncertainties that are covered by design or sensitivity studies are not easy to define ex ante. QSA allowed for an ex post analysis, and the identification of residual uncertainties that thus needed to be addressed especially by the means of the altered evolution scenarios 'SEAs'. It therefore helped to check that the SEAs provided for an as complete as possible description of foreseeable altered evolutions. It also helped to define additional quantitative assessments by identifying sensitivity cases, and by shedding light on possible couplings of different uncertainties (see Figure 3).



Figure 3: Illustrative example of QSA application

Finally, the QSA offers an integrated vision of all uncertainties by taking into account the various types of treatment (qualitative, calculation results, and scenarios). In that context, a set of four "Altered evolution scenarios" (SEA) were developed to provide an understanding of the potential impact of unlikely future evolutions related to specific system failures : (i) partial or overall deterioration of seal performance, (ii) waste disposal packages failure





Part 3: Uncertainty management and uncertainty analysis

Appendix A1: ANDRA (France)

(WPD), (iii) human intrusion and (iv) strongly degraded safety functions. As an end calculation, results (radionuclides flows through barriers & end-of pipe impact) based on these SEAs and sensitivity cases within the SEN and SEAs make it possible to evaluate overall repository feasibility and robustness, with information on the contribution of each component/barrier to safety.

Chapter 6 of TES [vii] presents the results of the qualitative analysis of safety, which consists of identifying the uncertainties of knowledge and studying their influence on the repository's behaviour. It allows not only characterising more completely the bounds of the normal evolution domain, but also identifying the situations that are not included in the normal evolution domain. It also allows setting up an initial hierarchy of the uncertainties according to their importance with respect to safety.

Documentation consisted in 40 data sheets (one per component) gathered into a Level-4 technical document [xi]. Chapter 6 of TES presents the:

- Results of uncertainty collection (by theme: knowledge of the geological medium, packages, phenomena regulating the internal evolution of the repository, technology, external events).
- Summary of uncertainties (for every component, a list of uncertainties/events that must be taken into account).
- Construction of altered situations (the combinations of the different uncertainties are verified in order to determine if they may lead to situations that are not covered by the scenarios, provided that those combinations are relevant).
- Conclusions

Some uncertainties can lead to an evolution of the repository which is not desired and no longer satisfies the expected safety functions. Such evolutions must be highly unlikely. It is advisable if this serves the objectives of the safety analysis to add other evolutions defined purely for safety reasons, which have no likelihood of occurrence and are studied simply to learn about the repository's behaviour faced with an unexpected influence. The definition of these situations qualified as «altered» is presented in chapter 7 of TES. It should be noted that beyond the simple definition of the altered evolutions the analyses of uncertainties are useful to finalise the design and to make it more robust to uncertainties, or to give priorities to the research programme.

Altered – reference situations, in this case a failure of the repository's seal devices, a failure of packaging elements, as well as an intrusive bore-hole intercepting the repository and left abandoned, were defined in advance based on the feedback from previous dossiers (of Andra and its counterparts). Their phenomenology is described in the PARS called «PARS of altered evolutions» [xviii, xix, xx], which allow learning how the processes controlling normal evolution can be modified. The situations of altered evolutions derived from the recording of uncertainties are attached to these major reference situations in order to form the altered evolution scenarios, which call for a special performance calculation. The objective is to check whether the repository remains safe under these worst-case conditions and to obtain additional information on the behaviour of the repository's components.

The uncertainties are not of the same kind depending on the time periods, components or parts of the repository and its environment. These phenomena determine the timescales and physical extent data used to support the safety analysis. In that respect, when performing





QSA, it relied upon the segmentation of the repository in time and space (called PARS) as mentioned in topic 3. The methodological approach to define the timescales relied upon spatial fractioning according to the main repository components and segmentation into "situations" corresponding to the phenomenological state with associated uncertainties of part of the repository or of its environment during a given period of time. Timescales relate to the evolution of features that are important to safety (on which the performance of the safety functions depend: for instance time frames of hydraulic phenomena are defined based on how long it takes to resaturate the various components of the repository).

To consolidate a comprehensive qualitative safety analysis, the Agency relied on the «features, events and processes» databases available internationally, in particular the FEP 2000 database of the OECD/NEA [xxi] and FEPCAT [xxii]. The FEPs databases list «features, events and processes» that are in principle important for safety analysis, which is a different approach from that of qualitative safety analysis which studies the uncertainties relating to these same features, processes and events [¡Error! Marcador no definido.]. The gualitative safety analysis emphasizes the uncertainties, component-by-component and by function approach; a FEP can therefore appear in several parties of the qualitative safety analysis (see details in chapter devoted to uncertainty management). Establishing a link between each FEP and each part of the analysis requires going into detail of the gualitative safety analysis arguments, but did prove possible in practice, and useful for verifying and clarifying the qualitative safety analysis [xvi]. Furthermore, the FEPs are intended to cover all of the phenomenology that could be found in different safety analyses, conducted in different geological contexts, and some require being adapted to become applicable to the Dossier 2005. This adaptation could be done without major difficulties, only a few FEPs concerning phenomena that could not occur in the particular context of the Meuse/Haute-Marne site could be identified in the databases, and were not included in the qualitative analysis.

SECTION 5 LESSONS LEARNED

Knowledge/Experience gained with the application of uncertainties management in the context of safety assessment

Application of the Qualitative Safety Analyses appeared to be:

- A more systematic method to identify and manage uncertainties (comparison with FEP databases)
- The impact of uncertainties on the disposal system was described and associated with predefined scenarios (normal-evolution scenario and sensitivity studies, calculation case of an altered-evolution scenario already identified and its sensitivity studies).
- Helped to verify that scenarios were consistent with the state of the knowledge: in general, the normal-evolution scenario encompasses a very large number of uncertainties (either as reference or as sensitivity).
- Border-line cases or combinations of uncertainties that may lead to situations outside the scope of the normal-evolution scenario, because they correspond to very unlikely deficiencies.
- Helped to identify uncertainties that had not been managed by the scenarios and have





a significant impact on safety functions: definition of a new case.

- But such analyses have some limits:
- At that "qualitative" stage, it is not possible to say whether those hypotheses have a significant impact or not on radionuclide transfers.
- No conclusion on the importance of the uncertainty with regard to safety \rightarrow quantification (cf. altered-evolution scenario).

On going or planned projects

The different national or international peer reviews of the Dossier 2005 agreed that the QSA method appeared as an interesting tool, quite efficient to manage uncertainties. However, it was recommended to better explain such a method, especially the link between QSA, Safety Functions, and PARS. It was also suggested to develop such a QAS method ahead of the "definition and description" of the scenarios. On going work will include a feed back on this methodology, and exchange on international level, in order to consolidate it in view of the future "dossiers" Andra will have to produce, not only for HAVL but also for other future Project or actual Centre.

Concerning the HAVL project, on going update of the QSA documentation is foreseen to take into account evolution of concept, functional analyses and acquisition of scientific knowledge. They will aim at verifying management of uncertainties but also to consolidate the choice of scenarios to be quantified.

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A2 ENRESA (Spain)









Proposal/Contract no.: FP6-036404

Project acronym: PAMINA

Project title: PERFORMANCE ASSESSMENT METHODOLOGIES IN APPLICATION TO GUIDE THE DEVELOPMENT OF THE SAFETY CASE

- Instrument: Integrated Project
- Thematic Priority: Management of Radioactive Waste and Radiation Protection and other activities in the field of Nuclear Technologies and Safety

WP1.1 Uncertainty management and uncertainty analyses. ENRESA contribution

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- PU Public
- PP Restricted to other programme participants (including the Commission Services)
- Х
- RE Restricted to a group specified by the consortium (including the Commission Services)
- CO Confidential, only for members of the consortium (including the Commission Services)





1 Background and introduction

This document describes the experience of Enresa regarding the management of uncertainties in the Performance Assessment (PA) of HLW repositories in granite and clay. The methods and results presented correspond to Enresa's most recent Performance Assessments of spent fuel repositories: ENRESA 2000 [2] for a granitic formation and ENRESA 2003 for a clay formation [3].

2 Regulatory requirements

The acceptance criteria for radioactive waste disposal facilities was set in 1987 by the following statement of the regulatory authority (CSN): "to ensure safety individual risk should be smaller than 10⁻⁶yr⁻¹, that is the risk associated to an effective dose of 10⁻ ⁴Sv/yr". There are no specific requirements on the treatment of uncertainties.

3 Key terms and concepts

No systematic definition of the concepts related with uncertainties is done in [2] and [3]. The different terms are used with the common meaning in this field of knowlwdge.

4 Treatment in the Safety Case

4.1 Methodology

Enresa Performance Assessment exercises have been done for synthetic sites, created on the base of limited data available for the Spanish favourable areas. Due to the lack of a real site, data must be taken from the literature or be based on the limited information obtained during the site searching programme. This leads in general to defining wide ranges of values for most host rock parameters, to cover the different potential sites.

Near field barriers are better defined in the preliminary repository concepts. Although R&D programmes have already provided a significant amount of data, much uncertainty remains due to the open decisions on the final design and the fitting to the geological environment. As a consequence, for the near field models an enlarged range of data taken from the bibliography has been adopted, leading also to quite wide ranges of values of near field parameters.

Taking into account the stage of the Spanish programme, great uncertainties are unavoidable, especially in the properties of the site. But this can have a beneficial effect, because the large uncertainty ranges considered ensure that potential combinations of parameter values that would lead to high doses can be identified. Uncertainties will be reduced at later stages when site specific information become available and engineered barriers properties are better known.







4.1.1 Classification and treatment of uncertainties

Three different types of uncertainties are considered in Enresa's PA exercises [2] and [3]:

- System evolution uncertainty, related to the prediction of the future evolution of the barriers of the system and the Biosphere.
- Conceptual uncertainty, related to the incomplete understanding of the nature of the processes involved in repository evolution.
- Data uncertainty, due to the limited amount of data available and the variability of the different input parameters to the models.

These three categories are equivalent to the classification identified by NEA in [1] as commonly used in PA exercises: scenario uncertainty, conceptual model uncertainty and parameter uncertainty.

Each one of the three previous types of uncertainties has been dealt following a different approach:

- System evolution uncertainty. In addition to the Reference (or Normal Evolution) Scenario, other scenarios are defined and evaluated to study the effect on system performance of alternative assumptions on future system evolution.
- Conceptual uncertainty. Calculations are performed for different conceptual models and variants derived from the Reference Scenario.
- Data uncertainty. This uncertainty is considered through the use of probability distributions in the probabilistic calculations. The acceptability of results is assessed by comparing the average dose to the dose acceptance criterion (see question above).

4.1.2 Probabilistic and deterministic approaches

In Spanish Safety Assessment exercises the probabilistic approach is preferred, although deterministic calculations are performed too, taking the best estimate (most likely) values for the latter. Then, deterministic calculations may be considered realistic in general, but for uncertain favourable processes which, in general, are not considered (for example: the hindering by hydrogen build up of radiolytic spent fuel matrix oxidation is ignored)

Deterministic calculations are performed using highly detailed codes. The calculation chain is formed by a set of individual calculations with manual transfer of the results from one code to the next one. As a consequence, a complete deterministic calculation can take several days and requires a significant human effort.

Probabilistic calculations allow including explicitly the parameter uncertainties in the calculations. In addition, all the models used in the global calculation are implemented in a single input file for computer code (GoldSim) and a calculation requires little human effort.

The self-contained probabilistic models, together with the fast algorithms used in GoldSim allow performing many calculations in a short time period. As a consequence, sensitivity and uncertainty calculations are performed mainly following the probabilistic approach.





4.1.3 System evolution uncertainty

Since generic synthetic sites are used in Spanish Safety Assessments there are significant uncertainties regarding the future evolution of the system, which is analysed through different scenarios. Covering a wide spectrum of future evolutions is useful at the current stage of the Spanish programme because it provides information that can help for site selection. At later stages, when a site becomes available, the number of potential future evolutions of the system can be reduced.

In the Safety Assessment of a repository in granite ENRESA 2000 [2] the following scenarios were defined and analysed, in addition to the Reference Scenario, to address the uncertainty in the system evolution: Climatic, Geodynamic, Human Intrusion, Shallow Well, Deep Well and Poor Backfill/Sealing.

In the Safety Assessment of a repository in clay ENRESA 2003 [3] the following scenarios were defined and analysed, in addition to the Reference Scenario, to address the uncertainty in the system evolution: Climatic, Deep Well and Poor Sealing.

Both probabilistic and deterministic calculations are performed, depending on the models that were considered to be more appropriate to model the particular scenario. Doses in the different scenarios are presented together with the Reference Scenario and the dose criterion (there is not consideration for the probabilities of the scenarios). In all scenarios considered the calculated doses were well below the acceptance criterion, showing that consequences remain acceptable in the different possible future evolutions of the system.

The next figure presents the results of the probabilistic calculations performed in ENRESA 2000 [2] for different scenarios.







4.1.4 Conceptual uncertainty

Due to limitation in knowledge, there can be different alternative conceptual models to represent a given process. To deal with this uncertainty, calculations are performed with the different models in order to identify their relevance for the global system (for instance, two alternative models of matrix alteration are considered: time decreasing matrix alteration due to alpha radiolysis and a small constant alteration rate in reducing conditions in presence of H_2). Hopefully, progress in scientific knowledge will allow selecting the right model, and decrease the model uncertainty.

In ENRESA 2000 [2] and ENRESA 2003 [3] many variants of the Reference Scenario were analysed using alternative models when there were significant conceptual uncertainties: different canister durations, constant spent fuel matrix alteration rate instead of the alpha radiolysis model, simultaneous failure of all the canister instead of failure spread over a long time period,...

In ENRESA 2000 [2] and ENRESA 2003 [3] conceptual uncertainty and sensitivity analyses were treated together through a great number of variants of the Reference Scenario. No explicit distinction is done and all the variants are presented together. Results of ENRESA 2000 variants are shown in the next figure and compared with the Reference Scenario. Most of the variants correspond to sensitivity cases and only a few to alternative models (to deal with conceptual uncertainty).

In future evaluations Enresa intends to completely segregate conceptual uncertainty treatment from sensitivity analyses.



4.1.5 Data uncertainty

Before making the global PA probabilistic calculations a significant effort was done to quantify parameter uncertainty. This was a collaborative effort between the R&D groups and the PA team using data generated within Enresa's R&D programme, data taken from the

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bibliography and the results of process level calculations performed for the particular exercise (hydrogeological calculations, solubility limits determination,...).

Enresa's main approach to performance assessment is probabilistic, using the Monte Carlo method. Some of the advantages of the probabilistic approach are that parameter uncertainty is explicitly taken into account in the calculations and the uncertainty in the input parameters is transmitted to the results.

Probabilistic PA calculations were done following the classical approach as follows:

- A probability distribution function (pdf) is assigned to each parameter (solubility limits, distribution coefficients, water travel time,..) which value has a significant uncertainty. Constant values (not pdf's) are assigned to well known parameters, such as the radionuclide inventory and the dimensions of the barriers.
- The pdf's of all the stochastic parameters are sampled and a value is assigned to each parameter. With these values a complete calculation is performed, modelling canister failure, radionuclide release from the matrix, near field and far field transport and dose to a representative member of the critical group is calculated. This individual calculation is called a "realization".
- The previous cycle of sampling-calculation is repeated many times (100 times in [2] an 500 times in [3], for instance) and the same number of dose vs. time curves are obtained for each radionuclide and the total dose. Averaging over all the realization at each instant, a mean dose vs. time curve is generated and compared with the reference value of 10⁻⁴ Sv/yr.

The next figure shows the mean dose due a spent fuel repository in granite, calculated in ENRESA 2000 [2]. Mean doses take into account parameter uncertainty, and obviously depend on the probability distributions assigned to stochastic parameters.







The next figure presents the results obtained in the probabilistic calculations of the Reference Scenario of ENRESA 2000 [2]. In addition to the mean, 5% and 95% percentiles are represented as well as the peak and minimum values obtained in all the realizations at each instant.

The next figure shows that there is no realization leading to peak doses close to the reference value. Even in the worst realization there exists a factor 30 of margin. Obviously, this is a very good result, but in other exercises some realizations can get close to or even surpass the reference value. The identification of these problematic realizations would help to identify the parameters which uncertainty should be reduced and guide R&D efforts.

The next figure clearly shows that there is no need to further reduce parameter uncertainty in order to fulfil safety criteria, provided that parameter uncertainty has been properly bounded before the evaluation. This result is a strong argument to show that there is no potential combination of values of the uncertain parameters that could lead to unacceptable results, and no efforts to further reduce uncertainties are necessary. This is satisfactory for the current stage of feasibility studies. Nevertheless in future stages, in particular when the safety authorities and the public opinion need to be confronted and comforted, this is not enough. We think that it will be necessary to demonstrate that a strong scientific base is available, that the uncertainties are identified and properly managed, and that every reasonable effort to reduce them has been made (this would be a very long and gradual process, extended to he whole development process, until the final closure, and probably beyond).



While the mean doses and the percentiles depend on the shape of the pdf's assigned to the stochastic parameters, the maximum and minimum values are only a function of uncertainty ranges. The shapes of the pdf's are hard to defend and open to criticism, while uncertainty





ranges are easier to justify.

We think that a figure such as the previous one is useful to demonstrate that uncertainty ranges are acceptable, and the classes of pdf's assigned to the different stochastic parameters are not critical because:

- all the realizations will be between the minimum and the maximum realizations, no matter the pdf's used, and
- although mean doses are sensible to the class of pdf's used, the calculated values will be always well below the reference value.

It must be clearly stated that Enresa follows the common approach of using as acceptance criteria that "mean dose must be below the reference limit". It is not Enresa's intention to use as acceptance criteria that all the realizations should be below the reference value, but if this is the case it would be a strong argument to justify that remaining uncertainties are acceptable and do not compromise the safety of the system.

4.2 Related topics

Uncertainty analysis and sensitivity analysis are related topics. In the exercises already done by Enresa conceptual uncertainty and sensitivity analyses have been treated together through a great number of variants of the Reference Scenario. For any future Safety Case, Enresa considers that a clear distinction between uncertainty and sensitivity analyses must be done.

The topic "definition and assessment of scenarios" is closely related to the treatment of the uncertainties regarding the future evolution of the disposal system.

4.3 Databases and tools

Not applicable.

4.4 Application and experience

Uncertainty management and analyses in Enresa PA exercises in a granitic formation [2] and a clay formation [3] are presented in section 4.1.

Since no site has been selected in the Spanish programme, there are great uncertainties in all geosphere data. Uncertainties in near field barriers are smaller, but remain significant.

The Safety Assessment exercises for repositories in both granite and clay rock were done assigning wide ranges of values to most parameters, and considering different conceptual models and scenarios. Doses in the different cases were well below the acceptance criteria. For instance, none of the individual realizations of the probabilistic calculations in the Reference Scenarios leads to doses greater than 3% of the reference value (10⁻⁴ Sv/yr).

In the future, when more data (mainly site specific) become available, uncertainties are expected to decrease while remaining bounded by the uncertainties already considered. As a consequence, doses are expected to be bounded by the estimates already performed too.





4.5 On going work and future evolution

Enresa is involved in several tasks of PAMINA RTDC2 "Management of Uncertainty during Safety Case Development":

- Task 2.1.D Techniques for sensitivity and uncertainty analysis. Enresa will test new techniques and codes developed by JRC in repositories in clay and granite.
- Task 2.2.A Parameter uncertainty. Enresa will participate in an expert elicitation exercise to generate pdf's for the solubilities of selected radionuclides under the near field conditions of a repository in granite.
- Task 2.2.E Probabilistic Safety Assessment for Deep Geological Repositories. Enresa will test the capability of GoldSim to perform a handle the requirements of a fully probabilistic analysis of a repository in clay defined by Nagra.

5 Lessons learned

At a very high level of the Safety Case, we think it is necessary to delineate an integrated and consistent view on the way to implement an appropriate management of uncertainty and how to show it: requirements, methods and tools.

The evaluation of uncertainties must be clearly separated from the sensitivity analyses. Although the tools used for uncertainty and sensitivity analyses can be similar, both topics should be clearly segregated in the Safety Case.

Variability, mainly of the geological formation, needs to be addressed explicitly in the Safety Case. This is a particular class of uncertainty that requires the use of specific tools in order to be taken into account properly. Including the variability in the Safety Case through the parameter uncertainty can be acceptable at an early stage of a disposal program but is not satisfactory for later stages.

Another area which in our opinion needs developments is the change of scale. This is in particular the case for the modelling of the far field, especially in the integrated performance assessment, where parameter values for coarse models must be selected on the base of field data.

In the Safety Assessments already performed by Enresa a limited post-processing of the probabilistic results has been done up to now: only mean doses, some percentiles and minimum/maximum values have been used. No formal analyses of which parameters control the uncertainties in the results have been done, but it is considered an interesting topic for the future.

The development of a methodology to extract as much information as possible from the fully probabilistic calculations would be useful. In particular, a systematic approach to identify the parameters that control the uncertainty in the results (doses) would help to focus R&D efforts. In the past, Enresa took place in the NEA PSA Group, and sustained an important activity in that area, which led to the proposal of a large number of sensitivity methods. Nevertheless, they did not prove to be very useful in safety assessment exercise. We think it is important to have a new verification of the potential usefulness of those, or new sensitivity







analysis methods.

We, as users of probabilistic approaches, think that an important weakness is the definition of pdf's. We think this is an important field for improvement, and it is one of the reasons why we proposed a task on expert judgement elicitation. At a general level, in our opinion the methods to define pdf's are of high priority.

Uncertainty management is intimately linked to the issue of confidence. The main element is the existence of a sound scientific programme subject to QA principles. The progress in terms of scientific understanding and data shall have to be submitted to critical analysis at different levels (assessment team, collaborating experts, overview groups, peer reviews, safety authorities). The step by step processes also plays an important role here, as the long time frames assure that new people come in and have a fresh look to the different issues. The third aspect is the robustness of the system (this mean that the system is a) reasonably predictable and b) forgiving in case of deviation (i.e. not very sensitive to uncertainties). Regarding the formal uncertainty analysis, both methodological approaches and mathematical methods are of fundamental importance.

6 References

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A3 GRS-B (Braunschweig, Germany)





Part 3: Uncertainty management and uncertainty analysis Appendix A3: GRS-B (Braunschweig, Germany)





Project acronym: PAMINA

Uncertainty management and uncertainty analysis GRS contribution

Reference: 1.1.b Version: 2 RTDC: 1 Work package: 1.1 Author: D.-A. Becker Date of working paper: September 2007





1 Background/Introduction

There are two basically different ways to handle uncertainties. One is using conservative models and parameter values instead of realistic ones, making sure that the reality cannot be worse than the calculated results. The other possibility is to establish probability distributions for all uncertain parameters and to perform a probabilistic analysis with a big number of separate runs. The first approach can cause some problems as conservativity is sometimes hard to prove. Moreover, too much conservativity can result in a failure of the proof of safety. Probabilistic analysis is always to be preferred, as it allows for an assessment of the probability of a failure, as long as the uncertainties of models and input parameter can be properly quantified. This, however, is not always possible. Therefore, normally, both approaches are combined by using conservative models and parameters only where the uncertainty is hard to quantify, and then performing a probabilistic analysis.

Probabilistic uncertainty analysis, though not required by valid regulations, is a common means for assessing the outcome of a repository model and has been in use in Germany for more than twenty years. The procedure was performed already in 1988 in the PAGIS study for a HLW/SF repository in rock salt and is described in the report [1]. In later studies it was applied in the same form and using the same tools up to now, recently for the LAW repository near Morsleben (ERAM) and the experimental LAW/MAW repository in the salt mine Asse near Wolfenbüttel. The methodology for uncertainty assessment is approved. The main problems lie in identifying the essential uncertainties, finding the adequate probability distribution functions and correct interpretation of the results.





2 Regulatory requirements and provisions

There is no valid regulation in Germany that requires the application of uncertainty analysis in a performance assessment. The German Atomic Energy Act merely requires the safe disposal of radioactive waste. There is an old German guideline ("safety criteria for the final disposal of radioactive wastes in a mine"), originating from 1983, which is formally still valid [2]. Concerning long-term safety, it simply requires that "even after decommissioning radionuclides that could reach the biosphere in consequence of non-excludable transport processes from a sealed repository must not lead to individual doses exceeding the value given in the Radiation Protection Ordinance". This value is 0.3 mSv/yr and is valid for all nuclear facilities. There are no rules for management of uncertainties.

There is, however, a consensus in Germany that the mentioned guideline is outdated and should be revised soon. A first draft for a new version, proposed by GRS, is currently under intense discussion. Among other new regulations, it requires a probabilistic analysis. This paper is, however, a controversial matter and will be essentially changed before being accepted by the authorities. Therefore, it is not presented here. Nevertheless, it can be said that the future guideline is very likely to contain a probabilistic criterion the fulfilment of which can only be proved by a probabilistic uncertainty analysis.



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3 Key terms and concepts

In the following, the general problem of uncertainties in long-term safety assessments is described as it is seen by GRS (Braunschweig).

Aleatory and epistemic uncertainties

Basically, it can be distinguished between two different kinds of uncertainties which require their specific handling: Uncertainties that are due to physical imponderabilities or principally unforeseeable processes are called aleatory; uncertainties, however, that originate from our lack of knowledge about the nature are called epistemic. Epistemic uncertainties are those of physical parameters that are only insufficiently known. Such uncertainties can be principally reduced by additional measurements, improvement of measurement techniques or other investigations. Aleatory uncertainties, however, can neither be avoided nor reduced and have simply to be accepted as they are. An example for an aleatory uncertainty is the time of failure of a single canister. This depends on things like pitting corrosion due to the existence of microscopic fissures in the container material from the fabrication process or from mechanical impacts during the emplacement. Of course, one can argue that it is possible to reduce this uncertainty by optimising the canister fabrication and handling processes, but such measures would change the system itself and not simply the knowledge about it.

The adequate handling of uncertainties depends on their type. Aleatory uncertainties should be quantified as exactly as possible and their influence on the uncertainty of the results should be analysed. This uncertainty has to be accepted and taken into account in the safety case. A sensitivity analysis normally makes little sense for parameters that are subject to aleatory uncertainties. In contrast to this, if applied to epistemically uncertain parameters, sensitivity analysis can identify those parameters that should be analysed or measured more thoroughly in order to reduce their uncertainty.

In the practice of long-term safety assessments for final repositories, there are very few, if any at all, purely aleatory uncertainties. Most uncertainties are a mixture of both types, since there are random influences as well as lack of knowledge. The epistemic character, however, is dominant in most cases, and if it is not, like in the mentioned example of the canister failure time, it can nevertheless make sense to treat the uncertainty as if it were epistemic. The reason has been indicated above: Normally, there are possibilities to reduce even aleatory uncertainties by technical or constructional measures, and it might be helpful to identify influential parameters by sensitivity analysis. Therefore, GRS decided not to distinguish between aleatory and epistemic uncertainties and to treat all uncertainties as epistemic ones.

Kinds of uncertainties

The most important uncertainties in long-term safety assessment are parameter uncertainties. As explained above, it is always assumed that these uncertainties are epistemic, i.e. due to insufficient knowledge about the actual natural conditions. Parameter uncertainties can origin from poorly known properties of the host rock, unclear flow conditions inside the mine, lack of knowledge about chemical conditions, etc. Parameter uncertainties are relatively easy to handle because they correspond directly with quantifiable numerical uncertainties. In many cases, a conservative value can be given, but this is only





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possible if the influence of the parameter to the result is monotonic.

Another kind of uncertainties is model uncertainties. In some cases, it is not clear which model has to be applied to describe a specific effect. Such uncertainties can be due to improper physical knowledge of the process, insufficient accuracy of the available models, or the inability to predict the correct physical situation. Model uncertainties are also always assumed to be epistemic. They are more difficult to handle than parameter uncertainties as they are hard to quantify. Where it is possible to specify a conservative model, this is the most convenient approach. If, however, there is no model that can be proved to be conservative, the model uncertainty can be mapped to an artificial parameter with discrete values, each representing one of the possible models. This parameter can be treated like a normal uncertain parameter in a probabilistic analysis.

Scenario uncertainties are the third kind of important uncertainties in long-term safety assessments. Normally, a number of different scenarios are developed which are considered more or less probable. Scenarios are derived from a FEP (features, events, processes) analysis and comprise things like the temporal evolution of the near field, transport through the far field and exposition paths in the biosphere. Since the probabilities of many FEPs can only roughly be estimated, scenario probabilities are very uncertain. The usual method to handle these uncertainties is investigating several scenarios independently, including a worst-case scenario and a scenario that is assumed to represent the intended evolution. Another possibility is to calculate risks which include contributions from all scenarios, but this requires a proper knowledge of the scenario probabilities.


4 Treatment in the safety case

4.1 Methodology

This section describes how uncertainties have been handled within the long-term safety assessment studies of GRS (Braunschweig). The general procedure has been basically the same for more than 20 years. The examples in the following are taken from the ERAM study for the LAW repository in an abandoned salt production mine near Morsleben. This is one of the most recent and most detailed studies by GRS.

Scenario uncertainties have been treated, as mentioned above, by investigating a normal evolution scenario, a worst-case scenario, and a limited number of additional scenarios that appear interesting by some reason. A quantification of scenario probabilities and calculation of risks has never been performed so far. Model uncertainties have mainly been handled by using conservative models. In some cases, however, model alternatives have been switched by use of artificial parameters as described above. In such cases, the model uncertainty is mapped to a parameter uncertainty and can be treated in the same way. Therefore, in the following only parameter uncertainties are considered.

Identification of uncertain parameters

Not all parameters in a safety assessment are uncertain. Geometrical dimensions of containers, distances in the mine building or well-known material constants like the mass density belong to the parameters that are more or less exactly known. Others may be less well-known, but are likely to have little influence on the results and can also be considered certain. In cases of doubt the value is chosen conservatively. In the ERAM study examples of such parameters are the void volumes in the different levels of the mine, or the radionuclide inventories, which have been collected over decades and can in some cases only be estimated.

The number of parameters that are really treated as uncertain should be kept limited, in order to allow a manageable uncertainty analysis. If for parameter a clearly conservative value can be given that is not too far away from the most probable value, are preferably simply assumed to be certain. Particularly those parameters that are suspected to have a nonlinear or unclear influence on the calculation results are selected for an uncertainty analysis. In the ERAM study, these are 43 parameters, comprising things like global and local convergence rates, reference porosity, corrosion rates, gas entry pressure, initial permeabilities of seals, distribution coefficients and diffusion constants.

Bandwidths and probability distribution functions

Each uncertain parameter has to be assigned a bandwidth interval. This can be a difficult task, as, if chosen too small, the bandwidth does not come up to the real uncertainty and, if chosen too big, it could jeopardise the proof of safety. Therefore, the interval boundaries have to be fixed carefully and with as much expertise as possible.

The next step is defining a probability distribution function (pdf) for each uncertain parameter.

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There is no unique procedure for this task. So far, mainly three types of distributions have been used:

Uniform distribution: If a parameter is known (or suspected) to lie anywhere between the boundaries with no preferred value, a uniform distribution is applied. In some cases the interval is divided into sub-intervals with different but constant probabilities. This is sometimes called a histogram distribution.

Triangular distribution: If the parameter has a clearly preferred value within its interval but no other information is available, a triangular distribution should be chosen. It can be symmetric or asymmetric.

Normal distribution: If a preferred value and a typical deviation is known, a normal distribution should be chosen. From a mathematical point of view, a normal distribution extends to infinity on both sides, which is physically doubtful and numerically problematic. Therefore, an interval is defined also for these parameters and the distribution must be cut at the boundaries. Sometimes, it seems plausible to choose a normal distribution within a given interval around some mean value but the standard deviation is unknown. In this case, the standard deviation has to be calculated from the interval boundaries. It is common practice to take the boundaries as the 0.001- and 0.999- quantiles of the distribution, which corresponds to a bandwidth of 3.09 times the standard deviation to both sides of the mean. This is unchangeably fixed in the EMOS code package, which has been used for all GRS studies. Therefore, it is neither possible to choose an asymmetric normal distribution nor to define the interval boundaries and the standard deviation independently.

All distribution types, except the triangular distribution, can be applied either on a linear or on a logarithmic scale. If the interval spans more than one order of magnitude, a logarithmical distribution is preferred. This pertains to parameters like diffusion constants, distribution coefficients or permeabilities. If the interval is smaller than one order of magnitude, normally a linear distribution is adequate.

Deterministic parameter variations

In the normal procedure of a safety assessment study a reference case is defined for each scenario under consideration. Every parameter is assigned a reference value within its bandwidth interval, which is either considered the most probable value or a slightly conservative one. The first exercise to investigate the influence of the uncertainty of a parameter is a deterministic parameter variation. The parameter is varied between several discrete values within its bandwidth interval, normally the boundaries possibly a few additional values, while all other parameters are kept on their reference value. Comparing the results with those of the reference case and interpreting the differences in detail often yields valuable information about the influence of the parameter. This information, however, has a qualitative character und must not be misinterpreted. If the results hardly change under variation of a specific parameter, this does not necessarily mean that the parameter generally has little influence. The observed behaviour can be due to the specific situation that results from the reference values of the other parameters and can be totally different for another combination of values.

The variation of a single parameter, keeping all others constant, is called a local parameter variation. The word 'local' does not mean that the variation is very small but refers to the fact





that only one of the parameters is considered.

Probabilistic uncertainty analysis

For a quantitative determination of the uncertainty of the result of a model calculation, a probabilistic uncertainty analysis must be performed, varying all parameters within their bandwidths and regarding their pdfs at the same time. The model is run for a number of times, each with a new set of parameter values. A complete set of *n* parameter value sets is called a sample of size *n*.

The necessary sample size can be derived from accuracy requirements. In Germany, there is no official regulation so far, but criteria of 90/90, 95/95 or 99/90 are discussed. The first of these numbers specifies the minimum percentage of adherence to some safety criterion normally given in form of a limit; the second number is the statistical reliability of this statement in percent. A criterion of this type specifies the admissible number of limit exceedings, but does not say anything about the acceptable amount by which the limit is exceeded. It can be shown that, if the sample is randomly chosen and all calculated results remain below the limit, a sample size of 22, 59, or 230 is sufficient to prove the 90/90, 95/95 or 99/90 criterion, respectively. This does not depend on the number of parameters. The actual number of runs, however, has been essentially higher in most studies.

There are different sampling strategies. GRS has most often used a random sampling strategy because it guarantees a statistical independence of the parameter values, which is often required by the mathematics. Intended parameter correlations can be taken into account as well in the sampling as in the evaluation. In some older studies Latin Hypercube Sampling (LHS) was applied, which allows a better covering of the total bandwidth of each parameter.

For evaluating the results and assessing the uncertainty of a model calculation, several statistical measures like mean, median or maximum are calculated. This can either be done for the absolute maxima of all runs or a specifically interesting point in time. If calculated in small steps for the total model time, the statistical values can be plotted as time curves. Another curve that is valuable for the uncertainty analysis and has always been plotted in GRS studies is the Complementary Cumulative Distribution Function (CCDF). It represents the relative frequency of runs with absolute maxima above some value versus this value. Typically, this curve has an s-shape, starting at 1 with a relatively steep decrease in the middle region and a flat tail at the end, finally reaching 0. It allows a much better assessment of the adherence to some limit than a simple statistical criterion like those mentioned above. Very useful information can also be gained from scatterplots with one dot for every run, each showing the maximum value and the time of its occurrence. These plots show, on the first sight, the highest maxima as well as the most critical time intervals. Additional interesting information can be extracted if the dots are coloured according to some properties of interest. In the ERAM study, the dots have been coloured after the radionuclide responsible for the absolute maximum. The plots show very clearly which radionuclides are responsible for the earliest, the latest, the highest, and the most maxima for each scenario.

Probabilistic sensitivity analysis

Sensitivity analysis is an own topic within PAMINA, but since probabilistic sensitivity analysis



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is closely related with uncertainty analysis it is briefly addressed here.

On the basis of a probabilistic set of calculations a global sensitivity analysis can be performed, meaning that the sensitivity of the calculation result to individual parameters under consideration of the influences of all others is investigated. A sensitivity analysis requires a much higher sample size than an uncertainty analysis. On the other hand, the sample size is limited by the computing time. By this reason, in older studies the sample size was typically a few hundred, while in the ERAM study it was chosen to be 2000. Generally spoken, the sensitivity analysis is the more accurate, the bigger the sample is.

There are a number of different methods for probabilistic sensitivity analysis. One simple approach, named after Pearson, is to calculate the correlation coefficients between the output of the model and each individual input parameter. The higher the absolute value of the correlation coefficient is, the higher is the sensitivity to the respective parameter. A positive coefficient means that the result increases if the parameter does so, a negative value indicates an inverse correlation. Another technique is performing a linear regression and determining regression coefficients for each parameter. A high regression coefficient means a high influence of the parameter to the result. There are some more similar, but more sophisticated, methods. All these methods are linear, which means that they work best for linear systems. Since, however, the models for final repositories are typically very complex and non-linear, the use of these methods is limited. A possibility of improving their significance is to perform a rank transformation. This means that each parameter value as well as the output value is replaced by its rank in the ordered list of all values in the sample. The rank transformation makes many models, at least monotonic ones, closer to linear, but at the cost of losing the quantitative relevance of the results. So far, GRS (Braunschweig) has always performed a rank transformation in sensitivity analysis studies.

A somewhat different approach to sensitivity analysis is two-sample tests like the Smirnov test. The sample values of the parameter under consideration are divided in two groups, one containing the upper 10 %, the other the rest. If there is a significant difference between the results obtained with the two groups, the parameter is considered important.

During the last years, variance-based sensitivity analysis methods have increasingly attracted attention. Such methods use the statistical variance for calculating sensitivity measures that do not require linearity or monotonicity of the model and can be quantitatively interpreted, but need high sample sizes. The most general theory was given by Sobol, but the technique proposed by him is complicated and computational expensive. A more practical approach is the Fourier Amplitude Sensitivity Test (FAST), which is based on the idea to scan the parameter space periodically with individual frequencies for each parameter, and to recover the frequencies in the model output value by means of a Fourier analysis. It can be shown that the sensitivity measures calculated with FAST are the same as those proposed by Smirnov. The FAST method has not yet been applied by GRS in practical studies, but it has been tested for demonstration purposes. It could be shown that the FAST technique works and can yield valuable additional information, compared with a linear sensitivity analysis.

Linear as well as variance-based sensitivity analysis can be performed with the software tool SIMLAB which is planned to replace the statistical components of the EMOS package in future.





4.2 Related topics

The issue of uncertainty management is related to a number of other PAMINA topics:

the assessment strategy,

the safety approach,

analysis of the evolution of the repository system,

definition and assessment of scenarios,

safety indicators and performance/function indicators,

sensitivity analysis,

modelling strategy,

criteria for input and data selection.

4.3 Databases and tools

The EMOS code package used for the GRS studies automatically calculates three linear sensitivity measures on a rank basis (Spearman rank correlation, partial rank correlation, standardised rank regression), and the Smirnov test. The methods are applied to the maximum value as well as to a number of points in time that may appear interesting. The parameters are ranked after the calculated significance for each method, and then an average ranking is calculated.

Linear as well as variance-based sensitivity analysis can be performed with the software tool SIMLAB which is planned to replace the statistical components of the EMOS package in future.

4.4 Application and experience

The results of uncertainty analysis are usually presented in different forms. In all German studies performed in the past, the complementary cumulated density function (CCDF) for the maximum was plotted. That means that the maximum output values of all runs, regardless of their time of occurrence, are evaluated together. The cumulated frequency of maxima above some value is plotted against this value. This results typically in an s-shaped curve starting at 1 for very low output values and ending at 0 for very high ones. Another method of presentation is a histogram plot directly showing the frequencies of maxima lying in specific intervals. Both diagrams are shown together exemplarily for the ERAM study in Figure 1. It can be seen that two of 2000 runs yield maxima slightly above the limit.







Figure 1: Complementary cumulative density function (CCDF) and frequency density for the ERAM study (2000 runs)

A very illustrative way of presenting the results of a probabilistic analysis is shown in Figure 2 for the ERAM study. The absolute maxima of all runs are plotted in a scatter diagram versus the time of their occurrence. Additional information is provided by colour-coding the radionuclides that are responsible for the respective maxima. Only five different radionuclides appear in the diagram. The earliest maxima occur after a few hundred years and are caused by 90 Sr or 137 Cs, which are relatively short-lived. These maxima are due to the extremely pessimistic assumption that the whole mine is flooded instantaneously after repository closure. The most maxima are caused by 126 Sn and remain well below the limit of $3 \cdot 10^{-4}$ Sv/yr. At medium times there are some maxima caused by 14 C, at very late times 226 Ra as a decay product of 238 U becomes dominant. A few maxima at medium times are caused by 226 Ra from the inventory.







Figure 2: Times and values of maximum radiation exposures

4.5 On-going work and future evolution

It is planned to create a basis for a more systematic uncertainty management. This comprises unique rules for establishing appropriate probability distribution functions according to the degree of knowledge, as well as applying standardised criteria for evaluation of the results.





5 Lessons learnt

Uncertainties can be managed by using conservative models or values or by probabilistic methods. Both approaches should be applied as they complement one another. A probabilistic uncertainty analysis should always be performed since it is the only possibility to provide quantitative measures that can be checked against formal criteria. The sample size has to be oriented at the formal criteria to be held, as well as the requirements of the methods to be applied.

A sensitivity analysis is a very useful supplement to a pure uncertainty analysis and should always be performed. Deterministic parameter variations help understanding the system behaviour and provide a qualitative local sensitivity analysis. A global sensitivity analysis requires probabilistic techniques and should be performed in combination with the uncertainty analysis.

The methods for defining bandwidths and pdfs are not very systematic so far. Often they are defined by a quick expert guess. This is not satisfying. There should be a clear and transparent procedure which leads to a unique bandwidth and pdf under consideration of all available knowledge. The development and testing of such a procedure is a task of the next years.

The linear methods of sensitivity analysis, which have been applied exclusively so far, seem to be insufficient to analyse the system behaviour correctly. It is possible that they yield even misleading results. Therefore, variance-based methods should be tested in detail, the more as the computational powers of modern hardware allow increasingly big sample sizes. It has been showed that such methods can yield some added value. There is, however, no experience so far about necessary sample sizes or specific problems like the considering of statistical parameter correlations.





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Appendix A4: GRS-K (Cologne, France)

A4 GRS-K (Cologne, Germany)









Project acronym: PAMINA

WP1.1 Contribution of GRS Cologne to the topic "Uncertainty Management and Uncertainty Analysis"





Appendix A4: GRS-K (Cologne, France)

Section 1: Background / Introduction

In Germany, no specific legal regulation exists which explicitly demands an uncertainty management in the framework of a safety case. The present safety criteria of 1983 /BMI 1983/ are currently revised and a new regulation on safety requirements is expected to be issued in 2008. One basis of the regulation will be the GRS-document "Safety requirements for the disposal of high active wastes in a deep geological formation" /BAL 07/, in which also uncertainty management and uncertainty analysis is addressed. The following sections will outline the present regulatory expectations which have developed since 1983 and will take special notice of the approach proposed by GRS.

Section 2: Definition of terms and used concepts

According to /BAL 07/, it has to be distinguished between scenario, model, and parameter uncertainties, being aware that such a categorisation is always somewhat arbitrary, subjective and dependent on the chosen modelling and assessment approaches. No distinction will be required between so-called subjective and so-called stochastic uncertainties because a clear distinction is not always possible. The terms scenario, model, and parameter uncertainties are usually understood in the following way.

<u>Scenario uncertainties</u>: Uncertainties concerning the potential development of the repository system, the time of occurrence as well as the impact of different evolutions on the safety of the repository system.

<u>Model uncertainties</u>: Uncertainties related to the applicability of conceptual, mathematical and numerical models (including codes) to be used in the safety assessments as regards the reproduction and representation of safety relevant site characteristics and processes.

<u>Parameter uncertainties</u>: Uncertainties resulting from the natural variability of the repository system, from statistical inexactness, lack of relevant data, or insufficient knowledge.

Section 3: Regulatory context

Section 3.1: Regulations and guidance

In Germany there is no legal regulation which explicitly demands an uncertainty management in the framework of a safety case. However, the general requirement for a state-of-the-art implementation of the safety case would imply for instance the application of stochastic methods in the framework of the safety analysis as well as a transparent documentation of uncertainty management in general.

Section 3.2: Requirements and expectations

Although the present safety criteria of 1983 /BMI 83/ do not include any requirements with regard to uncertainty management regulatory expectations have developed since 1983 reflecting a demand for a more sophisticated management of uncertainty. Theese



Part 3: Uncertainty management and uncertainty analysis



Appendix A4: GRS-K (Cologne, France)

expectations which are expressed in /BAT 07/ are outlined in the following.

In accordance with the IAEA safety requirements /IAEA 06/, it is a common sight that a safety case should clearly discuss the sources of each uncertainty and the measures which have been taken to reduce it. Additionally, the impact of remaining, non-reducible uncertainties on the system's safety performance should be discussed. The safety case should explain how uncertainties relevant to safety can be coped with in future project stages by means of an appropriate research programme and management strategy.

Robustness is an important attribute of the repository system to balance the existing uncertainties. According to /BAT 07/, important criteria for achieving robustness are e.g. good characterisation and possibility of prognosis of the features over at least 10⁶ years, a sufficient distance from active tectonic areas, a sufficient repository depth, and a repository concept which is based on multiple safety functions with complementary contributions to the overall system safety.

There is a common view on how scenario, model, and parameter uncertainties should be evaluated:

- Scenario uncertainties. A well-structured procedure for scenario development should be applied in order to ensure that a comprehensive set of reasonable scenarios is considered. The procedure should make use of national and international FEP databases and should ensure that every decision within the development procedure can be retraced. The implementer has to demonstrate the scenario development in a traceable manner. Scenarios have to be assigned to the three classes "likely scenarios", "less likely scenarios" and "scenarios not to be considered in the analysis". Human actions which take place in awareness of the repository are excluded from the scenario analysis. According to /BAT 07/ human intrusion scenarios should be described in a stylised way during siting and designing phases of a stepwise repository development process and should not be part of the consequences analysis due to their limited predictability. For the scenario classes "likely scenarios" and "less likely scenarios" the implementer has to demonstrate compliance with the equivalent criteria standards.
- Model uncertainties. The conceptual, mathematical and numerical models (including codes) to be used in the assessments should be developed according to established quality assurance procedures. Verification, validation and confidence building should be carried out according to the state of the art in science and technology. If modelling assumptions or the presence and nature of certain processes are subject to doubt alternative assumptions should be explored. The robustness of the system against model uncertainties has to be demonstrated.
- Parameter uncertainties. For parameter uncertainty, if not reducible by reasonable effort, either conservative choices are to be made or reasonable parameter bandwidths and probability density functions are to be derived. In either case, decisions how to deal with remaining uncertainties have to be justified and documented in a traceable manner. A probabilistic uncertainty analysis has to be carried out for each likely or less likely scenario. The GRS-document "Safety requirements for the disposal of high active wastes in a deep geological formation" /BAL 07/ proposes to consider the consequences of parameter uncertainties for each likely or less likely scenario explicitly. If stochastic methods are applied a confidence interval of 95 % for the 95-percentile of the respective safety indicator has to be met.







Appendix A4: GRS-K (Cologne, France)

There is a broad consensus that assertions of conservatism should be faced more critically, since conservatism has to be evaluated with regard to the assessment endpoints and not by only considering the behaviour of sub-systems.

Section 3.3: Experience and lessons learned

Section 3.4: Developments and trends

Section 4: Analysis and synthesis

Presently, uncertainty management within the safety case is not subject to regulations in Germany. Nevertheless, present regulatory expectations are in accordance with the international safety requirements and recommendation as documented in /IAEA 06/ and the ICRP publication 81, respectively. A new regulation including requirements for uncertainty management is expected to be issued in 2008 on the basis of the safety requirements draft of GRS /BAT 07/.

Section 5: References

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- /IAEA 06/ The International Atomic Energy Agency: Geological Disposal of Radioactive Waste. IAEA Safety Standards Series No. WS-R-4, Vienna, 2006.
 ISBN 92-0-10570-9, http://www-pub.iaea.org/MTCD/publications/PDF/Pub1231_web.pdf





Part 3: Uncertainty management and uncertainty analysis

Appendix A5: IRSN (France)

A5 IRSN (France)





Appendix A5: IRSN (France)

Uncertainty management

1 Background/ Introduction

The safety case must clearly identify the level of confidence in terms of uncertainties that the implementer can allocate to the site survey, the results of scientific programme, the design concept, and the assumptions taken in the assessment and in the sensitivity analyses. The development of a safety case is sorely related to the necessity of assessing, quantitatively or qualitatively, the uncertainties and of finding a process for managing them. This process consists in improving the understanding of the evolution of the disposal system and should allow reducing the number of uncertainties or mitigating their effects.

Performance assessment and safety assessment allow analysing the uncertainties on the phenomenon or on the components associated to the disposal system, including quantification of the performance and comparing with regulatory requirements. Thereby, the performance assessment must be supported by demonstrating the relevance of the choice of the assumptions through sensitivity.

2 Regulatory context

ASN (Nuclear Safety Authority) and its technical support organisation IRSN (Institute for Radioprotection and Nuclear Safety) develop the regulatory framework for the safety of the deep geological disposal. This framework follows the principles and recommendations enact by the international organisations being technically competent (IAEA, ICRP, OECD).

In June 1991, the Basic Safety Rule 3.2.f (BSR3.2.f) was edited by ASN, IRSN and ANDRA, as guidance for defining the situations providing demonstration of safety through evolution scenarios. A new version of this rule is currently under progress in order to account for the notions and the safety approaches developed in the 2005 Clay Dossier edited by ANDRA.

Uncertainty management enters in the process of development of scenario, since the understanding of the phenomenon or the behaviour of the disposal components, and then the reduction of the uncertainties, allows identifying the normal evolution of the system.

a) Regulations and guidance

The strategy for managing uncertainties must be clearly explained, and particularly, the uncertainties should be identified and tracked. This strategy is a stepwise process, where every different step allows either reducing the uncertainties or avoiding their effects.

As described in the BSR III.2.f, the safety approach is based on an iterative process performed at each phase of the development of the disposal system concept. The safety assessments lead either to confirm or to review the features issued from the last step in order to build the safety case. As a matter of fact, at each step from concept phase to post closure phase, the iterative process should focus on the three complementary sides:





- verification of the favourable behaviour of the disposal components when no interactions are expected,
- evaluation of the disturbances caused by the interactions between the different disposal components and assessment of the consequences of those disturbances,
- assessment of the future behaviour of the repository and checking that individual exposure is acceptable.

Within the safety case, the implementers must demonstrate sufficient confidence in the data gathered by laboratory or in-situ experiments. An expert judgement, based on the understanding of the evolution of the system, should be done to justify the relevance in using data for quantitative assessments. This judgement may be interpreted as a qualitative treatment of uncertainties trending towards conservatism in the build up of the system model. Reducing uncertainties can be also achieved by designing the disposal system with appropriate materials that mitigate the effects of poorly understood events or processes and avoid their occurrence.

b) Requirements and expectations

To optimize the efficiency of the iterative process described above and to build confidence on the safety assessment, the safety case should provide a classification of the uncertainties. This classification should highlight the various phases of the safety approach concerning the disposal system from the concept phase to the post closure phase in order to organize the management of the uncertainties to be tackled.

The uncertainties linked to the data can be due either to the measurement (e.g. experimental tool validity...), to the lack of data, to the lack of knowledge of the processes associated to the expected data or to the variability in time and space of the values.

Uncertainties linked to base data can be categorised into:

- data associated to wastes: inventory (amount and volume) of waste generated depending on operating options which may vary in time, degradation kinetics
- geological/hydrogeological data: occurrence and possible role of discontinuities in radionuclide transfer
- sismicity

Uncertainties linked to perturbations induced by repository design relate in:

- thermal load
- gas generation (corrosion) and transfer in poorly desaturated media and complex components (seals or plugs)
- mechanical behaviour of rock and extension of EDZ around excavations (on long term mainly)
- transient state of cement, steel and clay components under repository conditions and ageing of engineered component

Uncertainties also arise from modelling due to the simplifications done on the repository design, on the events and processes understanding. The uncertainties related to the





Appendix A5: IRSN (France)

numerical model assumptions are mainly due to the computational tools and methods used for the calculations.

c) Experience and lessons learned

For the modelling aspect, the deterministic approach was suggested by the BSR3.2.f edited in June 1991 to assess the quality of the sites in terms of safety. It consisted of studying a limited number of situations representative of different families of events or sequences of events, such that the associated consequences were greater than those of the situations of the same family. This approach was based on a selection of events considered to be reasonably conceivable. It included the following steps: identification of the events liable to occur, classification of the events on the basis of their probability or origin (the deposit, man and natural processes), sorting the events by criteria covering their probability; the effects induced relative to those of other events with comparable probability, or the importance of the radiological impact combination of events to form scenarios and sorting of the scenarios.

The deterministic approach is based on the comparison of different scenarios including whether or not uncertainties within the range of realistic possibilities. Therefore, for each scenario, the data taken into account within the calculations should be classified in "best estimate" or "conservative" assumption. A third choice consists of classifying data used in calculations in a "pessimistic" assumption based on unrealistic value. The combination of those assumptions for every data conducts to build scenarios; in order to represent the expected performance of the system, to compare different assumptions, and to test the robustness of the system.

Uncertainties may be encompassed by adopting hypotheses increasing their effects and studying the consequences on global installation safety of a partial or total loss of function of the various repository components. In a first stage of the safety assessment, uncertainties over the evolution of containment performances of engineered repository components (packages, over-packs, seals) may be taken into account by simply postulating deterministic failures of these components with varying degrees of severity.

d) Developments and trends

Probabilistic approach allows exploring a wide variety of situations and highlighting contributors to dose. Stochastic calculations for the long-term evolution of the total-repository-system assessment may provide insight on the soundness of the deterministic cases dealt with for safety assessment. But because of the large amount of memory and computer time required for running full 3D radionuclide transport models used in deterministic approach, it seems unrealistic to couple this kind of model with probabilistic subroutines. It is the reason why IRSN explores the possibility of deriving simplified models from the 3D model in order to perform probabilistic analysis. Probabilistic approach is judged by IRSN to be complementary to the deterministic modelling approach, which remains the reference approach for integrating processes on large scale for calculating radionuclide migration and radiological impact.

IRSN develops studies aiming at evaluating the means for predicting the performances actually reached in situ by the engineered components. Because the long term behaviour of the repository, i.e. the performance of its components and the quantification of the containment capabilities, depends on the initial and real state of the components reached







Appendix A5: IRSN (France)

after construction and operational phase, large uncertainties arise from the difficulty to:

- Practically measure the level of quality which will be actually reached in situ for the various components of the repository accounting for the effects of natural variability of the material, and of the in situ manufacturing as well as the interactions with the repository environment.
- Derive from sampled measurements the possible in situ performance of the component that requires to develop or adopt sampling techniques coupled to probabilistic approaches in order to derive probabilistic distributions of characteristics

The development of methods aiming at bridging the natural variability of some measurable parameters (for specified characteristics defined by the design) and the variability of the foreseeable perfomances (in terms of mechanics, chemistry...) is a key issue in reducing uncertainties in the long term performance of the components.

3 Analysis and synthesis

The management of uncertainties doesn't only rely on accurate deterministic or probabilistic calculation, since the demonstration that calculations over 10⁶ years make sense will ever face a credibility gap. The iterative process allows reducing the lack of knowledge all along the different phases of the project. As the stepwise process involves, accounting for updated experimental data and better understanding of the disposal system evolution, pessimistic and conservative assumptions can be replaced progressively by best estimate data.

4 References

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[3] Avis de l'Institut de radioprotection et de sûreté nucléaire - Rapport DSU n°106 IRSN – Décembre 2005 (French only)

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A6 NDA (United Kingdom)









Project acronym: PAMINA

WP 1.1 – Uncertainty management and uncertainty analysis





Section 1: Background/ Introduction

An appropriate treatment of uncertainty is an important part of the development of a safety case for a geological repository for radioactive waste. NDA has an on-going programme of work to develop the treatment of uncertainty as the safety case is developed for the forward programme.

Section 2: Regulatory requirements and provisions

The regulatory requirements in the UK drive the developer towards a probabilistic approach in the treatment of uncertainty. An important regulatory requirement is the calculation of the expectation value of risk for comparison with the regulatory risk target.

Section 3: Key terms and concepts.

The main uncertainties identified in Nirex's Generic post-closure Performance Assessment (GPA) are as follows:

- Data uncertainty: near-field solubility, near-field sorption, effect of organic complexants on solubility and sorption, far-field sorption, inventory, biosphere factors, groundwater travel time, groundwater flux through repository.
- Model uncertainty: gas generation and migration, waste container corrosion, groundwater pathway models.
- Scenario uncertainty: evolution of the near field, criticality events, evolution of geosphere and biosphere (e.g. climate change).
- Uncertainty regarding human behaviour: start of post-closure period, human intrusion.

Section 4: Treatment in the Safety Case

Section 4.1: Methodology

Strategies for handling uncertainty tend to fall into the following broad categories:

- 1. Demonstrating that the uncertainty is irrelevant, i.e. uncertainty in a particular process is not important to safety because, for example, safety is controlled by other processes.
- 2. Addressing the uncertainty explicitly, for example using probabilistic techniques.
- 3. Bounding the uncertainty and showing that even the bounding case gives acceptable safety.
- 4. Ruling out the uncertain process or event, usually on the grounds of very low probability of occurrence, or because other consequences, were the uncertain event to happen, would far outweigh concerns over the repository performance (for





example a direct meteorite strike).

5. Explicitly ignoring uncertainty or agreeing a stylised approach for handling an uncertainty (for example the 'reference biospheres' approach developed by the IAEA BIOMASS project).

Section 4.2: Related topics

None

Section 4.3: Databases and tools

None

Section 4.4: Application and experience

The preferred treatment of a particular uncertainty will depend on the context of the assessment. To build confidence in the safety case, the treatment of uncertainty should aim to be as rigorous as possible. For example, it may be possible to argue that a nuclear criticality incident is very unlikely to occur (strategy 4 above), but if it can also be shown that even if such an incident did occur there would be no significant impact on safety (strategy 1), this is a more robust position, which should lead to greater confidence. In a PA, NDA-RWMD uses a combination of these strategies to manage the different types of uncertainty.

Uncertainties in data can be quantified in terms of 'probability density functions' (PDFs) that give the relative likelihood of different parameter values. The PDFs can be based solely on measured values, or, more usually, are generated at a formal elicitation in which measured values are supplemented by the judgement of suitably qualified and experienced experts on the basis of various research data, and can take into account any scarcity of data, uncertainty or bias from measurements.

With the uncertainty quantified as PDFs, a probabilistic assessment can be carried out using Monte-Carlo methods. In such an assessment, a computer model is run many times (each run is called a realisation) with different sets of parameter values. In each realisation, the values of the parameters are chosen at random from the PDFs representing the range of possible values. This is a probabilistic approach and it ensures that wide ranges of possible parameter values are considered within a performance assessment. Statistical analysis of the results of a probabilistic calculation can be used to explore the sensitivity of the performance measure e.g. risk to the uncertain model parameters.

The probabilistic approach is also consistent with current regulatory guidance in the UK, as an important regulatory requirement is the calculation of the expectation value of risk for comparison with the regulatory risk target. The expectation value of risk is obtained by averaging the calculated risk from each probabilistic realisation.

The probabilistic approach is used to address most of the uncertainties in our post-closure assessments of the radiological risk from the groundwater pathway.

The challenge is then to be able to communicate this understanding of the relative impact of the uncertainties in a transparent manner. It is often helpful to include other presentations e.g. deterministic sensitivity studies and 'What if?' calculations to improve the understanding





and communication of the results of a performance assessment.

In performance assessment modelling, it is often necessary to make a number of simplifying assumptions, either because insufficient data are available or the modelling capability cannot represent some feature of the system in full detail. The aim is to address issues as realistically as possible, whilst erring on the side of caution. Therefore, some simplifications involve taking a conservative view, i.e. assumptions are made such that radiological risk will tend to be over- rather than under-estimated. Conservative assumptions are often the best way of addressing issues without introducing unnecessary complexity into the models.

However, this approach of making conservative assumptions can sometimes lead to models which, although robust from a safety point of view, are physically unrealistic. Also, it is important to note that the probability that all parameters in a system take their most pessimistic values is, in general, negligible, so that a calculation that assumes this would give a significant overestimate of the consequence and therefore provide a poor basis for making decisions. In particular, when optimising the design of a repository, it is important to have as realistic a view of the repository system performance as possible.

Expert judgement often plays a role in handling data uncertainty and may be combined with the available empirical data to elicit a full data set or manage the consequences of uncertainty associated with the available data. It is not possible to avoid expert judgement when handling uncertainty in performance assessments. Systematic frameworks and modelling processes provide tools to help the experts, but there will still be situations where judgements need to be made.

Expert judgement is based on scientific/technical understanding and experience, supplemented with appropriate evidence. However, there is still scope for different experts to have different views and for two groups to reach different conclusions regarding an elicited data set, even when they are both using the same empirical evidence. Ideally such a situation, if it occurred, should be resolved by discussion between the experts, or with an independent third party if necessary. Disagreement between experts can be one of the main reasons for undermining public confidence in any decision-making process. This emphasises the importance of peer review throughout the performance assessment process and the value in maintaining flexibility in the modelling process to allow the testing of alternative view-points. Where there is more than one expert view, it may be best to conduct two parallel sets of calculations to determine the relative impacts of the conflicting views.

In documenting a performance assessment it is important to ensure that all data and model inputs are traceable. This will mean being clear on the extent and role of expert judgement, for example recording all expert input in an appropriate database that can be easily linked to the models generated, thus creating an audit trail for the impact of such judgements.

Section 4.5 On going work and future evolution

NDA continues to keep a watching brief on developments in the treatment of uncertainty to ensure that we are aware of new methodologies and the possible application. For example, we have recently carried out work with Bristol University on the application of Bayesian Belief Networks to variant scenarios connected with climate change.





Section 5: Lessons learned

The GPA does not consider time-dependencies explicitly. Rather, the possible variation of a parameter in time is included implicitly in the uncertainty (in probabilistic calculations) for that parameter. Some stakeholders have challenged this approach and hence it is proposed that future assessments may use a more sophisticated treatment of the time-variation of parameter values, rather than treating time variation within parameter uncertainty. This is something which will be addressed in future assessments.

Section 6: References

Nirex, Generic Post-closure Performance Assessment, Nirex Report N/80, 2003.

Nirex, Handling Uncertainty in the Phased Geological Repository Concept, Nirex Technical Note, 2006.





Part 3: Uncertainty management and uncertainty analysis

Appendix A7: NRG (Netherlands)

A7 NRG (Netherlands)



note

to :	Topic coordinator 'Uncertainty managements and uncertainty analysis	
from :	J.B. Grupa	Petten/015.017
date :	03 December 2007	
reference :	21951/07.86199 RE/JG/ES	
subject :	NRG Final contribution to topic 4 'Uncertainty managements and uncertainty analysis'	

Section 1: Background/ Introduction

In the late 1980's the VEOS study (Safety evaluation of disposal concepts in rock salt) has been performed in the Netherlands [1, 2, 3, 4]. The aims of this study were the evaluation of the post-closure safety of some possible disposal concept and the determination of relevant characteristics. VEOS used a scenario approach followed by a deterministic consequence analysis and several deterministic sensitivity studies. The analyses resulted in a number of release scenarios with estimated exposure. For some scenarios with a relatively high exposure the probability of occurrence was also calculated. The resulting risk defined as the product of this probability and the health effect of the exposure was below the risk levels set in neighbouring countries and the IRCP.

In the early 1990's a generic probabilistic safety analysis (PROSA, [5]) of the Dutch generic reference disposal concept has been performed. In this study a systematic approach to scenario selection has been used that ultimately leads to a set of selected scenarios that covers all aspects relevant for the long term safety. The method used a FEP catalogue to show comprehensiveness of the obtained set of scenarios.

Section 2: Regulatory requirements and provisions

In The Netherlands a safety report has to show that risks and individual doses are below the regulatory limits. However, a license application will also include an EIS (Environmental Impact Statement), which follows more or less the ICRP principles for Radiation Protection, i.e.: (1) justification, (2) optimisation, and (3) compliance with limits. The EIS uses the safety report to show compliance. For optimisation the EIS needs more indicators to be able to compare with alternative options.

Presently the only indicators are dose and risk, for which there are reference values and constraints. However, no uncertainty bands have yet to be provided for these indicators.

Section 3: Key terms and concepts.

Uncertainties in general

Uncertainties about the short term behaviour of the engineered barriers and the near host rock are one of the reasons to implement the option of retrievability in the disposal system. Combined with adequate monitoring systems this allows better control of the conditions directly after disposal of the waste, i.e. less uncertainty in these conditions. If the conditions







Appendix A7: NRG (Netherlands)

start to deviate from the expected conditions, this will be detected by the monitoring systems and, when necessary, mitigative actions can be taken.

Moreover, a 'fail safe' requirement has been suggested. In [6] the term 'fail safe' was defined as the quality of a repository concept that, upon loss of control of the repository, natural processes in combination with the design make the repository finally stay in a safe condition.

The driving forces for the development into a 'fail-safe' condition come from natural sources such as pressure-induced compaction of salt or the self-healing properties of clay. These properties of salt and clay tend to isolate the waste canisters and therefore further restrict the extent of radionuclide release and migration.

Monitoring and retrievability can reduce uncertainty in the system development on the relative short term; 'fail safe' can reduce uncertainty on the long term.

Uncertainties in a safety assessment

In a safety assessment the disposal strategy and the disposal design are assumed to be fixed. In this context 'uncertainty' is limited to the scenarios of time tables for waste disposal and (far) future developments of the disposal system (scenarios). Given this context, only minor changes in the design (layout and material choice) may follow from the uncertainty assessment.

In the PROSA [5] study, the dose calculations were done taking into account best estimate values for most model parameters. For parameters of importance in the dose calculations and with large uncertainties, probability density functions were selected. Probability density functions (PDFs) were used for a selected number of model parameters for the salt compartment, the groundwater "compartment", and the biosphere "compartment".

In summary, the PROSA approach included:

- Scenario selection
- Determination of the probability of the scenarios (including human intrusion)
- determination of the calculation model
- determination of the parameters and their probabilities
- dose calculation
- sensitivity and uncertainty analysis investigation of the effects of uncertainties of mode parameters on the calculated dose rate

The PROSA report provides the following definitions for relevant topics:

- Probabilistic method of proof based on probability and not on certainty
- Deterministic a classical method of achieving a goal by understandable logical arguments
- Stochastic process a process with an ordered (logical) progression which is constructed by using probability methods or functions





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Section 4: Treatment in the Safety Case

Section 4.1: Methodology

There are many model parameters that have to be addressed in a full Performance Analysis. In a full probabilistic analysis for each of these parameters probability density functions have to be determined and also cross-correlation functions. Without an initial screening procedure, the total number of probability density functions and cross-correlations is unmanageable.

In practise, the uncertainty in most of the model parameters does not contribute significantly to the uncertainty in the endpoints of the calculations, and does not correlate with the uncertainty in most other model parameters. This allows a screening procedure that reduces the number of parameters to be addressed in a probabilistic analysis to manageable proportions.

The initial screening is essentially an expert judgement activity. Since the model parameters are inseparable from the associated model, and the model is connected to a process, (feature or event), in PROSA [5] the initial screening can be combined with the scenario identification procedure. This allows a systematic documentation of the expert judgement rationales for all models and associated model parameters.

In the PROSA [5] study, the dose calculations were done taking into account best estimate values for most model parameters. For parameters of importance in the dose calculations and with large uncertainties, probability density functions were selected. Probability density functions were used for a selected number of model parameters for the salt compartment, the groundwater "compartment", and the biosphere "compartment".

The definition of the PDFs was mainly done on the basis of careful engineering judgement. For several parameters, mean values and their standard deviation were obtained from literature or, in case of a few parameters, from measurements. For the majority of the considered parameters the statistical distributions were lognormal. For other parameters, e.g. the dose conversion factors, the 50 percentile value was used from a stochastic distribution that was calculated separately.

In addition to the dose calculations with statistically-spread input parameters, a sensitivity analysis has been performed aiming to find the input parameter(s) having the strongest influences on the exposure, as well as an uncertainty analysis aiming to quantify the output variability.

One of the outcomes of the study was that in the model calculations at least four different sources of uncertainty could be identified:

- (1) Uncertainty in the conditions of the biosphere;
- (2) Uncertainty in the model description;

(3) Uncertainty in the point of release point of radionuclides into the different "sferes", and

(4) Uncertainty in the model parameters.





Section 4.2: Related topics

The whole set-up of the PROSA study was to perform a systematic approach of the safety of a salt-based repository. This included:

- Scenario selection
- Determination of the probability of the scenarios (including human intrusion)
- determination of the calculation model
- determination of the parameters and their probabilities
- dose calculation
- sensitivity and uncertainty analysis investigation of the effects of uncertainties of mode parameters on the calculated dose rate

The topic of uncertainty analysis was therefore mainly addressed in the derivation of the scenarios and the determination of the parameters and their probabilities.

Section 4.3: Databases and tools

A large amount of data were applied in the Dutch studies. Sources of the model data were previous studies, engineering judgement, or, in some cases, measured values.

As already mentioned in Section 4.1 a full probabilistic analysis covers the determination of probability density functions each of these parameters also cross-correlation functions. Without an initial screening procedure, the total number of probability density functions and cross-correlations is unmanageable.

The initial screening is essentially an expert judgement activity. Since the model parameters are inseparable from the associated model, and the model is connected to a process, (feature or event), in PROSA [5] the initial screening can be combined with the scenario identification procedure. This allows a systematic documentation of the expert judgement rationales for all models and associated model parameters.

Section 4.4: Application and experience

No practical applications and experiences have yet been implemented in the Netherlands.

However, the basis of the methodology as applied in the PROSA study seems the way to go in the Netherlands, i.e. a systematic approach as bulleted under 4.2, which includes sensitivity and uncertainty analysis.

Arguments used in the safety assessment to support the conclusion that the repository is safe in the presence of uncertainties often lead to discussions about 'What is safety?" and "What is safe?", rather than to the conclusion that the repository is safe.

For well designed disposal systems, quantitative use of uncertainty (e.g. by probabilistic analyses) generally leads to the observation that for all different scenarios regarded in the uncertainty study, the regulatory limits for dose and risk are met.





Section 4.5: On going work and future evolution

Ongoing work in the Netherlands on this topic is presently not included in a national program. Activities and research mainly takes place in EU-funded FP programmes.

We expect that the PROSA procedure for identifying scenarios will be extended by the application of 'safety functions', and therefore also of safety/performance indicators in future safety studies.

Section 5: Lessons learned

Uncertainty analysis is a sound scientific ingredient of a safety assessment. A probabilistic analysis gives additional endpoints such as total risk or percentile values of dose rates. It can also help to identify parts of the disposal system that are robust as one of the characteristics of robustness is that it will not contribute significant to uncertainty in the endpoints.

Confidence, or trust, or acceptance, are primarily not provided by uncertainty analyses.

Presently the results of uncertainty analyses are not very helpful in showing robustness of the disposal system. This is probably due to the fact that only dose and risks were assessed, and the method used draws the attention to the parts of the system that dominate the uncertainty, i.e. the less robust parts of the system. Additional performance indicators as endpoint of an uncertainty analysis and a different look at the results of the analysis may help in showing robustness of the system.

Section 6: References

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A8: NRI, RAWRA (Czech Republic)









Proposal/Contract no.:	FP6-036404
Project acronym:	PAMINA
Project title:	PERFORMANCE ASSESSMENT METHODOLOGIES IN APPLICATION TO GUIDE THE DEVELOPMENT OF THE SAFETY CASE
Instrument:	Integrated Project
Thematic Priority:	Management of Radioactive Waste and Radiation Protection and other activities in the field of Nuclear Technologies and Safety

WP1.1 Uncertainty management and uncertainty analysis. NRI contribution

Jiri Landa, Jan Vejsada, Antonín Vokál

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Х

- PU
 Public

 PP
 Restricted to other programme participants (including the Commission Services)

 RE
 Restricted to a group specified by the consortium (including the Commission Services)

 CO
 Confidential only for members of the consortium (including the Commission)
- CO Confidential, only for members of the consortium (including the Commission Services)





Czech waste disposal programme and uncertainty management

Czech disposal programme is coordinated by Radioactive Waste Repository Authority (RAWRA), which established QA system, forcing all research and design institutions, taking part in DGR programme, to have their own QA system. This QA system can help decrease primarily "metric" uncertainty connected with measurements and documentation of data.

Regulatory criteria concerning uncertainty

Czech State Office for Nuclear Safety (SUJB) issued methodological guide for compilation of safety report for application for permission to locate radioactive waste repository (SUJB, 2004). This report includes also part focused to evaluation of uncertainties stemming from insufficient knowledge and complexity of natural environment. The following uncertainties were mentioned:

- Host rock environment
 - Insufficient description of host rock environment, e.g. fractures location
 - Effect of collector communication
 - Fluctuation of groundwater level
- Contaminant properties
 - Interaction of contaminants with other repository elements
 - Change of contaminant properties with change of environment, e.g. change of solubility with change of pH or change of redox conditions
- Limited application of data obtained
 - Insufficient mathematical model
 - Errors in sampling of data and chemical analyses
 - Errors of laboratory experiments, e.g. sorption or leaching experiments

It is required to include the uncertainty in input parameters of calculations or estimates, i.e. to express input data in the form intervals.

Conclusions of the safety assessments

Due to the initial stage of deep disposal programme in the Czech Republic, the total performance assessments were based on simplified, deterministic models (PAGODA, MIVCYL, GOLDSIM) (Vokál, Vopálka, 2003). Only the effect of limited number of parameters (e.g. solubility) has been also tested in probabilistic mode using GoldSim software. It was concluded that both sensitive or what-if deterministic analyses and probabilistic analyses could contribute to demonstration that all uncertainties have been taken into account in





safety assessments.

Types of uncertainties

In the Czech programme, the following uncertainty types have been discussed in previous reports (e.g. Vokal et al., 1996):

- Time uncertainty we do not know the behaviour of barriers in horizon of thousand of years.
- Structural uncertainty we do not know the effect of some factors (temperature, radiation, microbial) on the behaviour of barriers.
- Metric uncertainty we do not know whether the physical of chemical data have been well determined.
- Translation uncertainty We cannot explain causes of some effects.

It was concluded that the best way to express data is using probability distribution functions (PDFs), but it is felt that using the probability distribution functions, it will be difficult to explain the results in a simple way. For this purpose, it seems to be more convenient to apply variation sensitivity analyses. Therefore in future analyses both probabilistic and deterministic approaches were proposed to be used.

The approach of calculating the migration parameter's uncertainty

For calculating the uncertainty of migration parameters (such as distribution coefficient Kd), an approach that stems from chemical analytical measurement calculations has been applied. The "case" is divided into individual, well-defined steps that can be described using simple uncertainty calculation as described below. These steps are then connected together with increasing the measure and extending the calculation.

Uncertainty calculation (according to EURACHEM, 2000)

Individual component or groups of components of uncertainty expressed as standard uncertainties $(u(x_i))$ are calculated to the combined standard uncertainty $(u_c(y))$. The general relationship between the combined standard uncertainty $u_c(y)$ of a value y and the uncertainty of the independent parameters $x_1, x_2, ..., x_n$ on which it depends is

$$u_{c}(y(x_{1}, x_{2}, ...)) = \sqrt{\sum_{i=1,n} c_{i}^{2} u(x_{i})^{2}} = \sqrt{\sum_{i=1,n} u(y, x_{i})^{2}}$$
(1)

where: $y(x_1, x_2, ...)$ is a function of several parameters $x_1, x_2, ..., c_i$ is a sensitivity coefficient evaluated as $c_i = \frac{\partial y}{\partial x_i}$, the partial differential of y with respect to x_i and $u(y, x_i)$ denotes the





uncertainty in y arising from the uncertainty in x_i . For the case where variables are not independent, the relationship is

$$u_{c}(y(x_{i,j},...)) = \sqrt{\sum_{i=1,n} c_{i}^{2} u(x_{i})^{2} + \sum_{\substack{i,k=1,n\\i\neq k}} c_{i} c_{k} \cdot u(x_{i},x_{k})}$$
(2)

where: $u(x_i, x_k)$ is the covariance between x_i and x_k , and c_i and c_k are the sensitivity coefficients. The covariance is related to the correlation coefficient r_{ik} by

$$u(x_i, x_k) = u(x_i) \cdot u(x_k) \cdot r_{ik}$$
(3)

where: $-1 \le r_{ik} \le 1$

The spreadsheet software can be used to simplify the calculation by an approximate numerical method of differentiation. It requires knowledge only of the calculation used to derive the final result (including any necessary correction factors or influences) and of the numerical values of the parameters and their uncertainties. If either $y(x_1, x_2, ..., x_n)$ is linear in x_i or $u(x_i)$ is small compared to x_i , the partial differentials can be approximated by

$$\frac{\partial_{y}}{\partial_{x_{i}}} \approx \frac{y(x_{i} + u(x_{i})) - y(x_{i})}{u(x_{i})}$$
(4)

Multiplying by $u(x_i)$ to obtain the uncertainty $u(y, x_i)$ in y due to the uncertainty in x_i gives

$$u(y, x_i) \approx y(x_1, x_2, \dots, (x_i + u(x_i)) \dots x_n) - y(x_1, x_2, \dots, x_i \dots x_n)$$
(A5)

Thus $u(y,x_i)$ is just the difference between the values of y calculated for $[x_i + u(x_i)]$ and x_i respectively.

Application of selected approach

The approach described above has been tested for the distribution coefficient K_d determination in well-defined laboratory condition (Vejsada, 2006). The case has been divided into several individual steps that were analyzed in detail for theirs uncertainty contribution to the final uncertainty of K_d . For such a case it has been concluded, that this approach is applicable. But also several important problems were observed that require closer investigation. In our study, some steps were simplified because of lack of source data, which is the most important problem for uncertainty calculation. The second problem found was unknown quantities chaining (e.g. the mineral composition uncertainty depends on precise mineralogical analysis that depends on quality of sample, its chemical analysis, reference pattern matching and other factors; the unknown in one basic step is often chained to next steps without knowledge of uncertainties and connections, thus new parameter cannot be calculated correctly). The third important problem is the measure of evaluation and adequate appreciation of input data.

For closer identification of such problems in our approach, a new study has been started. The main aims are: lack of input data reduction, unknown quantities chaining reduction and




Appendix A8: NRI, RAWRA (Czech Republic)

cooperation with performance assessment codes requirements.

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Part 3: Uncertainty management and uncertainty analysis

Appendix A9: POSIVA (Finland)

A9 POSIVA (Finland)



Appendix A9: POSIVA (Finland)

PAMINA

WP1.1 Uncertainty management and uncertainty analysis

Posiva Oy – Contribution – 2nd Draft

PAMINA. WP1.1 Comprehensive Review of Methodologies and Approaches in the Safety Case

Uncertainty management and uncertainty analysis

1 Introduction

In Posiva's Safety Case uncertainties are linked to the understanding of the components of the repository system starting from the site and going through design to materials (bentonite, copper, iron, fuel) and processes. The understanding of the long-term behaviour of the components and processes includes uncertainties, which need of management to seek for improvements in understanding and safety, whenever possible.

2 State of the art in the treatment of uncertainties

The overall approach to treat uncertainties is deterministic. The protocol followed up to now consist of (e.g. POSIVA 2005):

- Listing and classifying the main uncertainties
- Determining the cause of the uncertainty (e.g. data inaccuracy)
- Noting whether the property and associated uncertainty has been determined using information from more than one source (incl. validation)
- Assessing the impact on other parts of the entirety under the assessment,
- Quantification of the uncertainty if possible,
- Determining whether there is a potential for an alternative representation and whether an alternative has actually been developed,
- Determining whether there are unused data which could be used to reduce uncertainty, and
- Deciding what new data would potentially help resolve uncertainty (input to further research programme)

The development programme is subject to regular updates, where uncertainties highlighted in previous steps are taken into account in planning research activity and/or implementation of design.

Although the treatment of uncertainties in a systematic way is being developed, uncertainties





Appendix A9: POSIVA (Finland)

are being treated in two levels:

2.1 Detail level in the understanding of individual component and processes

For example, in groundwater modelling parameter uncertainty is found in the values of hydraulic conductivity used, in the orientation of the deposition tunnels with respect to fractures, in the properties of the fractures (transmissivity), etc. These parameters influence the properties of the buffer, backfill, and sealings and their long-term behaviour. The forthcoming Process report (POSIVA 2007) addresses or highlights uncertainties in each of the process considered taking also into account the time frame for which the process is relevant in the overall safety of the system.

2.2 General level in the safety assessment

For the safety assessment most of conceptual and numerical uncertainties can be taken into account when defining scenarios and calculation cases. Varying one or several parameter values in the calculations, a wide rang of solutions is obtained that cover most of the uncertainties.

3 Challenges in the treatment of uncertainties

The Safety Case (SC) is a much broader concept than safety assessment. However the safety assessment is to include and address the uncertainties of all the other elements involved in the safety case (e.g. site, processes, design, evolution, etc.). The traceability of uncertainties from individual elements to the overall safety assessment should be made transparent to add confidence to the Safety Case. Most probable all uncertainties cannot be ruled out, but we need to be confident when addressing safety in spite of the remaining uncertainties.

4 References

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Part 3: Uncertainty management and uncertainty analysis Appendix A10: SCK-CEN, ONDRAF-NIRAS (Belgium)

A10 SCK-CEN, ONDRAF-NIRAS (Belgium)



14-09-2007

NIRAS/ONDRAF SCK.CEN Version: F

PAMINA PROJECT

WP 1.1 Review of Methodologies

Uncertainty management and uncertainty analysis

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1. Introduction

Uncertainty analyses have always been part of the safety assessments conducted in the past RD&D (Research, Development and Demonstration) of the Belgian disposal programme for radioactive waste, but the systematic integration and the level of details of these analyses have progressively evolved. Already in the PAGIS safety assessment [1] a first series of limited uncertainty analyses were conducted, mainly focussing on parameter uncertainty by making stochastic calculations, and on the analysis of a first set of scenarios, derived on the basis of expert judgement. In the SAFIR 2 report [2] more detailed uncertainty and sensitivity analyses were conducted and a first attempt in the direction of uncertainty management was made to discuss in a more systematic manner the different types of uncertainties and their impact on the level of confidence in the safety and feasibility of the studied disposal system and on future activities of RD&D.

2. Regulatory requirements and provisions

No disposal specific regulatory standards exist at the moment in Belgium, and the regulatory body (the Federal Agency for Nuclear Control) is currently defining protection criteria for disposal and is developing regulatory guidance. It is expected that this regulatory guidance will also treat the aspects of uncertainties and uncertainty analysis in safety assessments and in the safety case.

3. Key terms and concepts

In this paper a distinction is made between "uncertainty analysis" and "uncertainty



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management":

- uncertainty analysis is the analysis by different methods and tools that aims at the quantification of the uncertainty in the considered output variable (e.g. calculated doses or radionuclide fluxes);
- uncertainty management is the broader activity of deciding on the level of the disposal programme how to deal with the uncertainties, i.e. what measures have to be or will be taken in the disposal programme to systematically identify the uncertainties and decide for each of the identified uncertainties the way to treat them (e.g. reduction of uncertainties through additional RD&D, design modifications or site and waste characterisation actions, conservative assumptions in assessments).

Uncertainties are classified in the categories "scenario uncertainty", "model uncertainty", and "parameter uncertainty". We also make a distinction between poor knowledge (lack of data) and variability in space and time, but this distinction is not yet systematically introduced in the programme. Classification of the uncertainties is given below.

3.1 Scenarios

The scenario approach in general is used to deal with the uncertainties related to the evolution of the system.

Besides the definition and description of an expected evolution scenario in SAFIR 2, altered evolution scenarios are defined to treat the uncertainty in the evolution of the disposal system as a result of events and processes that are unlikely to occur but cannot be entirely ruled out. As the probability of occurrence of most of these events and processes cannot be accurately quantified, the decision to treat events and processes not in the expected evolution scenario, but in altered evolution scenarios is not always straightforward. Often the scenario uncertainties are related to the moment and magnitude of disturbance caused by these events and processes, and conservative assumptions are required (e.g. occurrence in an early timeframe, when the expected radiological consequences are highest).

"Variants" are considered within a specific scenario such as the distinction made between different possible evolutions of climate (Milankovitch or greenhouse) in the expected evolution scenario.

A specific category of scenarios are human intrusions, because the uncertainties related to future human actions that can potentially disturb the system and lead to exposures are largely irreducible. The use of one or several stylised scenarios is the approach taken.

3.2 Models

Uncertainties arise from the model representation of processes. These uncertainties may originate from a lack of knowledge of elementary steps underlying an observed process.

Choices regarding the mathematical representation of a process in the model is also a source of uncertainties (e.g. resolution level, simplifications introduced by applying one- or two- dimensional models, initial and boundary conditions).

Specific uncertainties that are largely considered to be irreducible (e.g. how to model human intrusions or evolutions in the biosphere) are treated in a stylised manner.





3.3 Parameters

The experimental uncertainties (measurement errors), the scarce number of experimental results and the discrete representation of parameters which have a continuous variability over space and/or time (e.g. the hydraulic conductivity K over a geological layer) contribute both to the parameter uncertainties.

4. Treatment in the Safety Case

4.1 Methodology

Management of the scenario uncertainties in SAFIR 2 is conducted through the analysis of the disposal system using different lines of reasoning and calculations within a structured and transparent scenario-development scheme. International peer reviewing and verification of the disposal-specific FEPs with internationally established databases provide additional support to the scenario uncertainties treatment.

Comparison of in situ experiments with a priori simulation results is an important way of testing models and to evaluate how good a process is modelled. This validation process is applied in the case of the migration of non-retarded radionuclides in the Boom Clay. However comparison between models and in situ experiments is not always possible because of scaling problems. It is impossible for example to perform validation tests of migration for retarded species because of the long time this experiment would require. Therefore, the models have to be tested in different and "indirect" ways, e.g., by analysing the distribution of the natural U and Th isotopes in the Boom clay in the case of studies of retarded radionuclides. Different conceptual models are used when there is uncertainty about the active processes that can occur and their representation. An example is the modeling of the transport of actinides through the Boom Clay in SAFIR 2. The uncertainty with respect to the mechanisms of radionuclide retardation has led to using two alternative conceptual models: one considering complexation by mobile organic species and another considering limitation of solubility and retardation by sorption on clay minerals or on immobile organic material.

Central part of the uncertainty analyses in the past safety assessments is the stochastic (Monte Carlo simulations) and deterministic assessments of parameter uncertainty. Parameter distribution functions (pdfs) have been estimated for the most important parameters (such as the transport parameters of critical radionuclides). However, for the majority of parameters there is not enough knowledge to quantify in a rigorous way the uncertainties, e.g. in the case of radionuclide transport parameter values (sorption coefficients, solubility limits, ...) and it is not possible to identify pdfs by applying statistical techniques. Therefore, most uncertainties are described by a log-uniform distribution for which a best estimate value and an uncertainty factor were estimated. Conservative parameter values are often used to avoid the problem in quantifying uncertainty.

Deterministic and stochastic (probabilistic) calculations are seen as complementary and both approaches are adopted. The deterministic approach presents advantages when interpreting the results in terms of compliance and when presenting the results to various stakeholders. Stochastic calculations are a tool for evaluating some types of uncertainties (combined parameter value uncertainty) and for identifying strongly influential parameter uncertainties (called sensitivity analysis).





4.2 Related topics

O/N has funded GSL to participate, on its behalf or within a partnership, to the other tasks of PAMINA related to uncertainty (Task 1.2, 2.1C, 2.2C and 2.3). Moreover SCK•CEN takes part to a clay benchmark exercise consisting of simulations using models of various levels of complexities (task 4.1) and of different dimensions (task 4.2).

4.3 Databases and tools

The catalogue of all FEPs relevant to the deep disposal of the high-level and long-lived waste into the Boom clay [3] has been checked for consistency and completeness with a more general and extensive FEP catalogue compiled by international working groups [4].

The software used for uncertainty and sensitivity analyses is based on SCK•CEN's version of the LISA code, which was developed in collaboration with JRC Ispra. The new version consists of a script that allows combining the statistical subroutines for preprocessing (sampling) and postprocessing (uncertainty and sensitivity analyses) with transport codes such as PORFLOW.

4.4 Application and experience

4.4.1 Monte Carlo simulations

The Monte Carlo simulations performed to evaluate the impact of the parameter uncertainties made use of both random and Latin hypercube sampling technique. However, the latter technique appeared to yield stable results (convergence) for a smaller number of runs. Fractional factorial design has also been tested as alternative technique for Monte Carlo simulations [5]. This technique yields satisfactory results for sensitivity analyses, but it cannot be recommended for uncertainty analyses.

To quantify the uncertainties in the considered output variable, e.g. the dose, 2 complementary sets of quantiles are used: the first set consists of the expectation values, i.e. arithmetic mean, and its 95% upper confidence limit, the second set consists of the 95th, 90th and 50th percentiles. An example of results of an uncertainty analysis is given in Fig. 1.

4.4.2 Measures taken to reduce or bound uncertainties

In the design of the repository the use of a long-lived (a few thousand years) container and/or overpack for the high-level heat emitting waste avoids that the uncertainties associated with temperature evolution (radionuclide releases from the waste form and radionuclide migration in a thermal gradient) and parameter values applicable at elevated temperatures have to be taken into account in the analysis of the expected evolution scenario.

Another conservative approach is the introduction of the robust concept: components that might, even significantly, contribute to the performance of the repository system are not considered in the evaluations if there is a lack of knowledge to evaluate their performance, e.g. the sorption of radionuclides on the iron(hydr)oxides that are formed in the near field during corrosion of the container.







Fig.1: Uncertainty analysis: evolution of the expectation value and percentiles of the dose for the case of disposal of spent fuel in the

The choice of the Boom Clay as the reference host rock leads to a case of diffusion controlled radionuclide transport, avoiding the complications of significant advective contributions that would be more sensitive to changes in the hydrogeological environment.

Given the very important contributions of the Boom Clay to the safe containment and isolation of the waste a central point of attention in the design of the EBS is the potential disturbance of the Boom Clay safety functions by the EBS components and materials. In the choice of EBS components the physico-chemical and mechanical impact on the Boom Clay is systematically considered. The underlying processes of Boom Clay performance have to be sufficiently understood to be able to convincingly assess these disturbances.

5. Lessons learned

In the SAFIR 2 report a final discussion was devoted to the systematic review of all types of uncertainties and their impact on the level of confidence with respect to safety and feasibility of the disposal system and on the future RD&D programme. This discussion was structured around the main types of uncertainties (scenario, model and parameter) and around a series of key questions (see below).

This evaluation, which is based mainly on the knowledge of the Boom Clay beneath the nuclear zone of Mol-Dessel, is an attempt to answer a series of fundamental questions about the development of an underground repository:





- 1. Can the disposal system be characterised?
- 2. Is the disposal system understood?
- 3. Can such a disposal system be designed and built?
- 4. Can values of parameters that relate to the disposal system be extrapolated in time and space ('upscaling')?
- 5. Can the safety of an underground repository be assessed?
- 6. Is the proposed disposal system safe?
- 7. Is the relative importance of the different types of uncertainty understood?
- 8. Can its costs be estimated and funding for its construction guaranteed?
- 9. What information and experience can be transposed to another location or another host formation?

This thorough analysis led to the general SAFIR 2 conclusion of system feasibility and safety, but also to the identification of the key remaining uncertainties and of the main priorities for the current RD&D phase of the disposal programme:

- Demonstrating the feasibility of implementing the disposal facilities;
- Improving the understanding of the processes of radionuclide retention at work in the Boom Clay and of the evolution of the retention properties in this formation for some specific radionuclides/elements (e.g. Se, some actinides);
- Analysing the heterogeneities and discontinuities in the Boom Clay and their effects on groundwater flows and on radionuclide migration;
- Analysing the effect on groundwater flows in the Boom Clay of changes in the regional hydrogeological conditions within the surrounding aquifers;
- Studying more in detail the aspects of chemical, biological, and physical compatibility of all of the repository materials with the host formation, and of the different disturbances induced by the various waste classes;
- Reviewing the choice of material for the packagings/overpacks and developing an integrated approach to defining the engineered barrier system based on preventing corrosion of the packaging/overpack;
- Creating a systematic and system-oriented design methodology for the disposal facilities for all waste classes, especially for the most demanding ones;
- Analysing the effects on the repository, on the host formation, and on safety of gas generation by the waste (mainly the category B waste), and evaluating ways in which the repository design might address these;
- Studying and demonstrating methods that can be used to characterise the waste and to verify and confirm their composition and heat emission. The nature and extent of these operations must be proportionate to the need for knowledge of the waste with a view to its deep disposal;
- Improving the methodology used to assess long-term safety, in particular as regards identifying and addressing uncertainties and alternative safety and performance indicators;





• Defining and developing a system for the long-term management and transfer of knowledge, in particular to enable the traceability of decisions and technical choices and the transmission, integration, and synthesis of multi-disciplinary information.

By submitting these findings to a peer review, as was the case for SAFIR 2 with the NEA peer review [6], a critical independent assessment is made of their correctness and completeness.

The main aspects of uncertainty analyses have been considered within the Belgian programme with a focus placed particularly on the parameter uncertainties. The complementary use of deterministic and stochastic calculations seems promising to evaluate and manage the impacts and dependences of the model and parameter uncertainties on the sensitivity of the system. A challenge for the running programme remains the development and the implementation of a methodology that can treat in a homogeneous way and in depth the various sources of uncertainty and evaluate their impact on the safety indicators.

6. References

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PART 4: SAFETY INDICATORS AND PERFORMANCE INDICATORS

(Prepared by D.-A. Becker. GRS, Germany)





1 Introduction and background

Measures for quantifying the results of performance assessment calculations, mainly dose and risk, have always been used. However, there is a wide international consensus today that it is necessary to use complementary indicators to improve the understanding of the system and to support the safety case. The long-term repository safety cannot be reduced to one single numerical measure and should at least be assessed using several independent indicators, which are called safety indicators. Other indicators are used to quantify or demonstrate the performance of subsystems or single barriers, or of the total system with respect to more specific aspects. Different names are in use for these indicators, such as performance indicators, function indicators, secondary safety indicators or safety function indicators. A unique and internationally accepted terminology does not exist at present.

While there is a consensus that using different indicators in addition to dose or risk in performance assessments is a good way to support the safety case, the different concepts and perceptions vary between the countries and organisations. Since the idea arose some ten years ago, national and international research programmes have led to these different views. Two international projects should be explicitly mentioned in this context:

- The IAEA Coordinated Research Programme on Safety Indicators (1999 2003) [IAEA-1372]
- The SPIN project (Testing of Safety and Performance Indicators) within the 5th EURATOM framework programme (2000 – 2002) [Becker et al. 2003]. The participants in SPIN were COLENCO, GRS, ENRESA, NAGRA, NRG, NRI, SCK·CEN and VTT.

The IAEA CRP was mainly aimed at creating a database of measured natural concentrations of radiologically relevant substances in materials from different geographical areas, as well as fluxes of such substances between geological compartments, showing their spatial variability. From this database 'safe' values were established for comparison with calculation results. In the SPIN project several numerical measures were identified as safety and performance indicators and tested for their usability in performance assessment by recalculating a number of recent national studies.

Although it was intended to keep the concepts developed in these two programmes compatible, this was not possible. Moreover, additional concepts using similar terms have been introduced in some countries and partially adopted by others. All these concepts have been interpreted and refined independently by individual organisations in view of their specific needs and perceptions during recent years. This is the reason why today there is a considerable inhomogeneity of views and understanding of this topic between different organisations. The differences and commonalities are presented in this topic report.

2 Regulations and guidelines

Numerical performance assessment is generally accepted as the most important means for assessing the long-term safety of repositories, and therefore, it is subject to detailed formal regulations in the various countries. Such regulations normally make use of at least one typical safety indicator such as dose or risk. The existing national regulations differ in their level of detail and in some countries the regulators are currently revising them. An overview





of the regulations regarding safety indicators in various countries is given below.

In the UK there is a specific requirement that individual risk should be below 10⁻⁶/yr. There is also a general requirement to look at indicators other than dose or risk, and additionally at qualitative safety arguments.

In Belgium there is currently no specific regulation for the disposal of radioactive waste. The general rules for protection against ionising radiation (2001) require a maximum dose limit of 1 mSv/yr for members of the public and 20 mSv/yr for workers.

In Spain, a regulation of 1987 requires that the individual risk should be below 10^{-6} /yr, which is directly associated to a dose constraint of 0.1 mSv/yr.

In the Netherlands there is a requirement that a safety report has to show that risks and individual doses are below the regulatory limits. These limits are defined for different scenario probabilities and different groups of people (adults / children). The limits lie between 0.04 mSv/yr and 40 mSv/yr for adults and between 0.015 mSv/yr and 15 mSv/yr for children. Moreover, the individual risk due to the releases from a repository must remain below 10^{-6} /yr.

In the Czech Republic it is required that the individual dose rate originating from the repository remains below 0.25 mSv/yr for normal evolution scenarios and below 1 mSv/yr for less probable "emergency scenarios". There is a regulation on general Environmental Impact, which will be additionally applied in performance assessment, addressing the impacts on flora and fauna, soil, water, climate, etc.

In France the basic safety rules require that the radiological impact of a repository to the environment be limited to levels that are as low as reasonably achievable (ALARA principle). The individual dose rate must not exceed 0.25 mSv/yr for the reference scenario, associated with certain or highly probable events. For situations considered as altered, the calculated impact is assessed according to the likelihood of the situation, the chronic or timely character of the exposures, the degree of pessimism of the calculation assumptions.

In Germany there is currently no legal regulation for the assessment of the long-term safety of repositories, but there is an official guideline that the individual dose rate originating from a sealed repository must not exceed a value of 0.3 mSv/yr. A supplementary regulation defines the time frame for which the dose rate should be evaluated to 10 000 years.

In Finland has the most detailed regulations of the countries participating in PAMINA (STUK 2001). The individual dose rate must remain below 0.1 mSv/yr for the most exposed members of the public and "insignificantly low" for others, within an assessment period that is adequately predictable by means of given assumptions about exposure pathways, human habits, etc. For long time frames beyond adequate predictability, constraints for the average release of specific radioactive substances from the repository are specified. Moreover, it is required that the repository shall not affect flora and fauna, which is to be demonstrated by assessing the radiation exposure of typical terrestrial and aquatic populations. Finally, it is required that the safety concept be based on redundant barriers, which should be assessed by means of adequate indicators.

In several countries the regulations are currently being revised. In France the Basic Safety Rule of 1991 is under revision, taking account of recently developed safety approaches. In Belgium a regulatory framework for disposal of radioactive waste is being developed. In Germany the guideline of 1983 is being replaced by a new one, which is still under intense





discussion. This guideline will require the evaluation of several independent safety criteria.

As a summary, the regulations of the different countries always establish at least one **safety indicator** for which an acceptance criterion is defined. The situation of **performance indicators** is quite different, since most regulations include no requirements on performance indicators. Performance indicators are usually selected and used by the implementers when building the Safety Case to understand, quantify and present to different audiences how the disposal system works.

3 Terminology

The terminology used by the different organisations is rather inconsistent. This means that identical or very similar concepts are sometimes denoted differently, while in other cases the same term is used with different meanings. Some formal definitions of the basic terms exist and are generally accepted, but these are interpreted differently by different organisations.

3.1 Safety indicator

Within the context of PAMINA 'safety' is understood to refer exclusively to the long-term safety of repositories, i.e. passive safety during the principally unlimited post-operational period. The term "safety" itself, however, is not clearly defined in a way that allows a principally unique decision whether or not a repository is safe.

IAEA definition

There is a definition of the term "safety indicator" in the IAEA Safety Glossary (2007):

Safety Indicator: A quantity used in assessments as a measure of the radiological impact of a source or practice, or of the performance of protection and safety provisions, other than a prediction of dose or risk.

Such quantities are most commonly used in situations where predictions of dose or risk are unlikely to be reliable, e.g. long term assessments of repositories. They are normally either:

(a) Illustrative calculations of dose or risk quantities, used to give an indication of the possible magnitude of doses or risks for comparison with criteria; or

(b) Other quantities, such as radionuclide concentrations or fluxes that are considered to give a more reliable indication of impact, and that can be compared with other relevant data.

This is a rather general definition which is not specifically made for repository safety. It only refers to radiological safety and does not speak of reference values, but only of comparison with "other relevant data".

There is a consensus about this definition for use in repository PA:

• The wording should not be understood to exclude dose and risk from being safety





indicators.

 The limitation to radiological impacts results from the responsibility of IAEA but is not always justified for repository PA since chemotoxical aspects can also be addressed by means of safety indicators.

SPIN definition

In the SPIN project a more precise definition was given for the project's purpose:

A safety indicator must

- Provide a statement on the safety of the whole system
- Provide an integrated measure describing the effects of the whole radionuclide spectrum
- Be a calculable time-dependent parameter
- Allow comparison with safety-related reference values

Unlike the IAEA definition, this definition clearly requires comparability with safety-related reference values, which means that, e.g., comparability between different options or repository types does not suffice to make a proper safety indicator.

This more precise definition describes the kind of safety indicators considered in SPIN. While some organisations find it too weak for their current work, it is considered too restrictive by others. Discussions during the workshop showed that it would be hard to reach an agreement on a common definition for the "safety indicators", because each organisation uses this concept with its own shades, which makes harmonisation difficult. Some organisations, such as GRS-K, are interested in developing the concept of "safety indicators" in more detail while others consider that a broad definition is enough.

Therefore, in view of the apparently different perceptions, it is strongly recommended that the term 'safety indicator' is not used without clearly stating what is meant.

3.2 Performance indicator, function indicator, safety function indicator

The use of the terms "performance indicator" and "function indicator" is rather inhomogeneous and there is no clear distinction between the two. The term "safety function indicator", however, has been introduced by SKB and denotes a characteristic measure for the integrity of a barrier that can be compared with a given technical criterion. Use of this term is generally restricted to SKB concept.





IAEA definition

The IAEA Safety Glossary gives a very short definition of performance indicator:

Performance indicator: Characteristic of a process that can be observed, measured or trended to infer or directly indicate the current and future performance of the process, with particular emphasis on satisfactory performance for safety.

If "process" is understood to mean not only physical or chemical processes but also the evolution of system components or even the total system, this definition seems to be in line with what the PAMINA participants have in mind. It is, however, not very precise and still allows a variety of different interpretations.

The term "function indicator" is not defined in the IAEA Safety Glossary.

SPIN definition

In SPIN a more precise definition of the term "performance indicator" was given for the specific purpose of the project:

A performance/function indicator must:

• Provide a statement on the performance of the whole system, a subsystem or a single barrier

- Provide a nuclide-specific or integral measure
- Be a calculable, time-dependent or absolute parameter
- Allow comparison between different options or with technical criteria
- Illustrates the functioning of the repository system.

This definition allows a wider variety of quantities to be used as performance indicators, compared with safety indicators. The main difference, however, is that no reference value is required that allows an assessment of safety, but only comparability between different options or with technical criteria. This includes the kind of indicators introduced by SKB to assess the compliance of barriers with technical criteria, called "safety function indicators", even if some of them are not the outcome of PA calculations.

3.3 Individual views

In the following the specific views of some organisations concerning the understanding of the terms in question are described in more detail.

ANDRA and **IRSN** indicators are in line with the SPIN definitions. The only *safety indicator* used so far is the individual effective dose per year within the context of a predefined biosphere and critical group, as well as a reference value of 0.25 mSv/yr. *Performance indicators* are understood to assess specific functions of the disposal system. Several such indicators are used in the recent safety evaluation (Dossier 2005 Argile).





ENRESA adopts the SPIN definitions and considers the annual effective dose as the main safety indicator. Moreover, the activity leaving the near field in a year is compared with the natural activity in a certain amount of natural soil, to show that radionuclide releases to the biosphere due to the repository are negligible compared with natural radioactivity. This is also considered a *safety indicator*.

Additionally, a number of different *performance indicators* are used. Some of these are timedependent, such as activity fluxes, while others are not, such as travel times or retardation factors. Performance indicators are primarily understood as indicators for the functioning of individual barriers.

NRG provides no definition but seems to follow SPIN. Dose and risk are the basic *safety indicators*. Closure times of plugs and seals are used as *performance indicators* for a repository in salt. The Dutch probabilistic safety study PROSA mentions the term *relevant characteristics*, indicating parameters such as glass dissolution rate, internal rise rate of a salt dome, groundwater velocity, distribution coefficients, and dose conversion factors.

NIRAS/ONDRAF and **SCK-CEN** follow largely the SPIN definition. The main safety indicators are the individual effective dose and the radiological risk (defined as the product of probability of exposure and the probability of a harmful effect on human health). Additionally, some indicators are defined that are supposed to be "complementary" to dose and risk and are called *performance indicators for the safety functions*. These indicators allow evaluation of the global and partial performance of the disposal system and the long-term safety functions.

DBE TEC accepts the SPIN definitions. For defining indicators they propose to follow either a top-down approach starting with legal regulations and deriving technical criteria, or a bottom-up approach defining comprehensive indicators that directly assess the fulfilment of regulations. DBE TEC remarks that even on the regulatory level there is no unique view of the terms. *Performance indicator* and *function indicator* are used synonymously.

Posiva distinguishes between *primary safety indicators* (PSI) and *complementary safety indicators* (CSI). The former refer to the radiological impact of the total repository system, including the biosphere path. The annual effective dose is the only PSI considered and practically the only indicator that fulfils this definition. Complementary safety indicators can be quantitative (numerical) or qualitative and can refer to the total system or a part of it. The radionuclide-specific flux from the geosphere to the biosphere is considered the most important CSI. The terms *performance indicator* and *function indicator* are considered unclear and not used. It is stated that unique definitions of all terms are needed.

AVN holds a more general view. According to them, a *safety indicator* should be considered as a key piece of information for the decision making process in order to proceed to the next step. It should assess the level of implementation of the safety strategy, addressing either the whole disposal system or a part of it. Comparability with reference values is not considered as a requirement. Dose and risk comply with this definition. The terms *performance indicator* and *function indicator* are not used.

NRI understands the term *safety indicator* to denote a value derived from natural concentrations or fluxes of radionuclides. Dose calculations are compared with two values, called *bottom safety indicator* and *upper safety indicator*. Performance indicators are understood in the sense of SPIN but have not been used so far.





NDA considers a range of safety indicators, but gives no formal definition or view of the terms under discussion. In the UK, consideration of qualitative arguments, as well as dose and risk calculations, is expected by the regulator.

GRS-K, the section of GRS that is closely associated with the regulator, has prepared a proposal for a new regulation to replace the old "safety criteria" of 1983. It defines safety indicators to show that the protection objectives are met by an integrated assessment of repository safety, and function indicators to assess the reliability performance of subsystems or components with regard to the requirements. The proposal for a new regulation contains six safety indicators the most of which are not directly related to radiological impact but indicate the containment isolation/containment capacity of the repository system. This concept is based on the perception that biosphere evolution and thus radiological impact can not be predicted on the long term. Consequently, the proof of long-term safety should focus on the safety function "isolation/containment" rather than on radiological impact. GRS-K arguments that protection objectives are met if containment is ensured.

GRS-B, the section of GRS that works for the implementer side, prefers an even more restrictive version of the SPIN definition. A *safety indicator* should address a specific part-aspect of safety and – as objectively as possible – quantify the respective degree of safety. A safety indicator of this kind makes no sense without a clearly safety-related reference value, and different reference magnitudes can make different safety indicators out of the same calculated measure. *Performance indicators* are understood in the sense of SPIN and are primarily used for demonstrating the functioning of the system.

The most important differences in the individual perceptions of safety indicators lie in the understanding of what "safety-related" means, as well as in the significance of reference values. Since "safety" is not a unique concept, there are different interpretations. There seems to be a consensus that additional safety indicators should support a single dose or risk criterion, but there are divergent opinions about how safety should be quantified using different measures.

Concerning reference values, the variety of opinions is even wider. While GRS-B holds a very restrictive view and insists on directly safety-related reference value for each safety indicator, ENRESA and GRS-K, for example, accept also less strict reference values. NRI does not clearly distinguish reference values from safety indicators. AVN's position is that safety indicators do not require reference values necessarily.

4 Methodology

Several organisations have already applied safety indicators other than dose or risk and/or performance/function indicators within their studies. Other organisations are planning to do so in the future. Due to the different conceptual perceptions described in the previous chapter, the approaches and methodologies differ between organisations. Understandably, organisations that have already used such indicators have more concrete concepts than the others. The participants of SPIN (COLENCO, GRS, ENRESA, NAGRA, NRG, NRI, SCK·CEN and VTT) seem to use the outcome of that project as a basis for their concepts.

In general, each approach consists of three steps. The first step is selection of the indicators to be evaluated, the second step is the numerical calculation and the third step is the presentation. This third step is important for conveying the intended message.





In the following, the basic principles of the different approaches are summarised.

The standard approach, applied, among others, by **ENRESA** and **GRS-B**, uses the standard safety indicator, the annual dose, calculated for specific scenarios. This can be done following a probabilistic approach, which means that a number of realisations with stochastically drawn parameter values are calculated and the mean annual dose is used as the safety indicator or following a deterministic approach using a set of constant values for the parameters. In both cases the peak value (of the mean dose or the dose) is compared with a reference value of, e.g., 10^{-4} Sv/yr. For presentation the total dose is plotted together with the contributions of the individual fission and activation products and the four decay chains in order to give a quick graphic impression of the most important radionuclides. Another kind of presentation for probabilistic investigations is to plot the time curves of the mean, the maximum, the minimum and specific percentiles of the dose in one diagram. The maximum and minimum are independent of the selected pdfs and only depend on the parameter ranges.

As an additional safety indicator **ENRESA** calculates the activity flux leaving the far field. The objective is to put into perspective the amounts of radionuclides that leave the disposal system in a year, making a comparison with the activity present in the natural environment (the reference value used is the natural radioactivity in 1 m³ of granitic soil).

A number of performance indicators have also been evaluated, and found useful, by ENRESA:

- the canister failure distribution, is seen as useful to describe the expected canister performance,
- the fraction of UOX altered serves as an indicator of the capability of the UO₂ matrix as a barrier,
- the activity flux leaving the near field is an indicator of the capability of the EBS in granite to limit the radionuclide release, and
- the water travel time, the retardation factor of the geosphere and the radionuclide travel time through the geosphere are useful parameters to quantify the capability of the host formation as a barrier.

GRS-B considers, apart from the annual individual dose, the two safety indicators identified in SPIN to be useful. These are the concentration of radiotoxicity in the aquifer (preferably for medium time frames) and the radiotoxic flux from the geosphere (preferably for long time frames). It is regarded necessary to find well-founded reference values that define a safe level for each safety indicator. The concentration of radiotoxicity in drinking water that is deemed to be radiologically harmless is a good reference value that can be easily determined. A suitable reference value for the radiotoxicity flux from the geosphere, however, is harder to find. This flux could be compared with the natural flux in a river near the repository that will finally collect all released radionuclides. Another possible reference value is the natural radiotoxicity flux in the groundwater. Since both reference values address different safety aspects (integrity of river water or integrity of groundwater) they are considered to make different safety indicators from the same calculated quantity. All safety indicators are presented as time curves, possibly normalised to their reference value.

Performance indicators have been used by GRS, following the SPIN methodology, for demonstrating the functioning of the system, which for this purpose is divided into





functionally separated parts or subsystems, called compartments. The compartment structure has to be established for every repository system individually, depending on its real structure.

Three performance indicators have been used preferably:

- the concentrations of radiotoxicity in the compartments,
- the fluxes of radiotoxicity between the compartments,
- the time-integrated fluxes of radiotoxicity from the compartments.

All performance indicators are presented as time curves. The last indicator yields monotonic curves that finally reach an asymptotic value. The differences between these values show how the radionuclides are retained in subsequent compartments.

NIRAS/ONDRAF and **SCK·CEN** uses, additionally to the annual dose, the radiological risk as a safety indicator, which is more suitable for scenarios that cannot be ruled out but have a low probability. The risk, however, is not calculated by multiplying the consequences of each scenario with its probability and summing up over all relevant scenarios, but both components are presented separately.

Two indicators of the type that **NIRAS/ONDRAF** and **SCK•CEN** call "performance indicators for the safety functions" have been considered in their study:

- the decayed fractions of the initial inventory activity, calculated for all actinides as well as for all fission and activation products, that is released to the aquifer,
- containment factor: ratio of disposed activity to cumulative released activity into biosphere.

Furthermore, two complementary indicators are given:

- the total maximum annual activity flux released to the aquifer, compared to the natural alpha activity present in the geological formation,
- the total initial inventory of uranium in the waste, compared to the natural alpha activity present in the formation.

The first of these indicators is similar to time-integrated radiotoxicity flux calculated by GRS and yields similar information. The other indicators put some typical properties of the system into perspective with the amounts of natural radioactivity.

ANDRA uses the annual dose as a safety indicator (as recommended by the French Basic Safety Rule), but performs a detailed system analysis considering three main safety functions of the repository. Each of these functions is addressed through performance indicators. Performance (as used in the dossier 2005 Argile) characterises a function. It is established by the designer according to criteria defined by the users. Among the analyzed indicators are:

- (i) the relation between convective and diffusive flux in the repository and the host rock,
- (ii) the overall activity leaving the waste packages, the underground structures and the





host rock, as compared to the initial quantity contained in the waste packages,

(iii) the activity flux at each of these components,

(iv) the concentration distributions of dissolved materials in the host rock and in surrounding formations.

Some of them, however, can be presented as lines of argument than performance indicators. More specifically for each function:

Resisting water circulation:

- Advective and diffusive flow from the near and far field.
- Distribution of radionuclide masses between the near field (including the shafts) and the far field. This is to show that there is no preferential pathway over the drifts and shafts.

Limiting the release of radionuclides and immobilising them in the repository:

- Analysis of the consequences of early water arrival at the waste allows assessment of the safety function with respect to isolating the waste from water as long as possible.
- Analysis of diffusion and advection in disposal cells via the Peclet number revealed to be adequate to ensure that a diffusive regime was effective in the cell.
- The system capability to limit the release of radionuclides from the waste was assessed by performing a sensitivity analysis against stronger release models.
- Solubility limits of specific elements allow assessment of the retention capability of the waste.

Delaying and reducing the migration of radionuclides:

- Three types of indicator associated with the molar flow of each radionuclide are used to assess the performance of the function:
 - the maximum molar flow,
 - the mass integrally corresponding to the molar flow over the simulation period,
 - the appearance time of maximum molar flow.

Comparison of values for each of these indicators, between two different surfaces (Si and Si+1), helps in assessing the confinement capability of barriers lying between these two surfaces. This concept is in some way similar to the compartment concept of SPIN.

NRI considers the annual dose and puts some effort in determining reference values, different from the regulatory limit, by analysing natural concentrations and fluxes. The values derived in this way are presented together with the annual dose curves and are called "safety indicators". This is, in a certain sense, in line with the concept of GRS-B, since the reference value is seen as an integral part of the safety indicator, and different reference values make different safety indicators. Performance indicators have not been considered in the Czech concept.







NRG has performed deterministic and probabilistic annual dose calculations. No clear distinction between safety indicators and performance indicators has been made. Instead, the term relevant characteristics has been used for calculated safety-related and performance-related parameters.

NDA is planning to investigate several numerical performance indicators, in addition to annual risk calculations, such as radionuclide fluxes. Additionally, a great importance is attached to qualitative arguments, which are considered to be more meaningful for non-technical audiences. Such qualitative arguments can include

- comparison with natural analogues,
- consistency with independent site-specific evidence, such as observations in nature,
- evidence for the intrinsic robustness of the system,
- passive safety features,
- general arguments related to radioactive waste management.

Posiva considers the annual dose as the only "primary safety indicator". It is calculated for two scenarios, one only considering the drinking water path, the other also integrating watering cattle and irrigating crops. Additionally, it is planned to consider "complementary safety indicators", which can be quantitative or qualitative. These indicators are not yet specified, but the radionuclide-specific flux from the geosphere to the biosphere is considered the most important one.

AVN has not yet developed a detailed view on the subject but sees the necessity to use safety indicators other than dose or risk to support the safety case and to communicate the system safety to the technical and non-technical public. Safety indicators should provide a quantitative indication of the level of implementation of the safety strategy, but it is not considered necessary that a reference value is available for comparison.

GRS-K has developed a proposal for a new German guideline to replace the old "safety criteria" from 1983. This proposal contains the following six safety indicators:

- the proportion of the cumulative released quantity of substance over the safety case period (to assess directly the containment capability),
- the concentrations of released uranium and thorium (to assess the modification of natural concentrations),
- the contribution to the power density in groundwater (to assess the modification of natural radioactivity),
- the contribution to the radiotoxicity in groundwater (to assess the modification of natural radiotoxicity),
- the radionuclide concentration in the usable water near the surface (to assess the modification of natural radionuclide concentrations),
- the effective individual dose per year (to assess the modification of natural ingestion of radiotoxicity).

For each of these indicators, individual limits are provided for both classes of likely scenarios





and less-likely scenarios.

DBE TEC does not perform own calculations but emphasises the necessity of distinguishing between the technical and the regulatory level. It is stated that, when defining new indicators for assessing the performance of a repository, different regulatory fields (mining, water protection, radiation protection and their different timescales) and levels should be regarded.

5 Application and experience

Only some organisations have experience of calculating and evaluating safety indicators other than dose or risk and/or performance/function indicators. The SPIN participants made some experiences in that project by re-calculating four granite studies and evaluating several safety and performance indicators. It was agreed that that these three safety indicators are useful and should be used with preference for different time frames:

- The annual effective dose is best for relatively short time frames up to 10 000 years, but should nevertheless be evaluated over the total assessment period.
- The radiotoxicity concentration in the aquifer is more robust because it is independent of the biosphere uncertainties and should be preferably used for medium time frames up to some hundred thousand years.
- The radiotoxicity flux from the geosphere is still more robust as it is additionally independent of the aquifer uncertainties. It is, however, hard to find adequate reference values. It should be evaluated preferably for long time frames up to millions of years.

Similar conclusions regarding the use of these three indicators and the corresponding time frames had already been drawn by NEA [6].

Moreover, it was found that performance indicators provide a good means for improving and communicating system understanding.

ENRESA has calculated for the ENRESA-2000 study the activity released in a year and compared it to the natural activity content of a certain volume of soil. It was found that this is an illustrative measure for communicating the system safety to the public. Moreover, several performance indicators have been calculated. By calculating and presenting the fraction of UOX vs. time it could be shown that even under pessimistic assumptions the matrix will release the total inventory only after several millions of years. Radionuclide travel times show that, e.g., plutonium is unlikely to start to leave the system earlier than 5 million years after repository closure. ENRESA experience is that any magnitude that can be useful to understand the system behaviour and to quantify the capabilities of the different barriers to delay and limit the releases of radionuclides from the repository, should be considered when building the Safety Case.

GRS-B has calculated the three safety indicators recommended in SPIN in a national project dealing with the existing Morsleben LLW repository (ERAM). Though the reference values were determined independently it was found that all three indicators yield nearly the same gap of three orders of magnitude between the maximum output and the reference value. This is interpreted as a mutual confirmation of the safety statements. Additionally, some performance indicators have been calculated to illustrate the functioning of the system.





Especially the time-integrated radiotoxicity flux was found to give a good impression of the efficiency of the different system components. It could be shown that even the worst of the emplacement areas releases only 10 % of its inventory, the other emplacement areas less than 0.3 %.

NIRAS/ONDRAF and **SCK-CEN** claims that for longer time frames the safety assessment should be based increasingly on concentrations and fluxes instead of dose or risk, and for very long time frames on qualitative arguments rather than on calculations. In SAFIR-2, the annual dose has been calculated for several scenarios and their probabilities have been discussed in a qualitative manner. This is not exactly a risk calculation but a semiquantitative risk assessment. It is concluded that this kind of investigation is more appropriate than a numerical risk calculation because the scenario probabilities are highly uncertain. Three performance indicators have been evaluated. The decayed fractions of radionuclide inventories are rather small for long-lived weakly sorbed radionuclide, and consequently, large fractions of these reach the biosphere, but spread over long times. Only a very small portion (about 10⁻¹⁰) of the initial total activity, however, reaches the biosphere.

ANDRA has calculated several performance indicators in association with the performance of functions in Dossier 2005. The migration delay of radionuclides, for example, is illustrated by presenting the molar flows at different points of the repository. It can be seen that the flows at the exit of the shaft are clearly more delayed than those at the top of the host formation. The maximum arrives at the shaft exit after approximately 800 000 years, at the top of the formation after 250 000 years. By evaluation of the attenuation of the maximum it has been confirmed that the host clay formation (Callovo-Oxfordian) has a very good capability for retaining actinides and delaying their release.

6 Developments

Several organisations are planning further developments of their methods or test further indicators. In PAMINA RTDC-3 there is a work package on safety and performance indicators (WP3.4) in which, among others, GRS-B, ENRESA, SCK-CEN, NRG, NRI and AVN are involved. These organisations will try to harmonise their views, at least partially, and perform calculations within their different national studies and apply a variety of indicators including risk indicators. Defining suitable reference values for safety indicators is considered of high significance and work on this topic is being performed within PAMINA WP3.4.

ANDRA will soon revise their safety case and will reconduct the calculation of safety and performance indicators using a similar approach to the one of the dossier 2005. NDA has not yet implemented and applied a methodology but will do that soon.

7 Conclusions

There is international consensus that a repository safety case can be enhanced by the presentation of a range of safety indicators, to complement the dose or risk calculations. There are different concepts of assessing repository safety and performance by means of other indicators. Several organisations have experience in using such indicators for supporting the safety case and communicating the results to the technical and non-technical public. In some countries the authorities are planning to revise their regulations and introduce





the obligation to consider additional indicators.

The review has shown that there is still a large variety of different views on the exact terminology used for safety indicators and performance/function indicators. The workshop recognised this and felt it was not a serious issue as long as the terms were clearly explained in each safety case.

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9 Appendices

A1 ANDRA (France)

A2 AVN (Belgium)

A3 DBETEC (Germany)

A4 ENRESA (Spain)

A5 GRS-B (Braunschweig, Germany))

A6 GRS-K (Cologne, Germany)

A7 IRSN (France)

A8 NDA (United Kingdom)

A9 NRG (Netherlands)

A10 NRI, RAWRA (Czech Republic)

A11 POSIVA (Finland)

A12 SCK•CEN, ONDRAF-NIRAS (Belgium)





A1 ANDRA (France)









Project acronym:

PAMINA

WP1.1 OVERVIEW OF PAST EXPERIENCE IN SAFETY AND PERFORMANCE INDICATORS

Updated version 26/11/07

Andra







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STRATEGY AND KEY ELEMENTS

This present contribution from Andra aims at giving an overview of methodologies that have been used by Andra in the framework of the Dossier 2005 Argile in the four topics selected by the steering committee: 1) safety functions, 2) scenarios, 3) safety indicators and 4) uncertainties management.

The first meeting hold in Amsterdam on June 12th, 2007 was an opportunity to review contributions and discuss them for the future workshop to be held in Paris in October. The present document completes the draft provided for the Amsterdam meeting and clarifies some points discussed during the October 2007 workshop at Andra. Its structure has been revised according to the DWG common structure.

The December 30, 1991 French Waste Act entrusted Andra, the French national agency for radioactive waste management, with the task of assessing the feasibility of deep geological disposal. The Basic Safety Rule RFS III.2.f of June 1991 [i], issued by the French nuclear safety authority, provides a framework for the studies to be conducted. The protection of man and the environment are to be demonstrated. Furthermore, studies should show the ability to limit potential consequences to a level as low as reasonably possible. The concept should include a multiple barrier system, and rely on passive repository evolution without institutional control beyond a given timeframe (500 years). The studies carried out within this framework are presented in the "**Dossier 2005 Argile**" (clay) [ii] and "**Dossier 2005 Granite**" [iii].

PRIMARY REFERENCES

In the present document, the « Dossier 2005 Argile » is used as reference. Primary references include the French Act and the series of reports submitted accordingly:

- The French Waste Act dated 30th December 1991 [iv]
- The French Safety rules namely RFS.III.2.f, guidelines [i].
- Synthesis Report, Evaluation of the Feasibility of a Geological Repository, Meuse/Haute-Marne Site (available in English and French) [ii].
- Architecture and Management of a Geological Disposal System Report (TAG; C.RP.ADP.04.0001) (available in English and French) [v].
- Phenomenological Evolution of the Geological Repository Report (TEP; C.RP.ADS.04.0025), (available in English and French) [vi].
- Assessment of Geological Repository Safety Report (TES; C.RP.ADSQ.04.0022) (available in English and French) [vii].

Other references such as the presentation made at the symposium hold in Paris in January 2007 [viii], and the INTESC questionnaire [ix] have been used when applicable.

STRATEGY AND KEY ELEMENTS

The feasibility assessment for the argillaceous site builds upon a number of key elements:

• Basic input: the inventory model of the waste [x] and the geological site [xi],





- Safety functions [xii] and requirement management,
- Technical solutions based on industrial experience,
- Reversible management and monitoring,
- Phenomenological Analysis of Repository Situations (PARS) [xiii] and detailed, coupled process modelling,
- Qualitative Safety Assessment (QSA) [xiv], uncertainty management and scenarios,
- • ALLIANCES simulation platform and calculation results.

Although the process thus summarized may suggest a linear progression from basic input data to designing a "solution" and assessing its safety, the process is in fact highly iterative, with repeated feedback exchanged between the various processes (see Figure 1). In addition to the routine feedback common to parallel engineering, three main iteration loops have been identified since 1991, each corresponding to a major milestone of the program: License application for construction and operation of the underground research laboratory (in 1996), submission of the Dossier 2001 (in December 2001), and the recent submission of the Dossier 2005.



In view of providing sound feedback to design, research and development and to determine residual uncertainties, the following tools have been carried out: **the functional analysis (FA)** [xii, xv] to determine the safety functions and associated requirements – what do we want? -; the **Phenomenological Analysis of Repository Situations (PARS)** [xiii] providing a good scientific understanding based on scientific studies from surface and underground laboratory – what do we get? - ; **the qualitative safety analysis (QSA)** [xiv] managing uncertainties and the quantitative assessment [safety and performance indicators] including sensitivity analysis – What is the impact of a given uncertainty (or set of uncertainty factors) on the robustness of the system? – And eventually: does the concept meet the safety/acceptability criteria?

The following sections of the document describe in more details each of those topics according to the sequence of the various stages of activities conducted in the dossier 2005 (see Figure 2).





SAFETY AND PERFORMANCE INDICATORS

SECTION 1: BACKGROUND/INTRODUCTION

The Dossier 2005 concentrates, first of all, on the radiological risk, without overlooking the other potential impacts of a radiological waste repository.

For the long term, the main safety indicator remains the committed individual effective dose at the outlet within the context of a predefined biosphere and a predefined critical group. A dose of 0.25 mSv/year at most in a normal situation set by the RFS III.2.f is retained by Andra.

The choice was made of the same constraint of 0.25 mSv/year for the repository's operating and closure situations, because it refers more broadly to the notion of equity between the generations: we do not accept for future generations detriments whic h would not be accepted for present-day populations.

For situations considered as altered, the calculated impact is assessed according to the likelihood of the situation, the chronic or timely character of the exposures, the degree of pessimism of the calculation assumptions (see details in chapter 7 of TES, [vii]).

This type of assessment presents specific problems when it is carried out over one million years. At this scale it is illusionary to pretend to have an assessment of the lifestyles of the beings that will inhabit the studied sector. More generally, the models used for the impact calculation do not pretend to have a predictive character with respect to the transfer times of the radionuclides to the biosphere. They are intended only to provide a view of the impact as large as possible. For those reasons, the long-term calculated dose is indeed an indicator of the impact and not a prediction of the latter.

Other indicators than dose can be proposed which show more clearly the repository's intrinsic performances without requiring any assumptions on the surface environment and the biosphere. In particular, radionuclide concentration flows assessed at relevant emplacements with respect to the safety analysis of the repository (typically at the host formation outlet) allow refining the judgement on safety and overcoming some of the uncertainties. They allow comparing different situations or different design provisions in order to see which one is the most favourable with respect to the limitation of the radionuclide transfers, but they cannot be compared to thresholds.

Some indicators allow assessing the performance of individual component with respect to their safety functions (see Figure 2) (for example, molar fluxes of radionuclides, which are independent of uncertainties on the future evolution of the biosphere).

Among the analyzed indicators are :

(i) the relation between convective and diffusive flux in the repository and the host rock,

(ii) the overall activity leaving the waste packages, the underground structures and the host rock, as compared to the initial quantity contained in the waste packages,





(iii) the activity flux at each of these components,

(iv) the concentration distributions of dissolved materials in the host rock and in surrounding formations.

Such indicators have been used in the safety analysis carried out in the Dossier 2005 Argile. They are developed in detail in the following chapters.



Figure 2 : Representation of the various stages of the analysis, showing where indicators have been used in the dossier 2005.

SECTION 2: REGULATORY REQUIREMENT AND PROVISIONS

The RFS III.2.f gives some radioprotection criteria [i]. They are expressed as follows:

« Critères de radioprotection

Les évaluations de sûreté comprendront la détermination des expositions individuelles exprimes en équivalents de dose. On supposera la constance des caractéristiques de l'homme (sensibilité aux rayonnements, habitudes alimentaires, conditions de vie, connaissances générales sans prise en compte de progrès scientifique, notamment dans les domaines technique et médical).

Il faut distinguer les expositions pouvant résulter du stockage en conditions d'évolution normale de référence et les expositions potentielles susceptibles de résulter d'événements aléatoires venant perturber l'évolution du stockage.





Situation de référence

- Les équivalents de dose individuels devront être limités à 0,25 mSv/an pour des expositions prolongées liées à des événements certains ou très probables. Cette valeur correspond à une fraction de la limite annuelle d'exposition du public en situation normale.

- Au-delà de cette période de stabilité de la barrière géologique, les incertitudes sur l'évolution du stockage augmentent progressivement avec le temps; l'activité des déchets aura notablement décru. Des estimations quantifiées majorantes des équivalents de dose individuels devront alors être faites. Elles seront éventuellement complétées, par des appréciations plus qualitatives des résultats de ces estimations, au regard des facteurs d'évolution de la barrière géologique, de façon à vérifier que le relâchement des radionucléides ne conduit pas à un équivalent de dose individuelle inacceptable. Dans cette vérification, la limite de 0,25 mSv/an précédemment citée sera conservée comme référence. »

RFS.III.2.f does recommend only one criterion (0.25 mSv/yr). In the Dossier 2005, the acceptability of the calculated impacts was then assessed case by case keeping in mind that:

- the criterion of 0.25 mSv/yr was taken as one benchmark among others, i.e. the calculation compares the results with this value, but it is not mandatory that the calculated dose comply with this limitation ;
- up to a few mSv/yr (10 at most), the impact can be regarded as acceptable on a case by case basis, provided the situation(s) described are sufficiently unlikely. In any event, even if the impact is considered acceptable, one seeks to reduce it by appropriate means, if any;
- particular attention must be paid to any calculated impact of more than 10mSv/yr. The scenario may be too over estimating; design methods that would prevent such a situation must be carefully examined. In the case of purely hypothetical situations, such an impact is not necessarily unacceptable as such, insofar as it does not refer to a situation that could actually arise.
- •

SECTION 3: KEY TERMS AND CONCEPTS

Impact calculation : In the Dossier 2005 Argile, impact calculation designates a performance calculation of the disposal (during operating and post-closure phases) conducted up to the environmental impact, usually based on conservatives hypothesis.

The radiological impact is characterised by the **Individual effective dose (annual)** committed (Sv/yr).

For the long term, the main **safety indicator** remains the dose at the outlet within the context of a predefined biosphere and a predefined critical group. A dose of 0.25 mSv/year at most in a normal situation set by the RFS III.2.f is retained by Andra.

Some indicators allow assessing the performance of individual component with respect to their safety functions. Performance is understood as:




Performance of the repository: Performance of the function "protection" of the disposal system.

Performance (as used in the dossier 2005 Argile): characterise a function. It is established by the designer according to criteria defined by the users. Among the analyzed indicators are:

 $(\ensuremath{\textbf{v}})$ the relation between convective and diffusive flux in the repository and the host rock,

(vi) the overall activity leaving the waste packages, the underground structures and the host rock, as compared to the initial quantity contained in the waste packages,

(vii) the activity flux at each of these components,

(viii) the concentration distributions of dissolved materials in the host rock and in surrounding formations.

Such indicators have been used in the safety analysis carried out in the Dossier 2005 Argile. They are developed in detail in the following section 4.

SECTION 4: APPLICATION AND EXPERIENCE

The methodology consisted in identifying pertinent indicators for assessing the three main safety functions:

1) resisting water circulation,

2) limiting the release of radionuclides and immobilizing them in the repository,

3) delaying and reducing the migration of radionuclides.

They are all tightly linked to the objectives of the safety functions. For that reason the overall methodology is explained together with the application made for the dossier 2005. Some illustrations of those indicators are given to sustain explanations in the following paragraph.

> Various and complementary indicators have been used for assessing and/or quantifying performances associated with the « **resisting water circulation** » function:

• The theoretical non dimensional Péclet (Pe) number, characterising the comparison of diffusive and advective transfer kinetics. For numbers greater than 2, advection becomes dominant. The theoretical Peclet number is equal to the ratio of the characteristic diffusive migration time (Td) over the advective transfer time (Tc). See Figure 3.







Figure 3 SEN – Distribution of « advective » and « diffusive » transfer paths

- **advective and diffusive flow indicators**, that provide a comparison of flow on exit from the argillite, around the repository ;
- Distribution between the radionuclide mass transiting along and/or in the structures made up of drifts and shafts and the mass migrating by diffusion in the unaltered Callovo-Oxfordian, before reaching the top or the bottom of the formation. In fact one of the main objectives of this function is to avoid that the system of drifts and shafts does not constitute a preferential path for radionuclides up to the biosphere. Taking into account the geometry of the repository (important horizontal extension with respect to vertical extension), migration should mainly take place in the geological barrier based on the vertical direction. Radionuclide flow exiting the shafts after having transited in the structures should therefore be negligible in face of the radionuclide flow reaching the top of the Callovo-Oxfordian after having migrated in the geological barrier. See Figure 6 and Figure 7.

> Pertinent indicators for assessing the « **limiting the release of radionuclides and immobilizing them in the repository** » function are deduced from the objectives associated with this function. It consists of :

- Prohibiting water arrival on waste (C, spent fuel) to avoid any release of radionuclides till the temperature of the waste or the surrounding medium is greater than the acceptable threshold (see chapter 3 of TES). Analysis of consequences associated with the premature release from C waste packages or spent fuel after initial failure of one or more containers allows us to asses, by difference, the interest of such provision at the scale of a package (the package failure alternate evolution scenario (SEA) will in addition highlight this aspect in section safety function. It is in fact possible to compare the attenuation functions in constructed components located in the field near the packages, but also in the geological barrier, between prematurely released radionuclides and radionuclides released at the end of containers' sealing period;
- Resisting transport of dissolved species in the vicinity of glass and spent fuel; this
 function is mainly ensured by the presence of mediums with **low diffusion coefficient**and permeability around waste (swelling clay buffer, if used, disturbed CallovoOxfordian (EDZ), plug...) that should induce diffusive transport in C waste disposal
 cells and spent fuel disposal cells. Analysis of the « resisting water circulation »





function has highlighted that effectively we were in **diffusive regime** in the C waste and spent fuel disposal cells with the **Péclet numbers** much less than 1 in disposal cell head as well as at the plug level than in the micro-fissured zone. The possibility of some C waste disposal cell plugs being defective was included in the analysis. Corresponding calculations show that the hydraulic regime is not modified in the disposal cell, due to redundancy ensured by sealing of drifts ;

- Limiting alteration of waste and consequently the release of radionuclides. Compliance with this objective is ensured by heat, water, mechanical and geochemical environmental conditions that are favourable and adapted to each waste type. The interest of this function can be assessed by difference when we conduct sensitivity analysis towards stronger release models for waste packages;
- Limiting the solubility of elements released by the packages. The performance of this part of the function is retranslated in the model by solubility limits applied to different radionuclides. This function plays a role for radionuclides such as actinides or selenium 79. The latter element illustrates the influence of the function especially well. In fact it is very similar to iodine, both representing non sorbed anions. The main difference between them is that in its reduced form, selenium has very low solubility in comparison with iodine. Comparing the masses of each of these radionuclides that migrate outside the disposal cell, with respect to the released masses, illustrates the precipitation effects well.

In the final analysis, this study allows to evaluate the added benefit of an overpack with respect to radionuclide transfer, by studying the results obtained with a failed overpack.

> Performance associated with the « **delaying and reducing the migration of radionuclides** » function in the repository is quantifiable with the help of three values associated with the molar flow of each radionuclide (see Figure 4):

- the maximum molar flow (? max);
- the mass (m) integrally corresponding to the « molar flow » over the simulation duration (1 million years);
- the appearance time of maximum molar flow (tmax = t(? max)).





Figure 4 Values retained to assess the delay and attenuation

Comparison of values for each of these indicators, between two different surfaces (Si and Si+1), helps in assessing the confinement capability of barriers lying between these two surfaces. This is assessed as follows :

- Attenuation of the radionuclide mass that corresponds to the fraction which does not exit the considered barrier(s) (especially the Callovo-Oxfordian) over the analysis duration. It tus integrates the radioactivity decay of radionuclides in the barriers resulting from more or less long migration time. The attenuation expression of the mass of each radionuclide is : 1- mi+1 / mi⁵ (or vice versa « mi+1 / mi » if we refer to the « crossing », that is, not attenuated fraction). As a general rule, we take the mass initially present in the waste as reference. The attenuation is assessed in all the components between the last considered barrier and the package ;
- The delay, corresponding to the duration between the molar flow maxima entering and leaving a barrier ; this indicator is mainly pertinent for very long-lived radionuclides having low decay. The delay is expressed by tmax i+1 t max i⁶;
- Attenuation of the molar flow maxima which illustrates the « attenuation » aspect of the function. Expression of the indicator is Φ_{max i+1} / Φ_{max i⁷}; it lets you assess the order of magnitude of the maximum flow emitted by the packages for each radionuclide.

Near field can also contribute to "delaying and reducing the migration of radionuclides". For instance:

- Elements strongly sorbed in the swelling clay buffer, mainly ⁹³Zr (→ ⁹³mNbm), ⁹⁴Nb, ¹⁶⁶mHo, ⁹⁹Tc, ¹²⁶Sn (retardation coefficient at least equal to 60 000);
- Selenium 79 not sorbed, but precipitating in the swelling clay buffer.



⁵ Where mi is the mass crossing the surface S during the entire calculation period.

⁶ tmax,i is the date on which maximum flow crosses the surface Si.

⁷ max,i is the maximum flow crossing the surface Si



SECTION 4: TREATMENT IN THE SAFETY CASE

The following section illustrates how indicators have been used in the Dossier 2005. For each one, an example is given, together with illustration of the information that can be obtained.

Committed Individual effective dose (annual)



Figure 5 : Effective dose from HLW

The radiological impact is characterised by the Individual effective dose (annual) committed (Sv/yr). For example, Figure 5 shows that the effective dose received by a critical group from all vitrified waste remains close to three orders of magnitude below the RFS.III.2.f value. Peak dose occurs after about 500 000 years. The only radionuclides shown to eventually leave the Callovo-Oxfordian are lodine 129, Chlorine 36 and Selenium 79.

Molar flow



Figure 6 Evolution of molar flows through the repository – ¹²⁹I – CU1 (Cox = Callovo-Oxfordian, COX exit : exit through the top and bottom of the formation, CU: spent fuel) [vii]

To illustrate this indicator, Figure 6 shows the evolution of the activity flow in different repository points (in the « package source term » release representation, we see two curves corresponding respectively to the failed packaging – beginning at 200 years and that due to the remaining inventory – beginning at 10 000 years) as a function of time for the entire UOx spent fuel zone. We observe that the flows on exiting the seal structures or shafts are clearly



more delayed than those at the roof of the host formation, which themselves already appear late. The flows on shaft exit arrive at their maximum after 800 000 to 1 million years, against around 250 000 years for rock.



Distribution of radionuclide mass

Figure 7 SEN – Distribution of mass through different calculation compartments (iodine 129 of CU1) (COX = Callovo-Oxfordian)

The quantitative assessment of distribution between the transfer paths through structures and unaltered Callovo-Oxfordian is based on the radionuclide (1291), Figure 7. In fact, this soluble long-lived and not sorbed radionuclide illustrates the transfer effects due to water, since it is almost insensitive to the medium's chemistry. The case of spent CU1 fuel has been used as an example, since they present the strongest iodine 129 inventory. Calculation highlights that the majority of the mass finally takes the transfer path through unaltered Callovo-Oxfordian. In fact the example concerning iodine of spent CU1 fuel gives the following conclusions.

Nearly the entire released mass (99.999 %) exits by the top or the bottom of the Callovo-Oxfordian after having migrated by diffusion in unaltered Callovo-Oxfordian ; in fact :

- 41 % of the mass emitted by the packages reaches the Callovo-Oxfordian directly after having migrated in the disposal cell's body ;
- 59 % of the mass emitted by the packages reaches the drifts by diffusing through the structures located between the packages and the access drifts (especially the clay plug). This distribution corresponds to a pessimistic estimation of what can migrate in the access drifts. It results from limiting conditions (null concentration imposed around the plug) that induce high concentration gradient between the package and the access drift, thereby, at the disposal cell's scale, favouring horizontal transport through the plug and the damaged near-field of the Callovo-Oxfordian up to the access drifts.





Maximum molar flow

For vitrified waste, the reference concept does not include a swelling clay buffer. Given that the geochemical characteristics that are very similar between the Callovo-Oxfordian host rock and the swelling clay, we do not expect any significant difference, from the point of view of near-field transfer, between the reference concept and the variant with swelling clay buffer. Also, in the near-field of the disposal cells, radionuclides whose molar flow maximum emitted by the packages is most reduced are the same as for the spent fuel (93 Zr (\rightarrow 93m Nbm), 94 Nb, 166m Ho, 99 Tc, 126 Sn) ; at about 7 metres in the Callovo-Oxfordian (distance conventionally chosen for ease of calculation), their molar flow is completely attenuated. As an example, for C2 reference packages and at the same distance, the most mobile radionuclides (129I and 36CI) present the maxima of molar flow reduced by about two orders of magnitude.



Figure 8 Preparatory SEN calculation - Concentration mapping of ²³⁷Np and ²³³U – in the near field at 200,000 and 500,000 years







Figure 9 Historical molar flow rates out of the repository at 7 metres from the cells – Reference package CU1 Uox3 (spent fuel).

The results confirm that the Callovo-Oxfordian has a very good capacity to « restrict the release of actinides and immobilise them in the repository » and « delay and attenuate their migration » due in particular to their very high retention (sorption and precipitation) in argillites. This phenomenon leads to almost total confinement in the near-field of the cells over the next million years. It is noted in particular that :

- After a few hundred thousand years, the actinides will only have covered a few metres in the geological barrier. Figure 8 illustrates this strong confinement for ²³⁷Np, which is one of the least-sorbed actinides, and ²³³U (both belonging to the 4N+1 filiation chain).
- Molar flow rate history in the Callovo-Oxfordian at 7 metres from the disposal cells, evaluated for the four actinide chains, confirm this very strong attenuation. Only ²⁴²Pu, ²³⁹Pu, ²³⁷Np, ²³⁸U, ²³⁵U, ²³³U ²³⁶U and ²²⁹Th in secular equilibrium with its ascendants present a molar flow rate in excess of 10⁻¹² mol/year (see Figure 9) for a three-package cell. The flor rates are nonetheless weak even for these isotopes. As an example, only 11 grams of ²³⁷Np, 2.3 grams of ²³⁸U and 0.0005 grams of ²³⁹Pu per cell covered more than 7 metres of argillite over the total analysis period (one million years).

The appearance time of maximum flow

It can be seen in Table 1 that diffusion properties in the Callovo-Oxfordian are nevertheless good enough to strongly delay the appearance of maximum of molar flow at the formation's exit, at the approximate scale of 200 000 years. Chlorine 36 and iodine 129 have a similar behaviour (not sorbed soluble anions). Iodine 129 has a long half-life with regards to the diffusive transfer time (1.57 10⁷ years), and decays only very little during its migration. The date when its maximum appears on exiting the host formation, 260 000 years for CU1⁸



⁸ CU : spent fuel



packages, is therefore the direct expression of its transfer time. On the other hand, chlorine 36, decays during its migration (its half-life is 300 000 years). The date when its maximum flow appears, 180 000 years, for CU1 packages is therefore earlier than that for iodine since it combines the effects due to migration and those due to radioactive decay.

Reference package	Radionuclides	Dates of molar flow maxima at the top of the Callovo-Oxfordian [years]
CU1 ⁹	129 I	260 000
	36Cl	180 000
	79 Se	400 000
C1/C2 ¹⁰	129 I	460 000
	36Cl	380 000
	79 Se	750 000
B2 ¹¹	129 I	465 000
	36Cl	200 000
	79 Se	165 000

 Table 1 SEN – Appearance dates of molar flow maxima on exiting the Callovo-Oxfordian for the three main impact contributors

SECTION 5: LESSONS LEARNED

KNOWLEDGE/EXPERIENCE GAINED WITH THE DEFINITION AND USE OF SAFETY AND PERFORMANCE INDICATORS

At this stage of the analysis, it is possible to appreciate the performance of safety function.

We observed that :

- The « resisting water circulation » function is efficiently ensured, since access paths to the repository are not the preferential migration paths. The sensitivity studies and the calculations of the altered evolution scenario allow appreciation of the robustness by testing the respective contribution of seals, host formation and the cul-de-sac architecture ;
- The « limiting the release of radionuclides and immobilizing them in the repository » function allows the retention of elements having weak solubility, in the C waste and spent fuel disposal cells. In addition, the interest of the function vis-à-vis management of heat transfer is important qualitatively (management of uncertainties in this field), but cannot be translated in terms of impact limitation with the data used at present;

¹¹ B : bitumen waste



⁹ CU : spent fuel

¹⁰ C1/C2 : vitrified waste



• The « delaying and reducing the migration of radionuclides » function highlights the preponderant importance of the host formation, which limits to four the radionuclides that can effectively exit at the end of over 150 000 years (in fact, mainly two: 129 and 36Cl). Due to its (weak) iodine sorbing capability, the B disposal cell concrete has a visible role. In normal situation, the clay engineered barrier of spent fuel does not contribute additional efficiency with respect to an equivalent thickness of Argillite.

The radionuclides that have exited from the repository system can induce an impact, the effective dose, as calculated in Figure 5.

ON GOING OR PLANNED PROJECTS

The approach used in the Dossier has been judged as an interesting tool to be developed. The so called AF/APSS/QSA approach is to be reconducted in future Andra's safety cases. It will be developed in order to take into account the evolution of the project (new act in June 2006, revision of the RFS III.2.f, site...). As for the dossier 2005, other indicators than dose will be used in order to appreciate the performance of safety functions.

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A2 AVN (Belgium)





European integrated "PAMINA" - Project WP 1.1 – AVN Contribution "Safety indicators"

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Examples of Safety Indicators

used by ONDRAF / NIRAS in SAFIR 2 Report

- 1. The decayed fraction10
- 2. Radionuclide fluxes to the aquifer
- 3. Total uranium inventory

ANNEX 2

Discussion on timeframe handling



1. Introduction

One of the aims of a safety report for a radioactive waste disposal is to assess the long-term radiological impact associated to the repository. Generally, parameters like dose or risk are used for carrying out this assessment.

Mainly the dose assessment is the result of two modelling stages. The first one allows the determination of the radionuclide flux at the interface geosphere - biosphere. The second one addresses the biosphere.

As it has already been demonstrated by many international studies on the long-term confidence in the disposal system components, some disposal components are more robust than others with respect to their long-term behaviour. For these more robust components, the variability of their characteristics, their evolution through time and the associated level of uncertainty can be regarded as more or less constant during a considered period of time. Likewise, amongst the parameters that can be used for characterizing the level of safety of a disposal system, some are more robust than others. Unfortunately, it is well known that dose or risk, as end-points of the safety assessment, do not belong to such robust parameters. This is mainly due to the high level of uncertainty attached to the evolution of the biosphere in the long-term (see figure 1 from SAFIR II report [1]).



With time, the confidence of the predicted properties of dose or risk diminishes. This is one of the reasons for which alternative safety indicators have been introduced.

Figure 1. Robustness of the main components of the disposal system and its environment [1].

In the present note, we will distinguish the two main safety indicators dose and risk from other safety indicators. The shortcomings identified earlier justify this distinction. A short



discussion on the use of dose and risk as safety indicators will be provided in chapter 3. However, considering that an international agreement already exists on the use of dose and risk as fundamental safety indicators, AVN decided not to put too much effort on this point but rather to focus on the alternative safety indicators which can be used, and on which an international agreement still needs to be reached.

2. Definition of terms and used concepts

Prior to discussing the use of safety indicator, we think necessary to precise our understanding of the concept of safety indicators. According to us, a safety indicator should be considered as key information for the decision making process in order to proceed to the next step. A safety indicator refers to a particular indicator that assesses the level of implementation of the safety strategy adopted by the implementer, addressing either the whole disposal system or a part of it. In this approach, some safety indicators can be compared to specific reference values, when they exist.

From the abovementioned definition, we clearly distinguish "Safety Indicator" from "Performance Indicator". The latter only aims at providing a measure of the level of quality, reliability or effectiveness of a given component of the system (or even of the whole system) looking at some technical specifications but without any direct link with the safety, while the former is intended to address the long-term safety of the disposal system by quantifying the level of implementation of the safety strategy.

The abovementioned definition has to be sustained by providing some "a minima" properties of Safety Indicators, especially on how to derive them from the safety strategy and by identifying what should be kept in mind when defining the associated reference values (complementary information will be provided in section 3.2 of the present document and will then be further developed by AVN in the framework of WP 3.4 of PAMINA).

As an application example, dose and risk comply with the preceding definition and are the most common safety indicators used in long-term safety assessment for radioactive waste disposal facilities. These two indicators inform the decision maker on the potential hazard (to human health) generated by the repository. As such, they constitute highly aggregated indicators that can be compared to reference values fixed for radiological protection.

Their use presents some evident advantages, but they also have some shortcomings e.g. with respect to the high level of uncertainties attached to their calculation in the long-term, which justifies other safety indicators to be used in complement.

3. Regulatory context

The national regulatory context in Belgium is developed by the Federal Agency for Nuclear Control (FANC) and its technical support AVN, starting from international regulations or standards issued by the IAEA, the ICRP and the European Union (Directive 96/29/Euratom).

The main principles set out by IAEA in its publication 111-F [2] are derived in 8 principles applicable to the management of radioactive waste on the whole Belgian territory. They are defined in a document named "strategic note related to the licensing procedure for radioactive waste disposal facilities", issued by the Belgian Nuclear Safety Authority in March 2007





Explanations about positions of the Belgian Nuclear Safety Authority on major issues linked to the long-term safety of radioactive waste disposal facilities can also be found in the document [3], written in collaboration with the French Nuclear Safety Authority and both the French and Belgian operators. The present note refers several times to this document, as it addresses, amongst others, the topic of safety indicators.

3.1 Regulations and guidance

So far, the existing regulation in Belgium does not impose the use of any particular safety indicator, although it is obvious that dose remains one of the most important one to be used in safety demonstrations, as indicated in the following paragraphs.

3.1.1 Dose and risk safety indicators

In document [3], which reflects basically the position of the Belgian Safety Authority, the necessity of reducing the **doses** resulting from a disposal facility to a fraction of the dose limit for members of the public is clearly mentioned. For doing so, reference is made to the values recommended by ICRP in its Publication 60 (0.3 mSv/yr as the maximum dose constraint and its risk equivalent of 10-5/yr considering reference scenarios¹²).

As concerns doses, ICRP recommends in its publication 81 that in the context of the assessment of the long-term performance of a repository, the individual doses should be preferred to the collective dose, as collective doses deal with high uncertainties such as the evolution of the size of the populations in the future. However, qualitative estimate of the sizes of the populations in question can also provide useful information to supplement the results presented in terms of individual doses, as stated in [3].

On a methodological point of view, for the use of risk, which combines the radiological consequences and the probabilities of the occurrence of events, it is recommended in document [3] to present separately the radiological consequences and the probability of occurrence, as this presentation contributes to the comprehension of the safety assessment in the context of a decision-making process.

3.1.2 Others Safety Indicators than dose or risk

No specific guidance or recommendations exist on the topic of using other safety indicators than dose or risk. Moreover, this item is not identified as a particular chapter of the safety assessment report.



¹² It is reminded that numerical values fixed in the regulation may have different status according to the characteristics of the considered scenarios. For instance, they can constitute real limit values to be strictly met for reference evolution scenarios, whereas they can generally be used only as comparison points for less probable scenarios.

This does not mean that the applicant is not allowed to develop qualitative and/or quantitative substantiations on this matter in order to sustain his safety demonstration, but it only means that such information is not required. From a regulator point of view, we also consider that the concept of safety indicator is not mature enough for developing a particular guidance or recommendation on their use at that stage.

3.2 Requirements and expectations

Before writing down any regulator's expectations concerning safety indicators, AVN, as technical support of the Belgian safety authority, deems necessary, in a first step, to clarify the use and the role of safety indicators in the safety demonstration, and the associated properties they should possess. In that concern, we clearly identify the need for trying to answer the following questions:

- What considerations should drive the choice of the safety indicators? What should be the place / weight of the choice of the safety indicators in the safety strategy developed by the operator? - Should safety indicators necessarily refer to quantitative parameters? Would it be possible to define some sort of safety indicators for assessing more qualitative elements of the safety strategy, such as for example the robustness of the system, its passivity, its simplicity or its demonstrability?
- What intrinsic properties a parameter should possess to be qualified as a safety • indicator?
 - Is there a need for being able to associate a level of human detriments to each • safety indicator?
 - In which matter should an alternative safety indicator be independent of dose and risk assessments (so that the alternative safety indicator would not have the same shortcomings as the fundamental safety indicators)?
- Can a safety indicator be a global indicator integrating the contribution of all radionuclides or could it be specific radionuclide by radionuclide?
- Should a safety indicator be an indicator of a collective impact or an indicator of an individual impact on human health?
- Are there any requirements on safety indicators concerning the time handling?

The results of these reflections will in a second step help AVN to better identify its regulatory expectation's concerning safety indicators.

Probably the main regulator's expectations will lie in the first steps of the licensing process, in the justification, by the operator, of the relevancy of the proposed safety indicators which will be used in the safety assessment, as regards their applicability in the considered timeframes, the level of uncertainty attached to them, their level of aggregation...

AVN plans to make benefit of the opportunity offered by the PAMINA project and its involvement in RTDC-1 and RTDC-3 to pursue its reflection on safety indicators.





3.3 Experiences and lessons learnt

As mentioned already in section 1, dose or risk, as end-points of the safety assessment, are not robust indicators.

ONDRAF/NIRAS thus proposed in "Safety Assessment and Feasibility Interim Report 2" (SAFIR 2) developed for geological disposal [1] to introduce the following alternative safety indicators, further detailed in annex 1:

- The decayed fraction;
- Radionuclide fluxes to the aquifer;
- Total uranium inventory.

No specific review has been made so far by the Belgian Safety Authority on the use of those parameters as safety indicators. Nevertheless, if the same safety indicators were to be used in the future safety cases the operator will submit in the framework of the geological disposal programme, it would be interesting to try to evaluate in what extent these indicators could comply with the expectations expressed in section 3.2.

Finally, we consider that the work carried out by both the French and the Belgian nuclear safety authorities together with the French and Belgian operators for establishing document [3] also contributes to our experience in reviewing safety cases of radioactive waste disposal facilities. In this work, it has been highlighted that in accordance with the application of the principle of optimisation, the evaluation of conformity with radiological protection objectives cannot be reduced to a simple comparison of the calculated doses or risks with the dose or risk constraints. This evaluation of conformity with the basic objective of protection is the result of a process that is based on a judgement and in which the calculated doses or risks are one of the elements to be taken into account, together with other considerations like the likelihood of the scenarios, the overall representativeness of the modelling, and the part of the environment affected by the release of activity as well as the size of the population potentially exposed.

3.4 Developments and trends

From the Belgian Safety Authority, no particular regulatory trend has been mentioned so far concerning the use of safety indicators, since as stated before, the tendency is rather to let the operator free of defining his own list of safety indicators, providing that the rationale behind it is clearly explained and justified.

Nevertheless, due to the recognized shortcomings of the use of dose and risk, AVN considers that there is a need for developing some considerations on the use of safety indicators in a safety case.

Consequently, in the scope of PAMINA RTDC-3 (WP 3.4), AVN proposes to try to develop the basis of a regulatory guidance concerning the use of safety indicators in a safety case.





4. Analysis and synthesis

4.1 Main advantages / possible difficulties linked to the use of safety indicators

The main advantages of the use of safety indicators lie in the fact that they provide a <u>quantitative</u> indication of the level of implementation of the safety strategy of a disposal facility, and also that ideally the assessment results of these safety indicators can be easily compared to specified reference values - when they exist -, in order to judge about the acceptability of the repository. For instance, the use of fundamental safety indicators such as dose and risk enables to directly quantify the potential biological hazard associated to the repository, which is essential to check for the ability of the disposal system to meet the fundamental objective of protection of human health against the effects of ionizing radiations arising from the waste disposed of.

As such, there is no doubt that safety indicators constitute an important element in the decision-making process.

From our experience about safety indicators, it appears that they could be easily used for communicating about the safety of the disposal to a large audience of stakeholders. In the context of the decision-making process, this feature gives more weight to the safety indicators.

According to AVN, besides dose and risk, other crucial, more quantitative and/or qualitative elements also intervene in the decision-making process as elements of the safety strategy such as for example, the robustness of the system, its passivity, its simplicity or its demonstrability.

Recognizing the specific issues of using dose and risk in a safety assessment, we also recognize that the use of other alternative safety indicators presents also their own difficulties. For example:

- Some safety indicators related to the safety principles may not have a direct link to human health. For some people, this absence of assessment of detriment to the human health could make these safety indicators less relevant or more difficult to use and understand than dose and risk.
- A well identified difficulty associated to the use of whatever safety indicator lies in the definition of the reference values which will be used as comparison points for judging it¹³.
- Another possible issue related to the use of non-radiological references could be the necessity of involving other authorities in order to validate these references. With work



¹³ To cope with it, NEA recommends to consider, as possible starting points for the definition of those reference values, either acceptable hazard (as for dose and risk) or negligible disturbance of nature (e.g. disturbances to the fluxes or concentrations of naturally occurring radionuclides that take place within natural systems). According to AVN, these starting points do not refer directly to the safety strategy. The compliance with our definition of safety indicators should be more carefully examined.



progress on the topic of Safety Indicators, the exact nature of this issue will be more precisely investigated.

Based on the further developments intended in the scope of PAMINA, some improvements on the use of the safety indicators and their intrinsic limitations may arise.

4.2 Communicating the safety of a deep geological disposal and the use of judgement of acceptability

In addition to the already identified weaknesses of dose and risk, dose and risk are generally regarded as not very understandable for non-technician public.

Hence, an argument commonly used for justifying the use of safety indicators other than dose and risk is that these concepts could be more understandable for non-technician public. This is of particular interest when presenting the results of the safety demonstration to the stakeholders involved in the decision-making process. The fact that safety indicators are suitable for communication, is a specific property of the Safety Indicators. This property is identified as one of the major advantages of them.

However, one should be aware that the possible problems of comprehension the public may have while consulting the safety assessment results should not be an excuse for according less importance to aggregated indicators, such as dose and risk, that give a direct link between the state of a disposal system and a human hazard.

Looking at the international developments on safety indicators, in some circumstances, one should be aware that confusion could exist in the use of safety indicators between the aim to communicate easily about the safety to stakeholders and to the objective of assessing the safety through the use of safety indicators. According to us, the positive feedback of safety indicator on communication should not progressively become more important than the original aim of using safety indicators in a safety case. The fundamental substantiation of the use of safety indicators should remain that through them operator should be able to demonstrate the safety of its concepts using other parameters than dose and risk.

4.3 Uncertainties and time frames handling

This paragraph should be developed in a further step. Example of discussion on this topic is provided in annex 2.

4.4 Improvement, Integration and harmonisation potencial

Taking into account the questions raised in the present note, some possible improvement potentials can be identified for the use of safety indicators. First of all, the expectations regarding safety indicators, the intrinsic properties they should possess as well as their role in the safety demonstration should be defined more clearly, and that is the objective of the questions asked in section 3.2 of the present note.

Based on the development made in this document, it appears that safety indicators appear to be a valuable tool for integration of safety case arguments. Safety indicators constitute a framework for developing qualitative arguments in a safety case without decreasing the







added value of quantitative arguments. For example, developing safety indicators will probably induce the use of some specific scenarios in order to assess them.

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ANNEX 1

Examples of Safety Indicators used by ONDRAF / NIRAS in SAFIR 2 Report

In Safety Assessment and Feasibility Interim Report 2 (SAFIR 2) developed for geological disposal [1], ONDRAF/NIRAS proposes to introduce the following alternative safety indicators:

1. The decayed fraction

A significant fraction of the radionuclides disposed of in the repository disappears before it can reach the biosphere, through radioactive decay in the waste form, during migration through the near field and during migration through the host formation. The decayed fraction is defined as the ratio of the amount of radionuclides that decays in the disposal system (before they are able to reach the aquifer or biosphere) to the amount initially disposed of. This can be expressed as a percentage. The amount which does reach the aquifer can then be calculated as the integral over time of 0 to 100 million years (the maximum time considered in the calculations) of the total flux to the aquifer (the calculation makes no allowance for radioactive decay in the aquifer). Finally, the containment factor offered by the disposal system, defined as the ratio of disposed activity to cumulative released activity in the aquifers, is calculated to illustrate the efficiency of the system to fulfill the two safety functions 'physical containment' and 'delaying and spreading the releases'.

2. Radionuclide fluxes to the aquifer

Just as with the calculated doses, these calculated radionuclide fluxes can be used as a safety indicator for the disposal system. Just as a calculated dose can be compared with the specified dose limit and dose constraint or with the dose due to natural background radiation, so radionuclide fluxes can be compared with the concentrations of fluxes of naturally occurring radionuclides, whether in the actual disposal system or elsewhere.

ONDRAF/NIRAS also mentions that if radionuclide fluxes are used as safety indicators, then steps similar to calculated doses can be followed:

- The first step is to compare the calculated fluxes with natural radionuclide concentrations or radionuclide fluxes;
- Then the intrinsic radiotoxicity of the fluxes from the repository and from the naturally present radionuclides can be compared (e.g. by allowing for the dose coefficients for ingestion and inhalation);
- Finally, ways for applying the ALARA principle with the help of the safety indicator used can be investigated.

3. Total uranium inventory

In SAFIR 2 report, ONDRAF/NIRAS indicates that the migration calculations as well as the dose calculations make clear evidence that the radiological impact of the actinides present in





the radioactive waste will only occur in the very distant future, i.e. after 1 million years. This poses a problem, in as much as the uncertainties in these migration and dose calculations become very great at this long time in the future. It is evident, however, that the impact in the far future will be determined by the U and Pu isotopes and their daughter products (including 226 Ra, 232 Th, 231 Pa).

ONDRAF/NIRAS thus considers interesting to compare the original U inventory in the highlevel and long-lived waste with the α activity already present in the host formation, taking into account the provisional volume of the geological formation that will host the repository.





ANNEX 2

Discussion on timeframe handling

The choice of different safety indicators for different timeframes could be an appropriate way of dealing with the increasing level of uncertainties through time concerning the possible evolution of the disposal system:

The IAEA working group on principles and criteria for radioactive waste disposal proposed in 1999 ([4]) to consider the following time intervals for identifying the more relevant safety indicators to be used in the safety assessment:

- In the first period following closure of the disposal (from closure of the repository up to 10⁴ years), although major changes in climate and human habits could occur, in general, the biosphere could be assumed for radiological protection purposes to remain comparable to present day conditions. **Dose** and **risk** can then be calculated during this period as main safety indicators, considering habits corresponding to those currently observed in the region.
- Following the first period described above, i.e. from 10⁴ to 10⁶ years, a glaciation event is expected to occur, that would bring about significant changes to man's environment. The range of possible biosphere conditions and human behaviour will thus be too large to allow reliable modelling. The calculations relating to the near-surface zone and human activity can be made on stylised sets of conditions. These calculations should be viewed as illustrative and the doses or risks as indicative. In this time frame, other safety indicators, requiring less information about near surface conditions, the biosphere and the human behaviour, will play an increasing role in assessing repository safety.
- For the period beyond 10⁶ years, unpredictable and/or large-scale changes could take place such as continental drift and massive erosion. Therefore, less credibility can be attached to assessments in this time frame than to assessments in earlier time frames. The use of dose and risk can only be purely indicative for such periods of time.

Furthermore, it is important to keep in mind that each safety indicator also has its own level of intrinsic uncertainty, as it only provides a representation of the level of safety of the considered disposal system and does not constitute a directly measurable quantity.

For instance, due to the fact that dose is the end product of a calculation of an exposure scenario based on simplifying assumptions and some sort of stylisation to compensate for the lack of knowledge on the evolution of the system and its environment in the long-term, the calculated dose cannot be regarded as a real prediction of future health consequences, but only as an indicator of the associated impact, with a set of particular hypotheses used for the purposes of the evaluation.

The value of the information provided by this dose indicator is related to these hypotheses and may vary significantly in accordance with the time scale and the scenario considered, and also depending on the confidence that can be placed in its evaluation. In particular, for some scenarios, it is possible that the assessment of the impact will consider highly stylized and pessimistic hypotheses on account of the lack of knowledge.





A3 DBETEC (Germany)









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- PU
 Public

 PP
 Restricted to other programme participants (including the Commission Services)
 X

 RE
 Restricted to a group specified by the consortium (including the Commission Services)
 X

 CO
 Comfidential only for members of the consortium (including the Commission
- CO Confidential, only for members of the consortium (including the Commission Services)

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This document describes the experience of DBE TECHNOLOGY GmbH regarding the use of safety and performance/function indicators in the context of licensing procedures of a radwaste repository in a deep geological salt formation in Germany. The scope of the present document is to highlight that there is a different meaning of the terms safety and performance/function indicator due to different regulatory fields which have to be regarded when licensing a radwaste repository in a deep geological formation in practice.

The terms safety and performance/function indicator are not used exclusively in PA, but in different regulatory fields. A detailed description is given below.

Introduction

Within the SPIN Project safety and performance indicators for different post closure timeframes have been discussed, requirements and expectations for appropriate indicators have been established, indicators were proposed and their effectiveness and applicability discussed. As a result of the SPIN project, preliminary recommendations for the use of safety indicators were established. They are summarized in Table 1 [1].

Indicator	Recommendation	
Effective dose rate	Continued application for all time periods Higher weighting at early time periods	
Radiotoxicity outside geosphere	Not to be used, unless safety related reference values are identified	
Time-integrated radiotoxicity flux from geosphere		
Radiotoxicity concentration in biosphere water	Preferred application for later time periods	
Radiotoxicity flux from geosphere	Preferred application for very late time periods	
Relative activity concentration in biosphere water	Not to be used, unless a new weighting scheme will be found	
Relative activity flux from geosphere	Apply if you asked to do so	

Table 1: Preliminary recommendations for the use of safety indicators

At present, there is an ongoing controversial discussion in Germany on safety and/or function/performance indicators besides dose and risk in the post-closure phase of a radioactive waste repository in a deep geological formation. The indicators proposed by GRS are based on the isolation of the waste inside an intact isolating rock zone and not on dose rate constraints for deep repositories which are between 0.1 and 0.3 mSv/a [2].

One reason for experts taking up different positions in the discussion is that when licensing a deep geological repository for radioactive waste not only radiological protection objectives have to be met but conventional protection objectives as well. There may be conventional protection objectives arising from the water protection act (e.g. applied to toxic waste repositories and backfill) as well as from the federal mining law, which have to be regarded in addition to radiological protection objectives of the atomic energy act. As the atomic energy







act, the water protection act, and the federal mining law are on the same regulatory level, protection objectives in these three fields have to be fulfilled in order to receive a license in practice.

This conclusion is based on the practical experience gained during the preparation of documents for licensing the closure of the Morsleben repository. The Morsleben repository for LILW represents a radwaste repository in a deep geological rock salt formation. The closure concept as well as the conventional and radiological protection objectives is described in [3].

Due to the different historical evolution of the federal mining law, the water protection act, and the atomic energy act, the protection objectives of these regulations differ and the manner to prove compliance with the respective protection objectives different approaches have to be used showing that the post-closure protection indicators are met which are of different nature as a consequence of different protection objectives.

Basically, it is not a problem to prove compliance with protection objectives in different regulatory fields if the protection objectives as well as the protection indicators are clearly different. In practice, problems arise if indicators are used that cannot be clearly and unambiguously assigned because different definitions of indicators may be available regarding the protection objective in the different regulatory fields. These definitions are given in the following and the present state is described, which may serve as a basis for further discussions.

Definition of protection objectives and related indicators

Figure 1 shows the different levels of statutory provisions for nuclear facilities in Germany. For conventional facilities a comparable structure exists as a result of the German judicial system. Different indicators are available on every level, they differ in their degree of definition becoming more and more concrete when moving from the top to the bottom. The bottom, however, is formed by a large amount of technical regulations describing state-of-the-art technology, in other words the real world of available technologies and established procedures. When defining or deriving indicators, both the top-down approach as well as the bottom-up approach can be used as the regulatory level must be based on the real world issues, however, must be regulated on abstract level.





	Laws and regulations	A Contraction of the second se
Top-down	Safety criteria Safety principles Protection objectives Basic/fundamental requirements	Bottom-up
	Guidelines/directives Criteria and methods of proof	
,	Technical regulations	

Figure 1: Levels of statutory provisions for nuclear facilities in Germany [4]

Regarding the level of statutory provisions in the paragraphs below, a top-down approach is carried out as well as a bottom-up approach and the definitions of protection objectives and related indicators are pointed out exemplarily. Finally, the consequences on the specifications of indicators are summarized.

Top-down approach - Regulatory level

At present, three main fields of regulations have to be taken into account to cover radiological and conventional protection objectives in the post-closure phase of a radwaste repository. Next, the yardsticks are listed. They are being applied, for example, in the closure of the Morsleben repository.

- Radiation protection [5]: Indicator "individual dose rate". The protection objective is the human being.
- Groundwater protection [6], e.g. near-surface, accessible groundwater: Indicator • "concentration of hazardous substances in groundwater". The protection objective is near-surface, accessible groundwater.
- Surface protection [7]: Indicator "surface subsidence". The protection objectives are • drainage capability, water, soil, cultural assets, and comparable subjects of protection. Surface protection is a generic protection objective covering a group of individual protection objectives.

Even if the protection objectives and the regulatory frameworks are different they have a common characteristic. The protection objectives are listed on the regulatory level. The protection indicators used as yardsticks are independent of the site and the technical repository concept. They are linked to the protection objectives "individual", "accessible groundwater", and "ground surface". The indicators do not prescribe how to prove







compliance with the protection objective. They are suitable for regulatory issues.

As the protection objectives are different the indicators are also completely different in their nature: dose, concentration, and geometric quantity. As a consequence of the different natures of protection objectives and protection indicators the three regulatory fields are clearly decoupled. In practice, this helps in the licensing process because due to the different issues the responsible authorities can easily be identified. When mixing the issues, e.g. regarding radionuclide concentration in groundwater, the protection objective moves from "human being" to "accessible groundwater". As a result the unequivocal attribution to a specific regulatory framework is lost and the question arises whether the regulations of the water protection act or the regulations of the atomic energy act and the related regulations apply. Remark: Currently, the opposite is true in Germany. If, for example, the exemption level for radionuclides is exceeded in a facility subject to the protection objectives of the water protection act, the proof of compliance of the radiation protection ordinance replaces the proof of compliance of the water protection ordinance replaces

The mixing of the issues causes additional problems when the proof of compliance with protection objectives is performed. As water protection is a conventional state-of-the-art issue, a number of regulations and guidelines on how to proceed and how to prove compliance with the protection objective, e.g. when disposing chemotoxic waste, are available. If isolation depending on the site and the technical concept is shown to be state-of-the-art, the quantity of chemotoxic waste inside the repository is not limited as this is unnecessary in case of complete isolation. If the proof of compliance is based on the state-of-the-art, the likely repository evolution - with some variation within certain limits - is considered. A timescale of 10,000 years is regarded.

For a radwaste repository, however, radiation protection has to be shown according to the state of science and technology. As a consequence, repository evolutions with a low probability are considered as well. A timescale of 1 million years is regarded.

If the same protection objective is used - in this case accessible groundwater – this will lead to much confusion and a lot of questions, e.g. how to handle the different timescales or manners of proof – state-of-the-art versus state of science and technology. At present, the consequences cannot be foreseen and the question of how to communicate the differences remains open as well.

Safety and performance/function indicators - regulatory level

Within the context of radwaste disposal safety indicators and performance/function indicators are distinguished. In [1] they are characterized as follows:

A safety indicator must

- Provide a statement on the safety of the whole system
- Provide an integrated measure describing the effects of the whole nuclide spectrum
- Be a calculable time-dependent parameter
- Allow comparison with safety-related reference values





A performance/function indicator must:

- Provide a statement on the performance of the whole system, a subsystem or a single barrier
- Provide a nuclide-specific or integral measure
- Be a calculable, time-dependent or absolute parameter
- Allow comparison between different options or with technical criteria.

There is not a doubt that the individual dose rate fulfils the criteria of a safety indicator.

Next, the question whether the characterization of a safety or performance/function indicator can be transferred to the issues of water protection and surface protection as well is discussed. The indicator "concentration of hazardous substances in accessible groundwater" is taken as an example. The indicator provides a statement on the safety of the whole system, however, formally it does not provide an integrated measure describing the effects of all substances in combination. The integrated measure is indirectly included [6]. The concentration is a calculable parameter, the comparison with safety-related reference values is performed indirectly, too. Thus, as per the above definition the concentration of hazardous substances would be regarded as a performance/function indicator. However, the experts involved in the water protection field call them safety indicators.

Next, the indicator "surface subsidence" is discussed. It provides a statement of the whole system, an integrated measure describing the effects of mining, and it is a calculable parameter. It allows the comparison with reference values. If these values are safety-related depends on the definition and classification of the protection objective. If the protection objective is adequately defined surface subsidence will fulfill the requirements of a safety indicator. In practice, however, it is – within certain limits – regarded as a function indicator.

As a result it can be concluded that the definition of safety and performance function indicators is considered and interpreted differently even on the regulatory level.

Bottom-up approach – technical level

A final repository for radwaste is a technical structure and has to be designed in compliance with technical standards and guidelines in force. The European Standard of civil engineering is typically applied to components installed in radwaste repositories, specifically to components which are decisive for repository safety. The European Standard regulates the basic structural design [8] as well as special fields of application, e.g. geotechnical design [9].

According to the European Standard of civil engineering, a structure is to be designed and executed in such a way that during its intended life it will – with an adequate degree of reliability and in an economical way –

- Sustain all actions and influences (= the technical definition of FEPs) likely to occur during execution and use, and remain suitable for the use for which it was designed.
- Not be damaged by accidental events or impacts (= the technical definition of low probable scenarios, which have to be regarded) or the consequences of human errors





to an extent inappropriate to the original cause (= technical definition of robustness).

Technical standards and guidelines describe state-of-the-art technology. Thus, in the case of conventional protection objectives a large number of regulations exist that have to be taken into consideration for licensing, and compliance criteria as well as procedures are defined. When constructing and closing a radwaste repository technical standards and guidelines have to be taken into account. The European Standards of civil engineering (Eurocodes) are harmonized on a European level. Terms and definitions used in these standards have to be carefully taken into account to avoid confusion later on when constructing the components of a radwaste repository.

To highlight the problem, the technical definition of safety and function/performance indicators is given next. It refers to the conventional protection objectives as well as to structures of the radwaste repository, e.g. geotechnical and geological barriers serving as radiation protection as well as lining in tunnels and boreholes. In this context, it has to be kept in mind that a radwaste repository is a technical structure and must be designed according to technical standards or at least according to state-of-the-art technology.

Safety and function/serviceability indicators - technical level

Especially the terms "safety indicator" and "function/serviceability indicator" are defined in different ways. However, it has to be mentioned that every technical standard has its individual field of application, i.e. they are applied in a defined technical field. In the context of European technical standards an indicator which guarantees a safety function, e.g. barrier integrity in case of dangerous substances, is called safety indicator. It is defined by the consequences of failure. The potential failure of a structure is related to the consequences in terms of risk of life, injury and potential economical losses. If risk of life or injury is a potential consequence it is called safety indicator, if economical loss is the consequence it is called serviceability indicator which is related to the function of the system in terms of usability. Thus, the term "function indicator" has completely different meanings. According to [8], it is a synonym for serviceability indicator.

In this context, all indicators, the indicators "individual dose" and "concentration of hazard substances in accessible groundwater" would be safety indicators. In addition to this, indicators describing the barrier integrity would be regarded as safety indicators describing a safety function. For a repository in rock salt the flow resistance of the undisturbed rock salt and the dilatancy criterion proving that the rock salt barrier remains free of cracks would be called safety indicators as well. These indicators show proper barrier performance, but not in terms of radionuclide release via transport modeling. Nevertheless, it is an effective site- and concept-specific indicator demonstrating the long-term safety function of the rock salt barrier. These indicators guarantee the site- and concept-specific safety function within the field of technical standards. Evidently, these two indicators have to be used in combination. Next, two safety indicators are given that can be applied alternatively. If instead of the dilatancy criterion the more conservative fluid criterion is applied, the dilatancy criterion is replaced as both criteria can be used to prove that the salt rock barrier remains free of cracks. In this case the application of one of the two safety indicators is sufficient.

In the context of technical standards, the ground surface subsidence would be regarded as a classical serviceability indicator because it is not related to risk of life or injury. It guarantees the usability of the ground surface after closure of a mine. The lack of ability to use the





ground surface is characterized by economical losses.

This fact is of major interest because in the context of technical standards the required levels of reliability of the system are different if the failure state leads to risk of life/injury or to economical losses. According to [8] it is required that:

The choice of levels of reliability for a particular structure should take into account the relevant factors, including:

- The possible cause and/or mode of attaining a limit state
- The possible consequences of failure in terms of risk to life, injury, potential economical losses
- Public aversion to failure
- The expense and procedures necessary to reduce the risk of failure

Different levels of reliability may be adopted inter alia:

- For structural resistance (complying with the safety function)
- For serviceability (complying with a function in terms of usability)

Indicative values of failure probability are defined by an upper bound, pf < 10-6/a losing a safety function and pf < 10-2/a to 10-4/a losing a serviceability function depending on the magnitude of economical loss.

Summary

Summarizing the discussion above the following conclusions must be drawn: When specifying indicators for deep geological repositories the three main regulatory fields relevant to the post-closure phase (mining, water protection, radiation protection and their different timescales) must be checked carefully. In addition to this, it has to be kept in mind that on the technical level the terms "safety indicator" and "function indicator" have a different meaning. Thus, when specifying a new indicator the following questions have to be answered:

- Does the indicator specify a protection objective (independent of site and concept)
- Is a comparable indicator already used in the three main fields
- Is the indicator related to risk of life/injury or economical losses
- Does the indicator depend on the site and the technical concept
- Does the indicator specify a safety function or a function regarding usability
- For which regulatory level is the indicator adequate (regulatory level or technical level)

Answering the questions before a new indicator is defined avoids mixing the issues of different regulatory fields as well as mixing the regulatory and the technical level of relevance when licensing a radwaste disposal facility in a deep geological formation.

For example, checking the indicators in Table 1 in the context of the discussion only the





indicator "effective dose" is clearly a safety indicator on the regulatory level because it is related to the individual. All other indicators describe a safety function, the function of the multbarrier system which often depends on the site and the technical repository concept, which is the technical level of standards and guidelines.

An indicator-related function in terms of usability according to technical standards is not given in Table 1.

Conclusions

This contribution draws attention to the aspect that when defining new indicators, different regulatory fields of final disposal must be taken into account carefully. Indicators are often used in other regulatory fields or on different regulatory levels. As illustrated, there are typically different indicators used in different regulatory fields leading to decoupling in the licensing process. When coupling the issues or levels this may lead to serious consequences when licensing a radwaste repository in a deep geological formation in practice and may touch conventional issues as well.

Lessons learnt during the assessment of indicators for a repository: Regard every field, even conventional aspects of safety, e.g. the mining law or the water protection act. Keep in mind the technical level. Often an indicator is already used in a regulatory field or in a different technical context, and this can cause utter confusion due to the different approaches "state-of-the-art" versus "science and technology" and because different levels of reliability are required in the different cases.

There is a controversial discussion in Germany regarding site- and concept-independent safety indicators. The individual dose rate is the only indicator commonly agreed to be a safety indicator. In this respect, this contribution is to serve as a basis for discussion.

Not mixing the regulatory field and the regulatory levels is essential in practice.

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Appendix A4: ENRESA (Spain)

A4 ENRESA (Spain)








Proposal/Contract no.: FP6-036404

Project acronym: PAMINA

Project title: PERFORMANCE ASSESSMENT METHODOLOGIES IN APPLICATION TO GUIDE THE DEVELOPMENT OF THE SAFETY CASE

- Instrument: Integrated Project
- Thematic Priority: Management of Radioactive Waste and Radiation Protection and other activities in the field of Nuclear Technologies and Safety

WP1.1 Safety indicators and performance/function indicators. ENRESA contribution

Due date of deliverable:	03.30.07
Actual submission date:	03.25.07

Start date of project:10.01.2006Duration:36 months

Enresa

Revision: 2

Project co-funded by the European Commission within the Sixth Framework Programme (2002-2006)

Dissemination level

 PU
 Public

 PP
 Restricted to other programme participants (including the Commission Services)
 X

 RE
 Restricted to a group specified by the consortium (including the Commission Services)
 X

 CO
 Confidential, only for members of the consortium (including the Commission Services)

[PAMINA] (D-N°: 1.1.1) – Task reports for the first group of topics Dissemination level: RE Date of issue of this report: 15/03/2008



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1 Introduction and background

This document describes the experience of Enresa regarding the use of safety and performance/function indicators in the Performance Assessment (PA) of HLW repositories in granite and clay. Enresa has been a partner in a recent EC project on this topic, called SPIN [1], which main findings are described elsewhere. The scope of the present document is circumscribed to the use of safety and performance/function indicators in the Enresa's most recent Performance Assessments of spent fuel repositories: ENRESA 2000 [2] for a granitic formation and ENRESA 2003 for a clay formation [3].

The terms safety and performance indicators are not used explicitly in Enresa's PA exercises, but the different indicators used to justify that the repository is safe or show how the disposal system works have been classified into these two categories in section 4.

2 Regulatory requirements and provisions

The acceptance criteria for radioactive waste final disposal facilities established by the Spanish Regulatory Body (CSN) was set in 1987 in these terms: to ensure safety individual risk should be smaller than 10^{-6} yr⁻¹, that is the risk associated to an effective dose of 10^{-4} Sv/yr. This is the only regulatory requirement in Spain.

3 Key terms and concepts

No formal definitions of safety and performance/function indicators are included in the PA exercises done by Enresa. Since the end of those exercises, Enresa has taken part in EC project SPIN (Testing of Safety and Performance Indicators) [1]. In SPIN project IAEA definitions for the different classes of indicators given by IAEA [5] are adopted, and requirements for each class of indicators are established [1].

In [5] IAEA states:

"For the purposes of the present report, an *indicator* is taken to by any characteristic or consequence of a disposal system that has a bearing on the ability of the system to perform ifs safety functions. Indicators may be:

- directly measurable characteristics of the disposal system (e.g. radionuclide concentrations in groundwater at different locations and depths),
- characteristics derived from system understanding (e.g. container lifetimes and radionuclide fluxes across different boundaries in the facility system), or
- characteristics derived from calculations of the long term evolution of the disposal system (e.g. dose).

A distinction is also drawn between a *performance indicator* and a *safety indicator*. A performance indicator provides measures of performance to support the development of system understanding and to assess the quality, reliability or effectiveness of a disposal





system as a whole or of particular aspects or components of a disposal system. A safety indicator, which may be regarded as a special type of performance indicator, is used to assess calculated performance in terms of overall safety."

In SPIN project final report [1] the following requirements for safety and performance indicators are established for the indicators considered in the project:

"A safety indicator must

- provide a measure of the safety of the whole system,
- allow a comparison with a safety-relevant reference value,
- take into account the contribution of all radionuclides,
- be calculable using performance assessment models

A performance indicator must

- provide a measure of the performance of the whole system or a subsystem,
- allow a comparison with different options or with technical criteria,
- take into account the contribution of all radionuclides or single radionuclide,
- be calculable using performance assessment models"

4 Treatment in the Safety Case

4.1 Methodology

Enresa has no systematic methodology to identify safety and performance indicators useful for the Safety Case. The terms safety and performance indicators are not used explicitly in Enresa's PA exercises either, but the different indicators used to justify that the repository is safe or show how the disposal system works have been classified into these two categories in sections 4.1.1 and 4.1.2.

4.1.1 Safety indicators

In Enresa PA exercises the different scenarios are analyzed separately. No estimations of scenario probabilities are done and consequences are not added, weighed by their probabilities. Under these conditions, using dose or risk as indicators is completely equivalent, since the different is just a factor (the dose to risk conversion factor).

In Enresa PA exercises only **dose** (not risk) is calculated. There are two reasons for this approach: first, at the stage of Enresa's programme, the calculation of scenario's probability is out of scope, and second so far the consequences for all the scenarios assessed comply with the dose constrain, and consequently also with the risk constrain in the most pessimistic hypothesis (probability=1). Consequently, the dose to an average member of the critical group is the key indicator to quantify the safety of the disposal system.







Enresa main approach to performance assessment is probabilistic, using the Monte Carlo method. The complete release and transport calculation is performed 100 [2] or 500 [3] times using different values for the stochastic parameters, sampled from their probability distributions. Each individual calculation is called "realization". The indicator used to verify compliance is the peak value of the mean dose (averaged over all the realizations) during the assessment period. Enresa makes deterministic calculations too, and the peak dose value during the assessment period is compared with the reference value of 10^{-4} Sv/yr.

The use of doses in the PA exercises is twofold:

- comparison of calculated doses with the reference value (10⁻⁴ Sv/yr) is used to demonstrate the safety of the disposal system,
- comparison of the doses calculated in different variants (what if cases) and in sensitivity calculations allows to identify the effect on the repository safety of alternative evolutions of the disposal system, models or parameter values.

Time dependent doses are the main result of the Enresa PA exercises in order to demonstrate repository safety. The next figure shows the mean doses per radionuclide and total in the probabilistic calculation for the Reference Scenario of ENRESA 2000 [2]. Similar results are obtained for the rest of scenarios and the deterministic calculations. Calculated doses are always compared with the reference value of 10^{-4} Sv/yr to show the significant safety margin provided by the repository system.



The next figure presents the results obtained in the probabilistic calculations of the Reference Scenario of ENRESA 2000 [2]. In addition to the mean, 5% and 95% percentiles are represented as well as the peak and minimum values obtained in all the realizations at each instant.







While the mean doses and the percentiles depend on the shape of the pdf's assigned to the stochastic parameters, the maximum and minimum values are only a function of uncertainty ranges, and uncertainty ranges are easier to justify than pdf's.

The previous figure shows that there is no realization leading to peak doses close to the reference value. Even in the worst realization there exists a factor 30 of margin. Obviously, this is a very good result, but in other exercises some realizations can get close or even surpass the reference value. The identification of these problematic realizations will help to identify the parameters which uncertainty should be reduced.

Since probabilistic calculations explicitly represent parameter uncertainty through the use of pdf's, above figure clearly shows that there is no need to further reduce parameter uncertainty in order to fulfil safety criteria.

The other safety indicator used is the **activity flux leaving the far field**. These fluxes are compared with the natural activity in granitic soils [4] to put into perspective how small are the fluxes of radionuclides arriving to the Biosphere due to the repository:

- for a repository in granite the activity reaching the Biosphere in a year is always smaller than the natural activity of 1 m³ of granitic soil, and
- for a repository in clay the activity reaching the Biosphere per year is always smaller than the natural activity of 0.05 m³ of granitic soil.

Above results show that the activity reaching the Biosphere due to the repository is expected to be negligible compared with natural amounts of radionuclides.

Similar results would have been obtained in terms of radiotoxicity fluxes reaching the Biosphere, multiplying by the radionuclide-specific Dose Conversion Factor and summing





over all the radionuclides. This magnitude was identified as a useful safety indicator in SPIN project [1], especially for long time periods, but Enresa has not used this indicator in the PA exercises already done.

4.1.2 Performance indicators

The **canister failure distribution** is used to describe how long the canisters are expected to maintain their integrity, providing total isolation of the radionuclides in the waste. In Enresa's most recent PA exercise [3] a realistic model for a 10cm thick carbon steel canister failure due to generalized corrosion was developed. A Weibull distribution was adopted to describe the failure of the 3600 canisters in the repository, with the following parameters:

- ε: 20.000 minimum duration of the canister
- μ : uniform distribution between 40.000 and 60.000 years
- α : uniform distribution between 2 and 3.

The resulting set of failure distributions of the canisters used in the probabilistic calculations can be seen in the figure



The figure shows that canisters are expected to provide a minimum isolation period of 20.000 years and then will fail gradually at a quite constant rate in several tens of thousands years. After 160.000 years all canisters are expected to have failed.

The **fraction of UOX altered vs. time** is an indicator of the capability of the UO_2 matrix as a barrier. The figure shows the range of values considered in the probabilistic calculations of Enresa2003. In all the cases the UO_2 matrix is a useful barrier that ensure that a significant





amount of the radionuclides remain immobilized in the waste after 1 million years or even more (depending on the particular realization).



The **activity flux leaving the near field** is used in ENRESA 2000 to illustrate the capability of the engineered barrier system (EBS) of a repository in granite to limit the radionuclide releases from the near field to the host formation. This indicator has been used in different alternative cases to assess the robustness of the engineered barriers and to support decisions on engineering issues, as for example, to analyze the thickness of the buffer.

In addition, these fluxes have been compared with the natural activity in granitic soils [4] to put into perspective how small they are. It was found that the activity leaving the near field of the whole repository in a year is always smaller than the natural activity in 100m³ of granitic soil.

In a repository in granite transport through the geosphere is controlled by water advection along fractures and the **water travel time** from the repository to the Biosphere is a critical parameter to quantify the capability of the host formation as a barrier. In Enresa 2000 a particle tracking code was used to generate a distribution of water travel times from the repository to a surface stream, and the following results were obtained:









The previous results were used to generate a distribution for the water travel times for the probabilistic calculations, ranging from 7.000 to 230.000 years. In the deterministic calculations a different approach was followed and a water travel time of 8.400 year was obtained.

The codes used in Enresa 2000 for transport in the far field of a repository in granite in general allow to explicitly model the diffusion in the granite matrix as a dynamic process. Only in the deterministic transport calculations of the members of a decay chain, it is necessary to use **global retardation factors in the geosphere (R**_G). These factors are calculated assuming that diffusion in the matrix is faster than advection in the fracture and assuming that solute concentration in the fracture and the matrix porewater are the same. The validity of this assumption was confirmed by the calculations performed in ENRESA 2000. The values of the different retardation factors in the geosphere were:

Element	Global retardation factor	
Curium	1.963	
Americium	1.963	
Plutonium	3.926	
Neptunium	1.963	
Uranium	1.963	
Protactinium	786	
Thorium	1.963	
Radium	1.178	





Although not used in the probabilistic calculations of ENRESA 2000, it is possible to calculate the global retardation factor in the geosphere for each radionuclide and realization. For a given chemical element it is possible to generate the cumulative probability distribution for R_G , as shown in the next figure for Plutonium.



Plutonium retardation factor in the geosphere of a repository in granite [2].

The parameter truly relevant for radionuclide transport in the geosphere is the product of the water travel time and the retardation factor, the so called **radionuclide travel time through the geosphere**. For actinides and daughters, the transport time from the repository to the Biosphere is at least 5 million years for actinides and daughters. A systematic presentation of the values or distribution of values for this parameter for the different radionuclides is a good performance indicator of the capability of the granitic formation as a barrier.

The next figure shows the cumulative distribution of Plutonium travel times through the geosphere in the probabilistic calculations. It ranges from 5 million years to 15 billion years (the age of the Universe). This figure clearly shows that in any case the granitic formation will be a strong barrier against Plutonium transport, providing plenty of time for radioactive decay of Plutonium isotopes.







Plutonium travel time through the geosphere of a repository in granite [2].

For non-sorbed species the transport time through the geosphere will be similar to the water travel time, while for sorbed species can be several orders of magnitude greater.

For repositories in clay advection is negligible and transport is controlled by diffusion. A **radionuclide transport time through the clay formation** can be calculated as L^2/Da , where L is the thickness of clay between the repository level and top or bottom of the host formation and Da is the apparent diffusion coefficient (that includes the sorption on clay).

The radionuclide travel time through the geosphere can be very useful as a supporting argument or an alternative line of reasoning to demonstrate long term safety of the repository system. Although Enresa has not used this parameter in previous PA exercises, we consider that it can be very useful in future Safety Cases.

The **dilution water flow in the Geosphere-Biosphere interface** is a key parameter because doses are inversely proportional to it. In the repository in granite (ENRESA 2000 [2]) releases from the geosphere reach a superficial stream which flow is 1.000.000m³/yr in the Reference Scenario and 200.000m³/yr in the Climatic Scenario. In the repository in clay (ENRESA 2003 [3]) radionuclides leaving the top of the formation enter an aquifer where a water production well is drilled. Calculations are performed for different well locations and pumping rates and finally a conservative value of 180.000m³/yr was adopted for the dilution water flow in the well.

Since the key safety indicator is dose, this dilution water flow is an important performance indicator that quantifies the diluting capability of the geosphere-biosphere interface. The same releases from the geosphere would translate into very different doses depending on the value of this dilution water flow.





4.2 Related topics

Safety and performance indicators are important elements of the Assessment Strategy/Safety Approach. These indicators are key tools to demonstrate the safety of the disposal system.

4.3 Databases and tools

Not applicable.

4.4 Application and experience

Section 4.1 presents Enresa particular experience with the application of safety and performance indicators in Spanish performance assessment exercises.

In addition, Enresa has taken part in EC project SPIN project "Testing of Safety and Performance Indicators", which results are presented in [1].

4.5 On-going work and future evolution

Enresa is making no in-house developments on this topic.

Enresa is involved in PAMINA WP3.4 "Safety indicators and performance indicators". Enresa will evaluate the applicability to a repository in clay of the indicators identified in SPIN project [1] and will participate in the identification and testing of new indicators.

5 Lessons learned

Enresa experience in different Performance Assessment exercises and SPIN project has led to identify the usefulness of using different safety and performance indicators when preparing the Safety Case for a deep geological repository.

The usefulness of dose, radiotoxicity concentration in biosphere water and radiotoxicity flux from the biosphere as safety indicators for different time frames has been identified both in SPIN project and by IAEA [5].

A safety indicator can be useful only if safety relevant reference values to compare with are available. Derivation of these reference values for the radiotoxicity concentration/flux is not a straightforward task and has been included within WP3.4 of PAMINA project.

Several performance indicators have been identified to be very useful to present how the system works, the role of the different barriers and the capability of the disposal system to minimize the releases of radionuclides to the environment.

It is useful to have a full array of safety/performance indicators to present the Safety Case to different audiences: from technical audiences such as the regulators and the scientific community to the layman.





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Part 4: Safety indicators and performance indicators Appendix A5: GRS-B (Braunschweig, Germany)

A5 GRS-B (Braunschweig, Germany))





Part 4: Safety indicators and performance indicators Appendix A5: GRS-B (Braunschweig, Germany)





Project acronym: PAMINA

Safety indicators and performance indicators GRS contribution

Reference: 1.1.a Version: 2 RTDC: 1 Work package: 1.1 Author: D.-A. Becker Date of working paper: Spetember 2007





1 Background/Introduction

Although, of course, measures for quantifying the results of performance assessment calculations, mainly dose and risk, were always in use, it is a relatively new concept to improve the understanding of the system and to support the safety case by using complementary indicators. Such indicators are calculable quantities resulting from a PA calculation. While safety indicators aim at providing a quantitative criterion for the overall safety of a repository system, other indicators are calculated and presented to show the functioning of the system or specific components. They are sometimes called 'performance indicators' or 'function indicators', but they differ, with respect to goals and intention, from what SKB calls 'safety function indicators'. It is therefore suggested, in order to avoid confusion, to use the term 'function indicator' only in the latter sense as a short form. In this paper, the term 'performance indicator' is used.

In former German safety assessment studies, the only safety indicator used was the individual ingestion dose per year, compared to a regulatory limit. The SPIN project [1] was initiated by a new way of thinking, based on the awareness that the robustness of the safety case could be improved by using more than one safety indicator, as well as performance indicators. Several safety and performance indicators were tested in SPIN, using four national granite studies as examples.

In 2004, a detailed performance assessment for the Morsleben LAW repository (ERAM), which is installed in a former salt mine, was performed. The safety indicators defined in SPIN, as well as some performance indicators, were successfully applied to support the safety statement. It has become clear in this exercise that a rock salt repository requires performance indicators that differ from those used for granite, while safety indicators, though possibly depending on local reference values, are independent of the site and formation type.

The concepts and understanding of safety and performance indicators have further evolved since the end of SPIN. Presently, a new study for a HLW/SF repository in salt, called ISIBEL, is being made. Several safety and performance indicators were or will be calculated and compared with one another. This is done in parallel to PAMINA and the new concepts and ideas developed





2 Regulatory requirements and provisions

In Germany, there is no legal regulation concerning the use of indicators in long-term safety assessments so far. Not even the criteria to be held by a final repository for radioactive waste are clearly defined. The German Atomic Energy Act merely requires the safe disposal of radioactive waste. There is an old German guideline ("safety criteria for the final disposal of radioactive wastes in a mine"), originating from 1983, which is formally still valid [2]. Concerning long-term safety, it simply requires that "even after decommissioning radionuclides that could reach the biosphere in consequence of non-excludable transport processes from a sealed repository must not lead to individual doses exceeding the value given in the Radiation Protection Ordinance". This value is 0.3 mSv/yr and is valid for all nuclear facilities. A supplementary regulation from 1988 defines the time frame for which the individual dose rate should be evaluated as 10 000 years. The consideration of other safety indicators is not required, nor are probabilistic criteria defined.

There is, however, a consensus in Germany that the mentioned guideline is outdated and should be revised soon. A first draft for a new version, proposed by GRS, is currently under intense discussion. It requires the consideration of six indicators with fixed reference values as well as a probabilistic analysis. This paper is, however, a controversial matter and will be essentially changed before being accepted by the authorities. Therefore, it is not presented here. Nevertheless, it can be said that the future guideline is very likely to contain the following regulations:

- The calculated individual effective dose rate must not exceed the reference value of 0.1 mSv/yr ,
- Several additional safety indicators have to be calculated,
- The time frame for which safety has to be proven is 1 million years or more.





3 Key terms and concepts

In the following, the concept of safety and performance indicators as it is understood by GRS (Braunschweig) is described. Since the subject is under intense discussion in Germany at present, the following should neither be seen as 'the German standpoint', nor should it be regarded as final.

Safety indicators

Repositories for radioactive waste must be proven to be safe in the long-term. But what does that mean? A very general definition of repository safety can be given in the following way:

A repository is safe if it does not significantly change or disturb the natural evolution of the environment outside a narrowly limited area of influence.

Safety, in this sense, cannot be reduced to one single aspect like human health, but comprises a nearly unlimited variety of protection goals like water quality, air quality, protection of species, etc. There can, of course, be overlap between such protection goals, or one goal can completely include another one, but the often-heard statement that protection of man comprises all other protection goals cannot be proven.

A numerical calculation of the dissemination of radionuclides from a repository yields, in general, radionuclide fluxes. These results are per se not suitable for assessing the long-term safety of the repository, as they give no information about whether or not the repository can be considered 'safe' as defined above. It is necessary to convert the results into some safety-related measure, or safety measure. 'Safety-related' means that the safety measure should quantify a specific aspect of repository safety.

The word 'significantly' in the definition above does allow a certain influence of the repository on the environment if it is very small and negligible in comparison with natural influences. If safety with respect to some specific aspect is to be assessed using a safety measure, it is necessary to quantify a reference value as the limit of acceptability with respect to the safety aspect under consideration. Reference values should be proven to maintain the protection goal.

It is possible that different safety aspects (or protection goals) can be quantified with the same safety measure, using different reference values. Therefore, only the combination of a safety measure and a suitable, safety related reference value, both related to the same protection goal, is appropriate to give an indication of safety of the repository and is called a safety indicator. A safety indicator should always take account of the effects of all radionuclides in the repository.

There are two kinds of safety indicators. Those of the first kind are calculated for specific scenarios and the results can be compared in order to assess the consequences of different scenarios. Safety indicators of the second kind, however, are summed up over all relevant scenarios, each weighted by its probability. Such indicators are preferably calculated in terms of risk. They can be compared with risks from daily life or from natural sources like earthquakes, meteorite impacts, etc. The main problem with risk indicators is that scenario







probabilities can, in most cases, only be roughly estimated.

For performing a safety assessment it is always necessary to use at least one safety indicator. The technique mostly applied in the past is to calculate the time-evolution of the annual ingestion dose to an individual or a group and to compare it with a regulatory limit. The protection goal underlying this safety indicator is human health and the reference value was, though fixed by a regulatory rule, originally derived from the demand to be negligible compared to the natural background. In Germany, a value of 0.3 mSv/yr has been used so far. This safety indicator is widely used and refers to a rather universal protection goal, but it depends on more or less uncertain assumptions about the geosphere and biosphere. Moreover, it could suggest covering all aspects of safety, while actually it does not. Therefore, it is regarded increasingly necessary to consider additional safety indicators.

Performance indicators

Safety indicators are a good means to assess the overall safety of a repository system, but they do not yield detailed information about the functioning of the system. Such information, however, can be very helpful or even necessary in the process of concept development. It can be gained by using performance indicators.

A performance indicator is a calculable measure for the performance of parts of the system. These parts, which are called compartments, can be things like single barriers, groups of barriers, emplacement fields, the complete near field, or even the total system. Compartments can include others. The compartment structure to be used for a specific repository system should be a sensibly simplified image of the real system structure and depends on the type of the repository.

Performance indicators should illustrate how the repository works. Radionuclide fluxes between or concentrations in the compartments, e.g., show how and where the radionuclides are retained during the transport through the system. The time-evolution of a performance indicator should be calculated and compared for different locations, but a comparison with an absolute value is normally not necessary.

Whereas a safety indicator always requires considering of all relevant radionuclides in order to derive a safety statement, a performance indicator can be calculated for a single radionuclide, a group of radionuclides or the total radionuclide spectrum, depending on what is to be demonstrated. In this way it is possible, e.g., to compare the system performance for sorbing and non-sorbing species, or for the uranium and the thorium chain.





4 Treatment in the safety case

This section describes which indicators have been used by GRS in the past, and why. It is pointed out how the indicators have been calculated and interpreted and which reference values were used.

4.1 Methodology

Safety indicators

According to the regulations mentioned above, in all German studies made before 2000, only the individual effective dose rate was calculated and compared with the limit of 0.3 mSv/yr, normally for different concepts or different scenarios. Additional numerical investigations were, in some cases, performed in order to explain the results, but not to derive independent safety statements. In contrast to what the valid guidelines require, however, the calculations were always executed over a model time of at least 1 million years.

The SPIN project (2000 - 2002) has triggered a new view of the problem. The three safety indicators identified in SPIN to be useful have been applied in two recent studies for real sites:

- ERAM: The long-term safety assessment study for the LAW repository in rock salt near Morsleben,
- Asse: The long-term safety assessment study for the experimental LAW/MAW repository Asse in rock salt near Wolfenbüttel.

Moreover, five of the six indicators defined in the GRS proposal for a new guideline have recently been calculated in the ISIBEL study which considers a generic HAW repository in rock salt. This, however, is a running project, and the indicators themselves are still under discussion at GRS. Therefore, the results and findings of this exercise are not presented here.

In the following, the application of the SPIN safety indicators in the ERAM study is explained more detailed.

The primary safety indicator evaluated in the study is, according to the regulations mentioned above, still the effective dose rate to an adult human individual, in combination with the regulatory reference value of 0.3 mSv/yr. It has been calculated as a function of time over 1 million years, using standardised biosphere dose conversion factors. These dose conversion factors have been defined by GSF considering a number of typical exposition paths, which comprise:

- ingestion of drinking water,
- ingestion of plants,
- ingestion of meat,
- ingestion of fish,





- inhalation of contaminated particles,
- exposition by external radiation.

Since these paths refer to the present human population, the dose conversions factors are increasingly uncertain for longer time frames.

There was no freedom about the reference value, but since it is about 10 % of the natural radiation exposure, the repository is considered to be safe if the additional radiation exposure originating from it remains below this limit. For the ERAM reference scenario, the maximum dose rate is more than three orders of magnitude below the reference value.

Two more safety indicators have been considered. The radiotoxicity concentration in the aquifer has been calculated using the ingestion dose coefficients by ICRP. This measure is more robust than the dose rate because it is independent of the biosphere, though it is still based on the radiosensitivity of present-day humans. There is no "official" reference value for this measure, but it is rather easy to determine one. Waters that have been drunk by humans for hundreds of years without causing harm can be considered radiologically safe. There are a lot of data about concentrations of radionuclides in German drinking waters, and a typical radiotoxicity concentration becomes a proper safety indicator. It has been found that for the ERAM reference scenario the maximum radioxicity concentration in the aquifer is a little more than three orders of magnitude below this reference value.

The third safety indicator considered is based on the radiotoxicity flux from the repository. This is an even more robust measure than the aquifer concentration because it is independent of the geosphere, which could be influenced by ice ages etc. The problem with this measure is to find a clearly safety-related reference value. Two different possibilities were discussed. One is the natural radiotoxicity flux in a river near the repository, which is likely to finally collect all radionuclides released from there. The other possibility is the natural flux of raditoxicity in the groundwater near the repository. It was found that the second value was about three orders of magnitude lower than the first one. This is an example for the argument that one single safety measure can yield different and independent safety indicators if compared with different reference values. If the first value is used, the safety statement will be, "there is no significant influence on the river", which could be relevant for the river fauna and is clearly a safety aspect. If, however, the groundwater flux is used as reference value, the safety statement will be, "there is no significant influence on the groundwater", which is a different and probably more relevant safety aspect. By this reason, and because the value is lower, it was decided only to consider the natural radiotoxicity flux in groundwater as reference value, though it was harder to determine and is considered less robust. It was found to be 0.2 Sv/yr. For the ERAM reference scenario the maximum radiotoxicity flux from the repository is a bit more than three orders of magnitude below this reference value.

Performance indicators

In order to investigate the functioning of the repository system in detail, several performance indicators have been calculated for the ERAM reference scenario. The compartment structure used for this purpose is based on the model structure which is a strong simplification of the real mine structure. There are three sealed emplacement areas, two non-sealed emplacement areas and a number of voids that have not been used for emplacement



Part 4: Safety indicators and performance indicators



Appendix A5: GRS-B (Braunschweig, Germany)

purposes and are called 'residual mine'. Depending on the specific requirements of the investigations, the performance indicators have been calculated for slightly different compartment structures, sometimes merging the non-sealed emplacement fields together with the residual mine, sometimes not. It has become clear that, unlike a granite repository as considered in SPIN, a rock salt repository, especially if erected in an abandoned production mine, does not allow a unique and hierarchical compartment structure.

In order to show the dissemination of radionuclides within the mine, the concentration of radiotoxicity in the different compartments has been calculated as a function of time. To distinguish between the influences of the different emplacement fields, three different investigations were performed, one considering the total inventory, one considering only the inventory of the sealed emplacement areas, and one considering only the inventory of the non-sealed emplacement areas. It could be showed that the sealed emplacement areas, though the seals are assumed to lose their effectiveness after about 20000 years, still contain 90% of that part of their inventory that has not decayed after 1 million years. Even the non-sealed emplacement areas hold the main part of their inventory for about 100 000 years.

As an additional performance indicator the integrated radiotoxicity flux from the compartments was calculated as a function of time, each normalised to the initial inventory of the appropriate compartment. As already detected in SPIN, this is a very illustrative indicator since the time curves reach asymptotic values and the comparison of these shows how much of the inventory is finally retained in each compartment. The results show that a part of less than 0.1 % of the inventory of the sealed emplacement areas leaves these and even from the worst of the non-sealed emplacement areas only 10 % of its inventory can escape. A part of 10^{-5} of the total inventory leaves the repository system and reaches the biosphere.

4.2 Related topics

The issue of safety and performance indicators is related to a number of other PAMINA topics:

- Assessment strategy,
- Safety approach,
- Safety functions,
- Analysis of the evolution of the repository system,
- Biosphere,
- Uncertainty management and uncertainty analysis,
- Sensitivity analysis.

4.3 Databases and tools

Reference data are of high importance for safety indicators and should be taken from environmental measurements, biological investigations, etc. Some of the available data needed for determination of reference values are rather incomplete and uncertain. This problem might make it hard to apply or even test some promising indicators.

The tools needed for calculating safety and performance indicators are the same that are



being used for conventional performance assessment calculations, with a few slight modifications or add-ons.

4.4 Application and experience

In the ERAM study three safety indicators were applied as described in section 4.1. The time-curves are similar in shape because they have been derived from the same calculations, but nevertheless yield independent safety statements since the reference values have been determined completely independently and with totally different assumptions. It is interesting to see that even so all three safety indicators yield nearly exactly the same gap of about three orders of magnitude between the maximum and the reference value. This is clearly a coincidence but it shows a certain robustness of the safety assessment. For the ERAM reference case, the results are shown in Figure 1 in units relative to the respective reference value. In this representation the three curves are very close to each other.



Figure 1: Three safety indicators, calculated for the ERAM reference case. Each curve is related to its respective reference value as given in section 4.1.

A very illustrative performance indicator is the time-integrated radiotoxicity flow from different compartments of the repository, related to the initial inventory of the compartment. The curves finally reach stationary values which show how much of the initial inventory leaves the compartment. For the ERAM case, this indicator has been calculated for five compartments, three of them being separated emplacement areas plus the complete mine and the total system including the geosphere. The results are shown in Figure 2. It can bee seen that even the worst (and non-sealed) emplacement area, which is not designed to retain anything at all, nevertheless retains nearly 90 % of its inventory and the total system releases only about ten millionths of the initial radiotoxicity.





Part 4: Safety indicators and performance indicators

Appendix A5: GRS-B (Braunschweig, Germany)



Figure 2: Time-integrated radiotoxicity fluxes from different compartments of the ERAM repository (reference case). Each curve is related to the initial inventory of the respective compartment

4.5 On-going work and future evolution

Currently, different safety and performance indicators are being calculated within the new ISIBEL study for a HLW/SF repository in rock salt. Within PAMINA it is planned to test a wider variety of indicators including those of the risk type. It is also planned to perform probabilistic analyses in order to identify the specific sensitivities of different safety and performance indicators.



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5 Lessons learnt

The application of different safety indicators does only make sense if they aim at different safety aspects and provide different and independent safety statements. A safety statement depends not only on the safety measure but also on the reference value. For a safety indicator to be robust it is necessary that neither the safety measure nor the reference value depend on uncertain data or assumptions. Therefore, the radiotoxicity flux from the repository can only be considered robust and adequate for long time-frames if combined with a robust and safety-related reference value, which is not easy to find. Establishing of reference values is a very important and sometimes difficult task. A good reference value should be provably safe and valid for a long or at least well-known time frame. Reference values can be global or site-specific. A safety indicator can never be better than its reference value.

So far, only safety indicators that aim at human health have been considered in actual studies in Germany. Other protection goals like protection of non-human biota or even the inanimate environment should be taken into account. Some of the indicators considered in ISIBEL are of a more general character and could be adequate for such a concept.

Since the number of possible protection goals is nearly unlimited, a classification of such goals with a hierarchical structure could be a sensible task. It should be tried to find a limited number of protection goals that cover large ranges of others, ideally the total field of 'safety'. This could help defining a limited but comprehensive set of safety indicators.

Safety indicators of the risk type have not been considered so far in German studies. The reason might be that scenario probabilities are hard to determine. This kind of indicators can, however, be very illustrative and helpful in communicating with the public and should therefore be tested.

Performance indicators are always helpful to better understand the functioning of the system. They should be defined specifically for each study. As already seen in SPIN, it is hard to give a general recommendation for the use of performance indicators. It can, however, be said that integrated fluxes from different compartments, if interpreted correctly, in many cases provide very illustrative and useful information.





6 References

- [1] Becker, D.-A. et al.: Testing of Safety and Performance Indicators (SPIN). EUR 19965, Brussels 2003
- [2] Bundesministerium des Innern (BMI): Sicherheitskriterien für die Endlagerung radioaktiver Abfälle in einem Bergwerk. GMBI. 1983





A6 GRS-K (Cologne, Germany)









Project acronym: PAMINA

WP1.1 Contribution of GRS Cologne to the topic "Safety Indicators and Performance/Function Indicators"





Section 1: Background / Introduction

The GRS-document "Safety requirements for the disposal of high active wastes in a deep geological formation" /BAL 07/ proposes definitions and uses of safety and function indicators in the safety case. The document serves as a basis for a regulation on safety requirements which is expected to be issued in 2008. The following sections will outline the approach proposed by GRS.

Section 2: Definition of terms and used concepts

The current safety criteria of 1983 /BMI 83/ do not define or use the term "indicator". In the draft safety requirements /BAT 07/ safety and function indicators are defined in the following way:

- (1) An **indicator** is a quantity for measurement and evaluation used to evaluate a required property:
- (2) **Safety indicators** are used to show that the protection objectives are met. They allow an integrated assessment of repository safety.
- (3) **Function indicators** are used to assess the reliability performance of subsystems and components of the repository system with regard to the requirements.

The indicator concept proposed in /BAT 07/ will be explained in Section "Developments and trends".

Section 3: Regulatory context

Section 3.1: Regulations and guidance

In Germany, no legal regulation exists with regard to the use of safety, function or performance indicators for the evaluation of the safety of a final repository for radioactive wastes. A new regulation on safety requirements is expected to be issued in 2008 by the Federal Ministry of Environment on the basis of the draft safety requirements published by GRS /BAT 07/.

Section 3.2: Requirements and expectations

See explanations in Section "Developments and trends".

Section 3.3: Experience and lessons learned

Between 1976 and 1998, low and medium radioactive waste has been disposed of in the Morsleben repository (ERAM). Originally ERAM had been licensed under the law of the German Democratic Republic (GDR, Eastern Germany). In 1992, BfS applied for a plan approval procedure for a continuation of waste emplacement after 2000. In 1997, however, this application was restricted to the decommissioning of ERAM. The applications for the decommissioning of the Morsleben facility are orientated on the ideas developed during the revision of the safety criteria /BMI 83/. In the application, performance indicators have been used to assess the long-term safety of the ERAM repository and to quantify the efficiency of





single barriers or components of the repository. The radiotoxicity inventory in the repository system and its sub-systems have been used as performance indicators as well as the radiotoxicity flux out of sub-systems of the repository.

Section 3.4: Developments and trends

The safety requirements draft /BAT 07/ requires the use of safety indicators and function indicators. A set of 4 safety indicators is proposed to serve as indicators for the performance of containment. Two more safety indicators are proposed to evaluate biosphere safety. The derivation of function indicators and corresponding evaluation criteria is left to the implementer due to the strong dependency of such indicators on site, concept, and design.

The radiological protection objectives have to be met over timeframes which are actually too long for biosphere prediction. In order to overcome the need for largely hypothetical biosphere models in safety assessments, GRS proposes to focus the assessment for long time frames on the safety function "containment" rather than on radiological impact and argues that protection objectives are met if containment is ensured. "Containment" is understood in a technical sense which allows limited release of radionuclides similar to the corresponding German term "Isolation".

On this basis, GRS proposes the following set of 6 safety indicators (see also table 1):

- 1. Proportion of the cumulative released quantity of substance over the safety case period
- 2. Concentrations of released U, Th
- 3. Contribution to power density in ground water
- 4. Contribution to radiotoxicity in ground water
- 5. Radionuclide concentration in the usable water near the surface
- 6. Effective individual dose

Indicators 1 to 4 are measured at the boundary or peripheral area of an isolating rock zone, indicators 5 and 6 in surface near aquifers and in the biosphere, respectively.

Apparently, the assessment time frames of indicators 5 and 6 are limited due to the limited predictability of hydrosphere and biosphere evolution. This is acknowledged by referring these indicators to current or plausible future hydrogeology and to a standardised biosphere.

The limited long-term predictability of indicator 5 and 6 implies a shift in what both indicators indicate with increasing time frame: for short time frames for which predictions of hydrogeology or biosphere evolution are permissible the impact on the hydrosphere or biosphere is indicated; for longer time frames, however, both indicators become indicators for containment. As mentioned before, it is argued that ensurance of containment is sufficient to meet the protection objectives.

Assuming that containment is ensured if the existing system is perturbed as little as possible, GRS proposes yardsticks which focus on preventing disturbances or changes in surrounding geosystems and biosystems. The yardsticks are as far as possible orientated on conditions





found in nature. Since artificial radionuclides are not present in nature radiological considerations have to be utilised for such radionuclides.

With regard to the criterions connected to the indicators it has to be distinguished between likely and less likely scenarios. For likely scenarios it is required that natural conditions are not significantly disturbed. For less likely scenarios natural conditions may be disturbed by release of radionuclides in the order of magnitude found in nature.

Determination of radiological consequences in terms of the proposed indicators has to consider data uncertainties. Using stochastical methods the evaluations of the results has to refer to the 95-percentile of the indicator determined with a confidence interval of 95%. Uncertainty analysis shall not be applied to the standardised models used in the calculation of indicators.

The draft safety requirements /BAT 07/ do not prescribe any form of presentation for the calculated indicators. However, the implementer should show that the repository system and its behaviour are well understood.

Table 1: Proposed safety indicators to be applied in safety assessment andoptimisation according to /BAT 07/. The indicators which will have to be considered in
the forthcoming regulation are still under discussion.

indicator	location	Criterion (likely scenarios)	Motivation / comments
Proportion of the cumulative released quantity of substance over the safety case period	boundary of isolating rock zone (cummulated flux)	Percentage of the quantity of substance disposed of	Assessment of containment
Concentrations of released U, Th	peripherial area of isolating rock zone	1 μg/l U, 0.1 μg/l Th	Modification of natural concentrations
Contribution to power density in ground water	peripherial area of isolating rock zone	1 MeV/I Pore water	Modification of natural radioactivity
contribution to radiotoxicity in groundwater	boundary of isolating rock zone (flux)	0,1 mSv/a	Modification of natural radioactivity
Radionuclide concentration in the usable water near the surface prediction timeframe limited!	Aquifers near to the surface	specific per nuclide (²³⁸ U, ²³⁴ U, ²²⁶ Ra, ²¹⁰ Pb, ²³⁵ U, ²²⁸ Th, ²³⁰ Th, ²³² Th)	Modification of natural concentrations Present-day or plausible future hydrology
effective individual dose per year	biosphere	0,1 mSv/a	Modification of natural radioactivity
prediction timeframe limited!			Present-day or plausible future hydrology, standardised "biosphere"





Section 4: Analysis and synthesis

GRS believes that the presented approach, which focuses on the aspect of containment by the isolating rock zone rather than on radiological impact, accounts for the often required, but less often implemented request to use safety and performance indicators additional to dose or risk. The discussion about the possibility of "compromising the ability of future generations to meet their needs and aspirations" (Joint Convention) loses importance. However, it has to be stressed that the proposed requirements and indicators as well as the role of the biosphere in safety assessments is still under discussion in Germany.

Section 5: References

- /BAL 07/ Baltes, B. et al.; Sicherheitsanforderungen an die Endlagerung hochradioaktiver Abfälle in tiefen geologischen Formationen, Entwurf der GRS, GRS- A- 3358, 2007 (only available in German language)
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Appendix A7: IRSN (France)

A7 IRSN (France)





Appendix A7: IRSN (France)

Safety indicators

1 Background/ Introduction

The assessment of the dose due to the disposal system must be verified through the safety case, in order to reach the safety objective of protecting man and the environment. In the performance assessment, the dose depends on the radionuclide decay due to their migration through the disposal facility and the geosphere. The calculated value is therefore related to the performance of the components of the disposal system, and then to the confidence on the choice of data used in calculations. The dose is generally used as a safety indicator in the safety assessment of the disposal system. Nevertheless, the dose must be relied to the potential risk of a population to be impacted by radioactive substances due to the disposal facility. In a practical point of view, the dose is calculated at the surface or at specific place located in possible water supplying zone (fractures, aquifers...) and is a product of a combination between calculated fluxes (or concentrations) of radionuclide and biosphere coefficients.

In its radionuclide transport calculations, IRSN quantifies the activity fluxes related to dose and also activity fluxes related to the performance of the components, which mainly depends on the ability to confine the radioactivity, and particularly to retard and to attenuate the radionuclide migration. Those indicators are performance indicators and they participate to the performance assessment of the disposal system.

This contribution gets onto the both notions of safety indicator and performance indicator as explained in the BSR III.2.f and as dealt with by IRSN.

2 Regulatory context

ASN (Nuclear Safety Authority) and its technical support organisation IRSN (Institute for Radioprotection and Nuclear Safety) develop the regulatory framework for the safety of the deep geological disposal. This framework follows the principles and recommendations enact by the international organisations being technically competent (IAEA, ICRP, OECD).

In June 1991, the Basic Safety Rule 3.2.f (BSR3.2.f) was edited by ASN as guidance for defining the situations providing demonstration of safety through evolution scenarios. A new version of this rule is currently under progress in order to introduce the notions and the safety approaches developed in the 2005 Clay Dossier edited by ANDRA.

a) Regulations and guidance

Adequacy must be checked in terms of protecting of man and the environment. To this end, assessments of the radiological impact will be made to verify that this objective is effectively





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met under all the situations considered (i.e. expected performance or failure of the components). The concept chosen for the repository must take it possible "to limit the radiological impact to levels which are as low as reasonably achievable in view of the technical, financial and social factors" according to the ALARA principle of the International Commission for Radiological Protection.

The safety assessments will include determination of individual exposure expressed as equivalent doses. A distinction must be drawn between exposure resulting from the repository in the normal reference scenario and that which would result from random events which disturb the repository. For the reference scenario, individual dose equivalents must be limited to 0.25 mSv/year for extended exposure associated with events which are certain or highly probable. This fraction corresponds to a fraction of the annual limit of exposure of public in a normal situation.

Regarding the BSR III.2.f, to maintain consistency between exposure in the reference situation and potential exposure associated with hypothetical situations, consideration may be given to using the risk concept (the product of the probability of the occurrence of the event and the effect of the associated exposure) to allow for the probability of occurrence of each situation giving rise to exposure. Under these conditions, the acceptability of individual exposure associated with the occurrence of random events is assessed with allowance made for the nature of the situations taken into consideration, the duration and the nature of the transfers of radioactive substances from the canisters to the biosphere, the properties of the pathways by which people can be affected and the sizes of groups exposed. Furthermore, the possibility of taking actions to mitigate the consequences (in the case where situations of the type considered should occur) must not be made allowance for in the design of the repository. Therefore, individual exposure expressed as a dose equivalent, associated with hypothetical situations for which allowance must be made in the design of the repository must be maintained well below liable to give rise to deterministic effects.

However, the safety approach preferred by IRSN is based on a stepwise collection of arguments conceived upon defence-in-depth principle but not risk-based principle. As a matter of fact, the definition of a criterion based on an individual risk limits precautions, as it may imply a debatable equivalence between reducing the probability and reducing the exposure. Furthermore, it can be expected to be difficult, if not impossible, to estimate the probabilities of the events which can result in exposure.

Regarding the performance indicators, activity fluxes and concentrations are not explicitly defined in the regulatory guidance. Performance indicators are defined by the implementers to quantify the containment capabilities of disposal components in various situations. The implementer must justify the choice of such indicators.

b) Experience and lessons learned

IRSN considers that besides the dose calculated for some situations considered as envelope of the possible situations to occur, activity flux is an important indicator. Calculated for different specific locations in the repository it allows assessing the design ability for confining the radionuclide activity as close as possible to the canisters. This indicator allows assessing influence of various hypotheses related to the possible behaviours of the various components in relation with the physico-chemical perturbations due to the repository or







design defect without any assumptions on the biosphere.

Concerning the transport calculations, the performance indicators inform on the amount of activity transferred out of the waste packages to the geosphere and on the transfer time, and consequently on the confinement performance of the components. The activity fluxes are performed a posteriori of the transport calculations and aim at:

- assessing the dominated regime transport, since the shape of the flux curves reflects the diffusive or advective transfer. This assumption can be verified by calculating Peclet numbers, which is a numerical indicator devoted to the determination of the dominated regime transport.
- quantifying the amount of radionuclide transferred, for a period of time, by integrating the fluxes.
- estimating the transfer time through a component and then assessing its effect on the amount of mass decayed for each radionuclide.

Activity fluxes are used for assessing the main pathway in the repository for various situations. As an example, activity fluxes can be calculated from the disposal tunnels to the access drift (through the disposal tunnel plug) and from the disposal tunnels to the non-disturbed host rock. The comparison of both fluxes for different evolution scenarios allows assessing the relative importance of radionuclide pathways and the influence of design options.

3 Analysis and synthesis

The safety indicators are used to evaluate the level of safety of the disposal system. Performance indicators allow quantifying the containment efficiency of the disposal components and then better understanding the functioning of the system as a whole.

Both quantitative indicators can be efficiently used to provide quantitative arguments concerning foreseeable behaviour of a disposal system during the technical exchanges between regulators and implementers within the framework of the review process.





Appendix A8: NDA (United Kingdom)

A8 NDA (United Kingdom)








Project acronym: PAMINA

WP 1.1 – Safety indicators and performance/function indicators





Appendix A8: NDA (United Kingdom)

Section 1: Background/ Introduction

In recent performance assessments it is fair to say that radiological risk was the main performance indicator reported. However other indicators are given - for example, the radionuclide flux out of the engineered barriers and out of the geosphere; and radioactivity levels in the main system components at various times. NDA has also investigated the use of alternative safety indicators (for example radionuclide concentrations in the environment) and the roles for more qualitative safety measures (for example comparisons with natural and anthropogenic analogues) in a safety case.

Section 2: Regulatory requirements and provisions

There is a general regulatory requirement to look at other indicators than dose/risk, but no specific alternative indicators or criteria are identified. The regulator does however expect to see qualitative safety arguments. It is for the developer to justify their safety case approach.

Section 3: Key terms and concepts.

Qualitative arguments can include:

- Comparisons with natural analogues, i.e. occurrences of materials or processes which resemble those expected in a proposed geological waste repository, for example the Magarin site in Jordan which provides a natural analogue for a cementitious repository.
- Showing consistency with independent site-specific evidence, such as observations in nature or palaeohydrogeological information.
- Evidence for the intrinsic robustness of the repository system, for example demonstrating that relevant features and processes are well understood, often supported by evidence from underground research laboratories.
- Describing the passive safety features of the repository and demonstrating that the design uses best practice scientific and engineering principles.
- The safety case may also include more general arguments related to radioactive waste management, and information to put the results of performance and safety assessments into perspective. For example, for the NDA (RWMD) ILW concept a repository at a depth below ground of about 650m is assumed. Such a depth offers a number of benefits to the long-term management of radioactive waste that would be of relevance to the safety case.

Section 4: Treatment in the Safety Case

Section 4.1: Methodology

In NDA (RWMD) plans for a Disposal System Safety Case (DSSC) these alternative safety arguments will take on a key role. The aim of the DSSC is to build confidence in the safety of the system using a wide range of arguments that will be meaningful to a wider spectrum of stakeholders than just performance assessment specialists. The numerical assessments







Appendix A8: NDA (United Kingdom)

and the performance indicators derived from them will still be important, but the DSSC audience will not be expected to rely solely on mathematical modelling for an assurance of long-term safety.

Numerical performance indicators (including radionuclide fluxes) will be derived from numerical modelling. The qualitative safety arguments will be derived directly from research (including, where possible, research that has been conducted by respected, independent parties outside the nuclear industry) and from analogue studies (such as those discussed in the EC NANet project). The aim will be to demonstrate our understanding of the performance and evolution of the repository system and its various components by reference to facts and situations with which the audience may be familiar.

Generally, a performance assessment will include a range of quantitative performance indicators, together with alternative lines of reasoning and qualitative considerations, such as the intrinsic quality of the repository design, to build understanding in the overall repository performance and hence determine whether it satisfies the relevant safety requirements.

There is also a role in many performance assessments for semi-quantitative arguments, for example applying physical and chemical understanding of the system to build more simple models to give an insight of repository system behaviour.

Qualitative arguments may be particularly important in performance assessments conducted at the earlier stages of a repository development programme. At these stages the focus is on building understanding of the processes that could affect the performance of a repository and on explaining how the repository concept will be able to provide safety over very long time periods. There may also be insufficient data at this stage to justify complex calculations, therefore other methods are required to build confidence in the viability of the proposals. Assessments at this stage are also more likely to be communicated, at least in summary form, to wider, non-technical audiences for whom qualitative arguments may be more meaningful than detailed, complex calculations.

It is possible to turn to the geological environment to find naturally occurring examples of how materials and processes present in a deep repository may operate over geological time periods measured in many millions of years, far in excess of the repository time scales. Analogues based on the study of archaeological and industrial artefacts (anthropogenic analogues) can also provide information relevant to processes occurring over time periods measured in hundreds to thousands of years.

Evidence from natural and anthropogenic analogues gives an indication of the extent and importance of processes over the timescales that are of relevance to the long-term safety of radioactive waste management and that would be impossible to investigate over laboratory experimental timescales.

Section 4.2: Related topics

Safety functions

Section 4.3: Databases and tools

None





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Section 4.4: Application and experience

This is currently on-going work. We have developed the methodology but have not yet produced a safety case based on the proposed methodology. The first generic safety case based on this approach is due for publication in 2009.

Section 4.5 On going work and future evolution

This work will continue to be developed as it is implemented.

Section 5: Lessons learned

Too early to say as the methodology has not yet been implemented. NDA will actively seek and respond to feedback.

Section 6: References

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A9 NRG (Netherlands)





note

to :	Topic coordinator 'safety indicators and perform	ance/function indicators'
from :	J.B. Grupa	Petten/015.017
copy :	J. Hart, A.D. Poley	
date :	03 December 2007	
reference :	21951/07.86200 RE/JG/ES	
subject :	NRG Final contribution to topic 3 'Safety indicat performance/function indicators'	ors and

Section 1: Background/ Introduction

In the late 1980's the VEOS study (Safety evaluation of disposal concepts in rock salt) has been performed in the Netherlands [1, 2, 3, 4]. The aims of this study were the evaluation of the post-closure safety of some possible disposal concept and the determination of relevant characteristics. VEOS used a scenario approach followed by a deterministic consequence analysis and several deterministic sensitivity studies. The analyses resulted in a number of release scenarios with estimated exposure. For some scenarios with a relatively high exposure the probability of occurrence was also calculated. The resulting risk defined as the product of this probability and the health effect of the exposure was below the risk levels set in neighbouring countries and the IRCP.

In the early 1990's a generic probabilistic safety analysis (PROSA, [5]) of the Dutch generic reference disposal concept has been performed. In this study a systematic approach to scenario selection has been used that ultimately leads to a set of selected scenarios that covers all aspects relevant for the long term safety. The method used a FEP catalogue to show comprehensiveness of the obtained set of scenarios.

Section 2: Regulatory requirements and provisions

In The Netherlands a safety report has to show that risks and individual doses are below the regulatory limits. However, a license application will also include an EIS (Environmental Impact Statement), which follows more or less the ICRP principles for Radiation Protection, i.e.: (1) justification, (2) optimisation, and (3) compliance with limits. The EIS uses the safety report to show compliance. For optimisation the EIS needs more indicators to be able to compare with alternative options.

Presently the only indicators are dose and risk, for which there are reference values and constraints.

Dose:

Reference value: The natural background radiation in The Netherlands is about 2.5 mSv/year.

Constraint:

The dose contraints given below is a dose level that should not be exceeded for more than 5% of the group under consideration.





probability of the event causing the exposure	adult	child
> 0.1 per year	0.04 Sv	0.015 Sv
> 1E-2 per year	0.4 Sv	0.15 Sv
> 1E-4 per year	4 mSv	1.5 mSv
> 1E-6 per year	40 mSv	15 mSv

Events with a probability less than 1E-6 per year must comply with the risk constraint.

Risk:

Reference value:

The average risk of dying is at its lowest level for persons of the age of about 15. This level is 1E-4 per year.

Constraint:

The total risk to a person from a (nuclear) instalation must be less than 1E-6 per year. To events that can result in 10 or more deaths more stringent constraints may be applied.

Section 3: Key terms and concepts.

Although the main indicators considered in the Netherlands are dose rate and risk, there is at present a focus on self sealing behaviour and closure times of plugs and sealing materials.

The Glossary in the PROSA study [5] gives the following definitions for:

Risk The radiation induced yearly probability for an individual to die by cancer in the period between t and t + 1 year after disposal

In addition, from [6], the "closure times" of plugs and seals in a salt-based repository are defined as the times for which compacted salt reaches the percolation limit (1% porosity), for which the possible water flow paths in the compacted salt are cut off through the ongoing compaction process.

Section 4: Treatment in the Safety Case

Section 4.1: Methodology

In the PROSA [5] study, the dose calculations were done taking into account best estimate values for most model parameters. For parameters of importance in the dose calculations and with large uncertainties, probability density functions were selected. Probability density functions were used for a selected number of model parameters for the salt compartment, the groundwater "compartment", and the biosphere "compartment".





In the PROSA [5] study, there has no clear distinction been made between safety indicators and performance / function indicators.

Section 4.2: Related topics

The whole set-up of the PROSA study was to perform a systematic approach of the safety of a salt-based repository. This included:

- Scenario selection
- Determination of the probability of the scenarios (including human intrusion)
- determination of the calculation model
- determination of the parameters and their probabilities
- dose calculation
- sensitivity and uncertainty analysis investigation of the effects of uncertainties of mode parameters on the calculated dose rate

Section 4.3: Databases and tools

A large amount of data were applied in the Dutch studies. Sources of the model data were previous studies, engineering judgement, or, in some cases, measured values.

Section 4.4: Application and experience

No practical applications and experiences have yet been implemented in the Netherlands.

However, the basis of the methodology as applied in the PROSA study seems the way to go in the Netherlands, i.e. a systematic approach as bulleted under 4.2, which includes sensitivity and uncertainty analysis.

Section 4.5 On going work and future evolution

Ongoing work in the Netherlands on this topic is presently not included in a national program. Activities and research mainly takes place in EU-funded FP programmes.

We expect that the PROSA procedure for identifying scenarios will be extended by the application of 'safety functions', and therefore also of safety/performance indicators in future safety studies.

Section 5: Lessons learned

Dose and risk do not show the strength of a disposal concept. They focus on the amount of material that escapes from the repository, rather than on the robustness of the isolation system provided.





Section 6: References

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- [6] Grupa, J.B., and M. Houkema, "Terughaalbare opberging van radioactief afval in diepe zout- en kleiformaties", NRG report 21082/00.33017/P, Petten, 16 June 2000 (in Dutch).



A10 NRI, RAWRA (Czech Republic)









Proposal/Contract no.: FP6-036404Project acronym:PAMINAProject title:PERFORMANCE ASSESSMENT METHODOLOGIES IN
APPLICATION TO GUIDE THE DEVELOPMENT OF THE
SAFETY CASEInstrument:Integrated ProjectThematic Priority:Management of Radioactive Waste and Radiation
Protection and other activities in the field of
Nuclear Technologies and Safety

WP1.1 Safety indicators and performance/function indicators. NRI contribution

Jiri Landa, Antonín Vokál

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PU	Public	
PP	Restricted to other programme participants (including the Commission Services)	Х
RE	Restricted to a group specified by the consortium (including the Commission	
	Services)	
CO	Confidential, only for members of the consortium (including the Commission	
	Services)	

[PAMINA] (D-N°: 1.1.1) – Task reports for the first group of topics Dissemination level: RE Date of issue of this report: 15/03/2008





Introduction

This document describes the work of NRI regarding the use of safety and performance/function indicators in the Performance Assessment (PA) of HLW repositories in granite. The work of NRI focused primarily on safety indicators based on analysis of available measurements of natural activity in the Czech Republic.

Regulatory issues

By regulations (Regulation No. 307/2002 Coll. on radiation protection), the only quantitative requirement from radiological point of view is potential individual dose raised by repository existence. It has not to exceed 0.25 mSv/yr for normal evolution scenarios and/or 1 mSv/yr for emergency scenarios. There exists no other quantitative limitation postulated by nuclear legislation.

Deep geological repository, however, will be assessed also according to Act 100/2001 Coll. On Environmental Impact Statement (EIA). This assessment shall comprise the impact of repository on public health and the impacts on the environment, including impacts on fauna and flora, ecological systems, the soil, the geological environment, water, air, climate and landscape, natural resources, tangible property and cultural monuments. The impact of DGR will be assessed in relation to the state of the environment in the affected territory at the time of submitting notification of the plan. The state of the environment can be described using so called **Environmental Indicators,** which express the state of the environment in the given territory (e.g. quality of water). Until this time, however, no such indicators were discussed in relation to radioactivity, but it cannot be excluded that this will required in future EIA requirements. The exact EIA requirements depend on the Ministry of Environment after agreement with Ministry of Health and possibly some regional authorities.

Safety indicators

In the Czech Republic there are three organisations, which measure regularly radiometric data of ground and surface water: Czech Hydrometerological Institute, Czech Geological Survey - Geofond and National Radiation Protection Institute, and which can form basis for safety indicators in the Czech Republic. Another data can be obtained from producers of mineral waters.

Available water radiometric data are mainly Gross Alpha, Gross Beta, concentration of U, activity of 226Ra, and concentration of 40K.

Numbers and types of measurement points of water quality are given in Table 1. Locations of surface water measurements points in Czech republic are on Figure 1.





Table 1: Number	of measurement	points in	Czech Rep	ublic [1]
	•••••••••••••••••	P • · · · · • · · ·		

Type of measuring point	Number of points
Measurement points on rivers	534
Points of water quality measurement	283
thereof radiotoxicity	82
Spring wells	507
thereof with water quality measurement	142
Boreholes	1925
thereof with water quality measurement	334

Figure 1: The state system of radiotoxicity measurement points [2]



In the following Table 2, natural radionuclides contributing to gross alpha and beta activity are listed. In the Table 3 Ingestion Dose Conversion Factors of total alpha and beta activity and Uranium are given.





Decay chain	Gross alpha activity	Gross beta activity		
Potassium		⁴⁰ K (89.33 %)		
Rubidium		⁸⁷ Rb		
4N	²³² Th, ²²⁸ Th, ²²⁴ Ra	²²⁸ Ra		
4N+2	²³⁸ U, ²³⁴ U, ²³⁰ Th, ²²⁶ Ra, ²¹⁰ Po	²¹⁰ Pb		
4N+3	²³⁵ U, ²³¹ Pa, ²²⁷ Ac (1.4 %), ²²³ Ra	²²⁷ Ac(98.6 %)		

Table 2: Natural radionuclides contributing to alpha and beta activity

Table 3: Ingestion Dose Conversion Factors for alpha and beta activity and uranium (Sv/Bq)

	mean	geom.	min.	max.
alfa	3.42E-07	1.71E-07	4.50E-08	1.20E-06
beta	4.94E-07	8.61E-08	1.50E-09	1.10E-06
U	4.70E-08	4.70E-08	4.50E-08	4.90E-08

In the Table 4 the gross alpha and beta activities in the Czech Republic in different types of water were summarised.

Table 4: Alpha and beta	activity of different s	sources of water in the	Czech Republic
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	Total alpha activity (Bq/I)			Total beta activity (Bq/I)		
	Median	Scale	Number of samples	Median	Scale	Number of samples
Near surface Groundwater in granite	0.1	0.01-14		0.195	0.05-4.4	
Deeper boreholes data	0.16	0.019 - 615	786	0.32	0.08 - 11.1	192
Near - surface data	0.14	0.03 - 8.3	632	0.39	0.05 - 3	120
Springs	0.09	0.02 - 3.2	490	0.13	0.05 - 1.27	81
Surface water	0.237		2044	0.231		2521

The concentration of uranium in the largest river in the Czech Republic (Elbe) at different measuring points is shown in Figure 2.







Figure 2: sample of measurement Uranium concentration in river Labe (Elbe) in some measurement points [3]

The content of uranium in common mineral water used in the Czech Republic is in the range from 0.002 to 0.009 mg/l and content of ²²⁶Ra about 0.26 Bq/l. Using the data given above, it can be easily derived that people drinking mineral water or water containing natural radionuclides from the wells (700 l/year) will obtain the doses in the range from 1 x 10^{-8} Sv to 1 x 10^{-4} Sv.

Comparing these values with the values obtained from conservative safety analyses (see Figure 3), we can see that the doses, which could obtain individuals from critical group of population from a repository, are in the range, which people can obtain by drinking water containing natural radionuclides.









Performance indicators

Performance indicators in the sense of SPIN project have not been considered in the Czech Republic in formal assessments of proposed repository designs, but the calculations of values of activity fluxes coming from different repository compartments (canister, near field) are common part of analyses.

References

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Appendix A11: POSIVA (Finland)

A11 POSIVA (Finland)





PAMINA. WP1.1 Comprehensive Review of Methodologies and Approaches in the Safety Case

Topic: <u>Safety indicators and performance/function indicators</u>

Regulatory guidelines

The Finnish regulation (STUK 2001) states that the annual effective dose to the most exposed members of the public shall remain below 0.1 mSv, and the average annual effective doses to other members of the public shall remain insignificantly low, in an assessment period that is adequately predictable with respect to assessments of human exposure. The regulation also gives some guidelines concerning assumptions that can be made about the potential exposure pathways:

- use of contaminated water as household water shall be considered
- use of contaminated water for irrigation of plants and for watering animals shall be considered
- use of contaminated watercourses and relictions shall be considered

In addition, the climate type as well as the human habits, nutritional needs and metabolism can be assumed to be similar to the current situation.

Over long time periods (beyond the adequately predictable time frame) the regulatory guidelines (STUK 2001) specify constraints for the average quantities of radioactive substances released from the disposed waste and migrated to the environment, stating that "the average quantities of radioactive substances over long time periods, release from the disposed waste and migrated to the environment, shall remain below the nuclide specific constraints defined by the Radiation and Nuclear Safety Authority. These constraints shall be defined so that 1) at their maximum, the radiation impacts arising from disposal can be comparable to those arising from natural radioactive substances, and 2) on a large scale, the radiation impacts remain insignificantly low. The nuclide specific constraints for the activity releases to the environment are as follows:

- 0.03 GBq/a for the long-lived, alpha emitting radikum, thorium, protactinium, plutonium, americium, and curium isotopes
- 0.1 GBq/a for the nuclides Se-79, I-129 and Np-237
- 0.3 GBq/a for the nuclides C-14, C-36 and C-135 and for the long-lived uranium isotopes
- 1 GBq/a for Nb-94 and Sn-126
- 3 GBq/a for the nuclideTc-99
- 10 GBq/a for the nuclide Zr-93
- 30 GBq/a for the nuclide Ni-59
- 100 GBq/a for the nuclides Pd-107 and Sm-151





Appendix A11: POSIVA (Finland)

These constraints apply to activity releases which arise from the expected evolution scenarios and which may enter the environment not until several thousands of years. These activity releases can be averaged over 1000 years at the most. The sum of the ratios between the nuclide specific activity releases and th respective constraints shall be less than one."

The regulators also requires that the repository shall not affect detrimentally to species of fauna and flora; which shall be demonstrated by assessing the typical radiation exposures of terrestrial and aquatic populations in the disposal site, assuming the present kind of living populations.

Furthermore, the regulation requires that the long-term safety of disposal is based on redundant barriers that effectively hinder the release of disposed radioactive substances into the host rock for several thousands of years. Performance targets for the long-term performance of each barrier shall be determined based on best available experimental knowledge and expert judgement; based on an assumption that, due to some unpredicted phenomenon, the performance of a single barrier as a whole may be significantly lower than the respective target value.

Safety indicators in the Posiva Safety Case

The complete set of safety indicators, and the approach to derive them, has not yet been defined in the Posiva Safety Case. Currently, the outcome of the SPIN project is evaluated, and will most likely result in the adoption of some of them. A safety indicator derived for non-human biota using the ERICA methodology is also under development. The safety indicators that are directly related to the regulatory guideline are briefly presented below.

Primary safety indicator

The term *primary safety indicator* (PSI) is in the Posiva Safety Case restricted to quantities related to the radiological impact of the whole repository system. To comply with the regulatory guidelines, the *annual effective dose*¹⁴ is chosen as the only quantity for the PSI in the Posiva Safety Case. The approach to implement it is by defining two stylised well scenarios: 1) a well only used for human drinking water (WELL), and 2) a well used for human drinking water, watering cattle and irrigate crops (AgriWELL).

The WELL scenario was introduced in Posiva's assessment of deep repositories in 1994 and has been applied, with minor modifications such as updated dose coefficients, in assessments since (WELL-94 \rightarrow WELL-96 \rightarrow WELL-97 \rightarrow WELL-2007). The WELL scenario is very simple and robust, it is based on an assumption that the annual releases from the repository into the biosphere are diluted in 100,000 m³ of water and an individual annually consumes 500 litres of contaminated water. An effective dilution volume of 100,000 m³/y is obtained, for example, if 1% of the total releases from the repository into the biosphere ends up in a well and the dilution in the well is 1,000 m³/y. Drinking of water is the only exposure pathway considered, thus the annual effective dose in the WELL scenario is identical to the committed effective dose from intakes of radionuclides in that year. Applying the WELL



¹⁴ The effective dose due to external exposure in a year plus the committed effective dose from intakes of radionuclides in that year.



Appendix A11: POSIVA (Finland)

scenario on the annual release rates of radionuclides from the geosphere into the biosphere results in an annual effective dose received by a member of the most exposed group; thus the WELL scenario can be used to compare to the first dose criteria of the regulatory guidelines.

The AgriWELL scenario is under development, currently a first version is under finalisation. The AgriWELL is based on the same well as in the WELL scenario, with the additional assumption that the well capacity is sufficiently high that the amount of water is enough for human consumption, cattle consumption, and for irrigation of crops. Added exposure pathways are consumption of irrigated crops (mainly vegetables, fruits and berries), consumptions of animal products (milk, meat and eggs). As for the WELL scenario, external exposure is not included as a pathway, which may be added in the future, at least for a few radionuclides. The irrigation water is contaminating the crops, taking both direct uptake of surface deposited activity and secondary uptake via the roots into account. The irrigation water is contaminating animal products due to the animals drinking water consumption and consumption of contaminated fodder. Furthermore, the AgriWELL uses data for a fictive farm with properties corresponding to the arithmetic average composition and production of the present farms (year 2004) in the region around the Olkiluoto site. Applying the WELL scenario on the annual release rates of radionuclides from the geosphere into the biosphere results in an annual effective dose received by a member of the most exposed group, which in this case is a member of the family running the farm who satisfy the nutrient need by eating and drinking products from the own farm. The excess products from the farm are sold to the local community, giving a base for deriving an indicative annual effective dose to other members of the public; thus the AgriWELL scenario can be used to compare to the first and the second dose criteria of the regulatory guidelines.

Complementary safety indicators

In addition to the PSI described above, *complementary safety indicators* (CSIs) will be used. These can be presented as various quantities and applied to provide confidence in the safety of as well the whole as parts of the repository system. The CSIs can be divided into two categories: numerical (e.g., radionuclide fluxes) and qualitative (e.g., evidence from natural and anthropogenic analogues). The most important CSI is the radionuclide specific flux from the geosphere into the biosphere, since the regulatory quantity to show compliance with for the assessment beyond the adequately predictable time is *"the average quantities of radioactive substances released from the disposed waste and migrated to the environment*".

Performance/function indicators in the Posiva Safety Case

The terminology with performance indicator (PI) and function indicator (FI) has not been adopted in the Posiva Safety Case as such. Up to now these terms have been used rather indistinctly and in many contexts without proper definitions. In the future the terminology of all indicators needs to be unambiguously defined, at least in the context in which they are used.

In the development of the alternative disposal concept KBS-3H, a first preliminary safety case is currently being developed. The treatment of indicators other than safety indicators resembles the Swedish approach in SR-Can (SKB 2006) where safety function indicators are introduced. In the KBS-3H Safety Case, FIs are used for properties of the disposal system components ("property" is like equal to "safety function indicator") to which a criterion





Part 4: Safety indicators and performance indicators

Appendix A11: POSIVA (Finland)

(quantitative) is given to fulfil the requirement. The criterion is assigned after thorough research on its significance. As an example, the interdependent function indicators and criteria on the density, swelling pressure and hydraulic conductivity of the buffer at the end of the early transient period of buffer saturation, adapted from criteria developed for the SR-Can safety assessment, are presented in Table 1 (from TKS 2006). Similar criteria have been set up for the compartment and the drift end plugs.

Table.1. Function indicators and criteria on the density, swelling pressure and hydraulic conductivity of the buffer after the early transient period of buffer saturation. Temperature criteria that apply at all times are also included for completeness (adapted for KBS-3H from Table 6-2 of SKB 2006)

Property			
indicator"	Criterion	Rationale	Notes
Applicable to throughout buffer after buffer saturation:			
Bulk hydraulic	< 10-12	Avoid advective	Isolated regions of higher around the super
Conductivity	ms	transport in butter	containers are not excluded
pressure	> 1 MPa	sealing	excluded
Minimum density	> 1650 kg m ⁻³	Prevent colloid- facilitated radionuclide transport	Isolated regions of higher hydraulic conductivity around the super containers are not excluded
Maximum density	Criterion yet to be defined	Avoid damage to rock	Tentatively set at < 2050 kg m-3, which would keep swelling pressure below about 13 MPa – (SKB 2006b, Figure 2.6)
Applicable between canisters and drift wall after buffer saturation (not relevant for distance blocks):			
Swelling pressure	> 2 MPa	Prevent microbial activity (to be confirmed)	Applicable adjacent to canister surfaces - "automatically" satisfied if criterion to prevent colloid-facilitated radionuclide transport also satisfied
	> 0.2 MPa	Avoid significant canister sinking	Criterion may be lower than for KBS-3V due to weight of canister being distributed over greater area
Maximum density	< 2100 kg m ⁻³	Ensure protection of canister against rock shear	May need modification if mineralogical alteration(cementation) of buffer cannot be excluded
Applicable throughout the buffer at all times:			
Minimum buffer temperature	> 0 °C	Avoid freezing	
Maximum buffer temperature	< 100 °C	Ensure mineralogical stability of buffer	

References

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Appendix A11: POSIVA (Finland)

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Part 4: Safety indicators and performance indicators Appendix A12: SCK-CEN, ONDRAF-NIRAS (Belgium)

A12 SCK-CEN, ONDRAF-NIRAS (Belgium)



Part 4: Safety indicators and performance indicators Appendix A12: SCK-CEN, ONDRAF-NIRAS (Belgium)





Proposal/Contract no.: **FP6-036404**

Project acronym: PAMINA

- Project title: PERFORMANCE ASSESSMENT METHODOLOGIES IN APPLICATION TO GUIDE THE DEVELOPMENT OF THE SAFETY CASE
- Instrument: Integrated Project
- Thematic Priority: Management of Radioactive Waste and Radiation Protection and other activities in the field of Nuclear Technologies and Safety

WP1.1 Safety indicators and performance/function indicators.

ONDRAF / NIRAS and SCK•CEN contribution

Version: F

Project co-funded by the European Commission within the Sixth Framework Programme (2002-2006) Dissemination level

- PU
 Public

 PP
 Restricted to other programme participants (including the Commission Services)
 X

 RE
 Restricted to a group specified by the consortium (including the Commission Services)
 X
- CO Confidential, only for members of the consortium (including the Commission Services)

[PAMINA] (D-N°: 1.1.1) – Task reports for the first group of topics Dissemination level: RE Date of issue of this report: 15/03/2008



456/456



1. Introduction and background

This document gives a short overview of the ONDRAF/NIRAS and SCK•CEN experience regarding the use of safety and performance indicators in the safety assessment (SA) of deep disposal of HLW and spent fuel. The information provided in this document mainly refers to what has been published on this topic in the most recent safety report (SAFIR 2,[1]) and in papers presented at international symposia and conferences [2,3].

2. Regulatory requirements and provisions

In the general rules for the protection against the risks of ionizing radiation (Royal Decree of July, 20, 2001) only dose as a safety indicators is defined. The general dose limits for members of the public (1 mSv/a) and for workers (20 mSv/a) are applicable to all types of nuclear installations, including disposal facilities. The concept of dose constraint has also been introduced but no specific values have been determined.

At this moment (September 2007) no specific regulations and regulatory guidance for disposal exists in Belgium. Regulatory documents for disposal are currently under development by the Federal Agency of Nuclear Control. ONDRAF/NIRAS has based its past safety assessments and safety reports (SAFIR 2) on the international recommendations (IAEA, ICRP).

3. Key terms and concepts

The terms safety indicators and performance indicators have up till now not been formally defined in the Belgian disposal programme, but are generally in line with the international definitions (EC SPIN, IAEA Tecdocs). In the past various indicators have been used for evaluating the safety and performance of a geological disposal system, without always making a clear distinction between safety and performance indicators. In the ongoing work, and with the current developments on regulatory standards and guidance in Belgium, a clearer distinction between the various indicators will be made.

In SAFIR 2 *Safety indicators* represent in general terms indicators of the safety provided by the system as a whole; they provide a measure of the safety level that can be obtained by comparing them with reference values. For the radiological safety of the system use is made of the following regulatory safety indicators:

The individual effective dose (Sv or mSv per year)

The radiological risk, defined as:

R = (the probability of exposure) x (the probability that the exposure will lead to a harmful effect on the health),

or

R = (the probability of exposure) x (the dose received) x (the risk factor),

has been expressed in a qualitative manner in SAFIR 2.

Besides effective doses as the main indicators, ONDRAF/NIRAS and SCK•CEN have





developed indicators "complementary" to effective dose and risk [1,2]. These indicators allow to evaluate the global and partial (sub-system level) performance of the disposal system and of the long-term safety functions. The major performance and complementary indicators are:

The decayed fractions: The ratio between the quantity of radionuclides that disappear through radioactive decay inside the disposal system and the initial quantity of activity placed in the system.

The containment factor: The ratio of disposed activity to cumulative released activity in the aquifer. This indicator is related more directly to the performance of the two long-term safety functions of the disposal system: 'physical containment' and 'delaying and spreading the releases'.

The Total activity fluxes at the interface between the Boom Clay and the above lying aquifer.

The evolution of the U-inventory compared to the natural α -activity in the Boom Clay and in the above lying aquifers.

4. Treatment in the Safety Case

4.1 Selection of indicators

While the effective individual dose was used for the normal evolution scenario (an occurrence probability of 1 or almost 1) it was argued in SAFIR 2 that for scenarios with a probability of occurrence significantly less than 1 the use of the individual effective dose alone is not appropriate. It is impossible to rule out certain highly improbable scenarios in which a dose higher than the dose constraint will be received, and in such cases the radiological risk is a more suitable safety indicator.

Uncertainty about calculated collective doses increases rapidly with time (even more than for individual annual doses) in deep disposal, because a reasonable estimate of the number of exposed persons becomes more difficult the further into the future considered. The collective dose therefore has limited value as a safety indicator for the post-closure safety of geological disposal of HLW and spent fuel and has not been calculated in the framework of the SAFIR 2 report.

The use of the various safety indicators is linked to the time frames considered in the safety assessments and to the general trend of increasing uncertainties with time. This is schematically presented in figure 4.5 in the Technical Overview of SAFIR 2 [ⁱ]. Also the "qualitative" arguments were introduced in this figure.

The interpretation of the results of the long-term radiological safety assessments for a deep disposal system must be based not only on quantitative indicators but also on qualitative arguments. More specifically, the interpretation of results is based both on a comparison of the values of (...) indicators with the relevant standards or threshold values and also on an assessment of the quality of the reasoning. These indicators and the bases of reasoning which accompany them have a certain relative importance or 'weight' that may vary, depending on which phase in the evolution of the disposal system is being considered









4.2 Reference values

The repository must be designed and constructed such that the calculated annual effective dose for a member of the reference group due to the release of radionuclides by gradual processes (expected evolution scenario) is less than the regulatory dose constraint. As this constraint was not defined in the national disposal regulations, NIRAS/ONDRAF made use of the international recommendations (ICRP 77 and 81), i.e. a value of not more than 0.3 mSv/a.

4.3 Application and presentation

4.3.1 Safety indicators

For all waste streams considered in SAFIR 2 (spent fuel, vitrified HLW from reprocessing, cladding waste) the post-closure radiological impact was calculated and expressed in *effective dose* as a function of time. One example is given in the figure below from SAFIR 2 for the case of spent fuel disposal. The effective doses attributable to the activation and fission products (A&F) and to the actinides (Act) for different types (and amounts) of spent fuel – resulting from different burn ups – are given.

We emphasized in the SAFIR 2 report that for extremely long time scales considered the





calculated dose only gives an indication of impact for the assumptions made (stable geology, hydrogeology and biosphere). In the assessments foreseen for our next safety case, we will apply cut-off calculations at 10^6 years.



A calculated or estimated *radiological risk* has two independent components, i.e. the individual effective dose and the probability of exposure. Both components are important in assessing the level of safety, therefore, it is appropriate to present both components separately.

For the altered evolution scenarios treated in SAFIR 2 (e.g. poor sealing and the drilling of a water well in the aquifer below the Boom Clay with the use of the pumped water for various applications) the individual effective doses were calculated but the obtained values were also discussed (in a qualitative manner) in terms of likelihood of occurrence of the scenario as described in the conceptual models. But the results were not expressed in an aggregated risk value.





4.3.2 Performance indicators

By comparing the total cumulative activity that is released from the Boom Clay into the above lying Neogene Aquifer over a period of 100 million years with the initial total activity placed in the disposal system, the *decayed fractions* within the disposal system can be calculated. This can be used as a way to assess the overall performance of the system's barriers.

In the case of the vitrified waste, only a very small portion of the initial activity reaches the aquifer (see Fig. 4.21 from SAFIR 2 below):

- about 2.10¹⁰ Bq of activation and fission products for a total initial activity of 7.10¹⁹ Bq;
- about 10⁷ Bq of actinides (i.e., the mean concentration of actinides in 0.2 m³ of category A radioactive waste) for a total initial activity of around 5.10¹⁷ Bq (mainly ²⁴¹Am and ²⁴⁴Cm).

From this information the containment factor, defined as in Sect. 3, was calculated: $4 \cdot 10^9$ for the activation and fission products and $5 \cdot 10^{10}$ for the actinides.

The total percentage of some long-lived radionuclides reaching the biosphere is high, however: 99 % for ¹²⁹I, 94 % for ²³⁸U, 92 % for ²³⁵U, etc. Nevertheless, this release is very much spread out over time, so that a future individual would only be exposed to a very small fraction of the total activity placed in the disposal system. Therefore, the annual maximum flux was also expressed as a fraction of the total RN activity disposed of. With the vitrified waste for example, the annual maximum flux of ¹²⁹I ($2 \cdot 10^6$ Bq per year) corresponds to $3 \cdot 10^{-6}$ of the total activity of ¹²⁹I disposed. For ⁷⁹Se, the annual maximum flux ($2 \cdot 10^7$ Bq per year) corresponds to $3 \cdot 10^{-7}$ of the initial activity of ⁷⁹Se.



Figure 4.21 (SAFIR 2) Cumulative activity reaching the Neogene Aquifer for the vitrified waste.



4.3.3 Complementary indicators

In SAFIR 2 the *total maximum annual flux of activity released from the Boom Clay in the above lying aquifer* was calculated and compared to the natural α -activity present in the Boom Clay and the sandy layers above the Boom Clay. The total maximum activity flux for the vitrified waste, spent fuel, and hulls and endpieces at the interface between the Boom Clay and the Neogene Aquifer is approximately $2 \cdot 10^7$ Bq per year. This is less than $100 \text{ Bq} \cdot \text{m}^{-2}$ per year, given that the area of the considered repository is 0.224 km² to accommodate the vitrified waste and 1.17 km² to accommodate the spent fuel.

The maximum total activity released in 1 year is compared to the *a*-activity of the uranium, thorium, and radium naturally occurring in a layer of Boom Clay, and was found to be equal to this natural activity in a layer of around 0.1 mm thick. (The mean activity of these isotopes in the clay is approximately $360 \text{ Bq}\cdot\text{kg}^{-1}$, or $7\cdot10^5 \text{ Bq}\cdot\text{m}^{-3}$.) In addition, the flux of radionuclides that leaves the Boom Clay and reaches the Neogene Aquifer only adds 0.0008 % per year to the natural activity already present in the Berchem Sands, a sub-layer of the Neogene Aquifer approximately 20 metres thick and situated just above the Boom Clay. (The natural activity of uranium, thorium, and radium in this layer is approximately $400 \text{ Bq}\cdot\text{kg}^{-1}$, or $6\cdot10^5 \text{ Bq}\cdot\text{m}^{-3}$.) Finally, the cumulative total activity due to the vitrified waste that reaches the Neogene Aquifer, integrated over a period of 100 million years (Fig. 4.21 in SAFIR 2), can be compared to the *a*-activity naturally present in the Berchem Sands. For the released activation and fission products, this corresponds to the *a*-activity present in an approximately 10-cm-thick layer of the Berchem Sands and, for the actinides, it corresponds to the *a*-activity in a layer approximately 0.1 mm thick.

A third alternative indication of the potential radiological impact in the very long term used in SAFIR 2 was to compare the *total initial inventory of uranium* in the vitrified waste and in the spent fuel with the quantity of α -activity naturally present in the Boom Clay around the disposal facility (approximately $7 \cdot 10^5$ Bq·m⁻³). For the total initial inventory of approximately $5 \cdot 10^{12}$ Bq U in the vitrified waste and approximately $2 \cdot 10^{14}$ Bq U in the spent fuel, the quantity of uranium isotopes that is disposed of in the repository is actually close to the quantity of α -activity already naturally present in the volume of clay that surrounds the disposal facility. Specifically, considering a clay layer that is 80 metres thick—the effective thickness of the Boom Clay layer—the equivalent volume of clay would have an area of 0.25 km² for the vitrified waste and 4 km² for the spent fuel.

4.4 Related topics

Within Task 3.4 of PAMINA, ONDRAF/NIRAS and SCK•CEN will reexamine the applicability of the SPIN indicators and will collect and evaluate some additional reference values based on naturally occurring radionuclides.

4.5 On-going work and future evolution

Since the publication of SAFIR 2 NIRAS/ONDRAF and SCK.CEN have continued to work, in interaction with the FANC, on the definition of the long-term safety functions and on their use in the designing the system and in assessing its safety. The definition of the functions and sub-functions have evolved. Further development in the field of safety and functional indicators will be based on these "stabilized" safety function definitions. We refer to the work that will be conducted within RTDC3.







5. Lessons learnt

The peer review of SAFIR 2 found the alternative safety indicators very informative (e.g. the assessment of how many of the radionuclides decay to insignificant levels while still within the engineered system) and useful for communication to less-specialized audiences. They recommended further development of the work on these indicators for both expected and altered evolution scenarios.

6. References

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- 3 J. Marivoet, P. De Preter, X. Sillen, Functional requirements and performance assessment: application to the case of spent fuel disposal in clay, Proceedings ICEM 2001, Brugge, 2001.